Organisation de Coopération et de Développement Économiques
Organisation for Economic Co-operation and Development

03-Dec-2010

English text only

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2010)10/PART1
Unclassified

Implementation of Severe Accident Management Measures, ISAMM 2009

Workshop Proceedings, Vol. I
Schloss Böttstein, Switzerland
26-28 October 2009

In collaboratioin with KKB, KKL, KKM, KKG and PSI

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JT03293928

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The mission of the NEA is:

– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
– to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

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Within the OECD framework, the NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, as well as representatives from regulatory authorities. It was set up in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety amongst the NEA member countries. The CSNI's main tasks are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and research consensus on technical issues; and to promote the co-ordination of work that serves to maintain competence in nuclear safety matters, including the establishment of joint undertakings.

The clear priority of the committee is on the safety of nuclear installations and the design and construction of new reactors and installations. For advanced reactor designs the committee provides a forum for improving safety related knowledge and a vehicle for joint research.

In implementing its programme, the CSNI establishes co-operate mechanisms with the NEA's Committee on Nuclear Regulatory Activities (CNRA) which is responsible for the programme of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with the other NEA's Standing Committees as well as with key international organizations (e.g. the IAEA) on matters of common interest.

Nuclear Safety
NEA/CSNI/R(2010)10
PSI Report Nr. 10-y
October 2010

OECD

Implementation of Severe Accident Management Measures, ISAMM 2009

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

WORKING GROUP ON ANALYSIS AND MANAGEMENT OF ACCIDENTS

IMPLEMENTATION OF SEVERE ACCIDENT MANAGEMENT MEASURES (ISAMM 2009)

Workshop Proceedings

Hosted by Paul Scherrer Institute

Schloss Böttstein
5315 Böttstein, Switzerland
October 26 - 28, 2009

Sponsored by
Nuclear Power Plant Beznau (KKB), Nuclear Power Plant Leibstadt (KKL)
Nuclear Power Plant Gösgen (KKG), Nuclear Power Plant Mühleberg (KKM)
and Paul Scherrer Institut (PSI)

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Organisation for Economic Co-operation and Development

The Convention establishing the Organisation for Economic Co-operation and Development (OECD) was signed on 14th December 1960.

Pursuant to Article 1 of the Convention, the OECD shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter : Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

The Nuclear Energy Agency

The Nuclear Energy Agency (NEA) is a specialised agency within the Organisation for Economic Co-operation and Development (OECD), an intergovernmental organisation of industrialised countries based in Paris, France.

The OECD's fundamental mission is to enable Members to consult and co-operate with each other so as to achieve the highest possible sustainable economic growth, improve the economic and social well-being of their populations, and contribute to development worldwide.

The primary objective of the NEA is to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes.

The NEA was established in 1958. Current membership consists of 27 countries, drawn from Europe, North America and the Asia-Pacific region. The Commission of the European Communities takes part in the work of the Agency.

NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Committee on the Safety of Nuclear Installations

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, and representatives from regulatory authorities. It was set up in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries. CSNI's main tasks are to exchange technical information and to promote collaboration between research, development, engineering and regulation organisations; to review the state of knowledge on selected topics of nuclear safety technology and safety assessments, including operating experience; to initiate and conduct programmes to overcome discrepancies, develop improvements and reach consensus on technical issues; to promote co-ordination of work, including the establishment of joint undertakings.

Working Group on Analysis and Management of Accidents

The Working Group on the Analysis and Management of Accidents (GAMA) is mainly composed of technical specialists in the areas of coolant system thermal-hydraulics, in-vessel protection, containment protection, and fission product retention. Its general functions include the exchange of information on national and international activities in these areas, the exchange of detailed technical information, and the discussion of progress achieved in respect of specific technical issues. Severe accident management is one of the important tasks of the group.

TABLE OF CONTENTS

- **Development of the SAM strategy for PAKS NPP on the basis of Level 2 PSA,**
  J. Elter (Paks Nuclear Power Plant, Hungary),
  G. Lajtha (NUBIKI, Hungary),
  É. Tóth ((Paks Nuclear Power Plant, Hungary),
  Z. Téchy (NUBIKI, Hungary)

- **Development of Technical Bases For Severe Accident Management in New Reactors**
  E. L. Fuller, H. G. Hamzehee (US-NRC, USA)

Session 3: PRA Modelling Issues

Chairs/Co-Chairs: *J. Primet /V. Dang*

- **Some international efforts to progress in the harmonization of L2 PSA development and their implication within OECD-CSNI, ANS, IAEA, EC (ASAMPSA2)**
  E. Raimond (IRSN, France)
  S. Guentay (PSI, Switzerland),
  C. Bassi (CEA, France)
  D. Helton (US-NRC, USA),
  A. Lyubarskiy (IAEA, Austria)

- **Accident Management and Risk Evaluation of Shutdown Modes at Beznau NPP**
  M. Richner, S. Zimmermann (KKB, Switzerland)
  J. Birchley, T. Haste, (PSI, Switzerland)
  N. Dessars (Westinghouse Electric Belgium S.A, USA)

- **The Role of Severe Accident Management in the Advancement of Level 2 PRA Modelling Techniques**
  D. Helton, J. Chang, N. Siu (US-NRC, USA),
  M. Leonard (Dycoda, LLC, USA),
  K. Coyne, N. Siu (US-NRC, USA)

- **Overview of the Modelling of Severe Accident Management in the Swiss Probabilistic Safety Analyses**
  V.h N. Dang (PSI, Switzerland),
  G. Schoen, B. Reer (ENSI, Switzerland),

- **Extended Use of MERMOS to assess Human Failure Events in Level 2 PSA**
  H. Pesme, P. Le Bot (*EDF R&D, France)*

Session 4: Code Analysis for Supporting SAMGs Part 1

Chairs/Co-Chairs: *E. Raimond/Y. Liao*

- **Best-Estimate Calculations of Unmitigated Severe Accidents in State-of-the-Art Reactor Consequence Analyses**
  C. G. Tinkler (US.NRC, USA),
  K.C. Wagne (SNL, USA),
  M. T. Leonard (Dycoda, LLC, USA)
  J. H. Schaperow (US-NRC, USA)

- **Deterministic Evaluation of Quantitative Health Objective and Target of Severe Accident Management**
  C.W. Huh N.D. Suh, (KINS , Korea)
  G.H. Jung (FNC Technology, Korea)

- **Verification of the SAMG for PAKS NPP with MAAP Code Calculations**
  G. Lajtha, Z. Téchy (NUBIKI, Hungary),
  J. Elter, É. Tóth (Paks Nuclear Power Plant, Hungary)

- **Treatment of Accident Mitigation Measures in State-of-the-Art Reactor Consequence Analyses**
  C. G. Tinkler (US.NRC, USA),
  K.C. Wagne (SNL, USA),
  M. T. Leonard (Dycoda, LLC, USA)
  J. H. Schaperow (US-NRC, USA)

- **Best Practices Applied to Deterministic Severe Accident and Source Term Analyses for PSA Level 2 for German NPPs**
  M. Sonnenkalb, N. Reinke, H. Nowack (GRS, Germany)

Session 5: Code Analysis for Supporting SAMGs Part 2

Chairs/Co-Chairs: *M. Leonard/ M. G. Cenerino*

- **Severe Core Damage Accident Analysis for a CANDU Plant,**
  P. Mani Mathew, S. M. Petoukhov, M. J. Brown and, B. Awadh (AECL, Canada)

- **Time Window for Steam Generator Secondary Side Reflooding to Mitigate Large Early Release Following SBO-Induced SGTR Accidents**
  Y. Liao, S. Guentay (PSI, Switzerland)

- **On the Effectiveness of CRGT Cooling as a Severe Accident Management Measure for BWRs**
  M. Weimin, T. Chi-Thanh (KTH, Sweden)

- **Ex-Vessel Corium Management for the VVER-1000 Reactor**
  B. Kujal (Nuclear Research Institute Řež plc Czech Republic)

Session 6: Decision making, Tools, Training, Risk Targets and Entrance to SAM

Chairs/Co-Chairs: *D. Helton/P. Le-Bot*

- **Criteria for the Transition to Severe Accident Management**
  B. Prio (Jacobsen Engineering Ltd (JEL, UK))

- **Use of The Software Module Sprint in the Netherlands**
  M. Slootman (NRG, Netherlands)

- **Safety Goals and Risk Targets for Severe Accidents in View of IAEA Recommendations**
  S J. Vitázková, E. Cazzoli (CCA, Switzerland)

- **Development, Validation and Training of Severe Accident Management Measures,**
  A. Torri, V. Pokorny, U. Lüttringhaus (RMA, USA)

- **Severe Accident Training in Spain: Experiences and Relevant Features**
  R. Martínez (Tecnatom, Spain),
  J.Benavides (C.N. Trillo, Spain),
  J. M. de Blas (C.N. Santa María de Garoña, Nuclenor, Spain),
  M. A. Catena (C.N. Almaraz, Spain),
  I. Sol (A.N. Ascó-Vandellós II, Spain)

Session 7: Design Modifications for Implementation of SAM

Chairs/Co-Chairs: *J. Primet/A. Lyubarskiy*

- **A Novel Process for Efficient Retention of Volatile Iodine Species in Aqueous Solutions during Reactor Accidents**
  S. Guentay, H. Bruchertseifer, (PSI, Switzerland),
  H. Venz,F. Wallimann (KKB, Switzerland),
  B. Jaeckel (PSI, Switzerland)

- **Development of Leibstadt NPP Severe Accident Management Guidelines for Shutdown Conditions (SSAMG)**
  W. Hoesel (KKL, Switzerland)

- **Design Modifications of the Mochovce Units 3 & 4 dedicated to Mitigation of Severe Accident Consequences, providing Conditions for Effective SAM**
  M. Cvan, D. Šiko (VUJE, a.s., Slovenské elektrárne, a.s., Slovenian)

Session 8: Physical phenomena affecting SAM Part 1

Chairs/Co-Chairs: *F. Kappler/M. D. Leteinturier*

- **Experimental Investigation of Melt Debris Agglomeration with High Melting Temperature Simulant Materials**
  P. Kudinov,A. Karbojian, C.-T. Tran (KTH, Sweden)

- **Approach to Prediction of Melt Debris Agglomeration Modes in a LWR Severe Accident**
  P. Kudinov (KTH, Sweden),
  M. Davydov(Electrogorsk Research and Engineering Center on Nuclear Power Plants Safety (EREC), Electrogorsk, Russia)

- **A Fuel Coolant Interaction Programme (FCI) devoted to reactor case**
  P. Piluso (CEA, France)
  S. W. Hong (KAERI, Kore)a,

- **Improved Molten Core Cooling Strategy in a Severe Accident Management Guideline**
  J.H Song (KAERI, Korea),
  C.W Huh, N.D Suh (KINS, Korea)

- **Summary and Outcome of OECD-Workshop on In-vessel Melt Pool Retention**
  B. Clément(IRSN; France)

- **Simulation of Ex-Vessel Debris Bed Formation and Coolability in a LWR Severe Accident**
  S. Yakush (Institute for Problems in Mechanics, Russian Academy of Sciences, Moscow, Russia),
  P. Kudinov(KTH, Sweden)

- **Substantiation of strategy of water supply recovery to steam generators at in-vessel severe accident phase for VVER-1000 Balakovo NPP**
  A. Suslov, V. Mitkin (RRC "Kurchatov Institute", Russian Federeation)

- **Ambient Pressure-Dependent Radionuclide Release from Fuel Observed In VEGA Tests under Severe Accident Condition and Influence on Source Term Evaluation**
  A. Hidaka (JAERI, Japan)

Appendices
- **Workshop Programme**
- **Organising Committee and Session Chairpersons**
- **List of Participants**

OECD WORKSHOP ON

THE IMPLEMENTATION OF SEVERE ACCIDENT MANAGEMENT MEASURES

ISAMM 2009

Organised in collaboration with Paul Scherrer Institute and Swiss Utilities Running Nuclear Power Plants Beznau, Leibstadt, Gösgen and Mühleberg

Schloss Böttstein, Switzerland

26-28 October 2009.

EXECUTIVE SUMMARY

Background and scope of the Workshop:

The subject of severe accident management (SAM) is part of a sequence of actions that are taken by utilities and national authorities to appropriately plan for a postulated severe accident at a nuclear power plant (NPP). Such actions involve a large number of steps such as preventive actions, installation of hardware, development and implementation of software, development and implementation of severe accident management strategies and programmes, training of those who would be called upon to deal with a severe accident, severe accident management, emergency centres, mitigation of consequences, emergency preparedness and emergency response plans. Several international organisations, including the Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA), have responsibilities and are active in this area (publication of a large number of related documents, organisation of periodic international emergency exercises, etc.)

In this respect, a joint OECD workshop of the Working Group on the Analysis and Management of Accidents (WGAMA) and the Working Group on Risk Assessment (WGRISK) on implementation of severe accident management measures was sponsored by the Committee on the Safety of Nuclear Installations (CSNI) of the Organization for Economic Cooperation and Development (OECD) NEA and was held in Schloss Böttstein on October 26 to 28, 2009, in Switzerland in cooperation with Paul Scherrer Institute (PSI) and the Swiss Utilities running NPPs Beznau, Leibstadt, Gösgen and Mühleberg.

This Workshop is a "follow-up" on the past CSNI activities in the area of severe accident management implementation, and to take stock of the progress made since the Rome Meeting on Severe Accident Management Programme Development held in September 1991 [CSNI reports NEA/CSNI/R(1991)16 and (1992)6], the Niantic Specialist Meeting on Severe Accident Management Implementation held in June 1995 [NEA/CSNI/R(1995)5 and 16], the Winnipeg Workshop on the Implementation of Hydrogen Mitigation Techniques held in May 1996 [NEA/CSNI/R(1996)8 and 9], the PSI-Villigen Workshop in September 2001 [NEA/CSNI/R(2001)20] and the Workshop on Evaluation of Uncertainties in Relation to Severe Accidents and Level-2 Probabilistic Safety Analysis, Aix-en-Provence, 7-9 November 2005 [NEA Report 6053]. A series of CSNI-sponsored meetings on dedicated topics covering iodine chemistry, fuel coolant interaction, operator aids for severe accident management, hydrogen mitigation, etc, were held over the last two decades.

In particular, the present workshop represents an update of the status of severe accident management measures and their implications since the OECD/CSNI workshop held in 2001 at PSI Switzerland. Since the 2001 workshop, additional work has been performed to integrate emergency procedures and SAM measures into risk assessments in order to better reflect operator responses to recover the plant from a damaged state. Therefore, a major focus of the workshop was to address SAM measures for both operating plants and new plant designs (as available) and the integration of SAM measures into contemporary/future probabilistic risk assessments.

Among the initially proposed 44 papers, 41 were presented in 8 sessions and addressed the following 6 areas:

- Current Status & Insights of SAM (in two sessions)
- PSA Modelling Issues

- Code Analysis for Supporting SAMGs (in two sessions)
- Decision making, Tools, Training, Risk Targets and Entrance to SAM
- Design Modifications for Implementation of SAM
- Physical phenomena

The last part of the workshop was devoted to presentation of the most striking highlights of the papers in the above topical areas, followed by two panellists giving presentations on:

- Human and Organizational Aspects of SAM: their importance vs. technical issues
- Effectiveness of current SAMG implementation - How can consequence analyses be used to improve the effectiveness of SAM.

The program of the workshop is reproduced in Appendix 1. Over 110 participants from OECD and non-OECD countries, i.e., Russian Federation, China and IAEA, attended the workshop, which contributed to the success of the workshop in terms of exchanging knowledge and experience among participants. Interesting discussions followed each paper, as well as the two presentations made in the panels. A list of participants is reproduced in Appendix 2.

Short summary of sessions

The Annex contains expanded summaries of each topical area. These summaries were prepared by the Session Chairpersons and discussed by the whole writing group.

During the two sessions, Session 1 and 2, of the topical area “Current Status & Insights of SAM”, 11 papers from regulatory bodies, Technical Safety Organizations (TSOs), several utilities and national research institutes were presented to outline the status of implementation of SAM programmes in countries like Canada, Germany, Japan, France, United States of America, Republic of Korea, Switzerland, Finland, and Hungary. Some of the papers also described the expansion of the SAM programmes to low power and shut down states. Finally one paper described the development of technical bases for SAM programmes in new generation III reactors based on the US process of reviewing design certification applications.

Session 3, for the topical area “Modelling PSA Issues” addressed the modelling of implemented SAMGs in Level 2 PSA, progress in Level 2 PSA, and the results of these studies. Three papers discussed different PSA modelling issues; development of some international efforts to progress in the harmonization of Level 2 PSA development and their implications within international organizations developing guidelines, the role of severe accident management in the advancement of Level 2 PSA modelling, and an overview of the modelling of severe accident management in the Swiss probabilistic safety analyses. Assessment of human failure events in Level 2 PSA and accident management and risk evaluation of shutdown states were the other topics covered in the other two papers.

During the two sessions, Session 4 and 5, of the topical area “Code Analysis for Supporting SAMGs”, nine papers presented different uses of best estimate codes for severe accident and source term calculations for unmitigated accident scenarios and treatment of accident mitigation measures. Specific applications were the determination of time windows for operator actions to mitigate large early release from steam generator tube rupture (SGTR) sequences, effectiveness of control rod guide tube cooling as a severe accident management measure for BWRs, and deterministic evaluation of quantitative health objectives and targets of severe accident management. The U.S. Nuclear Regulatory Commission’s State-of-the-Art Reactor Consequence Analyses project and verification of the SAMG developed for the PAKS plant were the subject of the other three papers. Best practices applied to deterministic severe accident and source term analyses from Gesellschaft für Anlagen und Reaktorsicherheit (GRS) provided the German experience for the use of MELCOR code for Level 2 PSA analyses and assessment of SAMGs.

Five papers presented in Session 6 covered each subtopic of the topical area “Decision Making, Tools, Training, Risk Targets and Entrance to SAM.” The paper on criteria for the transition to severe accident management provided an overview on available means helping plant operators to decide when to leave the emergency operational domain and enter the severe accident domain. The paper on the safety goals and risk targets for severe accidents in view of IAEA recommendations provided perhaps the most debated discussions on the safety goals, risk targets, and use of a common safety index. The other three papers focussed on (i) an accident diagnosis tool in the Netherlands, (ii) a perspective on the development validation and training of SAM measures, and (iii) SAM training activities in Spain.

In the topical area of “Design Modifications for Implementation of SAM” two presentations covered design modifications of Mochovce Units 3 & 4 for an effective SAM, and a new technique to be added to existing containment venting filter systems to suppress efficiently the release of all volatile iodine species. Development of Leibstadt NPP Severe Accident Management Guidelines for Shutdown Conditions was the subject of the third paper.

Eight papers were presented in Session 8 for the topical area “Physical Phenomena Affecting SAM” and covered the results of the recent research affecting SAM measures. Specific topics included experimental investigation of melt debris agglomeration and respective modelling, the status of the OECD project on fuel coolant interaction, a summary of the outcome of a very recent OECD workshop on ‘In-vessel coolability,” simulation of ex-vessel debris bed formation and coolability, substantiation of a strategy of water supply recovery to steam generators at the in-vessel severe accident phase for VVER-1000, ambient pressure-dependent radionuclide release from fuel observed in the VEGA tests under severe accident condition and influence on source term evaluation. One paper specifically dealt with the implementation of the research results for the improved molten core cooling strategy in a severe accident management guideline.

During the final session of the workshop, two invited speakers made presentations. Mr. C. Huh, from the Korea Institute of Nuclear Safety, made a presentation focusing on organizational aspects of decision making during postulated accidents. This presentation highlighted technical and organizational aspects of current SAMGs, the effects of group decision-making in the TSC, and provided an illustrative simulation of group decision-making. A second presentation, made by Mr. M. Leonard (dycoda, LLC), focused on the potential benefits of looking at severe accident management from an “outside-in” approach rather than an “inside-out” approach driven primarily by PSA results. As an example, the presentation made the point that drawing the link between potential mitigation actions and the associated averted offsite consequences might open up new perspectives on what types of actions should be pursued. A particular question that was posed is whether advances in modelling scope and fidelity in contemporary PSAs, combined with improvements in plant design and SAMG implementation, have driven the frequency of core damage sequences down to levels that rival those of events that are not treated by PSAs.

General conclusions

The overall picture achieved at the end of the workshop is that significant progress has been made since the 2001 OECD workshop on "The Implementation of Severe Accident Management Measures" in implementing and completing Severe Accident Management (SAM) programmes in many countries for full power states. Also, the workshop objective of prompting more interaction between probabilistic and deterministic analysts has been achieved, based on the breadth of papers submitted and the discussions that took place regarding the interface of these two approaches.

IAEA has published a safety guide on severe accident management programmes for nuclear power plants in 2009 to provide recommendations for the development and implementation of an accident management programme. The guide contains detailed recommendations for all steps in developing accident management guidelines. In addition, two safety guides on PSA have been approved by the Commission on Safety Standards (CSS) and are expected to be published by the end of 2009. The Safety Guides on PSA provides recommendations for performing or managing a Level 1 and Level 2 PSA and for using the PSA to support the safe design and operation of NPPs.

Numerous initiatives at different levels (national, European, international) are currently in progress in order to define standards and/or best practices for development of Level 2 PSA. One of the benefits of developing these standards is to give the experts the opportunity to share experiences. It was emphasized that intended use and applications of Level 2 PSA should drive the choices for modelling alternatives and level of details of Level 2 PSA. The severe accident management measures have become an integral part of Level 2 PSA, which in turn implicitly defines the needs and the scope for the SAMM.

The way implementation of SAM programmes has been made and the measures implemented vary as reported by the different countries presentations. Harmonisation, to the extent it may be desirable, does not seem feasible at this stage as was the case in 2001. All approaches cover both preventive and mitigative aspects for beyond-design basis and severe accidents at full power level.

The common goal of the SAM approaches implemented is to develop strategies to mitigate the consequences of accidents leading to significant core degradation. However, the measure of consequences varies between the countries since safety goals and risk targets vary between them. Adopting a risk measure definition that utilizes a common scale, such as IAEA's International Nuclear and Radiological Event Scale (INES) scale, was proposed by a paper as a way to promote consistency of individual applications with IAEA guidance.

The development and implementation of a SAM programme has been required or recommended by the safety authorities, but not necessarily translated into rules and regulations. It is ultimately the duty of the safety authority to review the work of the utilities. Therefore, the approaches followed in the different countries do not fit one single pattern. This situation has not been changed since the 2001 workshop. However, as a common rule, the responsibility of the plant owner for the safety of his plant remains untouched. The general aim of SAM strategy development is still to ensure containment integrity and mitigate release for containment by-pass sequences. Actual differences between SAM programmes in the same areas as highlighted in 2001 Workshop exist as:

- status of progress of work on specific programmes,
- the extent to which hardware modifications are part of the SAM approach at a plant,
- the availability and scope of SAM guidance in the mitigative domain,
- the extent to which the decision-making process at a specific plant is left in the hands of the operators in the main control room or becomes the responsibility of a specific Technical Support Centre.

Regarding implementation of SAM in probabilistic risk analyses, a major difficulty includes the large number of unique sequences, which requires screening out "unimportant" operator actions, and strategies for merging sequences. Operator response models need to be further developed for the severe accident regime. In order to realise the advantages of advanced Level 2 modelling approaches, additional work is needed.

The fact that once implemented, the SAM guidelines should be considered as a living product, as stated in the 2001 workshop, has been restated in the 2009 Workshop. Periodic review and update, verification and assessment, and improvements as new knowledge becomes available are the crucial reasons why SAM guidelines should remain as a living product.

The need for the extension of the SAM programmes to low power and shutdown states was highlighted in the 2001 workshop. It was apparent from several presentations that a lot of progress has been made in this area and several utilities have already developed Level 2 PSAs for shutdown and low power severe accident situations. Based on such Level 2 PSAs and in comparison to the full power Level 2 PSA states, it was apparent that for some plants the core damage frequencies (CDF) might be the same order of magnitude for both states. In the recent studies, it has been found that the risk of a large release of radionuclides might be comparable or even higher for severe accidents at low power and shut down states than for those that might occur at full power operation.

The role of research regarding in-vessel and ex-vessel debris coolability and fuel coolant interactions with respect to SAM was heavily discussed. Further R&D results in the melt coolability, in-vessel melt pool retention and fuel coolant interaction should reduce (but not eliminate) some of the remaining important uncertainties and should give more confidence in the robustness of guidelines, either existing or to be developed. It should provide some basis to better demonstrate the optimization of the SAM options.

A presentation and later discussions on the question of whether the operators should attempt to flood the degraded core if the water injection flow rate is below a certain value provided much controversy; discussions highlighted the adverse effect of low water flow rate on possibly enhancing core degradation, RCS pressure, hydrogen generation and fission product release, and raised the question of how to know how much water flow rate is actually entering into the core. It should be noted that most of the guidelines as implemented would instruct operators to inject water regardless of knowing its amount, whenever water injection is possible.

The importance of validation and training of SAM measures has been stressed and the need for focused attention to consider errors of commission and to prioritize activities toward making the reliability of SAM guidance commensurate with that of hardware through better validation and training was also highlighted.

In particular, the timing of emergency response organisation staffing relative to accident progression was discussed. The case was made that guidance should account for the impact that this relationship will have on decision-making for rapidly-evolving accident sequences. For these cases, the operators in the control room need instructions and training much like they have for emergency procedures in order to manage the initial phase of these accidents successfully.

SAM measures for new generation III reactors were covered only in one paper; the details on the SAM guidance have not been presented.

Recommendations

The presentations made during the technical and panel sessions and accompanying discussions have resulted in the following recommendations:

- SAM programme development remains a living process. SAM strategies and associated procedures and guidance should be reviewed periodically to make them as practical and efficient as possible. New knowledge, gained from experience or from research, should be integrated regularly in to the assessment, for the purpose of improvement. Regular operator training and emergency exercises are integral parts of the implementation of the severe accident management.
- It may be worthwhile to examine whether the present consensus in PSA, to credit actions to recover systems and functions only when these are supported by procedures, should apply for the severe accident context modelled in Level 2 PRA. This consensus is currently oriented to the preventive domain treated in Level 1 PRA; however, a number of decisions in severe accidents relate to the prioritization of recoveries and the implementation of some SAM strategies depends on these recoveries.
- Efforts should be pursued to complete the SAM programs for low power and shutdown states. SAM strategies developed for full power states could be used as guidance but shutdown specificities must be taken into account.
- Work should be undertaken to identify what existing methods and gaps exist for extending presently used Level 1 PSA human reliability methods to Level 2 PSA / SAM.
- The validation of SAM measures from the standpoint of the necessary operator action timings and the instructions to operators for the first 2 hours of an accident should be addressed more rigorously, relative to the verification of the efficacy of hardware
- The approaches followed at different plants do not fit one single pattern. Harmonisation, to the extent it may be desirable, does not seem feasible so far. An effort should be pursued to catalogue differences and provide explanations for alternate approaches and philosophies.
- A potential activity for CSNI would be the coordination of an activity to gather information and formulate lessons learned regarding the degree to which external events (and their effects) have been explicitly considered as part of SAM development. Such an effort would include both the effects of external events on equipment and structure damage, as well as the effects on evacuation and offsite mitigation resources.
- An effort is needed to discuss of equipment survivability and operator access (during severe accident conditions) issues, and identifying what experiments, data, and other information exist or need to be developed to better address this issue. In addition, another aspect that needs to be covered is the treatment of these issues within a PSA or SAM verification analysis. The next OECD/NEA SAM workshop should have this topic as one of its focuses.

- An international effort should be undertaken to investigate and better understand the relationship between the metrics used to quantify the benefit of SAM measures and the metrics used to satisfy offsite radiological consequence targets (e.g., dose, land contamination), in addition to traditional metrics available from severe accident analysis.
- Some comparison and information exchange activities regarding the technical basis for SAMG strategies should be maintained at an international level, especially for in-vessel water injection, cavity pit flooding, containment spray system actuation and filtration efficiency of containment venting. These activities should focus on the balance between the advantages and disadvantages of each chosen SAMG strategies, depending on reactor design specificities. The aim of this effort is to develop an international consensus on the effectiveness of SAM measures.
- The next workshop on SAM should cover the information exchange on SAM guidance and measures for generation III reactors in more detail.

Session Summaries

Sessions 1 and 2

Current Status and Insights of SAM

Session Chairpersons:

N. Suh (KINS)

A. Torri (Risk Management Associates, Inc)

H. Fujimoto (JNES)

M. Sonnenkalb (GRS)

Session summary

During the first and second session ***of "Current Status & Insights of SAM"***, 11 papers from regulatory bodies, TSOs, several utilities and national research institutes were presented to outline the status of implementation of SAM programmes in countries like Canada, Germany, Japan, France, United States of America, Republic of Korea, Switzerland, Finland, and Hungary. Some of the papers as well described the expansion of the SAM programmes to low power and shut down states. Finally one paper described the development of technical bases for SAM programmes in new generation III reactors based on the US process of reviewing design certification applications.

Summaries of the presentations

Recent IAEA Activities in the Area of Severe Accident Management and Level-2 PSA,
A. Lyubarskiy, IAEA

The paper introduced the activities of the IAEA to develop safety guides for severe accident management, PSA and other IAEA activities. The IAEA has published a safety guide on severe accident management for nuclear power plants in 2009 to provide recommendations for the development and implementation of an accident management programme. The guide contains detailed recommendations for all steps in developing accident management guidelines. One of the guidelines recommends considering the capabilities of the plant personnel in handling a severe accident. A question was raised on how to consider the capabilities of personnel because there is no way to know how the operators would behave under the real situation of a severe accident. There was also a question on how decisions would be made based on the plant information gathered. Training of the operators was said to increase their capabilities but the IAEA safety guide only describes very general principles, so it cannot answer this kind of detailed issues. Two safety guides on PSA have already been approved by the Commission on Safety Standards (CSS) and are expected to be published by the end of 2009. The Safety Guide on PSA provides recommendations for performing or managing a Level-1 and Level-2 PSA for a NPP and for using the PSA to support the safe design and operation of NPPs.

Technical Challenges in Applying SAMG Methodology to Operating CANDU Plants
K. Dinnie, AMEC NSS, UK

The paper explained that the unique CANDU plant design features made the development of CANDU SAMGs a challenge. SAMGs for CANDU reactors was successfully developed based on a modified Westinghouse Owners' Group (WOG) SAMG. Because a CANDU has no direct measurement of the core temperature available and fuel damage at the design basis limit alone is not an indication of an imminent transition to a severe accident, CANDU specific entry conditions were needed that are different from that of an LWR. The loss of moderator level below that of the upper fuel channels in conjunction with a loss of primary core cooling was identified as primary SAMG entry condition. Also for preventing the SAMGs from being entered prematurely or unnecessarily, the measured dose rate corresponding to calculation at specified locations assuming 3% FP release to containment was also chosen as a "rationality check". There was a question and discussion on the validity of 3% FP release saying that the FP percentage from the gap release should be considered in judging the onset of a severe accident. The author explained that 3% of the total FP released corresponds to about 3 times the gap release and therefore the entry condition represents a core damage state well beyond the initial fuel damage. The order of the seven CANDU SAGs are 1) Inject into the Heat Transport System, 2) Control Moderator

Conditions, 3) Control Shield Tank Conditions, 4) Reduce Fission Product Releases, 5) Control Containment Conditions, 6) Reduce Containment Hydrogen and 7) Inject into Containment. This prioritization of barriers to a severe accident progression was chosen to make best use of the resources available and maximize the chances of terminating an accident progression. Hydrogen management and filtered venting are still challenging technical issues and efforts are continuing to improve the understanding of these issues.

Accident Management in German NPPs: Status of Implementation and the Associated Role of PSA Level 2

P. Scheib, M. K. Schneider, Bundesamt für Strahlenschutz (Federal Office for Radiation Protection, Germany

The paper explained that in Germany it is mandatory to perform Periodic Safety Reviews including a plant-specific PSA in ten years intervals and that the efficiency of AM measures regarding the mitigation of the consequences of severe accidents is evaluated in the frame of Level 2 PSA. The RSK recommended the implementation of several AM measures in 1988 like additional off-site power supplies via underground cables and the implementation of primary and secondary bleed and feed in PWRs, etc. The requested measures have been implemented in German NPPs as part of backfitting actions. PARs in PWRs have mostly been implemented around the year 2000. But there is an ongoing discussion on the use of PARs because the PARs might act as igniters for the hydrogen-air-mix. Recently Level 2 PSAs for three reference plants have been performed to advance the methodology of PSA and to give feedback to the regulators in order to improve the regulatory framework. One of the insights on the effectiveness of AM measures from these analyses is that filtered containment venting is an effective measure for PWRs and not very effective for BWRs.

Circumstances and Present Situation of Accident Management Implementation in Japan

H. Fujimoto, K. Kondo, T. Ito, Y. Kasagawa, O. Kawabata, M. Ogino and M. Yamashita, JNES, Japan

The paper described that the implementation of AM measures in the fifty-two operating NPPs had already been completed by 2002 involving plant modifications. The effectiveness of AM measures was evaluated by utilities and the results are reported to the regulatory body. According to the results, the reduction ratio of CDF by AM measures, which is defined by the ratio of CDF after AM implementation to CDF before AM implementation, varies in the range of 0.3 to 0.6 for PWR. This confirms the effectiveness of the selected AM measures.

Progress in the Implementation of Severe Accident Measures on the operated French PWRs – Some IRSN Views and Activities

E. Raimond, G. Cenerino, N. Rahni, M. Dubreuil, F. Pichereau, IRSN, France

The paper presented the progress obtained in the severe accident management of French PWRs with practical implementations of measures to limit the accident consequences or to make the management of an accident easier. Since 1990, severe accident management guidelines have been developed in France to help the PWR plant operators and emergency teams in limiting the consequences of any postulated severe accident. Some key systems and material provisions implemented to limit the accident consequences or to facilitate accident management are containment filtered venting system, hydrogen recombiners, reinforcement of material-access-closure-systems, and instrumentation for hydrogen release measurement in the containment or means for vessel rupture detection. The severe accident guidelines for French PWRs place a high importance on the prevention of early containment failure and propose to terminate water injection to an already damaged core if this would increase the possibility of early containment failure. Also EDF has developed a two part severe accident safety standard consisting of 1) safety requirements and 2) synthesis of the operating plants status related to severe accidents. The Level 2 PSAs and the severe accident standard are now seen as helpful tools for the review of severe accident issues and the identification of new plant improvements.

Perspectives on Severe Accident Alternatives for US Plant License Renewal
T. Ghosh, R. Palla, D. Helton , US-NRC, USA

The paper presented severe accident mitigation alternative (SAMA) analysis, which is performed in the license renewal of the US plants by licensees. Major steps of SAMA include identification of leading contributors to risk, identification of candidate SAMAs, risk reduction/implementation cost estimates, potentially cost-beneficial SAMAs, and more detailed analysis for remaining SAMAs. Up to now, SAMAs for more than 50 plants have been completed. Numerous potentially cost-beneficial SAMAs have been identified. SAMAs can be categorized into five types, i.e. SAMAs related to SBO or loss of power sequences; internal floods, fire and external events; protection systems; support systems; and procedures and training. Specific examples include procurement of an additional portable 480V AC station DG for backup to EDGs; installation of watertight doors/wall around vulnerable equipment; provision of an alternate/additional compressor; and use of firewater systems as backup for containment spray.

From the floor, questions and comments on the cost effectiveness of SAM measures; treatment of shutdown state, fire, and seismic event in SAMA were raised and discussed.

Effect of SAMG on the Level 2 PSA of Korean Standard Nuclear Power,
Y. Jin, K.I. Ahn ,KAERI, Korea

The paper presented status of severe accident management programmes and effect of SAMG on Ulchin Unit 3&4 (UCN 3&4). In Korea, following the policy statement announced by MOST in 2001, KHNP, the operator of nuclear plants, had been completed PSAs for all operating and is conducting severe accident management programmes for his plants. For UCN 3&4, Level 2 PSA, which had been completed by 2004, was re-evaluated reflecting revision of EOPs and preparation of SAMG. HEP in sequence 37 of SGTR was much reduced, 0.59 to 0.02568, by the revision of EOP and, accordingly, the frequency of SGTR-37 and the bypass frequency were reduced. Consideration of restoration of spray system and use of fan cooler which are described in SAMG introduced a large reduction of late containment failure frequency.

From the floor, questions and comments on the consideration of degradation of material in the evaluation of SGTR occurrence frequency, and basis of allowable time in SGTR were made and discussed.

Insight from a full-scope Level 1/Level 2 all operational states PRA with respect to the efficacy of Severe Accident Management actions
J.U. Klügel[1], S. B. Rao[2], T. Mikschl[2], D. Wakefield[2], A. Torri[3], V, Pokorny[3]
[1]Kernkraftwerk Goesgen-Daeniken Switzerland,
[2]ABS Consulting, Irvine, USA,
[3]Risk Management Associates, Encenitas, USA

The paper presented scope and structure of Goesgen PSA model, main results of PSA, and insight gained from the results. All states and all events were integrated in Level 1/Level 2 Goesgen PSAs, which include 156 initiating events for power operation consisting of internal events subdivided into LOCAs, transients, SGTR, and ATWS; internal hazards subdivided into internal floods and fires; and external hazards subdivided into airplane crash, earthquakes, external floods, and loss of service water intakes; and 173 initiating events which are related to wind and tornado etc. The result of CDF was validated by the comparison with results of the Convoy and KKB plants. Insight gained from source term analysis shows that the availability of two SGs before transferring to the reduced inventory shutdown operation states is very beneficial. In addition, pre-damage post accident actions are more important for a reduction of LERF than the “direct” SAMG.

From the floor, questions and comments on accident management measures for shutdown state; multi-SGTR; safety goal and the need to evaluate SAM for the low frequency sequence; use of safety monitor to reduce shutdown risk; effectiveness of venting system; and shock model of operator action were made and discussed.

PRA Level 2 Perspectives on the SAM during Shutdown States at Loviisa NPP
S. Siltanen, T. Routamo, T. Purho, H. Tuomisto, Fortum Nuclear Services Ltd, Finland

The paper presented SAM strategy applied to Loviisa plant, its extension to the shutdown states, and Level 2 PSA results of shutdown states. Loviisa employed an integrated ROAAM, Risk Oriented Accident Analysis Methodology, approach to power operating states. Although there are some specific aspects to the shutdown condition, e.g. low level of decay heat, missing of containment function, and existence of maintenance works, same safety functions can be applied to the shutdown condition. Among these functions, she presented the results on the mitigation of hydrogen considering the effect of forcing open of the ice condenser doors, recombiners, and igniters.

From the floor, questions and comments on water source to be injected; condition of ice condenser in the shutdown condition; impact of open CV in shutdown; hydrogen management; existence of air in the core were made and discussed.

Development of the SAM strategy for Paks NPP on the basis of Level 2 PSA
J. Elter[1], G. Lajtha[2], É. Tóth[1], Z. Téchy[2]
[1]Paks Nuclear Power Plant, [2]NUBICI (former VEIKI), Hungary

The paper presented specific design features of the plant, Level 2 PSA results, AM strategies, and two phase plant modification plan. Analyses cover all sources of potential radioactivity releases, all plant operational states, and all types of initiating events. Using Level 2 PSA results, containment failure state and their reasons were identified and possible AM measures were derived. Two AM strategies taking account of recombiners, filtered venting, prevention of the reactor cavity door damage, reactor cavity flooding, and protection of basemat melt-through were established and evaluated. For the four key elements of AM strategies, i.e. prevention of the core damage, prevention of RPV failure by in-vessel retention, ex-vessel debris cooling, and release and containment management, specific components were identified. Plant modifications is planned to be made in 2 phase approach.

From the floor, questions and comments on leakage tightness of CV; failure of cavity due to high temperature; feed and bleed for SGTR; actuation criteria of cavity flooding system; plant modification for in-vessel retention were made and discussed.

Development of Technical Bases for Severe Accident Management in New Reactors
E. L. Fuller, H. G. Hamzehee, USNRC, USA

The paper presented AM programmes for existing reactors and their basis, SAM review for new reactors by USNRC, and insights obtained from design certification reviews. USNRC considers that the AM approach based on SECY-89-012, NEI 91-04, and EPRI TR-101869 used for the existing reactors can be applied to the design certification review of the new reactors. The new reactor designs address issues identified in SECY-90-016 and SECY-93-087, i.e. hydrogen control, core debris coolability, high-pressure core melt ejection, containment performance, containment bypass, and equipment survivability. In the presentation, insights from the review of AP1000, ESBWR and EPR designs were explained.

From the floor, questions and comments on detection of ex-vessel cooling; debris coolability; basis for 24 hrs to keep CV integrity were made and discussed.

General discussion and conclusions

- The overall picture received at the end of the session is that significant progress has been made since the 2001 OECD workshop on "The Implementation of Severe Accident Management Measures" in implementing and completing Severe Accident Management (SAM) programmes in many countries for full power states.

- IAEA has published a safety guide on severe accident management programme for nuclear power plant in 2009 to provide recommendations for the development and implementation of an accident management programme. The guide contains detailed recommendations for all steps in developing accident management guideline. On the other hand, two safety guides on PSA have been approved by the Commission on Safety Standards (CSS) and are expected to be published by the end of 2009. The Safety Guide on PSA provides recommendation for performing or managing a Level 1 and Level 2 PSA and for using the PSA to support the safe design and operation of NPPs.

- The way the implementation of SAM programmes is made and the measures implemented vary as reported by the different countries presentations. All approaches cover both preventive and mitigative aspects for beyond-design basis and severe accidents at full power level. The common goal of the SAM approaches implemented is to make up strategies to mitigate the consequences of beyond-design basis accidents.

- The approaches followed in the different countries do not fit one single pattern. As a common rule, the responsibility of the plant owner for the safety of his plant remains untouched. The general aim of SAM strategy development is still to ensure containment integrity during severe accidents, also, detailed safety objectives vary. Actual differences between SAM programmes exist also in terms of:

  - status of progress of work on specific programmes,
  - the extent to which hardware modifications are part of the SAM approach at a plant,
  - the availability and scope of SAM guidance in the mitigative domain,
  - the extent to which the decision-making process at a specific plant is left in the hands of the operators in the main control room or becomes the responsibility of a specific Technical Support Centre.

- In many plants, hardware modifications have been implemented in the frame of the SAM programmes, while in most of the plants SAM guidance is implemented at least in the mitigative domain.

The development and implementation of a SAM programme has been required or recommended by the safety authorities, but not necessarily translated into rules and regulations. It is finally the duty of the safety authority to review the work of the utilities.

- Once implemented, the SAM guidelines should be considered as a living product. They should be periodically reviewed and updated, and improved if necessary, as new knowledge becomes available and new assessment methods are developed.

- A clear tendency became obvious from several presentations to the extension of the SAM programmes to low power and shutdown states. Based on Level 2 PSA results which are available in most countries for full power states and now in several countries as well for low power and shut down (e.g. refuelling outages) states, it became obvious that the core damage frequencies (CDF) might be in the same order of magnitude for both states. In recently performed studies, it was found that the risk of a large radionuclide release might be comparable or even higher for low power and shut down states than for full power operation.

- For NPPs in Finland, Hungary, and some in Switzerland an extension of developed SAM programmes to the mentioned low power and shut down states has been presented. The basic idea of existing SAM strategy application assessment was to identify the sequences or states during low power and shutdown, for which the existing SAM measures were considered inefficient or not applicable. The accidents initiating from shutdown states involve much lower primary pressure and decay heat, and may take more time until the heat-up of the reactor core starts. The shutdown states are in many ways different from power operation states: the containment may be opened, the reactor vessel head may be open, the emergency systems as well as some SAM dedicated systems may not be available due to maintenance. As well additional openings and release paths which may exist allow coolant and radio nuclides to escape from the containment through different buildings into the environment. From such situations the return to a state, where containment integrity can be ensured and mitigation actions are available and efficient during severe accidents, may be very lengthy and may require many recovery actions. Some of the recovery actions have to be started rather soon after the initiating event, as at later stages the containment condition may prevent carrying out the operations.

- SAM measures for new generation III reactors were covered in one paper. Details on the SAM guidance have not been presented.

Session 3

Modelling PRA Issues

Session Chairpersons:

J. Primet (EdF, R&D)

V. Dang (PSI)

Session summary

Session 3 addressed the modelling of the implemented SAMGs in Level 2 PSA, progress in Level 2 PSA, and the results of these studies. One contribution reported that in view of the contribution of shutdown states to risk, SAMGs were developed for these plant states. The results of the Level 2 PSA for shutdown states were also reported. A second contribution is an overview of the Swiss Level 2 PSA studies, all of which have been or are undergoing updates to account for the SAMG. It contrasted the different methodologies being used and noted some of the open issues for modelling, especially in connection with the treatment of human failure events in accident management. A third contribution presented the extension of EDF's HRA method MERMOS to allow its application for actions in severe accidents and a first application of this method. These applications highlighted that decision-making in the severe accident phase may be a critical factor, due to the distribution of responsibilities among the crew and the various crisis centres. Several international efforts to harmonize Level 2 PSA were reviewed in another contribution, covering initiated efforts in Europe, the US, OECD and IAEA. These efforts address the collection of best practices, the development of standards, and coordinated efforts to identify and address Level 2 issues. Looking forward, a fifth contribution discussed advanced modelling techniques for severe accident modelling, highlighting the potential benefits of dynamic models that integrate phenomenological severe accident models with probabilistic tools and can include operator response modelling. It noted that there are significant challenges but also a significant body of earlier and on-going work that may support the implementation of such techniques.

This session and the related discussions highlighted

- The progress in Level 2 PSA models but also significant differences in approaches. These concern the Level 1/2 PSA model interface and the modelling of the human actions in severe accident management.
- Concerning the modelling of human actions, some PSAs are treating these in a level of detail comparable to the preventive actions in Level 1 PSAs. The methods in these analyses have to some degree been extended on a study-by-study basis.
- The workshop discussions suggested that there are significant differences in the degree to which different implementations of SAMGs are procedurised. Correspondingly, there were also diverging views concerning the extent of the flexibility and discretion with which Emergency Response Teams would apply the SAMG.

Summaries of the presentations

This session consisted of five presentations. Some general conclusions are drawn from the discussions.

Accident management and risk evaluation of shutdown states at Beznau NPP

M. Richner[1], S. Zimmermann[1], J. Birchley[2]. T. Haste[2], N. Dessars[3]
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[2]Paul Scherrer Institute, Switzerland,
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Beznau NPP is the oldest NPP in operation in the world. Since start-up a set of backfits have been implemented resulting in significant safety benefit, reflected by a reduction of CDF by two orders of magnitude. Beznau has also launched a comprehensive accident management programme including a wide range of hardware modifications as well as procedure changes and improvements.

However, shutdown states, which represent a significant risk contributor for light water reactors (LWRs), were not covered by Beznau specific set of procedures, the so-called Accident Management Procedures (AMPs) until 2005. That is why several accident management features were implemented at Beznau Nuclear Power plant to improve shutdown safety. In parallel, Severe Accident Management Guidelines (SAMGs) for full-power operation were extended to shutdown states.

In support and addition to this programme, a more realistic evaluation of shutdown risk was carried out. First it was shown that core degradation does not occur at start of core uncovery but at a reactor water level dropped much lower. This extends the time window for the operators to intervene in cases of loss of core cooling. It also enables operator actions according to the Severe Accident Management Guidelines (SAMGs) to restore core cooling during shutdown conditions even after the start of uncovery, using charging pump as an alternate recovery action. Due to the long time window, recovery of core cooling is also possible using mobile equipment such as firewater pumps.

The safety benefits of these measures were assessed by updating the Level 1 and 2 PSA models. The AM measures resulted in a reduction factor of at least 2 for cold shutdown CDF and LERF.

Overview of the modelling of severe accident management in the Swiss PSAs
V. N. Dang[1], G. M. Schoen[2], B. Reer[2],
[1]PSI, [2]ENSI, Switzerland

The presentation was given in two parts. The first one presented the regulatory basis of SAMG and PSA in Switzerland, the status of implementation of SAMG and PSA and main objectives of SAM actions. The second part focused on modelling and quantification of SAM actions in Level 2 PSA, main differences with Level 1 HRA that need to be considered and some results from Swiss PSA.

The Swiss regulatory framework is constructed in three stages: Nuclear Energy Law, Accompanying ordinance and regulatory guidelines. The accompanying ordinance consists of high-level requirements and precise decision guidance for severe accident management and high-level requirements for PSA development and applications. Three guidelines complement the ordinance about PSA and SAMG:

- ENSI-A05 - PSA : Quality and Scope
- ENSI-A06 – PSA : Applications
- ENSI-B12 – Emergency Preparedness for Nuclear Installations

Implementation of SAMG followed several step, starting in 1997 with a survey study and ending 2009 by stating detailed requirements in the regulatory guideline. Full Power SAMGs were implemented in Swiss NPP from 2001 to 2006 and Shutdown SAMG from 2005 to 2009.

As far as PSA is concerned, all 4 Swiss NPPs have developed specific Level 1/Level 2 PSA covering all kinds of events (internal and external), and all reactor states, except some development which are still needed for Level 2 low power and shutdown states for certain units.

SAM actions, which have to be stated in PSA Level 2, aim at terminating core degradation, ensure containment integrity and mitigate radiological releases. Examples of such actions are: alternative water supplies, especially alignment of firewater, flood for heat removal, flood or spray for radionuclide retention and containment venting. Number of SAM actions types and cases in each Swiss Full Power level PSA were presented and differ strongly from PSA to another, mainly due type of analysis (HRA type or APET state). It was also underlined that some PSA are currently being updated, which can result in significant changes for the results which were presented.

Following specific factors influencing performance of SAM actions were presented and discussed:

- transition to new, mitigation-oriented objectives;
- increased expertise available to and within the ERT;
- Open (by necessity) aspects of the mitigative response plan (less prescriptive than EOP);
- Increased uncertainty regarding plant state;
- Need for more parties to agree, more complex decision-making process;

- Personnel radiation exposure (for local actions) ;
- Possible dependencies between MCR crew and ERT.

Some statistics about failure probability for SAM actions were presented: it appears that only a very small part of the probabilities below 0,01 and that they are in majority in the 0.01-0.1 and 0.1-1 range. These values are in general higher than the human error probabilities in Level 1 PSA. However, the results may be biased by differences in the treatment of SAM actions in the PSAs, by analysis assumptions, or by limitations of HRA methods for Level 2 PSA applications.

Some international efforts to progress in the harmonization of Level 2 PSA development and their applications (European (ASAMPSA2), US-NRC, OECD-NEA and IAEA activities).

E. Raimond[1], S. Güntay[2], C. Bassi[3], D. Helton[4], A. Lyubarsliy[5],
[1]IRSN, France,
[2]PSI, Switzerland,
[3]CEA, France
[4]US-NRC, USA
[5]IAEA, Austria

The presentation drew attention to the fact that the expectations from a Level 2 PSA may indeed be large and can include:

- validation of severe accident measures,
- achieving safety goals or acceptability of the level of risk,
- cost-benefit analysis,
- support for decision regarding plant life extension,
- identification of R&D needs for unresolved generic safety issues,
- capitalization of knowledge,
- emergency preparedness.

Such expectations require robust and validated studies and authors consider that there is still a need in the international community to share experience in the development and applications of Level 2 and that the development of standards, best-practice guidelines and state-of-the-art methods can be a useful way to do so. The paper presented a summary of several initiatives from EC, US-NRC, OECD and IAEA, which are shortly repeated hereafter:

- SARNET (Severe Accident Research NETwork of Excellence)

This EC initiative covers two period 2004-2008 (SARNET1, 51 organizations) and 2009-2012 (SARNET2, 41 organizations). The main objectives of SARNET 2 are to spread knowledge and to propose research on high priority issues concerning severe accidents.

- ASAMPSA2 (Advanced Safety Assessment Methodology : Level 2 PSA)

This is a coordination project, which started in 2008 for 3 years and gathers 22 organizations (plant operators, Vendors, TSOs, Safety authorities, etc.). The objective of ASAMPSA2 is to develop best practice guidelines for the performance of Level 2 PSA methodologies with a view to harmonisation at EU level. The expected impact is that Level 2 PSA methodologies could be used with greater confidence in the further development of SAM procedures and could greatly assist in the decision-making associated with plant life management.

- State-of-the-Art Reactor Consequence Analysis (SOARCA) project

The goal of SOARCA is to generate realistic estimates of the offsite radiological consequences for severe accidents at US operating reactors using a methodology based on state-of-the-art analytical tools. Details about SOARCA project were provided during sessions 4 and 5 and are presented in the associated section of this document

- Development of New PSA standards

In the US, a consensus standard exists for the limited scope Level 2 model (so called "LERF model"). Currently three new standards are in development that are of interest for the accident management community: low power and shutdown PSA, Level 2 PSA and Level 3 PSA.

- OECD-NEA activities

OECD/NEA CSNI Risk and GAMA WG are supporting many collaborative actions related to severe accident and Level 2 PSA. Audience was informed about two NEA/CSNI recent papers on Level 2 PSA and SAM.

- IAEA activities

The IAEA activities were presented in a specific paper in session 1. Several document references were presented, included to safety guides for development of Level 1 and Level 2 development and application final versions of which are to be published in the near future.

Extended Use of MERMOS to assess Human Failure Events in Level 2 PSA
H. Pesme, P. Le Bot, EDF, France

This contribution presented the extension of EDF's Human Reliability Analysis (HRA) method MERMOS to address Human Failure Events in Level 2 PSA. The MERMOS method applications was extended for Level 2 PSA and evaluated on the basis of case studies dealing with Loss of (Steam Generator) Feedwater accident scenarios with the unavailability of Safety Injection. The studies considered scenarios with and without Station Blackout (SBO). Experts in severe accidents provided qualitative and quantitative inputs to the case studies. The studies highlighted the role of the crisis organization as one of the key aspects specific to HRA for Level 2 HRA. In particular, the examples discussed show that the complexity of the decision circuit is one of the ways in which Level 2 Human actions may fail.

Four main characteristics of Level 2 PSA drive the modifications of the MERMOS method. First, the Emergency Operating System (EOS), which in Level 1 PSA consists of the control room operators, their procedures, and their interface to the plant, is extended to include the emergency response organization. This organization consists of the on-site plant organization, EDF resources at the national level, and local (prefecture) and national public authorities. MERMOS Level 2 PSA applications thus include in the EOS the site's MCC, the local emergency response team, and the national emergency response team. Second, the procedures of this extended EOS supplement the Emergency Operating Procedures and need to be taken into account. These include the severe accident response guide (GIAG) used by the control room operators. Third, in severe accident conditions, MERMOS considers a prognosis function in addition to the usual functions of diagnosis, strategy, and actions. It consists of evaluating not only the scenario developments but also the plausible aggravating factors and proposed countermeasures. Fourth, the lack of data (fortunately) on real severe accidents, the small number of severe accident simulations (exercises) and their limitations for collecting HRA data, and the expertise on severe accidents in EDF's PSA teams increase the reliance on expert judgment. In the case studies, inputs were elicited from experts in severe accident and members of the national emergency response team.

The example of Loss of Feedwater combined with SBO from the case studies highlights that the failure probability for depressurization of the primary (Reactor Coolant System) by opening the pressurizer valves is not insignificant. The main scenarios involve failures related to in-situ actions (outside the control room), which are penalized by the short time available (15 minutes), and the complexity of the decision circuit, which limits the support of the crisis organization. In one of the contributing failure scenarios, the MCC is not available in time to confirm the decision to depressurize. The discussion addressed the assumptions concerning how long the emergency response teams (local and offsite) need to assemble and then to become capable to make decisions.

The Role of Severe Accident Management in the Advancement of Level 2 PRA Modelling Techniques

D. Helton[1], J. Chang[1], N. Siu[1], K. Coyne[1], M.Leonard[2]
[1]US-NRC, [2]Dytona, USA

This contribution explored the potential role of advanced methods for Level 2 PSA modelling for better capturing the effects of accident management (AM) guidance on severe accident risk. Some key aspects being considered as part of an exploratory long-term research project are the ability to model human actions as part of accident management and the coupling of these actions to deterministic models of severe accident progression to obtain more realistic and higher-fidelity results. The paper surveyed potential Level 2/3 PSA approaches and current and potential Level 1 PSA approaches in terms of human response modelling. Dynamic event tree methods are further examined as a promising technology for treating the human response in Level 2/3 PSA.

The presentation noted that post-core damage operator actions have either been neglected or incorporated in subjective probability assignments. The approach of relying on a subjective mixture of deterministic analysis, experimental data, and practical knowledge in the treatment of AM facilitates the treatment of a large number of sequences. Several advanced Level 2 approaches were compared in a scoping study with traditional methods: modified traditional approaches, hybrid event approaches that couple event trees statically to deterministic tools, dynamic event tree simulation methods, and sampling-based direct simulation methods. The dynamic event tree and sampling-based direct simulation methods were identified as the most promising. Some key advantages are the direct use of MELCOR in event tree construction, the elimination of pre-determined top events provided by dynamic event trees, and the coupling of an operator response model. These address the main limitations of the subjective traditional approaches to modelling AM actions in Level 2 PSA in terms of dealing with complex system/operator interactions and ensuring Level 1 / Level 2 consistency. Dynamic PSA modelling aims for a more time-based representation of sequence evolution and the direct modelling of accident scenario development, including all relevant phenomena, operator decision-making and actions, and physical accident development.

The major difficulties in implementation include the large number of unique sequences, which requires screening out “unimportant” operator actions, and strategies for merging sequences and truncating the dynamic event tree at prescribed frequencies. Operator response models need to be further developed. In applications of the dynamic models (in the analyses), oversight is needed to catch instances where the model may enter untested regimes. Strong non-linearities are expected that may magnify small modelling errors and lead to unrealistic contexts for operator actions. The benefits for pre-core damage modelling include clear links between actions and their proximate causes, the capability to model core damage based on actual fuel response rather than surrogates, and sequence-by-sequence modelling of EOP to SAMG transitions. In the post-core damage regime, the dynamic models provide the capability to define the context for SAMG decisions on a sequence-by-sequence basis. As a result, a more realistic source term is expected, combined with improved resolution concerning the importance of AM actions that impact the source term. In order to realise the advantages of advanced Level 2 modelling approaches, additional work is needed to implement these simulations. The existing body of work discussed in the paper as well as other on-going efforts can contribute to this objective.

The discussion highlighted the role of advanced Level 2 modelling approaches as detailed approaches to address specific issues and applications rather than a replacement for the full scope of Level 2 PSA. The significant difficulties and the level of effort to develop the inputs would tend to focus the applications to specific scenarios and sequences. The contributors also noted that the paper reflects on the status of Level 2 PSAs in the U.S., which may contain a different level of modelling detail than some of their international analogues. It was noted, for instance, some PSAs discussed in the workshop already use an integrated Level 1/2 methodology where AM and SAM actions have been modelled on a sequence-specific basis. While these may not address all of the objectives of the advanced Level 2 approaches discussed, the realisation of such approaches may also be viewed as an ambitious but challenging further step.

General discussion and conclusions on Session 3

The PSA has been an input to the development and implementation of the Severe Accident measures (hardware features) and the accompanying guidelines, the SAMGs. Updates of Level 2 PSAs have been performed to account for these technical measures and the strategies, mainly for full power but in some cases also for shutdown states. The Level 2 PSA practice is diverse, due to the Level 2 PSA approach and intended applications as well as the implementation of the SAMGs themselves. Based on these Level 2 PSA experiences, there are efforts a) to establish standards and b) to collect effective PSA practices. An outline of the general discussions and conclusions are provided below:

- The intended use and applications of the Level 2 PSA should drive the selection of the Level 2 PSA approach. Although many plants have Level 2 PSAs, a number of organizations are or will be initiating Level 2 PSA studies. The advantages and limitations of the respective approaches, in view of the intended use and applications of the Level 2 PSA, should be an important factor in this selection:

- In part due to scenarios where containment hatches are open and cannot be closed quickly (loss of power), shutdown scenarios may be significant contributors to risk in terms of release frequencies. Since 2001, SAMGs have been implemented for shutdown conditions for a number of plants. The development of Level 2 PSAs for shutdown is encouraged. It provides an updated risk perspective on shutdown L2 risk, insights for, and importance relative to release scenarios from full-power operation.
- It was recommended to model SAMG in Level 2 PSA and to assess and, if needed, to optimize SAM and SAMG.
- There are different views as to whether it is appropriate to incorporate SAMG-guided action in the Level 2 PSA. For those who believe that this should be done, there is no mature set of methods for comprehensively addressing human reliability aspects for SAM measures.

- Work should be undertaken to identify what existing methods and gaps exist for extending presently used Level 1 PSA human reliability methods to Level 2 PSA / SAM. As a first step, one could try to reach consensus on the question of whether existing methods can be extended, or whether the situation for Level 2 PSA / SAM is so different that entirely new methods are required.

- Whether a SAM measure is taken in a severe accident depends not only on hardware system availability and failure but also on the ERT decision to pursue the SAM measure. When the failure of the decision-making process is considered to be one of the possible contributor to the SAM measure failure, this has to be addressed in Level 2 PSA.

- The modelling of SAMG-guided actions for Level 2 PSA needs to address a number of characteristics specific to the Severe Accident context (accounting for the needs of the PSA application). These include e.g.

- Delay from ERT / ERO / TSC assembly to effective readiness
- Transition from preventive to mitigative objectives
- Transfer of some responsibilities from the main control room to the TSC / ERT.
- Timing of the entry into SAMG, after the entry conditions are satisfied
- Concurrent use of EOPs and SAMGs, at least for some time (in some SAMG concepts)
- Distributed decision process, complex "decision-path"
- Increased uncertainty regarding plant state
- Coordination of multiple teams (e.g. local operators, fire brigade)

- Most of the available HRA methods are mainly focused on modelling the failure of preventive actions guided by EOPs, i.e. those modelled in Level 1 PSA. As a result of a lack of guidance to model SAMG guided action in Level 2 PSA has led some studies not to account for the SAMGs and actions in the PSA to the degree expected, while others have developed approaches specific to their SAM and SAMG concepts. There is therefore a clear need to develop guidance on crediting and modelling SAMG-guided actions.

- Very different views were expressed concerning how much latitude the SAMG leave to the decision-maker(s), in order to allow for the uncertainties of severe accident contexts. Some stressed that the SAMG decision criteria are very clear-cut while other views highlighted the subjectivity of the decision-making process and the possible influence of group decision-making behaviours.

- Concerning actions to recover systems and functions, the present consensus in PRA is to credit these only when they are supported by procedures. This consensus is currently oriented to the preventive domain treated in Level 1 PRA. It may be worthwhile to examine whether this should apply for the severe accident context modelled in Level 2 PRA.

- SAMG training and emergency exercises go hand-in-hand with the implementation of hardware measures and SAM guidelines. The collection of observations from training and exercises could help to establish a basis for a consistent crediting of SAMG-guided actions.

- SAM implementations differ in how decision-making responsibilities are organized and in the structure of the guidance (e.g., philosophy of the entry criteria, concurrent or exclusive application of EOPs/SAMGs, and inclusion of back-up decision criteria). Assessments of SAM effectiveness need to take the specifics of an implementation into consideration and perhaps include comparisons with other implementations.

- The validation of SAM measures from the standpoint of the necessary operator action timings could be addressed more rigorously, relative to the verification of the efficacy of hardware. Further, errors of commission (which were important in the Three Mile Island and Chernobyl accidents) are still routinely neglected in PSA. Tools exist which can be used to scope these issues in the absence of (or in addition to) a full-scope simulator.

Sessions 4 and 5

Code Analysis for Supporting SAMGs

Session Chairpersons:

E. Raimond (IRSN)

Y. Liao (PSI)

M. Leonard (Dytona)

M. G. Cenerino (IRSN)

SESSION SUMMARY

Nine papers were presented in Sessions 4 and 5, on the general subject of "Code Analysis for Supporting SAMGs." The papers addressed a wide spectrum of topics and described deterministic analysis of a several different reactor designs. The papers stimulated several useful and interesting comments and discussions.

Discussion on the various topics can be summarized in three broad areas: (1) Deterministic consequence analysis, (2) Use of deterministic analysis to identify and verify SAM measures, and (3) Investigations of particular severe accident phenomena. Important aspects of these discussions are noted below.

Deterministic Consequence Analysis

Many types of deterministic analysis presented at the workshop clearly demonstrate measureable reductions in risk have been achieved by implementation of severe accident management measures. These reductions have been achieved by crediting or adding new resources for coolant injection and heat removal to prevent core damage, and by actively mitigating the consequences of core damage. Recent analysis by the U.S. NRC (SOARCA) identified any severe accident sequences in a representative BWR and PWR that would result in a 'large early' release to the environment. That is, their results suggest the QHO (or its surrogate, LERF) is satisfied in these U.S. reactors. However, comments from the audience noted that the PWR source term (9% of the initial core inventory of Caesium released to the environment within 10 hrs after core damage) for one of the U.S. PWR sequences would constitute a large early release in Switzerland. This highlighted the observation made at other times during the workshop that interpretation of the QHO in the form of surrogate parameters, such as LERF, are not uniform across the countries participating in the workshop. A contradictory conclusion was reached in a paper presented by KINS (Korea). The Korean study found that the QHO could only be achieved if radiological release to the environment was limited by design basis leakage from containment (0.1% volume/day), and was mitigated by early operation of containment sprays.

Verification of SAM Measures

Many workshop participants expressed the view that SAMGs should encourage restoration of water to core debris regardless of the conditions of the core, or the available coolant flow rate. This was however not a consensus view. Some participants expressed the view that the detrimental effects of adding water at low a flow rate might outweigh potential benefits. In particular, if the maximum available rate of coolant injection is significantly less than the amount needed to fully quench and cool debris, the water injection may increase the hydrogen generation (and its flow rate) and the risk of early containment failure. Depending on the design on the plant and on the considered accident, late in-vessel water injection may also increase the risk of direct containment heating if the vessel is already damaged by the relocated corium and fails while RCS pressure rises after water injection.

Investigations of Particular Severe Accident Phenomena

Severe accident sequences involving induced steam generator tube rupture (SGTR) remain an important contributor to risk for some PWRs and a significant challenge for developing SAM measures. Divergent assumptions in the analysis of induced SGTR, including the probability distribution among such events as cascading failure, tube failure followed by subsequent hot leg failure or no tube failure, and the differing assumption about the number of tubes involved can lead to significantly different results. Many studies continue to assume induced tube failure only affects a small number of tubes. In contrast; analysis by PSI in Switzerland assumed failure of a single tube with a sufficiently long crack could quickly propagate to failure of adjacent tubes. This, in turn, increases to effective break area in the reactor coolant system, causing depressurization and transfer of water from the accumulators into the SG secondary. Differences in these assumptions lead to differences in the effective decontamination factor for fission product retention in the SG secondary.

GENERAL CONCLUSIONS

- Technical progress was reported in modelling severe accident and consequence analysis, permitting realistic updates or revisions to past analysis of quantitative estimates of offsite radiological accident consequences. However, applications of these models to representative severe accident sequences led to very different conclusions regarding the extent to which quantitative health objectives (QHOs) have been achieved after severe accident measure implementation.
- Differences of opinion remain in the confidence with which SAM guidance can recommend the re-introduction of water to molten core debris, during either the in-vessel or the ex-vessel phase of the accident. Concerns regarding the detrimental side effects of the interaction between water and core debris (especially at coolant low flow rates), have not been fully resolved.
- Severe accident sequences involving induced steam generator tube rupture (SGTR) remain an important contributor to risk for some PWRs and a significant challenge for developing SAM measures. The relationship between the time of this event and the time at which creep rupture occurs at other locations in the RCS (e.g., hot leg) has a first-order impact on the radiological source term to the environment. If hot leg creep rupture occurs soon after tube rupture, a substantial fraction of fission products are discharged into containment, reducing the activity available for release to the environment through the ruptured SG tube(s).

RECOMMENDATIONS

- An international effort should be undertaken to investigate and better understand the relationship between the metrics used to quantify the benefit of SAM measures toward satisfying offsite radiological consequence targets (e.g., dose, land contamination) in addition to traditional metrics available from severe accident analysis.
- Some comparison and information exchange activities regarding the technical basis for SAMG strategies should be maintained at international level, especially for in-vessel water injection, cavity pit flooding, containment spray system actuation and filtration efficiency of containment venting. These activities should focus on the balance between the advantage and disadvantages of each chosen SAMG strategies, depending on reactors design specificities. The aim of this effort would be to develop an international consensus on the effectiveness of SAM measures.

Summaries of the presentations

Best-Estimate Calculations of Unmitigated Severe Accidents in State-of-the-Art Reactor Consequence Analyses

C. G. Tinkle1[r], K.C. Wagner[2], M. T. Leonard[3], J. H. Schaperow[1]
[1]U.S-NRC, [2]Sandia National Laboratories, [3] dycoda, LLC, USA

The paper described analysis of severe accident offsite radiological releases using state-of-the-art analytical tools for accident progression and consequence analyses (primarily MELCOR and MACCS). Calculations were performed for two U.S. plants (one PWR and one BWR), and addressed accident sequences identified as the dominant contributors to core damage frequency for internal and external initiating events. Calculations of accident progression, radiological source terms and offsite health consequences explicitly accounted for operator actions to follow SAMGs and use new plant hardware recently installed as mitigative measures to address security related hazards.

The analysis is an update to past (1982) analysis of offsite consequences at U.S. nuclear power plants and demonstrated significant reductions in radiological release and its impact on public health. None of the sequences resulted in a 'large early' release to the environment, based on U.S. standards definition.

Deterministic Evaluation of Quantitative Health Objective and Target of Severe Accident Management
C. Huh[1], N. Suh[1], G. Jung[2],
[1]Korea Institute of Nuclear Safety, [2]FNC Technology, Republic of Korea

This paper examined the conditions necessary to achieve a safety goal in expressed in a form of quantitative health objective (QHO) such that an additive risk of early fatality. In particular, the common QHO that the frequency of acute fatality or long-term cancer fatality caused by accident or operation of nuclear power plant should not exceed 0.1% of public risk to other common hazards. MELCOR and MACCS deterministic calculations were described, which suggested the only way to achieve the QHO in Korea for a representative severe accident (e.g., station blackout) is to limit releases to the environment to a value corresponding to maximum allowable containment leak rate with operation of containment sprays. Releases associated with filtered containment venting, or with containment leakage larger than allowable (Tech Spec) limits, combined with the nominal frequency of severe accidents, would not satisfy this QHO.

Verification of the SAMG for Paks NPP with MAAP Code Calculations
G. Lajtha[1], Z. Téchy[1],J. Elter[2], É. Tóth[2]
[1]NUBIKI, [2]Paks NPP, Hungary

The paper described a wide range of applications of the MAAP code, which verified of the efficiency of SAMGs in the Paks NPP. Specific actions studied included:

- depressurisation of the primary circuit,
- water injection into the primary system,
- in-vessel melt retention by external cooling of the vessel,
- preventing excessive vacuum in the containment,
- preventing containment overpressure,
- decreasing fission product release using the ventilation systems.

Treatment of Accident Mitigation Measures in State-of-the-Art Reactor Consequence Analyses
C. G. Tinkler[1], K.C. Wagner[2], M. T. Leonard[3], J. H. Schaperow[1]
[1]U.S-NRC, [2]Sandia National Laboratories, [3]dycoda, LLC, USA

The paper presented the effectiveness of resources for preventing core damage or mitigating radiological releases to the environment examined in deterministic calculations. The resources included plant systems that were not explicitly represented in the plant-specific PSAs and new equipment installed for responding to nuclear plant security requirements. When these resources were credited (assumed to function), core damage was avoided in most of the accident sequences identified in the plant-specific PSAs.

Best Practices Applied to Deterministic Severe Accident and Source Term Analyses for PSA Level 2 for German NPPs
M. Sonnenkalb, N. Reinke, H. Nowack GRS,Cologne, Germany

The paper described methods applying integral deterministic severe accident analyses to study severe accident phenomena in German NPPs. The description highlighted the importance of detail in spatial nodalization for severe accident codes like MELCOR or ASTEC, and appropriate modelling of relevant fission product release paths from the NPP buildings to the environment. Experiences gained in applying MELCOR in past analyses are being applied to future work using ASTEC.

Severe Code Damage Analysis for a CANDU Plant
P. Mani Mathew, S.M. Petoukhov, M.J. Brown, B. Awadh, AECL, Canada

The paper described a special version of MAAP4 which is being applied by AECL for CANDU-6 severe accident analysis. A key feature of the CANDU version of MAAP is explicit accounting for differences in core configuration from a traditional LWR (i.e., a horizontal fuel tubes within a horizontal cylindrical moderator tank, rather than a vertical configuration). This difference in physical configuration results in significant changes in fuel damage morphology and material relocation within the reactor vessel.

Preliminary calculations indicated the time frames for CANDU6 are much longer than LWRs of a similar power level. Initial calculations were performed for wide spectrum of sequences to understand event chronology and to identify opportunities for SAM measures. The paper highlighted specific natural SAM measure aspects regarding the in-vessel retention resulting from the fact that the reactor vessel (calandria tank) is submerged in a large light water shield tank.

Time Window for Steam Generator Secondary Side Reflooding to Mitigate Large Early Release Following SBO-Induced SGTR Accidents
Y. Liao, S. Guentay, Paul Scherer Institute, Switzerland

Steam generator secondary side reflooding has been implemented in some PWR power plants as a practical severe accident management measure to mitigate fission product release for spontaneous steam generator tube rupture severe accidents. The PSI work focused on station blackout induced SGTR accidents, which would progress much faster with the reactor uncovered and fission products released much earlier than spontaneous SGTR accidents. Therefore, the plant staff would have a much shorter time in response to SBO-induced SGTR. The time window available for the plant staff to prepare mobile pump and firewater for injection of water into the SG secondary side is a critical parameter governing if this SAM measure could be successfully achieved. The work used the MELCOR severe accident analysis code to analyze the SBO induced SGTR accident progression and to characterize the boundary conditions for fission product retention onto the SG secondary side. The main hypothesis was the propagation of damage from 1 tube to multiple tubes because of jet impingement due to high pressure RCS and disruption of steam generator inlet plenum mixing due to a large tube rupture flow. This resulted in a "cascade" of tube failures allowing a quick RCS depressurization until the accumulator started to inject coolant into the reactor core. As a conclusion, the paper highlighted the inherent plant safety features, such as establishment of a pool on SG secondary side following accumulator injection, which enables effectively retention of aerosols by inertial impaction as well as turbulence deposition on SG tube and structure surfaces. The effective SG retention was found to postpone a large early release by a number of hours, making time for accident management to refill SG to probably avoid large early release of fission product to environment

The following items were discussed after the presentation:

- Some concern was raised about the possibility for the accumulator water reaching the height of the SG breach if situated at the top of the tube sheet;
- Some participant questioned the quality of calculations on the jet velocity and particle size and velocity of big aerosols impacting the adjacent tubes surrounding the failed tube.

On the Effectiveness of BWR Control Rod Guide Tube Cooling as a SAM Measure for BWRs
W.-M. Ma., C.-T. Tran, KTH, Stockholm, Sweden

The paper reported the preliminary calculations performed to evaluate the potential for cooling core debris in the lower plenum of a BWR by operating (recovering) coolant flow through forest of control rod drive tubes (CRGTs). Unlike traditional forms of debris bed cooling, which provided direct contact between debris and water, this method would be in direct. Cooling would involve heat transfer from the debris bed through the walls of the cylindrical CRGTs and carried away by water flowing within the tubes. A simple, lumped-parameter model was developed for the MELCOR computer code to investigate the amount of coolant flow necessary to cool the debris bed. The CRGT flow was found to be adequate if it was restored soon after reactor scram; much larger flow rates were necessary if coolant flow was restored 2 hours after the initiating event (SBO).

Ex-Vessel Corium Management for the VVER-1000 Reactor

B. Kujal, Nuclear Research Institute Řež plc, Czech Republic

This paper was unfortunately not presented due to illness of the author. However, it is shortly summarised below to complement the related presentations made in the session. The paper describes an extensive work to analyze the effectiveness of ex-vessel corium management measures in the VVER-1000 containment. The VVER 1000 containment is built on the non-hermetic lower part of reactor building: The thickness of containment basement slab is only 2.4 m. The paper raises a concern due to the limited thickness of the basement slab that in the course of severe accident corium can melt through and hence may cause release of fission products into the non-hermetic lower part of the reactor building and eventually into the environment. As a result of this threat, two strategies are proposed for ex-vessel corium management in the VVER-1000 reactor cavity: i) corium spreading out of the cavity on containment floor and ii) water cooling of melt pool. Extensive calculations with the CORCON and the MEDICIS corium concrete interaction codes showed that the most effective measure is the combination of both the strategies. Nevertheless, this procedure does not provide assurance of terminating the corium-concrete interaction definitely, however, only retards the corium penetration through containment basement. On the other hand, supplementary studies conducted using the MELCOR code confirm the melting through and break down of containment basement slab as expected in one or a few days from the start of the accident. As a consequence, a massive fission products release is predicted to occur into the environment. A strategy with an attempt to mitigate the massive environmental release is proposed and presented. It consists of the following remedial measures: reinforcement and additional sealing of seven doors leading from lower part of the reactor building into environment, removal of cover and lids on the intermediate floors to facilitate corium transfer to the final destination, containment depressurization before containment basement slab failure, assuring long term heat removal from containment/reactor building and prevention of hydrogen detonation.

Session 6

Decision-making, Tools, Training, Risk targets and Entrance to SAM

Session Chairpersons:

D. Helton (US NRC)

P. Le Bot (EdF)

Session summary

Five papers were presented during Session 6. These papers covered a broad range of topics including:

- transition criteria used for moving from emergency procedures to severe accident procedures,
- an accident diagnosis and emergency response decision-making tool,
- views on the use of risk targets and safety goals in view of IAEA recommendations, and
- development, validation and training of SAM.

These papers/presentations are summarized below.

Summaries of the presentations

Criteria for the Transition to Severe Accident Management

B. Prior, Jacobsen Engineering Ltd (JEL), UK

The presentation on criteria for transition to SAM has provided a good survey of transition criteria and considerations. He has demonstrated that a wide, but explainable, variation exists with regard to the transition criteria employed. A WGAMA report providing additional information on this topic is forthcoming. An aspect of this issue, that is likely to receive more attention, is the extension of SAM entrance criteria to the shutdown states. A member of the audience noted that the OSSA criterion has changed relative to what is shown in the paper/presentation, but the new criterion has not yet been published.

Use of The Software Module Sprint in The Netherlands

M. Slootman, NRG, Arnhem, The Netherlands

The presentation on the use of the Sprint code in the Netherlands provided an overview of a useful tool for translating the state-of-knowledge in Level 2 PSA and deterministic severe accident analysis in to an accident diagnosis and emergency response decision-making tool. Of particular note was the strengths that Bayesian Belief Nets Offer in terms of probabilistic modelling of alternate scenario outcomes and accounting for missing or incomplete information. Limited discussion following the presentation focused on the need for understanding the strengths and limitations of the tool, and ensuring appropriate training is enacted (e.g., familiarizing TSC members with the probabilistic concepts employed).

Safety Goals and Risk Targets for Severe Accidents in View of IAEA Recommendations

J. Vitázková, E. Cazzoli , CCA, Switzerland

The presentation outlined international usage of safety goals and targets. It included a view as to how this usage is inconsistent with IAEA guidance and does not appropriately account for effects other than human health (specifically land contamination). The paper went on to provide illustrative offsite consequence analysis and recommended a means of adopting a risk target definition that utilizes the INES scale, through which consistency with IAEA guidance is promoted. It suggested that the definition of safety goals and risk targets is general enough to be used for both existing and future plants. Some ensuing comments related to clarifying use of risk surrogates relative to complementary deterministic goals, and acknowledgment of the benefit of debating these issues.

Development, Validation and Training of Severe Accident Management Measures
A. Torri, V. Pokorny, and U. Lüttringhaus, Risk Management Associates, Inc., USA

The presentation provided information on the historical development of, and status of design-basis emergency procedures and SAM measures and the linkage between the two. The remarks provided in the presentation focused attention on the need to consider errors of commission (based on the Chernobyl and TMI accidents), and to prioritize activities toward making the reliability of SAM guidance commensurate with that of hardware through better validation and training. The paper pointed out that many important accident sequences coming out of PSAs lead to core damage within 2 hours when for 70% of the time the emergency response organisation is not fully functional to make decisions and the control room operators must have the instruction and training much like they have for EOPs to manage the early accident phase alone. The paper went on to describe weaknesses in the current means of validating the use of SAM guidance by the plant operators, and provided a proposal for how this could be dealt with through the coupling of an operator action model with an accident simulation tool. A demonstration of this tool was provided during the presentation. Also of particular emphasis in Dr. Torri's arguments, particularly in the additional comments provided during the presentation, is the notion that the existing practices do not appropriately account for the necessary timings of operator action in relation to accident sequence evolution in guidance development, validation, and training.

Severe Accident Training in Spain: Experiences and Relevant Features
R. Martínez[1], J. Benavides[2], J. M. de Blas[3], M. A. Catena[4], I. Sol[5],
[1]Tecnatom,
[2]C.N. Trillo,
[3] C.N. Santa María de Garoña, Nuclenor
[4]C.N. Almaraz,
[5]A.N. Ascó-Vandellós II
Spain

The presentation covered severe accident management training activities in Spain, including description of a rotating curricula for the annual retraining. It also described hardware and software upgrades that have been being made to the technical support centres of each NPP, as well as updates that have been recently completed or underway for the NPP's severe accident management guidance. Finally, it described the Tecnatom's implementation of a full-scope (meaning both design-basis and severe accident capabilities) simulator at the Laguna Verde NPP in Mexico, which uses the MAAP code to augment the previously existing design-basis simulation capability.

General discussion and conclusions

- The transition from emergency operating procedures to severe accident management domain varies by plant designs. These variations are generally explainable. An upcoming WGAMA report will provide views on the appropriateness of core exit temperature as the transition criteria in use.
- Work on the use of Bayesian Belief Network (BBN)-based tools for real-time source term estimation and emergency response continues, and these tools have seen specific and positive use in the Netherlands.
- A diverse set of views exist on the appropriateness of current risk targets and safety goals, including the appropriateness of surrogates (e.g., large early release frequency) used to show compliance with these goals.
- The validation of SAM measures from the standpoint of the necessary operator action timings could be addressed more rigorously relative to the verification of the efficacy of hardware.
- Training and facility upgrade continues to be an important aspect of SAMG implementation.

Session 7

Design Modifications for Implementation of SAM

Session Chairpersons:

J. Primet (EDF)

A. Lyubarskiy (IAEA)

Session summary

Three presentations were delivered during this relatively short session. Two of the presentations discussed the proposed design changes that will increase the efficiency of severe accident mitigation and the third provided the details of the SAM procedures developed for shutdown conditions.

All presentations were appreciated by the participants of the workshop and several clarification questions were asked following each presentation.

It should be mentioned that during the discussion after presentation of the paper "Design modifications of the Mochovce units 3 & 4 dedicated to mitigation of severe accident consequences" it could be a common practice to evaluate the safety benefits of the proposed design modifications using results of Level-2 PSA and/or cost-benefits analysis, whenever it is possible.

The brief summaries of the papers are provided below.

Summaries of the presentations

A Novel Process for Efficient Retention of Volatile Iodine Species in Aqueous Solutions during Reactor Accidents,

S. Guentay[1], H. Bruchertseifer[2], H. Venz[3], F.Wallimann[3], B. Jaeckel[1]
[1]Paul Scherrer Institut, 5232 Villigen-PSI
[2]Private consultant, Frick,
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In severe accidents, elemental iodine and organic iodides are the main gaseous iodine species in the containment atmosphere. Although a great progress in the understanding and modelling of several basic aspects of iodine chemistry has been achieved, there still exists a deficit in the scientific understanding of the underlying processes, which determine ultimately speciation and magnitude of the gaseous iodine species in the containment; e.g., elemental iodine and organic iodides. Long-term research has not led to a consensus within the international research community on the generation mechanisms of highly volatile organic iodides and numerous dedicated research projects did not lead to effective measures to provide a sufficiently good retention of highly volatile organic iodides after their thermal and radiolytic generation in the containment. The specific research and development programs conducted for Filtered Containment Venting Systems (FCVS) already installed at many plants to avoid containment failure at high pressure already demonstrated high retention of the particulates, including metallic iodides by these engineered systems. However, demonstration of the high retention of volatile gaseous iodine species and in particular organic iodides under certain conditions was either not secured or not systematically studied.

In order to manage iodine retention during, a severe accident a new efficient technical process leading to a fast, comprehensive and reliable retention of volatile iodine species in a containment of a nuclear reactor during a severe accident has been developed in Paul Scherrer Institut (PSI).

The PSI research demonstrated that the concurrent use of a phase transfer catalyst, specifically, Aliquat336 eliminates the problems of unsatisfactory, undefined and ineffective retention of organic iodides in the existing designs of FCVSs in a wide range of severe accidents conditions. At the same time the researches has proved that the efficiency of the containment venting filter for removing aerosol particles and iodine is not impaired when doped with the additive mixture of Aliquat336.

This research has resulted in the development of a passive add-on to existing containment venting filter systems. The hardware modification of existing containment venting filter systems is also proposed in the paper. The implementation of the novel process will lead to significant safety benefits and will not require major changes in the existing designs of FCVSs.

Design modifications of the Mochovce units 3 & 4 dedicated to mitigation of severe accident consequences, providing conditions for effective SAM

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The paper outlined the design modifications foreseen in the four units of VVER440/V213 (Bohunice units 3 and 4 and Mochovce units 1and 2) which are in operation in Slovakia and for those planned for the additional two other units (Mochovce units 3 and 4), which are already in an advanced state of construction and are candidates for completion in the near future.

The NPP operator initiated the projects focused on the identification of the potential design modifications aimed at enhancement of mitigation capabilities for the consequences of the severe accident and at development of SAMGs (Severe Accident Management Guidelines).

After the decision to complete Mochovce 3 and 4 units (MO34), specification of the relevant design modifications of these units was the first priority. The MO34 units are planned to be put into operation in 2012/2013. Within the completion phase, the full scope SAMGs will be developed, tested and included into basic set of operation procedures. The paper presents and summarizes the proposed design modifications that will be further considered in the SAM guidelines.

The following key modifications are proposed for the incorporation into the design of MOV3, 4 units:

1. Modifications aimed to manage containment atmosphere
    - Group of measures to manage hydrogen concentration inside containment
    - Measures to prevent decompression of the containment
2. Modifications aimed to enhance in-vessel retention of corium
    - Modification of shielding at the bottom of the reactor pressure vessel
    - Provision of sufficient coolant inventory and circulation of coolant in the channel along the RPV wall
    - Modification of the drain line from the reactor cavity
3. Modifications aimed to improve the mitigation of the severe accidents when reactor vessel is opened
    - Adding of delivery pump for supply of coolant into the spent fuel pool or into the open reactor.
4. Installation of external sources of coolant
    - A system of three tanks together with all necessary auxiliary systems for mixing of the solution, heating, draining and operation, appropriate pipelines and valves.
5. Modifications aimed to increase reliability of electricity supply for the systems used for severe accidents mitigation, and
6. Measures aimed at improved monitoring of the parameters needed to control of severe accidents.

The proposed modifications of the original design of the Mochovce 3 and 4 units dedicated to severe accident mitigation are expected to provide sound basis for effective SAMGs and finally the effective mitigation and control of severe accidents.

Development of shutdown severe accident management guidelines (SSAMG) for the Leibstadt NPP

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In 2004 the Leibstadt NPP introduced SAMGs that cover all plant operating states from full to low power with the exception of shutdown conditions. Following the completion of the Leibstadt Shutdown PSA, the Swiss authority has required the development of additional Shutdown SAMG (SSAMG) by the end of 2009.

The important strategy and main procedure for severe accident management of the Leibstadt BWR-6/MARK-III boiling water reactor is containment flooding. The same strategy is implemented for the shutdown conditions although specific difficulties could be expected due to removed of the hatch that otherwise prevents an effective flooding of the containment.

As a basis for managing core degradation situations, understanding of plant-specific severe accident conditions are compulsory. Based on the Shutdown PSA results a set of eight scenarios was defined and analyzed using the MELCOR/MELSIM code equipped with Leibstadt specific complete models of all safety systems, all relevant plant I&C systems, and consideration of management actions.

The Leibstadt specific insights of these analyses are currently utilized to expand the existing EOP and SAMG procedures to cover the shutdown conditions.

General discussion and conclusions

- There is still a room for new, alternative measures to support SAM. In particular, as many efforts have already been carried out to reduce frequency or conditional probability of containment failure, more attention could be focussed to improve actual measures or to investigate new ones capable of reducing the severity of potential releases by retaining the fission products inside the containment.
- Before a plant starts its operation, there are more possibilities to make comprehensive hardware modifications and to optimise SAM. Thus, a good practice would be to start considering SAM as early as possible in the design and construction phases;
- Shutdown states have some particularities that are to be taken into account for development of shutdown Level 2 PSA, such as longer delays in general, various possible initial conditions, including status of containment hatch and RPV status (open or not).
- It may be necessary to develop specific entry criteria for SAMG to address different potential initial conditions and taking into account the fact that core exit thermocouples may be unavailable during shutdown.
- For many plants, shutdown states constitute significant contributors to risk (CDF and/or LERF).
- A comprehensive SAM programme should address the shutdown states specifically. This could lead to adapt existing Full Power SAM measures or define new shutdown-specific ones. A Level 2 PSA including shutdown states could be a useful tool to support this work.
- Level 1 and Level 2 PSAs are good tools to evaluate safety benefit of already implemented or potential SAM Measures. They can be useful in the earlier stage of SAM programme development to rank different alternatives in terms of safety benefit and cost/benefit ratio and to support decision-making.

Session 8

Physical Phenomena affecting SAM

Session Chairpersons:

F. Kappler (EDF)

D. Leteinturier (IRSN)

Summary of the session

Eight papers were presented during the session and the technical scope of the session appeared somewhat larger than suggested by the title. The papers covered:

a) Physical phenomena related to debris coolability and modelling,

b) Physical phenomena related to the source term in Severe accident,

c) Status of SERENA experiment on steam explosion,

d) Conclusions of the OECD /NEA workshop on in-vessel Coolability,

e) Example of strategies implementation on SAM examples.

General outcome of the accompanying discussions showed that while there is a common understanding on the physical phenomena, the comprehension of importance given to the phenomena dealt in the session by the different countries may largely vary and may therefore naturally impacts the strategies of SAM in the countries.

Examples given on SAMGs optimization have raised the question of having a clear harmonized reference document recalling the basic principles of SAM and the degree of freedom given in this domain (with regard to complexity, information, etc.).

- Physical phenomena dealing with debris coolability and modelling

Three presentations were provided on the debris coolability which focussed mainly on Swedish BWRs.

Many experiments have been performed in the field of corium concrete and fuel coolant interactions, within which some important issues have already been concluded but others are still under investigation.

The presentations showed an improved understanding of debris bed behaviour in a pool of water and demonstrated possibilities for debris coolability. Based on a conservative-mechanistic modelling of the debris agglomeration state it may be well possible to expect the debris coolability in a very deep pool of water (> 10 m). However, it remains necessary to complete the approach with regard to the reactor case involving shallower water pools and other debris temperatures.

- Physical phenomena related to the source term in severe accident

Presentation made by the representative of JAEA provided an interesting complementary set of data for fission product release from fuel under SA conditions obtained from experiments aiming at measuring the pressure influence on Caesium releases from PWR UO2, ATR/MOX and BWR UO2 fuel subjected to high burn up. Results show that pressure has minor influence on the fission product release from MOX and ATR fuels; however, a 30% reduction was measured from UO2 fuel. Impact of the experimental findings on the general expression of the global fission product source term should thus be minimal. However, the investigations have provided opportunity to the international community to evaluate the radiological consequences using a more validated set of data at elevated pressures.

- Status of SERENA experiment on steam explosion

The presentation made by CEA provided the status of the former OECD SERENA 1 and of the present OECD SERENA 2 programmes. Objective of the currently undergoing programme is to evaluate the risk of confinement failure due to corium ejection in a flooded reactor pit in case of reactor pressure vessel breach.

- Conclusions of the OECD /NEA workshop on in-vessel Coolability

The reader should refer to the resume of the in vessel coolability Workshop introduced below for details. The main outcome of the workshop is the consensus expressed by the majority of the participants favouring for water injection without consideration of flow amounts.

Concerning in-vessel reflooding a large common practice has been established favouring for reflooding with all the available means without considering any restrictions for flows. Even if the consequence of a non-reflooding would automatically involve vessel failure and possible consequent melt concrete interaction, the examination of the potential negative effects has to be undertaken seriously. Objective is to identify, on a realistic basis, if any potential cliff edge effects could lead to early large releases.

Following the same objective of identifying any potential coolability threshold effects, SAMG related to reactor cavity flooding before a potential vessel failure will have to be further investigated with regard to the results of the dedicated test programs (OECD SERENA2 and KTH DEFOR).

- Example of strategies implementation on SAM examples

Two presentations were made in this area:

The Korean presentation from KINS and KAERI dealt with an improved molten core strategy in SAMG on the basis NPP KORI 1, and ULCHIN 1-2. The approach presented raised the idea of taking advantage of a high probability of return of external power sources after a SBO scenario to favour core or corium coolability rather than a quick depressurization process. The conclusion of the study concluded on a better corium concrete erosion stabilization than with the strict application of SAMG depressurization chart. The presentation led to an animated discussion with regard to not aggressively follow the depressurizing principle in SAMGs. Controversial discussions also started related to the MCCI model to estimate the basemat erosion, stressing out the issue of code qualification for very difficult phenomena on which the R&D programme has not yet concluded.

The second presentation by the Kurchakov Institute of Moscow presented the analyses of the advantages of the secondary feed & bleed for recovering the core coolability. The MELCOR simulation provided very interesting results showing a quite good recovery and stabilization of the situation without reaching the core relocation. Nevertheless the analysis needs to be completed regarding engineering feasibility and implementation time of the secondary side feed & bleed in SAMGs (very short grace period) or EOPs (anticipation process in the EOPs of SAMGs type decisions).

Summaries of the presentations

Experimental Investigation of Melt Debris Agglomeration with High Melting Temperature Simulant Materials
P. Kudinov, A. Karbojian, C.-T. Tran, KTH, Stockholm, Sweden

Reactor cavity flooding is the cornerstone of severe accident management strategy adopted in Swedish and Finnish BWRs in case of a severe accident involving core melting and reactor vessel melt through. It is assumed that the melt ejected into a deep water pool will fragment and eventually form a coolable porous debris bed. If the coolability cannot be achieved then corium debris will reheat, remelt and ultimately attack the containment base-mat, by which the containment integrity might be threatened. Coolability of the debris bed depends on its properties as a porous media. Agglomeration of the debris in the process of fuel coolant interaction and further formation of debris bed may significantly affect hydraulic resistance for the coolant ingression inside the bed and thus have an effect on coolability of the corium debris. Although agglomeration of debris and formation of “cake” were observed in previous fuel-coolant interaction (FCI) experiments with prototypic corium mixtures and with corium simulant materials there is a lack of understanding of the governing physical phenomena. Presented work is a part of the DEFOR (Debris Bed Formation) research programme initiated at the Division of Nuclear Power Safety (NPS) Royal Institute of Technology (KTH). The aim of the DEFOR programme is understanding and quantification of phenomena that govern formation of the debris bed in different scenarios of corium melt release into a deep water pool.

First of a kind systematic experimental data on the mass fraction of agglomerated debris as a function of water pool depth was obtained in the experiments with high melting temperature simulant materials. Particle size distribution is in a good agreement with the data from the FARO fuel coolant interaction experiments with corium, which confirms that the simulant material well represents corium fragmentation behaviour.

Main finding is that fraction of agglomerated debris decreases rapidly as the depth of the coolant is increasing. Debris collected at depth 1.5m was completely fragmented in all DEFOR-A experiments. The highest mass fractions of agglomerates were obtained in experiments with relatively small jets and relatively high water subcooling and melt superheat. Further investigation of the mechanisms which lead to such result is necessary. Preliminary analysis suggests that steam production rate and upward steam flow may significantly affect sedimentation velocity of the particles and eventually fraction of agglomerated debris.

Approach to prediction of melt debris agglomeration states in a LWR severe accident
P. Kudinov, KTH, Stockholm, Sweden
M. Davydov, Electrogorsk Research and Engineering Center on Nuclear Power Plants Safety (EREC), Electrogorsk, Russia

The goal of the presentation was to show the development and validation of approach for prediction of debris agglomeration in severe accident analysis. Key element of the proposed approach is so called "debris agglomeration mode map" which is obtained by parametric study for prototypical ranges of conditions of severe accident in Swedish type BWR. Present work is focused on further development and validation of simulation tools used in development of the map for prediction of different agglomeration states of debris at various conditions of melt coolant interaction.

The VAPEX code is used as computational model for the fuel coolant interaction (FCI) simulations in the present work. Thermo-mechanical state of the debris immediately before deposition (pre-deposited) on the debris bed is considered as the main factor for onset of different agglomeration states. High sensitivity of pre-deposited state of the melt debris to the parameter of melt-coolant interaction and especially to jet breakup state was identified. Two different possible mechanisms of melt jet breakup are considered; (i) erosion of jet side surface due to stripping of Kelvin-Helmholtz instabilities, and (ii) leading edge breakup due to Rayleigh-Taylor instability. Epistemic uncertainty in pre-deposited state of corium debris due to the influence of different states of jet breakup is addressed in the work with bounding approach. Validation of simulation methods is performed on the data from specially designed for study of melt debris agglomeration experiment DEFOR-A (Debris Bed Formation and Agglomeration). In the DEFOR-A experiment up to 3 litres of high density, high melting temperature oxides mixture simulating corium were poured in a test section filled with water. Comparison of experimental and simulation data shows reasonable agreement and degree of conservatism in agglomeration prediction with taking into account the influence on agglomeration of epistemic uncertainty in the jet breakup state and intrinsic uncertainties of the experiment. Validated tools were applied for prediction of debris agglomeration in various plant prototypic conditions of melt ejection during the severe accident.

One of the factors which may significantly affect ex-vessel debris bed coolability is debris agglomeration. There are considerable aleatory and epistemic uncertainties in scenarios and physical phenomena of the debris agglomeration and cake formation. Therefore, in the present work conservative-mechanistic approach for quantification of the debris agglomeration state map has been developed. The approach is based on combined use of conservative assumptions in modelling and mechanistic simulations tool the VAPEX-P code. Experimental data from the DEFOR-A experiments are used for development and validation of semi-empirical conservative-mechanistic closure. It is demonstrated that conservative treatment of epistemic uncertainties in agglomeration phenomena and aleatory uncertainties in scenario (melt properties and superheat) creates sufficient margin and simulation data are enveloping the set of physically reasonable mass fractions of agglomerated debris obtained at various conditions. Application of the developed models to the plant accident conditions allows quantification of "partial agglomeration" domain on the agglomeration state map. Plant scale analysis confirms that it is possible in principle to achieve completely fragmented debris bed within the present design of Swedish BWRs. Important and encouraging finding is that mass fraction of agglomerated debris reduces rapidly with increasing of the pool depth or decreasing melt jet diameter even if there is a considerable degree of conservatism in the analysis.

OECD SERENA phase 2: A Fuel Coolant Interaction Program devoted to ex-vessel situation reactor case

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The presentation presented the results of Phase 1 of OECD SERENA programme on fuel-coolant interaction under the aspect of the implications that uncertainties on FCI phenomena may have an impact on the predictability of steam explosion induced loads. SERENA, in its Phase 1, produced a status of the predictive capabilities of the FCI codes through comparative calculations of most relevant existing experiments and reactor cases. All the codes were able to calculate reactors situations. Concerning in-vessel steam explosion, all the calculated loads are far below the capacity of a typical pressure vessel, which allows concluding that the safety margin for keeping in-vessel FCI induced loads is sufficient. For ex-vessel steam explosion, calculated loads, although low, are partly above the capacity of typical cavity walls. The scatter of the results have raised the problem of the quantification of the safety margin for ex-vessel FCI. OECD SERENA phase 2 has started at October 1, 2007 and will be finished September 30, 2011. The role of void (gas content and distribution) and corium melt properties on initial conditions (pre-mixing) and propagation of the explosion are the key issues to be resolved to reduce the scatter of the predictions to acceptable levels. OECD SERENA phase 2 is designed to resolve the uncertainties in these issues by performing a limited number of well-designed tests with advanced instrumentation reflecting a large spectrum of ex-vessel melt compositions and conditions, and by the required analytical work to bring the code capabilities to a sufficient level for use in reactor case analyses. These goals will be achieved by using the complementary features of KROTOS (CEA) and TROI (KAERI) test facilities including fitness for purpose oriented analytical activities.

Improved Molten Core Cooling Strategy in a Severe Accident Management Guideline

J. Song[1], N. Suh[2], C. Huh[2],
[1]KAERI, [2]KINS, Korea

The basic philosophy of SAMG is to utilize the available equipments to minimize the consequences of severe accidents under given circumstances. The experience from reviewing the SAMG by KINS (Korea Institute of Nuclear Safety) tells us that the present SAMG provides a reasonable guideline to cope with severe accidents in harsh conditions, but it is not clear whether these SAMGs could really contribute to mitigation of severe accidents or not. A flooding of the reactor cavity is suggested as an accident management strategy. However, the success probability of a stabilization of the molten core is still subjective due to the complexity of the phenomena including the molten core concrete interaction and the energetic fuel and coolant interaction, which are still unresolved safety issues.

The analysis above provides three insights about what is lacking in the present strategy.

- The first one is the need of an appropriate way to detect either the breach of the reactor vessel or discharge of corium into the reactor cavity. It would be very helpful if one can implement a proper instrumentation to detect a breach of a reactor vessel and the subsequent corium discharge into the reactor cavity.
- The second one is the need for a calculational aid, which would give a direction as to whether one should initiate a pre-flooding or a post-flooding. If there is little chance of delaying the failure of the reactor vessel by a pre-flooding, for example, due to a shortage of the water inventory or an inadequacy of the reactor vessel structural details regarding the insulation geometry in providing effective path for steam flow, there is clearly no need for pre-flooding the cavity.
- The last one is that when an operator should depressurize the reactor coolant system, should he carefully consider an optimal number of valves to be opened relying on a calculation aid, since such delicate operations can easily affect the timing of reactor vessel breach and coolability of the molten corium in a reactor cavity. Therefore, a calculation aid at the site is compulsory for implementing the depressurization strategy to determine the optimal capacity of depressurization.

From the evaluation of the molten core cooling strategies as implemented in the SAMG of the operating Korean plant, it was observed that the current SAMG has weak points in handling the cooling of the molten core either inside the reactor vessel or inside the reactor cavity. To improve the current SAMG, it was suggested to have an appropriate way to detect either the breach of the reactor vessel or discharge of corium into the reactor cavity, and a calculation aid which would give us a direction as to whether initiation of a pre-flooding or a post-flooding is to be decided. It was also suggested that an optimal choice of depressurization capacity would delay the timing of the reactor vessel breach and increase the coolability margin of the molten corium in a reactor cavity, which can be easily implemented using a calculation aid.

Main Outcomes of the OECD/SARNET Workshop On In-Vessel Coolability
B. Clement, IRSN, France

Since, the In-vessel coolability workshop was held a few weeks before this workshop only the outcome of the workshop is reported in this presentation. Therefore, the reader should refer the proceedings of this workshop for the details.

The presentation summarized the main outcomes as introduced below:

- Present studies reinforce the view that introducing water in a degrading core (the reflooding issue) is not straightforward due to many not resolved issues, such as:
    - The efficiency of reflooding for significantly delaying or stopping core degradation is not demonstrated for all situations;
    - In particular effective cooling becomes increasingly problematic as the core degradation escalates
    - Thorough investigations on degraded core reflood taking into account available experimental data and analytical work resulted in a preliminary reflood map to identifying main parameters influential for in-core coolability
    - About 1g/s/rod flooding rate was given as a guideline figure for minimum water flow rate (that given value was deeply questioned during the ISAMM workshop)
    - In addition to the phenomenological issues related to cooling a degraded core, the probability for recovery of water sources has to be addressed
- Similarly, presented results reinforce the view that trying to cool the RPV externally to assure In-Vessel Retention is also not straightforward since:
    - The maximum amount of molten corium that can be retained in the RPV lower head has been estimated by different methods to lie between about 30 and 100% of total core mass – at a first glance, not all the results seem to be consistent, but for small and medium size reactors below 1000 MWe size, there is a good prospect for success.
- The possibility of stopping/delaying the progression of a core melt accident by the use of a recovered water source or taking benefit of specific engineered systems is taken into account in a number of PSA studies:
    - It is understood that the plant and its engineered systems are not designed specifically for severe accident conditions, and there is no guaranteed successful cooling. The measures are very plant specific.
    - In addition to the phenomenological issues related to cooling of a degraded core, the probability for recovery of water sources has to be addressed.
    - The uncertainty on the likelihood to stop the progression of a core meltdown by water injection is generally considered high and however, depends on reactor specific features.
    - This need calls for a sustained R&D effort, both on experimental and analytical point of views.

- Ongoing, starting or planned experimental programmes address the coolability issue in different configurations, i.e. reflooding of bundles, debris beds, molten pools, RPV external cooling.
- Difficulty with present models persists in the assurance of reliable prediction of cladding oxidation runaway – oxidation of melts to be or not to be triggered by reflooding attempted during early core degradation and prediction of the related many thermal-hydraulics phenomena.
- Code developments are promisingly directed towards a more mechanistic approach using porous medium modelling able to treat different configurations; – validation is expected again the results of ongoing experimental programmes
- Transposition of results to reactor scale, where multi-dimensional effects are expected to become important, needs to be evaluated; however, larger scale experiments are probably not feasible.

• The questions of uncertainty and adequacy of the codes were discussed, revealed some divergence of view:
  - While some irreducible uncertainty is unavoidable, uncertainties should be interpretable in terms of inherently stochastic effects or to modelling limitations that point out to needs for new data.

• Another way to cope with uncertainties is to implement specific engineered features and/or management procedures to act on influential parameters such as increase the available water flow rate, specific examples were given during the workshop:
  - There are good prospects for external RPV cooling in VVER-440/213,
  - Use of spray found to be efficient for Sizewell PWR for reducing source term,
  - Potential of CRD flow to cool molten pool in the lower plenum of BWRs,
  - Feedback experience from the analysis of safety cases of NPPs having, planning and/or contemplating the implementation of specific engineered features would be of great benefit.

Simulation of Ex-Vessel Debris Bed Formation and Coolability in a LWR Severe Accident
S. Yakush, Institute for Problems in Mechanics, Russian Academy of Sciences, Moscow, Russia
P. Kudinov, KTH, Stockholm, Sweden

The presentation reported that severe accident management strategy for Swedish type BWRs adopts reactor cavity flooding for termination of ex-vessel accident progression. It is assumed that core melt materials ejected from the reactor vessel into a deep pool in the reactor cavity will be fragmented, quenched and will form a porous debris bed coolable by natural circulation. A criterion generally accepted for successful long-term cooling of the porous medium with decay heat release is that the flow rate of coolant through the debris bed should be sufficiently high, so that no local dry out occurs. The possibility of local dryout occurrence in the natural circulation (gravity-driven) flow is determined by the distribution of coolant and vapour volume fractions which depend on debris bed geometry, as well as on the properties of the porous medium (particle size, porosity, homogeneity etc).

Numerical simulations performed by DECOSIM code have been performed for two severe accident scenarios. In the case of gradual melt release, it is shown that “self-organization” plays an important role in particle sedimentation and debris bed formation. Generally, interaction of particles with the natural circulation flow results in lateral spreading of particles over the pool bottom, which prevents formation of a tall compact debris bed and improves coolability of debris. It was shown that smaller particles are more affected by the flow, so that their lateral spreading is more pronounced. A consequence of this is that debris bed expected to be non-homogeneous, both in vertical and horizontal directions in case of polydisperse particles. Implications of this effect for debris bed coolability have to be studied.

In the scenario of massive melt release, it is expected that the debris bed will have some heap-like shape, although, ad hoc specification of the shape is used currently in all coolability studies. Here, Gaussian-shaped debris bed is considered, for which the effects of particle diameter, internal porosity and "cake" are studied. It is shown that generally, dryout in the heap-like debris bed occurs more readily than in an equivalent flat layer, even despite the fact that side ingress of coolant is possible. Formation of a low-permeability "cake" on the top of debris bed has a pronounced negative effect. The effect of encapsulated particle porosity on the coolability of debris bed was found to be system pressure-dependent and requires more thorough analysis.

Substantiation of strategy of water supply recovery to steam generators at in-vessel severe accident phase for VVER-1000 Balakovo NPP
A.Suslov, V.Mitkin, RRC "Kurchatov Institute", Russia

The presentation reported that the Severe Accident Management Guidance (SAMG) for Balakovo NPP, Unit 4 with VVER-1000/V-320 reactor was developed in 2008. It also provided a brief description of SAMG development and reviewed the current related activities.

Results of PSA Level 1 for the units of Balakovo NPP show that initial events and failures leading to dryout of steam generators in secondary circuit make a large contribution into frequency of core meltdown. In such scenarios, the decay heat cannot be removed from the primary circuit and consequently the severe accident begins at high primary pressure. In these scenarios, it is necessary to consider possibility of SG tube creep failure under impact of hot gases flowing from the core and also possibility of core melt release from the failing reactor vessel at high primary pressure. These phenomena and their consequences with respect to severe accident progression were discussed. Recovery of heat removal from primary circuit is possible if water supply to steam generators is successful. At Balakovo NPP the strategy of water supply into steam generators from fire engines has been implemented. This strategy has been included into the SAMG of Balakovo NPP, Unit 4 and was presented.

To evaluate efficiency of this strategy depending on water flow rate and time of water supply the computer analyses have been performed. The results show the influence of water supply from fire engines on severe accident progression during the in-vessel SA phase. The analyses have been conducted using the MELCOR 1.8.5 code and show a positive influence of water supply from fire engines on the severe accident progression during the in-vessel SA phase.

Ambient Pressure-dependent Radionuclide Release from Fuel Observed in VEGA Tests under Severe Accident Condition and Influence on Source Term Evaluation
A. Hidaka, Japan Atomic Energy Agency, Japan

The radionuclide release from fuel during severe accidents is a primary issue for the source term evaluation. Although numbers of experiments[)] have been conducted in this domain of research in the world, information is still insufficient for the precise evaluation. For example, radionuclide release could mostly occur at high temperature under elevated pressure but very few studies have investigated the pressure effect so far due to difficulties in experimental operation. This presentation described the dependency of radionuclide release on ambient pressure observed in VEGA tests, proposed release mechanisms and release model with pressure effect, limitations of VEGA tests and future issues, possible influences on source term evaluation and accident management measures. In the VEGA programme, totally 10 tests were performed under the highest pressure and/or temperature conditions from 1999 to 2004. Tests with PWR fuel at 1.0MPa showed experimentally first that Cs release rate was suppressed by about 30% compared with that at 0.1MPa. Observed pressure effect could be explained by two-stage diffusion in UO2 grains & pores, and predicted by a simplified $1/\sqrt{P}$ CORSOR-M model. In BWR and MOX fuel tests, however, this effect was not observed clearly due to domination of vaporization from Cs deposited at peripheral pellet as a result of higher linear heating rate during reactor operation, differences in test conditions such as fuel oxidation and eutectic reaction with cladding. Relationship between the pressure effect and the factors described above is desirable to be further examined by other future tests considering the scale effect and irradiation history of fuel.

The decrease in radionuclide release under elevated pressure may affect PWR source terms, accident management such as intentional primary system depressurization. Present analyses suggested that the intentional depressurization has many advantages such as delay in accident progression and mitigation of the source terms at time of early CV failure in spite of increase in radionuclide release into primary system. The effect of pressure on consequences needs to be evaluated systematically for various combinations of accident sequences and AM measures considering their occurrence probabilities.

General discussion and conclusions

- Application of the results to configurations with shallow pools would require additional investigations to confirm the fragmentation and coolability.
- On the issue of Steam Explosion following statements were made:
  - ✓ Concerning In-vessel steam explosion: it has been recalled that no risk for vessel integrity in case of steam explosion has been identified by the participants in the SERENA programme.
  - ✓ Concerning Ex-vessel Steam Explosion: the issue is still under investigation. Specific new measurements devices have been introduced in the mock-ups in France and Korea to have a better view on the premixing zone of a corium jet entering the water.
- Analysis of code results in the SERENA 1 showed a significant set of discrepancies within the code results. For SERENA 2 a final decision-making process related to a reasonable risk evaluation should be implemented using experimental available data while the uncertainties of the codes models are still high.
- SAMGs optimization should be guided in an upper level document recalling the basic principles of SAM and the degree of flexibility allowed.
- Before implementing optimization of SAMGs strategies it has to be checked that the codes used to draw conclusions are well adapted to the phenomena subject to the analysis. The currently available experimental data sets and the modelling of the phenomena within codes may lead to differences in the evaluation of critical issues and hence a better common approach should be aimed at addressing, as far as possible, the critical issues.

Papers

Session 1

Recent IAEA Activities in the Area of Severe Accident Management and Level-2 PSA

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1. INTRODUCTION

Consideration of beyond design basis accidents of nuclear power plants (NPPs) is an essential component of the defence in depth approach underpins nuclear safety [1-3]. Beyond design basis accidents that may involve significant core degradation are of particular interest for accident management - a set of actions taken during the evolution of a beyond design basis accident made to prevent the escalation of the event into a severe accident; to mitigate the consequences of a severe accident and to achieve a long term safe stable state. The IAEA Safety Standards Safety Guide[1] "Severe Accident Management Programmes for Nuclear Power Plants" [4] provides recommendations on meeting the requirements of Refs. [5,6,7] for the establishing of an accident management programme to prevent and mitigate the consequences of beyond design basis accidents including severe accidents. The guiding principles for design and operation of NPPs are deterministic requirements with the implications that if deterministic criteria are met, the plant would be safe enough, and the risk of unacceptable radiological releases would be sufficiently low. The PSA technology provides the possibility to assess the risk dealing with a particular NPP. The application of PSA techniques to severe accidents is of particular importance due to very low probability of occurrence of a severe accident, but significant consequences resulting from degradation of the nuclear fuel. In order to address the need for standardization of the technical content of PSA the IAEA is developing two new Safety Guides: "Development and Application of Level-1 Probabilistic Safety Assessment for Nuclear Power Plants" [8] and "Development and Application of Level-2 Probabilistic Safety Assessment for Nuclear Power Plants" [9]. The Safety Guide on Level-2 PSA among others applications addresses the use of PSA for identification and evaluation of the measures in place and the actions that can be carried out to mitigate the effects of a severe accident after core damage has occurred.

2. THE GENERAL PROCESS OF DEVELOPMENT OF IAEA SAFETY STANDARDS

The general process of development of the publications in the IAEA Safety Standards Series foresees several stages that ensure close involvement of Member States, thorough review, and achieving a consensus position. Two safety Guides on PSA have been approved by the Commission on Safety Standards (CSS) and are expected to be published by the end of 2009.

[1] The IAEA Safety Standards Safety Guides are publications that provide recommendations on different aspects of NPP design and operation. They are governed by the general principles and objectives stated in Safety Fundamentals (Ref. [1]) and safety requirements presented in Safety Requirements publications.

3. THE SAFETY GUIDE ON SEVERE ACCIDENT MANAGEMENT PROGRAM

The Safety Guide on Severe Accident Management Program published this year provides recommendations on meeting the requirements for accident management, including severe accidents that are established in IAEA Safety Requirements [5,6,7]. The Safety Guide focuses on the development and implementation of severe accident management programmes for NPPs. Although the recommendations of this Safety Guide have been developed primarily for use for light water reactors, they are anticipated to be valid for a wide range of nuclear reactors, both existing and new.

The recommendations of this Safety Guide have been developed primarily for accident management during at-power states; however, it is also applicable, in principle, to other modes of operation, including shutdown states. Safety Guide consists of two main parts that are briefly described below.

3.1 Concept of the Accident Management Programme

A structured top down approach that should be used to develop the accident management guidance and main principles that should be followed while developing accident management guidance are presented in the Safety Guide. The top down approach should begin with the definition of objectives and strategies, follow a systematic process throughout the development course, and finally result in procedures and guidelines that generally should cover both the preventive and the mitigatory domains.

The Safety Guide presents recommendations to the structure and features of the accident management guidance for different possible domains (*Preventive, Mitigatory or both Preventive and Mitigatory domains)* and discusses the effective organization of the accident management process, the roles and responsibilities for the different members of the emergency response organization at the plant or the utility involved in accident management and communication between members of the emergency response organization. General recommendations to the upgrade of the equipment that is necessary for the development of a meaningful severe accident management programme and recommendations to the update of the accident management guidance where existing equipment or instrumentation is upgraded are also given in the Safety Guide.

3.2 Development of an Accident Management Programme

The recommendations to the process of the development and implementation of an accident management programme are presented in the Safety Guide. A brief summary of the key aspects of the process is given below.

Identification of sufficiently comprehensive spectrum of credible beyond design basis accidents (BDBA) is the main goal of the process for the preventive regime. An effective tool achieve this goal is to use insights from Level-1 PSA.

Identification of the full spectrum of credible challenges to fission product boundaries due to severe accidents is the primary task for *mitigatory regime*. Safety Guide recommends to use insights from Level-2 PSA for determination of the full spectrum of challenge mechanisms and to check whether risks are reduced accordingly after the accident management guidance has been completed. In view of the inherent uncertainties in determining the credible events, the PSA should not be used a priori to exclude accident scenarios from the development of severe accident management guidance. The Safety Guide considers the following main steps to set up an accident management programme:

1. Identification of plant vulnerabilities to find mechanisms through which critical safety functions may be challenged;
2. Identification of plant capabilities under challenges to critical safety functions and fission product barriers;
3. Development of suitable accident management strategies and measures; and
4. Development of the procedures and guidelines to execute the strategies.

STEP 1 The *identification of plant vulnerabilities* should be based on a comprehensive set of insights on the behavior of the plant during a beyond design basis accident and severe accident, including identified phenomena that may occur and their expected timing and severity are discussed.

STEP 2 *Plant capabilities* available to fulfill the safety functions, including unconventional line-ups, temporary connections and adaptation of equipment necessary to use these capabilities should be also identified. At this process the capabilities of plant personnel to contribute to unconventional measures to mitigate plant vulnerabilities should be considered.

STEP 3 The *accident management stra*tegies should be developed for each individual challenge or plant vulnerability in both the preventive and mitigatory domains. The development of strategies in the preventive domain should be aimed to preserve safety functions important to prevent core damage, and in the mitigatory domain - to enable terminating the progress of core damage once it has started, maintaining the integrity of the containment as long as possible; minimizing releases of radioactive material; and achieving a long term stable state. The systematic evaluation and documentation of the possible strategies that can be applied and particular consideration of the strategies that have both positive and negative impacts is essential. The overall goal of this systematic evaluation is to provide the basis for a decision about which strategies constitute a proper response under a given plant damage condition.

STEP 4 *Development of the procedures and guidelines* is the next step of the process. The strategies and measures should be converted to the Emergency Operating Procedures (EOPs) for the preventive domain and to the Severe Accident Management Guidelines (SAMGs) for the mitigatory domain. Procedures and guidelines should contain the necessary information and instructions for the responsible personnel, including the use of equipment and associated limitations as well as cautions and benefits. The guidelines should also address the various positive and negative consequences of proposed actions and offer options. Interfaces between the EOPs and the SAMGs should be addressed, and proper transition from EOPs into SAMGs should be provided for, where appropriate. However, where EOPs and SAMGs are executed in parallel it is important that hierarchy between EOPs and SAMGs is established. The recovery of failed equipment and/or recovery from erroneous operator actions that led to a beyond design basis accident or severe accident should be a primary strategy in accident management, and this should be reflected in the accident management guidelines. Safety Guide recommends to develop precalculated graphs or to use simple formulas ('computational aids') to avoid the need to perform complex calculations during the accident. It is also recommended to definite "rules of usage" for the actual application of SAMGs. The adequate background material that provides the technical basis for strategies must also be presented.

Hardware provisions for accident management (e.g. specific safety systems dealing with accidents) are essential to fulfill the fundamental safety functions (control of reactivity, removal of heat from the fuel, confinement of radioactive material) for beyond design basis accidents and severe accidents. For the new plants there are usually design features present that practically eliminate some severe accident phenomena; however, for existing plants, it may not be possible to develop a meaningful severe accident management programme that would make use of the existing hardware configuration; therefore, modification of the plant should be considered accordingly. Changes in design should also be proposed where uncertainties in the analytical prediction of challenges to fission product barriers cannot be reduced to an acceptable level. Equipment upgrades aimed at enhancing preventive features of the plant should be considered with high priority. For the mitigatory domain, in upgrading equipment the focus should be placed on preservation of the containment functions.

The role of instrumentation and control in the accident management is defined by the ability of the instrumentation to estimate the magnitude of key plant parameters needed for both preventive and mitigatory accident management measures. The instrumentation qualified for global conditions may not function properly under local conditions; therefore it's failures in severe accident conditions should be identified and method to verify that the reading from the dedicated instrument is reasonable should be developed. In the development of the SAMGs, the potential failure of important nonqualified instrumentation during the evolution of the accident should be considered and, where possible, alternative strategies that do not use this instrumentation should be developed.

The functions and responsibilities in accident management, in both preventive and mitigatory domains, need to be defined within the documentation of the accident management programme. A typical layout of the on-site emergency response organization is shown in the Safety Guide. The Safety Guide gives detailed recommendations to the responsible persons for the decision making in different domains, key recommendations to the technical support centre personnel, decision makers and implementers. In addition, the Safety Guide recommends to clearly define any involvement of the regulatory body in the decision making process.

The verification and validation process of all procedures and guidelines is aimed:

- to confirm correctness of the written procedure or guideline;
- to ensure that technical and human factors have been properly incorporated; and
- to confirm that the actions specified in the procedures and guidelines can be followed by trained staff to manage emergency events.

The review of plant specific procedures and guidelines and proper quality assurance programme is an essential part of the process.

An important factor is the education and training. It is recommended that education and training should be given for each group involved in accident management, including the management of the operating organization and other decision making levels, and, where applicable, also regulatory personnel. The training should be commensurate with the tasks and responsibilities of the functions (e.g. in-depth training should be provided for those performing the key functions in the severe accident management programme; others should be trained so that they fully understand the basis of proposed utility decisions). The training programme should be put in place prior to the accident management programme being introduced. The results from exercises and drills should be fed back into the training programme and, if applicable, into the procedures and guidelines as well as into organizational aspects of accident management.

The next point emphasized in the Safety Guide is dealing with processing new information and supporting analysis. This is an essential part of the procedures and guidelines development process. The revisions of EOPs and SAMGs and organizational aspects of accident management should be made in any change in plant configuration or change in background information used in the development of the procedures and guidelines (e.g. update of the PSA that identifies new accident sequences that were not a part of the basis of the existing accident management guidance; new insights from the research on severe accident phenomena).

The key aspects of the analysis of a potential beyond design basis accident or severe accident sequence performed in support for SAMGs are considered in Safety Guide for three consequential steps. In the first step of the analysis a full set of sequences should be analysed that would, without credit for operator intervention in the beyond design basis accident or severe accident domain, lead to core damage (typically identified in the PSA). In the second step - the effectiveness of proposed strategies and their potential negative consequences should be investigated. In the third step of the analysis, once the procedures and guidelines have been developed, they should be verified and validated. It is generally recommended that supporting analysis should be of a best estimate type performed with the appropriate computer codes and a consideration should be given to uncertainties in the determination of the timing and severity of the phenomena.

Several examples and recommendations given for the practical use of severe accident management guidelines and categorization scheme for accident sequences are presented in the Safety Guide (in Appendixes).

4. THE SAFETY GUIDES ON PSA PERFORMANCE AND APPLICATION

The Safety Guides on PSA [8,9] provide recommendations for performing or managing a Level-1 and Level-2 PSA for a NPP and for using the PSA to support the safe design and operation of NPPs. The recommendations aim to provide technical consistency of PSA studies to reliably support PSA applications and risk-informed decisions.
An additional aim is to promote a standard framework that can facilitate a regulatory or external peer review of a Level-1 and Level-2 PSAs and their various applications. The Safety Guides addresses the necessary technical features of a Level-1 and Level-2 PSAs for NPPs, as well as its applications, based on internationally recognized good practices. This paper briefly describes the Safety Guide on Level-1 PSA and with more details the Safety Guide on Level-2 PSA (with emphasis on application for severe accident management).

4.1 Safety Guide on Level-1 PSA and Applications

The PSA scope addressed in the Safety Guide includes all plant operational modes (i. e. full power, low power, and shutdown), internal initiating events (i.e. initiating events caused by random component failures and human errors) internal hazards (e.g. internal fires and floods, turbine missiles) and external hazards, both natural (e. g. earthquake, high winds, external floods) and man-made (e.g. airplane crash, accidents at nearby industrial facilities). The Safety Guide is focused on the damage to the reactor core; it does not cover other sources of radioactive material on the site, e. g. the spent fuel pool. However, while considering PSA for low power and shutdown operational modes, the risk from the fuel removed from the reactor is also addressed. The consideration of hazards dealing with malevolent actions is out of the scope of the Safety Guide. In Level-1 PSA aimed at assessing the core damage frequency, the most common practice is to perform the analysis for different hazards and operational modes in separate modules having a Level-1 PSA for full power operating conditions for internal initiating events as a basis. The Safety Guide on Level-1 PSA and applications follows this consideration.

4.2 Safety Guide on Level-2 PSA and Applications

This Safety Guide includes all the steps in the Level 2 PSA process up to, and including, the determination of the detailed source terms that would be required as input to a Level 3 PSA. Different plant designs use different provisions to prevent or limit the release of radioactive material following a severe accident. Most designs include a containment structure as one of the passive measures for this purpose. The phenomena associated with severe accidents are also very much influenced by the design and composition of the reactor core. The recommendations of this Safety Guide are intended to be technology neutral to the extent possible. However, the number and content of the various steps of the analysis assume the existence of some type of containment structure. General aspects of performance, project management, documentation and peer review of a PSA and implementation of a management system are described in the Safety Guide on Level 1 PSA [8] and are therefore not addressed here. This Safety Guide addresses only the aspects of PSA that are specific to Level 2 PSA. The Safety Guide describes all aspects of the Level 2 PSA that need to be carried out if the starting point is a full scope Level 1 PSA as described in Ref. [8]. The objective of this Safety Guide is to provide recommendations for meeting the requirements of Refs. [5,7] in performing or managing a Level 2 PSA project for a NPP. The Safety Guide is structured in accordance with the major task as discussed below.

PSA project management and organization: Specific recommendations relating to the management and organization of a Level-2 PSA project are provided in the Safety Guide. In particular the following aspects are addressed: definition of the objectives of Level 2 PSA; scope of the Level 2 PSA; project management for PSA; and team selection.

Familiarization with the plant and identification of aspects important to severe accidents: The aim of this task should be to identify plant systems, structures, components and operating procedures that can influence the progression of severe accidents, the containment response and the transport of radioactive material inside the containment. Safety Guide provides detailed recommendations dealing with acquisition of information important to severe accident analysis.

Interface with Level-1 PSA: grouping of sequences: This task is aimed at establishing the interface between Level-1 and Level-2 PSAs to definite plant damage states. The Safety Guide addresses recommendations for plant damage states definition for all initiating events and hazards, and plant operational states. The recommendations on how the existing Level-1 should be expanded to address specific aspects of the Level 2 PSA (when it is an extension of a Level 1 PSA performed originally without the intention to perform a Level 2 or Level 3 PSA) are also provided.

Accident progression and containment analysis: The key recommendations regarding the analysis of containment performance during severe accidents, analysis of the progression of severe accidents, development and quantification of accident progression event trees or containment event trees, treatment of uncertainties, and interpretation of containment event tree quantification results are provided in Safety Guide.

Source terms for severe accidents: The important step in the Level 2 PSA is the calculation of the source terms associated with the end states of the containment event tree. Source terms determine the quantity of radioactive material released from the plant into the environment. Since the containment event trees have a large number of end states, for practical reasons this requires the end states to be grouped into release categories for which the source term analysis is then carried out. Safety Guide gives detailed recommendations for definition of the release categories, grouping of containment event tree end states into release categories, source term analysis, uncertainty evaluation, and interpretation of results of the source term analysis.

Documentation of the analysis: The specific issues related to the presentation and interpretation of results and to organization of Level-2 PSA documentation are also in focus of Safety Guide.

Use and applications of the PSA: The Safety Guide provides the key recommendations for a number of Level-2 PSA applications. The following applications are covered among others: design evaluation; severe accident management; emergency planning; off-site consequences analysis; prioritization of research.

Three annexes of the Safety Guide provide an example of a typical schedule for a Level-2 PSA, information on computer codes for severe accidents, and details of the severe accident phenomena.

4.3 Application of Level-2 PSA for Severe Accident Management

The Safety Guide [9] provides recommendations on the use of Level 2 PSA for the evaluation of the measures in place and the actions that can be carried out to mitigate the effects of a severe accident after core damage has occurred. The aim of mitigatory measures and actions should be to arrest the progression of the severe accident or mitigate its consequences by preventing the accident from leading to failure of the reactor pressure vessel or the containment, and controlling the transport and release of radioactive material with the aim of minimizing off-site consequences. In particular the Safety Guide recommends to use the results of Level 2 PSA to determine the effectiveness of the severe accident management measures that are described in the severe accident management guidelines or procedures, whether they have been specified using the Level 2 PSA or by any other method. In addition Safety Guide emphasize that an accident management measure that is aimed at mitigating a particular phenomenon might make another phenomenon more likely due to the fact that the phenomena that occur in the course of a severe accident are highly uncertain and often interrelated. Therefore it is recommended to identify using the Level 2 PSA all interdependencies between the various phenomena that can occur during a severe accident to take them into account in the development of the severe accident management guidelines. Several examples illustrate this statement: depressurization of the primary circuit may prevent high pressure melt ejection but might increase the probability of an in-vessel steam explosion; introducing water into the containment may provide a cooling medium for molten core material after it has come out of the reactor pressure vessel but might increase the probability of an ex-vessel steam explosion; and operation of the containment sprays may provide a means of removing heat and radioactive material from the containment atmosphere but might increase the flammability of the containment atmosphere by condensing steam. It is also recommended that the updates of the Level 2 PSA and updates of the severe accident management guidelines should be performed in an iterative manner to facilitate the progressive optimization of the severe accident management guidelines. These recommendations correspond to those, provided in Ref. [4].

5. RELATED IAEA SERVICES

The IAEA mandate authorizes the IAEA to develop Safety Standards and to provide support for the application of these standards. A number of Services are made available by the IAEA for the Member States; amongst them there are also those related to severe accident management and Level-2 PSA.
The IAEA RAMP service is an activity to support individual Member States with the **R**eview of **A**ccident **M**anagement **P**rogrammes at their plants. Review of AM program at particular plant is performed on request by a Member State. The review team usually includes four experts plus an IAEA staff-member. The review focuses on studying the relevant documents, interviews with plant staff and regulators. The output of the review is a detailed report with assessment and recommendations for the improvements/refinements to the existing **A**ccident **M**anagement **P**rogramme. IAEA has prepared a manual in support of RAMP service (Ref. [12]) that contains a detailed questionnaire for the self assessment of the existing accident management programme. The following topics are covered in the manual:

- Selection and definition of AMP
- Accident analysis for AMP
- Assessment of plant vulnerabilities
- Development of severe accident management strategies
- Evaluation of plant equipment and instrumentation
- Development of procedures and guidelines
- Verification and validation of procedures and guidelines
- Integration of AMP and plant Emergency Arrangements
- Staffing and qualification
- Training needs and performance
- AM Programme revisions.

Several successful RAMP missions have been already conducted during which extensive review activities have been performed, a feedback provided, and findings discussed with the plant specialists. A formal review report was produced by the IAEA and forwarded to the counterpart.

Numerous workshops, training seminar and expert missions were provide by IAEA to China, Romania, Russia, Ukraine, Pakistan, Slovakia, Lithuania, etc. before the RAMP mission. The first RAMP mission was held at Krisko NPP in Slovenia in 2001, and other missions to Chinese PWR in China and Ignalina NPP in Lithuania were also conducted in 2006 and 2007, respectively. In 2009 the RAMP was performed for KANUPP (Pakistan). So far the mission for PWR, PHWR and RBMK were conducted. The RAMP for Cernavoda NPP (Romania) etc. are expecting for future service.

- For Ignalina NPP, several design modifications (core exit temperature measurement and an additional shutdown system) were made during the establishment of SAMP. It is the first SAMGs for RBMK reactors. It will therefore constitute a source of valuable information for other RBMK reactors.
- For Krisko NPP, assessing the possible impact of non-uniform hydrogen distribution and of the adequacy of the hydrogen source term and reconsidering the availability of the systems due to their potential failure during scenarios dominating core damage frequency were recommended during the mission.

International **P**robabilistic **S**afety **A**ssessment **R**eview **T**eam (IPSART) service was established in 1988. The dedicated guideline [13] is used for conduct of the review missions. Review of PSAs for plants from different countries, of various designs, and all PSA levels, hazard scopes, and operational modes is performed on specific request submitted to the IAEA by the Member State. Depending on the scope of the PSA the review duration is 1 to 2 weeks and review team composition is from four to seven international independent experts plus an IAEA staff-member. The review focuses on the check of methodological aspects, completeness, consistency, coherence, etc. of the PSA. The output of the review is the IPSART Mission Report that describes the review performed, the review findings, the technical aspects of the PSA study, strengths, and limitations and provides suggestions and recommendations for improvement of the PSA quality and its sound use for enhancing plant safety and risk management applications.
The IPSART service helps to achieve high quality of PSA and therefore assists in further enhancing the nuclear safety. More than 60 IPSART mission have been conducted so far in many countries all around the world helping to achieve high quality of PSA and proliferating advanced methodology and knowledge in nuclear safety assessment.

6. CONCLUSIONS

The IAEA has developed a comprehensive set of new Safety Standards including Safety Guides for Level-1 and Level-2 PSAs and severe accident management. The Safety Guides provide a common standardized platform for safety assessment and severe accident management that represent widely accepted good practices and consensus amongst Member States. These publications will promote a consistent development of the severe accident management programme, and development, application and review of PSA studies, as well as the use of PSA results and insights in different applications, including application for severe accident programme development.

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Technical Challenges in Applying SAMG Methodology to Operating CANDU[1] Plants

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1. Introduction

The Canadian Industry, through the CANDU Owners' Group, undertook a Joint Project to develop and implement SAMG to operating plants in Canada. This project began in 2001 with a review of available SAMG approaches, which recommended that an approach based on a modified Westinghouse Owners' Group (WOG) be used. A set of generic SAMG Guidance was produced in 2006 and subsequently plant-specific versions were developed for all seven plants operating in Canada. Canadian Utilities are currently in various stages of customizing the guidance for implementation. This paper reviews some of the major challenges encountered in SAMG development and implementation, and discusses how these challenges were addressed.

Overall, the application of the modified WOG approach to SAMG to CANDU plants is regarded as successful and as having the potential to contribute significantly to the reduction of residual risk. Some areas where SAMG implementation will have an impact on the conduct and results of Probabilistic Safety Analysis (PSA) are discussed.

2. Features of Multi-Unit CANDU Station Design Relevant to SAMG

CANDU reactor cores (Fig. 1) in the operating units contain either 380 (CANDU 6, Pickering B), 390 (Pickering A) or 480 (Bruce, Darlington) individual horizontal fuel channels comprising a pressure tube containing fuel and primary coolant, surrounded by a calandria tube forming the boundary with the low pressure moderator system. Heavy water coolant passes through the channels in the core,

[1] CANDU® is a registered trademark of Atomic Energy of Canada Ltd.

which are connected by feeder pipes at each end to headers located above the reactor (Fig. 2), and then to pumps and steam generators.

The channels span the length of a cylindrical vessel containing the heavy water moderator known as the “calandria”. This vessel is in turn surrounded by a large biological shield tank containing light water. In CANDU 6 and similar plants, this tank is a thick walled concrete vault, whereas in Bruce/Darlington designs the tank is a relatively thin-walled steel structure. In the multi-unit Bruce/Darlington designs, this tank sits above a large duct used for on-line fuelling access to all units. For the remainder of the paper, the focus will be on the Bruce/Darlington design, although plant-specific SAMG has been developed for all variants of the operating CANDU fleet[2].

Figure 1: CANDU Reactor Assembly

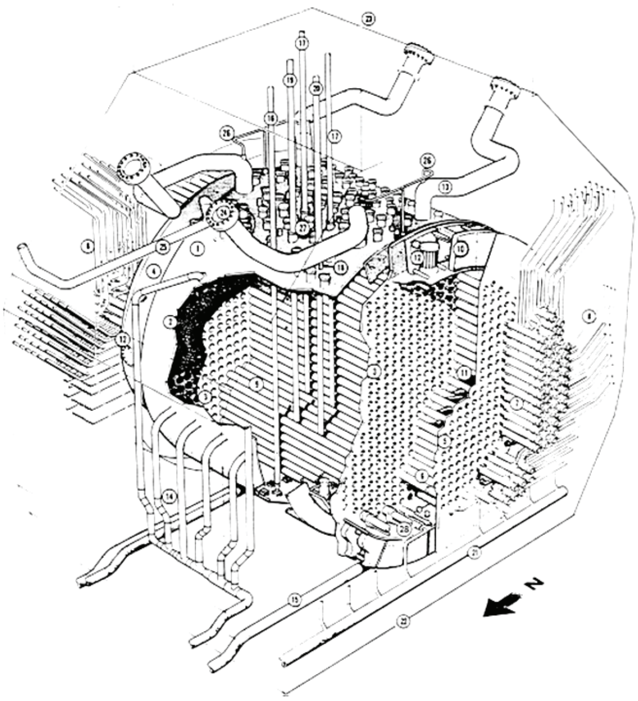
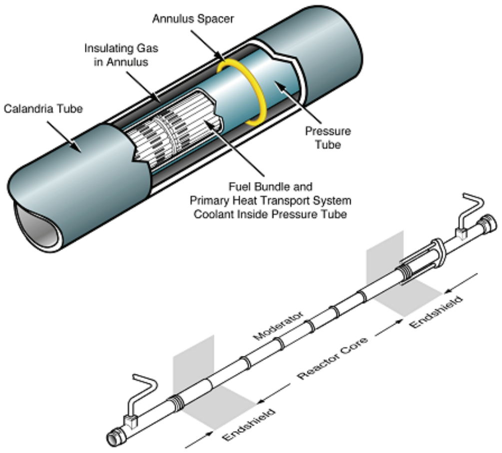


Figure 2: CANDU Fuel Channel Schematics



3. SAMG Entry Criteria

The most direct indicator that core cooling conditions are degraded is the temperature of the fuel. Most LWR cores use the exit thermocouple readings located above the core as a surrogate for this. In a horizontal pressure tube reactor there is no equivalent measurement. The only available temperature measurement is from thermocouples on the feeder pipes outside the reactor structure, whose response under degraded core conditions is unknown.

In addition, the design basis of CANDU plants includes certain accidents for which limited but significant fuel damage could occur, for example from a LOCA and failure of emergency core cooling. The stable core state after loss of core cooling is by rejection of decay heat to the moderator system (upper part of Fig. 3). The

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2 R. Fluke, K. Dinnie, M. Chai, R. Jaitly, M. O’Neill, “Developing Guidelines for Severe Accident Management”, Conference of the Canadian Nuclear Society, June 2005.

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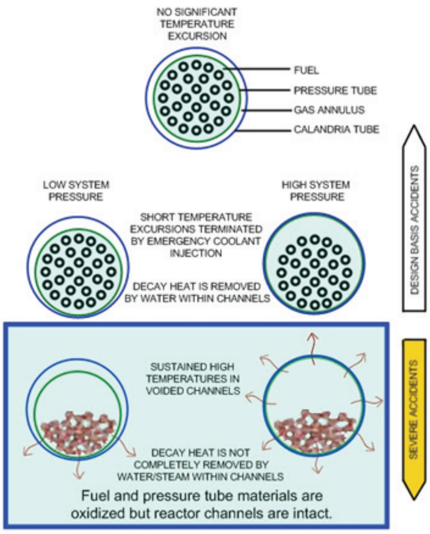
plant Emergency Operating Procedures (EOPs) contain guidance to respond to this type of event and there was clear direction from the Canadian industry that SAMG not encroach on this territory. Only if this heat removal pathway is lost would severe accident conditions beyond the design basis exist (lower part of Fig. 3).

Two types of indications were available on which to base the SAMG entry criteria:

1. System conditions (loss of core cooling plus loss of moderator cooling), and
2. Evidence of fuel damage beyond that expected in design basis events.

Fortunately, there is very good instrumentation available regarding the status of the moderator system due to its importance to core reactivity and power distribution. It has been shown that ensuring that fuel channels are covered by moderator water is sufficient to remove decay heat if primary cooling is lost. Therefore, loss of moderator level below that of the upper fuel channels in conjunction with loss of primary core cooling were identified as primary SAMG entry criteria.

Figure 3: CANDU Fuel Channel Under Degraded Cooling Conditions



Conditions in CANDU fuel channels with degraded fuel cooling but moderator cooling available remain highly uncertain. Depending on analysis assumptions regarding steam availability, it is possible to predict the oxidation of a significant fraction of fuel cladding with the accompanying hydrogen generation and fission product release to containment. Such conditions do not represent the expected behaviour but if they were to occur would represent containment conditions much more closely aligned with those addressed by SAMG. This unexpected behaviour might indicate the presence of accident progression or phenomena different from those assumed to exist in the design basis analysis. Accordingly, it was concluded that, should there be indications of fuel damage well beyond that expected in design basis accidents, then entry to SAMG would be appropriate. Such indications would be in the form of elevated dose rate measurements outside the containment boundary. This condition also provides a “rationality check” that would prevent SAMG from being entered prematurely or unnecessarily.

The above considerations led to the development of generic CANDU SAMG entry criteria as shown in Table 1.

Table 1: SAMG Entry Conditions

Condition	Parameter	Typical Instrumentation
1. Loss of core cooling	No subcooling margin in inlet headers for >15 minutes	Heat transport system (HTS) temperature and pressure instrumentation
AND ***either***		
2. Loss of moderator cooling to fuel channels	Moderator level below top of highest channels	Moderator level instrumentation
or		
3. Major release of fission products from the fuel	Plant radiation levels > setpoints	Ex-containment gamma measurements

The remaining challenge was to establish locations and associated setpoints for the radiation measurements used as a surrogate for degree of core damage. It has been a long-established practice at multi-unit stations to measure post-accident gamma dose at pre-determined locations outside containment for which pre-calculated correlations to core damage levels associated with specific accidents were already available. This information is intended to adjust release source terms to support offsite emergency response but could be readily adapted to infer a first-order indication of the likely degree of core damage. Curves of measured dose vs. time after reactor shutdown were provided in the form of a Computational Aid.

Choosing an appropriate setpoint was one of the most controversial issues in the development of CANDU SAMG. Best-estimates of likely degree of core damage associated with design basis accidents suggested that up to about 1% of the core inventory of volatile fission products could be released, although some worst case estimates were higher. Another consideration was the large uncertainty associated with the dose calculations and core damage correlations. Eventually, a dose rate measurement correlated to 3% release of volatile fission products was selected. This represents the logarithmic mean between an expected design basis source term of ~1% release and a 10% release that would clearly be in the severe accident domain.

4. Prioritization of Barriers to Severe Accident Progression

CANDU fuel channels are housed in a cylindrical calandria vessel containing heavy water moderator, which is itself surrounded by a large shield tank filled with light water (Fig. 4). These three components represent barriers to accident progression that can be "defended" by actions taken as part of SAMG. The question to be resolved was in what order should these barriers be prioritized in SAMG to make best use of the resources available and maximize the chances of terminating accident progression?

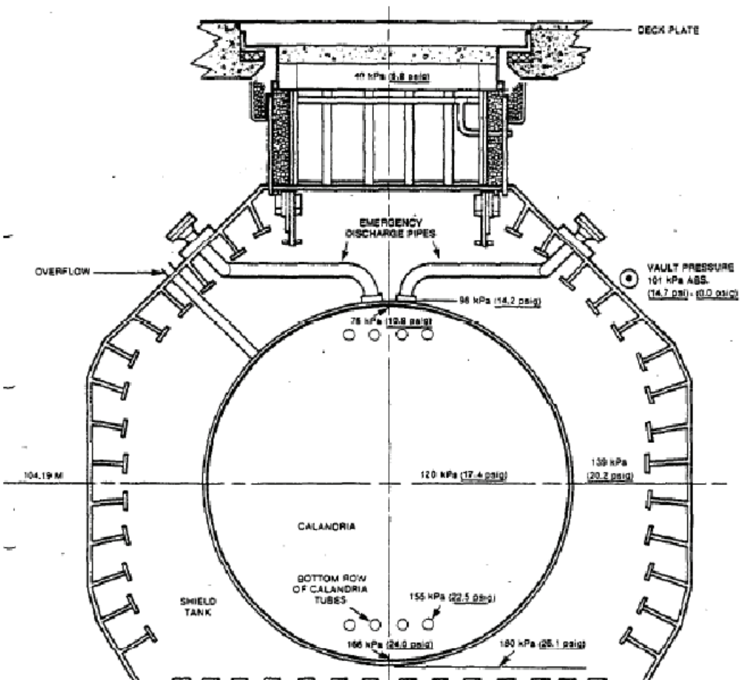
In conventional PWRs, the first two decisions in the Diagnostic Flow Chart (DFC) relate to protecting steam generator tubes from consequential rupture and depressurizing the Reactor Coolant System. These Severe Accident Guides (SAGs) are not required in CANDU plants because a) steam generators are well buffered from the conditions in the core by long pipe runs and headers and, b) the thin walled pressure tubes that fail early in a severe accident sequence, automatically causing the primary circuit to depressurize.

In the WOG SAMG, the next SAG is related to protecting RCS integrity and involves actions designed to maintain coolant inventory. For CANDU, the challenge to be addressed was whether priority should be given to protecting the intact barriers of the shield tank and calandria structures or whether actions should be taken to attempt to restore fuel cooling through the primary circuit, even though it is likely that the systems involved would be unavailable initially (otherwise there would be no severe accident).

In their current form, the seven CANDU SAGs are as follows:

1. Inject into the Heat Transport System
2. Control Moderator Conditions
3. Control Shield Tank Conditions
4. Reduce Fission Product Releases
5. Control Containment Conditions
6. Reduce Containment Hydrogen
7. Inject into Containment.

Figure 4: Barriers to Accident Progression



In its initial development, the order of the first three SAGs was reversed with the first being to maintain shield tank inventory. The idea behind this approach was that priority would be given to protecting the intact barriers. In the early stages of accident progression, the entry criteria for SAG-1 and SAG-2 would likely not be met and attention directed towards adding water to the fuel channels. If the fuel channels are already failed and fuel debris is in the moderator, the entry criteria for SAG-2 would be met and the calandria vessel would be the priority barrier to be protected and should this fail, the shield tank. The "nested" geometry of the three boundaries makes this approach attractive.

This prioritization has disadvantages, because the SAG priorities carry across to recovery actions and it is always better to initiate efforts to recover the emergency core cooling (ECC) system as soon as possible. In the original order this would not necessarily be the case.

In the alternative approach, with "inject into the HTS" as SAG-1, there is the risk that needed actions to protect the calandria may be delayed because of attempts to add water to the HTS, which in all likelihood will be unsuccessful during the early stages of accident progression (otherwise SAMG would not be entered at all). On the other hand, if water can be injected into the HTS, it will find its way to the intact barrier.

After debate it was decided to place "inject into the HTS" first. During the first validation exercise, admittedly with an inexperienced TSG, the concern discussed above regarding delay in completing SAG-1 was encountered and one of the outcomes of the exercise was to provide clearer direction about what to do if the initial pass through SAG-1 ("Inject into the HTS") indicated that no water sources were available. The general nature of this guidance is "initiate actions to recover ECC and exit SAG-1".

5. Diagnosing and Mitigating Challenges to Containment

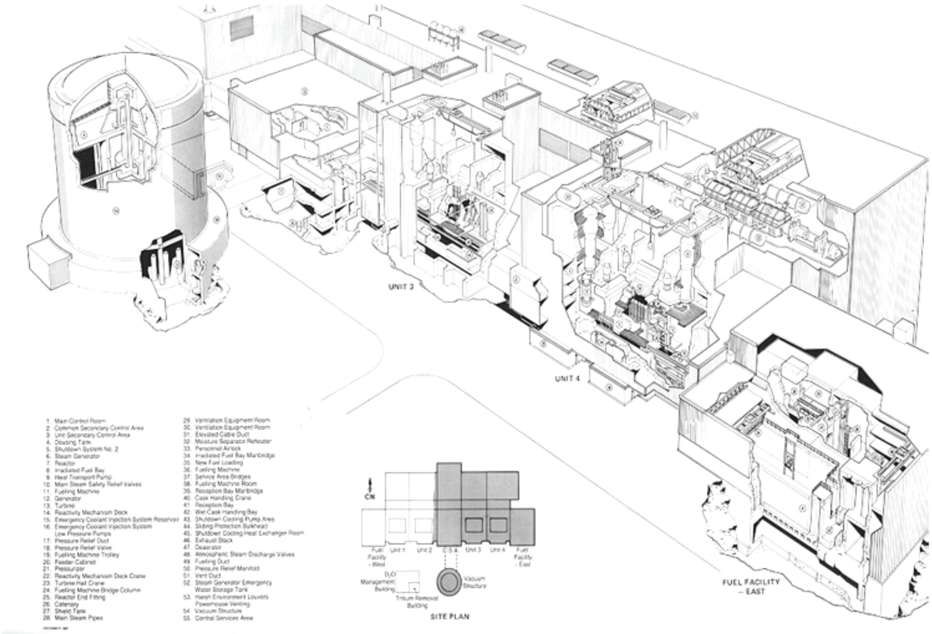
A third category of challenges is related to the response of the multi-unit, negative-pressure containment (Fig. 5). While the two single unit stations in Canada possess fairly conventional reactor containment designs, the large multi-unit stations in Ontario employ a low pressure, large-volume containment envelope shared by up to four units, protected initially by a Vacuum Building (VB) providing a large energy sink. The degree to which steam and other gases produced during accident progression are capable of challenging the structural integrity of the containment boundary and its sub-compartments remains an area of uncertainty, partly due to diagnosis and measurement limitations, to the extent that the single unit stations are proposing to add a filtered vent system to protect the containment structure from over-pressurization. Global containment analysis for the multi-unit plants suggests this is not required but local accident progression effects are still being evaluated including the role of the existing emergency filter system.

Two of the more challenging technical issues are discussed below: hydrogen management and filtered venting

5.1 Control of Hydrogen

CANDU reactor cores contain a large mass of zirconium and its alloys, of the order of 50 Mg, used to fabricate the fuel pellets, fuel bundles, pressure tubes and calandria tubes. Oxidation of a large fraction of this during severe accident progression, should it occur, has the potential to result in significant concentrations of hydrogen in the containment atmosphere. Actual concentrations would depend crucially not only on how much oxidation takes place but how hydrogen/deuterium is distributed throughout the four reactor units, the fuelling machine duct, pressure relief ducts and the vacuum building (Figs 5 and 6). Each unit also contains a system of glow-plug igniters. There is currently no engineered hardware measurement or sampling capability.

Figure 5: Darlington Multi Unit Containment



In the current SAMG, the hydrogen source term is evaluated by means of a Computational Aid (CA). This in turn relies on:

(a) MAAP4-CANDU[3] sensitivity calculations on source term related to rate of accident progression;

(b) correlations between hydrogen production and degree of fission product release to containment;

(c) an expectation that hydrogen will be well-mixed in the accident unit and that there will be mass transfer between the accident unit and other containment volumes (can be estimated by comparing relative radiation measurements at similar locations outside each reactor building);

(d) tracking of mass transfer to the VB (analogous to "venting" from the containment to the VB) for which pressure and temperature measurements are available;

(e) an assumption that igniters, when available, will maintain local hydrogen concentrations close to the flammability limit.

The resulting CA is complex and requires staff to be trained as a specialist in its use. Even the availability of measurement capability would not necessarily eliminate the need for a CA, due to the complexity of the containment geometry.

Maximum predicted quantities of hydrogen are well below typical PWR assumptions of 75% of active cladding. This, together with a well-mixed containment atmosphere, would result in containment conditions in which hydrogen burning is not expected to threaten containment integrity. Retention of

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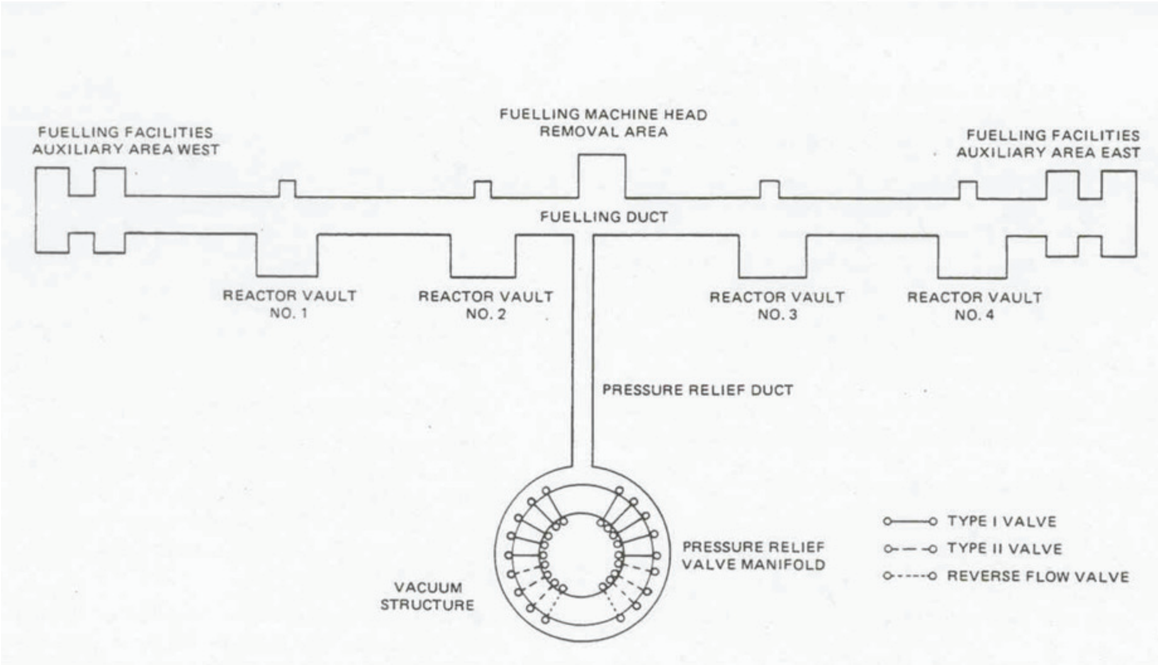
3 MAAP4-CANDU – Modular Accident Analysis Program for CANDU Power Plant Volume 1 &2 (User Guidance and Code Structure and Theory), Prepared by FAI, April 1998.

most of the hydrogen in a single reactor unit would represent a concern if igniters were unavailable or ineffective for a period of time. Efforts are continuing to improve the understanding of hydrogen generation and distribution into the long term phase.

5.2 Role of Containment Venting

The primary objective of the negative pressure containment design was to limit positive design pressure by providing a large energy sink (a large volume maintained at negative pressure to which steam released during an accident could be directed and condensed). Subsequently, the issue of containment pressure control in the long-term led to the design of a low pressure pumped filtered venting system whose purpose was to maintain containment pressure slightly subatmospheric by means of a controlled, monitored and filtered discharge to atmosphere from the vacuum building. The system is called the (emergency) filtered air discharge system or EFADS.

Figure 6: Outline of the Multi-Unit Containment Envelope



The EFADS was designed to be used following a design basis accident in which most of the energy release to containment would occur in the early stages of accident progression (typically within 1000s). The vacuum is slowly depleted over time due to various sources of inleakage until the EFADS is required to be initiated as the containment approaches atmospheric pressure per the existing Emergency Operating Procedures. The filters themselves are located outside containment in a conventional metal housing. The design pressure of the housing is 7 kPa(g) and the pressure across the filter limited to about 2.5 kPa (d).

The challenge facing SAMG is that the energy release to containment may occur over a protracted period of time, potentially many hours or even days, and conditions that would require operation of EFADS under EOPs may be reached before accident progression is complete. Even though the EFADS is buffered from the events in the main containment by the VB and associated pressure relief valves, occurrence of any significant pressure surge while the EFADS is operating has the potential to damage the filter and cause significant fission product release (does not apply to the Pickering containment due to design differences).

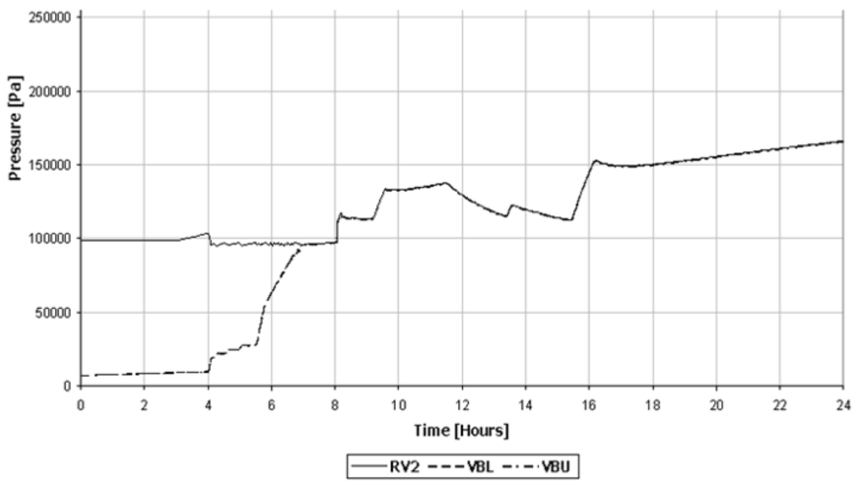
In SAMG, the Technical Support Group has to assess the state of accident progression and the likelihood of a future steam surge or hydrogen burn. The possibility of filter damage must be weighed against the impact of leakage from containment and expectations of external emergency response organizations to ensure any releases are controlled and monitored.

6. Impact of SAMG on PSA

The fact that SAMG has yet to be formally implemented at any plant in Canada at the present time to some extent makes the issue a hypothetical one, because this is a prerequisite to credit SAMG actions in any of the plant PSAs. However, it is worthy of note that, because current EOPs already address the prevention and mitigation of events leading to limited fuel damage, current PSAs contain some of the early preventative and mitigative actions typically found in SAMG.

Because of this, the focus for crediting SAMG in PSA will be primarily in support of Level 2 and the establishment of a long term stable plant state. In the current versions of Level 2 analysis, the only human actions credited are the operation of EFADS per the EOPs and an implicit assumption that recovery actions can be taken to reduce the likelihood of long term failure to no more than a 5% chance.

Figure 7: Containment Pressure vs. Time for Total Loss of Heatsinks



The process of developing SAMG has indicated that there may be a greater role for SAMG-based recovery actions in PRA. Analysis using recent versions of MAAP4-CANDU has shown that there is potential for unusual failure modes. For example, the pressure plot in Fig. 7 shows the pressure inside the major containment volumes (RV2 – reactor vault, VBU – vacuum building upper chamber, VBL – vacuum building lower chamber). Following a total loss of heatsink event, the current EOP would require that EFADS is initiated as the VBL pressure approaches atmospheric at about 7 hours. Shortly after there is a pressure spike associated with accident progression that would be sufficient to risk damaging the filter system, which is designed only for operation close to atmospheric pressure. Action to isolate EFADS from containment on entry to SAMG would reduce the likelihood of this failure mode in the Level 2 model.

The ability to credit human interactions in a containment event tree requires well-defined diagnosis information, clear decision criteria and enabling procedures. These requirements may not be met for many of the strategies considered in SAMG, especially those involving transient conditions where there may be a number of possible alternative options for mitigation. Therefore, inclusion in the event tree of SAMG actions requiring TSG judgement remains problematic.

SAMG is expected to be helpful in supporting PRA assumptions related to the initial conditions for Level 2, attempting recovery actions and establishing confidence regarding the achievement of a long term stable state. On the other hand, the Level 2 is expected to be able to provide insight to customize SAMG to anticipate station-specific phenomena and actions to mitigate them.

7. Conclusions

A number of design, procedural and operational aspects of CANDU plants present challenges in implementing SAMG as it has been applied in most LWRs. The CANDU approach has been to retain the principles of the international approaches while introducing modifications to address the unique aspects of CANDU plant design. The complexity of the multi-unit containment structures and the associated diagnostic processes needed to support SAMG remain a continuing area of investigation.

The initial role of SAMG in Level 2 PSA is expected to be in providing a higher level of confidence that existing actions in the containment event tree will be completed successfully. Actions to mitigate specific phenomena will be included if they can be defined with sufficient clarity to meet the requirements of PSA.

Accident Management in German NPPs: Status of Implementation and the Associated Role of PSA Level 2

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1. History of Accident Management in Germany

Triggered by WASH 1400, the first PSA in Germany (German Risk Study Phase A[1]) was started by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) in 1976 and published 1979. This survey had a more generic character, trying to estimate the risk caused by German NPPs in general and applying PSA methods. A plant-specific survey - the so-called Phase B[2]- has been elaborated in 1981 through 1989. During this second phase more detailed studies on the progress of accidents were conducted. Concerning accident sequences, the progression in time, the resulting loads and the necessary actions by the safety systems were investigated. During these studies it was discovered that existing safety margins could be used to prevent core damage or at least mitigate the consequences. The measures conducted for this purpose are commonly known as accident management (AM) measures by now and added a fourth layer to the defence in depth concept.

After the Tschernobyl accident the German Reaktor-Sicherheitskommission (RSK, Reactor Safety Advisory Committee) stated that AM-measures should be implemented into the existing NPPs if it would be possible with reasonable effort. The RSK also gave first recommendations on what kind of measures should be included. In addition, the RSK was also asked by the responsible Federal Ministry of the Interior to conduct a comprehensive safety review of all NPPs operating and being constructed. The RSK reported to the newly created Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), which is responsible for nuclear safety since mid-1986, in November 1988. As part of this report[3] the recommendations to introduce AM-measures were compiled and supplemented.

2. Safety Review and associated PSA requirements

Based on the recommendation of the RSK to make every operating NPP subject to comprehensive safety review, regulatory guides have been filed for the different elements of a PSR by a task force of

[1] Deutsche Risikostudie Kernkraftwerke, ISBN 3-921059-67-4, BMFT 1979
[2] Deutsche Risikostudie Kernkraftwerke Phase B, ISBN 3-88585-809-6, BMFT 1990
[3] Abschlussbericht über die Ergebnisse der Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland durch die RSK, Bundesanzeiger Nr. 47a, 08.03.1989

the Federal States Committee for Atomic Nuclear Energy, headed by the BMU, in the middle of the 1990s: the basic principles of PSR, the guide on safety status analysis, the guide on analysis of physical protection and the guide on probabilistic safety analysis (PSA).

The first German regulatory PSA guide, which has been issued in 1997, covered the fundamental and minimal requirements concerning the performance of PSAs. The technical details regarding the performance of PSA were set out in two technical documents, PSA Methods[4] and PSA Data[5]. These were developed by a working group of PSA experts (Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke - FAK PSA) coordinated by the Bundesamt für Strahlenschutz (BfS). In these documents, PSA requirements were restricted to Level 1+ (loss of design basis accident safety functions plus active containment systems at full power states but without consideration of core degradation) covering internal events as well as internal flooding events. Fire PSA was required, low-power and shutdown (LPSD) states and external events were not yet included explicitly.

Since a major amendment of the Atomic Energy Act in 2002 it is mandatory to perform PSRs including plant-specific PSA at ten years interval. Based on the experiences of performed plant-specific PSAs and PSA studies performed in Germany, the guide on PSA[6] as well as the technical documents[7,8] were published in a revised version end of 2005. Essentially, it contained the following major extension of scope compared to the 1997 version:

- calculation of core damage states[9], taking into account preventive accident management measures,
- extension of Level 1 PSA to LPSD,
- extension of the event spectrum to external hazards,
- performance of Level 2 PSA for full power operation.

Also, as part of Level 1 PSA, the entire safety related design features and the approved procedures of the operating manual get evaluated in order to achieve a well balanced safety concept. This includes the AM-measures described in the emergency manual and the evaluation of the efficiency of AM in preventing accident progression. In the frame of Level 2 PSA the efficiency of AM-measures regarding the mitigation of the consequences of severe accidents is evaluated.

3. Status of Implementation of AM-measures

As mentioned in paragraph 1, the RSK recommended the implementation of several AM-measures in its report to the BMU in 1988[3]. In order to deal with loss of offsite power, it was asked for additional off-site power via underground cable. Additionally, it was stated that the battery capacity for instrumentation should be sufficient to deal with a station black-out for two to three hours. Another preventive measure was to implement primary and secondary bleed and feed in PWRs.

Also first recommendations on how to deal with severe accidents were given. In order to limit the risk of hydrogen combustion, the inertisation of the containment was demanded for BWRs. For PWRs research about the effectiveness of igniters and passive autocatalytic recombiners (PARs) was started. A sampling system in the containment was also considered necessary in order to gain more precise information about the condition of the reactor core in case of a beyond design basis accident. Since

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[4] Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke: Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, *BfS-KT-16/97*, 1997

[5] Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke: Daten zur Quantifizierung von Ereignisablaufdiagrammen und Fehlerbäumen, *BfS-KT-18/97*, 1997

[6] Sicherheitsüberprüfung für Kernkraftwerke gemäß §19a des Atomgesetzes -Leitfaden Probabilistische Sicherheitsanalyse-, Bundesanzeiger 207a, 2005, English Version available: http://www.bfs.de/de/bfs/recht/rsh/volltext/A1_Englisch/A1_08_05.pdf

[7] Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke: Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, *BfS-SCHR-37/05*, 2005

[8] Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke*:* Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, *BfS-SCHR-38/05*, 2005

[9] Prior it was common for German Level 1 PSAs to end with so-called “hazard-states“, which do not take AM-measures into account.

maintaining the integrity of the containment is of highest priority in case of a severe accident, also the implementation of filtered containment venting (FCV) was requested.
The requested measures have been implemented in German NPPs as part of backfitting actions since then. Table 1 and Table 2 show the current status of implementation. Most AM-measures have been implemented in the 1990s or in some cases already in the late 1980s. PARs in PWRs have mostly been implemented around the year 2000. The installation of sampling systems in the containment was started in 1999 and is not yet finished for all plants.

Table 1: Implementation of Accident Management Measures at PWRs (dated Feb. 2009)

Measure	**KWB A**	**KWB B**	**GKN 1**	**GKN 2**	**KKU**	**KKG**	**KWG**	**KKP 2**	**KBR**	**KKI 2**	**KKE**
Emergency management manual	●	●	●	●	●	●	●	●	●	●	●
Secondary side bleed	●	●	●	√	●	●	●	●	●	●	√
Secondary side feed	●	●	●	●	●	●	●	●	●	●	●
Primary side bleed	●	●	●	●	●	●	●	●	●	●	●
Primary side feed	●	●	●	√	●	●	●	√	●	●	√
Assured containment isolation	●	●	●	√	●	●	√	●	●	●	√
Filtered containment venting	●	●	●	●	●	●	●	●	●	●	●
Passive autocatalytic recombiners	g	●	●	●	●	●	●	●	●	●	●
Supply-air filtering for the control room	●	●	●	●	●	●	●	●	●	●	√
Emergency power supply from neighbouring plant	●	●	●	●	□	□	□	●	□		□
Increased capacity of the batteries	●	●	●	●	●	●	√	●	●	●	●
Restoration of off-site power supply	●	●	●	√	●	●	●	●	●	●	●
Additional off-site power supply (underground cable)	●	●	●	●	●	●	●	●	●	●	●
Sampling system in the containment			●	●	●	●	●	●	●	●	●

√ design ● realised through backfitting measures g license granted □ not applicable

Empty fields: not yet applied for (KWB A + B) or not planned (KKI 2).
In case of KKI 2 the emergency power supply from the neighbouring plant (KKI 1) is not considered necessary since there is a direct connection to a nearby water power plant (see also Table 2).

Table 2: Implementation of Accident Management Measures at BWRs (dated Feb. 2009)

Measure	KKB	KKI 1	KKP 1	KKK	KRB B	KRB C
Emergency management manual	●	●	●	●	●	●
Independent injection system	●	●	●	●	□	□
Additional injection and refilling of the RPV	●	●	●	●	●	●
Assured containment isolation	●	●	●	●	√	√
Diverse pressure limitation for the RPV	●	●	●	●	●	●
Filtered containment venting	●	●	●	●	●	●
Containment inertisation	●	●	●	●	●*	●*
Supply-air filtering for the control room	●	●	●	●	●	●
Emergency power supply from neighbouring plant	□		●	□	●	●
Increased capacity for batteries	●	√	●	●	√	√
Restoration of off-site power supply	●	●	√	●	●	●
Additional off-site power supply (underground cable)	●	●	●	●	●	●
Sampling system in the containment		●	●		○	○

√ design ● realised through backfitting measures ○ applied for □ not applicable

* wetwell inerted, drywell equipped with catalytic recombiners

Empty fields: not applied for (KKB + KKK) or not planned (KKI 1).
KRB B+C: The independent injection system is for older BWRs (SWR-69) only in order to cover the whole pressure range between 1.5 MPa and operating pressure. This is not necessary for BWRs of the series 72, since injection systems for the complete pressure range are already implemented.

Concerning the use of PARs there is an ongoing discussion about the usefulness of these devices. This started in May 2005, when the BMU informed the RSK that there were doubts about the use of PARs in German PWRs because the PARs might act as ignitor for the hydrogen-air-mix and asked the RSK to answer its questions about whether the installation of PARs is safety-oriented or not. By now the responsible committee on plant- and system-engineering finished its statement and the RSK is discussing this statement. The statement includes insight gained from Level 2 PSA of the plant used as reference plant to determine the original design of the PARs for German PWRs. In this context it should be mentioned that quantitative results of Level 2 PSAs have not been used during the assessment which mitigative AM-measures should be implemented in German NPPs.

4. Conduct of Level 2 PSA in Germany

The following paragraph deals with the requirements concerning Level 2 PSA as described in PSA Methods[7].
As mentioned before, since 2005 it is mandatory to perform PSA Level 2 for full power operation as part of periodic safety review. This can either be done as part of an integrated PSA project comprising

Level 1 PSA and Level 2 PSA or an existing Level 1 PSA can be used as starting point. The latter is more common since Level 1 PSAs already existed for all German NPPs before it became mandatory to perform Level 2 PSA. Performing Level 2 PSA based on an existing Level 1 PSA will be called 2-step approach throughout this paper. In order to use the 2-step approach there must be an interface between Level 1 and Level 2 PSA. That means that certain information about availability of systems or the status of the containment is needed. This information will be fed into the grouping of the core damage states into plant damage states (PDS). It is not necessary to model plant damage states with low frequencies explicitly in the accident progression tree (APET) analysis if the sum of the frequencies of the PDSs screened out is less than 1% of the sum of the frequencies of plant damage states with similar properties. PSA Methods[7] contains tables for PWRs and BWRs showing which characteristics of the core damage states are needed for the grouping into plant damage states (PDS) for both the integrated and the 2-step approach. For the 2-step approach, the PDSs are the only link between the Level 2 and the Level 1 model, so this approach leads to a larger and more complex set of PDSs than the integrated approach.

Starting with these plant damage states, the accident progression event tree analysis is to be performed. This analysis should cover the timeframe until the release of radioactive material is basically over. The necessary deterministic accident analyses can either be done using integral computer codes, like MELCOR or MAAP or by using more specific models. The recommend procedure is to use integral deterministic accident analyses and, when necessary, supplement those by using more detailed codes.

The APET has to address all relevant features important to behaviour of containment, its failure and the source terms. It is recommend to distinguish different time phases, deduced from key events in the accident progression like:

- processes in the primary circuit before loss of RPV integrity,
- processes in the containment before loss of RPV integrity,
- processes in the containment during RPV bottom destruction by molten core material,
- processes in the containment after loss of RPV integrity.

The APET should also include branches for processes outside the containment and for containment bypass.

The resulting end states of the accident progression event tree are to be grouped into release categories (RC) which result in similar releases of radionuclides to the environment. For these RCs the source term analysis has to be performed. Therefor the radioactive inventory at the middle of a fuel cycle can be used. In general, approximately 10 release categories should be sufficient to cover all relevant end states. This is in accordance with international standards as in the current draft of the new IEAE safety standard on Level 2 PSA, which states *typically tens of release categories*[10]. The presentation of the results of the PSA should also include sensitivity analyses, which also cover the effectiveness of AM-measures.

PSA Methods[7] also gives hints on how the results of Level 2 PSA could be presented in order to support emergency planning, though it does not give any requirements. It is up to now not the case that emergency management gets oriented to PSA results.

5. Recent PSA Results on SAM Efficiency

On behalf of the BMU and the BfS, Level 2 PSAs for three different kinds of NPPs have been performed. It is not the scope of these PSAs to conduct a risk assessment of the reference plant but rather to advance the methodology of PSA and give feedback to the regulators in order to improve the regulatory framework. The NPP types used for these studies were BWR Series 69, BWR Series 72 and the Konvoi type PWR. Besides that, first Level 2 PSAs performed as part of PSR are completed by the licenses. The following paragraphs shall give an overview about the insights gained on the

[10] Performance and Application of Level 2 Probabilisitc Safety Assessment for Nuclear Power Plants, Draft Safety Guide , DS393 Draft 10, dated 05 March 2009

effectiveness of accident management measures in general and severe accident management measures in special.

Results for Konvoi-type PWR[11]:

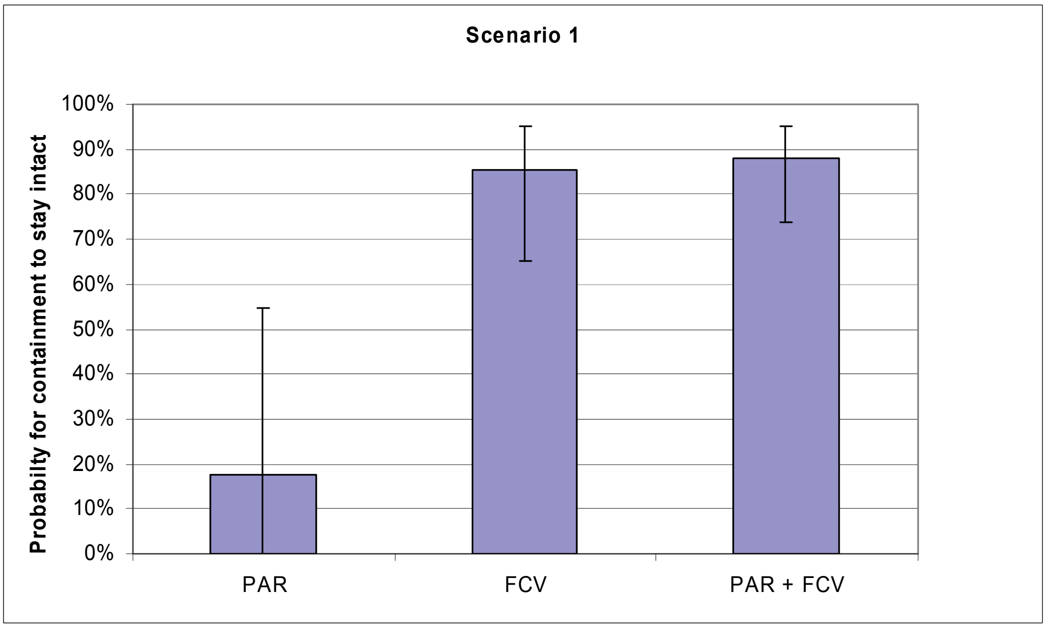
For the Konvoi type PWR the effect of PARs and FCV on the probability for the containment to stay intact was examined for three accident scenarios. Especially FCV shows a great influence on the containment performance.

Scenario 1 consists of a total loss of the steam generator feed using primary bleed and feed. During the in-vessel phase of the accident there is an inertisation of the containment due to high steam content. After failure of the RPV the pressure and the steam content inside the containment drop. Combustion of Hydrogen becomes possible. In the long term, the pressure inside the containment rises again until venting has to be initiated after two days. This explains why PARs alone are not very useful in this scenario. That the mean for the use of this measure is not equal to zero is caused by the possibility in the APET that emergency core cooling is recovered early enough to retain the core inside the RPV.

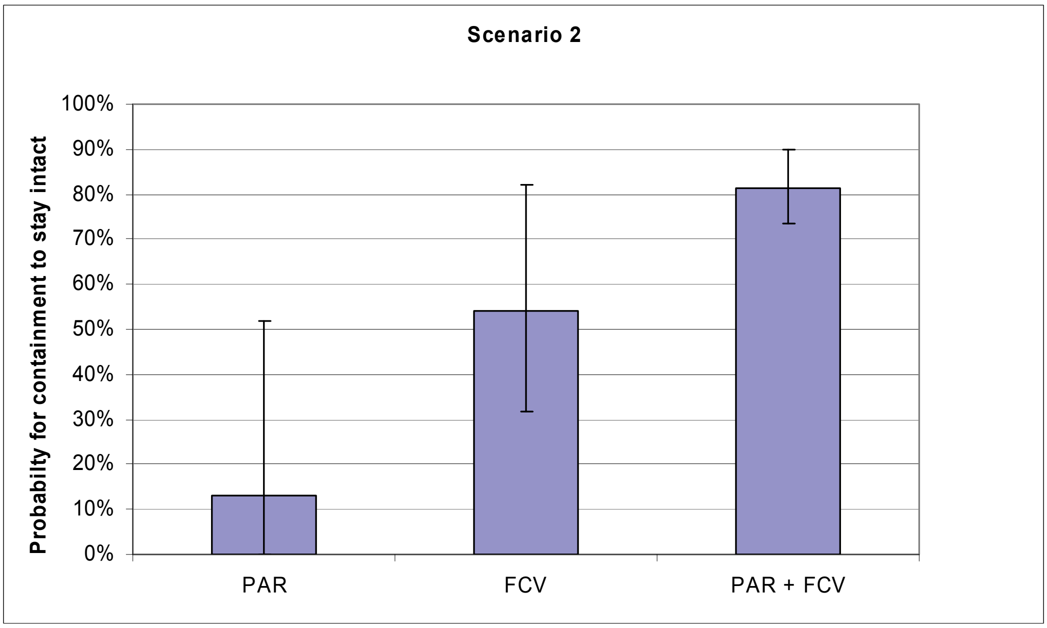
The second scenario, a break of the pressurizer connecting pipe, shows a large effect of the recombiners combined with FCV because the steam content inside the containment is fairly small in this case and so the probability for hydrogen combustion is higher than in the two other cases.

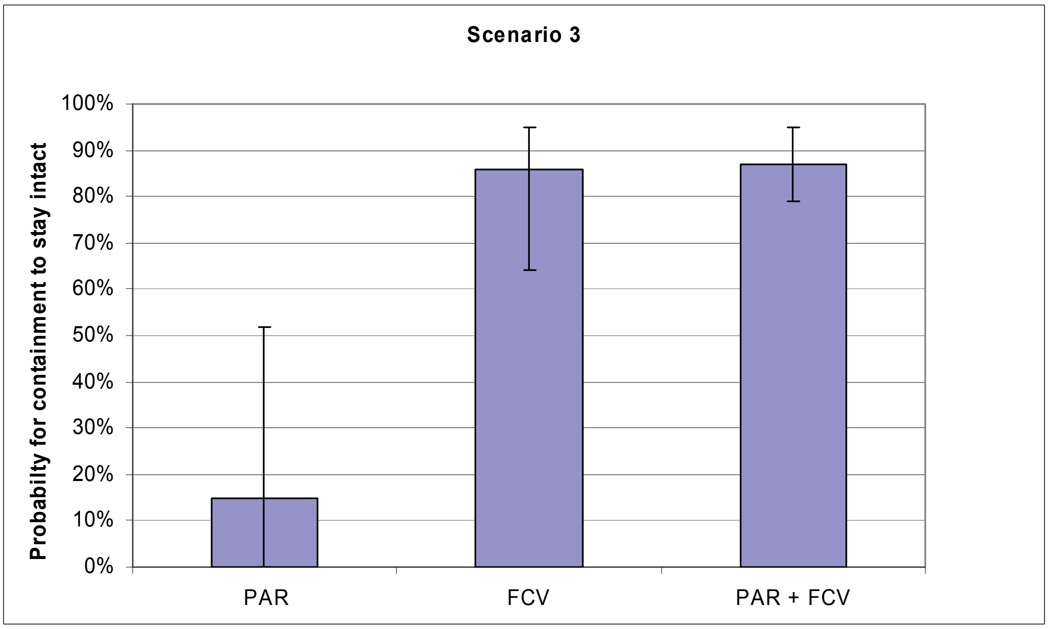
Scenario 3 describes a small leak in the hot leg. The conditions inside the containment are similar to scenario 1, just the progression in time differs. That explains why the probabilities for the containment to stay intact are comparable for scenarios 1 and 3.

Figure 1-3: Effect of PARs and FCV on selected accident scenerios[11]



[11] G. Bönigke et al.: Untersuchungen von Maßnahmen des anlageninternen Notfallschutzes zur Schadensbegrenzung für LWR, BMU-1999-536





Results for SWR 69*[12]*:

The SWR 69 is one of two types of BWRs used in Germany, although they exist in different versions with power ratings between 800 and 1400 MW. The reverence plant for the survey has a rated power of 890 MW net. The Level 2 PSA, which was conducted in form of a 2-step approach, shows that in case of a core damage there is a high probability for large and early release since the plant has low potential to prevent containment failure after failure of the RPV. Considering the RPV, there is also a low probability to restrain the core inside. This is only possible in the case that core degradation is not

[12] H. Löffler, M. Sonnenkalb: Methods and Results of a PSA Level 2 for a German BWR of the 900 MWe Class, presented at EUROSAFE 2006, Paris

far advanced and the core damage starts at high system pressure. The restraint is either possible using flushing water for control rods and pump seal water and keeping the pressure above 1 MPa or by releasing the pressure and using the low-pressure injection system. However, since the probability for high pressure core melt scenarios is low, the overall probability that one of these measures is successful is only about 0.015. Another measure intended to keep the integrity of the RPV would be to flood the control rod driving room (CRDR) with water using the drywell spray system. Analyses show that this is not effective to keep the RPV intact. Because of steam preventing water to reach crucial parts of the RPV, it will eventually fail on a large area instead of locally like it would if it is not cooled from the outside. A large-area failure of the RPV would damage the containment mechanically. After the failure of the RPV, which in most cases happens at low system pressure locally due to melt attack at the multiple penetrations in the bottom head, the melt and remaining water are released into the CRDR, where the steel shell of the containment will melt through after short time. This will happen at elevated but lower than initiating pressure for FCV. The shock pressure and combustion of released hydrogen will damage the reactor building with high probability, creating new release paths to the environment.
As part of the Level 2 PSA additional strategies to limit the consequences of a core melt accident were discussed. Emphasis is on preventing or at least delaying containment failure. This could be done by modifications inside the CRDR in order to ensure fragmentation of the core melt and increasing the flooding rate for the CRDR. Also, it should be possible to cool the outside of the CRDR with water. Another proposal to mitigate the effect of containment failure is to use FCV down to lower pressure than according to the emergency manual in order to reduce the shock pressure when the containment fails in order to reduce the damage to adjacent building. Venting would also reduce the risk due to hydrogen combustion since part of the hydrogen stored in the containment could be released to the environment in controlled conditions.

***Results for GKN 1*[13]**

GKN 1 is one of Germanys older PWRs. It entered commercial operation in 1976. It is a 3-loop PWR with a rated power of 840MWe. The plant has AM-measures implemented according to Table 1. The Level 2 PSA for GKN 1 has been conducted as part of PSR due December 2007, which is currently being reviewed by the responsible authority. As the Level 2 PSA for the SWR 69, it was conducted in a 2-step approach based on an existing Level 1 PSA. Assessment of the containment showed that gross failure of the containment under static or dynamic overpressure condition is the dominant failure mode since there is no preferred location for the development of a small leakage.
The results of the Level 2 PSA show large influence of FCV. Table 3 shows the release categories defined in the PSA. The dominant characteristics for the classification of the release categories were time and mode of containment failure. The table also shows the percentage of the plant damage states that lead to each release category and the relative contribution of each RC to the total activity risk, which is defined as the sum of the activity released for all release categories. This sum, however, does not include the noble gases. About ¾ of all initiating events leading to core damage eventual lead to a RC involving FCV in order to prevent gross failure of the containment. These end states, however, account for only 1.6% of the total release if the noble gases, whose release cannot be prevented by FCV, are neglected. The main contributors to the risk –about 95% of the risk– are scenarios for which the containment already fails before core damage (RC-D) or is bypassed by a LOCA outside the containment or steam generator tube rupture (RC-A,B,E) or FCV is initiated and the filtration capability fails (RC-I). Since these events account for only 12.5% of all initiating events leading to core damage, one can see that in most cases the consequences of a severe accident can be significantely reduced.

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[13] A. Strohm et al.: An Approach to quantification of Uncertainties in the Risk of Severe Accidents ant Neckarwestheim Unit 1 Nuclear Power Plant and the Risk Impact of Severe Accident Management Measures, presented at PSAM 9, 2008

According to the quoted paper, more detailed studies about the effectiveness of AM-measures have been done as part of the sensitivity studies by considering the following assumptions and/or conditions: recovery of alternating current, recovery of containment isolation failure, additional credit to reactor coolant system depressurization, steam generator decontamination factor, failure to actuate filtered containment venting system, impact of on-demand annulus filtration, containment leaking control, passive autocatalytic recombiners. Unfortunately there is no information available since the PSA is still being reviewed and no details have been published yet.

Table 3: Release Categories used for GKN 1 (Values according to [13])

Release Category	Containment failure mode	Description of release path	Relative proportions of the total PDS frequency	Relative contribution of RC to the total activity risk (excluding noble gases)
RC-A	LOCA outside containment	Large containment bypass → annulus → Unfiltered release	0.31%	21.50%
RC-B	Uncovered steam generator tube rupture (SGTR)	Release via uncovered steam generator tubes	0.05%	3.02%
RC-C	Early containment rupture	Containment failure at or before vessel breach → Annulus → unfiltered release	0.04%	1.58%
RC-D	Containment isolation failure	Containment failure before core damage → Annulus → unfiltered release	1.42%	12.58%
RC-E	Covered SGTR	Release via covered steam generator tubes	6.66%	51.43%
RC-F	Sump line failure	Containment failure after vessel breach → Annulus → unfiltered release	0.02%	0.03%
RC-G	Late containment rupture	Containment failure long after vessel breach → Annulus → unfiltered release	0.22%	0.29%
RC-H	Basemat melt-through	Release via penetration of concrete basemat	0.51%	0.62%
RC-I	Unfiltered containment venting	Containment venting with loss of filtration capability	4.02%	7.36%
RC-J	Filtered containment venting	Containment venting to stack with filtration	77.48%	1.60%
RC-K	No containment failure	Small containment leakage → Annulus → filtered or unfiltered release	9.27%	0.00%

6. Conclusion and Outlook

Based on recommendations by the RSK, various AM-measures have been implemented in German NPPs during the last 20 years. Some of the preventive actions for PWRs have been developed as part of the German Risk study Phase B[2]. Apart from that, all the measures are based on deterministic considerations. The analysis of AM-measures in PSA, especially in Level 2 PSA, was not demanded by the authorities before 2005. So, no detailed picture about the effectiveness of (severe) accident management measures is available for all German NPPs yet. First examinations have been conducted in the frame of research initiated by the BMU and by now first Level 2 PSAs for German NPPs conducted as part of PSR are completed and insights into the effectiveness of AM-measures are becoming available. One result e.g. is, as can be seen in paragraph 5, that FCV is an effective measure for PWRs and not very effective for BWRs.

For the Level 2 PSAs conducted as part of PSR, both the integrated approach and the 2-step approach have been used. Feedback from developers and reviewers will be fed into the next version of the German PSA guidance documents. To do this, the FAK PSA has just set up a working group on Level 2 PSA, which will start its work in November this year. With this feedback and Level 2 PSAs being conducted for more plants, more insights into the effectiveness and potential for improvement of AM-measures should become available.

Circumstances and Present Situation of Accident Management Implementation in Japan

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1. Introduction

The beginning of accident management (AM) implementation in Japan can be traced back to 1992. Through in-depth researches and discussions regarding the severe accidents and AM, the Nuclear Safety Commission (NSC) of Japan issued a decision entitled "Accident Management as a Measure against Severe Accidents at Power Generating Light Water Reactors"[1] in May 1992. In this decision, the NSC strongly recommended the regulatory body and utilities to introduce AM measures to nuclear power plants (NPPs), although sufficient safety level has been maintained by current safety systems at operating NPPs.

Responding to the decision issued by the NSC, the Ministry of International Trade and Industry (MITI), which was the regulatory body of NPPs at that time, encouraged the utilities to establish AM implementation plans using benefit of insights obtained from PSA in July 1992. With an investigation period of one year, the utilities submitted their plans of AM implementation to MITI in March 1994. MITI reviewed these utilities' plans from the technical point of view and made a report of "AM for Light Water NPPs" [2] in October 1994, in which MITI recommended the utilities to undertake the AM implementation plans with preparation of AM operating procedures and establishment of administrative framework toward 2000.

The utilities completed implementation of AM to their NPPs by February 2002 and reported to the Nuclear and Industrial Safety Agency (NISA), which is the new regulatory body of NPPs founded in

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1 Nuclear Safety Commission, "Accident Management as a Measure against Severe Accidents at Power Generating Light Water Reactors," May 28, 1992

2 Ministry of International Trade and Industry, "Accident Management for Light Water Nuclear Power Reactors," October, 1994

January 2001. In addition, the utilities submitted evaluation of effectiveness of AM measures for eight representative BWR and PWR plants to NISA. NISA reviewed those results with the assistant of the Japan Nuclear Energy Safety Organization (JNES) and confirmed validity of them. The results of evaluation performed by JNES were presented in the previous SAMM conference held in 2001.[3, 4]

Meanwhile, NISA recognized it was also important to evaluate effectiveness of AM measures for NPPs other than eight representative plants and requested utilities to perform evaluation on them. Following this request, the utilities performed evaluation of effectiveness of AM measures for each NPP and submitted the results to NISA as "PSA Evaluation Report following AM Implementation" in March 2004. NISA also reviewed these results with the help of JNES and confirmed appropriateness of these evaluation. This paper presents the results of this review.

Besides fifty-two operating NPPs, AM have been studied and implemented to four newly constructed NPPs up to now. This paper also presents current situation of AM implementation for these newly constructed NPPs.

2. Accident management measures and their effectiveness at the representative plants

The utilities selected AM measures focusing on essential safety functions of NPPs. Specifically, reactor shutdown, coolant injection to the reactor vessel and the containment vessel, heat removal from the containment vessel, and supporting function to the safety systems are chosen as the four essential functions for BWR and then relevant AM measures were selected for each safety function. Table 1 summarizes those AM measures adopted for BWRs. Similarly, reactor shutdown, core cooling, confinement of fission products, and supporting function to the safety systems are chosen as the essential safety functions for PWRs and AM measures were selected, which are shown in Table 2.

Although, similar AM strategies are used for BWR or PWR, respectively, regardless of varieties of plant design, specific AM plans depend on the design of plant as well as the preference of the utilities. For example, as an enhancement of electric power supply, which is categorized as one of AM measures of supporting functions to the safety systems, it is realized in various ways for BWRs as follows;

- Electric power from the adjacent unit is used by connecting safety buses of both units, in case of both offsite power and emergency diesel generators (EDG) being unavailable simultaneously.

[3] M. Kajimoto et. al., "Evaluation of Technological Appropriateness of the Implemented Accident Management Measures for BWR by Level 1 and Level 2 PSA Methods," Workshop on the Implementation of Severe Accident Management Measures, September 2001

[4] H. Takahashi et. al., "Evaluation of Technological Appropriateness of the Implemented Accident Management Measures for PWRs by Level 1 and Level 2 PSA Methods," Workshop on the Implementation of Severe Accident Management Measures, September 2001

- For the single-unit site, HPCS-DG is used as an alternate AC source to another safety bus.
- In case that one EDG is shared by two adjacent units, an additional EDG is installed so that each unit is equipped with two dedicated EDGs.

With regard to PWRs, following three alternatives are used for ECCS recirculation;

- Cross-tie between the low pressure injection line and the containment vessel spray injection line, which can make the low pressure recirculation using a containment spray pump in case of ECCS recirculation failure
- Alternative recirculation pump put in place in the recirculation sump
- A redundant valve to the recirculation sump isolation valve

Effectiveness of AM measures is assessed using level 1 and level 2 PSA. Reflecting highly standardization of plant designs in Japan and considering commonality of them, all BWR plants and all PWR plants are divided into eight groups, four for BWRs and another four for PWRs, and then PSA was performed for a representative NPP in each group. Categorization of BWRs and PWRs as well as their safety features are presented in Table 3 and 4, respectively. Studies on effectiveness of AM measures were conducted both by utilities and JNES. The result performed by JNES were presented previous ISAMM meeting held in 2001.

3. Effectiveness of accident management measures of individual plant

Upon the request from NISA, the utilities performed evaluation of effectiveness of AM measures for individual plant other than the eight representative plants, and submitted the results of these evaluation to NISA in March 2004 as “PSA Evaluation Report after AM Implementation.” NISA reviewed these reports with assistance from JNES. In the course of this effort, JNES made an investigation focusing on the large differences in the core damage frequencies (CDFs) between individual plant and the representative plant in the same group. In addition, PSA models of the representative plant were modified and sensitivity studies were done in order to clarify the causes of these large differences. The results of studies on the effectiveness of AM measures of individual plant are shown below.

3.1 BWR plants

Figure 1 shows the comparison of CDFs of individual BWR plant before and after AM measures implementation. Those values are normalized by CDF of type C (BWR5) representative plant before AM implementation. Figure 1 also shows reduction ratio of CDF by AM measures in each plant. This value is defined by the ratio of CDF after AM implementation to CDF before AM implementation in each plant. Similarly, Figure 2 shows the comparison of the containment functional failure frequencies (CFFs) of individual NPP and reduction ratios of CFF. These CDFs and CFFs are the results evaluated by the utilities.

When comparing CDFs among plant types, CDFs of type D plants before AM implementation are much less than CDFs of type A, type B and type C plants, while the reduction ratios by AM of type D plants are greater than the ratios of other plant types. For type D plants, the alternate rod insertion (ARI) and recirculation pump trip functions, which are designated as AM measures for the other types of plants, are adopted in the basic design of the plant for the purpose of additional reactor shutdown. In addition, highly redundant systems are used for the coolant injection and residual heat removal functions in the basic design of type D plants. These factors make CDFs before AM implementation much smaller than CDFs of other types. On the other hand, because additional reactor shutdown measures are already installed and additional AM measures are considered unnecessary for the highly redundant coolant injection and residual heat removal function, overall reduction ratios of CDF by AM measures of type D plants are greater than the other.

Some differences can be found among CDFs and CFFs of individual NPP before AM implementation and the reduction ratios by AM measures even in the same plant type. This is because there are some small differences in the design and operation of plants and AM measures adopted are sometimes unique to individual plant. One typical example of this difference is the design and operational of CCWS. While there are a lot of plants which belong to type C, they can be further divided into three subgroups. The plants in the first subgroup have a similar design of CCWS to the representative plant of the group. The design and operation of CCWS in the second subgroup, such as Kashiwazaki-Kariwa-2, and the third subgroup, such as Hamaoka-3, is not same as the first subgroup. This difference yield low unavailability of ECCS and, thus, smaller CDFs of the plants. On the other hand, an example of difference in AM measures can be found in Onagawa-1 in type B. In Onagawa-1, redundant CCWS pumps are installed as an AM measure, which makes a large reduction of CDF after AM implementation comparing the other plants in type B.

Because the differences in CFFs chiefly come from the differences in CDFs, thorough investigation on the differences of CFFs are not performed.

Reduction ratios range from 0.02 to 0.6 for CDFs and from 0.01 to 0.08 for CFFs. The effectiveness of AM measures can well be confirmed.

3.2 PWR plants

Figure 3 shows the comparison of CDFs of individual PWR plant before and after AM implementation. These values are normalized by CDF of type D (four-loop PWR with large dry containment vessel) representative plant before AM implementation as is in the BWR case. Figure 3 also shows the reduction ratios of CDF by implementing AM measures. Similarly, Figure 4 shows the comparison of CFFs of individual NPP and the reduction ratios. These results are evaluated by the utilities as well.

Some differences can be observed among CDFs of individual NPP and their reduction ratios. They are originated from the difference in the plant design or AM measures adopted, as discussed in the BWR case.

An example of the variation of the plant design which causes the difference in CDFs and CFFs is ECCS system design. CDF of Ikata-3 in type B group is much smaller than CDFs of other NPPs in the same group. In Ikata-3, the high pressure injection (HPI) pumps do not require boosting by the low pressure injection (LPI) pumps during ECCS recirculation mode while the other NPPs in the same group require boosting by LPI pumps. This plant design of Ikata-3 leads to smaller overall unreliability of ECCS during recirculation mode and thus smaller CDF of the plant. Same situation also can be found in the type D plants. Amongst type D plants, Turuga-2 is the only one plant which needs the boosting by LPI pump to HPI pump and, therefore, CDF of Turuga-2 is higher than CDFs of the other plants in type D group.

Another example can be seen in type A group. ECCS switch-over from the injection mode to the recirculation mode is done automatically for Tomari-1 and 2, while this operation is done by operator manually in other NPPs of type A group. This design difference makes CDFs of Tomari-1 and 2 smaller than CDFs of the other plants in the type A group.

In contrast, an example of the variation of AM measures which causes the differences in CDFs can be found in a measure of alternative ECCS recirculation. The CDF reduction ratio of Turuga-2 in type D group is smaller than the reduction ratios of other plants in the same group. The cross-tie between LPI line and CSI line is adopted as an AM measure for the alternative ECCS recirculation in type D plants generally, and this AM measure is applied to only one train for the plants other than Turuga-2. On the other hand, this AM measure is applied to both two trains of LPI and CSI at Turuga-2, and thus CDF reduction ratio of this plant is lower than the other.

The differences in CFFs chiefly come from the differences in CDFs and a thorough investigation is not performed for CFFs.

Although there are some differences in CDFs and CDF reduction ratios among plants according to the difference in the design of plants and AM measures adopted as mentioned above, reduction ratios of CDF and CFF lie in the range of 0.3 to 0.6 and 0.1 to 0.6, respectively, and the effectiveness of AM measures can well be confirmed.

In general, a large variation of CDFs can be found among the types of BWRs even before AM implementation comparing to CDFs of PWRs. This is because the basic design concept of ECCS is basically similar even in the different types of the plants for PWRs, whereas it depends on the types of the plants for BWRs. For PWRs, necessity of boosting by LPI pumps to HPI pumps during recirculation mode has a large effect. In addition, there is a tendency that the reduction ratios by AM measures are large for BWR plants.

4. Implementation of accident management measures for the newly constructed NPPs

Implementation of AM measures to the operating fifty-two NPPs had been completed by 2002 involving plant modifications. Meanwhile, for the newly constructed NPPs which begin commercial operation in 2002 or later, it is recommended by the NSC to establish an AM implementation plan and to submit the plan to the regulatory body for review soon after the detailed design of the plant was accomplished, and to complete AM implementation before the first fuel loading to the core.[5] According to this process, AM measures for Higashidori-1, Hamaoka-5, Shika-2, and Tomari-3 have been investigated, reported to NISA and reviewed by NISA and the NSC until now.

AM implementation plan and evaluation of effectiveness of AM measures for Tomari-3 were reported to NISA last year and they were reviewed by NISA and the NSC until the beginning of this year.[6, 7] Similar AM measures to the operating plants shown in Table 4 are used for this plant, but some of them, i.e. train separation of CCWS actuated by a low CCW surge tank level signal against loss of CCWS function and redundant intake lines from CV recirculation sump, are incorporated as a part of basic design. The reduction ratio of CDF and CFF taking a credit of AM measures including the measures considered as the basic design described above are 0.4 and 0.1, respectively. Although Tomari-3 belongs to type B group in Table 2, the design of the plant and the results of CDF, CFF, and the reduction ratio of CDF and CFF are not close to those of the representative plant of the group, rather close to those of Ikata-3.

In AM review, possibility of adverse effects on the essential safety functions of the plant and conformance to the basic requisites stipulated by NISA are examined in addition to the evaluation of the effectiveness of AM measures using PSA. These reviews are performed for the newly constructed plants in a similar way to the operating plants.

Specifically, the adverse effects on the essential safety functions of the plant are reviewed from the following points;

- Conformance to the safety guidelines of NPPs

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5 Nuclear Safety Committee, “Future Policy on Implementation of Accident Management for Light Water Nuclear Power Reactor Facillities,” Ocotrber 20, 1997

6 Nuclear and Industrial Safety Agency, “Report for Studies on Accident Management of Hokkaido Electric Power Company Tomari Nuclear Power Plant Unit No.3,” October 6, 2008

7 Nuclear Safety Committee, “Implementation of Accident Management for Hokkaido Electric Power Company Tomari Nuclear Power Plant Unit No.3,” January 19, 2009

Conformance to the safety design guidelines, the safety analysis guidelines, and the seismic design guidelines was reviewed to check if there is no adverse effect by implementing AM measures.

- Adverse effect on the safety systems
  To check if there is no adverse effect on redundancy, independence, and essential functions of the safety systems in case of modification of these systems being made in order to incorporate AM measures.
- Effect to the results of safety analyses
  To check if there is no effect to the results of safety analyses which are reviewed in the plant licensing in case of any failure in AM features being assumed in normal operation.

With regard to AM basic requisites, the following five points are reviewed by NISA;

- AM enforcement structure (organization, roles of staffs)
- Facilities and equipments (communication system, plant information transmission system, data acquisition system like radiation monitors, emergency dose prediction system, manuals (operating manuals and AM guidelines))
- Knowledgebase of AM
- Notification and communication
- Training of staffs

The results of AM review for Tomari-3 by NISA were reported to the NSC in October, 2008. Upon receiving this report, the NSC reviewed the results and corroborated adequacy of AM measures for Tomari-3. The NSC also raised the followings as the future issues of AM implementation;[8]

- Reconsideration of the treatment of AM in the nuclear safety regulatory framework
- Efficient scheme of AM development
- Improvement of quality to confirm the effectiveness of AM measures
- Points of concern to use PSA
- Consideration of external events
- Contribution to grow up the security of public to the nuclear safety

5. Concluding Remarks

Introduction of AM measures to the Japanese NPPs began with the decision by the NSC issued in 1992, followed by the study of AM measures for the operating plants. Modifications of the plants as well as the establishment of AM enforcement framework and the preparation of the relevant AM

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[8] Nuclear Safety Committee, "Future Issues for Implementation of Accident Management," January 19, 2009

procedures have been completed by 2002. The effectiveness of AM measures is evaluated by utilities and results of these evaluations are reported to the regulatory body. The effectiveness of AM measures was confirmed through the reviews on these reports performed by the regulatory body.

Meanwhile, for the newly constructed NPPs, it is recommended to establish AM measures and to complete installation of AM measures by the first fuel loading to the core of the plant. Up to now, AM plans for four newly constructed plants are studied and reviewed in this process. In some cases, AM measures are incorporated as a part of basic design of the plant, reflecting the outcomes achieved by the AM studies for the operating plants.

In the latest AM review, the NSC pointed out some future issues for AM implementation; i.e. reconsideration of the treatment of AM in the nuclear safety regulatory framework, improvement of the quality of PSA, AM for external events and others.

Table 1 Reactor types and safety systems (BWR)

<table>
<tr><td colspan="2"></td><td>type A</td><td>type B</td><td>type C</td><td>type D</td></tr>
<tr><td colspan="2">Type of reactor</td><td>BWR2, 3</td><td>BWR4</td><td>BWR5</td><td>ABWR</td></tr>
<tr><td colspan="2">Type of containment vessel</td><td>MARK-I</td><td>MARK-I</td><td>Mod. MARK-I
MARK-II
Mod. MARK-II</td><td>RCCV</td></tr>
<tr><td colspan="2">Name of plant
(Bold : representative plant)</td><td>Fukushima1-1
Turuga-1</td><td>Onagawa-1
Fukushima1-2
Fukushima1-3, 4, 5
Hamakoka-1, 2
Shimane-1</td><td>Onagawa-2, 3
Fukushima1-6
Fukushima2-1
Fukushima2-2, 3, 4
Tokai-2
Kashiwaza
-Kikariwa-1, 2, 3, 4, 5
Hamaoka-3, 4
Shika-1
Shimane-2</td><td>Kashiwazaki
-Kariwa-6
Kashiwaza
-Kikariwa-7</td></tr>
<tr><td colspan="2">Safety systems</td><td></td><td></td><td></td><td></td></tr>
<tr><td colspan="2">Reactor scram</td><td>CRDHS
SLCS</td><td>CRDHS
SLCS</td><td>CRDHS
SLCS</td><td>CRDHS
SLCS
ARI
FMCRD</td></tr>
<tr><td rowspan="2">ECCS</td><td>High press.</td><td>HPCI
IC(2 trains)</td><td>HPCI
RCIC</td><td>HPCS
RCIC</td><td>HPCF(2 trains)
RCIC</td></tr>
<tr><td>Low press.</td><td>CS(2 trains)</td><td>CS(2 trains)
LPCI(2 trains)</td><td>LPCS
LPCI(3 trains)</td><td>LPFL(3 trains)</td></tr>
<tr><td colspan="2">Containment heat removal</td><td>SHC(2 trains)
CCS(2 trains)</td><td>RHR(2 trains)</td><td>RHR(2 trains)</td><td>RHR(3 trains)</td></tr>
</table>

RCCV:	Reinforced concrete containment vessel
Fukushima1:	Fukushima Site No.1
Fukushima2:	Fukushima Site No.2
CRDHS:	Control rod drive hydraulic control system
SLCS:	Standby liquid control system
ARI:	Alternate rod insertion
FMCRD:	Fine motion control rod drive
HPCI:	High pressure core injection (system)
IC:	Isolation condenser
RCIC:	Reactor core isolation cooling (system)
HPCF:	High pressure core flooder
CS:	Core spray (system)
LPCI:	Low pressure coolant injection (system)
LPFL:	Low pressure flooder
SHC:	Shutdown reactor cooling (system)
CCS:	Containment cooling system

Table 2 Reactor types and safety systems (PWR)

		type A	type B	type C	type D
Type of reactor		Two-loop	Three-loop	Four-loop	Four-loop
Type of containment vessel		Large dry SSCV	Large dry SSCV	Ice condenser	Large dry PCCV
Name of plant (Bold: representative plant)		Tomari-1, 2 Mihama-1, 2 Ikata-1 **Ikata-2** Genkai-1, 2	Mihama-3 Takahama-1, 2 **Takahama-3, 4** Ikata-3 Sendai-1, 2	**Ohi-1, 2**	Turuga-2 **Ohi-3, 4** Genkai-3, 4
Safety systems					
Reactor protection system		2 trains, Relay type	2 trains, SSPS	2 trains, SSPS	4 trains, SSPS
ECCS	High press. injection (No. of pumps)	2(High press. injection pump), Boosted by LPI pump during recirculation mode	3(Charging SI pump), Boosted by LPI pump during recirculation mode	2(Charging SI pump), 2(High press. injection pump), Boosted by LPI pump during recirculation mode	2(High press. injection pump)
	Low press. injection (No. of pumps)	2	2	2	2
	No. of accumulators	2	3	4	4
Auxiliary feedwater No. of M/D pumps No. of T/D pumps		 2 1	 2 1	 2 2	 2 1
Containment vessel spray (No. of pumps)		2	2	2 with 2 RHR spray pumps	2

SSCV: Steel containment vessel
PCCV: Pre-stressed concrete containment vessel
SSPS: Solid state protection system
ECCS: Emergency core cooling system
M/D: Motor-driven
T/D: Turbine-driven
RHR: Residual heat removal (system)

Table 3 Accident management measures (BWR)

<table>
<tr><th>Safety function</th><th>Purpose</th><th>Accident management measures to prevent core damage</th><th>Accident management measures to mitigate core damage</th></tr>
<tr><td>Reactor shutdown</td><td>Alternate reactivity control</td><td>● ARI(Control rod insertion by high reactor pressure or low reactor level)
● RPT(same signal)
*These signals are independent to current scram signals or ECCS actuation signals
ABWR adopts alternate reactivity control as the basic design.</td><td>-</td></tr>
<tr><td rowspan="2">Coolant injection to reactor and containment vessel</td><td>Automatic reactor depressurization</td><td>● ADS automatic actuation by low reactor level(L-1) with delay (except BWR2,3 and ABWR)</td><td>-</td></tr>
<tr><td>Alternate coolant injection</td><td colspan="2">● MUWC
● Fire extinguish system (except Onagawa), Filtrate water system (Onagawa)</td></tr>
<tr><td rowspan="3">Heat removal from containment vessel</td><td>Hard vent system</td><td colspan="2">● Hard vent system</td></tr>
<tr><td>Alternate cooling</td><td>-</td><td>● Alternate cooling by dry-well cooler or CUW</td></tr>
<tr><td>Recovery of RHR</td><td colspan="2">● Recovery of RHR</td></tr>
<tr><td rowspan="2">Supporting function</td><td>Electric power supply</td><td colspan="2">● Electric power supply from adjacent unit on 6.9 kV and 480 V (Fukushima Site No.1, Fukushima Site No.2, Kashiwazaki-Kariwa, Tokai-2, Tsuruga-1) or 460 V (Other Plants)
● Electric power supply from HPCS-DG (Single-unit site: Shika-1 and Tokai-2)
● Installation of dedicated emergency diesel generators (Fukushima Site No.1)</td></tr>
<tr><td>Recovery of emergency diesel generator</td><td colspan="2">● Recovery of emergency diesel generator</td></tr>
</table>

ARI: Alternate rod insertion

RPT: Recirculation pump trip

MUWC: Makeup water system condensated

CUW: Reactor water cleanup (system)

Table 4 Accident management measures (PWR)

Safety function	Purpose	Accident management measures to prevent core damage	Accident management measures to mitigate core damage
Reactor shutdown	Reactor shutdown	● Diversity of emergency secondary cooling (use of main feedwater in case of ATWS)	-
Core cooling	ECCS injection	● Use of LPI with depressurization by turbine bypass valves	-
	ECCS recirculation	● Alternative recirculation ➢ Tie-line between LPI and CSI ➢ Alternate recirculation pump ➢ Recirculation sump isolation valve bypass line	-
	Isolation of coolant leakage	● Cooldown and recirculation	-
Confinement of radioactive materials	Heat removal from containment vessel	● Natural convection heat removal ➢ Use of non-safety CV heat removal system ➢ Outside CV spray	● Natural convection heat removal ● Coolant injection to CV ● Forced depressurization of primary system ● Hydrogen igniter (Ice condenser CV plant)
Supporting function	Supporting function	● Alternate component cooling ➢ Air conditioning system ➢ BOP CCWS ➢ CV cooling system ➢ Fire extinguish system	-
		● Electric power supply from adjacent unit ➢ Connection between high voltage buses ➢ Connection between low voltage buses	-

LPI: Low pressure injection
CSI: Containment spray injection
BOP: Balance of plant

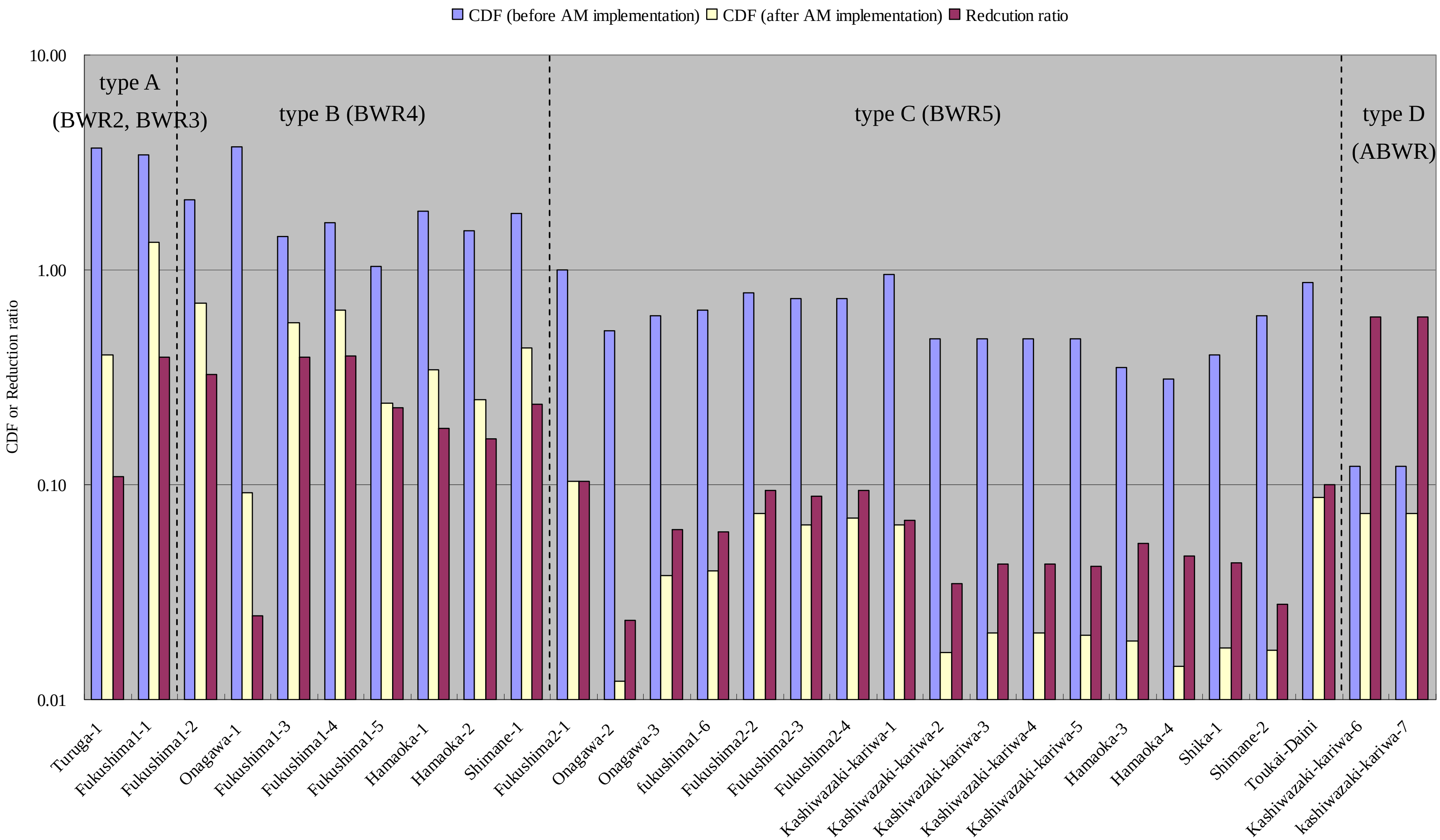

Figure 1 Comparison of core damage frequencies before and after AM implementation (BWR)

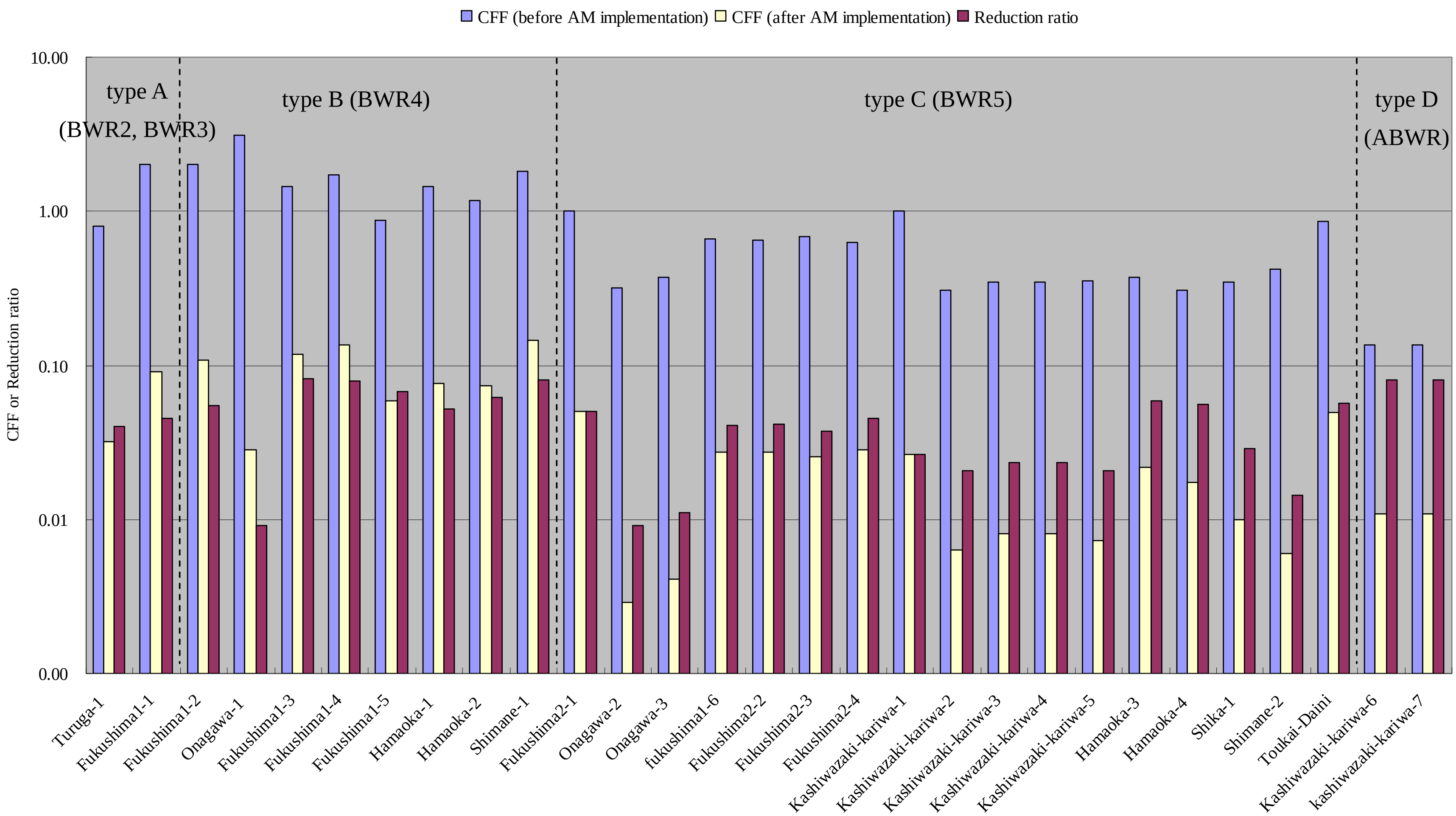

Figure 2 Comparison of containment functional failure frequencies before and after AM implementation (BWR)

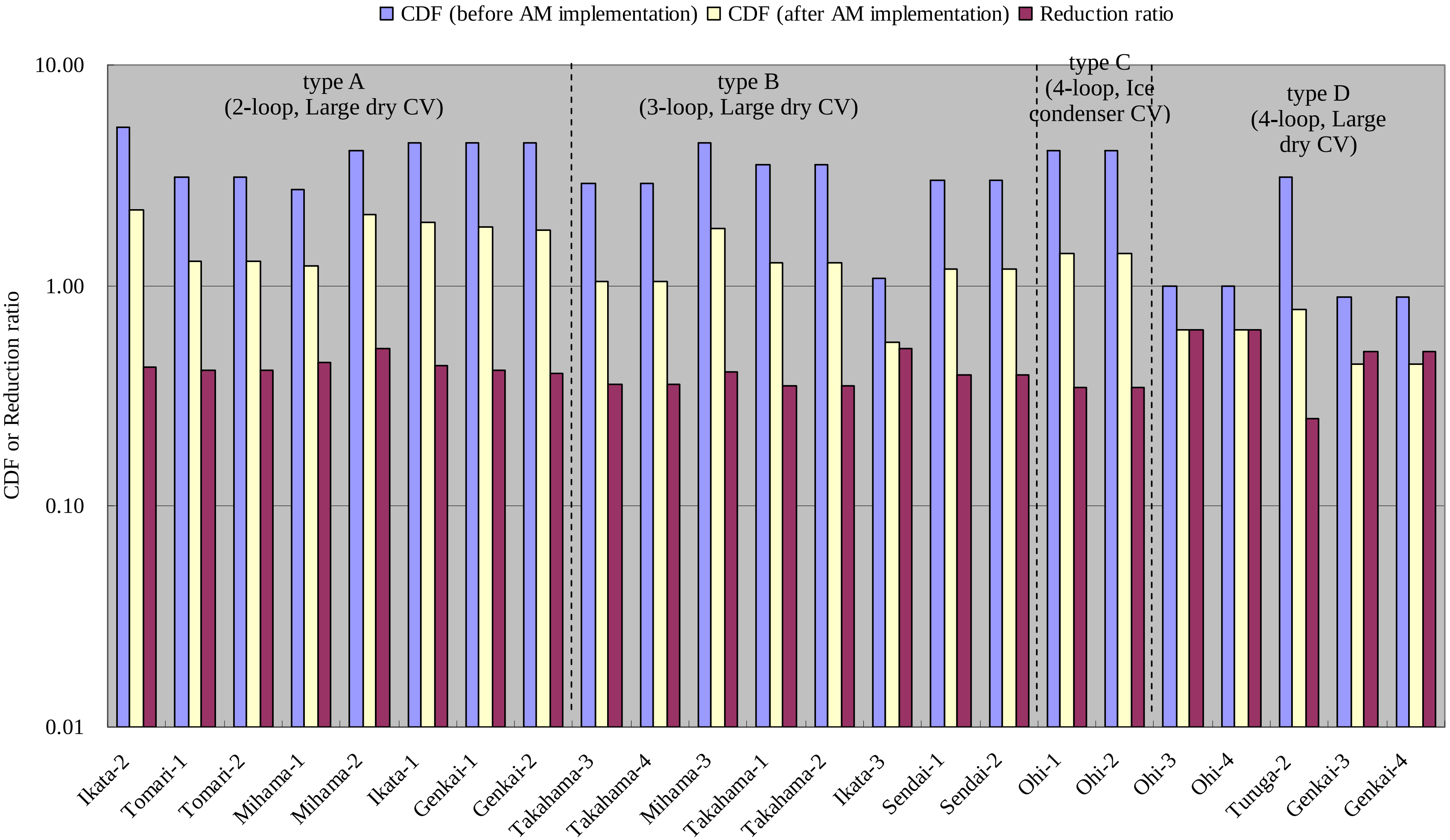

Figure 3 Comparison of core damage frequencies before and after AM implementation (PWR)

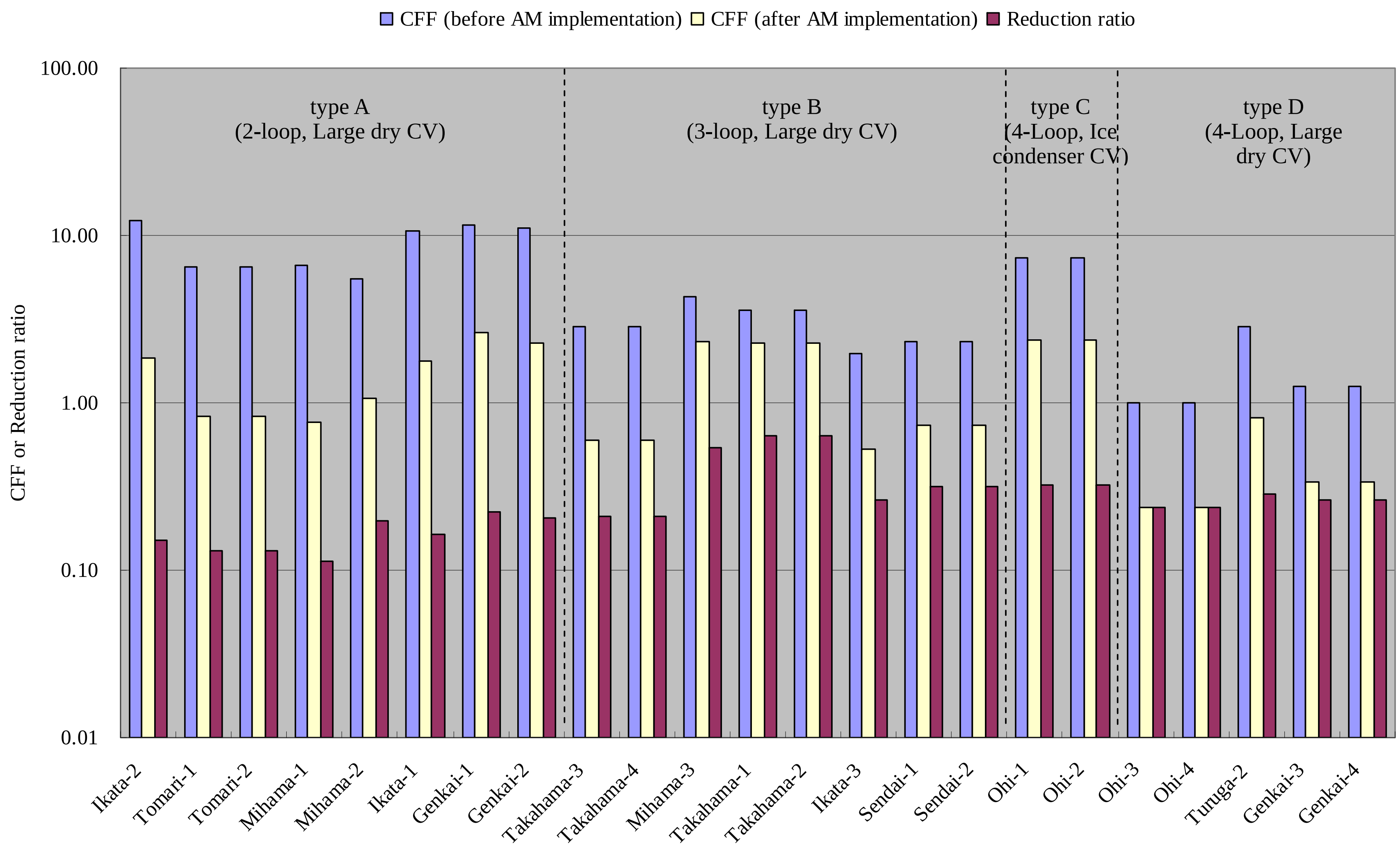

Figure 4 Comparison of containment functional failure frequencies before and after AM implementation (PWR)

PROGRESS IN THE IMPLEMENTATION OF SEVERE ACCIDENT MEASURES ON THE OPERATED FRENCH PWRs – SOME IRSN VIEWS AND ACTIVITIES

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Since the 1990's, Severe Accident Management Guidelines have been developed in France to help the PWR plant operators and emergency teams in limiting the consequences of any postulated severe accident. These guidelines have been progressively improved by taking into account the results of R&D activities and reactor studies. Some plants modifications have been decided accordingly and are (or will be) implemented, such as containment filtered venting system, hydrogen recombiners, material access closure system reinforcement, instrumentation for hydrogen release measurement in containment or means for vessel rupture detection. Significant progress has been currently made from the methodological point of view, with the achievement of L2 PSAs (by EDF and IRSN), and the development by EDF of a severe accident safety standard. This document presents the current knowledge on the plant behaviour in the case of a severe accident, and some "severe accident" specific requirements concerning the systems. The L2 PSAs and the severe accident standard are now seen as some helpful tools for the review of the severe accident issues and identification of new plant improvements.

The paper provides some details on specific SAM designed for the French PWRs and the methodology used in France to review the severe accident safety issues. It presents also some examples to show that complementary research activity results are still needed to definitely validate some parts of the SAM strategy (e.g. impact of in-vessel core flooding , advantage and disadvantage of the reactor cavity flooding before vessel rupture, hydrogen combustion in case of spray system activation, MCCI, …).

The final part of the paper provides some prospects taking into account the future situation in France with the simultaneous operation of Gen III reactors (EPR) and Gen II PWR, with large design differences impacting the risk of severe accident. This situation will certainly maintain the motivation at IRSN to continue the effort in identifying possible improvements of current PWR design vs. the severe accident challenge.

1. Introduction

Since the 1990's, Severe Accident Management Guidelines have been developed in France to help the PWR plant operators and emergency teams in limiting the consequences of any postulated severe accident. These guidelines have been progressively improved by taking into account the results of R&D activities and reactor studies. Some plants modifications have been decided accordingly and are (or will be) implemented, such as containment filtered venting system, hydrogen recombiners, material access closure system reinforcement, instrumentation for hydrogen release measurement in containment or means for vessel rupture detection. Significant progress has been currently made from the methodological point of view, with the achievement of L2 PSAs (by EDF and IRSN), and the development by EDF of a severe accident safety standard. This document presents the current knowledge on the plant behaviour in the case of a severe accident, and some "severe accident" specific requirements concerning the systems. The L2 PSAs and the severe accident standard are now seen as some helpful tools for the review of the severe accident issues and identification of new plant improvements.

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The final part of the paper is aimed at provides some prospects taking into account the future situation in France with the operation of Gen III reactors (EPR) and Gen II PWR, with large design differences impacting the risk of severe accident. This situation will certainly maintain the motivation at IRSN to continue the effort in identifying possible improvements of current PWR design vs. the severe accident challenge.

2. PWRs in operation

Three series of Gen II reactors are operated in France (900 MWe, 1300 MWe and 1450 MWe). Like most Gen II reactors, the severe accidents were not taken into account in the initial plant design. The following table provides some information on the design of these plants.

Table 1 – Some features of the French PWRs in operation

	900 MWe PWR	1300 MWe PWR	1450 MWe PWR
Started			
Loops	3	4	4
Safety injection	2 high pressure trains (HP) 2 low pressure trains (LP)	2 medium pressure (MP) trains, 2 BP trains	2 MP trains 2 BP trains
Accumulators	3	4	4
Specific procedures for additional water injection means	Yes, including connection with neighbouring plant	Yes	Yes
Containment	Single, liner, design pressure: 5 bar abs -CPY series	Double, design pressure: 4,8 bar abs -P4 series, 5,2 bar abs - P'4 series	Double, design pressure: 5,3 bar abs

Regarding severe accident issues, the oldest 900 MWe series is somehow in advance due to the recent 3rd periodic safety review (2004-2008), which included a large part for severe accident issues. For the first time, L2 PSA has been used during a periodic safety review (PSR).

Some of the conclusions from the 900 MWe series are now reported to the 1450 MWe series during the 1st periodic safety review but the next effort (in the regulation process) will concern the 3rd periodic safety review of the 1300 MWe series (2010-2014). A reference L2 PSA is being developed by EDF and an independent LS2 PSA is developed by IRSN (acting as TSO) to support the review.

3. Existing severe accident measures on operated PWRs

Since many years and in different contexts (WASH-1400 in 80's follow-up, TMI2 accident follow-up in 90's and today achievement of PSA2), activities have been performed in France to understand what could be the progression of a severe accident for this type of reactor and to identify some reasonable improvements of the initial design Gen II PWRs in order to mitigate the consequences of such accidents.

This activity complements the efforts in increasing the plant safety (since TMI2 accident) that have mainly concerned the prevention of the severe accidents (core degradation).

The following chapters provide some general information on material and provisions useful for the mitigation of a severe accident and on the current severe accident guidelines.

This information is not exhaustive but tries to introduce the current situation.

3.1. Some key systems and material provisions to limit the accident consequences or to make easier the accident management

<u>Reactor Containment building</u>

For all French Gen II PWRs, the normal behaviour of containment (in the design) is associated to leakage rates that are low enough to guaranty that the radiological consequences of a severe accident would be limited enough to be managed by the emergency organization. Main issues regarding severe accident concern the situations that may lead to some degraded containment tightness and the demonstration that the probability of such situations is very low (practically eliminated).

For all French Gen II PWRs, the design pressure of reactor containment building is about 5 bar abs, which is below the extreme loading that could be calculated for a severe accident with pessimistic assumptions (in case of DCH and hydrogen deflagration for example). This situation justifies the achievement of detailed analyse of the beyond design behaviour of the reactor containment building and the implementation of severe accident measures aiming at limiting the potential loading on the containment.

For most reactors of the 900 MWe series, the detailed study of the beyond design behaviour has shown that realistic mechanical resistance is well above the design pressure thanks to the internal steel liner and that a relative weak point was the closure system of material access penetration. For each reactor, a reinforcement of this closure system is planned at the 3rd decennial inspection.

For the 1300 MWe series reactors, which were not equipped with an inner steel liner, but with an annular space with filtration/ventilation ducts, the beyond design behaviour analysis is still in progress but the ultimate (calculated) resistance pressure of the internal containment should be somehow lower than for the 900 MWe series reactor. For the most pessimistic severe accident loading, the containment efficiency is supposed to depend also on the release collection (and filtration) through the annular space. This issue will be examined in detail during the preparation of the 3rd PSR for this PWR series (2010-2014).

Pressurizer safety valves

The RCS safety valves have a key role in case of severe accident to limit the in-vessel pressure (and avoid DCH or induced steam generator tube rupture). Opening the pressurizer safety valves is one of the first actions that should be achieved by the operator at the beginning of the core degradation.

To avoid any unwanted closure of these valves (due for example to electrical cables failure after irradiation) during the in-vessel progression of accident, EDF has proposed a modification of the electrical command of the valves. This modification will be implemented during the 3rd decennial visit for 900 MWe reactors and is being examined for other series.

Containment venting system

A containment venting system has been installed on all French PWR in the 90's to avoid any containment failure in the long term phase of accident (MCCI). A metallic filter in the containment can retain a large part of aerosol and a sand filter, outside the containment should retain the remaining aerosols. The venting line is heated to avoid the steam condensation and to limit the risk of hydrogen combustion within the venting line.

This system is supposed to retain efficiently the aerosols and limit the long term impact of a severe accident. Some technical exchanges are now in progress between EDF and the French Safety Authority plus IRSN on the interest to improve the capabilities of this venting system for iodine filtration.

For some plants with particular design of the foundations (earthquake), it may be necessary to depressurize with more efficiency the containment during MCCI phase; the containment venting has an increased capacity and a specific procedure is available. Some technical reviews are still in progress at IRSN to check the compatibility of such procedures with emergency preparedness.

Passive Autocatalytic Recombiners (PARS)

PARs have now been installed on all operated French PWRs and are designed on the following basis:

- hydrogen combustion pressure peak in the containment should not exceed the beyond design containment strength,
- the molar hydrogen mean concentration in the containment should stay below 8 %,
- the local molar hydrogen concentration should stay below 10 % (indicative value).

The development of L2 PSA provides today the opportunity to validate the design of PARS and to identify some low probability sequences that may conduct to exceed the design criteria (in particular the situations that may lead to high kinetics of hydrogen production).

Instrumentation for hydrogen

Following a requirement of the French Safety Authority, EDF has developed some specific instrumentation that should help the operators and emergency teams in understanding the situation regarding hydrogen release during a severe accident. This instrumentation is based on thermocouples installed on PARs and uses the high temperature of the catalyser plates during the hydrogen recombination with oxygen.

It will be installed for the 900 MWe series during the 3rd PSR but some technical elements are still expected from the utility (justification of the number of captors and their localization, guideline for the operators or emergency teams).

Instrumentation for the vessel failure detection

Following a requirement of the French Safety Authority, EDF has developed a specific instrumentation able to inform the operators and emergency teams on the occurrence of a vessel rupture. This instrumentation is based on a thermal couple located in the reactor cavity. Some technical elements are still expected from EDF on the availability of the measure in all situations but it will be installed also during the 3rd PSR of 900 MWe series.

Containment Heat Removal System (spray system)

For IRSN, the containment heat removal system must be considered as a key system in case of severe accident because it allows the deposit of fission product and may be the unique solution to avoid the containment pressurization.

Today, the only requirement specific to severe accident on this system concerns the abilities to close the isolation valves in severe accident conditions in case of leakage in the auxiliary building.

Role of the CHRS for the short and long term phase of a severe accident has been discussed and proposals are expected from EDF by the Safety Authority. This issue may be difficult to deal with, in particular for the demonstration of operability of a long term sump recirculation.

Isolation system

Some specific procedures have been established by EDF (within EOPs) to control the efficiency of the containment isolation system. Specific requirements are being defined for the circuits (called "3rd barrier extension") that may stay open during the accident.

The studies have been mainly based on a deterministic basis and, for IRSN, the development of L2 PSA should provide the possibility to check the efficiency of the system and procedure. Some modelling proposals are expected from EDF for the next version of L2 PSA. Nevertheless this topic is considered by IRSN as technically difficult to deal with, in relation with the periodic test of isolation components).

Safety Injection system

The safety injection may be crucial in the management of a severe accident, either to stop the in-vessel accident progression (see TMI2 accident) or to maintain some long term corium cooling. Like CHRS, the demonstration of the operability of a long term operation of safety injection system through sump recirculation is still not done.

3.2. Severe accident guidelines

Severe accident management guidelines (SAMG) have been developed by EDF since many years, with the objective to define actions based on the containment protection (in the emergency operating procedures (EOP), before SAMG application, the main objective is to assure the short and long terms core cooling).

The latest versions of SAMG include some specific recommendations regarding in-vessel water injection to limit the risks on the reactor containment, for example:

- water injection should be avoided at the beginning of core degradation if the flow rate is not sufficient to compensate both residual power and oxidation power (the idea is to avoid hydrogen production with high kinetics regarding PARs (passive autocatalytic recombiners) capabilities); from a practical point of view, the safety injection system is the only mean able to cope with this recommendation;

- water injection should be avoided after few hours of core degradation if a sufficient break does not exist on the reactor cooling system (RCS); this condition has been drafted to avoid RCS pressurization by injected water vaporization and then DCH;

The spray system activation may also be delayed (6 hours) to keep as far as possible the containment atmosphere inert during the in-vessel hydrogen production phase.

Regarding the international practice, the severe accident guidelines for the French PWRs may appear singular because it gives a very high importance on the prevention of early containment failure and conducts to limit the possibility of core cooling when the water injection is prohibited.

For IRSN, the current situation is justified regarding the state of knowledge on severe accident in France but a better understanding of the technical basis used in other countries to establish the severe accident management guidelines (case where water injection is recommended whatever the situation) would be certainly useful. Unfortunately, this level of information is rarely available in the public domain …

Some updated versions of the SAMG are expected from EDF in near future with complements related to the progress in the severe accident knowledge, the new materials installed on the plants and mostly the management of the long term phase of an accident.

4. A new tool for the safety regulation: the severe accident safety standard

As explained above, the severe accidents were not included in the initial design of the PWR. Nevertheless, some specific plant modifications are implemented to improve the plant robustness in case of accident (mainly for the mitigation of the consequences of a severe accident). Progressively the situation became difficult to manage in terms of safety regulation due to the lack of clear safety requirement that should be applied on the operated plants for the severe accidents issues, while many progresses were obtained on the knowledge on the severe accident phenomenology knowledge.

In that context, and after several meetings of the French "Permanent Group", the French Safety Authority asked EDF in 2001 to propose a severe accident safety standard containing at minimum the approach and objectives for prevention and mitigation of risks associated with serious accidents, the studies necessary to demonstrate compliance with the objectives and the practical provisions and their design basis. This standard should also take into account aspects related to radiation protection of workers and rely on the initial results of level 2 PSA in order to prioritize requirements in function of the level of potential releases for the accidental scenarios considered.

Several versions for this standard have now been established by EDF and successively reviewed by IRSN. The last version of the safety standard includes two parts:

- the safety requirements (approach and safety objectives in terms of prevention and mitigation of severe accident, the studies necessary to demonstrate compliance with the objectives, the current practical provisions and their design basis, the requirement applied to materials),
- the synthesis of the operated plants status related to severe accident (synthesis of existing knowledge on severe accident progression, the status of material behaviour in severe accident conditions, a demonstration that the probabilistic safety goals are achieved and the results of radiological consequences assessment for reference scenarios); this synthesis is supposed to show that the safety requirements are met.

The last review by IRSN and positions of the “French Permanent Group” has conducted the Safety Authority to ask for some complements (objective of a continuous improvement of plant safety, in particular for radiological consequences or probabilistic safety goals; requirements linked to the long term management of the plant in case of severe accident, materials classification…) but the main conclusion is that this standard is now seen as a progress and can be used for the identification of the plant improvements related to accident prevention and consequences limitation. It should be applicable during the next PSR of the 1300 MWe PWRs.

For IRSN, the use of a safety standard for the severe accident, in conjunction with both deterministic studies, progress of R&D and development of L2 PSA will certainly help in the analyse of the severe accident issues and also in the capitalization of knowledge needed in a perspective of potential plant life extension.

5. Severe accident risk quantification and reduction – Present and future activities at IRSN for the PWR severe accident management

The severe accident risk quantification and identification of reduction possibilities for the French PWRs will orientate IRSN futures activities in that field for Gen II reactors. This activity remains based on IRSN independent analyses (R&D programmes, codes developments, L2 PSA developments, deterministic studies…) whose conclusions are used during the safety review process.

The present chapter provides some insights on these activities.

5.1. Some conclusions from the L2 PSA of the 900 MWe PWRs developed by IRSN

Last version of the 900 MWe PWR L2 PSA developed by IRSN was achieved beginning of 2009[1] and help in ranking the different remaining risks. It covers the internal accident initiators for shutdown and power state of reactor. This study is not the reference study for the plants (provided by EDF) but tries to gather the different contributions of the IRSN teams involved in severe accident activities. An equivalent study is being performed for the 1300 MWe series but the first global results will be available only at the end of 2009.
It is therefore intended through level 2 PSA developments to contribute to the continuous improvement of reactor safety approach. The results obtained are used to identify relatively weak points in facilities, or specific issues for which additional knowledge may be required.
Few examples from the 900 MWe PWR are described below:

- the frequency of the heterogeneous dilution sequences remain relatively high, considering the potential associated impact of such accident and the utility efforts to limit as far as possible the possibility to send some non borated water in the reactor and justify to pursue activities on this topic (elimination of such situation, assessment of the consequences);
- the calculated frequency of the loss-of-containment-integrity sequence after a steam explosion in the reactor pit appears relatively high; this point is commented in next chapter;this issue is currently the subject of technical effort by EDF and IRSN and will be reviewed again in a near future. Additional studies regarding induced loads and containment strength under this type of loading seem to be necessary (IRSN participates actively in SERENA programme to improve the validation of MC3D code for steam explosion loading calculation and develops some 3D modelling of the structure around the vessel pit to obtain a better characterization of their

[1] Rapport Scientifique et Technique de l’IRSN – 2008 - L’EPS de niveau 2 pour les REP 900 : du développement aux enseignements de l’étude. E. Raimond, N. Rahni, T.Durin, K. Chevalier-Jabet (www.irsn.fr).

mechanical strength); in addition, the presence of water in the reactor pit and its favourable effect on cooling the corium after vessel failure and preventing basemat penetration must also be considered and make this issue quite difficult to close;

- the study indicates a risk of containment failure due to hydrogen combustion after in-vessel water injection; the calculated frequency of this type of scenario is low, due to the precautions already taken by the operator and emergency teams through SAMG application (prohibition of low-flow water injection at the beginning of core degradation); nevertheless, IRSN considers that the actions recommended in the severe accident guidelines could and should be optimized;
- certain sequences correspond to core-meltdown-with high vessel pressure, with risk of containment bypass in the case of steam generator tube rupture, despite the implementation of specific control measures to depressurize the reactor coolant system before (or during, at the latest) core degradation; these sequences will be re-examined in detail with the objective to check if current provision to prevent release in such situation are sufficient;
- some sequences seem potentially leading to the opening of the containment venting system in less than 24 hours after to the beginning of core degradation (while the SAMG recommends to avoid opening the containment venting system before 24 hours); this will be re-examined in detail in relation with the validation of the severe accident codes;
- the study shows the importance of the ultimate pressure capacity of the containment (i.e. beyond the initial design pressure) to limit the accident consequences for the more extreme loading (mainly H2 combustion and DCH) and recalls the importance of maintaining containment structures in excellent condition during plant life. It also shows the relevance of making changes to reinforce containment structures beyond their initial design strength (reinforced equipment hatch).

Generally speaking, the level 2 PSA study and the associated supporting studies provided a vast amount of knowledge on the potential consequences of a severe accident on the PWR reactor. This knowledge is established using state-of-the-art R&D techniques and numerical simulation tools. It has improved communication with EDF on these very complex issues but exhibits also some limits in the state-of-the-art. The following chapter provides some examples.

5.2. The management of water during a severe accident : a key issue with no sufficient technical basis?

As explained above, the management of the water during a severe accident is seen as a topic where some optimization of the management in severe accident is needed. Some details have been provided during recent the joint OECD-NEA / EC-SARNET meeting on the in-vessel coolability[2] and are summarized here.

Water injection on the corium during the severe accident progression would be the more efficient way to stop the accident progression on a Gen II PWR (like in TMI2 accident). It may be crucial because these plants were not designed with a core catcher for the case of vessel rupture and the demonstration that the basemat will not be penetrated by the corium is still to be done. The gravity of an accident with basemat penetration would nevertheless be higher (ground contamination, uncontrolled leakage) than without, and the "accident managers" would certainly keep this in mind.

[2] Importance of the in and ex-vessel coolability in case of severe accident for the French PWRs. Some views from L2 PSA and perspectives. E.Raimond, C. Caroli, R. Meignen, N. Rahni, B. Laurent. OECD-NEA / EC-SARNET workshop on in-vessel coolability, Issy-Les-Moulineaux, France, Oct 2009.

But for IRSN (and also EDF), this cannot justify to introduce in the SAMG any risk of early containment failure due to the water injection.

At IRSN, we have to consider that today, and after 30 years of research on severe accident, the technical basis to deal with some of the following issues remains poor:

- what would be the increase of hydrogen production rate in case of in-vessel water injection? Does it really justify avoiding water injection in some reactor configurations? Can the spray system be used to decrease the containment pressure and limit the amplitude combustion peak?
- what would be the RCS pressure rise in case of late in-vessel water injection? what would be the vessel behaviour? what is the link with the DCH risk?
- is the presence of water in the reactor pit (before vessel rupture) positive (corium cooling) or negative (steam explosion, containment pressurisation, corium spread area) on the accident progression?

This situation had an impact on the IRSN priority for existing severe accident programmes in order to complete the needed technical basis for SAMG:

- the development of a validated 2D modelling for degraded core is now in progress in ICARE-CATHARE then ASTEC V2 codes, supported by the experimental PEARL programme; hopefully will the new versions of ASTEC V2 help in reducing the uncertainties on positive and negative consequences of in-vessel water injection,
- the comprehension of the hydrogen combustion development mechanism under spray conditions is studied through collaborations with CNRS,
- the comprehension of the vessel failure condition (delay and break size) is still studied with some specific experimental and modelling effort,
- the analysis of ex-vessel steam explosion risk remained at high priority through the improvement and the validation of the simulation tools (MC3D code, SERENA programme…).

The spreading capacity of the corium when it falls in the water of the reactor cavity is now seen as a subject of interest (from 1300 MWe PWR L2 PSA, because for these reactors, the reactor cavity is connected to a corridor that increase significantly the corium spreading area) and with no experimental data. Some modelling efforts have been planed at IRSN in 2010 (with MC3D and ASTEC V2 codes) and may conduct to some complementary need in terms of experiments. Exchange of experience with other countries may have interest.

Following the IRSN and the French permanent Group recommendations, the French Safety Authority also required from EDF to present, at the 3rd PSR of the 1300 MWe series, a synthesis of the advantages and disadvantages of the different strategies for the flooding of reactor cavity, and also the specific means associated to each strategy. This synthesis must be reinforced by the results of a level 2 PSA. The French Safety Authority also required comparing the following strategies:

- to let the reactor cavity be filled by the CSHRS water (spray system),
- to fill voluntary the reactor cavity to the primary loop level or to sump level,
- to keep the reactor cavity empty until vessel rupture and to flood voluntary the reactor cavity after the vessel rupture.

5.3. The source term assessment

In France, the emergency preparedness (distances of counter-measures applications) was defined on the basis on a reference source term (S3) for severe accident (core degradation and vessel rupture with late containment venting). This approach is evolving progressively with the development and use of L2 PSA allowing a more precise categorisation of the accident scenarios and source term calculations.

The integration of the results of the ISTP programme in the basic assumptions for the source term calculation is now in progress (either in ASTEC code or in the very fast-running release code of L2 PSA). The new modelling developed at IRSN for iodine and ruthenium behaviour in containment and will justify an update of the reference source term calculations in 2010.

Further evolutions of these assumptions and calculations are already planned (integration of the CHIP programme result on the iodine form transferred from RCS to containment) and some complements to the ISTP programmes are also proposed, in particular to validate the assumptions concerning the long term phase of a severe accident or examine some specific mean for the release reduction.

The position of the updated reference source terms regarding the objectives defined in the severe accident safety standard will be examined during the next periodic safety reviews. Some complementary provisions may be examined to limit as far as possible the amplitude of the release.

6. Towards some higher requirements in relation with plant life extension?

For 900 MWe and 1300 MWe reactors, the preparation of the 3rd decennial review has and will provide an opportunity to make an inventory of the severe accident risks, with a better formalization (development of severe accident safety standard and L2 PSAs). Some plant design modifications have been defined (or will be for the 1300 MWe reactors) for issues with undeniable ratio cost / safety benefits.

The exercise shows also clearly some field where the situation remains complex, in particular the management of water during severe accident progression, and where some progress from the R&D are needed.

But, in near future, will be examined in France the EDF request for plant extension of life beyond 40 years. Gen II and Gen III (EPR) reactors will cohabit during a long period of time and this will conduct to a societal wish of progress in the safety of Gen II reactors. For IRSN, both accident prevention and accident consequence mitigation will have to be examined. Future safety improvements cannot be limited to prevention and the consequence mitigation of a severe accident is considered as a key issue. In the framework of plant life extension, the current difficulties on topics like water injection will have to be solved.

The severe accident safety standard should be a relevant tool to define possible additional requirements in relation with the Safety Authority demands. For IRSN, this near future should be a turning point in the severe accident activities, passing from a long period of knowledge acquisition to the definition of practical (reasonable) provisions allowing a better control of the accident consequences.

First discussions between EDF, the Safety Authority and IRSN have been initiated in 2009 in the broader framework of plant life extension and will be intensified in 2010.

7. Conclusion

The paper tries to present the progress obtained in the severe accident management on French Gen II PWR with practical implementations of measures to limit the accident consequences or to make easier the accident management. It shows also that some results from the R&D field are still expected for some specific issues, in particular for the water management during the accident and the source term assessment. The future activities will be linked to the plant life extension with the definition of possible additional safety requirement and a research of practical and reasonable provisions allowing a better control of the accident consequences.

Session 2

Perspectives on Severe Accident Mitigation Alternatives for U.S. Plant License Renewal

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As part of the environmental review performed for license renewal for U.S. plants, licensees perform a severe accident mitigation alternative (SAMA) analysis. A SAMA analysis is a systematic search for potentially cost beneficial enhancements to further reduce nuclear power plant risk. This paper will provide the history of events that led to the requirement for conducting SAMA analyses and the process by which this analysis is performed. The paper will review the results from the SAMA analyses completed to date, including: (i) the onsite and offsite economic impacts of a severe accident and their typical estimated values; (ii) the types of enhancements considered/evaluated in a SAMA analysis; (iii) examples of the potentially cost-beneficial improvements (SAMAs) identified through the analyses; and (iv) the level of risk reduction that can be achieved through SAMA implementation. Finally, the paper will offer perspectives and insights on the process and results.

1. Historical context and regulatory basis

Section 5.4 of the U.S. Nuclear Regulatory Commission's (NRC), "Generic Environmental Impact Statement for License Renewal of Nuclear Plants" (NUREG-1437[1]) provides background information on the genesis of the SAMA regulatory requirement. This discussion is summarized briefly here for the purpose of providing the necessary context to the reader. Note that NUREG-1437 is in the process of being revised, but this does not affect the historical discussion provided below.

In 1980 NRC issued an interim policy statement on the consideration of severe accidents in environmental impact statements (EISs) applicable to Construction Permit and Operating License applications submitted on or after July 1, 1980[2]. The policy statement states that it is "the intent of the Commission that the staff take steps to identify additional cases that might warrant early consideration of either additional features or other actions which would prevent or mitigate the consequences of severe accidents." These features have become known as severe accident mitigation design

[1] NUREG-1437, Volume 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," U.S. Nuclear Regulatory Commission, May, 1996.
[2] 45 *Federal Register* 40101, Statement of Interim Policy, "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969," June 13th, 1980.

alternatives (SAMDAs) when applied at the design stage, or SAMAs when applied in the context of extending an existing license. But the scope of the analyses is the same.

In August 1985, NRC issued its policy statement on severe reactor accidents. That policy statement presented NRC's conclusions that existing plants pose no undue risk to public health and safety and that there was no present basis for immediate action on generic rulemaking or other regulatory changes for those plants because of severe accident risk. Nevertheless, it called for each licensee to perform an analysis designed to discover instances of particular vulnerability to core melt or unusually poor containment performance given a core-melt accident. NRC believed that this policy statement was a sufficient basis for not requiring a consideration of SAMDAs at the operating license review stage for previously constructed plants. However, a 1989 court decision ruled that such a policy statement was not sufficient to preclude a consideration of SAMDAs and that such a consideration is required for plant operation[3].

Relative to the evaluation of potential improvements for existing reactors in the U.S., the NRC gained considerable experience during the 1980s and 1990s via (a) staff assessments of SAMDAs for the Limerick, Comanche Peak, and Watts Bar plants performed as a result of the aforementioned *Limerick Ecology Action* court decision, (b) the containment performance improvement program[4], (c) the individual plant examination (IPE) program[5], and (d) the implementation of severe accident management programs at all nuclear power plants as part of an industry initiative. These regulatory programs and initiatives provide assurance that any major vulnerabilities to severe accidents have been identified and addressed, and that the residual level of risk is low. As a result, major plant modifications would not be expected as a result of a SAMA analysis. As stated in NUREG-1437, "the NRC expects that a site-specific consideration of severe accident mitigation for license renewal will only identify procedural and programmatic improvements (and perhaps minor hardware changes) as being cost-beneficial in reducing severe accident risk or consequence." This expectation has generally been met as discussed below.

2. Definition and scope

As described above, the term SAMA refers to an additional feature or action which would prevent or mitigate the consequences of serious accidents. SAMA analysis includes consideration of (i) hardware modifications, procedure changes, and training program improvements, (ii) SAMAs that would prevent core damage as well as SAMAs that would mitigate severe accident consequences, and (iii) the full scope of potential accidents (meaning both internal and external events).

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[3] *Limerick Ecology Action v. NRC*, 869 F.d 719 (3rd Cir. 1989)

[4] NRC examined each of five U.S. reactor containment types (BWR Mark I, II and III; PWR Ice Condenser; and PWR Dry) with the purpose of examining the potential failure modes, potential enhancements, and the cost benefit of such enhancements. This examination has been called the containment performance improvement (CPI) program and was documented in a series of reports (NUREG/CR-5225; NUREG/CR-5278; NUREG/CR-5528; NUREG/ CR-5529; NUREG/CR-5565; NUREG/CR-5567; NUREG/CR-5575; NUREG/CR-5586; NUREG/CR-5589; NUREG/CR-5602; NUREG/CR-5623; NUREG/ CR-5630).

[5] In accordance with NRC's policy statement on severe accidents, each U.S. licensee was requested to perform an individual plant examination (IPE) to look for vulnerabilities to both internal and external initiating events (Generic Letter 88-20, Supplements 1-4). These examinations consider potential improvements on a plant-specific basis. Results are described in NUREG-1560 and NUREG-1742, respectively.

3. Major steps in a SAMA evaluation

3.1 Identification and characterization of leading contributors to risk

The first step of a SAMA evaluation is to identify and characterize the leading contributors to core damage frequency (CDF) and offsite risk based on a plant-specific risk study or applicable studies for other plants. In practice, maximum use is made of the plant-specific risk model for characterizing the dominant contributors to risk and identifying candidate SAMAs to address these contributors. The contribution of external events is considered to the extent that it can be supported by available risk methods, because external events can affect whether or not a SAMA is cost-beneficial (greater reduction of risk). In some cases, the SAMA may specifically relate only to external events (e.g., a modification related to a piece of hardware that is only damaged during seismic events). In other cases, a SAMA that may have been identified based on internal event considerations (e.g., use of portable generators to power equipment in a station blackout (SBO)) may also have benefits in externally initiated events (e.g., a seismic induced SBO).

3.2 Identification of candidate SAMAs

The next step in the process is to identify candidate SAMAs. Although the greatest level of risk reduction might be achieved by a major plant modification, lower cost alternatives might eliminate a substantial fraction of the risk and have a greater net benefit. In identifying SAMAs, the lowest cost means of achieving the functional objectives should not be overlooked. As an example, developing procedures to connect hydrogen igniters to portable on-site generators, rather than installing additional igniters with dedicated batteries, would be more cost-beneficial if it achieved the same reduction in risk. One key tool used in identifying SAMAs is the use of PRA importance measures (e.g., Risk Achievement Worth, or RAW) to identify important basic events from the PRA (e.g., equipment failures and operator actions) and candidate SAMAs to address these basic events. In addition, a list of SAMAs that have been found to be cost-beneficial at other plants in the past should be reviewed to identify candidate SAMAs for the plant being analyzed.

3.3 Estimation of risk reduction and implementation cost estimates

Once candidate SAMAs have been identified, an initial screening is performed to determine which SAMAs can not be cost-beneficial. A rough implementation cost estimate is developed for each SAMA. If the cost estimate exceeds the bounding condition of the maximum attainable benefit (i.e., the benefit of eliminating all plant risk) then the SAMA is screened out from further consideration because it cannot be cost-beneficial. In addition, candidate SAMAs from other plants that are not applicable to the plant being analyzed (e.g., due to design or risk-profile differences) are screened out.

For each SAMA that survives this initial screening, a benefit assessment is performed to address how the change would affect relevant risk measures (core damage frequency, offsite population dose in person-Sv [person-rem], offsite economic cost risk - OECR). This includes a description of how the change was implemented/credited in the PRA model (i.e., what changes were made to the basic events, fault trees, or event trees). For example, the impact of a procedural change might be estimated by reducing the associated human error probabilities. In some cases, bounding assumptions are used that capture the maximum possible benefit of the change, such as assuming that improvements to assure reactor cavity flooding would eliminate all containment failures due to core-concrete interactions.

A cost assessment is also performed for each SAMA. Cost estimates for hardware modifications can be taken from past studies performed for a similar plant, or developed on a plant-specific basis. Cost estimates are generally conservative in that they neglect certain cost factors (e.g., surveillance/maintenance, the cost of replacement power during implementation), therefore tending to increase the number of potentially cost beneficial SAMAs. Typically screening estimates are used for initial assessments and refined as appropriate if a SAMA is potentially cost-beneficial. In general, hardware costs are several hundred thousand to a million dollars; procedure changes range from ~$50K to $200K for complex changes with analysis and operator training impacts.

The licensee is expected to assess the impact of major uncertainties on the results, to demonstrate the robustness of the conclusions. Sensitivity analyses are typically performed, examples of which include: (1) the estimated benefits are increased by the ratio of the 95th percentile CDF to mean CDF (to address uncertainty in the CDF analysis) and (2) alternative discount rates are used in the cost-benefit analysis (e.g., 7% versus 3%) to assess sensitivity of results to the assumed discount rate.

3.4 Identification of SAMAs that are potentially cost-beneficial

To identify SAMAs that may be cost-beneficial, the net value of each SAMA is estimated. The NRC maintains two documents that provide guidance in this area: NUREG/BR-0058[6] and NUREG/BR-0184[7].

The net value of a particular SAMA can be generated from the following basic equation:

$$\text{Net Value} = (\text{APE} + \text{AOC} + \text{AOE} + \text{AOSC}) - \text{COE}$$

where:

APE = averted public exposure costs
AOC = averted offsite property damage costs
AOE = averted occupational exposure costs
AOSC = averted onsite costs = averted cleanup and decontamination costs (ACC) + averted replacement power costs (ARPC)
COE = cost of enhancement

Table 1 provides information on each of the averted cost components (offsite and onsite economic components of the maximum attainable benefit [MAB]), including references to the relevant sections of NUREG/BR-0184, the relevant supporting parameters, and aggregated values from the licensee submittals for all approved U.S. license renewals as of August 2009. The costs represent the dollar value of completely eliminating all internal event risk[8]. The MAB cost factors and total can vary widely from plant to plant due to differences in baseline risk (e.g., baseline CDF), and differences in population and land values surrounding the plant site.

[6] NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," U.S. Nuclear Regulatory Commission, September 2004.
[7] NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," U.S. Nuclear Regulatory Commission, January 1997.
[8] With the exception of data from a few licensees whose dollar values include eliminating external event risk as well.

Table 1. Supporting Information for Averted Cost Components

Cost Factor	Significance	NUREG/ BR-0184 Section	Related Parameters	Average (and Ranges) of Maximum Attainable Benefits from Licensee Submittals for All Approved License Renewals
APE	Offsite exposure	5.7.1	Δperson-Sv [Δperson-rem] (from the Level 3 PRA analysis)	$370K ($12K – $1,500K)
AOC	Offsite economic	5.7.5	ΔOffsite Economic Cost (from Level 3 PRA) and accident frequency (from Level 2 PRA)	$400K ($10K – $2,700K)
AOE	Onsite exposure	5.7.3	Immediate occupational dose (33 person-Sv [3,300 person-rem][a]) Long term occupational dose (200 person-Sv [20,000 person-rem][a])	$17K ($1K – $130K)
ACC	Onsite economic	5.7.6.1	Onsite cleanup and decontamination cost ($\$1.1\cdot10^{9}$ single event[a], present worth)	$870K ($37K – $6,300K)
ARPC	Onsite economic	5.7.6.2	Plant power level	
Total[b]				$1,700K ($110K - $8,700K)

[a] From NUREG/BR-0184

[b] The range for total costs represents the range of total costs cited in the licensee submittals, not a summation of the ranges for the individual components.

3.5 More detailed analysis for remaining SAMAs

The final step in the process is a more detailed analysis of the SAMAs that were identified as being potentially cost-beneficial in the steps above. This may include a more detailed (i.e., more realistic and less bounding) evaluation of the potential benefits of the SAMA (i.e., rather than assuming that the SAMA eliminates all CDF contributors, only those sequences relevant to the SAMA are included). It may also include a more detailed development of the cost associated with the proposed modification (including such things as engineering support, training, hardware costs, and implementation costs). Additional guidance for conducting this step is available in a Nuclear Energy Institute (NEI) document NEI-05-01, Revision A[9]. The NRC staff has recommended that applicants for license renewal follow the guidance provided in NEI-05-01, Revision A, in the staff's Final License Renewal Interim Staff Guidance LR-ISG-2006-03[10].

[9] NEI-05-01 [Rev. A], "Severe Accident Mitigation Alternatives (SAMA) Analysis: Guidance Document," Nuclear Energy Institute, November 2005.

[10] LR-ISG-2006-03, "Final License Renewal Interim Staff Guidance LR-ISG-2006-03: Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses," U.S. Nuclear Regulatory Commission, August 2, 2007.

4. Current Status of SAMA Reviews (as of August 2009)

SAMA/SAMDA evaluations have been completed for initial plant licensing of the following three operating plants[11]: (1) Limerick (1989); (2) Comanche Peak (1989); and (3) Watts Bar 1 (1995). SAMDA evaluations have been completed for the following advanced light-water reactor certified plant designs: (1) CE System 80+ (1995); (2) General Electric Advanced Boiling Water Reactor – ABWR (1995); (3) Westinghouse Advanced Passive 600MW – AP600 (1999); and (4) Westinghouse Advanced Passive 1000MW – AP1000 (2004). To date, SAMA evaluations have been completed for operating plant license renewal applications that were approved for over 30 sites encompassing over 50 units. Table 2 lists the completed SAMA evaluations by plant, nuclear steam supply system (NSSS), and containment type.

Table 2. Completed SAMA Evaluations for Plants with Approved License Renewal

Plant Type	NSSS	Containment Type	Plant Name	Year of License Renewal Approval
BWR	GE 2	Mark I	Nine Mile Point 1	2006
			Oyster Creek	2009
	GE 3	Mark I	Dresden 2 & 3	2004
			Quad Cities 1 & 2	2004
			Monticello	2006
	GE 4	Mark I	Edwin I. Hatch 1 & 2	2002
			Peach Bottom 2 & 3	2003
			Browns Ferry 1, 2 & 3	2006
			Brunswick 1 & 2	2006
			James A. FitzPatrick	2008
	GE 5	Mark II	Nine Mile Point 2	2006

[11] NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Main Report and Supplements available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/.

Plant Type	NSSS	Containment Type	Plant Name	Year of License Renewal Approval
PWR	W 2-Loop	Dry Ambient	R.E. Ginna	2004
			Point Beach 1 & 2	2005
	W 3-Loop	Dry Ambient	Turkey Point 3 & 4	2002
			H.B. Robinson 2	2004
			V.C. Summer	2004
			Joseph M. Farley 1 & 2	2005
			Shearon Harris 1	2008
		Dry Subatmospheric	Surry 1 & 2	2003
			North Anna 1 & 2	2003
	W 4-Loop	Dry Ambient	Wolf Creek 1	2008
			Vogtle 1 & 2	2009
		Dry Subatmospheric	Millstone 3	2005
		Ice Condenser	McGuire 1 & 2	2003
			Catawba 1 & 2	2003
			D.C. Cook 1 & 2	2005
	CE	Dry Ambient	Calvert Cliffs 1 & 2	2000
			St. Lucie 1 & 2	2003
			Fort Calhoun	2003
			Arkansas Nuclear One 2	2005
			Millstone 2	2005
			Palisades	2007
	B&W	Dry Ambient	Oconee 1, 2 & 3	2000
			Arkansas Nuclear One 1	2001

B&W: Babcock and Wilcox
CE: Combustion Engineering
PWR: Pressurized-Water Reactor
BWR: Boiling-Water Reactor
GE: General Electric
W: Westinghouse

5. Insights from SAMA Evaluations

In general, the estimated CDFs for operating plants are relatively low (i.e., less than 10^{-4} per year). In addition, many of the weaknesses uncovered through the IPE and individual plant examination of external events (IPEEE) programs have already been addressed. It is therefore difficult to identify additional changes that both reduce risk substantially and are cost-beneficial, for the above reasons and because: (1) risk is generally driven by multiple sequences while a SAMA generally acts on only one contributor; (2) risk reduction potential is highest at operating plants (versus new reactors still under design), but the cost of implementing design changes within an operating plant is much higher too; and (3) the cost of design changes are lower in advanced light-water reactors given that the plant has not yet been constructed, but the calculated residual risk is so low that even complete elimination of all severe accident risk, if it were possible, would not warrant spending substantial funds.

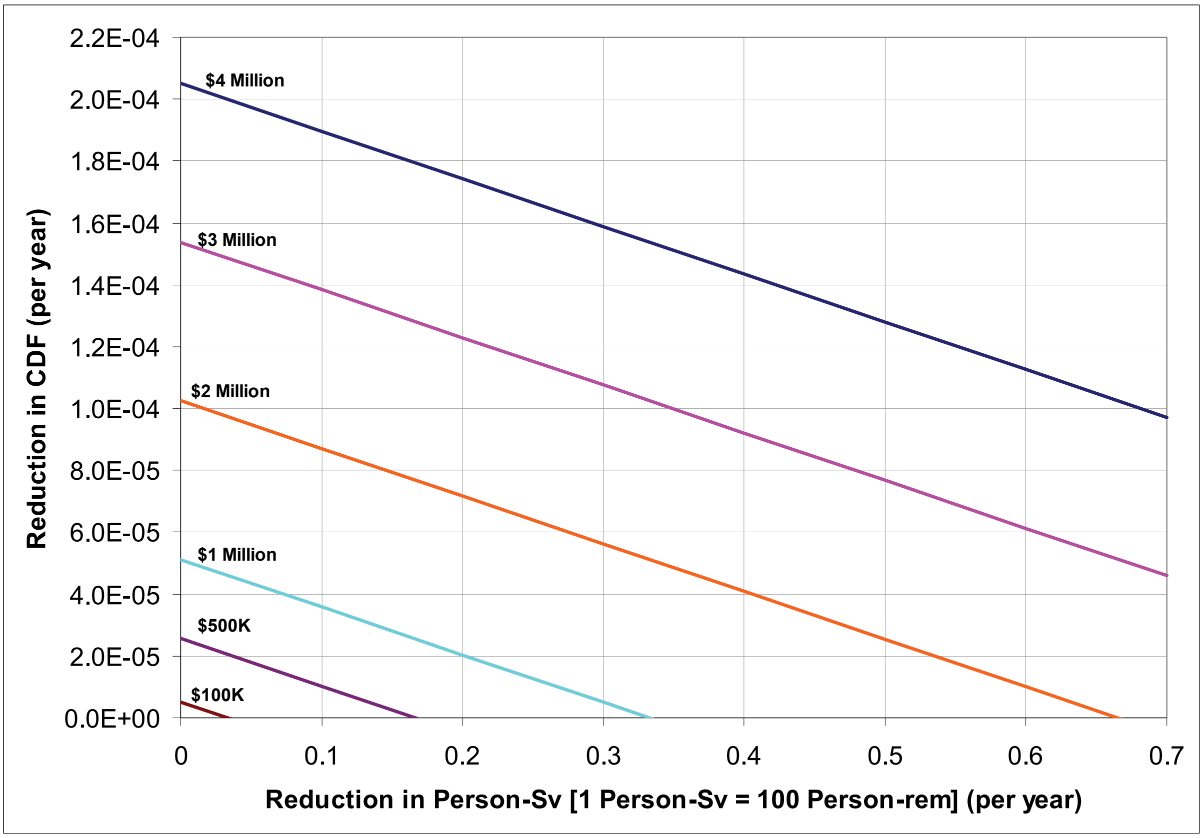
Identification of cost-beneficial changes is most likely for operating plants, where reductions in CDF could be on the order of 10^{-5} per year. At these plants, reduction in averted onsite costs and offiste impacts could justify the expenditure of several hundred thousand dollars. Cost-beneficial SAMAs would most likely be limited to procedure changes and minimal hardware changes. Averted onsite costs (AOSC) is a critical factor in cost-benefit analyses and tends to make preventive SAMAs more attractive than mitigative SAMAs (which improve containment performance but do not impact CDF).

Table 3 shows the average and ranges of CDF, population dose, \$/event, \$/person-Sv [\$/person-rem], and MAB computed for all approved U.S. license renewals as of August 2009. Figure 1 shows typical cost benefit thresholds for different reductions in CDF per year and person-Sv per year, assuming a 3% discount rate, a 20 year term, and other cost factors provided in NUREG/BR-0184 and NUREG/BR-0058.

Table 3. CDF, Population Dose, and Maximum Attainable Benefit Associated with Completely Eliminating Severe Accidents

	Average	Range
CDF (/yr)	4.0×10^{-5}	$1.9 \times 10^{-6} - 3.3 \times 10^{-4}$
Population Dose (person-Sv/year [person-rem/year])	0.15 [15]	0.006 – 0.69 [0.6 – 69]
\$/event	\$2.8 billion	\$49 million – \$12 billion
\$/person-Sv [\$/person-rem]	\$220,000 [\$2,200]	\$69,000 - \$670,000 [\$690 – \$6,700]
Total MAB	\$1.7 million	\$110K – \$8.7 million

Figure 1 . Typical Cost Benefit Threshold (3% Discount, 20 Year Term)



The SAMA identification and evaluation process has matured over the years, and typical analyses for nuclear power plant license renewal are now identifying multiple cost-beneficial SAMAs for most plants.

6. Potentially cost-beneficial SAMAs

Numerous potentially cost-beneficial SAMAs have been identified to date in U.S. operating nuclear power plant license renewal applications that have been approved. Most of these SAMAs are low-cost improvements such as modifications to plant procedures or training, minimal hardware changes to enable cross-tying existing pipes or electrical buses, and using portable equipment (e.g., generators and pumps) as backups. Below we provide examples of the specific potentially cost-beneficial SAMAs that have been identified for different operating U.S. plants.

SAMAs related to station blackout or loss of power sequences:

- Use portable generator or portable battery charger to extend coping time in loss of AC power events, or extend DC power availability
- Procure an additional portable 480VAC station diesel generator for backup to EDGs
- Install minimal hardware modifications and modify procedures to provide cross-tie capability between 4 kv AC emergency buses
- Modify plant procedures to allow use of a portable power supply for battery chargers, which would improve the availability of the DC power system
- Use the security diesel generator to extend the life of the 125 VDC batteries
- Modify plant procedures to use DC bus cross-ties to enhance the reliability of the DC power system
- Install key-locked control switches to enable AC bus cross-ties
- Develop procedures and operator training for cross-tying an opposite unit diesel generator

SAMAs related to internal floods, fire, seismic, and other external events:

- For internal floods, install watertight door or watertight wall around vulnerable equipment
- Install interlocks to open doors on high water level in order to divert flood water to a safe area, and change door swing direction to prevent opening a flood path to battery rooms
- Waterproof motor operators for vulnerable valves to mitigate floods caused by service water line breaks
- Enhance protection of critical fire targets by improving separation or providing cable tray protection
- Modify RHR valve yokes to reduce risk from seismically-induced ISLOCA
- Provide additional diesel fire pump for fire service water system (develop procedure for the use of a fire truck to pressurize and provide flow to the fire main)
- Increase fire pump house building integrity and combustion turbine building integrity to withstand higher winds, so that fire system and combustion turbines would be capable of withstanding a severe weather event

SAMAs related to protection systems:

- Change logic in under-voltage, block, and/or actuation signals, e.g., to 3 out of 4 logic
- Modify procedures to allow operators to defeat the low reactor pressure interlock circuitry that inhibits opening the LPCI or core spray injection valves following sensor or logic failures that prevent all low pressure injection valves from opening
- Install additional fuses in control panel to enable direct torus vent valve function during loss of containment heat removal accident sequences

SAMAs related to support systems:

- Various SAMAs to improve cooling of EDG rooms, e.g., revise operator procedure to provide additional space cooling to the EDG room via the use of portable equipment; modify plant procedures to open the doors of the EDG building upon receipt of a high temperature alarm; install diverse fan actuation logic for starting EDG room fans or operating exhaust dampers
- Provide an alternate/additional compressor that can be aligned to the instrument air supply header

SAMAs related to procedures and training:

- Increase operator training on the systems and operator actions determined to be important from the PRA
- Modify procedures and training to operate the isolation condensers with no support systems available
- Develop guidance/procedures for local, manual control of RCIC following loss of DC power
- Emphasize timely recirculation swap-over in operator training
- Develop emergency procedures for refilling the condensate storage tank using the fire service water system
- Use firewater systems as backup for containment spray
- Develop procedure for local manual operation of AFW when control power is lost

7. Conclusion

PRA has been used to identify cost-beneficial improvements at numerous operating U.S. nuclear power plants. Importance measures are used to identify risk-significant basic events from the PRA, and SAMAs are identified to address these basic events. SAMAs that are found to be potentially cost-beneficial tend to be low-cost improvements such as modifications to plant procedures or training, minimal hardware changes, and use of portable equipment.

Effect of SAMG on the Level 2 PSA of Korean Standard Nuclear Power Plant

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1. Introduction

Korea government pronounced a severe accident policy in 2001[1]. This policy requests the utility to perform Probabilistic Safety Assessment (PSA) and to develop severe accident management plan. KHNP which operates the nuclear power plants in Korea, submitted his implementation plan to comply with this policy. According to his plan, KHNP had performed PSA for all nuclear power plants and had developed severe accident management plan including development severe accident management guidance (SAMG) for all pressurized water reactors (PWRs) and is developing severe accident management plan for pressurized heavy water reactors (PHWRs) in Korea. Level 2 PSA had completed for Ulchin Nuclear Power Plant Units 3&4 (UCN 3&4) in 2003, which are Korean standard nuclear power plants. At that time SAMG for UCN 3&4 had not developed yet, so PSA result did not reflect SAMG.

SAMG for UCN 3&4 was developed in 2006. This SAMG includes strategies for the prevention of the failure of reactor vessel, the prevention of containment failure, and the reduction of fission product release[2]. How much each action affects to the frequency of the containment failure is evaluated in this paper.

2. Strategies for the prevention of the reactor vessel failure

Strategies for the prevention of the reactor vessel failure composed of three actions, i.e. the injection into the RCS, the injection into steam generators, and the injection into reactor cavity.
Level 2 PSA starts from the onset of the core damage. Most safety systems have already failed when the core damage occurred. It does not take much time from the onset of the core damage to the failure of the reactor vessel. MAAP predicts that it takes less than 3 hours from the core uncover to the vessel failure without a supply of feedwater and a safety injection. The possibility to restore the failed equipment and supply water to the steam generators or the reactor vessel within this short time is very low. So the benefits of these strategies are not evaluated quantitatively.

3. Strategies for the prevention of the early containment failure

The major mechanism which causes an early containment failure for a large dry containment is a direct containment heating (DCH). During high melt ejection process a hydrogen burn occurs simultaneously. The peak pressure depends on the amount of corium ejected out of the reactor cavity, temperature of the corium, and an amount of hydrogen accumulated in the containment. Depressurization of the RCS before the reactor vessel rupture is important to eliminate the DCH phenomena. UCN 3&4 has a safety depressurization system. This system operates manually. A containment of the UCN 3&4 is very robust. The median pressure in the fragility curve of the UCN 3&4 is 179 psig. The peak pressure at the reactor vessel failure for the high RCS pressure sequence is far below the median failure pressure. The early containment failure frequency is very low, i.e. 0.2 % of the total core damage frequency (CDF). The total CDF of is 5.3 x 10^{-6}/ry. So the actions in SAMG (depressurization of the RCS) are applied not to reduce the early containment failure frequency.

4. **Strategies for the prevention of the late containment failure**

UCN 3&4 have a relatively high late containment failure frequency. It is 5.38 x 10^{-7}/ry (10.1% of the total CDF). This high frequency comes from a long mission time which is 3 days from the accident happened, no credit to the recovery of failed equipment, and no credit to the non-safety graded equipment.

UCN 3&4 have four fan coolers which are non-safety grade equipments. We did not give any credit to the non-safety grade equipment when we perform a Level 2 PSA. But SAMG encourages the use of any available equipment, even though they are non-safety grade equipments. When the containment spray system failed, the fan coolers can be used to control the pressure and temperature within the containment. The fan coolers may experience a harsh environment during a severe accident progression. The failure frequency of the fan cooler is low during a normal operation, but may increase significantly under a harsh condition. We don’t know the failure frequency of fan cooler under a harsh condition. If fan coolers start to operate just after a spray system failure, the failure frequency of the fan coolers may be low because the temperature and pressure is not too high. The failure frequency may increase as the operation of fan coolers delays after a spray system failed. We assign a value of 0.5 to the failure frequency of the fan coolers because it is hard to quantify when an operator start fan coolers after a spray system failed and the containment conditions exactly. Consideration of fan cooler reduces the late containment failure probability to 2.992 x 10^{-7}/ry (5.6 % of total CDF)

SAMG has steps which orders operators to recover the failed equipment or system. At present PSA model, the recovery of failed equipment is not considered except a power recovery for a station blackout accident. In addition to the power recovery, we may give credit to recovery of the failed system if we have enough time to repair it. The spray system is the representative one. The time required to repair the spray system depends on what component in the spray system is failed. The repair time of failed valve is relatively short and the repair time of failed pump is long. The failed spray pump requires much time to repair it. The spray pump requires 47 hours to disassemble and assemble again completely. The maintenance staffs may restore the spray pump within this time even though they experience high stress. The available time to repair is key factor in the estimation of human reliability for the pump recovery. If the spray pump failed on demand, then we have sufficient time to fix it. If it failed while operating, we may not have enough time. If spray pump is restored before the containment failure, the containment failure due to overpressurization can be prevented. Considering these factors which affect spray pump recovery, we assign 0.9 to the spray system recovery. When considering the restoration of the spray system, the late containment failure frequency reduced to 1/10 of the base case.

The consideration of the fan coolers and the recovery of the failed spray system reduce the late containment failure frequency to the half of the original frequency. But the frequency of the basemat melt-through increased slightly.

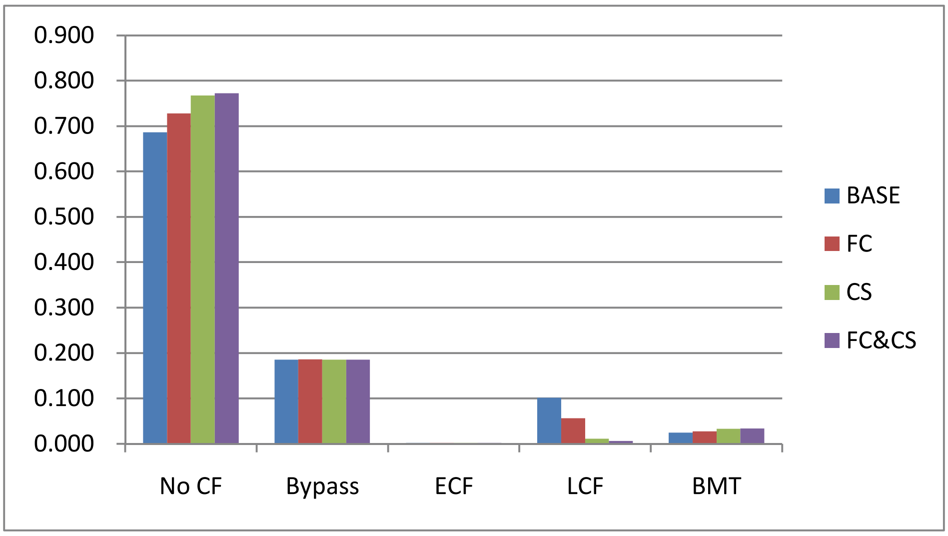
References

1. The severe accident policy statement, Ministry of science and technology, 2001
2. Severe Accident Management Guidance for Ulchin Unit 3&4, Korea Hydro and Nuclear Company, 2007

Table 1. The variation of the containment failure frequency when considering SAMG activities

Containment failure mode	Base Case	Fan Coolers	Spray Recovery	Fan Coolers & Spray Recovery
Intact	3.635E-06 (0.686)*	3.856E-06 (0.728)	4.067E-06 (0.768)	4.091E-06 (0.772)
Early containment failure	1.192E-08 (0.002)	1.192E-08 (0.002)	1.192E-08 (0.002)	1.192E-08 (0.002)
Late containment failure	5.376E-07 (0.101)	2.992E-07 (0.056)	5.938E-08 (0.011)	3.259E-08 (0.006)
Basemat Melt-through	1.286E-07 (0.024)	1.462E-07 (0.028)	1.750E-07 (0.033)	1.775E-07 (0.034)
Containment Bypass	9.841E-07 (0.186)	9.841E-07 (0.186)	9.841E-07 (0.186)	9.841E-07 (0.186)
Total frequency (/ry)	**5.297E-06**	**5.298E-06**	**5.297E-06**	**5.297E-06**

* Note : Values in the parenthesis is the contribution to the total frequency


0.900
0.800
0.700
0.600
0.500
0.400
0.300
0.200
0.100
0.000
No CF
Bypass
ECF
LCF
BMT
BASE
FC
CS
FC&CS

Insights From a Full-Scope Level 1/Level 2 All Operational Modes PSA With Respect to the Efficacy of Severe Accident Management Actions.

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1 Introduction

According to the Swiss national regulations, licensees of nuclear power plants have to perform a comprehensive Periodic Safety Review (PSR). The update of the plant specific PSA (Probabilistic Safety Assessment) as well as the update of the plant specific safety state analysis contribute a major part of this review effort. As a part of the PSR it is also required to review the effectiveness of the existing accident management procedures. At NPP Goesgen accident management procedures have been integrated into an Integrated Emergency Management system [1] that combines preventive and mitigative accident management procedures (SAMG) assuring a swift shift from preventive actions to mitigative actions without organisational delays.

NPP Goesgen completed its periodic safety review (the second during the lifetime of the plant) at the end of 2008. As a part of this effort, a major update of the plant-specific PSA was performed [2]. This update was aimed at addressing new regulatory requirements of maintaining an all operational modes, all events plant-specific PSA up to PSA level 2. Besides the general methodological and requantification effort taken, large attention was directed to incorporate the results of a new probabilistic seismic hazard study (the PEGASOS project) into the seismic part of the plant-specific PSA. The update of the PSA also included a re-evaluation of operator actions required to peform post-accident actions or accident management actions, including actions intended to mitigate the consequences of a core damage sequence.

The main insights gained with respect to an assessment of the efficiency of operator actions with respect to the mitigation of the consequences of severe accidents as well as of other administrative measures taken are presented below.

2 Scope and Structure of the Goesgen PSA model

The Goesgen PSA is an all modes, all events integrated level 1/level 2 PSA model. For quantification, the software RISKMAN® for Windows is used, which is based on the linked event tree approach. The quantification of system fault trees is fully based on the use of Binary Decision Diagrams (BDDs) allowing use of non-coharent logic in a mathematically correct way and the use of multi-state top events in the model event trees. Figure 1 shows the assignment of the different operational modes to the corresponding Goesgen PSA models.

Figure 2 shows the structure of the linked event tree PSA model for full power operation. The set of linked event trees can be subdivided into a set of pre-trees which set the boundary conditions for the subsequent quantification of support system event trees and frontline event trees for external or large scale internal hazards (e.g. fires). The model presented in Figure 2 is used for full power operation as well as for low power operational modes (below 40%).

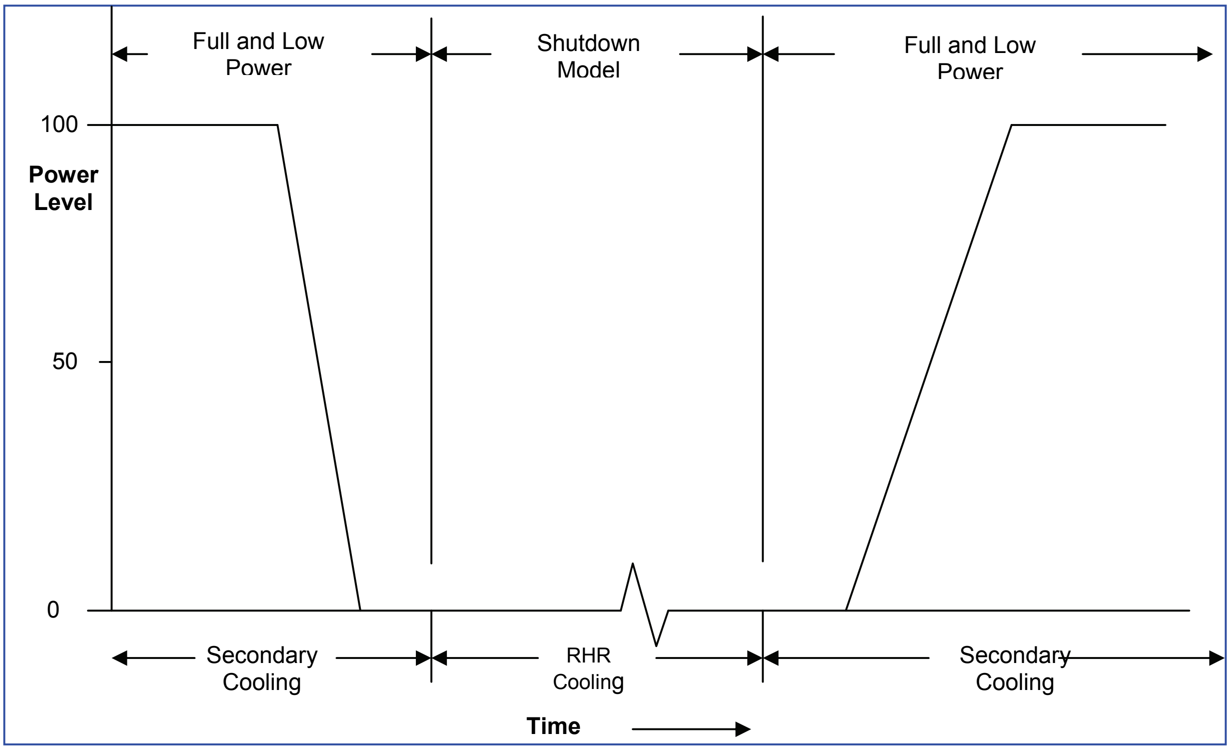

Figure 1. Assignment of Operational Modes to Goesgen PSA Models

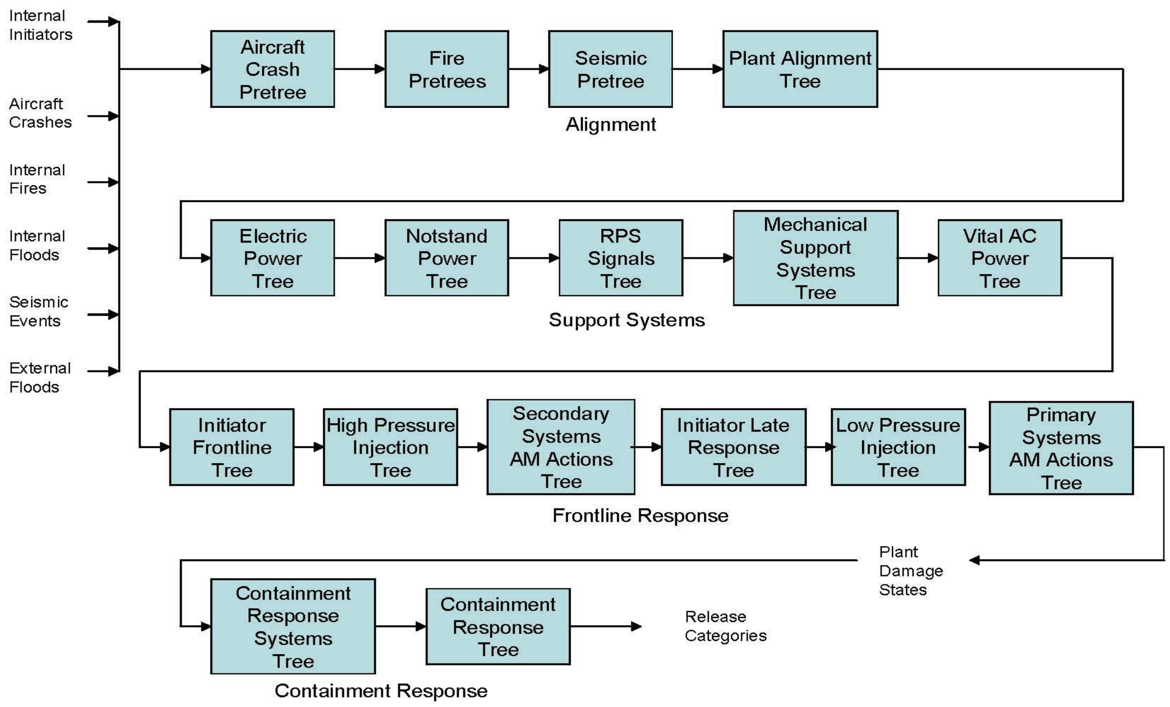

Figure 2. Structure of the Goesgen PSA Model for Power Operation

A total of 156 initiating events are quantified by the power model including directly the following event groups:

- Internal events subdivided into
  - LOCAs
  - Transients (including reactivity transients and precurser events (e.g. partial loss of main feedwater, partial loss of primary flow))
  - Steam Generator Tube Ruptures (SGTR – 4 initiators ranging from leaks to multiple ruptures)
  - ATWS (including failure of scram for all transients, small LOCAs and SGTRs)
- Internal Hazards
  - Internal Floods
  - Fires (more than 300 scenarios including small, room and propagating fires, as well as component fires)
- External Hazards
  - Airplane crash (different classes of impactors)
  - Earthquakes
  - External floods

- Loss of service water intakes

The following external hazards are considered as a part of forced shutdown initiators (7 initiators with different availability of safety trains) or of a manual reactor scram scenario with all safety systems available:

- Wind and Tornado (below screening threshold of 10^{-10}/a)
- Forest fire (forced shutdown or manual scram)
- Hail (forced shutdown)
- Extreme snow loads (slow transient, forced shutdown)
- Climate change (extreme river temperature conditions; forced shutdown)
- Transportation and industry accidents (e.g. gas pipeline, forced shutdown or manual scram)
- Turbine missiles (below screening threshold)

The Goesgen PSA considers also complex scenarios where an external initiating event is combined with possible dependent events, e.g. earthquake induced LOCAs, ATWS, internal Floods, external fires (transformer fires) and dam breaks (external floods), tornadoes combined with missile impacts (considered in the screening process), high river conditions combined with water intake plugging.

The model for low power operation includes initiating events which are specific for these conditions or have different response under specific boundary conditions. This is the case for steam generator tube ruptures because below a power level of 40% the N-16 reactor scram signal and the following automatic actions (depressurization via pressurizer cold spray) are not available.

The model for shutdown conditions considers three different types of shutdown activities:

- Plant shutdown for repair without opening of the reactor primary circuit – Type A
- Plant shutdown for repair with opening of the reactor primary circuit – Type B
- Plant shutdown for refuelling (normal annual outage) – Type C

For the shutdown operational modes the same groups of initiating events are considered as for the power model. The total number of initiating events for shutdown operational modes is 173.

The Goesgen PSA models represent integrated level 1/level 2 PSA models. Therefore the results for CDF and the different plant damage states as well as the frequencies of the different release categories and for the LERF (Large Early Release Frequency), LLRF (Large Late Release Frequency), VENTF (Frequency of a release via the containment venting system), SRF (Small Release Frequency) release groups are quantified in a single run of the model.

3 Main results of the Goesgen PSA

The following tables and figures present an overview of the main results of the Goesgen PSA for power and shutdown operation. On a first glance the results for CDF for the Goesgen plant appear to be rather low for people not familiar with the Goesgen plant’s design and safety upgrade programmes. For interpreting the results, it is important to note that for most initiating events (transients and small LOCAs) NPP Goesgen has a 6 times 100% design including two service water intakes and two well water systems each with 100% capacity (for transients). Simulations performed with the Goesgen PSA model with the aim to approximate the degree of redundancies of a three train Westinghouse plant (3 x 100%, 1 seismically designed service water intake, 2 loops) or a typical Convoi (KWU) plant (4 x 100% design) result in similar results for core damage frequency as are published for such designs (3.1 E-5/a for the Westinghouse plant, 4.3 E-6/a for a Convoi type [3]). Therefore, the Goesgen PSA model is not optimistic. It simply reflects the increased degree of functional redundancy as is characteristic for the Goesgen plant in comparison to other designs.

Table 1. Key results of full power PSA

Initiating Event Group (Number of Initiators)	CDF Contribution [1 /a]	CDF Contribution, [%]
LOCAs (10)	2.61E-7	40.4
Transients (40)	8.07E-9	1.2
SGTRs (6)	2.87E-9	0.4
Internal Events (Total)	2.77E-7	42.9
Aircraft crashes (7)	1.13E-8	1.8
External Floods (1)	1.42E-8	1.8
Fires (23, more than 300 scenarios)	2.37E-8	0.4
Cooling Water Intake Plugging (2)	2.66E-9	0.4
Internal Floods (20)	1.34E-9	0.2
Seismic Events (41)	3.37E-7	52.1
External Events (Total)	3.65E-7	56.5
Other	3.87E-9	0.5
Total CDF	**6.46E-7**	**100%**

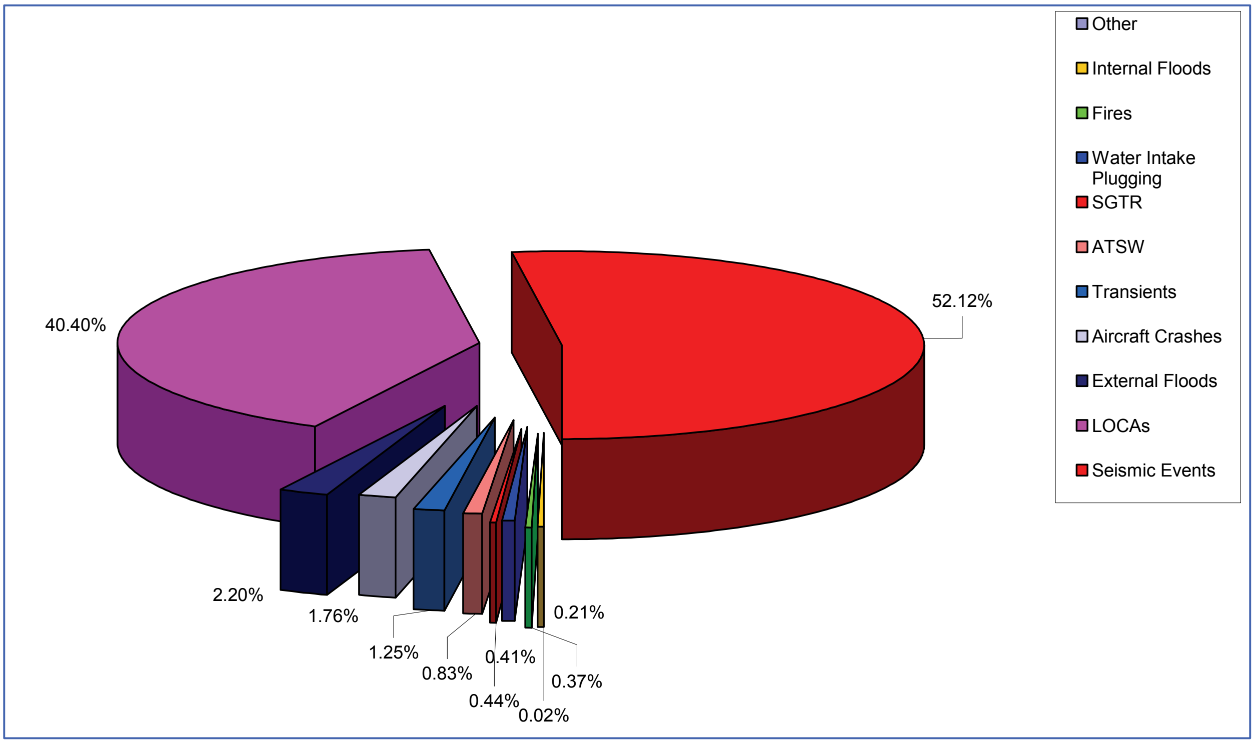

Figure 3. Contribution of Major Initiating Event Groups to Core Damage Frequency

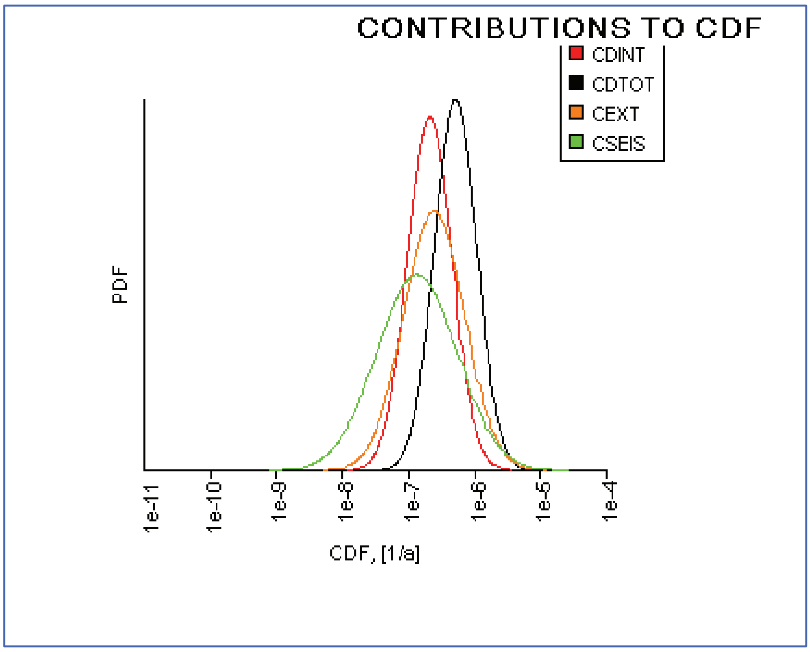

Figure 4 . Uncertainty Analysis – Main Contributions to CDF

Table 2 presents the results of the level 2 PSA for power operation, for major release categories. The table illustrates the benefit of the venting system which assures that a large amount of the potential large late releases will be vented by the filtered venting system.

Table 2 Frequency of major release categories for power operation

Denotation	Release Category	Release Frequency, [1/a]
VENTF	Release via filtered venting	1.70E-07
LERF	Large early release (within 10 hours after core damage)	5.08E-08
LLR	Large late (offsite) release	5.01E-08
SMREL	Small and moderate releases	2.92E-07

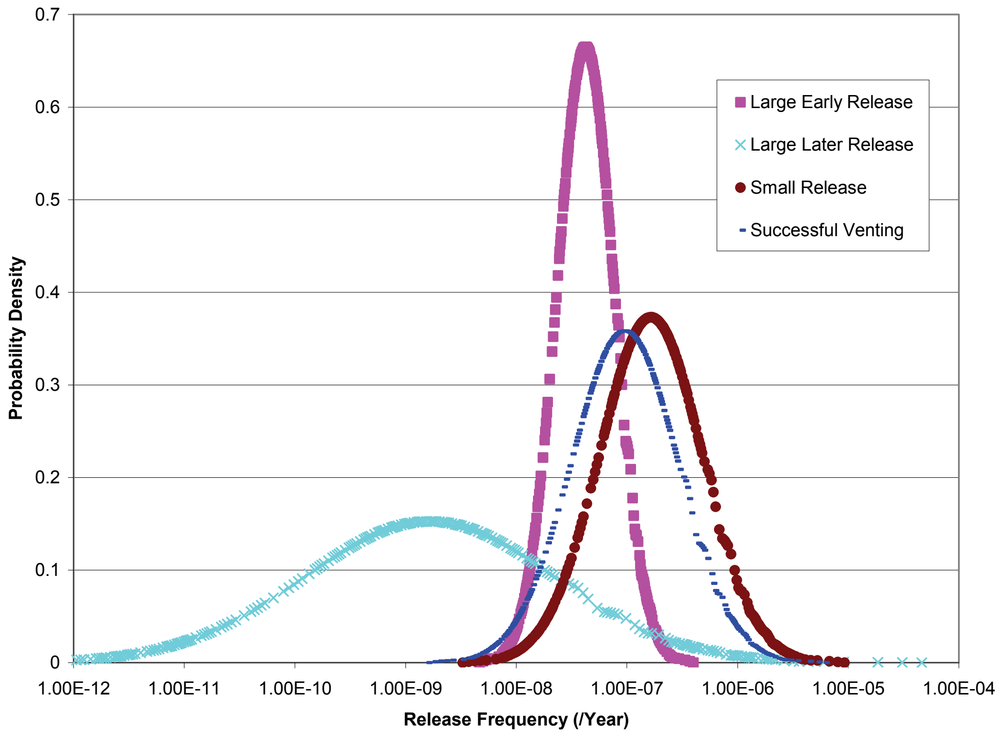

Figure 5. Results of the Uncertainty Analysis for Major Release Categories for Power Operation PSA

Figure 5 presents the results of the uncertainty analysis performed for the major release categories. Figure 6 shows the contribution of major groups of initiating events to the frequency of large early releases for power operation only. As it can be seen, the dominating contribution results from earthquakes, the second largest contribution results from high-speed aircraft crash. The contributions from earthquakes are caused by seismic failures of ventilation ducts penetrating the reactor building, leading to a containment bypass for some of the core damage sequences caused by earthquakes. This observation leads to an important conclusion. The importance of physical phenomena like steam explosions, hydrogen burns, direct containment heating or the “rocket failure mode” is rather low with respect to the calculated large early release frequency for NPP Goesgen, because the results are dominated by accident sequences leading directly to an unisolated containment.

Initiating Event Group	FDF per year	% of External Events Group	% of Total FDF
External Events & Internal Hazards	1.96E-06		81.8%
Fires	1.31E-06	66.9%	54.8%
Seismic Events	5.47E-07	27.9%	22.8%
External Floods	5.34E-08	2.7%	2.2%
Internal Floods	2.86E-08	1.5%	1.2%
Cooling Water Intake Plugging	1.88E-08	1.0%	0.8%
Aircraft Crashes	3.85E-10	< 0.1%	< 0.1%
Internal Initiating Event Group	**FDF per year**	**% of Internal Event Group**	**% of Total FDF**
Internal Events	4.36E-07		18.2%
LOCAs	2.67E-07	61.1%	11.1%
Transients	1.70E-07	38.9%	7.1%
All Initiating Events	2.40E-06		

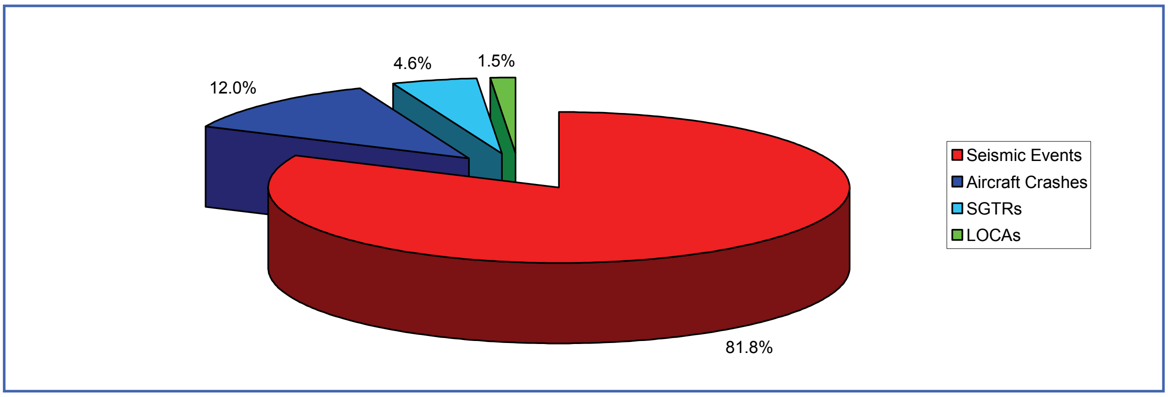

Figure 6. Major Contributions to Large Early Release Frequency for Power Operation

Table 3. Major Contributors to Fuel Damage Frequency, All Shutdown Modes

An analysis of the results shows that the contribution of earthquakes to fuel damage frequency is significantly smaller than to the CDF for power operation. The reason for this is the larger total damage frequency (due to the lack of automatic reactor protection signals) and the larger contribution of fires to risk. The increased fire risk is mainly driven by the contribution of fires in the large annulus compartment ZB0104 where the RHR pumps are located. Although the pumps are spatially separated, a propagating fire may fail RHR functions either directly or fuel damage occurs because of a concurrent independent failure of RHR-equipment.

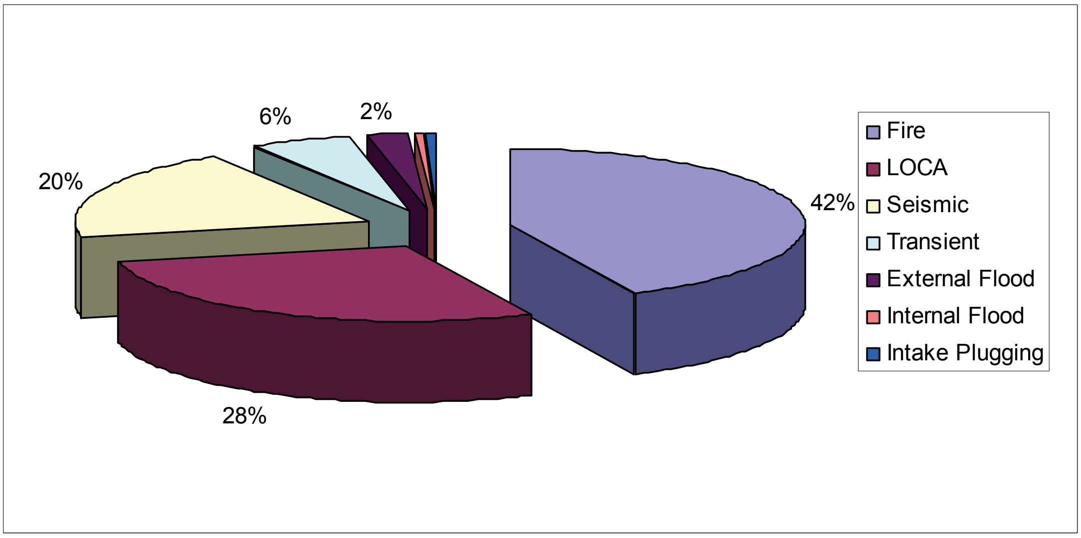

Figure 7. Major Contributions to LERF, All Shutdown Modes

Similarly an analysis of the contributions of major initiating event groups to LERF shows that fires drive the risk, while LOCAs and seismic events represent the next important contributors.

Figure 8 shows the frequencies of major release categories for shutdown conditions with the associated uncertainties.

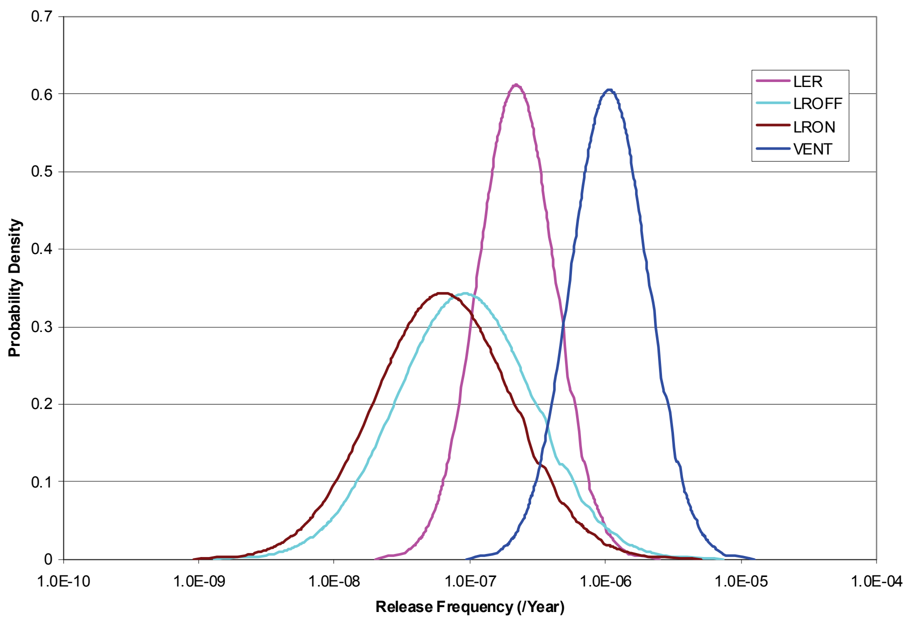

Figure 8. Frequencies Of Major Release Categories For Shutdown Operational Modes

The analysis of the results shows again the importance of the containment venting system. In compliance with Swiss regulations, LERF is defined as a large release within 10 hours after fuel damage (for fuel damage in pool configurations this includes time windows up to 58 hours after occurrence of the initiating event since fuel damage is delayed) it is understandable that closing the containment (the equipment hatch) is the most important mitigating acccident management action. The success of this action allows relying on the venting system to assure a filtered release thereby avoiding unfiltered large early releases.

4 Insights gained from an analysis of the PSA results

After completion of the update of the PSA, a detailed analysis of the benefit and limitations of severe accident management actions was performed. For this purpose a LERF-based importance analysis for operator actions was performed. Additionally some special analysis of source terms for the most critical plant configuration during shutdown " the ¾ loop operation" (sometimes called midloop-operation) was performed to check the impact of general plant alignments (pre-accident actions) on available time windows and resulting source terms. The two parts of this investigation are presented below.

4.1 Insights gained from source term analysis for shutdown conditions- ¾ loop operation

As a support for the development of the level 2 PSA for shutdown conditions a MELSIM_KKG (MELCOR based plant specific accident simulator for severe accidents) model for shutdown operational modes was developed [4]. The following model configurations (standard input decks) are available:

- "Early" ¾ loop ("midloop") operation with reactor vessel head in place (high residual heat case, 23.5 hours after scram)
- "Early" ¾ loop ("midloop") operation with reactor vessel head removed (high residual heat case, 23.5 hours after scram)
- "Late" ¾ loop ("midloop") operation with reactor vessel head removed (low residual heat case, 467 hours after scram)
- Fuel unloaded (pool configuration) with a typical mixture of freshly unloaded and older spent fuel in the spent fuel pool (located inside containment)

The models were used to study the influence of availability of passive secondary side cooling (availability of steam generators) on the available time windows for operator actions as well as on the timing of radioactive releases. As the reference scenario, a complete station blackout during ¾ loop operation with vessel head in place was selected. Additionally the source term related with a large release after fuel damage in a spent fuel pool configuration was determined. This analysis was performed under the assumption of an open equipment (material) hatch. Table 4 shows a comparison of the accident progression for different pre-accident plant configurations. Case 1 represents the case when all 3 steam generators are filled and depressurized (open). Case 2 represents the case when 2 steam generators are filled and depressurized, while the 3rd steam generator is assumed to be drained and to be isolated. This corresponds to the minimal requirement on the availability of steam generators according to the Goesgen plant operational procedures. These two cases are compared with the case when passive heat removal via the secondary side is not available. In all cases the pressurizer is assumed to be maintained in a hot standby condition. As it can be learned from the results in Table 4, the availability of filled steam generators during plant outage states with reduced inventory in the reactor circuit is very beneficial and provides significantly increased time windows for operator actions; for example closing the equipment hatch.

Table 4. Anaylsis of the Effect of Pre-Accident Boundary Conditions on Severe Accident Progression, Station Blackout, Early ¾ Loop Operation

Event	Time (Case 1: 3 S/Gs Filled)	Time (Case 2: 2 S/Gs Filled)	Time (Case 3: No S/Gs Filled)	Comment
Swollen level at Top of active fuel (TAF)	1081s (0.3 h)	1463s (0.4 h)	1366s (0.4 h)	Local boiling starts earlier
Swollen level oscillates between 75% and 100% TAF	1886s (0.5 h)	1642s (0.5 h)	1571s (0.4 h)	Heat removal via the secondary side of the filled SGs, reflux-condenser mode of heat transfer
Evaporation of water in SGs completed (<5m), start of pressure increase	27000-27300s (7.5 to 7.6h)	22356-22573s (6.2 to 6.3h)	0	Level below 5 m, some heat transfer is possible until water level drops below 1-1.5 m
Setpoint of safety Valve THxxS090 achieved	44600s (12.4h)	33615s (9.3h)	8080s (2.2h)	Induced LOCA in the containment (40 cm^2)
Start of Gap release	49473s (13.7h)	38457s (10.7h)	13424s (3.7h)	Onset of core damage, core damage state A according to SAMG
First clad melting	50860s (14.1h)	39638s (11.0h)	14724s (4.1h)	Core damage state B according to SAMG
Reactor vessel rupture	66880s (18.6h)	51891s (14.4h)	32795s (9.1h)	Core damage state C according to SAMG

Figure 9 illustrates the accident progression for case 1. During the course of the accident it is assumed that the safety valve located on the RHR-line (THxxS090) will not reseat after opening causing an induced LOCA condition (worst case scenario).

For the “minimal availability configuration” according to the plant operational manual (2 filled SGs) at least 8 hours are available to restore cooling and/or to depressurize the reactor circuit by accident management actions. A possible accident management strategy is to cool the steam generators actively; for example by fire extinguishing water. Such an accident management action is prepared at NPP Goesgen. For this purpose a special injection nozzle allowing for a simple connection of a fire extinguishing water hose is available in the fourth train of the emergency feedwater system. This train can be connected to all three steam generators. Alternatively, accumulators can be activated by jogging open the motor-operated check valves using spurts of injection of coolant into the reactor circuit. Table 5 shows the activity release into the environment for the three different cases under the assumption of an open equipment hatch (worst c se scenario without accident management actions).

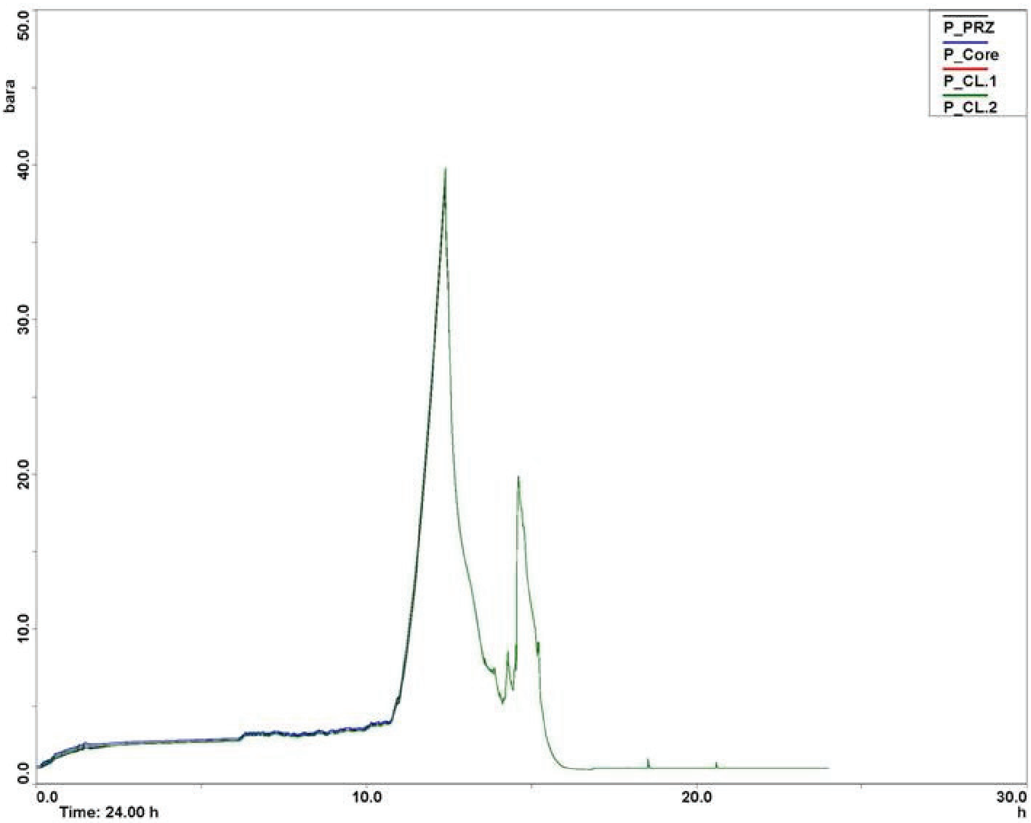

Figure 9. Pressure [bar] in the Primary Circuit for Case 1 (3 SGs Available), Time in Hours

Table 5 Source terms (LERF) for ¾ loop operation, vessel head in place (Bq), SBO with induced LOCA, mission time 48 hours

CASE	TOTAL	NOBLE GAS (NG)	CS	CSI	BA	TE	I	RU	MO	CE	LA	CD	SN
2 SGs (mean)	6.54E18	5.55E18	3.56E16	3.19E17	7.67E16	6.67E16	1.79E14	2.84E15	3.90E17	7.49E16	9.75E14	8.68E15	1.57E16
3 SGs (lower limit)	5.98E18	5.17E18	3.16E16	2.91E17	8.44E16	8.57E16	1.393E14	3.57E15	7.14E16	2.22E07	6.56E15	5.60E15	8.35E15
0 SGs (upper limit)	6.68E18	5.70E18	2.61E16	2.79E17	7.51E16	1.03E17	1.91E14	2.53E15	3.26E17	1.39E17	3.89E15	6.32E15	1.14E16

A similar station blackout scenario was analyzed for the ¾ loop operation with vessel head removed. Here the availability of steam generators is completely unimportant. A comparative analysis was made to study the effect of reduced residual heat generation during the late stage of plant refuelling outage on the available time windows for operator actions. The results of this comparison are presented in Table 6 The analysis shows the significantly increased time windows in case of the "late" configuration (467 hours after reactor scram).

The analysis of source terms demonstrated that despite the activity decay, the release in case of an open equipment hatch can be very large. Even for noble gases it can exceed 10^{18} Bq. Such large releases have to be considered even in case of pool configuration, when the fuel is completely unloaded. Assuming again a station blackout, it takes about 50.1 hours to cause fuel damage (gap release) due to the large coolant inventory in the spent fuel pool of NPP Goesgen. Nevertheless the associated release can be very high. At 50.9 hours the release (in case of an open equipment hatch) will exceed the limits defined as a large release in terms of Cs release ($2.0 * 10^{14}$ Bq). The exceedance of the large release within one hour after gap release is related to Zr-air oxidation which is modeled by an accelerated oxidation rate in the MELSIM_KKG simulation. Table 7 gives an overview of accident progression for this case.

Table 6. Accident Progression – Station Blackout in Configurations C6 and C11 (Open RCS, "Early" Vs. "Late" Configuration)

Event	Time, C6	Time, C11	Comment
Swollen level at top of active fuel (TAF)	435.7s (0.1h)	10088s (2.8h)	Oscillation of swollen level in the vessel assures sufficient heat removal
Swollen level at 75% of TAF	1305s (0.4h)	28601s (7.9h)	Start of core heat up
First cladding damage, start of gap release	11201s (3.1h)	55677s (15.5h)	Core damage state A according to SAMG procedures
First Clad melting	12669s (3.5h)	60316s-61043s (16.8 to 17.0h) for the three different channels modelled)	Core damage state B according to SAMG procedures
Vessel Rupture	27943s (7.8h)	108927s (30.3h)	Core damage state C according to SAMG procedures

Table 7. Accident Progression - Station Blackout During Pool Configuration (Fuel Completely Unloaded)

Event	Time	Comment
Swollen level drops to top of active fuel	159400s (44.3h)	Fuel cooled by oscillating water/steam mixture untill this point in time, start of fuel heat-up and oxidation
Start of gap release	180265 (50.1h)	"Fuel damage state B" "approximate" definition acccording SAMG
Exceedance of large release threshold (2.0E14 Cs)	183320s (50.9h)	Unfiltered release of this amount of Cs may lead to a radiation dose of 100 mSv
Failure of fuel racks, start of core (melt) -concrete interaction	203080s (56.4h)	Molten fuel starts to progress towards the sump area (possible bypass scenario)

The accident and source term analysis performed underlines the importance of well prepared accident management actions. For the most critical plant configuration of a western PWR during an outage – the "midloop" operation (for Goesgen a ¾ loop operation) the preaccident boundary conditions were found to be even more important. The possibility of "passive" secondary side cooling via depressurized and open to atmosphere steam generators was found to be crucial to assure sufficiently large time windows for the implementation of accident management actions. As an important administrative action NPP Goesgen does not allow scheduled maintenance work in the nuclear part or in the elctrical power supply systems during the period of "3/4 loop" operation. This reduces the risk of inadvertent drain-down events or loss of power supply scenarios which are frequently caused by maintenance work.

Source term analysis performed for "small releases" in the format of a sensitivity analysis indicated that small releases during shutdown can be significantly higher than for power operation conditions. The main reason for this is the larger size of "left" leakages due to difficulty to close all open penetrations during a plant accident during shutdown.

4.2 Analysis of important accident management actions with respect to LERF

The importance of post-accident actions was analyzed with the help of a PSA importance analysis for LERF. It is an advantage of the integrated level 1/level 2 PSA of Goesgen that such an analysis can be performed directly. In this paper we present the results from the full power PSA. A similar analysis was performed with respect to the shutdown PSA. The analysis for full power operation (based on RAW) was very revealing. The most important operator actions are found to be related to the pre-core-damage domain (post-accident operator actions and preventive accident management actions), while the mitigating accident management actions were established to be relatively unimportant. This does not come as a surprise looking at the dominant contribution of external events to LERF (about 96% by earthquakes and airplane crash combined).

Therefore, it has to be concluded that the implementation of SAMG measures which was recommended and requested by Swiss regulations on deterministic grounds appeared to be of limited benefit if evaluated on a risk-informed basis. The venting system can be regarded as the only exception, as long as the increased noble gas releases associated with the venting process are regarded as less important then the achievement of the goal to avoid long term land contamination, e.g by Cs releases. Swiss regulations give the avoidance of long term land contamination higher priority than the avoidance of early noble gas releases.

4.3 Summary insights from the analysis of PSA results

Summarizing the insights drawn from a comparison of the results for full power operation and shutdown operational modes it can be concluded that the risk during shutdown operations is notably larger than the risk during full power operation. This can be partially explained by the increased importance of fires in the reactor annulus building during RHR-cooling modes. The main reason has to be seen in the lack of automatic reactor protection signals which significantly increases the importance of post-accident operator actions during shutdown operational modes. It was found that by the help of simple administrative measures like the removal of scheduled maintenance in the nuclear part of the plant from configurations with reduced reactor circuit inventory to configurations with increased time windows for operator response the risk can be reduced significantly. The requirement in the Goesgen operational procedures to maintain at least 2 steam generators available for passive secondary side cooling during "3/4 loop" operation with closed reactor circuit was found to be an effective measure to increase available time windows for post-accident operator actions.

In general it was established that mitigative accident management actions have a lower importance for the reduction of the frequency of large early releases than pre-core-damage accident management actions. This is a result of the large contribution of external events to the frequency of large early releases.

Nevertheless, analysing the risk profile of the Goesgen NPP, it was found that further overall risk reduction can be achieved by removing scheduled maintenance activities from the plant shutdown to full power operation as far as such a shift is feasible.

5 Conclusions

In the framework of the 2008 periodic safety review, NPP Goesgen performed a full upgrade of its probabilistic safety assessment (PSA) extending it to the scope of an all operational modes, all hazards and events, integrated level 1/level 2 study. The upgrade of the study included a detailed evaluation of source terms for power and shutdown operational modes. In 2005 NPP Goesgen implemented the concept of integrated emergency management applying a holistic approach combining preventive and mitigative accident management guidance into an integrated accident management handbook. The updated PSA study provided the capability to assess the efficacy of severe accident management actions and of the hardware changes associated with the implementation of SAM directly by evaluating the corresponding risk importance measures.

The risk profile of the plant is in compliance with studies performed for new generation 3 nuclear power plants.

The results of the risk study show that the importance of physical phenomena like steam explosions, hydrogen burns, direct containment heating or the "rocket failure mode" is rather low with respect to the calculated large early release frequency for NPP Goesgen. This is caused by the high risk contribution of extreme low frequency external events such as earthquakes and high speed airplane crash scenarios causing the failure of reactor building or ventilation lines . Similarly it was found that the importance of mitigative severe management actions is rather low in comparison to the importance of pre-core damage actions (preventive actions). The source term analysis performed for shutdown operational modes indicates that the source terms for small releases of radioactivity are higher than for similar scenarios at power operation due to the higher containment leak rate due to "left leakages" (unisolated penetrations or pipings leading to atmosphere instead to a closed circuit due to maintenance work) during shutdown.

In general it was found that the risk of a large early release is significantly higher for a refuelling outage than for full power operation. The Goesgen practice of performing a substantial amount of planned maintenance during power operation thereby limiting the amount of planned maintenance during refueling has to be regarded as an effective and efficient method of overall risk reduction.

Also, administrative measures taken by NPP Goesgen to avoid scheduled maintenance during shutdown operational modes with reduced inventory of the reactor coolant circuit have proven to be a successful measure for reducing the risk during plant shutdown.

The Goesgen integrated level 1/level 2 PSA has proved to be an effective tool for analyzing the benefit of severe accident management actions as well as of hardware upgrades.

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PRA Level 2 Perspectives on the SAM during Shutdown States at the Loviisa NPP

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Abstract

The Loviisa NPP went through an extensive severe accident management (SAM) programme during 1990's and early 2000's. Programme included a wide range of experimental and analytical studies and the selected SAM approach led to significant plant modifications. The SAM approach was developed and the SAM systems were designed to cope with the accidents starting from power operation states. More recently the severe accidents initiating during shutdown have been studied for Loviisa case and the applicability of the SAM strategy for shutdown states has been evaluated. The basic requirements for severe accident management during shutdown are the same as for power operating states. However, during shutdown states initial conditions, such as decay power and primary pressure, differs significantly from conditions during power operation. Also conditions in the containment are different when maintenance work is going on. Recovery actions in order to resume containment tightness and the operability of the SAM systems will be needed in accidents starting during shutdown. The overall efficiency of the SAM strategy has been evaluated with PRA level 2 for power operating states. For shutdown states the work involved with assessment of the SAM strategy applicability and development of level 2 PRA have been closely connected. Based on the work, new guidance and many procedures facilitating the SAM during shutdown will be implemented in the Loviisa NPP. The PRA level 2 shows the development towards efficient SAM also in shutdown states. However, work still has to be carried on.

1 Introduction

The aim of SAM strategy development was to ensure containment integrity during severe accidents initiating from power operation states. The basic idea of SAM strategy was in ensuring the mitigation of such accident sequences, for which the mitigative actions could be considered reliable, and in screening out the sequences, for which the mitigation was considered inefficient. Particularly, the screening was applied to containment bypass sequences, high-pressure core melt scenarios and reactivity initiated sequences such as boron dilution accidents. The top level critical safety functions of the Loviisa NPP SAM approach relate to ensuring subcriticality of the core, containment isolation, primary system depressurisation, absence of energetic events, coolability of molten core and long-tern containment pressure control.

The accidents initiating from shutdown states involve much lower primary pressure and decay heat, and they often have very long delays until the heat-up of the reactor core. The shutdown states are in many ways different from power operation states: the containment may have no leaktightness requirements, the primary circuit may be open, the emergency systems may not be available due to maintenance, and some openings may allow coolant to escape from the containment, etc. From such situations the return to a state, where containment integrity can be ensured during severe accidents, may be very lengthy and require many recovery operations.

During shutdown states, the SAM functions do not differ from the power operation states, but their fulfilment requires much more actions. In-vessel retention of core melt and hydrogen mitigation are the cornerstones of the Loviisa SAM strategy, and ensuring their reliability is essential for shutdown states, as well. For successful in-vessel retention adequate amount of water is needed to flood the reactor cavity, the flow paths around the reactor pressure vessel have to be available, and the escape of water from lower parts of the containment has to be prevented. During shutdown, the hydrogen recombiners are protected against possible poisoning, and thus the protections need to be removed well before the core heat-up. Furthermore, for efficient hydrogen mixing in the containment, the flow paths through the ice condensers have to be recovered. Some of the recovery actions have to be started rather soon after the initiating event, as at later stages the containment condition may prevent carrying out the operations.

This paper presents briefly the SAM strategy implemented for the Loviisa NPP. Results arising from the Loviisa NPP level 2 PRA for the containment sequences initiating from power operation states are presented. These results confirm the adequacy of the SAM implementation for power operating states. Work done in order to extend the applicability of the SAM strategy for shutdown states is explained and some results arising from PRA level 2 for containment sequences are presented. There are a number of guidelines and procedures that are being developed to improve the SAM during shutdown states at the Loviisa NPP. These are briefly described and the effect of these procedures on level 2 PRA results is studied in order to study their adequacy in supporting the SAM measures. Some sights on continuation of the work are also described.

2 SAM strategy in the Loviisa NPP

2.1 Unique features of the Loviisa NPP

The Loviisa NPP is a two-unit VVER-440 plant with ice condenser containment. Units 1 and 2 were commissioned in 1977 and 1981 respectively. The original plant concept didn't include the containment and since it was definitely required in Finland, ice condenser containments were built. Also other significant modifications to the original plant design were carried out, most notably modifications of the ECCS, the reactor coolant pumps, and the inclusion of Siemens I&C systems.

In the containment two ice condenser (IC) sections connect the lower (LC) and upper compartment (UC). In the UC, a dome part and the reactor hall can be separated. The containment is surrounded by the outer annulus (OA). The total free volume of the containment (excluding the dead-ended compartment) is 58 000 m^3. The containment and the global convective loop flow inside the containment in power operating states are shown in Figure 1. The absolute design pressure of the containment is 1.7 bar, which is rather low considering the loads that could take place during severe accidents.

Studies considering severe accident management for the Loviisa NPP have been structured around the identified containment-threatening mechanisms. The aim has been to find solutions that would reliably protect the containment. It has to be recognised that even though the Loviisa NPP has certain vulnerabilities to severe accident phenomena, it also presents some unique opportunities for selection of mitigation strategies. For example, water from melting the ice would quickly (and passively) flood the small-sized cavity in an accident. This feature, in combination with the fact that the decay power level is low and the reactor pressure vessel lower head has no penetrations, makes in-vessel retention of molten corium feasible through external cooling of the RPV. A well-known vulnerability is that the ice condenser containment has a rather low estimated failure pressure in relation to loads that could take place during severe accident (e.g. from global hydrogen deflagrations). On the other hand the ice condenser configuration would ensure efficient mixing of the containment atmosphere, in case the ice condenser doors would be forced open. The containment steel shell makes it possible to control long-term pressurisation through external cooling. All of these elements (and many more) are now part of the overall SAM approach for ensuring the selected SAM safety functions (Chapter 2.2).

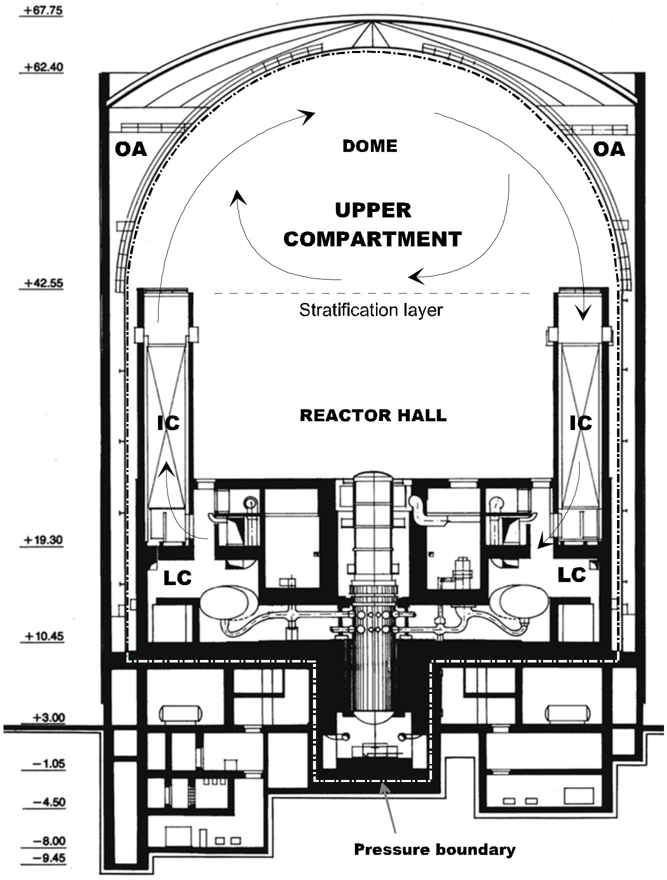

Figure 1. The Loviisa NPP containment

2.2 Overall SAM approach

Implementation of the SAM approach at the Loviisa NPP includes several different lines of action. The most notable tasks are the following:

- Hardware modifications have been carried out at the plant in order to ensure that core damage can be reliably prevented and severe accident phenomena can be mitigated.
- Substantial new I&C (instrumentation and control) qualified for severe accident conditions has been installed.
- New SAM guidelines and procedures, as well as a SAM Handbook have been written.

– The emergency preparedness organisation has been revised.

– Versatile training approaches, including the development of a severe accident simulator, APROS SA[1] are being developed.

The Integrated ROAAM (Risk oriented Accident Analysis Methodology) approach was applied for the development of an overall SAM strategy for Loviisa.[2 3] The strategy consists of four steps:

– The reliability of prevention of core damage should be demonstrated by a PRA level 1 to meet our prevention requirements.

– Prevention of core melt sequences with imminent threat of a large release (usually sequences with an impaired containment function) that cannot be mitigated has to be demonstrated to be sufficiently reliable according to PRA.

– Reliable mitigation of severe accident phenomena that could pose a threat to containment integrity should be demonstrated for all relevant accident scenarios

– In order to show a compliance with Finnish safety requirements with regards to SAM, we have to demonstrate that radioactive release limits[4] are not being exceeded due to normal leakages out of an intact containment in a severe accident.

The idea of Integrated ROAAM is shown in Figure 2. Integrated ROAAM ensures the sound balance between preventive and mitigative parts of the SAM strategy. The most important issues threatening the containment integrity are handled with ROAAM in issue resolution context. The use of Integrated ROAAM and the use of ROAAM in issue resolution context for the Loviisa NPP case are thoroughly explained in.[5]

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[1] Raiko, E., Salminen, K., Lundström, P., Harti, M. and Routamo T. Severe Accident Training Simulator APROS SA. The 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Seoul, Korea, October 5-9. 2003.

[2] Theofanous, T.G. On the Proper Formulation of Safety Goals and Assessment of Safety Margins for Rare and High-Consequence Hazards. Reliability Engineering and System Safety 54 (1996), pp. 243-257.

[3] Lundström, P., Tuomisto, H. and Theofanous T.G. "Integration of severe accident assessment and management to fulfill the safety goals for the Loviisa NPP", International Topical Meeting on Probabilistic Safety Assessment PSA96, Park City, Utah, September 29 - October 3, 1996, USA.

[4] In Finland the degree of the Council of the State for the safety of nuclear power plants (733/2008) says that the limit for the release of radioactive materials arising from a severe accident is a release which causes neither acute harmful health effects to the population in the vicinity of the nuclear power plant nor any long-term restrictions on the use of extensive areas of land and water. For satisfying this requirement applied to long-term effects, the limit for an atmospheric cesium-137 release is 100 TBq. The possibility that, as the result of a severe accident, the above mentioned requirement is not met shall be extremely small.

[5] Siltanen, S., Routamo, T., Tuomisto, H. and Lundström, P. Severe Accident Management at the Loviisa NPP - Application of Integrated ROAAM and PSA Level 2. Proceedings of the Workshop on evaluation of Uncertainties in Relation to Severe Accidents & Level 2 Probabilistic Safety Analysis. Aix-en-Provence, France, 7-9- November 2005.

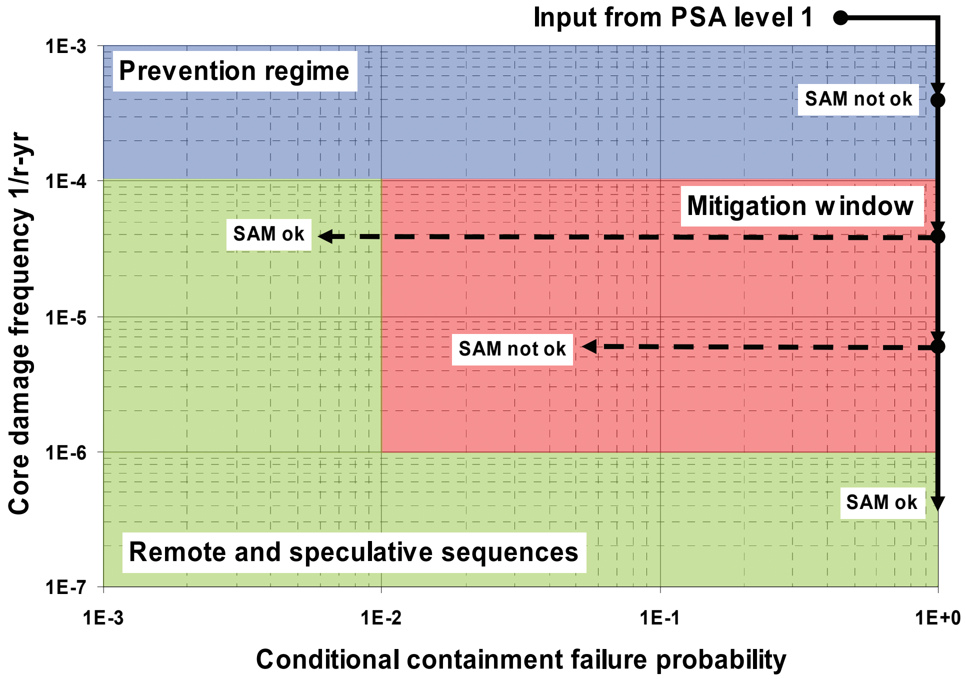

Figure 2. Two dimensions of Integrated ROAAM approach; prevention of core damage when the frequency of sequence is above the accident prevention goal and mitigation of containment threatening phenomena when frequency of sequence is inside the mitigation window.

The mitigation part of the SAM approach is built around the following SAM safety functions:

– Successful containment isolation: New approaches for actuating isolation signals, ensuring isolation status, and monitoring containment leak-tightness have been developed.

– Primary system depressurisation: Installation of high-capacity depressurisation valves (manually operated relief valves), which are separate from the primary system safety relief valves.

– Absence of energetic events (mitigation of hydrogen combustion, since successful in-vessel retention of molten corium excludes other energetic events.) A new hydrogen mitigation scheme based on containment mixing through forcing open ice condenser doors, and controlled removal of hydrogen through passive autocatalytic recombiners (PARs) and deliberate ignition has been developed.[6]

– Cooling of reactor core or core debris (reactor pressure vessel lower head coolability and melt retention). The basic preconditions for RPV lower head coolability and melt retention are a flooded cavity, a lower decay power level, and a RPV lower head without penetrations are fulfilled in case of Loviisa.[7] Certain plant modifications were necessary in order to ensure e.g. access of water to the vessel wall and sufficient flow paths for steam at the boiling channel.

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[6] Lundström, P., Kymäläinen O. and Tuomisto, H. Implementation of Severe Accident Management Strategy at the Loviisa NPP. Workshop Proceedings, PSI-Villigen, Switzerland, 10-13 September, 2001.

[7] Kymäläinen, O.,Tuomisto, H. and Theofanous T.G. In-Vessel Retention of Corium at the Loviisa Plant. Nuclear Engineering and Design 169 (1997), pp. 109-130.

– Mitigation of slow containment overpressurisation (long-term containment cooling): The approach was taken to install a containment external spray system instead of filtered venting due to certain Loviisa-specific features such as sensitivity to subatmospheric pressures and low steaming rates.[6] No other non-condensible gases than hydrogen are generated and containment steel shell makes it possible to cool containment from the outside.

All aspects of the strategy, like hardware and I&C modifications have been targeted towards ensuring the safety functions in a highly reliable manner. The SAM guidelines and procedures and the SAM Handbook have also been structured around the SAM safety functions. SAM safety functions and success or failure of mitigation systems affect PRA level 2 and are taken into account in yhe level 2 PRA modeling and source term calculations.

3 Extension of SAM strategy to shutdown states

The SAM strategy for Loviisa NPP was originally designed for severe accidents starting from power operating states. Even though the situation during shutdown states in the Loviisa NPP differs significantly from the situation during power operating states, the main requirements for severe accident management are not different and the mitigative part of the SAM strategy in shutdown states is very similar as it is in power operating states.

One of the main challenges in the shutdown states in Loviisa is that the containment function might be impaired and some recovery actions are needed. Also system maintenance, performance tests and inspections are done in shutdown states. Every second year the yearly outage in the Loviisa NPP is so called refuelling outage (15-16 days) and every second year the outage is longer; either short maintenance outage (18-24 days), 4-year maintenance and inspection outage (30 days) or 8-year maintenance and inspection outage (37-38 days). Besides these scheduled maintenance also un-scheduled repair outages are possible, though rather rare. The amount of work done during outage depends on the type and length of the outage. The maintenance of the SAM systems is also done during yearly outage, which means that the state of the systems is different from power operating states. In order to successfully mitigate the severe accident starting in shutdown states, some actions are needed to recover the functionality of containment and severe accident management systems. Conditions in the containment and initial core conditions are also different from power operating states and these conditions will change during shutdown. In some cases this means that success criteria will be not only different from power operating states but also different depending on the outage stage.

Extensive work has been done for the Loviisa NPP in order to analyse the state of containment and the state of accident management systems during shutdown. Each of the SAM safety functions (listed in chapter 2.2) has been evaluated. The success criteria have been re-assessed and the state of hardware implemented for the SAM purposes have been studied during different stages of outage. The aim has been to study the applicability of the SAM strategy and to recognise possible actions and procedures in order to improve the SAM applicability. The work has included large amount of observation work done in the Loviisa NPP during several consecutive plant outages, background studies and code calculations. The work continues further. The work done at this point has generated many reports including one Master's Thesis[8].

3.1 State of containment during shutdown

Shutdown accidents in the mitigation window (see Figure 2) are typically rather slow sequences, where the cause of core damage is either leak from the primary circuit or the transient in heat removal

[8] Björklöf, A-S. Isolation of the Loviisa NPP containment during shutdown states, Master's Thesis, Helsinki University of Technology, Laboratory of Energy Engineering and Environmental Protection, March 2006 (in Finnish).

systems. The consequences of other accidents (i.e. boron dilution, drop of heavy load) are very difficult to mitigate and the goal according to SAM strategy approach with these sequences is to prevent the accidents with high confidence (accident becomes remote and speculative).

The containment flow pattern in the accident situation in shutdown might differ significantly from the flow pattern during accident in power operation (see Figure 1). When the pressure vessel lid has been removed the primary circuit is in atmospheric pressure and the water pool in reactor shaft is the only barrier between core and containment air space. If the core overheats the steam produced will be released directly to the upper compartment. There are additional flow routes in the containment between the lower and upper compartments because maintenance hatches from the main deck are open in some stages of outage. It is also possible that access doors (one or two) to the steam generator room are open. The flow conditions differ in many ways from conditions during power operation, and the flow conditions can also change during the shutdown accident progression.

The typical flow pattern in shutdown when the ice condenser doors have been opened and hydrogen recombining has started is presented in Figure 3. Steam releases from pressure vessel to the upper compartment, from which the flow goes downwards trough both ice condenser. From the lower compartment the main deck openings are the main flow route to the upper compartment. Containment is fully mixed.

Containment flow pattern has an influence on hydrogen management and also to the melting of ice condensers. Different kind of flow patterns where studied with COCOSYS.[9] [10] [11] [12]Altogether more than 20 analyses were made for the base case and for 7 variations. The analyses were made for the situation in cold shutdown just before removing of the pressure vessel lid (primary circuit already untight) with respective decay heat. The influence of main deck maintenance hatches and steam generator room doors to the flow pattern was studied as well as the effects of the timing of the ice condenser doors opening. Also the hydrogen concentration in the containment with different hydrogen recombination capacities was studied.

[9] Hongisto, O. Loviisa 1&2, Severe reactor accident influences on the conditions in the Loviisa containment in accident starting during cold and refuelling shutdown states. Fortum, report, 14.4.2000 (in Finnish).

[10] Hongisto, O. Loviisa 1&2, Severe reactor accident influences on the conditions in the Loviisa containment in accident starting during cold and refuelling shutdown states, additional analysis. Fortum, report, 2.12.2002 (in Finnish).

[11] Hongisto, O. Loviisa 1&2, Containment conditions in severe reactor accidents starting during shutdown states. Fortum, report, 16.4.2007 (in Finnish).

[12] Hongisto, O. Loviisa 1&2, The effect of the amount of recombiners to the containment conditions in the severe accidents starting during shutdown states. Fortum, report, 3.1.2008 (in Finnish).

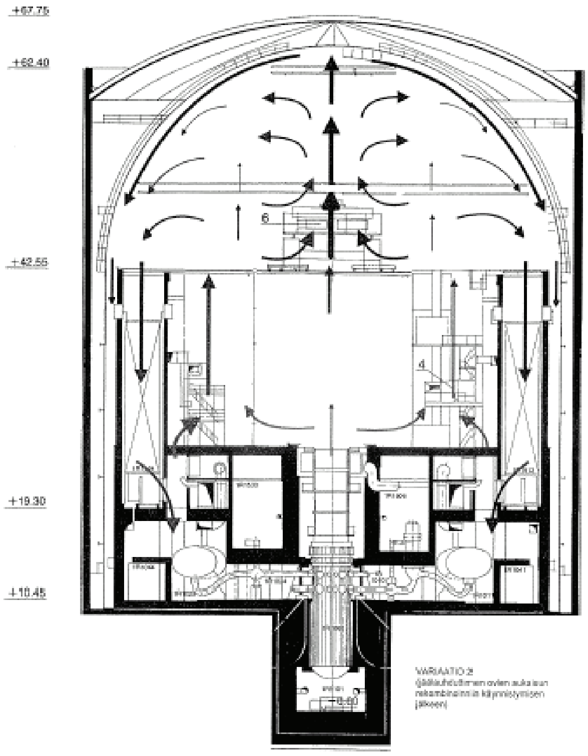

Figure 3. Typical flow pattern in shutdown when ice condenser doors are opened and hydrogen recombining has started

During shutdown containment is not tight. The material hatch might be open and there are temporary summer doors in main access hatch. Also the on-going maintenance work might cause untightness of the containment. However, in certain stages, defined by the operational limits and conditions of the Loviisa NPP, the containment will be closed (i.e. tightness will be recovered). This is the case during fuel refuelling, during spent fuel handling and during heavy load lifting. In these stages containment has basically the same requirements considering the leaktightness as during power operation.

On-going maintenance work inside the containment has an influence on the conditions in the containment. During outage there might be plenty of people working inside the containment. Systems are under maintenance and periodical testing and inspections are being made. There are more equipment and loose material inside the containment when systems are opened for maintenance and material needed for maintenance work is brought in. Also systems used for severe accident management are under maintenance work. Consequences of the different issues have to be taken into consideration when severe accident management is evaluated.

3.2 Fulfillment of the SAM safety functions in shutdown

In severe accident initiated in shutdown the same SAM safety functions have to be fulfilled as during power operation. In this chapter the mitigative part of the SAM strategy and each SAM safety function is addressed separately and the issues important for shutdown states SAM are concerned. Only the most important issues have been brought up here.

<u>Successful containment isolation</u>

Containment isolation has to be ensured in an accident situation during outage. In stages when there are no requirements for the containment leaktightness this means that recovery actions are needed.

During cold shutdown stages automatic containment isolation signals are not in use and isolation have to be manually activated in order to close the isolation valves. In some cases successful isolation has to be ensured manually on site (i.e. cases where maintenance work for an isolation valve is done). Penetrations, which are opened for maintenance purposes i.e. for the cables of inspection equipments, have to be closed and sealed. The material hatch and access hatches have to be closed and sealed. Special concern has to be given to the leaktightness of the lower compartment since there's a possibility of loosing the coolant outside the containment. The untightness of lower compartment has influence both on prevention of core damage and in-vessel melt retention.

Several procedures were found to improve the containment isolation function. Also guidance for executing the containment isolation in an accident arising from shutdown states was found to be necessary.

Primary circuit depressurization

Primary circuit is at atmospheric pressure during refuelling when the pressure vessel is open and pressurization of the circuit is not possible. During shutdown and again during start-up when the circuit is closed the high-capacity safety depressurization valves installed for SAM purposes are fully operable. It is evaluated that no further procedures are needed to fulfil the safety function.

Mitigation of hydrogen

Containment flow pattern during accidents starting in outage differs from the flow pattern during power operation as explained earlier in chapter 3.1. Even though there are more open hatches between the lower and the upper compartment, the efficient mixing for hydrogen mitigation purposes has to be ensured by forcing open ice condenser doors. During yearly outage there are maintenance and inspection work going on inside the ice condenser. Ice baskets are weighed and filled with ice to cover the ice loss and inspections for flow routes and drainage systems are made. Because of the on-going work the systems used for forcing open ice condenser doors are disabled and recovery actions inside the ice condensers are needed in order to make door opening systems usable again. Guidance for executing these recovery actions is needed.

Besides efficient mixing, the hydrogen management in the Loviisa NPP is based on controlled hydrogen removing through recombiners and deliberate ignition and burning of hydrogen with igniters, when the hydrogen release rate is high. Hydrogen igniters system consists of two sub-systems. Some maintenance work, inspections and periodical tests are made for igniters system, but at least one of two sub-systems is fully operable during the whole outage.

Recombiners, however are not operable since at the beginning of outage they are protected against possible poisoning of the catalytic plate material. Recombiners in Loviisa are located throughout the containment and recovery actions are needed in order to make the recombiners operable again. This is found rather challenging since recovery actions might take quite a while and in some accident sequences either lower compartment or upper dome might not be accessible because of water or steam leaking or evaporating (or boiling) from primary system. However since the hydrogen recombining capacity is designed with rather high margins it is possible to succeed even in those cases where recovery of all recombiners is not possible. Because of the containment flow pattern, it is even found that recombiners in the lower compartment (steam generator space) might not be needed in many sequences. Guidance for executing the recovery actions for recombiners is needed.

The hydrogen management strategy was re-assessed for shutdown purposes and new calculations were made in order to define success criteria for the recombiners.[13] COCOSYS-calculations were used to study the flow pattern in containment during shutdown as explained earlier (see chapter 3.1).

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[13] Routamo, T. Loviisa 1&2, Hydrogen management during shutdown states in the Loviisa NPP. Fortum Nuclear Services, report, 18.3.2009.

Reactor pressure vessel lower head coolability and melt retention

In order to succeed with the in-vessel retention it has to be ensured that the cavity will be flooded and steam produced can flow out from the cavity. The water shall be able to get in touch with the reactor pressure vessel bottom, which means that the thermal shield below the pressure vessel has to be lowered down (hardware modifications for executing this were made during SAM implementation).

In order to flood the cavity it has to be ensured that there will be enough water and water will not be lost outside containment through open penetrations in the lower compartment floor or through cavity door or cable penetrations in the cavity.

Access to the reactor cavity consists of two doors, which have to be closed and sealed tight in order to maintain leaktightness in conditions typical for severe accidents. Cavity door is opened during shutdown in order to do inspections and maintenance in the reactor cavity. Also periodical testing of the thermal shield lowering system are done in yearly outage. In refuelling outage and in short maintenance outage there are not that much work to do in the cavity and the period, during which the cavity door is open, could be limited. It is also possible to recover the cavity tightness if the door is not closed and sealed, but the actions will take some time and guidance for the executing of these actions is needed.

During long (4- and 8-year maintenance) outages material inspection of the reactor pressure vessel will be done using the inspection device located in the reactor cavity. In this situation there are a lot of devices and cables in the cavity. The possibilities to recover the cavity so that external cooling with water could succeed are not very good. The recovery is possible but it will take very long time. However, these inspections are only made in long maintenance outages, in which the fuel is removed from the reactor pressure vessel to the fuel pool. Inspection work in the cavity can be limited to the period where the fuel will not be in the reactor pressure vessel and these problems can be avoided.

Water flows to the cavity from steam generator room floor. Before water can open the passively operating valves, through which it flows to the cavity, the water level has to rise to a certain level. It has to be ensured that also penetrations in the steam generator room (lower compartment) floor are sealed in order not to lose water outside the containment. Also certain maintenance work done for example to the emergency core cooling system valves (sump valves) might cause the untightness of lower compartment. When this kind of maintenance work is executed, it has to be ensured that the period of containment untightness is very short. In fact in the Loviisa NPP metal flanges are used when sump valve maintenance is going on and the lower compartment leaktightness would be otherwise lost.

Lower compartment leaktightness is found to be very important not only for mitigating the consequences of severe accident management but also for defensive part of accident management. The goal is to ensure the tightness of lower compartment as far as it is possible. Guidance for the recovery actions will be needed in order to ensure the tightness in an accident starting in shutdown state.

Besides cavity tightness, it also has to be ensured that there is enough water to fill the cavity and ensure the possibility of natural cooling circle. This, in fact, is found to be problematic in some accident sequences. The decay heat is lower in accidents starting in shutdown states and the melting of ice condensers will not be as effective as it is during accidents starting in power operating states. If the core cooling and make-up water feed is lost, there is not any other water reserves in use than the water coming from melting ice and the primary circuit water inventory, which is different depending the stage of the outage. It was found that the ice condenser doors should be in many sequences opened earlier than in accidents starting during power operating states. This means that the criterion for executing the SAM action is different than in accidents starting during power operating states.

Water inventories were analysed and code calculations with APROS containment code were made in order to analyze the adequacy of water flowing from the ice condenser for cavity flooding.[14] The effects of ice condenser doors early opening were studied in analysis done with COCOSYS code (see chapter 3.1).

Long-term containment cooling

Containment code calculations with COCOSYS have shown that even in the fastest sequences the need for external spraying of steel shell for the long-term containment cooling will not be until after certain period of time.[9] [10] Even if there were some maintenance work going on for the external spray system at the time of initiating event, the external spray system can be recovered in all sequences with high reliability. Actually, in most of the sequences there will be more than 48 hours from beginning of the core melt until the need of external spray system. It is believed that specific procedures or guidance are not needed to fulfil the safety function.

3.3 Procedures and guidelines facilitating SAM strategy in shutdown

In order to mitigate the consequences of severe accident initiated during shutdown, active recovery actions are needed in order to recover the operability of severe accident management systems. In the Loviisa NPP, based on the work done during previous years, guidance for needed SAM system recovery actions has been made as well as many procedural improvements. The work is still going on. At the moment guidance is going through validation and verification. Some of the procedures, which are introduced during the process, have already been implemented and some procedures are still under further discussions. Some examples are given here:

– Guidance for SAM system recovery actions has been prepared. There is guidance for the actions needed in order to isolate the containment, recover the ice condenser operability and recover recombiners operability. The recovery actions needed as well as time and man-power needed to execute these recovery actions have been evaluated and based on these evaluations and typical accident sequences, the criteria for taking the guidelines into use has been defined.
– In order to ensure the containment lower compartment tightness some changes in the outage work planning has been made. For example all the work done in the reactor cavity has been grouped together in a work package. This way the time period during which the cavity door is not closed and sealed can be limited. Also using of the material inspection devices in the reactor cavity is limited to the time period, during which the fuel will be removed from rector to the fuel pool.
– Some restrictions have been made included in the operational limits and conditions considering the maintenance of the SAM systems i.e. for SAM dedicated electrical systems

Besides above mentioned examples many more issues are under further examination. The validation and verification of the SAM systems recovery guidelines is one of the most important on-going tasks. The further work with the criteria to take the guidance in use is one important part of the continuation work. Need for further code calculations have been recognised.

It has to be noted that guidance mentioned above, is guidance for recovering the SAM systems operability. These actions have to be started already before the core damage before the conditions in the containment will make the containment compartments non-accessible. The goal of these guides is to ensure that the SAM systems, which are needed in order to successfully mitigate the consequences if the accident would proceed to core damage, are available when needed. Difference has to be made

[14] Vierimaa, J. Modelling of severe accidents starting during shutdown states with APROS code. Fortum, report, 10.2.2009 (in Finnish).

between SAMGs which guides the execution of the mitigation actions. The Loviisa NPP has SAMGs and a part of the on-going work is also to assess their applicability to shutdown states.

4 Level 2 PRA for Loviisa NPP

First level 2 PRA study for the Loviisa NPP was done at the late 1990's in order to study the effectiveness of SAM strategy with was under implementation at that time. The study was done for internal initiators in power operating states. Later the PRA 2 was extended with studies of internal flood hazards and external events (i.e. weather hazards, oil spill accidents in Gulf of Finland etc.). At spring 2009 the first part of the shutdown extension was reported to Finnish Radiation and Nuclear Safety Authority (STUK). This part of the shutdown study included internal initiators. Currently work with the second part is going on. In the second part the shutdown study will be extended to cover also internal floods and external events.

The level 2 PRA for Loviisa will be further extended with the fire hazard study both in power operating states and in shutdown states. This study will be done after the main parts of currently on-going extensive I&C renewal project[15] at Loviisa are implemented.

4.1 Results arising from PRA level 2 for power operating states

Figure 4 shows the results from the Loviisa NPP level 2 PRA study for power operating states for containment sequences.[16] The study includes internal initiators, internal flood initiators and external hazards. Results will be presented only for containment sequences to show the efficiency of implemented SAM strategy. As explained earlier (see chapter 2.2) the SAM strategy goal with sequences posing the imminent threat for containment integrity (i.e. bypass sequences) is to prevent core damage.

If all initiator groups are taken into consideration it can be noticed from Figure 4 that in approximately 85 % of core damage sequences the consequences can be mitigated and there will be no releases to the environment. Releases will be large approximately in 8 % of all sequences and very large in 7 % of the sequences. In both classes; large release and very large release, the sequences will lead to the release which is higher than Finnish release limit (See chapter 2.2). However sequences in which the release is stated to be large instead of very large the release will happen because of late containment failure and the release is rather close to the release limit. The consequences in the environment will be different for large and very large releases.

The results show the adequacy of the SAM strategy during power operating states.

[15] Honkoila, K. Simulators for Loviisa Automation Renewal Project. Presented at IAEA Technical Meeting on Simulators, Advanced Training Tools and Technologies for the Nuclear Industry, June 2-5.2009, IAEA's Headquarters in Vienna, Austria.

[16] Routamo, T., Siltanen, S., Tarkiainen, S. and Björklöf, A-S. Loviisa 1, PRA level 2, Fortum, report, 25.8.2006 (in Finnish).

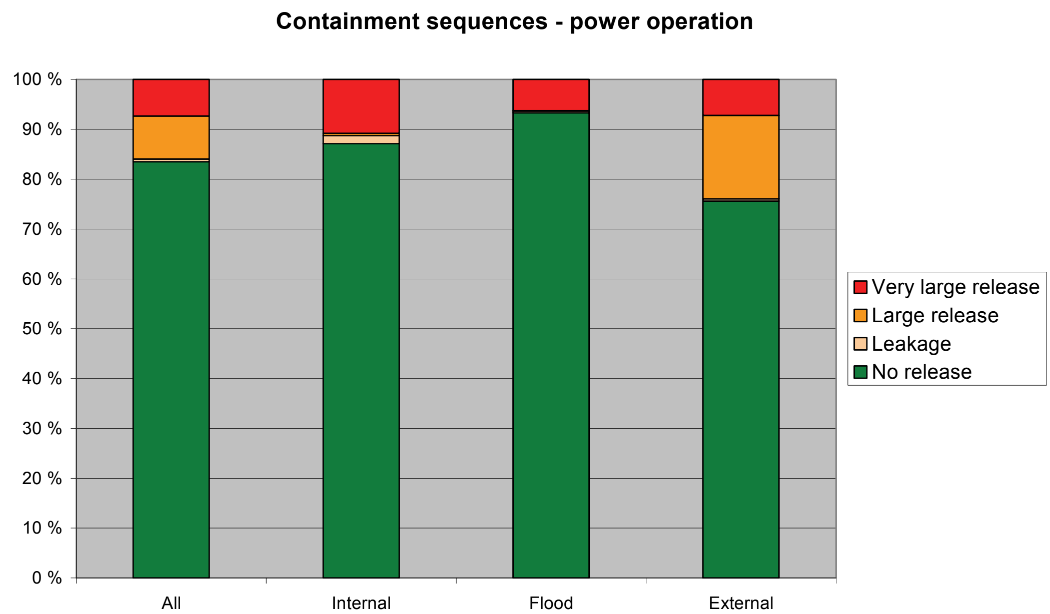

Figure 4. Level 2 PRA results for containment sequences in power operation.

4.2 Results for shutdown states

Figure 5 shows the results from the Loviisa NPP level 2 PRA study for shutdown states for containment sequences.[17] At the moment the study includes only internal initiators and results will be presented only for containment sequences to study the efficiency SAM strategy in shutdown states.

On the left in the Figure 5 (Internal) are shown the results without implementation of any specific guidance or procedures to facilitate the severe accident management in shutdown states. It can be recognised that the situation is not very good. As an example on the right (Internal - SAM) are shown the results after implementation of the SAM system recovery guidance and the procedures explained earlier (see chapter 3.3). In these figures it is expected that the guidance, which is now going through validation and verification, and procedural changes explained earlier (in chapter 3.3) are in use. It can be recognised that the situation will become better. However, it can be seen that continuation of the work will still be needed in order to achieve acceptable level.

Primary circuit leakages are rather problematic. The primary coolant leakage, in the situation where emergency core cooling or any other make-up water is not available, will lower the water level in the pressure vessel to the level of cold leg. How fast this is happening depends on the leak size. Water from the leak will end-up to the steam generator room floor (lower compartment). Water in the lower compartment has an effect on accessibility of the lower compartment and the recovery actions which might be needed will not be possible in some point. Conditions in the upper compartment dome will also suffer from steam evaporating from the water which is heating-up above the core. In current results (shown on the left in Figure 5) the assumptions made considering the accessibility might be rather conservative. Some new analysis needs have been recognised and after performing the analysis more realistic and best-estimate assumptions can be made.

[17] Siltanen, S., Purho, T., Routamo, T. and Tarkiainen, S. Loviisa 1, PRA level 2, Internal initiators, Results of the frequency calculations. Fortum, report, 19.3.2009 (in Finnish).

When shutdown results of the Loviisa NPP are concerned it should also be pointed out that currently the work related to shutdown emergency operating procedures (EOPs) is going on. This work will influence the overall accident management, especially on the preventive side, which will have an effect on the core damage frequency and also to level 2 PRA results.

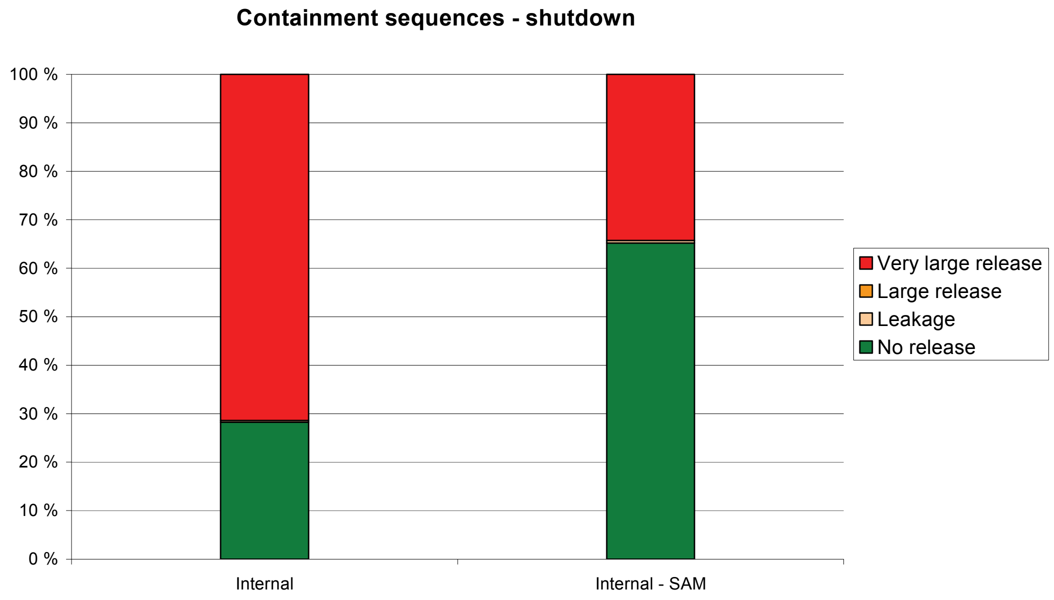

Figure 5. Level 2 PRA results for containment sequences in shutdown

5 Summary and conclusions

SAM strategy for the Loviisa NPP was originally developed and SAM systems were designed to be applied to the accidents starting from power operating states. Strategy has been implemented and PRA level 2 shows the adequacy of the mitigation part of the strategy for power operating states.

During recent years severe accidents starting in shutdown states have been widely studied. The work has shown that main parts of the SAM strategy are applicable also in shutdown states. The same safety functions are to be fulfilled in shutdown states also. However, the state of the containment and the state of SAM systems in shutdown states differs significantly from the power operating states. In an accident situation work has to be done in order to recover the operability of the containment and SAM systems already before core damage. Work with shutdown states has been started and it will be continued with further studies and implementation of procedures facilitating SAM during shutdown states.

When shutdown states of the Loviisa NPP are studied, a lot of attention has been paid also to the defensive part of the SAM strategy and the work will be further carried on also on this side of the SAM strategy. Besides the challenges with the mitigation part of the SAM strategy, which has been the main focus of this paper, a lot of work has been done in the area of sequences which pose imminent threat to the containment (boron dilution, drop of heavy load). Also this work will be continued.

Development of the SAM strategy for Paks NPP on the basis of Level 2 PSA

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1. Introduction

Level 2 PSA was performed for the Paks NPP. The main objective of the study was to quantify the frequency of large radioactivity releases due to severe accidents and to develop risk-informed recommendations for accident management strategies and measures. The study identified the important plant damage states that were used for severe accident and containment event tree analyses in the Level 2 PSA of the plant. Analyses covered all sources of potential radioactivity release, not only the reactor core. All plant operational states (full power, low power and shutdown) and all types of important initiating events (internal and external) were within the scope of the study. Results of the Level 2 PSA served as a basis for development of mitigative accident management strategies. Level 2 PSA and a following uncertainty study provided the basis to establish the strategy of the SAMG.

2. Specific design features with SAM implication

Paks NPP has 4 units of VVER-440/213 type with bubbler condenser. These reactors have extremely large water reserves both in the primary and the secondary side. Expressed in terms of time, the reserves provide a longer time span, than it is usually available for reactors of other design. The relatively small reactor core, with a very low power density is situated in a long reactor vessel containing a large amount of iron structures below the core. The six-loop arrangement leads to a high volume of the primary circuit and this ensures that the core or the pressure vessel remains intact for a longer period of time even if the core cooling has been lost.

Specific design features are characteristic to the fuel assemblies and the core, reactor pressure vessel, primary coolant loops with the horizontal steam generators, the reactor cavity, and the containment structure.

Core

There are absorber assemblies equipped with fuel follower assemblies in the VVER cores. In case of a tripped reactor about the 10 % of the assemblies are below the bottom of the core. This arrangement is influencing the core melt progression and relocation process. The relatively small core contains large amount of zirconium, thus the hydrogen generation and the threat of hydrogen combustion is a major vulnerability of the VVER units.

Vessel

The reactor pressure vessel has an elliptical bottom. In case of a core melt accident the highest elevation of the relocated debris bed will be in the cylindrical part of the vessel. That affects the outside heat transfer processes in case of external cooling, and the debris bed dynamics in the bottom of the reactor vessel would also be influenced. The RPV has relatively high surface area compared to the low decay power, which makes the external cooling more effective.

Cavity

The reactor cavity has a relatively narrow design with a door at the bottom level leading to the non-hermetic compartments. The flow-paths between reactor cavity and the rest of the containment are small. Due to this arrangement the pressure transients in the cavity may lead to peak pressures that threaten the cavity wall or the access door integrity. Since these are containment boundaries, such loss of integrity would represent a containment failure. As designed, the cavity remains dry during any type of accident and presently there are no flow-paths available to flood the cavity.

Loops

The horizontal SG design, in conjunction with the loop seals in the hot legs would very likely preclude any counter-current gas flow in the hot legs. This feature strongly influences the potential of induced primary system failures.

Containment

The containment is a system of interconnected compartments. There is a pressure suppression system consisting of a vertically mounted series of bubbler condenser trays and air traps to maximise condensation efficiency. This bubble condenser tower is connected to the main confinement by a large flow area opening and it contains about 1200 m^3 of water.

The compartment walls are approximately 1.5 m thick and have an internal steel lining. The design pressure of the containment is 2.5 bar. The mean overall containment capacity calculated by the aggregation of the capacities of all the structural elements was computed to be 0.35 MPa, and the "high confidence of low probability of the failure" capacity as 0.235 MPa. The containment integrity can be threatened by combustion of hydrogen inventory of about 45-50 % Zr oxidation fraction (350-400 kg hydrogen). This value is within the possible range of in-vessel hydrogen production.

Presently the containment is not equipped with any hydrogen control measures designed for severe accident conditions. Random ignition sources sufficient to initiate deflagration will probably be present, but a random nature of the ignition process is expected due to the high uncertainty over the time and availability of ignition sources.

The containments of VVER units have relatively high (14.7 %vol/day) design leakage rates. Recently a lot of efforts have been made to improve the leaktightness of the containment. Now, leakage rates around 5-10 %/day are quoted for these units. Despite the improvements, management of the environmental releases due to the pre-existing leakage has a higher priority in the accident management, while containment over-pressurisation remains a relatively low level concern.

If the pressure vessel failure occurs at high primary system pressure, then different energetic events may lead to the loss of containment integrity. Therefore primary system pressure reduction is a measure of major importance.

3. Summary of Level 2 PSA results

Level 2 PSA analyses can be used for the development of a severe accident management strategy with the identification of the accident sequences that result in core damage, containment failure (or containment by-pass) and the release of fission products into the environment. It is assumed as ideal tool to assess the risk impact and the risk reduction efficiency of selected SAM actions and to identify reasonable design basis for mitigative systems.

The performed severe accident analyses for Level 2 PSA[1] were the basis for the development of SAM. Taking into account the frequency of different release categories, Level 2 PSA and uncertainty studies[2] could show the importance of the elements of the accident management procedures.

At first, containment failure modes caused by different physical phenomena were determined (Table 1.). All possible reasons of the containment failure modes were assessed and their probabilities were calculated. The total sum of the probability of different containment failure modes exceeded the value 10^{-6} 1/year. Early containment failure is caused mostly by hydrogen burn. Early release from the containment can also occur in case of steam generator tube or collector break (by-pass). These two elements are the main contributors to the early large release that should be avoided.

Environmental consequences of the late enhanced containment leakage are smaller, than that of the early failures, but the frequency of late failures is higher. Late containment leakage is mainly caused by the corium melt attack on cavity door. Possible accident management measures for prevention of containment failure are shown in Table 2.

Table 1. Containment failure modes and their reasons

Containment failure modes	Main reason of the cont. failure (physical phenomena)
High pressure RPV rupture	Failure of primary depressurization (human error, valve failure)
By-pass	Steam generator tube/collector rupture
Early containment rupture	Hydrogen burn
Early enhanced containment leakage	
Late containment rupture	Containment slow overpressurization
Late enhanced containment leakage	Cavity door seal failure due to high temperature (corium near to the door)
Early containment rupture with spray	Hydrogen burn
Early enhanced containment leakage with spray	-
Late containment rupture with spray	
Late enhanced containment leakage with spray	Cavity door seal failure
Intact containment	
Intact containment with spray	

[1] Level 2 Probabilistic Safety Assessment of the Paks Nuclear Power Plant, August 2000-December 2003, Final Report, Budapest, December 2003, AEKI-PSA2-2003-778-04-11, VEIKI 20.11-214
[2] Z. Techy, G. Lajtha, Z. Karsa, P. Siklossy, A. Bareith: Uncertainty Analysis for Level 2 PSA, September 2004, Budapest, VEIKI Biztonsagtechnika+ 194-31-00 (in Hungarian)

Table 2. Possible accident management measures

Main reason of the cont. failure (physical phenomena)	Possible accident management measures
Failure of primary depressurization	SAMG
Steam generator tube/collector rupture	Bleed from ruptured SG to the containment
Hydrogen burn	Hydrogen recombiner, igniter or inerting
Cavity door seal failure	Isolation of room A004 or prevention of RPV failure
Containment late overpressurization	Filtered venting and/or spray

According to the identified main challenges two severe accident management strategies were elaborated[3] (Table 3.):

- the first strategy included hydrogen treatment using recombiners, filtered venting and prevention of the reactor cavity door damage as accident management measures;
- the second strategy included in addition reactor cavity flooding for in-vessel retention and also for protecting from the basemat melt-through.

Table 3. Possible AM strategies

	Base case	Strategy I	Strategy II
Prevention of RPV failure	ECCS recovery	ECCS recovery	ECCS recovery + reactor cavity flooding
Hydrogen treatment	-	30 recombiners	30 recombiners
Limitation of radioactive releases	Spray recovery	Spray recovery	Spray recovery
Prevention of cont. overpressurization	-	Filtered venting	Filtered venting
Safe integrity of the reactor cavity	-	Isolation of room A004	Solved by cavity flooding
(External cooling of the molten material)	-	-	(Not challenged)

Both strategies were investigated and the corresponding probabilities were quantified. There were only minor differences between the strategies in terms of the probabilities of the early containment failure. However, if other considerations, such as the cost of the modifications and mitigation of the consequences are taken into account, then the two options differ substantially.

As a summary evaluation of the results the following conclusions were drawn.

- Due to AM measures, the distribution of release category frequencies is significantly modified compared to the present state.
- The frequency of most severe high-pressure sequences is not influenced by the AM measures, however, their frequency is not significant ($<10^{-7}$/unit/year).
- The frequency of early containment failures can be reduced to 1/3 of the present value by the AM measures.
- The frequency of early containment failure with spray can be reduced to 1/4 of the present value by hydrogen treatment.

[3]Level 2 Probabilistic Safety Assessment of the Paks Nuclear Power Plant, August 2000-December 2003, Final Report, II. Assessment of the Accident Management Strategies, Budapest, December 2003, AEKI-PSA2-2003-778-04-12, VEIKI 20.11-215/1.

The two AM strategies do not differ significantly in terms of atmospheric release, however, a difference can be found in the basemat failure. The reactor cavity flooding protects the basemat from melt through in many cases, decreasing its frequency from $1.83 \cdot 10^{-5}$ to $1.6 \cdot 10^{-7}$/unit/year. The cavity door protection does not affect the basemat failure probability. As a result of the studies a unified strategy has been developed. The main points of the proposed strategy are:

- hydrogen mitigation with recombiners,
- in-vessel melt retention via flooding the cavity,
- using a ventilation system for filtered venting to prevent late overpressurization.

After the quantification of event trees and the binning of end states into release categories, one can sum up the frequencies of the various release categories[4]. The result is given in the Table 4. (seismic initiating events are not included).

It was found that the released amount of ^{137}Cs isotopes and the doses at 1 km distance from the plant in the early and later phase of the accident are more or less proportional with each other in the case of release categories 1-13. In the other categories due to the extended decay periods the iodine content of the fuel is much less and the released caesium activity is no more proportional with the consequences. For these cases the releases have milder health effects.

It is doubtless that releases belonging to release categories 1, 2, 3, 14, 15, 16 and 17 have to be considered to large radioactive releases while the release categories 8, 9, 10, 12 and 13 are really harmless.

Table 4. Frequency of the release categories

Release category		**Frequency [1/unit/year]**
Initially closed containment, full power and shut-down states		
1	High pressure RPV rupture	$6.38 \cdot 10^{-8}$
2	By-pass	$1.73 \cdot 10^{-6}$
3	Early containment rupture	$3.82 \cdot 10^{-6}$
4	Early enhanced containment leakage	$< 10^{-8}$
5	Late containment rupture	$< 10^{-8}$
6	Late enhanced containment leakage	$1.19 \cdot 10^{-5}$
7	Early containment rupture with spray	$1.01 \cdot 10^{-6}$
8	Early enhanced containment leakage with spray	$< 10^{-8}$
9	Late containment rupture with spray	$< 10^{-8}$
10	Late enhanced containment leakage with spray	$4.95 \cdot 10^{-8}$
11	Intact containment	$< 10^{-8}$
12	Intact containment with spray	$8.51 \cdot 10^{-6}$
13	Partial core damage	$6.65 \cdot 10^{-6}$
Open containment, shut-down states		
14	Loss-of fuel cooling (high decay heat)	$1.47 \cdot 10^{-6}$
15	Loss-of fuel cooling (low decay heat)	$5.73 \cdot 10^{-7}$
Open containment, spent fuel pool accidents		
16	Loss of cooling	$1.14 \cdot 10^{-6}$
17	Loss of coolant	$4.13 \cdot 10^{-7}$

[4] A. Bareith, G. Lajtha, Z. Techy: Level 2 Probabilistic Safety Assessment of the Paks Nuclear Power Plant, Summary report, March 2008, Budapest, VEIKI 22.22-712/4 (in Hungarian)

Level 2 PSA analyses were performed also for shutdown mode operation and for the spent fuel storage pool. Shutdown mode analyses were arranged in two groups by the state of the reactor: open and closed reactor.

- Closed reactor: In these cases the PDS and APET for nominal power were used. For mitigation the SAMG identified for nominal power operation can be used.
- Open reactor: Different initial events (heavy load drops, loss of decay heat removal from the core, before and after refuelling) were analysed. Releases in these cases are passing into the reactor hall, which is out of the containment, so the releases are going directly to the environment. Therefore, a new type of SAMG is necessary.

For the spent fuel storage pool the following initial events were considered[5]:

- loss of heat removal (pump failure);
- loss of coolant (pipe break).

Releases in these cases are also leading from the reactor hall to the environment. Therefore, mitigative actions are practically impossible, thus AM measures should focus on the preventive measures. For prevention these accidents the development of extended shutdown EOP and some hardware modifications (additional automatic closing valves) are needed.

4. Accident management strategies and their components

The target of the accident management is the overall capability of the plant to respond to and recover from a severe accident situation. This capability could be increased by hardware modifications and with a guide to use the available resources in an optimal way. One of the key elements of this AM program is the SAMG that are already under development for Paks NPP.

The approach is structured around the four safety objectives of prevention of core damage, prevention of reactor vessel failure, prevention of containment failure and limitation of fission product release. Because of the variety of processes there is a need for different interventions and the selection of the right intervention needs a strategic decision. The main demand is consistency: this means that there should be a comprehensive connection between each element and they should not disturb or impair the effect on each other.

The key elements of AM strategies are the following: (1) prevention of the core damage, (2) prevention of RPV failure by in-vessel retention, or in the case it appears unfeasible (3) ex-vessel debris cooling, and (4) release and containment management.

(1) The ***prevention of the core damage*** would be accomplished by recovery from inadequate core cooling or from loss of heat sink situations. There are certain strategies to cope with the station black-out and assure long term water sources from the bubbler tower for loss of emergency recirculation events as well. All those actions should be included in the EOP.

(2) An indispensable part of the ***in-vessel retention*** is the strategy for *depressurisation* of the primary circuit. Since the depressurisation has not only mitigative but also preventive aspects it should take place early enough and in a very reliable way, according to the relevant EOP. In a few special cases the in vessel flooding can prevent the reactor pressure vessel failure but creditable in-vessel corium retention can only be accomplished by the active cavity flooding. It is already clear that such IVR is potentially feasible but certain measures and modifications would be required.

[5] A. Bareith, E. Hollo, J. Nigicser, P. Siklossy: Reliability studies of systems in Paks NPP, Level 1 PSA analysis of the spent fuel pool for internal hazards, November 2006, VEIKI 22.22-503/2 (in Hungarian)

(3) ***Ex-vessel debris cooling*** has no more challenges, as it is should be solved by successful early cavity flooding.

(4) ***The containment strategies*** have three interconnected aspects. The first priority is given to the management of *pre-existing containment leakages*, which will predetermine the amount of the early release and will inherently affect *the late containment overpressurization*. The influence of the possible mitigation techniques (i.e. containment spraying) should be taken into account when developing the *hydrogen mitigation* scheme.

According these aspects AM program for Paks NPP has been developed and approved by the regulatory body. This program contains a set of planned ***plant modifications***. The most important ones are described as follows.

Primary circuit depressurisation is a necessary measure to prevent high-pressure core meltdown sequences and to reduce the risk for induced steam generator tube rupture through the circulation of hot gases. In case the in-vessel retention of corium strategy is selected, a low primary system pressure is also a definite requirement. The recently implemented depressurisation capability would reduce the pressure sufficiently as it was designed for potential releases of steam, two-phase mixture and water. In order to ensure sufficient opening reliability an independent so-called SAM power supply of the valves has yet to be installed.

The PSA results indicate that the risk of large releases dominated by *containment by-pass* sequences that caused leaks from primary to secondary side (PRISE) of the steam generators. It is an effective precaution against containment bypass to implement blow-down lines on the bottom of the steam generators that are directed to the containment.

Severe accident hydrogen is confirmed to be as a major threat to containment integrity. The rapid onset of flammable conditions in an unmitigated severe accident necessitates a means of control. With the help of level 2 PSA it was showed that implementation of about 30 large PARs would ensure that the containment would not experience high pressures loads in all those sequences that dominate the overall risk.

Both, *in-vessel corium retention* or ex-vessel debris cooling can only be accomplished by the active cavity flooding. It is already clear that IVR is potentially feasible, but the potential for coolability of corium or core debris on the concrete basemat is still under investigation. There are double hermetic steel doors with rubber sealing in the sidewall of reactor cavity, which is a part of hermetic boundary. In case of ex-vessel cooling the thermal protection of those doors against temperature loads should also be solved. In order to avoid a number of specific loading mechanisms caused by the eventual melt ejection into the reactor cavity the necessary plant modifications needed for corium localisation and stabilisation inside RPV gets higher priority in the SAM programme.

An IVR concept with simple ECVR loop based only on minor modifications of existing plant technology was proposed for Paks NPP[6][7]. Solution is shown on Figure 1. The analyses supported the assumption that the proposed solution is effective in preserving RPV integrity in the case of a severe accident.

[6] P. Matejovic, M. Bachraty, M. Barnak: In-vessel corium retention for Paks NPP. Analysis of LB and SB LOCA sequences, IVS reports, Trnava, August 2009

[7] R. Berky, J. Bosansky: Computational Analysis of Reactor Pressure Vessel Dilatation, IBOK Technical Report, Bratislava, August 2009

Efficiency of the ERVC loop in given IVR geometry should be proven experimentally by AEKI on CERES facility, which is under construction in Budapest.

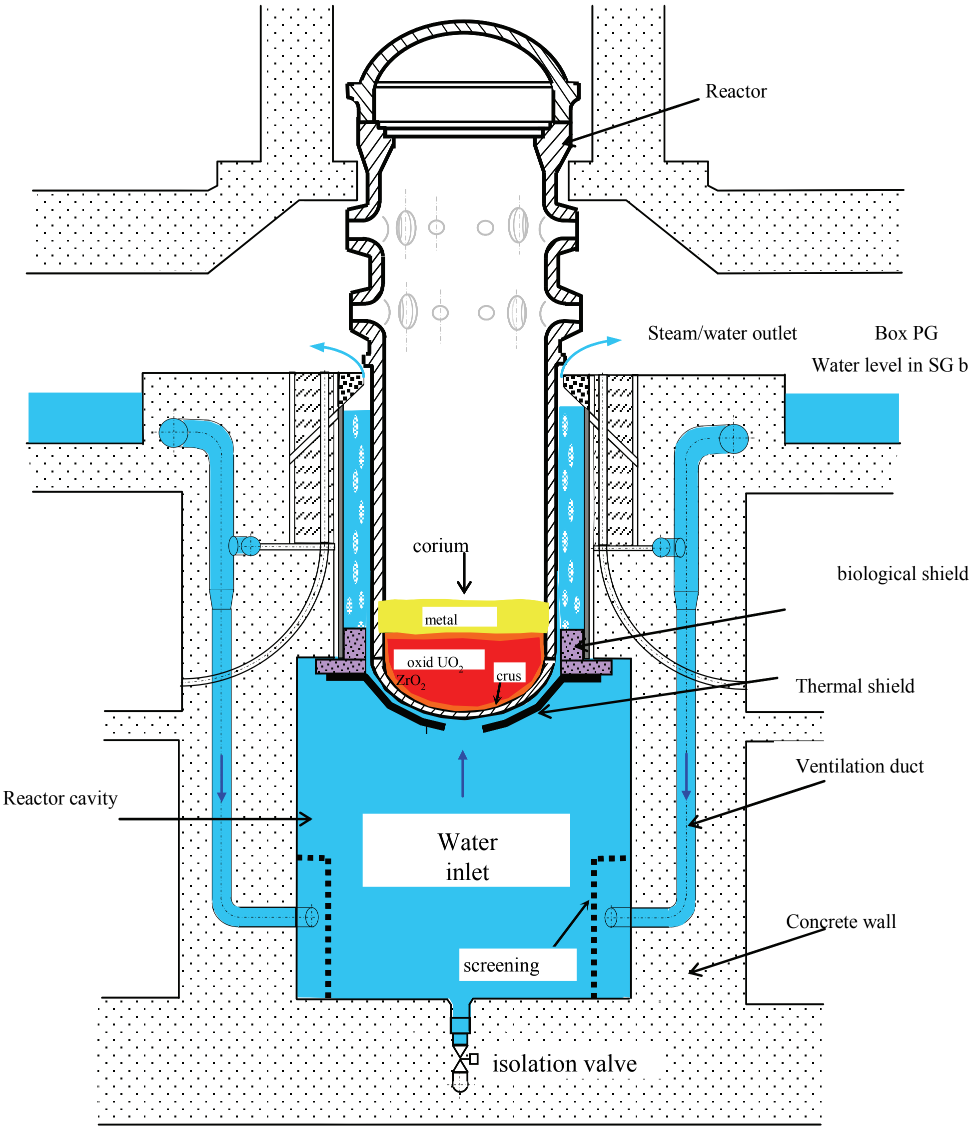

Fig 1. In-vessel retention by cavity flooding, proposal with central hole in thermal shield

Filtered venting is used to prevent late containment failure. It is an effective precaution against late containment overpressure to modify operating confinement vent system to use as filtered venting.

SAMG development should be based on the implemented plant modifications and measures. Important standpoint also, that SAMG should be linked with the already implemented Westinghouse type EOPs. Therefore, SAMG development was done also in cooperation with Westinghouse Co.

Preventive measures for open reactor and spent fuel storage pool have also high priority because of large directly releases to the environment. They have two important aspects: one of them is the extension of Westinghouse type EOPs for shutdown mode, this work is just finishing. An other important aspect coming from Level 1 PSA for storage pool is the reinforcement of the storage pool cooling system (installation of fast closing valves).

5. Plant modifications planned by a 2-phase schedule

Severe accident management measures and plant modifications in Paks NPP have been selected by two priorities:

- *1. priority measures:* They will be taken anyway, independently of the lifetime extension of the units. They are essential plant modifications, procedure development and organizational arrangements. These measures are scheduled up to 2012, which is the data of receiving the lifetime extension licence for 1st Unit.
- *2. priority measures:* They will be taken only in case of life time extension has been permitted by authority. These measures are scheduled after 2012.

These 2 phase AM measures can be selected also by different safety objectives/goals are listed below.

1. priority measures selected by safety goal:

Prevention of the core damage:

- Extension of EOPs for shutdown mode and storage pool accidents (development is just finishing).
- Implementation of 2 severe accident diesels for autonomous electrical supply for SAM equipment: e.g. PRZ valves for successful primary depressurization, drainage valve of bubble condenser trays, new inlet valves for cavity flooding, severe accident instrumentation, etc. (a preliminary licence is obtained, implementation is planning).
- Implementation of new PRISE strategy: bleed from ruptured SG to the containment before it filled up (implementing now).
- Reinforcement of storage pool cooling system, installation of new automatic by water level closing valves (licence application document has been finished).
- Implementation of new strategy for ECCS tests (under development).
- Arrangement of duties for the other, non-damaged units.

Prevention of reactor vessel failure and early containment failure:

- Development of SAMGs, still with provisional elements (just finishing).
- Establishment of Technical Support Centre.
- Modification of the operating procedures: to ensure the availability of containment spray system and water from bubbler condenser trays for open reactor and spent fuel pool cooling (under development).
- Installation of high capacity PARs to solve hydrogen issue (a preliminary licence is obtained, selection of manufacturer is ongoing).
- Design and installation of cavity flooding flow path (licence application document has been finished).
- Installation of new independent severe accident measurement system (preliminary licence is obtained).

2. priority measures selected by selected by safety goal:

Prevention of late containment failure:

- Increase reliability and protection of the spray system from common cause failures.
- Modification of containment vent system TN01 to be used for filtered venting.
- Finalisation of SAMGs on the base of hardware modifications.

6. Conclusions

Nuclear power plants with VVER-440/213 type reactors have specific design features, therefore these plants should have specific AM strategy and SAMG. Selection of the possible severe accident management strategy was based on the results of a Level 2 PSA study. The main points of the proposed strategy are: hydrogen mitigation with recombiners, in-vessel melt retention by flooding the cavity and using an existing ventilation system for filtered venting to prevent late overpressurization.

Severe accident management measures and plant modifications for Paks NPP are planned to be implemented by a 2-phase schedule:

- 1. priority measures up to 2012;
- 2. priority measures after 2012.

Abbreviations

APET	Accident Progression Event Tree
AM	Accident Management
CERES	Cooling Effectiveness on Reactor External Surface
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
IVR	In-Vessel Retention
MCP	Main Coolant Pump
MLIV	Main Loop Isolation Valve
MCCI	Molten Corium Concrete Interaction
NPP	Nuclear Power Plant
PAR	Passive Autocatalytic Recombiner
PDS	Plant Damage State
PRISE	Primary to Secondary Leakage
PSA	Probabilistic Safety Assessment
RPV	Reactor Pressure Vessel
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidance
SG	Steam Generator

Development of Technical Bases for Severe Accident Management in New Reactors

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Introduction

The NRC has taken an active role to ensure that utilities adopt acceptable management (AM) practices. In January, 1989, the Staff issued SECY 89-012, "Staff Plans for Accident Management Regulatory and Research Programs,' discussing essential elements of a utility AM plan and offering an approach for its implementation. Subsequently, the NRC worked with the industry to define the scope and attributes of a utility AM plan and to develop guidelines for plant-specific implementation. The result was Section 5 of NEI 91-04, Revision 1, ("Severe Accident Closure Guidelines"), which lays out the elements of the industry's severe accident management (SAM) closure actions that have been accepted by the NRC staff.

The AM programs are based on a technical basis for systematically evaluating and enhancing the ability to deal with potential severe accidents. Vendor-specific AM guidelines were developed for use by individual utilities in establishing plant-specific procedures and guidance. From these, guidance and material to support utility activities related to training in severe accident prevention and mitigation was developed. From these, each utility has prepared and implemented plant-specific AM plans.

The regulatory basis for existing plants is described in NEI 91-04 (originally NUMARC 91-04), "Severe Accident Issue Closure Guidelines," which includes a summary of the important information pertaining to the agreement between the nuclear industry and the NRC on severe accident management, and contains the binding implementing guidance. This industry initiative has been endorsed by the NRC. The current industry technical basis stems from EPRI's "Severe Accident Management Technical Basis Report" (EPRI TR-101869), which was used in developing vendor-specific guidance. This guidance was provided to the NRC by the various owners groups and constitutes the technical basis of severe accident management for existing plants.

Although new reactor designs are to have enhanced capabilities for preventing and mitigating severe accidents, AM remains an important element of defense-in-depth for these designs. However, the increased attention on accident prevention and mitigation can be expected to alter the scope and focus of AM relative to that for operating reactors. For example, increased attention on accident prevention and the development of error-tolerant designs can be expected to decrease the need for operator intervention somewhat, while increasing the time available for such actions if necessary. This permits a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), the need for human intervention and accident management will continue.

For both operating and advanced reactors, the overall responsibility for AM, including development, implementation, and maintenance of the AM plan, lies with the nuclear utility. However, the vendors' guidance has continued to serve as the technical basis for SAM procedures and for training utility personnel in carrying them out. Computational aids for technical support have been developed, information needed to respond to a spectrum of severe accidents has been provided, decision-making responsibilities have been delineated, and utility self-evaluation methodologies have been developed. The NRC's Office of New Reactors (NRO) staff expects that this approach will be adopted by the applicants for new reactor licenses as well. Accordingly, the applications for design certifications are being reviewed in such a manner as to ensure that the technical basis for AM will be provided by the vendors for each design.

As stated in NEI 91-04, AM consists of those actions taken during the course of an accident by the Emergency Response Organization (ERO), including plant operations, technical support and plant management staff, in order to:

- Prevent the accident from progressing to core damage;
- Terminate core damage progression once it begins;
- Maintain the capability of the containment as long as possible; and
- Minimize on-site and off-site releases and their effects.

The latter three actions constitute a subset of accident management referred to as severe accident management (SAM). They provide guidance when Emergency Operating Procedures (EOPs) are no longer effective. For existing plants the approach is to make full use of existing plant capabilities, including standard and non-standard uses of plant systems and equipment.

Severe accident management review for new reactors

During the process of reviewing design certification applications, the NRC staff is requesting the applicants to provide the revised technical bases for the new plants, to ensure that the new features for accident prevention and mitigation are included. The NRC staff is not asking for specific severe accident management guidelines. Instead, it needs to evaluate the technical bases that would support the guidelines.

Upon completion of a design certification review, a Safety Evaluation Report (SER) will be issued and, once the design certification is granted, a utility can obtain a combined license (COL) to build and operate such a plant. Before operation can commence, the NRC must approve the AM procedures. Thus, it is important for the NRO staff to accept the vendors' description of the technical basis. Before a COL is issued, an applicant needs to demonstrate that it understands the new technical basis and has established a framework for incorporating it into the AM procedures for its new plant. After a COL is issued, it is imperative that the new plants also have their SAMG implementation in place, including procedures and training, prior to initial fuel load.

Insights regarding severe accident mitigation features in new reactors

The new reactor designs all include features that increase the capability for mitigating severe accidents. These address the concerns expressed by the NRC staff in SECY-90-016 and SECY-93-087, and the associated Staff Requirements Memoranda (SRM) issued by the Commission that describe the new requirements that must be met. Noteworthy among the issues addressed are: hydrogen control; core debris coolability; high-pressure core melt ejection; containment performance (including the possible effects of molten core/coolant interactions); containment bypass, including from steam generator tube ruptures; and equipment survivability. The increased attention on accident prevention and mitigation for new reactors can be expected to alter the scope and focus of AM relative to that for operating reactors, while remaining an important element of defense-in-depth. For example, increased attention on accident prevention and the development of error-tolerant designs can be expected to decrease the need for operator intervention somewhat, while increasing the time available for such actions if necessary. This permits a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), the need for human intervention and accident management will continue.

Since the new designs include safety enhancements not present in existing plants, a major focus of the NRO staff review is on the roles new severe accident mitigation features would play in AM strategies. The new-plant vendors have all used insights derived from their probabilistic risk assessments (PRA), and are using the MAAP4 code to simulate accident progression for the more-likely severe accident scenarios, as bases for modifying their existing AM guidance. In some cases, different strategies from those used for existing reactors must be adopted. For example, the ESBWR design includes a device (the BiMAC) that is intended to arrest core melt progression in the lower drywell by cooling the debris both from above and below. For all currently operated LWRs, severe accident management requires that, provided a sufficient floor area available for spreading and a sufficient amount of water to cover the molten core debris, the debris will become quenched and will remain coolable thereafter. While the ESBWR satisfies the basic conditions for this approach, the core-on-the-floor approach is further improved, it is also necessary to ensure that a large ex-vessel steam explosion would not occur immediately after vessel breach. To prevent this, the lower drywell (LDW) needs to be kept dry until after the debris enters. Consequently, the vendor is recommending that the strategy for flooding containment currently in place for the existing boiling water reactors in the United States be modified for ESBWR plants. A similar argument can be made for ABWR plants.

The PWR vendors are also modifying their strategies related to depressurizing the primary system, due to the incorporation of severe accident-related depressurization valves into their designs. Such valves would reduce the risk from induced steam generator tube ruptures in high-pressure scenarios, as well as greatly mitigate the consequences of high-pressure core melt ejection.

A discussion is presented below of some of the more important mitigation features in designs for which COLA applications have been filed. Insights pertinent to severe accident management are also discussed.

AP1000 design. The AP1000 design has already been certified by the NRC and a number of COL applications are currently being reviewed. A design certification amendment is also under review.

External reactor vessel cooling (ERVC). The AP1000 design incorporates ERVC as a strategy for retaining molten core debris in-vessel in severe accidents. The objective of ERVC is to remove sufficient heat from the vessel exterior surface so that the thermal and structural loads on the vessel (from the core debris which has relocated to the lower head) do not lead to failure of the vessel. By maintaining RPV integrity, the potential for large releases due to ex-vessel severe accident phenomena, i.e., ex-vessel fuel/coolant interactions (FCIs) and core debris/concrete interactions (CCI) is eliminated.

The AP1000 design includes several features that enhance ERVC relative to operating plants, specifically:

- safety-grade systems to provide automatic or manual RCS depressurization and reactor cavity flooding;
- a "clean" lower head that has no penetrations; and
- an RPV thermal insulation system to limit thermal losses during normal operations, while providing an engineered pathway to supply water for cooling the vessel, and to vent steam from the reactor cavity, during severe accidents.

The AP1000 Level 2 PRA estimates that the ERVC would enable the majority (~97 percent) of core melt accidents (that do not involve containment bypass or containment isolation) to be arrested in the vessel. Depressurization of the RCS and reactor cavity flooding contribute to the success of ERVC.

Combustible gas control. The containment hydrogen control system serves the following functions:

- hydrogen concentration monitoring;
- hydrogen control during and following degraded core or core melt scenarios (provided by hydrogen igniters).

In addition, non-safety-related passive autocatalytic recombiners (PARs) are provided for defense-in-depth protection against the buildup of hydrogen following a LOCA. The hydrogen ignition subsystem meets the requirements of 10 CFR 50.44 for future water-cooled reactors, whereby the design must limit hydrogen concentrations in containment from a release of a 100% fuel clad-steam reaction to less than 10% by volume, and maintain containment structural integrity and appropriate accident-mitigating features. This requirement was promulgated to address the lessons learned from the TMI accident.

The hydrogen ignition subsystem is designed to promote hydrogen burning soon after the lower flammability limit is reached in the vicinity of an igniter, and prevent the concentration from reaching 10%. This would provide confidence that containment structural integrity can be maintained during hydrogen burns.

Core debris coolability. The AP1000 design relies primarily on safety grade RCS depressurization and reactor cavity flooding capabilities to prevent RPV breach and CCI, but also incorporates plant features consistent with the guidance in SECY-93-087 regarding debris coolability. In the unlikely event of RPV failure, these features would reduce the potential for containment failure from CCI. The AP1000 design features include the following items:

- a cavity floor area and sump curb to allow debris spreading without debris ingression into the reactor cavity sump;
- a manually-actuated reactor cavity flooding system to cover core debris with water and maintain long-term coolability;
- a 0.85 m (2.8 ft) thick layer of concrete to protect the embedded containment shell, with an additional 1.8 m (6 ft) thick concrete layer below the liner elevation.

The applicant calculated that adequate reactor cavity flooding is achieved in about 98 percent of the sequences identified in the AP1000 PRA. About half of the core damage events require operator actuation of the cavity flooding system to ensure successful cavity flooding, but the remaining half would adequately flood as a direct consequence of the accident progression, even without manual actions.

High-pressure core melt ejection (HPME). Two features have been incorporated into the AP1000 design to prevent and mitigate the effects of HPME (including direct containment heating (DCH)), specifically, the automatic depressurization system (ADS) and reactor cavity design features.

The ADS is one of the major features of the AP1000 design. It is an automatically-actuated, safety-grade system with four different valve stages that open sequentially to reduce RCS pressure sufficiently to ensure long-term cooling. If automatic actuation fails, then operator action from the main control room could initiate depressurization. The ADS valves are designed to remain open, thereby preventing repressurization of the RCS. It is estimated in the PRA that sufficient depressurization is achieved in about 95 percent of the core melt sequences.

The design of the reactor cavity is such that most of the ejected core debris would not reach the upper containment. The pathways for debris transport from the cavity include the following:

- annular openings between the coolant loops and the biological shield wall leading to the SG compartments;
- the area around the reactor vessel flange leading directly to the upper compartment; and
- a ventilation shaft leading to the SG compartments.

Each pathway is such that the debris particles would change direction and encounter obstacles before reaching the upper containment region. It should be noted, however, that deposition of core debris and aerosols in pump rooms and steam generator compartments is possible, and could affect accident management strategies.

ESBWR design. The ESBWR design is under review has not yet been certified by the NRC. However, a number of COL applications are currently being reviewed.

Combustible gas control. Just as for existing BWRs, the ESBWR containment would be inerted during full-power operation. Consequently, insufficient oxygen would be present to cause a hydrogen burn during a severe accident.

Results from the applicant's MAAP 4.0.6 simulations show that the time required for the oxygen concentration to increase to the de-inerting value of 5 percent is significantly greater than 24 hours for a wide range of fuel cladding-steam interaction and iodine release assumptions. Accumulation of combustible gases that may develop in the period after about 24 hours would be managed by implementing the severe accident management guidelines. Risk and safety of operations when the containment is not inerted (e.g., when the containment is open during shutdown) will need to be considered, because combustible gas generation can occur in locations with oxygen present at combustion-supporting levels.

Core debris coolability and fuel-coolant interactions. Two design features, the Gravity-Driven Cooling System (GDCS) and the Basemat Internal Melt Arrest and Coolability device (BiMAC), act to prevent significant ablation of the concrete in the lower drywell (LDW). The deluge mode of GDCS operation provides flow to flood the LDW when the temperature in the LDW increases enough to be indicative of RPV failure and core debris in the LDW. The GDCS pools supply water to the BiMAC device via squib valves that are activated on the deluge lines. Flooding the LDW after the introduction of core material minimizes the potential for energetic FCI at RPV failure. Covering core debris with water provides scrubbing of fission products released from the debris and cools the corium, thus limiting potential CCI. Failure of the squib valves to function properly would cause an order-of-magnitude increase in the large release frequency (LRF) for the ESBWR, but the total LRF would be well below the goal of 1.0×10^{-6} per reactor-year.

The BiMAC gives additional assurance of debris bed cooling by providing an engineered pathway for water flow through the debris bed. It is a passively cooled barrier to core debris on the LDW floor. The design features a series of side-by-side inclined pipes, forming a jacket, which is passively cooled by natural circulation when subjected to thermal loading. Water from the GDCS pools enters the pipes via connecting downcomers. Once the pipes fill up, the debris is also cooled from above from water that flows out of them. The timing and flows are such that cooling becomes available immediately upon actuation. Timely flooding of the LDW and a properly-functioning BiMAC device would make the issue of corium-concrete interactions inconsequential. Moreover, the design procedure of not immediately adding water greatly reduces the probability of a highly energetic steam explosion.

High-pressure core melt ejection. The probability of a HPME is significantly reduced by the highly reliable ADS. In addition, the following ESBWR containment design features mitigate the possible effects of high-pressure core melt:

- The containment is segregated into an upper drywell (UDW) and LDW that communicate directly, but the design mitigates the ability of high-pressure core melt, ejected within the lower drywell, to reach the UDW.
- The UDW atmosphere can vent into the wetwell through a large vent area, making it virtually impossible to overpressurize the drywell volume.
- The containment steel liner is structurally backed by reinforced concrete, which cannot be structurally challenged by DCH.

The upper drywell head is immersed in a water pool during normal operation. Consequently, analyses have shown that thermally-induced failure of the upper drywell head and its seals would be physically unreasonable. Moreover, bounding calculations have shown that upper drywell liner temperatures would be considerably below its melting point. However, the calculations also show short periods of potentially very high temperatures in the LDW atmosphere (up to 4000 °K) under some highly unlikely conditions. These temperatures, and the presence of potentially large quantities of melt in the LDW, indicate that the LDW liner could be subject to local failures.

Containment performance. Because of the passive cooling function in the ESBWR containment, the vacuum breakers between the wetwell and UDW are designed be essentially leak-proof, to prevent the possibility of containment bypass during a severe accident. Three vacuum breakers are installed in the diaphragm floor, to limit the magnitude of a negative pressure differential between the drywell and the wetwell. They operate passively in response to a negative drywell-to-suppression pool pressure gradient and are held closed by a combination of gravity and the normally positive pressure gradient.

Four position sensors are located around the disk periphery of the primary vacuum breakers to confirm to the plant operator that the disks are securely seated. The analysis in the ESBWR PRA assumes that the position switch that provides annunciation in the control room can sense a very small gap between the disk and the seating surface.

Each vacuum breaker is equipped with a diverse, redundant, passive, process-actuated check-type isolation valve, which provides isolation capability if the vacuum breaker sticks open or leaks in its closed position. The isolation valve is normally in the closed position and, like the vacuum breaker itself, is process-actuated by differential pressure between the structure and component (SC) and drywell. In this manner, the isolation valve is more like a redundant vacuum breaker than an isolation valve, and both valves would have to leak simultaneously to create a leakage path from the SC to the drywell. Including the isolation valves significantly decreases the large release frequency.

U.S. EPR design. The U.S. EPR design is under review has not yet been certified by the NRC. However, a number of COL applications are currently being reviewed.

Combustible gas control. The containment has a dedicated combustible gas control system (CGCS) to avoid containment failure from rapid deflagration or from accidental ignition of a critical gas mixture. The CGCS system is divided into two subsystems: the Hydrogen Reduction System (HRS) and the Hydrogen Mixing and Distribution System. The HRS consists of both large and small passive autocatalytic recombiners (PAR) installed in various parts of the containment. In the presence of oxygen, the PARs would automatically start if the threshold hydrogen concentration is reached at the catalytic surfaces.

The PAR locations and arrangement inside the equipment rooms and containment dome are such that they support global circulation within the containment, and thereby homogenize the atmosphere and reduce locally high hydrogen concentrations to below 10 percent by volume during various phases of accidents resulting in oxidation up to 100 percent of the zirconium surrounding the reactor core fuel, and ensure that the global hydrogen concentration can be maintained below the lower flammability limit of 4 percent by volume of the containment atmosphere in the long term.

The hydrogen mixing and distribution system would ensure that adequate communication exists throughout the containment to facilitate atmospheric mixing. Several of the equipment rooms surrounding the RCS are isolated from the rest of the containment during normal operation. In the event of an accident, communication is established between these equipment rooms, thereby eliminating any potential dead-end compartments where non-condensable gases could accumulate. This ability to transform the containment into a single convective volume is supported by a series of mixing dampers and blowout panels.

The hydrogen concentration and its distribution within various compartments of the containment are continuously monitored, and information would be available to the main control room.

Results of 59 uncertainty analysis simulations carried out by the applicant show that the global hydrogen concentration did not reach or exceed 10 percent by volume for any one of the scenarios, due to the effectiveness of the PARs.

Core debris coolability and containment performance. The Core Melt Stabilization System (CMSS) and the Severe Accident Heat Removal System (SAHRS) are features designed to ensure core debris coolability. The CMSS would stabilize core debris exiting the RPV before it could challenge containment integrity. Initial stabilization would take place in the reactor cavity, where temporary retention is achieved by a layer of sacrificial siliceous concrete that must be penetrated by the melt before it can escape from the cavity by failing a melt plug. This delay would provide a means for allowing practically the entire molten core inventory to be collected in the cavity. The sacrificial layer is backed with a protective refractory material that has a low thermal conductivity and a mechanical strength greater than concrete, to confine the melt and insulate the RPV support structure. The protective layer "guides" heat transfer from the melt toward the melt plug. Once the melt plug fails, the melt would flow down a discharge channel into a spreading compartment, which consists of a large horizontal concrete surface over which the molten core debris can be dispersed.

Arrival of the melt into the spreading compartment triggers the opening of spring-loaded valves that initiate the gravity-driven flow of water from the in-containment refueling water storage tank

(IRWST) into the spreading compartment. The compartment floor is covered by a sacrificial concrete layer that protects a cooling structure against thermal loads resulting from melt spreading. The cooling elements form a series of parallel cooling channels that serve as flow paths for water from the IRWST to flow under the melt, along the sidewalls and onto the top of the molten core debris, cooling and stabilizing the melt.

The SAHRS is a dedicated single-train, non-safety related, thermal-fluid system used to control the environmental conditions within the containment following a severe accident. There are four primary modes of operation, including:

- Passive cooling of molten core debris in the spreading compartment,
- Active spray for environmental control of the containment atmosphere,
- Active recirculation cooling of the molten core debris and containment atmosphere, and
- Active back-flush of the IRWST.

During the passive cooling process, water covering the core debris would boil off as steam and be released into the free volume of the containment through the steam chimney directly above the spreading compartment. As this process continues, the temperature and pressure within the containment would steadily increase, until the SAHRS is configured to operate in the containment spray mode. Active recirculation cooling of the CMSS would occur once the containment spray has sufficiently reduced containment pressure. In this mode of operation, the water level in the spreading compartment would rise to the top of the steam outlet chimney, overflow onto the containment floor, and drain back into the IRWST, where it could recirculate back into the spreading area cooling system.

It is evident that the CMSS method of assuring debris coolability is fairly complex, involving both passive and active cooling modes. A properly-functioning CMSS would keep the debris cool, and prevent sustained concrete ablation in the core spreading room. It definitely would be addressed in severe accident management procedures. The active spray and recirculation cooling modes of a properly-functioning SAHRS would effectively act to keep the pressure in the containment well below the ultimate pressure. Together, the two systems would act together to significantly reduce the LRF in the U.S. EPR and contribute to a successful accident management strategy.

High-pressure core melt ejection. The U.S. EPR design includes two dedicated severe accident depressurization valve trains, each of which consists of a DC-powered depressurization valve in series with an isolation valve connected to the pressurizer. The objective of this design is to convert high-pressure core melt sequences into low-pressure sequences, so that a high-pressure vessel breach can be excluded. The operator would actuate these valves when the core exit temperature exceeds 1200°F (829K). The anticipated loads within the reactor cavity in the event of successful RCS depressurization (i.e., pre-vessel breach RCS pressure) would be well below the reactor cavity design load. Timely operation of the depressurization valves is part of the AM strategy, and is also very important to avert possible induced ruptures of damaged steam generator tubes.

Even if the RPV would fail at high pressure, the pathways for the melt and aerosols dispersed through the reactor cavity cooling ventilation ducts are expected to be tortuous, causing entrainment and de-entrainment, and consequent significant reduction in materials entering the upper containment. It should be noted, however, that deposition of these materials in pump rooms and steam generator compartments is possible; this could affect AM strategies.

Containment bypass. The U.S. EPR design strategy for reducing potential radioactive release in an SGTR is based on having the medium head safety injection pump shut off head at a pressure below the steam generator safety relief valve set point. As a consequence, the likelihood of a SGTR progressing to containment bypass due to secondary system pressure increasing enough to open a safety valve and fail to reseat has been significantly reduced. Automatic isolation of the affected steam generator on its high level signal coincident with the end of partial cool down prevents overfilling and limits liquid release to the environment. No operator actions are required to mitigate the SGTR accident, and the secondary system remains sealed against releases to the environment after the relief valve or its block valve is closed. The subsequent plant cool down is accomplished using the remaining three intact loops.

US-APWR design. The US-APWR design is under review has not yet been certified by the NRC. One COL application is currently being reviewed.

Combustible gas control. The US-APWR containment has a hydrogen gas control system to avoid the risk of containment failure due to fast deflagration or accidental ignition of a critical gas mixture. For controlling hydrogen generated during a severe accident, a number of hydrogen igniters are provided.

Containment response would be monitored to ensure that the pressure loads resulting from the accumulation and combustion of hydrogen could not exceed the containment ultimate capacity pressure limit. To provide reasonable assurance that structural integrity is not compromised, the containment is qualified to withstand global hydrogen deflagration, and flame acceleration.

Although the igniters control the combustible gas concentration, certain scenarios involve the failure of the containment spray system (CSS). In this case, the containment atmosphere may become steam-inerted for some period of time. Condensing the steam upon CSS recovery may lead to combustible conditions. If the hydrogen monitoring system detects high concentrations of combustible gases, the sprays would be turned off.

External reactor vessel cooling (ERVC). In-vessel retention of core debris by external RV cooling is considered as effective potential mechanism for severe accident mitigation. Various physical phenomena related to severe accidents such as steam explosions and CCI, which are the consequences of a result of core debris relocation to the reactor cavity, would be prevented by attaining in-vessel retention. Since the US-APWR is designed to fill the reactor cavity with coolant water when a severe accident occurs, ERVC may be possible. However, in-vessel retention is not credited for the US-APWR severe accident treatment or in the Level 2 PRA study due to its inherent uncertainty.

Flooding of the reactor cavity would occur be manually initiated when core damage is detected, provided the water level is below a certain level. The objective would be to cool down molten debris in the cavity after vessel failure.

Core debris coolability. The US-APWR design includes a large area in the reactor cavity to provide floor space for debris spreading and quenching capability to cool the debris. The design would provide retention and long-term stabilization of the molten core debris inside the containment. It has been calculated that the core debris would spread to a depth of between 7 and 10 inches. The melt would be cooled by the water from two independent sources: the in-containment reactor water storage pit (RWSP) by manually activating containment spray; and fire protection water supply. There would be no cooling from below. Water would flow into the cavity through a drain line. In order to utilize the fire water service system for reactor cavity flooding, it is necessary to establish lineup before activating the fire water service pump.

High-pressure core melt ejection. The US-APWR severe accident dedicated depressurization valves (DVs) design consists of a flow path with two redundant motor-operated remote manual valves connected in series. Non-condensed gas or steam is directly discharged to the containment vessel. The valve arrangement with two normally-closed valves in series minimizes the possibility of inadvertent actuation. The motor-operated valves are controlled from main control room.

The objective of this approach is to convert high-pressure core melt sequences into low-pressure sequences, so that a HPME can be excluded. Timely operation of the DVs is part of the AM strategy, and is a key action to avert possible induced ruptures of damaged steam generator tubes.

Containment bypass. The RCS depressurization feature would act to reduce the probability of temperature-induced SGTR. The capacity of the depressurization valve is considered to be sufficient to reduce RCS pressure for preventing temperature-induced SGTR.

ABWR design. The ABWR design has already been certified by the NRC and one COL application is currently being reviewed.

Combustible gas control. Just as for existing BWRs, the ABWR containment would be inerted during full-power operation. Consequently, insufficient oxygen would be present to cause a hydrogen burn during a severe accident.

Core debris coolability. Numerous features are incorporated into the ABWR design to help mitigate the effects of CCI. The most important are: a large lower drywell floor area with minimal obstructions to the spreading of core debris; a lower drywell flooder (LDF) system; an ac-independent water addition (ACIWA) system; use of sacrificial basaltic concrete for the lower drywell floor; a thick reactor pedestal wall; and a Containment Overpressure Protection System (COPS). The LDF consists of ten piping lines from the suppression pool to the lower drywell, with thermally activated flooder valves attached to them. The thermally activated flooder valves open when the LDW air temperature reaches 260 °C (500 °F), which would be soon after the core debris enters the LDW. The time delay would effectively eliminate energetic steam explosions.

Injection to the reactor vessel using the ACIWA system is intended to prevent core damage. In the event that it is not initiated in time, and reactor vessel melt-through occurs, the ACIWA would provide water to the lower drywell through the breach in the reactor vessel to assist in cooling ex-vessel core debris. This flooding of the lower drywell could be in addition to or in-place of the flooding provided by the LDF. The actual circumstances are accident-sequence specific.

Operation of the ACIWA in the containment spray mode controls atmospheric temperatures in the upper drywell and provides fission product scrubbing. This system is very beneficial in delaying the time to or preventing the opening of the COPS. The COPS passively relieves containment pressurization before containment pressure reaches ASME Service Level C limits. This system provides for a controlled release through a containment vent pathway with fission product scrubbing provided by the suppression pool. The COPS would also prevent catastrophic overpressurization failure of the containment for severe accident sequences involving prolonged periods of CCI.

High-pressure core melt ejection. The probability of HPME is significantly reduced by the highly reliable ADS. In addition, the following ABWR containment design features mitigate the possible effects of high-pressure core melt:

- The containment is segregated into upper drywell (UDW) and LDW regions that communicate directly. However, the design mitigates the ability of high-pressure core melt, ejected within the lower drywell, to reach the UDW.

- Once the horizontal vents have been cleared, the gas and debris leaving the lower drywell would split into two paths: one to the upper drywell and the other to the suppression pool.
- The inerted containment prevents pressurization from combustion of hydrogen generated from oxidation of the metallic constituents of the core debris.

The NRC staff approved this approach because the ADS system is to be provided with a reliable nitrogen supply and dc power to ensure its operability, and the containment design would sufficiently reduce the amount of core debris that would reach the upper drywell. The staff concluded that the criteria of SECY-93-087 would be met.

Severe accident management insights from NRC confirmatory assessments

A complementary activity to NRO's severe accident evaluation is confirmatory assessment by the NRC's Office of Research (RES). Severe accident scenario simulations are done using the MELCOR code, and the results are compared against the MAAP simulations. These results are shared with NRO, and insights obtained from these calculations are factored into the Safety Evaluation Reports (SER) prepared by for each design. Some insights from the confirmatory assessments pertaining to the prevention and mitigation features are provided below in the context of the AM technical basis for each design. Although a number of other examples are described, the focus is on core debris coolability because it is a concern for all reactor designs.

AP1000 insights from confirmatory assessment

Ex-reactor vessel retention. Confirmatory analyses were performed by the staff to assess lower head thermal behavior under severe accident conditions. The following configurations were evaluated:

- Configuration I: a molten ceramic (oxide) pool above a molten metallic layer; and
- Configuration II: a molten ceramic pool sandwiched between a bottom heavy metallic layer and an overlying metallic layer

These configurations are considered bounding in terms of their impact on the lower head integrity.

For Configuration I, a thin top metallic layer could form that is that could cause significant focusing of heat onto the RPV wall. For a low ceramic pool mass, the lower core support plate would not be submerged, and the amount of steel in the metallic layer would be limited, resulting in increased heat fluxes to the RPV wall. For higher ceramic pool masses, the core support plate would be submerged, resulting in a thick metallic layer and reduced heat fluxes to the RPV wall. Results show that the critical heat flux (CHF) would not be exceeded within the molten oxide region. However, the probability of exceeding CHF is about 0.15 within the metallic layer region.

For Configuration II, parametric calculations were performed using point estimate mean values of the masses from Configuration I. The mass fraction of uranium in the bottom layer was fixed at 0.4, and had a density greater than that of the oxide layer, consistent with this configuration. The results of these calculations indicate that the heat fluxes from the vessel remain below CHF at all locations. Thus, the vessel would not be expected to fail if partitioning of the heavy metals from the ceramic pool were to occur.

The applicant did not consider Configuration I to be applicable to the AP1000 because its analyses indicated that the lower plenum debris pool would contact the lower support plate and create a thick metal layer, and in the transient stages before the debris contacts the lower support plate, the debris would be either water cooled or quenched rather than a fully developed naturally circulating pool. For Configuration II, the applicant provided an analysis that produced results similar to those from the staff analysis, and concluded that RPV failure would be physically unreasonable.

The staff concluded that the applicant's position was not justified in light of the uncertainties in the late-phase melt progression and the melt configuration in the lower head. Nevertheless, the probability of vessel failure was judged to be small, and this assumption is inconsequential from the overall risk perspective. From an accident management perspective, the consequences of a breach of the RPV must be taken into account.

Debris coolability. The applicant performed deterministic calculations of CCI for a postulated vessel breach event using MAAP4 for two different reactor cavity/basemat concrete compositions, i.e., limestone/common sand and basaltic concrete. With limestone concrete (for which noncondensable gas generation is maximized and concrete ablation minimized), basemat penetration would occur after about 3 days following the onset of core damage. With basaltic concrete (which maximizes concrete ablation and minimizes noncondensable gas generation), the predicted time of basemat melt-through is reduced to about 2 days, with containment overpressure failure expected some time later.

The staff performed calculations using the MELCOR code to confirm the degree of basemat ablation. The calculations indicate a maximum ablation depth of about 1.3 m (4.3 ft) for both limestone and basaltic concrete 2.5 days or more after accident initiation, assuming a dry reactor cavity and uniform distribution of debris within it. The ablation rates predicted by MELCOR are considerably lower than those predicted by MAAP4, partially as a result of a later time of RPV failure in the MELCOR calculation (8 hours in MELCOR versus 2 hours in MAAP). While not directly comparable to the applicant's calculations, the MELCOR calculations support the applicant's finding that basemat penetration would not occur for several days.

The staff concluded that if core debris were not retained in the vessel, the AP1000 design would still provide adequate protection against early containment failure and large releases resulting from CCI. In short, the AP1000 incorporates features that adequately address the guidance called out in SECY-93-087 related to core debris coolability.

ESBWR insights from confirmatory assessment

An independent assessment of the ESBWR design response to selected severe accident scenarios was performed using the MELCOR 1.8.6 computer code. The assessment examined 13 accident scenarios from the ESBWR PRA, most of which were simulated by the applicant using MAAP 4.0.6. The results generally support and confirm the PRA accident progression analysis methodology and the applicant's interpretations of its severe accident analyses.

Core debris coolability. The applicant provided the results of sensitivity studies using MAAP 4.0.6, given a depressurized RPV, performed to estimate concrete ablation for both limestone and basaltic concrete to assess the potential for RPV pedestal failure. The calculated times from RPV failure to pedestal failure ranged from 26 hours (dry LDW with basaltic concrete) to 55 hours (dry LDW with limestone concrete), to beyond 72 hours (either limestone or basaltic concrete in a flooded LDW). An independent assessment of CCI using MELCOR 1.8.6 confirmed that concrete ablation depths in the axial direction would be of similar or somewhat smaller magnitudes than those predicted by MAAP 4.0.6. The staff believes that, while it is possible that a horizontal "blowout" may occur into the lower reactor building somewhat before 20 hours because of local thinning of the pressure boundary in the region of the BiMAC trough, further analysis of this event is of questionable value given the very low probability of a dry CCI event. It is reasonable to assume that the containment would fail from over-pressurization before basemat melt-through or pedestal failure.

U.S. EPR insights from confirmatory assessment

Extensive MELCOR confirmatory calculations were performed using the MELCOR 1.8.6 computer code to analyze the representative accident scenarios identified by the applicant and simulated using MAAP 4.0.7. Insights from some of these are discussed below.

Combustible gas control.

Generally, the MAAP-predicted in-vessel hydrogen generation was higher than the MELCOR predictions. This was attributed, for the large part, to the conservative enhancement of the oxidation rate as modeled in MAAP. However, ex-vessel hydrogen production, in the absence of passive flooding of the containment, was reported to be higher in MELCOR, even though the concrete erosion rate was lower than the MAAP prediction. For either mode of hydrogen production, both MAAP and MELCOR results showed that hydrogen concentration in the containment to remain low due to the effective recombination of hydrogen and oxygen by PARs.

MELCOR calculations for the representative accident scenarios have confirmed that, due to efficient recombination by PARs, there is little potential for formation of pockets of high hydrogen concentration inside the EPR containment and hence deflagration or detonation is unlikely.

Molten fuel/coolant interactions.

The time duration from vessel breach to reactor pit melt plug failure was found to be much shorter in MELCOR as compared to MAAP predictions, even though the MAAP-predicted debris temperature in the reactor pit before the melt plug failure was found to be 500 °K to 600 °K higher than the MELCOR predictions. The MELCOR results showed that the entire core debris content may not be in the reactor pit by the time the reactor melt plug melt-through occurs. Nonetheless, the mass of any remaining core debris arriving into the reactor pit after melt plug failure was calculated to be small (~5%). This delayed relocation can have implications in terms of ex-vessel energetic fuel coolant interactions, which the EPR design of the reactor pit was intended to circumvent.

Core debris coolability. Differences in MELCOR and MAAP prediction of concrete erosion were found with the MELCOR-calculated erosion rate being lower than that of the MAAP prediction. For most of the scenarios examined, the MELCOR-predicted debris temperature in the spreading compartment was shown to be lower due to lower initial debris temperature as compared with MAAP.

Provided that full uniform spreading of the melt over the floor of the spreading compartment occurs, and provided that IRWST passive injection is initiated as designed, melt cooling and stabilization were predicted to take place by both MAAP and MELCOR. Nonetheless, for most of the scenarios examined, the predicted core debris cool-down rate on the spreading compartment floor was shown to be faster in MELCOR as compared with MAAP.

US-APWR Insights from Confirmatory Assessment

A number of severe accident scenarios were simulated for the US-APWR using the MELCOR 1.8.6 computer code. A total of six representative scenarios were selected, based on the accident sequences described in the US-APWR PRA. Two additional calculations were also performed to examine the effect of vessel depressurization as a result of creep rupture of the hot leg. The results of the MELCOR calculations were compared to MAAP simulations documented in the US-APWR PRA. Differences in the modeling approach and assumptions in the initial and boundary conditions, resulted in variations in the details of the accident progression. Overall, the timing of accident progression and failure of the lower head were in relatively good agreement given the uncertainties in severe accidents. A specific comparison on debris coolability is briefly described below.

Core debris coolability. Results of accident progression analyses using MAAP 4.0.6 for selected representative accident sequences, in which both features of the diverse reactor cavity flooding system are available, indicate that molten debris is appropriately cooled down in a reactor cavity water pool and no concrete erosion occurs for accident sequences in which molten debris drops into water pool. Results of accident progression analyses for characteristic accident sequences, in which no continuous reactor cavity flooding means is available, indicate that the earliest possibility of complete erosion of the reactor cavity floor concrete (i.e. more than 40 in. erosion of concrete) is approximately 28 hours after onset of core damage.

MELCOR predicts that cavity ablation can be averted as long as water remains in the lower cavity region. The core concrete interaction shows that after vessel breach and relocation of the core to the cavity, the core debris is cooled by the cavity water and reaches a near steady state, where the decay heat generation is balanced by the heat loss to the cavity water. As soon as the cavity water is boiled off, ablation of the 40 inch basemat results in the failure of the containment pressure boundary. Basemat melt through does not occur before 24 hours following the initiation of the accident. Thus, the two simulations are in good agreement with respect to core debris coolability.

ABWR Insights from Confirmatory Assessment

Core debris coolability. General Electric carried out analyses for each accident sequence using a very early version of the MAAP code, adapted to be able to simulate the ABWR. These analyses generally indicate core debris coolability and little, if any, CCI if LDW flooding would occur. The time-to-release of fission products ranged from 8.6 to 50 hours from the start of the transient, with the most likely fission product release location through the containment overpressure protection system (COPS). In most cases the time to fission product release exceeded 24 hours. For cases where the LDW would not be flooded and containment heat removal is lost, the time to COPS initiation would be less – sometimes less than 24 hours after accident initiation.

The staff analyzed in-house the response of the ABWR using the MELCOR code. In addition, the staff performed additional analyses using the MELCOR code. The MELCOR results generally reproduced the event sequences predicted by MAAP, albeit usually with timing shifts. These timing shifts did not affect the safety insights for the containment analyses.

Conclusions

The design application reviews, both complete and ongoing, are confirming that the new reactors will be safer by including the new severe accident mitigation systems that address the concerns expressed in SECY-90-016 and SECY-93-087. The PRAs all calculate significantly lower core damage frequencies and large release frequencies than for existing reactors. All of the applicants claim that the new regulatory requirements emanating from SECY-90-016 and SECY-93-087 will be met. Both the preparation of the design certification applications by the applicants and the technical reviews by the NRC staff are revealing insights on how the use of these design features will enhance the technical basis now in place for the existing reactors, so that appropriate accident management procedures will be put in place.

Session 3

Some International Efforts to Progress in the Harmonization of L2 PSA Development and Their Applications (European (ASAMPSA2), U.S.NRC, OECD-NEA and IAEA activities)

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1. Introduction - Some views on international contexte

Most of the existing Nuclear Power Plants (NPPs) are designed with the principles of defence-in-depth and incorporate a strong containment and engineering systems to protect the public against radioactivity release for a series of postulated accidents.

Nevertheless, in some very low probability circumstances, severe accident sequences may result in core melting and plant damage leading to dispersal of radioactive material into the environment and thus constituting a health hazard to the public.

A major issue for all stakeholders is to keep the probability of such circumstances as low as possible and in addition to have implemented appropriate accident management measures allowing an efficient limitation of the consequences of such events.

Following the initial US effort in the 80's, in most countries, level 1 and level 2 Probabilistic Safety Assessment (L1 and L2 PSA) have now been developed for the existing and future plants and are used to demonstrate that the probability of occurrence of a severe accident is low enough and that, if such an accident occurs, all reasonable provisions are taken to limit the consequences. These studies, updated in function of plant modifications, new knowledge and scope extension, contribute to the continuous improvement of plants safety, while identifying remaining dominant risks.

Nevertheless, regarding the severe accident phenomenology, the remaining uncertainties, and also the diversity of accident scenarios considered, the development of L2 PSA is still a very complex activity often conducted by rather small teams. In parallel, the expectation of these studies may be large, for example:

- validation of severe accident measures (SAM),
- achieving safety goals or acceptability of the level of risk,
- cost-benefit analysis,
- support for decision regarding plant life extension,
- identification of R&D needs for closing issues,
- capitalization of knowledge,
- emergency preparedness ...

Such expectations require robust and validated studies. But one should recognize that, in some cases, discrepancies may exist between the real quality of the L2 PSAs (regarding the complexity of the different issues) and the expected applications. For that reason, the L2 PSAs are generally used very carefully in their applications.

In that context, there is still a need in the international accident management community to share experience in the development and the application of L2 PSA. The development of standards, best-practice guidelines, and state-of-the art methods is a useful way for allowing experts to share their experiences and to formalize some best-practices.

This paper aims at providing some information on current or recent initiatives regarding the harmonization of the assessment of severe accident risks and its applications. Initially dedicated to the European Advanced Safety Assessment Methodologies: Level 2 PSA (ASAMPSA2) project, it has been fruitfully extended to the initiatives of the International Atomic Energy Agency (IAEA), the Nuclear Energy Agency (NEA) and the US Nuclear Regulatory Commission (US NRC), allowing some broader views on this topic.

2. Ongoing activities within the European Framework Programmes

2.1. SARNET (Severe Accident Research NETwork of Excellence)

In the European context, the Severe Accident Research NETwork of Excellence (SARNET[1]) gathers a large part of activities concerning severe accident issues. A first project SARNET was initiated in 2004 with 51 organisations involved in severe accident research in Europe plus Switzerland and

[1] FISA 2009 – Sustainable integration of EU research in severe accident phenomenology and management – JP. Van-Dorsselaere (IRSN) & al.

Canada. A second project, named SARNET2, gathers, from 2009, 41 organizations from 21 countries (Europe plus Switzerland, Canada, USA and Korea).

SARNET2 includes some integration activities (database on experimental results, integration of knowledge in the ASTEC IRSN-GRS integral code, extension of ASTEC to BWR and CANDU reactors), spreading of knowledge (conferences, seminars, courses, encouragement of exchange of students …) and some research activities considered with a high level of priority (in-vessel core coolability, ex-vessel melt-pool configuration during molten core concrete interaction (MCCI) and coolability by top flooding, hydrogen mixing and deflagration/detonation in containment, melt relocation into water and fuel coolant interaction (FCI), ruthenium behaviour and iodine chemistry in reactor cooling system and containment).

Activities concerning L2 PSA were performed within the first project in 2004-2008 (general methodology, uncertainties assessment and dynamic reliability methods[2]) and have been used to define and initiate the ASAMPSA2 project of the 7th Framework programme that is described hereafter. Technical exchanges between SARNET and ASAMPSA2 will continue in particular on the update of the knowledge of the severe accident physical phenomena and management measures and on the L2 PSA requirements for computer codes such as ASTEC.

2.2. ASAMPSA2 (Advanced Safety Assessment Methodology : level 2 PSA)

a) Objectives and context

Within the European community responsible for fission reactor safety (plant operators, plant designers, Technical Safety Organizations (TSO), Safety Authorities), a need to develop best practice guidelines for the level 2 PSA methodology has been repeatedly expressed, with the aim of both fulfilling the requirements of safety authorities in an efficient way, and also promoting harmonization of practices in European countries in order to use results from level 2 PSAs with a greater confidence.

Many existing guidelines, like those developed by the IAEA, propose a general stepwise procedural methodology, mainly based on the US NRC's NUREG-1150 study and high level requirements (for example on assessment of uncertainties). While it is clear that such a framework is necessary, comparisons of existing level 2 PSA, performed and discussed in the SARNET L2 PSA group, have shown that the detailed criteria and methodologies of current level 2 PSAs strongly differ from each other in some respects. Currently in Europe integration of probabilistic findings or insights into the overall safety assessment of NPPs is quite differently understood and implemented.

Within this general context, the project objectives are to highlight common best practices, develop the appropriate scope and criteria for different level 2 PSA applications, and promote optimal use of the available resources. Such a common assessment framework will support some harmonized view on nuclear safety, and help formalize the role of probabilistic safety assessment.

A common assessment framework requires that some underlying issues are clearly understood and well developed. Some important issues are the following:

- the PSA tool should be fit for purpose in terms of quality of models and input data,
- the scope should be appropriate to the life stage (e.g. preliminary safety report, pre-operational safety report, living PSA) and plant states (e.g. full power, shutdown, maintenance) considered,
- the objectives, assessment criteria and presentation of results should facilitate the regulatory decision making process.

[2] ERMSAR Nesseber, Bulgaria, September 23-24, 2008, L2 PSA methods harmonization, B. Chaumont, E. Raimond & al.

The main characteristic of this coordination action is to bring together the different stakeholders (plant operators, plant designers, TSO, Safety Authorities, PSA developers), regardless of their role in the safety demonstration and analysis: this should promote some common views and definitions for the different approaches for L2 PSA.

The project started at the beginning of 2008 for 3 years and gathers 22 organizations from 13 European countries. IRSN coordinates the project. It is mainly focused on BWRs and PWRs of Gen II and III, but includes also a small extension on Gen IV reactors.

b) Structure of the ASAMPSA2 best-practices guidelines

The aim of the coordination action is to build a consensus on the L2 PSA scope and on methods deemed to be acceptable, according to the different potential applications. In any methodology, and especially one developed from a wide range of contributing perspectives, there will be a range of outcomes that are considered acceptable.

To represent this range, the project initially tried to distinguish between a 'limited-scope' methodology and a 'full-scope' one, based on what is currently technically achievable in the performance of a level 2 PSA.

In this respect, it was considered that what is technically achievable may not be cost effective, but for the purpose of this project is taken to represent the upper bound of what may be considered 'reasonable'. The notion of 'limited-scope' methodology may correspond to the case where the study is performed to answer some precise question (for example the quantification of LERF), allowing simplification of some parts of the study, and limitation of the needed resources and delay.

The distinction between limited-scope and full-scope methodologies has been widely discussed in the initial phase of the project and the possibility to establish two separated guidelines has been examined. But from a practical point of view, it appeared that many variations in the definition of what is a 'limited-scope study' exist in relation with the different applications. Consequently, the Partners of the project have decided to build a unique guideline including all issues related to level 2 PSA development and applications. For each issue, the different level of details and acceptable methods will be described with some recommendations.

At the end of the project, a correspondence table between the final application of a L2 PSA and the required level of detail or methodology required for each issue will be built if possible.

c) Content

The guideline will be composed of three parts.

The first part will include a general description of L2 PSA content and structure but should mainly discuss the applications of L2 PSA studies conducted by the Partners with comprehensive experience. The project will use (as much as possible) information available on public domain, mainly from other international collaboration initiative, for example on the description of safety criteria. This part is considered to be the most difficult part of the guideline to be established but is crucial because the targeted applications drive the objectives and scope of a L2 PSA.

The second part of the document will contain all technical recommendations gained from the experience of the ASAMPSA2 Partners and external sources. This part will concern the methodological topics (level 1- level 2 PSA interface, Human Reliability Assessment, the event tree structure, the uncertainties assessment …), the quantification of severe accident progression and containment loading, the containment performance (tightness), the plant system behaviour in severe accident conditions and the source term assessment. While establishing this outline, we had to recognize the very large number of issues that may be examined in a L2 PSA. The treatment of each issue with enough details is another difficulty of the project (with limited available resources) but the efficient working plan developed and the current distribution of tasks between the Partners with the related experience will enable a complete coverage of all issues.

The last part of the document concerns the applications for Gen IV reactors, with the objective to describe how far the existing recommendations for Gen II and III reactors L2 PSA may apply for the Gen IV reactors concepts.

d) Relationship with the L2 PSA "End-Users"

In designing the ASAMPSA2 project, the relationships with the L2 PSA 'End-Users' (establishing the needs of the 'End-Users' for the performance of a L2 PSA as well as assuring the acceptance of the guidelines to be prepared at the end of the project by a majority of the 'End-Users') were considered as a key point. A dedicated working group, coordinated by PSI, has been established to help in formalizing these relationships.

These elements were planned in two steps in the project. At the beginning of the project, a survey was conducted to establish more precisely the needs of the 'End-Users' community regarding many aspects of performing a L2 PSA. The results of the survey were discussed during a dedicated workshop, hosted by Vattenfall in Hamburg (Germany) in October 2008. At the end of the project, an external review of the guidelines will be organized to receive the response from the End-Users community. The review will be discussed during a workshop organized by the end of 2010 and the resolutions will be sought to eliminate possible differences in especially key areas. This review, like the initial survey, will be asked from European stakeholders but also from other organizations, especially those members of the OECD CSNI-WG-Risk.

ASAMPSA2 will take in consideration the positions provided by End-Users irrespective of their role (plant operators, plant designers, Technical Safety Organizations (TSO), Safety Authorities). The guideline will try to propose some technical solutions, regarding some high level requirement that may exist in national or international regulation policies. Moreover the project provides an opportunity to discuss these requirements (like probabilistic safety criteria) at a technical level.

e) Some outcomes of the initial End-Users survey

Feedback on the 2008 End-Users survey helped in the identification of some technical issues where harmonization or best-practices are particularly needed, e.g.:

- L1 PSA – L2 PSA Interface: advantages and disadvantages of the integrated and non integrated studies, use of L1 PSA probabilistic tools or dedicated tools for L2 PSA,
- methods for uncertainty assessment (issue by issue, in the event tree, propagation, for results presentation), may depend on the L2 PSA objectives, plant design and may be limited to some relevant issues (the assessment of all uncertainties is not reasonable …),
- the closure of issues in accident progression regarding research activities: in that context, an issue is 'closed' when L2 PSA developers find enough knowledge or validated codes for the assessment of risks (it can be dependent on the plant design),
- the assessment of initial containment leakage, use of historic data (tests), assessment of containment isolation failure …

The End-Users survey also showed that there is a lack of uniformity between the countries in the objectives and applications of L2 PSAs. Only a few EU Safety Authorities have precise safety goals regarding severe accidents, and in general the legislation or rules, when they exist, are not strictly applied. Very few utilities have a voluntary approach for 'risk-informed' application of L2 PSA (Finish utilities as mandated in legislation, EDF recently developed application for periodic safety review). Some utilities may still have an unclear view on how and mainly why to develop a L2 PSA.

It is expected that the project should help in harmonization of technical issues by providing a global (but practical) vision of how the different risks can be assessed within a level 2 PSA taking into account the existing knowledge and codes. It should also help in harmonization on application of L2 PSA, in particular it should help to identify some plant 'risk reduction options'.

f) Link with the international scientific research activities related to severe accidents

A level 2 PSA is mainly based on a set of deterministic studies on the different phenomena related to severe accident progression. A large part of the guideline will concern the way of quantifying each part of the accident progression. For example, the guideline will examine how to quantify the delay before reactor vessel rupture and what uncertainties should be taken into account.

The first draft of the different chapters will gather the methodology currently used by the partners PSA experts and describe some rationale. To improve its final quality regarding the state-of-art for each topic, the guideline will be open for review by specialists involved in the SARNET Network of Excellence or NEA/CSNI/WGAMA members.

g) Link with other existing standards

Others countries, outside the European Union, may have developed such guidance at a technical level and comparison may be very beneficial. The activities of the US NRC, American Nuclear Society (ANS), NEA and IAEA, presented in the next chapter are of course of high interest in relation to the ASAMPSA2 effort.

3. Ongoing NRC activities of interest to the international Accident Management community

The US Nuclear Regulatory Commission (US NRC) has a number of ongoing activities related to Level 2 Probabilistic Safety Assessment and Accident Management which are either performed in collaboration with the international community or are of interest to the international community. Each of these activities is highlighted below, along with any relevant links to other international activities.

The US NRC's State-of-the-Art Reactor Consequence Analyses (SOARCA) project involves the reanalysis of severe accident progression and consequences to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. In addition to incorporating the results of more than 25 years of research, the objective of this updated plant analysis is to include the significant plant safety improvements and updates, which have been made by plant owners but were not always reflected in earlier assessments by the US NRC. In particular, these plant safety improvements include system enhancements, training and emergency procedures, and offsite emergency response. In addition, these improvements include the recent enhancements in connection with security-related events.

The goal of SOARCA is to generate realistic estimates of the offsite radiological consequences for severe accidents at U.S. operating reactors using a methodology based on state-of-the-art analytical tools. These estimates account for the full extent and value of defense-in-depth features of plant design and operation, as well as mitigation strategies implemented in the form of Severe Accident Management Guidelines or other procedures. This project is expected to lead to new opportunities for collaboration with international organizations on the topic of best-estimate consequence assessment, both through the existing Cooperative Severe Accident Research Program (CSARP) and more broadly.

Results of the SOARCA project may also impact the application of deterministic calculations of severe accident behavior and offsite consequences in Level 2 and Level 3 PRA. For example, comparisons of radiological release estimates from SOARCA to those from past analyses that were based on older modelling technology or that incorporated selected conservatisms, illustrate the extent to which these results impact numerical estimates of risk or revise our understanding of the characteristics of accident sequences that impact offsite radiological consequences.

In the US, a consensus standard exists for the application of an at-power Level 1 and limited Level 2 (large early release frequency - LERF) probabilistic risk assessment (PRA)[3] for internal and external hazards for light-water reactors. The US NRC's position on this standard is articulated in Regulatory Guide 1.200[4]. There are three additional light-water reactor standards that are under development that are of interest to the Accident Management community. These involve low power shutdown PRA, Level 2 PRA, and Level 3 PRA. The second item is the focus of the discussion here. This standard is being developed to provide requirements for a full (as opposed to a limited, e.g., LERF) Level 2 PRA. The standard is intended to integrate well with the existing Level 1/LERF standard as well as the Level 3 standard under development. This means that Level 1/2 and Level 2/3 interface issues are being addressed. The standard is also intended to be applicable to both existing and advanced light-water reactors, and will accommodate the differences in the Level 2 PRA risk surrogates used for each type. The target date for a draft of the new Level 2 standard is late 2009. Subsequent to its issuance, the US NRC will issue supporting implementation guidance. This activity shares some commonalities with other recent and ongoing international activities such as the European Commission ASAMPSA2 project described above and the IAEA Safety Guide 393, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants."

The US NRC is also participating in an ASME-led effort aimed at developing a PRA standard for advanced non-light water reactors. This standard is intended to cover Level 1, Level 2, and Level 3 PRA for all potentially significant onsite sources of radioactivity, and for all potentially significant initiators and hazards.

The US NRC is also reviewing a number of applications for design certification and combined license for advanced light-water reactors. These reviews include deterministic severe accident analysis, probabilistic severe accident mitigation design alternative (SAMDA) analysis, and Level 2 PRA development. These review activities are discussed further in a separate US NRC submittal to this same workshop. In addition, the US NRC is developing the necessary guidance for operational oversight of these new reactors, including the risk metrics (e.g., large release frequency) and target values to be used. In the initial stages of developing this guidance, consideration has been given to the risk metrics used in other countries. Interaction with external stakeholders has been a focus of this effort. The US NRC is also interacting with the international community on new reactor issues through the Multinational Design Evaluation Program (MDEP), such as the series of meetings being conducted by the EPR probabilistic risk assessment and severe accident evaluation sub-groups.

The other ongoing activities for the NRC in the area of accident management concern (i) accident management issues for operating reactors, (ii) severe accident mitigation alternatives (SAMA) analyses for license renewal, and (iii) development of advanced Level 2/3 PRA methods. Each of the latter two items is covered by a separate US NRC submittal to this workshop.

4. Recent OECD/NEA activities

Many collaborative actions related to severe accident and L2 PSA are conducted through the OECD/NEA, especially by the CSNI Risk and GAMA working group members. The present paper

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[3] ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 : Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers, February 2009.

[4] Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," US Nuclear Regulatory Commission, March 2009.

provides the opportunity to relay some of the recent references that may be of key importance for the development of L2 PSAs.

Table 1. OECD references on severe accidents, severe accident management and Level 2 PSA

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NEA/CSNI/R(1997)19R. Documentation on the use of severe accident computer codes in selected level-2 PSAs for nuclear power plants, 1998.
NEA/CSNI/R(1997)18R. Results and insights from level-2 PSAs performed in Germany, Japan, The Netherlands, Sweden, Switzerland, the United Kingdom and the United States, 1998.
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NEA/CSNI/R(2001)20. Implementation of severe Accident Management Measures - Workshop Proceedings - 10-13 September 2001.
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NEA/CSNI/R(2007)12 Use and Development of Probabilistic Safety Assessment A CSNI WGRISK Report on the International Situation.
NEA/CSNI/R(2007)16 Recent Developments in Level 2 PSA and Severe Accident Management.
NEA/CSNI/2007 Technical opinion Paper N°9 - Level-2 PSA for Nuclear Power Plants.
NEA/CSNI/R(2009)3 Ability of Current Advanced Codes to Predict Core Degradation, Melt Progression and Reflooding - Benchmark Exercise on an Alternative TMI-2 Accident Scenario.

Note: **R** at the end of the report code means that the report has a limited distribution.

An important publication is the Technical Opinion Paper (TOP) No.9 (ISBN 978-92-64-99008-1) entitled Level-2 PSA for Nuclear Power Plants.

The CSNI TOPs are short statements giving a summary and a position of WGRISK concerning an important topic, generally written after a State-of-the Art Report or after a Workshop. The level 2 PSA TOP was published in 2007 and its conclusion is remained hereafter.

“The main message of this Technical Opinion Paper is that the Level 2 PSA methodology may now be seen as mature. This is reflected by the large number of high quality analyses that have been performed in recent years and used to identify the potential vulnerabilities to severe accidents and the accident management measures that could be implemented.

The Level 2 PSA is now seen as an essential part of the safety analysis that is carried out for all types of nuclear power plants worldwide. The information provided by the Level 2 PSA is being used by plant operators and Regulatory Authorities as part of a risk informed decision making process on plant operation and more specifically on issues related to severe accident management.

A consistent framework has been established with the development of the individual components of the Level 2 PSA methodology and guidance has been produced by international organisations for carrying out the analysis. In practice, however, there are still differences in the approach and the level of detail in the individual steps that have been carried out in different analyses, partly due to the different objectives that have been defined for these studies. Quality standards and guidelines are currently being developed for Level 2 PSA which should address many of these differences.

The acceptability of the methodology since the early studies in the 1980s is due largely to the significant progress made in the understanding of severe accident and source term phenomenology and in the model development in the current generation of integrated severe accident analysis codes. The research and development activities have continued internationally, albeit at a reduced scale, with emphasis on improving the state of knowledge and providing further data for model validation and improvement.

Further development in Level 2 PSA is likely to see its integration within a Living PSA and its use for risk-informed applications. This requires improvement in the Level 2 PSA methodology in a number of areas, including: the Level 1/ Level 2 PSA interface, the modelling of safety system recovery and human reliability analysis.

The epistemic uncertainty related to some Level 2 PSA issues is regarded as being quite large. The impact of this on risk-informed decision making will also require further consideration of uncertainty treatment in a more integrated manner.

Finally, given the role that integrated severe accident codes (supported by research) have played in the acceptance of Level 2 PSA, future Level 2 PSA research and development activities should be aimed at making these codes play a more central and integral role in the PSA quantification process. Such a shift is likely to alter (and quite possibly diminish) the role of expert judgement and phenomenological event tree modelling in the quantification.

5. IAEA activities

The recent IAEA activities in the area of Severe Accident Management and level 2 PSA are described in a dedicated paper of the workshop. IAEA activities include the development of Safety Standards, the Review of Accident Management Program (RAMP), which is a service provided to Member States, the activity of the International Review Team (IPSART) and the organization of workshops, training courses and technical meetings.

The IAEA Safety Standards Safety Guide[5] “Severe Accident Management Programmes for Nuclear Power Plants” (NS-G-2.15) provides recommendations on meeting the requirements listed in different Safety Requirements publications for the establishing of an accident management programme to prevent and to mitigate the consequences of beyond design basis accidents including severe accidents. The Safety Guide presents the overall concept of an accident management programme and the process of its development and implementation. The established requirements on severe accidents and accident

[5] The IAEA Safety Standards Safety Guides are publications that provide recommendations on different aspects of NPP design and operation. They are governed by the general principles and objectives stated in Safety Fundamentals and safety requirements presented in Safety Requirements publications.

management in the design[6] and in the operation[7] of nuclear power plants, as well as the requirement to determine whether adequate provisions have been made to accident management measures at each of the levels of defence in depth[8] are addressed in the Safety Guide.

The application of PSA techniques in severe accidents is of particular importance due to very low probability of occurrence of a severe accident, but significant consequences resulting from degradation of the nuclear fuel. In order to address the need for standardization of the technical content of PSA, the IAEA develops two new Safety Guides: 'Development and Application of Level-1 Probabilistic Safety Assessment for Nuclear Power Plants' and 'Development and Application of Level-2 Probabilistic Safety Assessment for Nuclear Power Plants'. The Safety Guide on Level-2 PSA among others applications addresses the use of PSA for identification and evaluation of the measures in place and the actions that can be carried out to mitigate the effects of a severe accident after core damage has occurred.

IAEA RAMP service is an activity to support individual Member States with the **R**eview of **A**ccident **M**anagement **P**rogrammes at their plants. Review of the AM program at a particular plant is performed on request by the Member State. The review focuses on the studying of the relevant documents, and interviews with plant staff and regulators. The output of the review is the detailed report with assessment and recommendations for the improvements of the existing **A**ccident **M**anagement **P**rogramme. IAEA has prepared manual supporting RAMP service (IAEA Services Series No. 9) that contains a detailed questionnaire on different related topics (selection and definition of AMP, accident analysis for AMP, assessment of plant vulnerabilities, development of severe accident management strategies, evaluation of plant equipment and instrumentation, etc). Several successful RAMP missions have been already conducted (e.g. Krsko NPP, Slovenia, 2001, Chashma NPP, Pakistan, 2005, Ignalina NPP, Lithuania, 2007), beneficial for the 'fresh look' and in-depth discussions during about one week of missions.

International **P**robabilistic **S**afety **A**ssessment **R**eview **T**eam (IPSART) was established in 1988 and is conducted in accordance with dedicated guideline (IAEA TECDOC 832). Review of PSAs for plants from different countries, of various designs, and all PSA levels, hazard scopes, and operational modes is performed on request by the Member State. Depending on the scope of the PSA, the review duration is from one to two weeks and review team's composition is from four to seven international independent experts, plus an IAEA staff-member. The review focuses on the check of methodological aspects, completeness, consistency, coherence, etc. of the PSA. The output of the review is the IPSART Mission Report that describes the review performed, the review findings, technical aspects of the PSA study, strengths, and limitations and provides suggestions and recommendations for improvement of the PSA quality and its sound use for enhancing plant safety and risk management applications. The IPSART service helps to achieve high quality of PSA and therefore assists in further enhancing nuclear safety. More than 50 IPSART mission have been conducted so far in many countries all around the world, helping to achieve high quality of PSA and proliferating advanced methodology and knowledge in nuclear safety assessment.

The Safety Guide on L2 PSA and applications and the Safety Guide on Severe Accident Management programs are two key references in the perspective of harmonization of practices. The following table provides some other important references.

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6. Conclusions

The present paper presents an overview of some international activities on harmonization of risk assessment regarding postulated severe accidents at NPPs.

This overview shows clearly that these harmonization activities appear useful within a perspective of continuous plants safety improvement in all countries, especially for existing plants which are subject in many countries to life extension programs. It shows also that this harmonization can progress at different levels:

- on high level requirements as provided in IAEA standards,
- on recommendations that support high level requirements as provided in IAEA Safety Guides,
- on the fundamental analysis of the severe accident phenomena as provided within SARNET activities, some OECD projects like SERENA or through the development and the validation of the severe accident codes,
- through the comparison and sharing of experience in L2 PSA development and applications allowing, for example, the drafting of the state-of-the-art report (by OECD CSNI/WG-Risk),
- through the development of L2 PSA best-practice guidelines or standards as conducted today within the EC ASAMPSA2 project and also by the American Society of Mechanical

Engineers and the American Nuclear Society; it offers a structured framework to discuss in detail how to make the best use of existing knowledge and codes for the quantification of risks,

- through international review services aimed at proliferating advanced methodology and knowledge in nuclear safety assessment (RAMP, IPSART).

Authors deem that activities at each level are ultimately useful and help stakeholders to make risk assessments more robust, and to identify or confirm plant risk reduction options and severe accident measures.

ACCIDENT MANAGEMENT AND RISK EVALUATION OF SHUTDOWN MODES AT BEZNAU NPP

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ABSTRACT

Shutdown states represent a significant risk contributor for light water reactors (LWRs) according to several shutdown PSA studies performed. Therefore, Accident Management and risk evaluation of shutdown modes becomes an important safety issue for LWRs.

The paper presents several Accident Management features that were implemented at Beznau Nuclear Power plant to improve shutdown safety. It also shows how the existing Severe Accident Management Guidelines (SAMGs) for full-power operation were extended to shutdown modes.

In addition, a more realistic evaluation of shutdown risk is presented. First it is shown that core degradation does not occur at start of core uncovery but at a reactor water level dropped much lower. This extends the time window for the operators to intervene in cases of loss of core cooling. It also enables operator actions according to the Severe Accident Management Guidelines (SAMGs) to restore core cooling during shutdown conditions even after the start of uncovery. Due to the long time window, recovery of core cooling is possible using mobile equipment such as fire water pumps.

The paper describes how such recovery actions were implemented at KKB and how these actions were modelled in the shutdown PSA study. In addition, an approach to perform a simplified Level 2 PSA for shutdown conditions is presented. This approach enables to evaluate the Level 2 risk for all modes of operation for all initiating events.

Finally, conclusions are drawn with respect to risk of shutdown modes at Beznau NPP.

Key Words: Shutdown Accident Management, shutdown Level 2 PSA, realistic shutdown risk.

1. INTRODUCTION

In light water reactors (LWRs), conditions during shutdown differ significantly from those during power operation and as a result pose a different set of safety-relevant questions. Some aspects are more favourable during shutdown conditions, such as the low core decay heat and the large time windows for recovery actions. However, certain other aspects of the defence-in-depth concept are more vulnerable during shutdown conditions, such as the reduced water inventory, the absence of automatic actuations, the need for permanent decay heat removal, and the open containment hatch. As a result, several PSA studies have shown shutdown conditions to be significant risk contributors for LWRs.

This paper presents several Accident Management measures that were implemented at Beznau Nuclear Power Plant (KKB) to improve shutdown safety. In addition, a more realistic evaluation of shutdown risk including an extension to a Level 2 PSA study is presented.

Section 2 includes a short description of the Beznau NPP and its Accident Management capabilities, especially during shutdown. Section 3 illustrates the refinements performed to the Beznau shutdown PSA study to more realistically evaluate the risk of shutdown modes and to model all possible Accident Management recovery actions. In addition, an approach to perform a simplified Level 2 PSA for shutdown conditions is presented. This approach includes using a simplified Shutdown Accident Management and Containment Event Tree of only few nodes and linking it with the Level 1 PSA model. Section 4 illustrates the actual results of the Beznau shutdown Level 2 PSA study. Finally, conclusions are drawn with respect to the most important risk contributions of shutdown conditions at the Beznau NPP.

2. BEZNAU PLANT AND ACCIDENT MANAGEMENT CAPABILITIES

2.1. Beznau Plant and Hardware for Accident Management

The Beznau plant is located in Northern part of Switzerland and consists of two Westinghouse pressurized water reactors of early design. Unit 1 went into operation 1969 and is actually the oldest operating PWR worldwide. The net power output per unit is 364 MWe. The plant was backfitted in the early 1990s by the construction of additional safety systems. In addition, several Accident Management measures were implemented at the Beznau plant. Examples are the construction of a containment filtered vent system, the installation of passive autocatalytic hydrogen recombiners and the installation of several fire water connections for injection using mobile pumps. The main injection capabilities using mobile pumps are illustrated in Figure 1.

Using these capabilities, water can be injected by mobile pumps to the following locations:

- into the steam generators
- to emergency feedwater tank
- injection of borated water from the spent fuel pit into the reactor
- injection of unborated water into the reactor
- to the containment sump
- into the containment spray line
- to containment fan coolers
- to alternate spent fuel pit cooling train
- into the spent fuel pit.

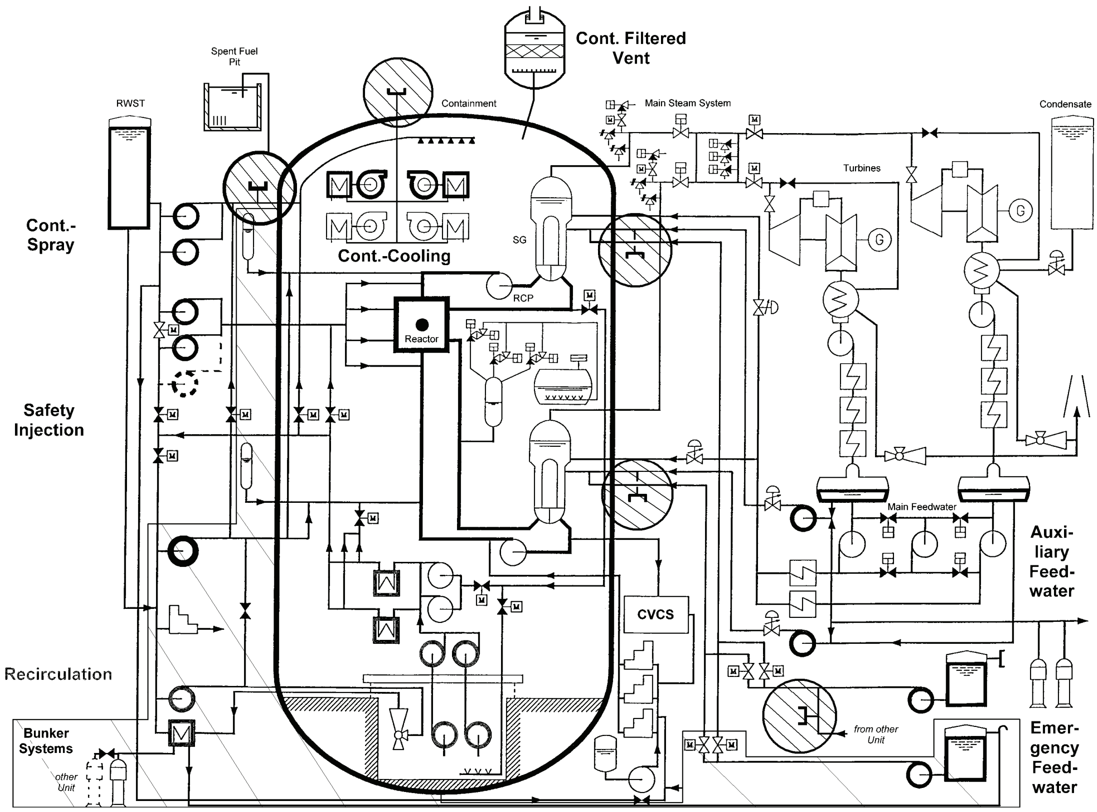

Figure 1. Beznau NPP Accident Management Capabilities

Most of these injection capabilities can be used to prevent core damage as well as for the mitigation of severe accidents. In addition, the alignment of these pumps requires coordination between at least two different emergency crews: operating crew and fire brigade. Therefore, the alignment of mobile pumps is guided by a Beznau specific set of procedures, the so-called Accident Management Procedures (AMPs).

2.2. Responsibilities and Procedural Guidance

The main responsibilities and the procedural guidance for accident management actions during different stages of accident severity are illustrated in Figure 2.

All actions to prevent core damage are included in the Emergency Operating Procedures (EOPs; [1]) and are performed under the main responsibility of the operating crew. During this phase of accident severity, the emergency staff has only some technical responsibility of surveillance and to give guidance to the operating crew if questions arise from the EOPs to the emergency staff.

If an accident exceeds the limit of core degradation, the main technical responsibility is shifted to the emergency staff which then uses the Severe Accident Management Guidelines (SAMGs, [2]) as the leading document. The decisions of the emergency staff are executed by the operating crew.

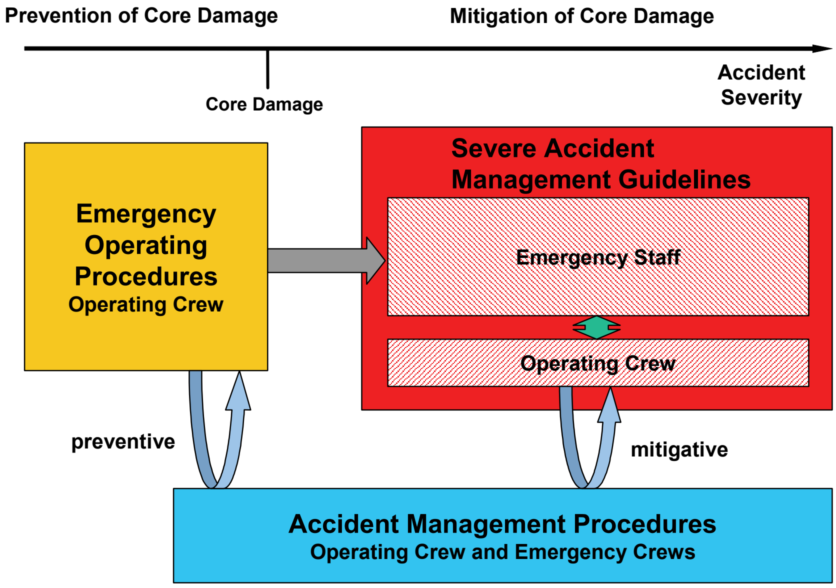

Figure 2. Beznau Accident Management Procedural Guidance and Responsibilities

Actions to align mobile equipment can be used for prevention as well as for mitigation of core damage. These actions are included in specific Accident Management Procedures (AMPs [3]). Their implementation requires actions of at least the operating crew and one additional emergency crew, for example the fire brigade. The AMPs can be called from either the EOPs by the shift crew or from the SAMGs by the emergency staff. During execution of the AMPs, the coordination between the different emergency crews is performed by the operating crew.

2.3. Special Features for AM during Shutdown

The fire water connections to feed into the reactor using mobile pumps can be used for Accident Management in any plant operating state. However, these measures are suitable especially during shutdown states due to the following reasons:

- Longer time window during shutdown than after a reactor trip
- Low RCS pressure during shutdown states.

Therefore, the Emergency Operating Procedures (EOPs) used during shutdown states were extended to fully refer to the Accident Management measures mentioned in Section 2.1.

In addition, the Beznau Severe Accident Management Guidelines (SAMGs) were extended to shutdown modes. This included the addition of special transition evaluation table to cope with those plant configurations where the core exit thermocouples are removed. This transition evaluation table used the following parameters as an alternate to the core exit temperature:

- Containment radiation
- Hydrogen concentration inside containment
- Hot Leg and pressurizer temperatures
- Reactor neutron flux.

In addition, the extended SAMGs include a procedure for the spent fuel pit.

3. REALISTIC EVALUATION OF SHUTDOWN RISK

3.1. Start of Core Degradation

In most shutdown PSA studies core damage is assumed to be equivalent with fuel uncovery. However, the following facts are important during shutdown conditions due to the low core decay heat:

- Slow steaming rate and therefore slow reduction of RCS water level
- Steam cooling of uncovered fuel as long steam production is available.

Due to these reasons, core damage is not expected to occur as long as the lower half of the fuel is covered by water. As specific analysis using MELCOR 1.8.5 QZ was performed [4] to evaluate that time window for the Beznau PWRs. The following boundary conditions were assumed:

- Loss of RHR cooling during Mid-Loop operation 22 hours after power operation
- The Reactor Pressure Vessel (RPV) head is in place, the bolts are detensioned
- Injection into the core is recovered by start of a charging pump about when the water level in the RPV is at the middle of the fuel length (15000 sec, 15200 sec and 15400 sec after loss or RHR). The injection flow rate of a charging pump is 3.5 l/sec.

The results of that analysis are shown in Figure 3 for the liquid level in the core and in Figure 4 for the maximum cladding fuel temperature.

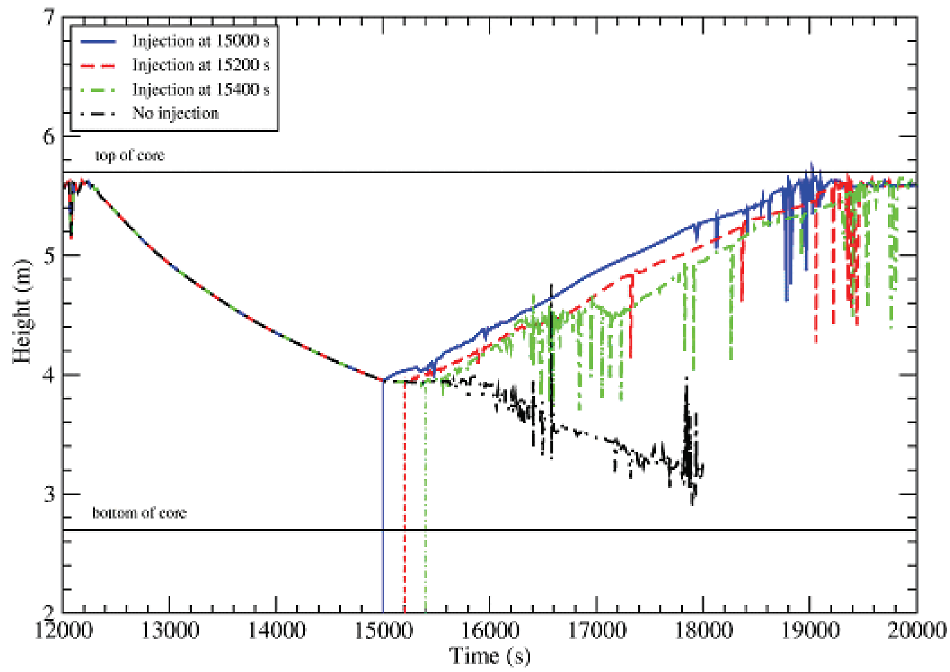

Figure 3. Liquid Level in Core for Loss or RHR Cooling at Mid-Loop with Injection Recovery

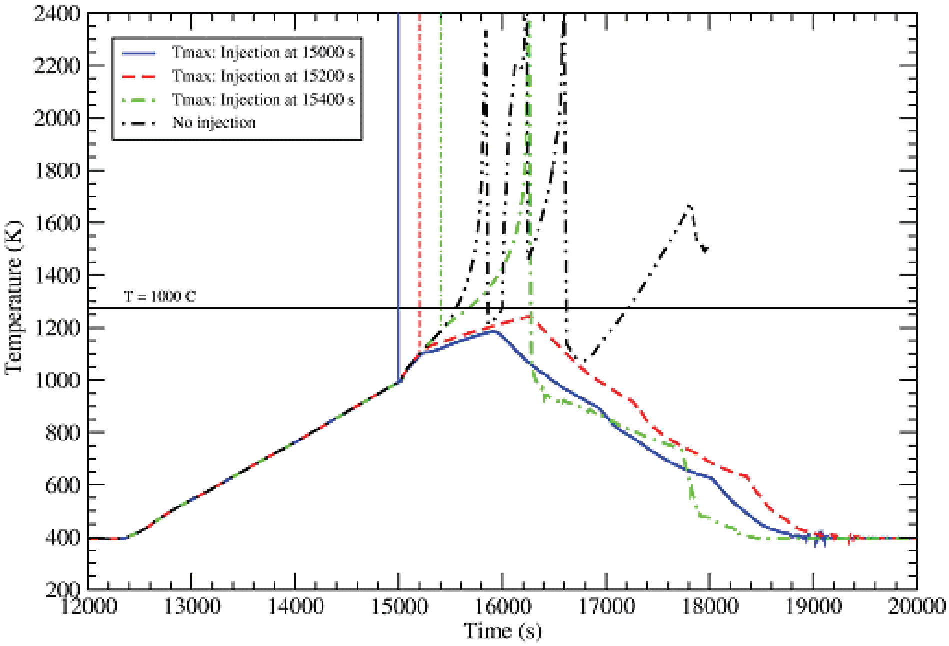

Figure 4. Maximum Fuel Rod Temperature for Loss or RHR Cooling at Mid-Loop with Injection Recovery

Based on this analysis, the following conclusions can be drawn:

- Fuel degradation does certainly not start before the core liquid level falls below the middle of the fuel length.
- After core uncovery has started, there is an additional time window of about 3000 sec to recover core cooling by start of an injection pump.

3.2. Accident Management Measures in Shutdown PSA

Based on the sections above, the following facts were considered in the update of the earlier Beznau shutdown PSA study [5]:

- The additional time window between core uncovery and start of core degradation.
- The additional physical parameters of the transition evaluation table which represent a significant and diverse indication for fuel uncovery to the operators and emergency staff.
- The transfer of the main technical responsibility after fuel uncovery from the operating personnel to the emergency staff as a new control and command group.
- The mobile equipment such as fire water pumps which can be used for recovery of core cooling.

As a result, the following extensions were performed to the earlier Beznau shutdown PSA [5]:

- Consider start of a charging pump as an additional recovery action after core uncovery has started. The dependency of the human error rate with previous operator failures was fully considered.
- Addition of a specific Shutdown Accident Management and Containment Event Tree that models the hardware as well as the human error rates of all recovery actions using mobile equipment.

- Consider the emergency staff as an independent control and command group for Accident Management measures using mobile equipment.

3.3. Simplified Shutdown Level 2 PSA

In addition, the earlier Beznau shutdown PSA study [5] was extended to a simplified Level 2 study by implementing the following additional nodes into the Shutdown Accident Management and Containment event tree:

- Operator actions to recover containment isolation, especially to close the containment hatch.
- Include a conditional failure rate for containment failure due to accident progression phenomena such as hydrogen burn, containment pressurization etc. This containment failure rate was implemented into the Accident Management and Containment Event Tree as one single node and used a failure rate that bounds the results of the detailed full-power Level 2 PSA study of the plant [6].

3.4. Shutdown Accident Management and Containment Event Tree

Figure 5 shows the Beznau Shutdown Accident Management and Containment Event Tree.

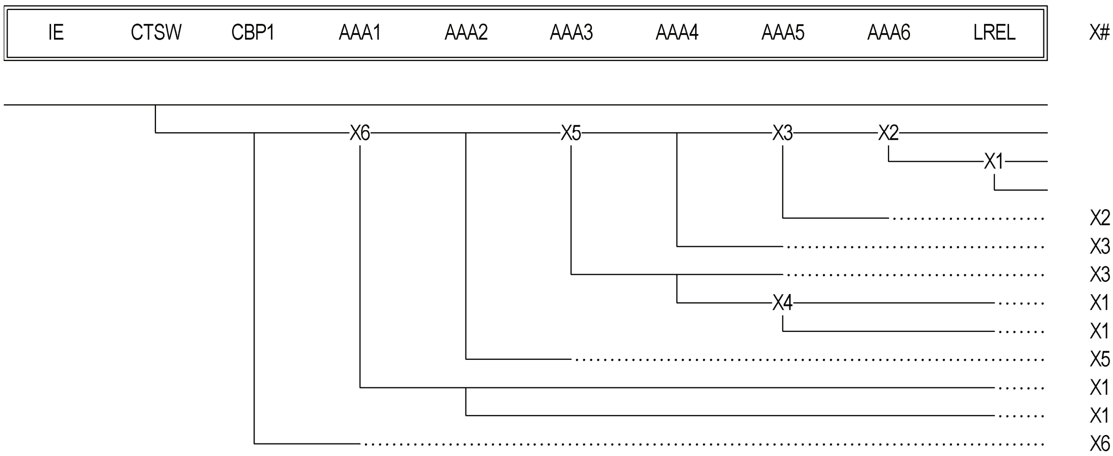

Figure 5. Beznau Shutdown Accident Management and Containment Event Tree

The nodes and content of the individual nodes of the Shutdown Accident Management and Containment Event Tree are:

CTSW ACCIDENT MANAGEMENT AND CONTAINMENT EVENT TREE SWITCH: Switch to enter or bypass the Accident Management and Containment Event Tree.

CBP1 SCENARIO DOES NOT REPRESENT A CONTAINMENT BYPASS PDS: Node to distinguish between scenarios with the containment hatch open or closed.

AAA1 OPERATORS AND EMERGECY STAFF FOLLOW AM PROCEDURES OR TRANSFER TO SAMGS: Questions if the operators and the emergency staff follow the Accident Management Procedures and Severe Accident Management Guidelines.

AAA2 CONTAINMENT ISOLATION IS SUCCESSFUL OR RECOVERED BY OPERATORS: Questions of the containment is isolated or if the open containment hatch is reclosed by the operators.

AAA3 MOBILE FIRE WATER PUMP OPERATES FOR 10 HOURS: Questions if one of the two normal fire water pumps operates for 10 hours.

AAA4 FIRE WATER TRUCK OPERATES FOR 10 HOURS: Questions if the fire water truck operates for 10 hours.

AAA5 RCS DEPRESSURIZAITON: Questions if the RCS stays at low pressure of if the hardware necessary for depressurization is available.

AAA6 OPERATORS RECOVER CORE COOLING USING MOBILE PUMPS: Questions if the operators and the fire brigade reestablish core cooling using mobile equipment.

LREL NO LARGE RELEASE: Questions if containment integrity is maintained through the accident progression.

The Shutdown Accident Management and Containment Event Tree is fully linked with the Level 1 PSA model. This enables to calculate importance measures of structures, systems and components as well as for operator actions with respect to the Large Early Release Frequency (LERF).

4. RESULTS OF BEZNAU SHUTDOWN PSA

Figure 6 illustrates the actual core damage frequency (CDF) and the large early release frequency (LERF) of the Beznau plant for all modes of operation. All events were investigated in all modes of operation, for example fires, floods, seismic events, severe weather.

As illustrated in Figure 6, the risk of core damage during shutdown modes is lower than the contribution from full-power operation. However, the LERF contribution from non-seismic events is higher for cold shutdown than for power modes. This is mainly due the contribution of unisolated containment hatch during cold shutdown.

Figure 7 shows the contributions of the individual initiating event groups to the Beznau cold shutdown CDF and LERF. While seismic represent the dominating significant risk contributor during power operation according to Figure 6, seismic risk is also considerable during shutdown.

Figure 8 illustrates the contributions of the individual system and human failures to the Beznau cold shutdown CDF. As it can be seen from this figure, human failures dominate the shutdown risk after implementation of a dedicated shutdown Accident Management program.

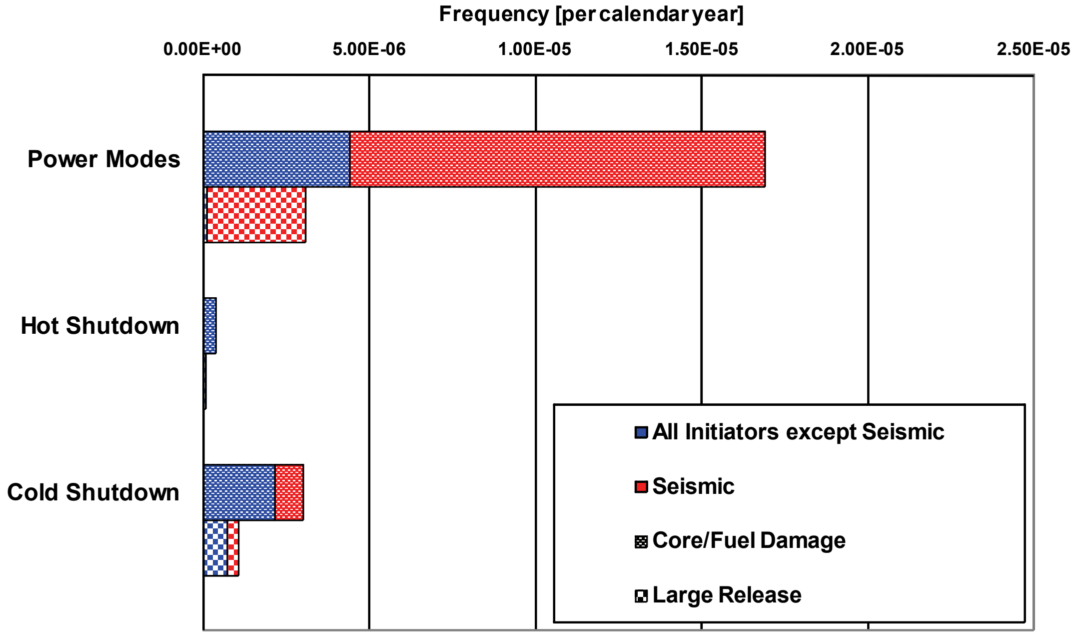

Figure 6. Beznau CDF and LERF for all modes of operation

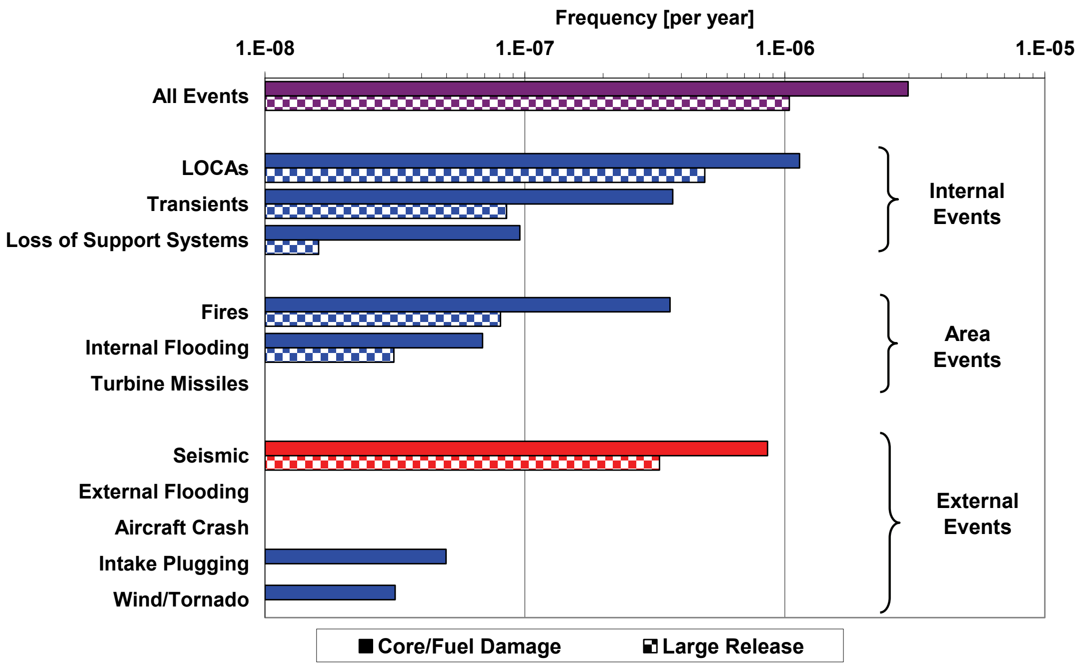

Figure 7. Contributions of Initiator Groups to Beznau Cold Shutdown CDF and LERF

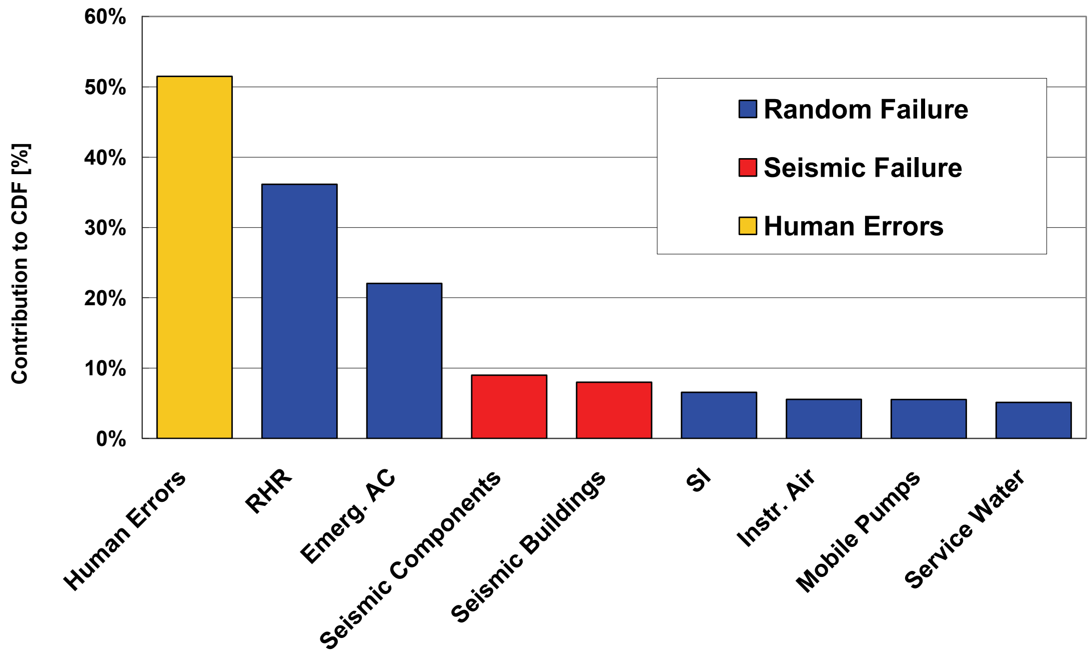

Figure 8. Contributions of System and Human Failures to Beznau Cold Shutdown CDF

5. CONCLUSIONS

Based on the last sections, the following conclusions can be drawn:

- During shutdown modes, several conditions are favorable with respect to restoration of core cooling by alternate Accident Management measures such as mobile equipment. These conditions are the long time windows and the fact that core degradations starts significantly after fuel uncovery.
- Procedures for Accident Management during shutdown (EOPs, SAMGs for shutdown modes) and fire water connections represent very cost effective measures to improve shutdown safety.
- After implementation of a shutdown Accident Management program, the shutdown core damage frequency is expected to be lower than the CDF from power modes and is mainly dominated by human error rates.
- A simplified containment event tree can be used for shutdown modes that covers the conditional containment failure rates of the detailed full-power containment event tree. This enables to perform a Level 2 PSA for all modes and all events.
- Shutdown risk with respect to large releases is mainly dominated by scenarios with failure to reclose the containment hatch.
- Shutdown risk with respect to large releases is comparable to the risk from power modes.

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The Role of Severe Accident Management in the Advancement of Level 2 PRA Modeling Techniques

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This paper explores the potential role of advanced methods in Level 2 probabilistic risk assessment (PRA[1]) for better capturing the effects of accident management (AM) guidance on severe accident risk. As part of an exploratory long-term research project, the US Nuclear Regulatory Commission (US NRC) is investigating various methodologies for conducting Level 2/3 PRAs. Key aspects that are being considered include the ability to model human actions as part of accident management and the coupling of these actions to deterministic models of severe accident progression to arrive at more realistic and higher-fidelity results. In exploring this subject area, the paper surveys potential Level 2/3 PRA approaches and current/potential simulation-based Level 1 PRA approaches with regard to modeling human response. The general area of dynamic event tree methods is further explored as a promising technology for improving the treatment of human involvement in Level 2/3 PRA.

1. Overview of the current treatment of accident management guidance in Level 2 PRA

Historically, operator actions taken after the onset of core damage have either been entirely neglected in Level 2 PRA, or have been incorporated as part of the subjective probability assignment process leading to containment event tree split fractions (using either existing or new top events). The practical need to minimize the number of unique accident progression scenarios requiring detailed deterministic analysis (i.e., variations in the time and manner in which available plant equipment could be used to change the course of the accident), has often outweighed the desire to explicitly recognize every opportunity for operator intervention.[2] In some Level 2 PRAs, results of a coarse or conservative

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[1] The term PRA is used interchangeably in this document with the term probabilistic safety assessment (PSA).
[2] A good practical perspective on these issues from the time-period following NUREG-1150 can be found in: Ang, M. L., and N.E. Buttery, An Approach to the Application of Subjective Probabilities in Level 2 PSAs, Reliability Engineering & System Safety, 1997. 58: p. 145-156.

(unmitigated) treatment of severe accident progression provide sufficient information to demonstrate compliance with numerical risk targets, and a realistic incorporation of accident management actions has been deemed unnecessary.

Having said this, studies have been undertaken to assess the effects of accident management on Level 2 PRA results, and evaluate the level of detail at which they can be represented.[3] Further, best practice guidance and consensus standards have also encouraged/required incorporation of important operator actions. As an example, the US Level 1/large early release frequency (LERF) PRA standard requires "realistic treatment of feasible operator actions following the onset of core damage consistent with applicable procedures" for Capability Categories II and III.[4] International Atomic Energy Agency (IAEA) guidance on Level 2 PRA also cautions that crediting operator actions taken after significant core damage has occurred should carefully account for factors that affect their probability of success (particularly if the Level 2 model is not directly coupled to the Level 1 logic model).[5] These factors include dependencies on prior human performance, physical dependencies in the availability of equipment, and the impacts of changing and adverse environmental conditions.

These efforts/approaches rely primarily on the use of subjective (expert) judgment to apply a combination of deterministic analysis (computer simulation), experimental data, and practical knowledge of plant/operator behavior to arrive at scalars or probability distributions that are applied to branch points in a static (containment/accident progression) event tree. The static event tree approach does not capture complex system/operator interactions well; its primary strengths are facilitating the treatment of a large number of accident sequences and facilitating the use of the aforementioned subjective probabilities. A related shortcoming of the existing approaches is difficulty in ensuring consistency between the Level 1 and Level 2 portions of the analysis, especially if: (i) separate models are used, (ii) model development and execution is handled by separate teams, and/or (iii) plant damage state binning disconnects the post-core damage sequences from their Level 1 origins.

A similar limitation with the existing framework was identified in IAEA TECDOC-1352[6], as follows:

> "There have been some attempts to quantify the influence of accident management on core damage frequency or fission product (FP) releases, with varying results. A possible reason for only few available quantifications is that the state of the probabilistic safety assessment (PSA) methodologies applied has often not been adequately defined to account for the recovery actions, particularly when complex human behaviour is involved. The modelling of the decision making process is a difficult task, and it is further complicated by the balancing of positive and negative consequences. Many decisions within the severe accident management framework depend on the outcome of actions, the consequences of which cannot be predicted, such as the time up to restoration of power in a station blackout sequence, the time to recover safety injection after it has failed, or deliberations to use a last remaining water source for the containment spray system, RPV injection, or containment flooding."

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[3] E.g., Lutz, R. and M. Lucci. "Modeling Post-Core Damage Operator Actions in the PRA." and Leonard, M.T. et al., "Optimization Study of Filtered Vent and Drywell Flooding for a BWR/4 Mark I." both available in the proceedings of ANS PSA 2008 Topical Meeting, September 2008, Knoxville, TN.

[4] American Society of Mechanical Engineers/American Nuclear Society. Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications. ASME/ANS RA-Sa-2009.

[5] International Atomic Energy Agency, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants," Draft Safety Guide DS-393 [in publication].

[6] International Atomic Energy Agency, "Application of Simulation Techniques for Accident Management Training in Nuclear Power Plants,"IAEA-TECDOC-1352, May 2003.

Again, the concern deals with the level of specificity in the means by which operator actions are included in the PRA, as well as the ability to account for the dynamics of the scenario progression.

These shortcomings are primarily a function of the computational (probabilistic) infrastructure, the state of methods and simulation tools, and the state of accident management at the time that the current generation of methods were developed, combined with the desire to have a methodology that could address a full-scope Level 2 analysis. The current paper focuses on the advantages of dynamic methods in addressing this issue, but also implicitly explores the higher-fidelity results that can be achieved when one has the motivation and resources to focus on a particular portion of the Level 2 spectrum (e.g., addressing a specific aspect of accident mitigation such as containment venting).

2. Overview of dynamic models in PRA

As discussed in the preceding sections, the current framework for conducting Level 2 PRAs implements significant accident management steps by adding top events to the model, or by modifying the point estimates or distributions associated with specific split fractions. The main disadvantage of this approach is that the actions are added to the model a priori (i.e., without the full context of the specific sequence being modeled). This context includes the time-dependent evolution of key phenomena (and the indications of these phenomena), as well as the occurrence of the top events. Moreover, in some cases, the arrangement of top events, which provides a nominal representation of accident scenario development, may not accurately represent the actual development for specific sequences. These concerns are important for the treatment of accident management actions because, as recognized in the human reliability analysis (HRA) community, proper analysis of operator actions requires a good understanding of the context for these actions.[7]

One natural approach for improving the representation of context and the associated operator actions involves the direct modeling of accident scenario development (including all relevant phenomena, operator decision making and actions, and physical accident progression) in a probabilistic framework.[8] The development of practical modeling and computational methods to implement this approach, sometimes referred to as "dynamic PRA," has been the subject of investigation for a number of years.[9] Some of these methods are discrete logic extensions of the classical event tree/fault tree methodology that employ off-line predictions of accident progression; other methods are based on a more integrated, computer-based simulation approach. For reasons further discussed in Section 3 of this paper, the following discussion focuses on the latter.

Although it was not aimed at supporting PRA, an early application in using computers to simulate the interactions between a nuclear power plant's (NPP) operators accident response model and the NPP thermal hydraulic model to generate event scenarios was the Cognitive Environment Simulation (CES).[10] The CES approach used a neural network approach that resulted in a large number of simple rules to model crew response that proved to be unmanageable. Originally, CES was a "semi-automatic" simulation using two manually-coupled computer models to simulate NPP response and crew response separately, only generating one scenario in a given simulation. Later work made the simulation automatic and capable of generating multiple scenarios in one simulation.

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[7] NUREG-1792. Good Practices for Implementing Human Reliability Analysis (HRA). US Nuclear Regulatory Commission, 2005.
[8] E.g., Mosleh, A. and Y.H. Chang. Model-Based Human Reliability Analysis: Prospects and Requirements. Reliability Engineering & System Safety, 2004. 58: p. 145-156.
[9] E.g., Siu, N. Risk Assessment for Dynamic Systems: An Overview. Reliability Engineering & System Safety, 1994. 43(1), p. 43-73.
[10] E.g., Woods, D.D., et al., Cognitive Environment Simulation: An Artificial Intelligence System for Human Performance Assessment. 1987. U.S. Nuclear Regulatory Commission: Washington D.C.

A number of computer-based methods have been developed to treat crew-plant interactions in a PRA environment (i.e., an environment requiring the identification of possible scenarios, the assessment of the consequences of these scenarios, and the estimation of the likelihood of these scenarios). These methods involve the explicit construction of dynamic event trees (i.e., event trees that branch at discrete, and not necessarily pre-specified, points in time in accordance with specified branching rules). Some of these branching rules employ the results (e.g., the achievement of system actuation set points) of accident progression calculations being performed as an integral part of the simulation. The DYLAM[11], DETAM[12], ADS[13], RRAF[14] / ADAPT[15], and MCDET[16] methods can all be considered as belonging to this class. (Note that the original DYLAM method[17], which can be viewed as the predecessor of current dynamic PRA methods, was focused on hardware reliability and was not framed as a dynamic event tree method).

Current implementations of dynamic PRA methods have three common elements: the plant model, the crew model, and a simulation manager. Predicted behaviors are generated by the plant model and the crew model, and their dynamic interactions are managed by the simulation manager. In general, most dynamic simulation tools have been developed with plant-crew interactions in mind. However, their emphases vary. For example, the ADAPT and MCDET efforts have emphasized hardware / phenomenological dynamics, while the ADS and DETAM efforts have emphasized crew dynamics.

The plant response models typically include separate deterministic and stochastic sub-models. The deterministic models typically utilize existing system codes (e.g., TRETA in DYLAM; RELAP in ADS and ADAPT; MELCOR in MCDET and ADAPT) to simulate plant response. The stochastic models include Monte Carlo simulation (e.g., MCDET), deterministic sampling from cumulative distribution functions (e.g., ADAPT), and simple rules (e.g., ADS) to determine the possible hardware state transitions and corresponding probabilities.

As compared with the modeling of plant response, computer-based modeling of NPP crews interacting with the plant is a relatively new and still evolving research area. In the context of PRA applications, a variety of models have been developed, including the COSIMO model[18] in DYLAM, the IDAC model[19] in ADS, and the OPSIM model[20] (the last has not been integrated into a specific dynamic PRA

[11] Cacciabue, P.C. and G. Cojazzi, A Human Factors Methodology for Safety Assessment Based on the DYLAM Approach. Reliability Engineering and System Safety, 1994. 45: p. 127-138.

[12] Acosta, C. and N. Siu, Dynamic event trees in accident sequence analysis: application to steam generator tube rupture. Reliability Engineering and System Safety, 1993. 41: p. 135-154.

[13] Chang, Y.H.J. and A. Mosleh, Cognitive Modeling and Dynamic Probabilistic Simulation of Operating Crew Response to Complex System Accidents -- Part 5 Dynamic Probabilistic Simulation of IDAC Model. Reliability Engineering & System Safety, 2007. 92: p. 1076-1101.

[14] Rutt, B., et al. Distributed Dynamic Event Tree Generation for Reliability and Risk Assessment. in Proceedings of the International Workshop on Challenges of Large Applications in Distributed Environments (CLADE'06). 2006. Los Alamitos, California: IEEE Computer Society.

[15] E.g., Hakobyan, A. et al. Dynamic Generation of Accident Progression Event Trees. Nuclear Engineering & Design, December 2008. Volume 238, Issue 12, p. 3457-3467.

[16] E.g., Kloos, M. and J. Peschke, MCDET: A Probabilistic Dynamics Method Combining Monte Carlo Simulation with the Discrete Dynamic Event Tree Approach. Nuclear Science & Engineering, 2006. 153(2): p. 137-156.

[17] Cacciabue, P. C. and Amendola, A., Dynamic logical analytical methodology versus fault tree: The case study for the auxiliary feedwater system of a nuclear power plant. Nuclear Technology. 74 (1986), p. 195-208.

[18] Cacciabue, P.C., et al., COSIMO: a cognitive simulation model of human decision making and behaviour in accident management of complex plants. IEEE Transactions in Systems, Man & Cybernetics, 1992. 5(22): p. 1-17.

[19] E.g., Chang, Y.H.J. and A. Mosleh, Cognitive Modeling and Dynamic Probabilistic Simulation of Operating Crew Response to Complex System Accidents -- Part 3 IDAC Operator Response Model. Reliability Engineering & System Safety, 2007. 92: p. 1041-1060.

[20] Dang, V.N., Modeling operator cognition for accident sequence analysis: development of an operator-plant simulation, Department of Nuclear Engineering. 1996, Massachusetts Institute of Technology.

framework). All of these models address cognitive aspects of crew behavior. For example, the IDAC model uses a set of sub-models treating information perception, decision making, action execution, memory and knowledge, psychological state, and operating procedures to predict key aspects of crew behavior (e.g., procedure following, knowledge-based decision making, and within-crew communication). The ADS-IDAC model has been recently exercised in an international HRA benchmark study and has been shown to provide a feasible approach for modeling procedure-driven and knowledge-driven operator responses (based on event symptoms) in conjunction with the accident simulation..[21] Recently, the structure of performance influencing factors in the IDAC model has been refined and represented with the use of a Bayesian Belief Network based on the data in the US NRC's Human Event Repository and Analysis (HERA)[22] database.

Outside of a PRA context, the Micro Saint[23] approach has been used for improving procedures and task planning during the design stage. Micro Saint calculates an operating crew's workload to estimate the level of workload requirement based on given procedures or task design. Task analysis needs to be performed before simulation to evaluate demands on the crew's visual, auditory, cognitive, and psychomotor (VACP) functions. These demands are updated by the plant state provided through the thermal-hydraulic simulation. In the first workshop of the aforementioned HRA benchmark study, the Micro Saint simulation generated human error probabilities (HEPs) for various tasks as a function of workload.

The aforementioned simulation manager component of a dynamic human response model coordinates all models during simulation and typically includes generating branches (in the case of dynamic event trees) or state transitions (in the case of state-based modeling approaches), calculating probabilities of the branches/transitions, and controlling the development of the scenario model (e.g., using truncation rules) to reduce computational requirements. In recent years, distributed computation has been implemented in some simulation tools, thereby allowing the use of parallel processing with these tools.

3. Overview of Level 2 approach classes

As part of a long-term research activity, the US NRC is currently investigating advanced modeling techniques in Level 2 and Level 3 PRA. The first step of this process was a scoping study to investigate the spectrum of potential methods that might be used in future PRAs.[24] This spectrum was divided into four classes of approaches to allow for more direct comparison and assessment of the relative benefits of the approaches. The four classes identified are characterized in the following manner:

1. Modified traditional approach – This is a containment event tree (CET)-based approach which could utilize fault trees, decomposition event trees, and/or Bayes Nets as the basis for quantifying top event nodes.

[21] NUREG/IA-0216. International HRA Empirical Study - Pilot Phase Report: Description of Overall Approach and First Pilot Results from Comparing HRA Methods to Simulator Data. US Nuclear Regulatory Commission, 2009 (In publication).
[22] NUREG/CR-6903, Vol. 1. Human Event Repository and Analysis (HERA) System, Overview. US Nuclear Regulatory Commission, 2006.
[23] Laughery, R. Using Discrete-Event Simulation to Model Human Performance in Complex Systems. Proceeding of the Winter Simulation Conference, Phoenix, AZ, December 5-8, 1999.
[24] US Nuclear Regulatory Commission, "Scoping Study on Advancing Modeling Techniques for Level 2/3 PRA," Donald Helton, May 2009. Available via the Agencywide Document Access and Management System, accession no. ML091320447 at http://www.nrc.gov/reading-rm/adams.html.

2. Hybrid event tree approach – This is an event tree-based approach in which multiple event trees are constructed based on specifics of the accident signature, and the system code (e.g., MELCOR) analysis results are used iteratively to adjust the event trees.
3. Dynamic event tree simulation approach – This is an approach whereby an executive program uses branching rules to develop a time-based event tree using the real-time results of a system code and crew module.
4. Sampling-based simulation approach – This is an approach where the system code input parameters, phenomenological models, and operator action logic have probability distribution functions that are randomly sampled, for each of a large number of calculations, to arrive at a distribution of results.

The high-level structure of these approaches (except for approach 4) retain the event tree structure familiar to Level 2 PRA, but do not preclude other approaches (e.g., influence diagrams[25]) as supporting elements. Other approaches that have been developed and employed in severe accident management (e.g., the Risk-Oriented Accident Analysis Methodology) are not explicitly depicted in the taxonomy above, but many are implicitly within the envelope associated with a particular category.

The approaches were assessed in terms of their ability to meet a number of desirable characteristics, which spanned a range of Level 2 PRA considerations, all of which have some relevance for accident management:

- Reduces reliance on modeling simplifications and surrogates (i.e., more phenomenological)
- Addresses methodological shortcomings identified by the State-of-the Art Reactor Consequence Analysis (SOARCA) project
- Improves treatment of human interaction and mitigation
- Makes process and results more scrutable
- Allows for consideration of alternative risk metrics
- Leverages advances in computational capabilities and technology developments, but is computationally tractable
- Allows for ready production of uncertainty characterization
- Permits simplification for regulatory application at a later time (i.e., after it has been sufficiently developed and applied)

With regard to the potential overall benefit for advancing Level 2/3 PRA modeling, approaches 3 and 4 were considered to be the best suited. Limited development and testing of Approach 4 has been attempted in prior work on a particular severe accident issue[26] and shows some promise for expansion to fully-integrated severe accident simulations in the future. However, Approach 3 was viewed as the best candidate for development to extend and refine current Level 2 PRA modeling techniques. As emphasized elsewhere in this paper, the key attributes of this approach with regard to better modeling of accident management are:

1. Its use of a phenomenological tool (MELCOR) directly in the event tree construction, to ensure that the event tree accurately reflects the expected evolution of the accident sequences,
2. Its use of (dynamic) event trees as the top-level structure, which are not constrained to having the same set of top events for every sequence and can aid in the understanding and communication of results, and

[25] Moosung, J. and G. Apostolakis. The Use of Influence Diagrams for Evaluating Severe Accident Management Strategies. Nuclear Technology, 1992. Volume 99, p. 142-157.

[26] R.O. Gauntt, "An Uncertainty Analysis for Hydrogen Generation in Station Blackout Accident Using MELCOR 1.8.5," NURETH-11, Avignon, France (2005).

3. Its ability to couple the operator response model directly to the phenomenological tool, such that the operator response is informed by the information (correct or incorrect) that an actual operator would have at the time the decision is being made.

A follow-on effort is being led by Sandia National Laboratories to further develop approach 3, including assessing many of the implementation decisions that were not addressed as part of the high-level scoping effort.

4. Potential benefits for accident management implementation

This section will address the benefits of incorporating accident management in to a dynamic framework, in terms of its benefit to the PRA (better modeling) and accident management (improved understanding of what is important and ways to reduce risk). We will start with some practical considerations (both positive and negative) relative to implementation. Following this, potential benefits will be outlined for the various stages of the accident (pre-core damage, post-core damage, and offsite response modeling).

4.1 Capturing accident management guidance implementation in a dynamic framework

As described above, the current framework for conducting Level 2 PRAs implements significant accident management steps by adding top events, or modifying the probabilities associated with specific branches in the tree. Some studies have concluded that this framework is appropriate, and not in need of major development. In at least one case, such a study drew this conclusion while simultaneously calling for major developments in the area of mitigative regime (e.g., severe accident) human reliability quantification[27]. In the current paper, the proposition is that better quantification of human reliability requires a new framework for developing Level 2 PRA models.

By adopting a dynamic framework, many of the major constraints identified in Section 1 of this paper are relieved. First, the action will only be incorporated in to a sequence once the associated rules have been satisfied, based on the full context of the accident (e.g., steam generator level). Second, the event can occur in the model in the order that it actually is predicted to occur during the accident, and is not constrained to happening at the same relative point for all accident sequences modeled in a given event tree. Finally, the dynamics between the plant response and the operators' actions are coupled.

The major difficulties with this approach are similar to those for Level 1 PRA, namely:

- More rigorous treatment of operator actions can lead to sequence explosion, which must be dealt with by (i) limiting the number of sequences by screening out operator actions that are not expected to have a significant effect on the course of the accident progression, (ii) merging sequences that are substantively the same and/or (iii) truncating sequences that (due to their conditional probabilities) are not expected to contribute to the overall risk result. A potential pitfall is that these determinations will have to be made based on Level 2 metrics for risk, rather than the true measures of risk that would result if they were carried through to the Level 3 analysis.
- To accomplish dynamic operator modeling, one must create a model (e.g., rule sets) for when the computer program should take specified operator actions based on the procedures, training, experience or knowledge. The development of this model, as well as the testing

[27] Ang, M. L. et al., The development and determination of integrated models for the evaluation of severe accident management strategies – SAMEM. Nuclear Engineering and Design, 2001. 209: p. 223-231.

needed to ensure consistency/cohesiveness, can be very time consuming. Data for validating the model is likely to be very limited.

- Application of the model requires oversight (or post-processing tools) to determine when the simulation has entered an untested (or an inapplicable) regime, or monitoring tools to avoid this situation.
- If strong nonlinearities exist in the model, they can magnify small errors, potentially leading to unrealistic contexts for operator actions (observed in earlier studies using simplified thermal hydraulic models).

On one final note, the use of a dynamic framework offers different strengths and limitations from the existing framework with regard to uncertainty. In either framework, it is straight-forward to model the uncertainty associated with an action taking place or not taking place. In a dynamic framework, it is arguably more-straightforward to also account for the variation in timing of a particular action being taken. For instance, when the rule set determines that a particular action will be taken by the operator, three branches could be created: one each for the action being taken with a 0, 15 and 30 minute delay. If it is determined downstream that this variation did not substantively effect the accident progression, the sequences can be re-merged to reduce computational burden. This same approach can be accomplished with a traditional approach, but the manual bookkeeping, the lack of operator context, and the (potential) lack of a 1-to-1 mapping between accident simulations and event tree paths make it more difficult.

4.2 Potential benefits in pre-core damage accident progression modeling

The dynamic simulation approach seems promising in providing practical, systematic, and detailed risk analyses on a level that can capture the dynamics of crew and plant interaction, much more so than a conventional PRA approach (i.e., event tree/fault tree). Most of the shortcomings identified in the early development of dynamic simulation tools (e.g., long computation time, event sequences explosion, programming challenges) have now become more manageable. This is mainly due to the fast advancement in computational technologies over the past twenty years, including (i) increased computing speed, (ii) object-oriented programming languages, (iii) distributed computational technology, and (iv) user-friendly programming environments.

Besides the above potential benefits, translating models into computational programs could force the model developers to re-think their models in terms of behavior (i.e., how the element would respond to the situations) instead of performance (i.e., failure or success of performing a pre-defined task). Also the relationship between the behavior and the reasons for the behavior must be expressed in an explicit manner in order to be understood by computers. This provides a clear path to identify the causes of behavior. In addition, since the implementation includes an integral simulation environment, the causal identification can be performed holistically. The root causes can be identified by starting with the proximate cause and moving to intermediate causes. The process is likely to cover all hardware and human aspects modeled in the simulation to provide detailed narrative on how a failure or an undesired consequence occurred. Such information provides a far more rich and specific context for risk assessment and management than current practice.

On a practical note, modeling the system response explicitly with a system code provides at least two additional advantages relative to the pre-core damage portion of the accident. First, the determination of core damage can be assessed based on the actual response of the fuel rather than a generic core damage surrogate parameter. This capability can be important for identifying and correcting weaknesses in the Level 1 PRA success criteria, if they exist. Second, the transition from the emergency operating procedures (EOPs) to the severe accident management guidelines (SAMGs) can be assessed on a sequence-by-sequence basis using real-time simulation results, rather than making generic assumptions.

4.3 Potential benefits in post-core damage accident progression modeling

Along with the general benefits described previously (e.g., greater context), there are some additional benefits that can be realized through the use of a dynamic framework for the post-core damage portion of the accident. One such benefit is the natural extension from Level 1 PRA dynamic work, that of capturing sequence-specific contextual information in a manner to inform the treatment of the operator's cognitive process and decision-making when executing severe accident management guidance. This aspect combines with the greater degree of realism, owing to greater reliance on phenomenological modeling, to result in a higher degree of simulation fidelity.

Related to this, dynamic methods provide an opportunity for directly coupling the phenomenological/operator response elements of the Level 1 and Level 2 portions of the analysis. Such an approach may limit the scope of the application (due to computational constraints) by requiring the re-analysis of the pre-core damage phase, but has many potential advantages in terms of ensuring consistency between the boundary conditions (both hardware and operator-related) associated with a particular accident sequence. A seamless Level 1/Level 2 analysis could avoid many of the pitfalls associated with the traditional 'pinch-point' approach.

Another advantage of capturing accident management actions more rigorously is the ability to generate new insights as to which specific actions have the largest effect on the Level 3 results. This has been done before using the traditional framework, and in fact, these studies have played a part in the development of accident management. However, the ability to address action timing variability and to ensure that the sequence-specific context supports a given action should lead to better resolution with regard to the order of importance. The dynamic framework also provides an opportunity for performing more rigorous "what if" sensitivity studies to assess the effects of key uncertainties.

A final example is the potential for treating communications pathways specific to severe accidents in a more explicit fashion. The effects that shift changeover (assignment handoffs), communications with field personnel (task assignments), and communications with offsite technical support center decision-makers (information sharing and task receipt) can be more readily explored, in much the same way that dynamic Level 1 PRA allows greater exploration of the effects of within-crew interactions.

4.4 Potential benefits in offsite response modeling

A more rigorous treatment of accident management has several inherent advantages with respect to offsite response modeling as well. First, the source terms (both magnitudes and timings) and their associated release frequencies are expected to be a more accurate reflection of reality. Second, the modeling of accident management described above can include explicit modeling of Emergency Action Level (EAL) declarations and their variability, leading to more realistic modeling of the timing, and variability in timing, associated with Protective Actions (PAs). Third, some of the logic used above to model accident management within the context of a Level 2 PRA can be extended to the Level 3 PRA (e.g., modeling of variability in decision-making regarding sheltering versus evacuation to address jurisdictional differences). Such work might be useful in addressing some of the issues identified in a 2006 US NRC conference paper on emergency management[28] (e.g., identification of problems in inter-organizational communication).

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[28] Siu, N., Current Applications of PRA in Emergency Management: A Literature Review, International Conference on Probabilistic Safety Assessment (PSAM 8), New Orleans, LA, USA, May 14-19, 2006.

5. Conclusions and recommended future work

For the reasons outlined above, the use of dynamic simulation in Level 2 PRA has the potential to improve the fidelity of modeling the actions and effectiveness of severe accident management in Level 2 PRA. A body of work exists for dynamic PRA, and this experience can be directly applied to dynamic Level 2 PRA. Further work is needed, and planned, associated with gaining experience in the treatment of these types of models in Level 2 PRA, including some aspects that are unique to this portion of the accident (e.g., non-prescriptive guidance). Such work is also being undertaken by others (e.g., The Ohio State University, Gesellschaft für Anlagen und Reaktorsicherheit - GRS). If the limitations associated with dynamic simulation can be overcome, the resulting benefits of this approach could prove very useful for assessing the variability and effectiveness of severe accident management.

Overview of the Modelling of Severe Accident Management in the Swiss Probabilistic Safety Analyses

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Abstract

A full-scope Level 2 Probabilistic Safety Analysis (PSA) including all relevant internal as well as external events was required in Switzerland at the end of the eighties. During the nineties the corresponding studies were conducted by the licensees and reviewed by HSK[1]. When HSK required the introduction of Severe Accident Management Guidance (SAMG) in the late nineties, the Level 2 PSA comprised an important technical basis. However, the Level 2 PSA had to be also updated and refined in order to reflect Severe Accident Management actions (SAM actions). This work is currently still in progress for some Swiss nuclear power plants (NPPs). Moreover, the results of the Level 2 PSA can give valuable results for an improvement of the SAM strategy. This means that the development of SAMG and Level 2 PSA can be an iterative process. In this process the probabilistic assessment of the SAM actions can play a key role.

The focus of this paper is the probabilistic assessment of the SAM actions. The paper briefly presents the legal and technical basis of SAMG as well as PSA in Switzerland. Furthermore, an overview of the SAM actions modelled in the Level 2 PSAs, the methods used for the assessment of their failure probabilities and some of the factors modelled as contributing to their failures is given. The paper discusses the particular uncertainties involved in the probabilistic quantification of the Human Failure Events (HFEs) related to SAM.

1. Definitions

This section defines a number of terms as used in this paper.

1.1 SAMG

A Beyond Design Basis Accident (BDBA) according to ENSI regulatory guideline A01/d [1] is an accident which exceeds the design basis concerning initiating events or number of failures. A BDBA that includes significant core degradation is termed a severe accident [2].

According to ENSI-guideline B-12/d [3] the objectives of Severe Accident Management (SAM) are as follows:

1. Terminate core degradation,
2. Ensure containment integrity, and

[1] On January 1, 2009, HSK, a department of the Swiss Federal Office of Energy, became ENSI, the formally independent nuclear regulatory authority.

3. Mitigate radiological releases.

Severe Accident Management Guidance (SAMG) are written decision guidance documents designed to support the Emergency Response Organization (ERO) and in particular the decision-making part of the ERO, the Emergency Response Team (ERT), during severe accidents, such that the ERT can determine the optimal strategy to follow in order to reach the above-mentioned objectives.

In addition to the written decision guidance, the SAMG includes other help tools, such as calculational guidance and computer simulation models, to support decision-making. In many EROs, a SAMG Group reporting to the ERT typically uses these tools. Finally, the SAMG includes procedures for implementing the SAM measures.

1.2 PSA

ENSI guideline A05/d [4] defines two "levels" of Probabilistic Safety Analysis (PSA) and the basic risk measures as follows:

- Level 1 PSA: Probabilistic analysis to identify and quantify the accident sequences leading to the onset of core damage.
- Level 2 PSA: Probabilistic analysis of the processes taking place after core damage and quantification of the frequency and quantity of radioactive releases.
- Core Damage Frequency (CDF): The Core Damage Frequency is the expected number of events per calendar year that occur during power operation resulting in uncovery and heatup of the reactor core and leading to a significant release of radioactive material from the core.
- Large Early Release Frequency (LERF): The Large Early Release Frequency is the expected number of events per calendar year with a release of more than $2 \cdot 10^{15}$ Bq of Iodine-131 within the first 10 hours after core damage.

The SAM actions are those SAMG-guided actions associated with the three objectives listed in the preceding section. This paper focuses on the SAM actions relevant to the Level 2 PSA for accidents initiated during power operation.

In line with ENSI-guideline A05/d, the analyses used to group (bin) accident and core damage sequences with similar conditions, in terms of the expected subsequent severe accident progressions, into the so-called plant damage states (PDS) are considered part of the Level 2 PSA. This grouping is part of the Level 1 / Level 2 analysis interface and is frequently documented as part of the Level 1 PSA, unless the PSA is implemented as an integrated Level 1 / Level 2 model.

2. Regulatory Basis of SAMG and PSA in Switzerland

2.1 SAMG

In February 2005, a new Nuclear Energy Law and an accompanying ordinance were enacted in Switzerland. The ordinance requires that written decision guidance for severe accident management is derived and is a basic technical document.

The requirements on SAMGs are further outlined in the ENSI guideline ENSI-B12. This guideline states that the SAMGs shall cover all relevant operational modes. Furthermore, it requires PSA results and insights are considered in the technical basis of SAMG.

2.2 PSA

The ordinance introduced in February 2005 requires a full scope, plant-specific Level 1 and Level 2 PSA for all relevant operational modes. It also anchors a number of PSA applications in the law. Furthermore, the ordinance authorizes the nuclear regulator body to derive two PSA-guidelines:

- Guideline ENSI-A05/d [4]: PSA quality and scope
- Guideline ENSI-A06/d [5]: PSA applications

The use of PSA as an element of regulatory decision-making process is based on the premise that the results of various plant-specific PSAs are comparable. In order to further harmonize the quality and the scope of the Swiss PSAs, it was decided to develop prescriptive guidelines as far as possible.

3. Overview on the Status of Implementation of SAMG and of PSA for Swiss NPP

3.1 SAMG

In 1997, HSK initiated a study of the development of Severe Accident Management (SAM) in various countries in order to establish in Switzerland a regulatory policy on SAMG, as it was felt that SAMG could be an enhancement to the existing procedures and guidelines at Swiss plants. Based on the survey summarized in [6], HSK required the development and implemention of SAMG by all licensees in 1998. The requirements for SAMG [7] were recently anchored in the ENSI regulatory guideline B12/d [3].

A summary of the current status of SAMG implementation in Switzerland is shown in Table 1. At all four Swiss NPP sites, SAMGs for accidents initiated during power operation have been implemented (e.g. [8]) and the SAMG has been addressed so far by emergency exercises. Moreover, SAMGs for accidents initiated during shutdown have been implemented at two sites with pressurized water reactors (PWRs) and one site with a boiling water reactor (BWR). The implementation is under preparation in one BWR site. Emergency exercises addressing shutdown accidents have not taken place so far.

In view of the advanced status of SAMG implementation and validation for accidents initiated at-power, it is worthwhile to compare the respective entry criteria. Table A.1 in the appendix shows both the commonalities and the differences among the NPP sites. The two PWR sites have in common that the core exit temperature is used as an indication. One BWR site and one PWR site have in common that the indication of hydrogen concentration is used. In one PWR site, all criteria are purely symptom-based (self-standing) in the sense that physical indications are not combined with contextual conditions regarding the procedure in-use or the progress of actions initiated so far. The differences in the entry criteria reflect various aspects including the philosophy of the owner group and the features of the emergency operating procedures available before the start of SAMG development.

Table 1. Status of SAMG implementation and validation in the four Swiss NPP sites.

	KKB PWR	KKG PWR	KKL BWR	KKM BWR
SAMG implemented for Full Power (FP)	2001	2006	2004	2004
SAMG implemented for Shutdown (SD)	2005	2006	in preparation	2007
SAMG trial in emergency exercises	FP	FP	FP	FP

KKB: Beznau NPP; KKG: Gösgen NPP; KKL: Leibstadt NPP; KKM: Mühleberg NPP

3.2 PSA

The development of a PSA for a Swiss nuclear power plant was started in 1983. This initiative was aimed at the development of a Level 1 PSA for the Beznau nuclear power plant. Subsequently, in 1987, HSK required the utilities to perform full power Level 1 and Level 2 PSAs for all Swiss nuclear power plants. Four years later, HSK additionally required the licensees to develop plant-specific low power and shutdown PSAs including external events.

PSAs for all Swiss nuclear power plants have been performed by the licensees and independently assessed by HSK. The plant-specific PSAs include internal events as well as external events such as fires, flooding, earthquakes, aircraft impacts and high winds. Level 1 PSAs have been developed for full power as well as for shutdown mode. In parallel Level 2 PSAs for full power state has been developed for all Swiss NPPs considering all relevant internal and external events. Several intermediate updates of the PSAs have been performed. For every periodic safety review a fully updated PSA needs to be submitted to ENSI by the licensees.

The new ordinance in 2005 requires the quantification of the release risk for all relevant operational modes. This implies the introduction of a Level 2 PSA for low power and shutdown modes. Till end 2008 for two plants a full scope Level 2 PSA for low power and shutdown modes have been submitted to ENSI.

4. Overview of SAM Actions and their Modelling in Swiss Level 2 PSA

4.1 SAM actions and Level 2 Scenarios

As discussed in a 2001 review of the status of SAMG implementation in Switzerland [6], some of the SAM measures supported by the SAMG existed prior to the development of the SAMG.

At a high level, looking at both BWRs and PWRs, some of the SAM measures and associated hardware and systems include:

- Containment venting
- Flood for heat removal, e.g. of the reactor pressure vessel, of the drywell
- Flood or spray for radionuclide retention
- Alternative water supplies, especially alignment of firewater
- Recombiners, igniters etc. for hydrogen control (containment atmosphere)

In addition, there exists hardware relevant to SAM that is also used in preventive actions, in response to the initiating event and prior to core damage. Such hardware includes for instance, valves for the depressurization of the primary circuit. The implication is that SAM measures and the operator actions to implement these measures cannot be identified solely in terms of hardware and systems.

A typical example of a SAM action is the initiation of a Filtered Containment Venting System. However, SAM actions can also be operator actions to stop or block SAM-related systems when their function could be detrimental to mitigation. Examples include turning off active hydrogen recombiners at higher H_2 concentration levels (e.g. >6% at one plant) or stopping containment venting in order to avoid a large release source term.

Table A.2 in the Appendix lists the types of SAM actions modelled in the most recent Level 2 PSAs for the Swiss NPPs [9],[10],[11],[12]. The actions modelled in the PSAs for the PWRs correspond to the systems and measures listed above. On the other hand, the BWR Level 2 PSAs address only a small subset of the SAM measures; the latter have not been fully updated to consider the SAMG.

In this survey of the Swiss Level 2 PSAs for initiating events during full-power operation, there were marked differences in the number of SAM actions included in the PSAs. It is important to note that several of the Level 2 PSAs, in particular the PSAs for the Swiss BWRs, are in an update phase. These interim PSA updates account only partially for SAMG and, at this time, do not fully treat the impact of the introduction of SAMG.

4.2 Methods used to Quantify SAM Actions

The performance context of SAM actions

SAM actions share some similarities with the operator actions modelled in a Level 1 PSA, those actions in response to an initiating event and aiming to bring the plant to a safe shutdown state as foreseen in the Emergency Operating Procedures (EOPs). However, there are significant differences in the performance conditions and context that make the quantification of SAM actions different from that of the actions addressed in a Human Reliability Analysis for Level 1 PSA.

Some of the most significant differences are:

- The prescriptive character of EOPs vs. the informative nature of SAMG.

  EOPs represent a plan that should be followed to the extent possible while SAMG are more akin to a set of options with informative character. This distinction is not completely unambiguous since there are a few areas with scope for the control room operators' judgment in the EOPs and, conversely, some accident measures with clear criteria within the SAMG.

- The optimal response (whether or not to implement the SAM measure) cannot be fully determined in advance.

  One of the reasons for the informative rather than prescriptive character of SAMG is that the uncertainties concerning accident progressions hinder the determination of the optimal response in advance. Some of these uncertainties will not be eliminated during the accident, such that the determination of the optimal response for the situation "at hand" remain subject to uncertainties.

- The responsible staff for making the decisions within the EOPs and the SAMG.

  The decision-making responsibility for SAMG actions lies with the head of the Emergency Response Team (ERT), who is advised by the SAMG team, the ERO, and, in some cases, other external experts. Some SAM measures require the agreement of the authorities.

- The need to consider radiation exposure in assessing the actions to implement the SAM measure, in terms of feasibility as well as constraints on the execution.

The time window for some SAM actions may be on the order of hours; however, the modelled SAM actions include some with small time windows, of about 20-30 minutes. This will affect the decision aspects as well as the execution aspects of the SAM action.

Concerning the analysis of dependencies between AM actions and previous human failure events, some of the analyses also consider whether the Level 1 PSA sequence, the scenario leading up to core damage, includes an operator error. Finally, it should be noted that the actual execution of the actions required to implement an SAM measure can to some extent be modelled as in HRA for Level 1 PSA, provided that the conditions specific to severe accidents are accounted for in the performance shaping factors.

Quantification methodology

As can be seen in Table 2, two approaches to derive the probabilities assigned to the SAM actions in the Level 2 PSA are being used:

1. an expert / engineering judgment process that results in probabilities of occurrence that combine the probability that the Emergency Response Team will decide that a given SAM measure is

optimal (should be taken) and the probability of successfully implementing the AM measure. The latter includes the success of manual implementation as well as the availability of the hardware.

2. an HRA-type analysis that results in human failure event probabilities and the hardware is modelled separately.

Table 2. Modelling of SAM actions in the Swiss PSAs.

	KKB	**KKG**	**KKL**	**KKM**
Types of SAM actions *	10 types	7 types	Manual alignment of alternate injection per SAMG	3 types
Cases **	34 cases	19 cases	2 cases	7 cases
Approach to quantify SAM Actions	HRA-type analysis: Assignment to categories "very simple", "simple", "complex", referring to the difficulty of the action	HRA-type analysis: ASEP nominal per software	Engineering/expert judgment	Engineering/expert judgment (APET questions)
Dependence	Dependence among L2 actions is addressed.	Dependence on previous HFEs and among SAM actions are considered.	Treated integrally in expert judgment.	
Remark			This interim update of the PSA focuses on the impact of SAMG on release. The incorporation of SAMG-guided actions is on-going.	This PSA treats the SAM actions within a classical L2 PSA approach (in APET questions); an HRA-type analysis is not used to quantify these actions.

* The types of actions are listed in Table A.2 of the Appendix.

** The quantification of an action (of a given type) may be subdivided into cases that represent different scenarios. The listed number of cases excludes guaranteed failures.

An HRA-type process is used in both of the PWR Level 2 PSAs, where 20-34 specific cases (split fractions) have been modelled. The expert/engineering judgment process is used when the AM actions are being modelled within the responses to the Accident Progression Event Tree (APET) questions.

In the engineering/expert judgment process, a clear distinction is not made among feasibility (principally hardware availability), plant state assessment issues (what is the state of plant and core and what is the expected progression?), strategy issues (what is the best response or AM-measure for this plant state?), and challenges (if any) to the manual implementation of the measures.

In the category-based assessment used in the Level 2 PSA for KKB, the definitions for the difficulty categories encompass both the assessment/decision issues and the implementation/execution aspect.[9], (for details, see also [13]) These failure probabilities do not include the hardware availability.

Finally, in the ASEP-based assessment used for KKG's Level 2 PSA, the decision component of the SAM action is modelled with a time reliability curve.

Probabilities for the SAM Actions

Table 3 shows the ranges of the probabilities assigned to the SAM actions in the various PSAs. These probabilities correspond to failure probabilities in the case of the PWRs (first two rows) and to probabilities of non-occurrence for the BWRs (last two rows). The main distinction between the two types of probabilities is that the probability of non-occurrence typically includes a) whether the ERT will decide that the measure is optimal for the scenario as well as b) the hardware availability.

Table 3. Probabilities assigned to SAM actions in the surveyed PSAs.

Counts	Probability of failure or non-occurrence				
	$p < 0.001$	$0.001 \leq p < 0.01$	$0.01 \leq p \leq 0.1$	$p > 0.1$	**Total**
KKB PWR	0	2	20	12	34
KKG PWR	0	2	14	3	19
KKL BWR	0	0	0	2	2
KKM BWR	0	0	4	3	7

Table 3 shows the number of SAM actions within different probability bins for the surveyed PSAs. Although there are differences among the studies, many values have a probability of failure or non-occurrence exceeding 0.01 and a fair number of these exceed 0.1. In general the values are larger than the human failure event probabilities often found in Level 1 PSAs. Note that some PSAs are currently being updated or reviewed.

4.3 Discussion

Section 4.2 has outlined some of the major differences in the performance contexts of preventive actions and SAM actions. It is worth noting that both the engineering/expert judgment approach and the HRA-type analyses can in principle address the specifics of the performance context of SAM actions. This has been seen in the surveyed PSAs although there are differences in how they have done this. This section discusses in more detail the major issues for the HRA of SAM actions and the open issues.

With regard to the execution of the actions needed to implement the SAM measures, the severe accident conditions need to be kept in perspective, in particular because of the local actions needed for some SAM measures. These factors are radiation exposure (as mentioned earlier), the level of stress, which may indeed exceed the stress experienced prior to core damage, the amount and type of training on SAM actions, and the availability of personnel. The level of training may not be comparable to that for the actions guided by Abnormal Operating Procedures and EOPs.

There are significant differences in the decision aspects of SAM actions compared to preventive operator actions. The severe accident state suggests that hardware failures, human failure events, or a combination of these have occurred and the systems and responses intended to prevent core damage have in some way failed. The potential for dependence between the SAM actions and the previous human failure events needs to be examined, both for previously failed preventive actions as well as previously failed SAM actions. The entrance of the Emergency Response Team into the situation, as a new set of actors, should generally reduce potential dependence; on the other hand, their assessment, at least initially, will not be independent of the operating crew's assessment. **The factors affecting potential dependence of SAM actions on previous HFEs need further study.**

The multiple strategies for mitigating the accident and the multiple options for the implementation of a specific SAM measure constitute a second challenge for modelling the decision aspects of SAM actions. In the surveyed PSAs, the modelling of an SAM action typically considers only one option among the alternative means of implementing the SAM measure. In one way, this is a conservative assumption in terms of not crediting several means for accomplishing a given AM-measure, given that dependence among the failures of the options may be significant. **On the other hand, the selection of the option that is in fact modelled for the SAM action in the PSA may represent a decision with**

a significant potential for error. Is the modelled alternative the option designated in the SAMG as the preferred option or is it instead the option with the highest availability?

A capital difference between preventive required actions and SAM actions lies in the status of EOPs and SAMG, respectively. The EOPs are essentially prescriptive while the SAMG are intended to be informative. This aspect of the SAMG is deliberate and reflects the recognition that there are large uncertainties in severe accident conditions and in the knowledge of their evolution. In light of these uncertainties, it is less possible and not desirable to fully plan and define a prescriptive mitigative response. When the optimal SAM measures can be determined in advance, a prescriptive instruction may be formulated and included in the SAMG. In the more general case, however, **the decisions associated with some SAM actions will involve not only identifying the plant state and the applicable guidance but also determining (judging) whether the SAM measure could be effective in the given severe accident condition**. This second aspect of decision-making is not required for the preventive actions instructed by EOPs.

5. Further Considerations

In the following a number of further considerations are noted:

- The entry criteria of SAMG are designed such that entry to the SAMG occurs prior to core damage. In contrast, the boundary between the Level 1 and the Level 2 PSA is core damage. It is worth noting briefly the general reasons of this difference:

  Aiming to enter SAMG before core damage reflects the precautionary orientation of the ERO and control room crew. Anticipating core damage in this way ensures that the mitigation measures would be ready when needed. Furthermore, this approach allows defining robust and simple entry criteria.

- With regard to SAMG for shutdown, three of the sites have implemented the guidance but only two of the licensees have addressed the SAMG in their Level 2 PSAs for shutdown.
- The fact that some of the probabilities for the SAM measures account for hardware failure can cause some difficulties for some PSA applications required by ENSI A06 [5]. However, it has to be considered that some of the L2 PSAs as well as SAMG are being currently updated or are under review.
- The introduction of SAM Guidance and the associated training have contributed in an increased reliability of Level 2 mitigative actions. Sensitivity studies conducted by the licensees have identified a number of SAM actions that significantly reduce risk. This implies that to obtain a realistic estimate of risk, it is important to model the actions and measures supported by SAMG in Level 2 PSA. However, there are significant challenges for quantifying the impact of SAMG on the reliability of these actions.

6. Conclusions

Updates of the Level 2 PSAs, which account for the SAM Guidance implemented in Switzerland between 2001 and 2006, have been completed for two of the four Swiss NPPs so far. The remaining updates are ongoing. Nevertheless, the overview of the Swiss PSAs with respect to the modelling of the SAMG provides some useful results and highlights some differences in the treatment of the main issues.

In providing guidance and other tools to support the decision-making of the Emergency Response Team as well as instructions for the manual implementation of the SAM measures, the SAMG can be expected to support a more comprehensively informed strategy for the mitigative response to severe accidents. However, a difficulty for PSA modelling of SAMG relates to the role of SAMG as informative (non-prescriptive) guidance. This role accounts for the fact that the optimal response to a severe accident cannot be fully determined in advance due to the still significant limitations in the state-of-knowledge on severe accident progression and phenomena.

With respect to the (Human Reliability) Analysis of the SAM actions, a number of differences and issues arise when compared to the HRA of the preventive actions modelled in Level 1 PSA. Besides the above mentioned non-prescriptive nature of the guidance, the need of the Emergency Response Team to rely partially on information provided by the control room crew and on this crew's assessment introduces elements that may lead to some dependencies. The transition to a new set of mitigation-oriented objectives and the increased expertise available to and within the Emergency Response Team should work in favor of effective decision-making. Potentially negative factors include the increased uncertainty regarding the plant state, the expected progression of the severe accident, the (by necessity) open aspects of the mitigative response plan, and the need for more parties to agree. In contrast, the preventive response reflected in the EOPs have been analyzed more deeply.

The analyses of SAM actions in the surveyed PSAs have addressed these issues in different ways. This may reflect justifiable differences in the accident management "philosophy" underlying the SAMG for a given plant. However, it may also reflect the experts' different perspectives on the key factors. Foremost among the differences in these analyses is the informative, non-prescriptive character of SAMG. Given the scope of judgment (intentionally) left to the ERT in severe accident situations, it is difficult to predict reliably what the ERT will decide in a given scenario and to quantify this prediction.

In this light, it seems particularly useful a) within the PSA, to define clearly the limitations / assumptions of the analysis of SAM actions, and b) to perform sensitivity analyses to address those assumptions that can be expected to have a large impact on the PSA results.

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Appendix: Tables on technical details

Table A.1. SAMG entry criteria for accidents initiated at-power in the four Swiss NPP sites.

BWR site 1	1. Core cooling cannot be ensured after entry in procedure on alternate core cooling OR 2. Emergency reactor depressurization fails after entry in procedure on alternate reactor depressurization OR 3. ATWS (Anticipated Transient without Scram) and temperature conditions in the decay heat removal system exceed capacity limits OR 4. Hydrogen concentration in the containment exceeds 5.0 %
BWR site 2	1. Reactor level cannot be increased and dropped below -79cm (fuel range) after entry in procedure on reactor level recovery OR 2. Reactor level cannot be increased and dropped below -79cm (fuel range) after entry in procedure on reactor level control under ATWS conditions OR 3. Insufficient number of safety relief valves (SRVs) opens after entry in procedure on reactor flooding OR 4. Adverse difference of reactor vs. containment pressure after entry in procedure on reactor flooding OR 5. ATWS and no SRV open after entry in procedure on reactor flooding OR 6. ATWS and reactor pressure below steam mass pressure after entry in procedure on reactor flooding
PWR site 1	1. Core exit temperature above 650°C and no indication of success of the initiated actions on core cooling
PWR site 2	1. Core exit temperature above 620°C OR 2. More than one protection goal is violated and the dose level in the containment is >10'000 mSv/h OR 3. Hydrogen concentration inside the containment exceeds 0.5 %

Table A.2. Overview of the SAM actions credited in the Level 2 PSAs for the four Swiss NPPs

	KKB PWR	KKG PWR	KKL BWR	KKM BWR
Types of actions	• Transfer to SAMG • Control containment under-pressure • Control containment under-pressure after venting • Align makeup to SGs • Depressurize RCS • Recovery of core cooling • Mitigate containment bypass by water pool or spray • Align emergency containment spray • Inject water into containment • Unlock containment vent system	• Containment isolation • Primary depressurization using PDE valves • Align firewater to RCS • Initiate H2 recombiners • Containment venting • Refill scrubbing tank for containment venting system • Close venting line (to avoid large release source term)	• Manual alignment of alternate injection for in-shroud injection	• Operation of Drywell Spray and Flooding System (DSFS) • Manual venting using Containment Depressurization System • Injection of firewater into reactor pressure vessel (per SAMG)
Cases (excluding guaranteed failure)	34 cases	19 cases	2 cases	7 cases

Extended Use of MERMOS to assess Human Failure Events in Level 2 PSA

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Abstract

MERMOS is the HRA (Human Reliability Assessment) method used by EDF in order to assess post-intitiator HFEs (Human Failure Events) for Level 1 PSA. To take into account the contribution of Crisis Organization, decision making and field operator actions, simple HRA models as Time-Reliability Correlations could be insufficient. Then EDF R&D has extended the domain of MERMOS to Severe Accident Management to provide analysts with a method that help them to take into account Human contribution for ensuring limitation of radioactive discharge after the core meltdown. Since few HFEs are to be analyzed in Level 2 PSA, generic conservative analyses can be performed and then analysts would derive them to the specific context of the analyzed HFE by the "delta approach" of MERMOS. To take into account the specificities of the considered HFEs, the systemic modeling of MERMOS takes into account the "Prognosis" function devolved to the extended EOS (Emergency Operating System) that includes plant and national Crisis Organisations teams. The "Prognosis" function allows MERMOS analyses to take into account the anticipation of the future state of the reactor and the containment that is realized by the extended EOS. Another important specificity of Level 2 HRA is the lack of data from actual or simulated events. Even if severe accident simulation exercises are regularly organized and have been observed, experts judgments are needed in order to imagine, to quantify and to validate the MERMOS scenarios. For Level 1 PSA, a small MERMOS experts team belonging to the HRA analysts team is enough, but for Severe Accident Management, it is necessary to organize dedicated collective assessment with crisis organization members to build qualitative and quantitative data from their judgments. In the paper we will describe the analysis of one Level 2 HFE. The HFE consists in the failure of the immediate depressurization of the primary circuit by opening of the pressurizer relief valves, at the beginning of the application of the Severe Accident procedure. The MERMOS analysis has been completed as a demonstration of the method. A successful expert judgment collection process has allowed us to collect data (or, better to say, knowledge) for the worst case.

1. Introduction

The purpose of this paper is to show from an example how EDF R&D proposes to more accurately model the "Human Factor" aspects to Level 2 PSA (Probabilistic Safety Assessment), and the findings resulting from the first case of it being applied. The MERMOS[1] HRA[2] method and the specific features of Level 2 PSA are briefly described first, to enable the example to then be presented in greater detail.

2. MERMOS: a benchmark HRA method at EDF for level 1 PSA

MERMOS is the benchmark HRA method at EDF, initially designed to analyse the failure of HF[3] mission post-accident in Level 1 PSAs. Its objective is to qualitatively explain and structure the reasons for potential failures of the operating system, for a given HF mission. The "operating system" here means the team in the control room with its procedures and the interface with the process. Consequently, the results of a MERMOS analysis are presented in the form of various "failure scenarios" which are a set of "little stories", structured and quantified, leading to the failure of that aspect [ref 1][4].

For Level 2 PSAs, the French nuclear safety authority asked EDF to develop an HF model which includes in a realistic way assessment of the risks by the emergency response teams, along with any resulting reluctance they may have had to take certain actions. As a fixed-factor model does not satisfactorily meet this request, an extension to the MERMOS method was developed, based on conducting two case studies [ref 2][5].

3. MERMOS extension for Level 2 PSA: inclusion of the emergency response organisation and specific features of L2 PSA modelling

Extending MERMOS to L2 PSA required the inclusion firstly of the emergency response organisation, organized in France as the National Crisis Organization (cf figure 1), and secondly the specific features of L2 PSA modelling.

As regards the emergency response organisation, this in fact involves including it within the socio-technical system hitherto considered under the MERMOS method (operating system confined to the control room), taking the specific operating procedures of this extended system into account, and resolving the problem of a lack of feedback data on severe accidents.

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[1] MERMOS : French acronym for method for evaluating fulfilment of operator safety actions (*méthode d'évaluation de la réalisation des missions opérateur pour la sûreté*)

[2] HRA : Human Reliability Assessment

[3] HF : human factor

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[5] [ref 2] IEEE / HPRCT, 26-31/08/07, Monterey CA, USA " MERMOS : an extended second generation HRA method" P. Meyer, P. Le Bot, H. Pesme

The "operating system" considered with MERMOS in Level 1 PSA is limited to the team in the control room, with its procedures and the interface with the process; the emergency response teams are included only for some scenarios and handled as a fixed-factor. In Level 2 PSA, the operating system under consideration is extended to the site's MCC[6], the local emergency response team and the national emergency response team.

This extended system operates in a specific way; it must not only draw up a diagnosis, strategy and actions, but also a prognosis of the situation, and it is reiterated that this prognosis consists not only of evaluating developments in the situation but also consideration of aggravating factors and proposed counter measures. A new "function" in the system is incorporated into MERMOS to consider failure scenarios, i.e. the prognosis functions. Hence the SAD[7] functions used in the method become PSAD functions.

The lack of feedback data from severe accidents is a difficulty when it comes to devising potential failure scenarios. Firstly, there is (fortunately) little data available on severe accidents; secondly there are few simulations and their objectives, which differ from those for gathering HRA data, make them difficult to use (as they target communication methods, etc.); lastly, knowledge of serious incidents is still rare in EDF's PSA teams. The gathering of specific data was therefore organised with experts in controlling severe accidents, members of the national emergency response team, for both case studies conducted.

As regards the specific features of L2 PSA modelling, this involves including the aggregation of very different initiators which lead to a comparable physical situation, which is one difficulty from a HF viewpoint. Conversely, the limited number of HRA aspects in Level 2 PSA is an advantage. To take these elements into account, the modelling choices for MERMOS are the running of a limited number of conservative generic analyses, which could be versioned into more specific contexts thanks to the "delta" approach of the MERMOS method. The MERMOS "per delta" method consists of adapting the analysis for one aspect to that of a similar aspect.

The HRA analysis for the two generic aspects chosen was submitted to several experts in Severe accidents who, by following the MERMOS method procedure with help from HRA experts, were able to augment them, add details to them and justify them on the basis of their knowledge of accident contexts and control. One of these analyses is presented in the next section.

This gathering of opinions from severe accident experts proved very worthwhile, and also enabled knowledge to be collected which went beyond the two case studies and which will be of use in future analyses, in particular as regards:

- the context of severe accidents (for example, the fact that those involved did not rely on provisional maintenance data until it was confirmed)

- procedure management (for example, the severe accident response guide actions are not based on the status of power plant parameters, since it is assumed that these indications could be wrong in these situations, but on the availability states of power plant systems)

- team training (for example, the messages given in training request operators to strictly observe the order of instructions).

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[6] MCC : Main Control Center (cf Figure 1 next page)

[7] SAD : strategy, action, diagnosis

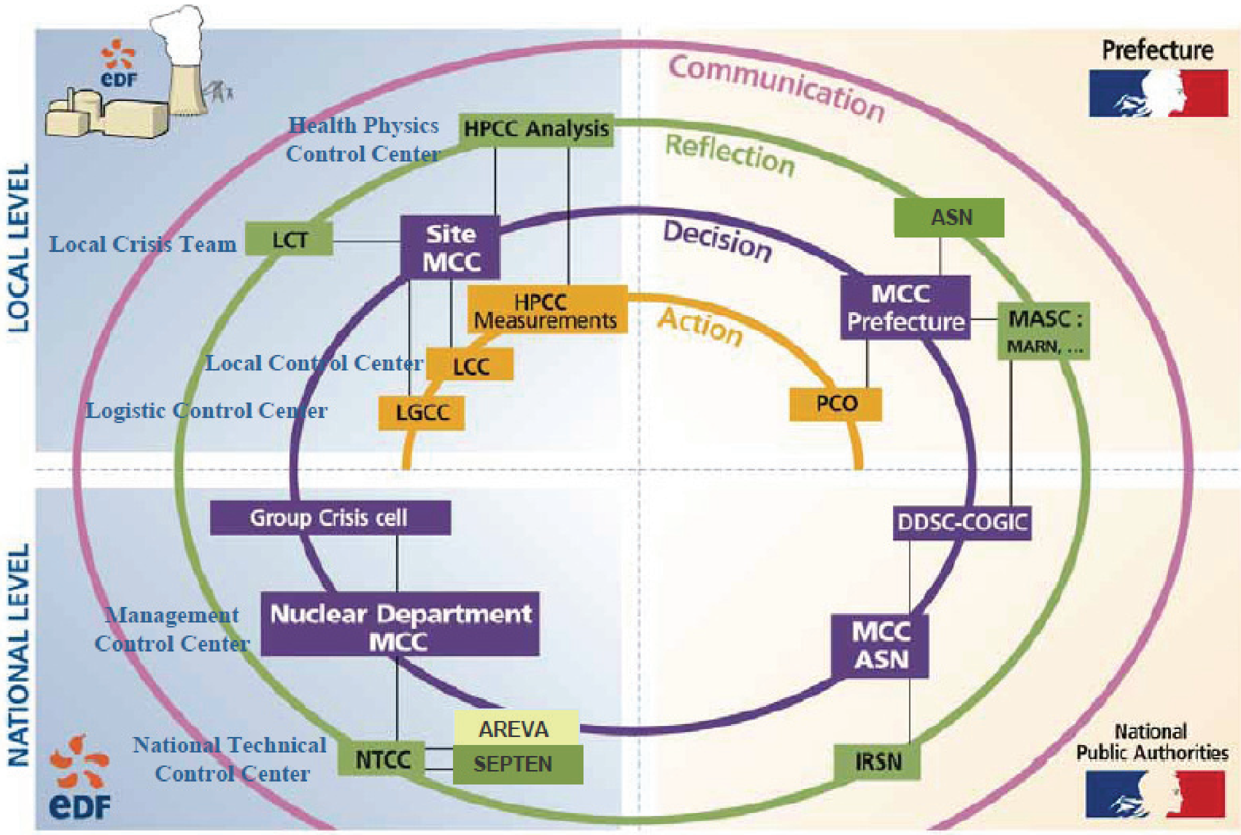

Figure 1: National Crisis organization

4. Analysis example for a Level 2 PSA aspect: depressurisation of the primary cooling system by opening pressuriser valves

To develop and illustrate the method, two generic aspects were analysed by R&D, i.e. depressurisation of the primary cooling system through a serious LOFW[8] accident with the SI[9] unavailable, one in a situation combined with SBO[10] and the other not. If necessary, other versions of these generic aspects could be designed for more discriminating L2 PSA initiators.

In this paper, we will present the most unfavourable example of the two, i.e. the aspect analysed for the combined LOFW+SBO situation. This aspect consists of the opening the pressuriser valves within 15 minutes of reaching the RCIT[11] > 1100°C criterion (for applying the GIAG[12] procedure used by the local emergency response team), with transmission of the RCIT to the control room unavailable. This aspect is in fact required from the start of core melt in L2 PSA (even before in L1 PSA in respect of feed and bleed) and in our case study we assume that this has not been carried out before reaching the RCIT criterion.

[8] LOFW : loss of feedwater (for the steam generators)

[9] SI : safety injection system

[10] SBO : station blackout

[11] RCIT : Reactor core instrumentation temperature

[12] [ref 3] GIAG: Serious accident response guide : GIAG V4 CPY (EDF DIN : ENFCRI050018A). GIAG is the EDF's procedure used by operators in the control room for severe accidents

For this type of initiator, the start of core melt is reached after two hours from LOFW+SBO combined, the RCIT criterion is reached after 4 hours and it is assumed that the PUI[13] is triggered after 20 minutes if the SBO power loss is total and sudden. It is further assumed within our PSA that the local emergency response team is operational 2 hours at the latest after the PUI is triggered, and the national emergency response team 4 hours afterwards at the latest.

It should be noted that no dependency with other aspects has been included for this first methodological case study because the PSA events tree within which this aspect is incorporated was not finalised at the time of the study. The MERMOS method allows dependencies between HR aspects to be taken into account (by describing the outcome within failure scenarios).

The resulting aspect analysis sheet is as follows:

	HF MISSION IDENTIFICATION
Name	Primary cooling system depressurisation by opening pressuriser valves
Accident category	SBO
Reactor status	Full Power
Initiator	Core melt in an SBO situation
Series	900MW-CPY
Procedure	GIAG - SPE

	HF MISSION DESCRIPTION
Functional objective	To depressurise the primary cooling system, i.e.: Open the available pressuriser valves to limit the risk of pressure vessel fracture, from the resulting creep or SGTR.
Success criteria (with timing)	Available pressuriser valves open 15 minutes after the GIAG criterion reached i.e. RCIT = 1100°C

[13] PUI :Internal Emergency Response Plan

TIME REQUIREMENTS	
Time T0 (where required)	RCIT = 1100°C - primary cooling system under pressure with no available back-up water (failure of restore actions)
Time T1 (where data is visible in control room)	Rapid rise in RCIT between 800°C and 1100°C (T1 before T0 here, therefore the system can anticipate the need for this aspect)
Time T2 (where instructed)	GIAG application criteria on criterion RCIT > 1100°C (data to be gathered in situ if need be, if power supply to cooling monitoring system is lost) Entering the GIAG procedure occurs on the MCC's agreement, approved after the PUI is triggered, which happens 20 minutes after the initiator event, in the sequence requesting an intermediate shutdown. Secondly, for the safety engineer in the SPE procedure: - the criteria for local RCIT monitoring are no power from back-up LHA and LHB distribution systems, and AFW Back-up turbine-driven pump out of service - power supply to the pressuriser valves and preparing to open them is instructed where there is no power from back-up LHA and LHB distribution systems and the AFW Back-up turbine-driven pump is out of service and SG level < 9.8m or RCIT > 330°C)
BETWEEN T0 AND T1	
Perceptible signs of change	Rapid rise in RCIT (and SG levels if SBO+LOFW combined)
Counter measures to attempt	Restoring power supply prior to re-establishing primary or secondary back-up, or use of a neighbouring unit's volume control tank.

This accurate description of the aspect, relative to requirements for its success and the time-scale requirements, provides a full specification of its characteristics before undertaking an analysis.

To be in a position to devise and build all possible failure scenarios for the aspect, the systematic MERMOS procedure requires a functional breakdown of the requirements on the basis of strategy, action, diagnosis and prognosis, stating the following elements in particular:

	FUNCTIONAL BREAKDOWN OF REQUIREMENTS

Strategy:

1. Rival objectives to the required strategy		Containment isolation (phase 1 and 2, final alignment, effluent) Cooler system isolation (valves outside reactor building) U5 filter preheating Management of SGs[14] (filling unavailable SGs and maximum cooling on available SGs) Refilling of the pool water treatment tank Restoring power Restoring injection mechanism External communications (MCC to regional authority)
2. Resources for this strategy (depending on the emergency)	To restore power to the valves (I&C):	If the back-up supply LCA is in battery-saving mode: confirm appropriate LCA output in service and trigger the others. If batteries cut off: targeted re-supply of outputs + position jumpers.
	For opening (and confirmation) of valves:	From control room or in situ

Action:

1. Actions necessary (and special sequence if any):	
	(Re-) establish power supply to pressuriser valves Control room-pressuriser room synchronisation for power supply Opening of valves and confirmation of opening under P11.

Status diagnosis:

[14] SG: steam generator

1. Parameters affected	RCIT, primary cooling, SG levels
2. Accident in progress	Core melt in an SBO situation
3. Systems unavailable	Restoring power to back-up distribution system RCIT re-transcription in control room (conservative assumption for L2 PSA modelling) Back-up turbine-driven feedwater pump
4. Systems available	RCV003PO pump, neighbouring unit volume control pump , LLS (in the majority of cases) valve control mechanism using solenoid valve LCA battery: note (see note ENFCFF040119A) that battery life of 8 hours is assumed before the low voltage alarm is reached, then power restored independently by the residual charge (after re-establishing supply) of 10 hours.

Prognosis:

Changes in parameters	RCIT increase, pressure changes, battery charge
Possible aggravating factors Protective counter-measures	Appearance of breaches (primary pump joints) Containment by-pass Application of containment GIAG (containment isolation) +U2
Ineffective or inadequate counter-measures	Restoring electricity source 48V Low voltage back-up distribution system Restoring injection mechanism Following status-oriented approach procedures Restoring coolant source (re-establishing Auxiliary feedwater system tank, restoring Back-up turbine-driven feedwater pump, etc.)

These elements then allow failure scenarios to be built by systematically considering the failure of each PSAD function on the basis of the different possible combinations (it should be noted that the failure of one function at a time is considered):

- Strategy: scenarios with no strategy and scenarios with the wrong strategy (higher priority rival objective, hesitation about resources, etc.)

- Action: scenarios with no action, or incomplete actions, and scenarios with the wrong actions

- Diagnosis: scenarios with no diagnosis carried out, and scenarios with a wrong diagnosis

- Prognosis: scenarios with no prognosis carried out, and scenarios with a wrong prognosis.

The results obtained for the generic severe accident aspect combining LOFW+SBO are as follows (a very brief summary presentation has been selected here, with no details of precise components):

HFE	Depressurisation by opening SEBIM safety valves		Failure probability 1.1×10^{-1}
			Residual 10^{-4}
Fault mode	**Failure scenario**	Number	Probability
No strategy	As the system does not recognise the urgency of the situation relative to the risk of vessel fracture, it suspends the recommended actions until the diagnosis is carried out by the CE-IS GIAG and does not open the valves in time	1	8.1×10^{-4}
	The system suspends immediate actions to be carried out in order to prioritise them	2	2.7×10^{-6}
Wrong strategy	No plausible scenario identified: given the situation and other correct P-A-D functions, no rival objective identified seems plausible to explain the non-completion of the aspect by the operating system.	3	
No action	No scenario: considering the other correct functions, it is not plausible that the system does not initiate the action.	4	
Wrong action	The system does not correctly confirm valve opening	5	8.1×10^{-3}
	The system does not correctly restore power supply to the valves in time	6	8.1×10^{-2}
No status diagnosis	No scenario: given the situation, it seems implausible that the system has not drawn up any status diagnosis	7	
Wrong status diagnosis	The system is waiting for confirmation from the MCC and its technical support to depressurise the primary cooling system, and is not opening the valves (MCC waiting for technical support from local emergency response team which is not available in time)	8	7.3×10^{-4}
	The system is waiting for confirmation from MCC to de-pressurise the primary cooling system and is not opening the valves (MCC not available in time)	9	2.2×10^{-2}
Wrong prognosis	No scenario identified: given the situation and the available time-scales, and considering the other correct functions	10	
No prognosis	No scenario identified: given the situation and the available time-scales, and considering the other correct functions	11	

The quantification procedure is not developed here for lack of space. This involves using expert judgement to evaluate each element making up each failure scenario (context elements, selection of inappropriate control action for the context, what can be restored in the time available). These scenario components are quantified using the following scale of values (taking account of dependencies):

Very probable: 0.9

Somewhat probable	0.3
Unlikely:	0.1
Extremely unlikely:	0.01

The probabilities for the components in a scenario are then multiplied (conditional probabilities) to obtain the probability of the failure scenario occurring. Lastly, the probabilities for all the failure scenarios in the aspect, built separately, are summed and added to a residual probability (1 E^{-4}) to give the probability of failure for this aspect.

The results obtained for the depressurisation aspect of a serious primary cooling system accident combining LOFW+SBO highlight that, even if this aspect is well-known to everyone, its probability of failure is not insignificant given the short timeframe (which particularly penalises actions in situ, see scenarios 5 and 6) and the complicated decision circuit (scenarios 8 and 9).

5. Conclusion

Analysis of the "human factor" of two aspects of Level 2 PSA, by the MERMOS method extended into this area and by knowledge gathering from experts in severe accidents, has enabled the applicability of the method for L2 PSA requirements to be shown. The procedure and its application have garnered favourable opinions from severe accident experts at EDF. This gathering of expert opinion took a day for each of the two examples, but proved very worthwhile. The severe accident experts contributed their knowledge both qualitatively, to describe the various possible ways of failing through failure scenarios, and quantitatively to evaluate the probability of these scenarios occurring, with the help of expert in the MERMOS method structuring the procedure.

The MERMOS method provides clarification and traceability of expert judgements, through a detailed description of possible failure scenarios and justifications for the probabilities selected, and human reliability thus enables knowledge of high-risk socio-technical systems to be capitalised and enhanced [ref 4][15].

These first two analyses produced with MERMOS for L2 PSA have been forwarded to the French nuclear safety authority. They could lead to opting to roll-out the procedure for all PSA, or to using it for specific cases.

[15] [ref 4] Using expert judgments with MERMOS: from static Assessment towards Knowledge Capitalization - P. Le Bot, European Commission DG Joint Research Centre, Institute for Energy & Commissariat à l'Energie Atomique (June 21st - 23rd, 2005 Aix - En – Provence, France)

Session 4

Best-Estimate Calculations of Unmitigated Severe Accidents in State-of-the-Art Reactor Consequence Analyses

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Extended Abstract

This paper discusses the prediction of severe accident offsite radiological releases in conjunction with the U.S. Nuclear Regulatory Commission (US NRC) program entitled, State-of-the-Art Reactor Consequence Analyses (SOARCA).

The US NRC has undertaken the SOARCA program to perform an updated realistic evaluation of severe reactor accidents and their offsite consequences. It is the intent that these analyses reflect the accumulated improved understanding of severe accident behavior and potential consequences developed through the considerable research conducted by the US NRC, the industry, and the international research community over the last 25 years and that the analyses would provide a body of knowledge on the more likely outcomes of such remote events. This information would be the basis for communicating that aspect of nuclear safety to government authorities, licensees, and the general public. It is also an objective that SOARCA would update and replace quantification of offsite consequences documented in earlier studies, that in some cases was based on overly conservative assumptions and simple bounding analyses to the extent the earlier results are unnecessarily conservative and can be misleading.

The approach used in SOARCA has been to (1) use state-of-the-art analytical tools for accident progression and consequence analyses; (2) credit the use of Severe Accident Management Guidelines (SAMGs) and other new plant procedures, such as mitigative measures resulting from security related assessments and other like programs; and (3) use realistic site-specific evacuation scenarios and emergency planning modeling along with updated population and meteorological data. The focus of SOARCA is the application of detailed, realistic scenario-specific and consistent modeling using an integrated severe accident code, MELCOR and the offsite consequence code MACCS2. Analyses have been completed for two pilot plants, the Surry plant, a pressurized water reactor design, and the Peach Bottom plant, a boiling water reactor design. Accident scenarios adopted for analyses were selected by their potential severity for offsite consequences together with the frequency of occurrence.

Evaluation of the accident scenarios was conducted in a two-fold manner. First, the scenarios were assessed with consideration of mitigation measures, including emergency operating procedures, SAMGs and new security related mitigation measures. Secondly, we assessed those same scenarios assuming SAMGs and the new security related mitigation were not implemented in order to assess the benefit of those measures. It was our intent to perform an updated, realistic assessment of potential offsite releases

with full consideration of severe accident research, using detailed, integrated modelling, and fission product phenomenological behaviour revealed in testing programs such as Phebus and ARTIST.

The analysis of the accident progression and radiological release was performed using the MELCOR code, NRC's detailed mechanistic model that incorporates our best understanding of plant response and severe accident phenomenology. MELCOR was used to provide state-of-the-art modelling of the reactor system, containment and auxiliary building (or reactor building) thermal hydraulics, core degradation, fission product release, transport and deposition, and containment leakage/failure criterion. The analyses revealed that realistic best estimate modelling of unmitigated events can yield smaller and delayed radiological releases and that in some instances, e.g., thermally induced steam generator tube rupture, integrated analyses predict that phenomena previously associated with large early releases do not result in large releases.

Deterministic Evaluation of Quantitative Health Objective and Target of Severe Accident Management

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1. Introduction

The Korean government issued a policy statement on severe accident of nuclear power plant on August, 2001. According to the policy utility was required to perform PSA (probabilistic safety assessment), develop and implement accident management programs for operating plants. Also a safety goal was proposed in a form of quantitative health objective (QHO) such that an additive risk of early fatality and cancer fatality caused by accident or operation of nuclear power plant should not exceed 0.1% of early fatality and cancer fatality resulting from other base accidents and cancer mortality, respectively. And it was recommended that a performance goal to achieve this safety goal should be developed.

This quantitative health objective (QHO) of additional 0.1% risk was adopted from the US NRC safety goal and the adequacy of this QHO for Korean nuclear power plants needs to be evaluated. This QHO is rather widely accepted one also in other countries and generally Level 2 and 3 PSA are needed to assess whether a specific plant satisfies this safety goal. Also the developed severe accident management guidelines (SAMG) were reviewed by Korea Institute of Nuclear Safety (KINS). But in reviewing the QHO and SAMG for operating plants, we had some conceptual difficulties.

The first conceptual difficulty came from applying the concept of risk to show public that the nuclear power plant satisfies the QHO. In nuclear business, risk is defined as Risk = Frequency × Consequence. According to this definition, we could show that the risk is not significant in case the frequency is extremely low even though the consequences from a severe accident are huge. Nuclear power plant was shown to satisfy the safety goal using this logic for most cases. But the frequency does not have any meaning for the people living near a nuclear power plant at the time of accident. For that situation, the risk accepted by the public must be considered as Risk = Hazard + Outrageous which is developed for risk communication with the public.

The difference between the definitions of risk is depicted conceptually in Figure 1 below. If we suppose fatality from base accident is 20 in a city with 2000 people, the average risk becomes 0.01(=20/2000) per year. Now suppose that a nuclear power plant operates normally for 49 years but a severe accident occurs at 50th year and mortality rises up to 1000. In this case we, the nuclear community, calculate the risk to be 0.01 (=1000/2000/50) per year, but the risk recognized by the public living near the plant at the time of accident is just 0.5 (=1000/2000), 50 times higher than our estimation. So the acceptance of the current QHO developed on the premise of the above logic needs to be critically evaluated.

Figure 1. Concept of Fatality

For the public to accept that the risk of nuclear power plant is low, therefore, we need to evaluate the consequence caused from severe accident by removing the frequency multiplication factor. This is why we tried to evaluate the consequence from accident deterministically. Deterministic evaluation means we are following the progression of severe accident as it occurs and simulate the release of source terms through leak area which could be calculated from a structure analysis of containment.

The second difficulty came from assessing whether the implemented SAMG could really increase the capability of plant in handling the severe accident or not. The Korean SAMG was developed referencing the Westinghouse Owners Group SAMG and the basic philosophy of the WOG SAMG is to do one's best with the equipments available at the plant. The thing is that the operating plants are designed not taking into account the severe accident, so there are no specific engineered safety features (ESFs) against severe accident in most cases. This means that it is quite uncertain whether a specific strategy recommended in the SAMG could really prevent or mitigate an accident even though the SAMG are useful in a sense that it is a well structured and software like procedure. Also because a target of accident management is not clearly defined, it is hard to conclude whether the accident management activities satisfy the target and thus are acceptable or not.

These two main difficulties made us to think seriously on what should be the target of accident management to show the public that nuclear power plant is safe. In this paper we will show a preliminary result of our evaluation efforts tried from this point of view.

2. Quantitative Health Objective (QHO) of Different Countries

In this section we will describe how the Korean QHO has been derived. The QHO of U.S. and Japan are also explained and compared each other.

2.1 Korean Quantitative Health Objective

In order to assess the accident fatality risks and cancer fatality risks of operating nuclear plants compared to the health objectives of severe accident policy, we have surveyed Korean statistics of mortality from 1983 to 2006 using Korean statistical information service (KOSIS) [(1)] of National Statistical Office. The data on accident and cancer mortality by year using analysis results on the cause of death allows us to calculate the acceptable risks of safety goals based on these mortality data. The QHO could be calculated as 0.1% of these mortality data and figure 2 shows the acceptable risks of early and cancer fatality by year we got. Average acceptable risks of early and cancer fatalities during 24 years could be defined as 6.935×10^{-7} and 1.115×10^{-6}, respectively. But it is conceptually confusing whether it is right to take the average value of fatalities in this way because the mortality data change through out the period of data accumulation.

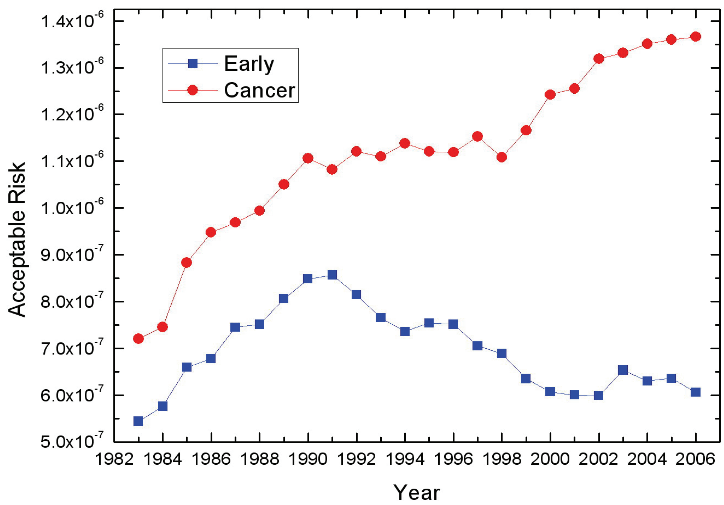

Figure 2. Derived Acceptable Risks by Year

2.2 U.S. Quantitative Health Objective

Since United States Nuclear Regulatory Commission (U.S. NRC) started making an effort to set safety goals in 1979, U.S. NRC issued the policy statement through 6 years evaluation period. Safety concept of U.S. had been basically kept by guarantee of sufficient safety margin with Defense-In-Depth principle. But this concept could not apply to the question of 'how safe is safe enough'. Also, there was no quantitative analysis methodology of light water reactor before WASH-1400 in 1975.

After the accident at Three Mile Island, Advisory Committee on Reactor Safeguards (ACRS) of U.S. NRC recommended that quantitative safety goals of nuclear power plants should be established. And the President's Commission on the Accident at Three Mile Island and the NRC's Special Inquiry Group recommended that safety goals and philosophy should be represented more clearly and announced to the public. Following that, U.S. NRC announced the plan for the development of safety goals and ACRS suggested trial approach for development of safety goals.

U.S. NRC held workshops on April and July in 1981 for the development of formal safety goals policy and issued safety goals policy statement for comment on February, 1982. Reflecting the opinions of ACRS, industry, and public U.S. NRC adopted safety policy statement that would be used during 2-year evaluation period on 14th March, 1983. After that, policy statement on safety goals was announced on 4th August, 1986.

The individual mortality risk of prompt fatality in the U.S. is about 5.0E-4 per year for all accidental causes of death [(2)]. Thus, on the average, approximately 5 persons out of 10,000 die annually as a result of accidents in the U.S. The prompt mortality risk design objective would limit the increase in an individual's annual risk of accidental death (5 in 10,000) by an increment of no more than 5 in 10,000,000 per year.

On the other hand, roughly 19 persons per 10,000 population die annually in the U.S. as a result of cancer on the average. The risk of developing a fatal cancer is subject to large variation depending on geographic and demographic factors [(3)].

2.3 Japanese Quantitative Health Objective

In case of nuclear power plants, safety assurances by licensee and safety regulations by government are based on Defense-In-Depth principle, which considers 3-step safety measures as prevention of abnormal condition occurrence, prevention of abnormal condition extension and expansion to accident, and prevention of abnormal release of radioactive materials.

Japanese nuclear safety inspection guidelines and criteria do not mention the risk restriction level to the public quantitatively except radiation limits during the normal operation of nuclear power plant. Japanese nuclear safety commission decided that effective safety assurance could be possible if safety goals of probabilistic risk concept as risk restriction measures that could be achieved by nuclear safety regulatory activity are used to make a decision about safety regulatory activity. Accordingly, safety goals special group composed of expert advisers of various fields was founded on September, 2000.

Safety goals special group has investigated and reviewed the concept of safety goals and submitted an interim report to Japanese nuclear safety commission. Safety goals special group held panel forums to explain the meaning of safety goals and to collect citizen's views at Tokyo on July, 2002 and at Kyoto on October, 2002. Safety goals special group examined forum results and made progress on more deep inspection. And now, interim safety goals are determined, but it does not enforce any legal binding.

2.4 Comparison of Quantitative Health Objectives

Quantitative safety goals of different countries are compared in Table 1. Quantitative safety goals of Korea and of U.S. are similar and Japanese quantitative safety goals has a different feature in that performance objectives supporting health objectives are set. The QHOs shown in concrete numbers are marked in Figure 3.

Table 1 : Different Quantitative Safety Goals

		Korea	U.S.	Japan
Health Objectives	Early	< 0.1 %	< 0.1%	$< 10^{-6}$
	Cancer	< 0.1 %	< 0.1%	$< 10^{-6}$
Performance Objectives		N.A.	N.A.	CDF $< 10^{-4}$ CFF $< 10^{-5}$

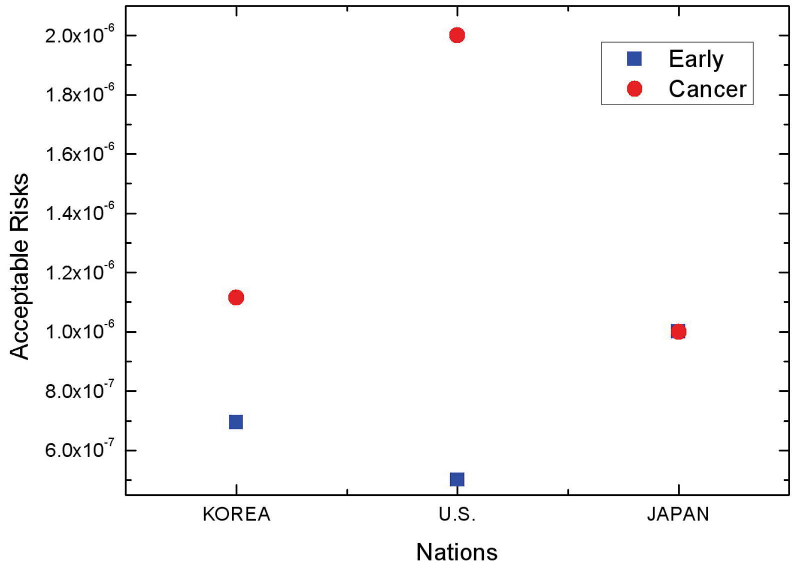

Figure 3. Comparison of Korean, U.S. and Japanese Health Objectives

3. Evaluation of QHO using MELCOR-MACCS2 Code Package for Ulchin Unit 3&4

The QHO of Ulchin unit 3&4 plant was assessed for severe accident scenarios using MELCOR 1.8.5 [(4)] and MACCS2 [(5)] codes. At first try, we have assessed the QHO for LBLOCA, SBLOCA and SBO accident scenarios just to have a rough value on the magnitude of fatalities. After an initiation of accident, neither engineered safety features nor operator actions are assumed for simplicity of calculation. For a second analysis, we have chosen SBLOCA accident because it is the main contributor of source term category (STC) 6 according to Ulchin PSA [(6)] and tried to simulate the accident progression as much realistic as possible. The STC 6 represents an average source term category both from the frequency and the consequence of the accident event. In this SBLOCA scenario, we assume that both the high pressure safety injection (HPSI) and low pressure safety injection (LPSI) pumps are available initially but the recirculation fails and the accident progresses to a severe accident.

The characteristic of this calculation is to follow the accident progression as it occurs and then simulate the source term release using the containment leak model available. This process removes the frequency multiplication factor and could calculate the risks as viewed from the public. The accident progression and the source terms are calculated using the MELCOR 1.8.5 code and the fatalities are calculated using MACCS2 code. These evaluation processes are explained in the following sections.

The Ulchin unit 3&4 plant is a two loop plant of 2826 MWt with 2 steam generators, 1 pressurizer and 4 reactor coolant pumps. The MELCOR model for Ulchin 3&4 is shown in the Figure 4 below. The modeling is a typical one KINS uses for its regulatory audit calculation.

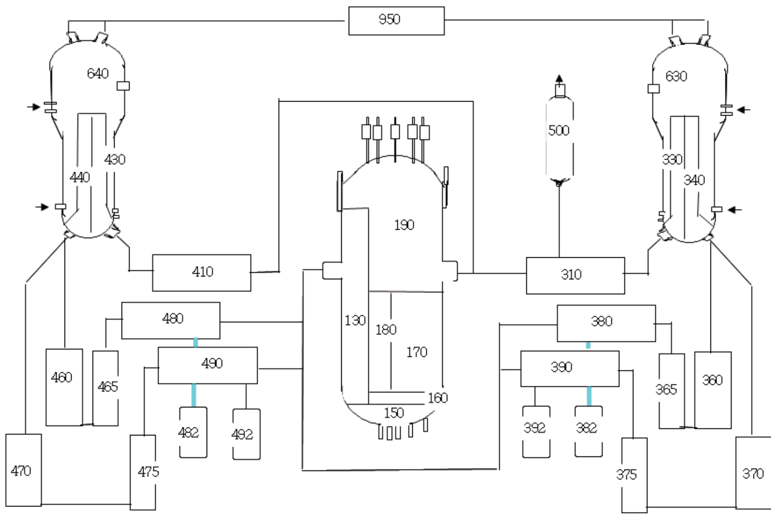

Figure 4. MELCOR modeling of Ulchin 3&4

3.1 ORIGEN-S

The basic data of the source term for the offsite consequence analysis are the core inventory of in-core

Table 2 : Accident Core Inventory of Ulchin 3&4 NPP

Radionuclides	Radioactivity(Bq)	Radionuclides	Radioactivity(Bq)	Radionuclides	Radioactivity(Bq)
Kr-85	2.98977E+16	Te-129m	1.43E+17	La-142	3.91E+18
Kr-85m	5.63937E+17	Te-131m	5.33E+17	Nb-95	4.1E+18
Kr-87	1.09641E+18	Te-132	3.71E+18	Nd-147	1.67E+18
Kr-88	1.45107E+18	Sr-89	2.04E+18	Pr-143	3.65E+18
Xe-133	5.24883E+18	Sr-90	2.31E+17	Y-90	2.42E+17
Xe-135	1.23993E+18	Sr-91	2.63E+18	Y-91	2.75E+18
I-131	2.62269E+18	Sr-92	2.9E+18	Y-92	2.93E+18
I-132	3.81363E+18	Co-58	4.2E+13	Y-93	3.41E+18
I-133	5.23848E+18	Co-60	3.82E+14	Zr-95	4.06E+18
I-134	5.82636E+18	Mo-99	4.75E+18	Zr-97	4.24E+18
I-135	5.02458E+18	Rh-105	3.05E+18	Ce-141	4.14E+18
Cs-134	5.29092E+17	Ru-103	4.47E+18	Ce-143	3.74E+18
Cs-136	1.62219E+17	Ru-105	3.34E+18	Ce-144	3.34E+18
Cs-137	3.36651E+17	Ru-106	1.79E+18	Np-239	5.89E+19
Rb-86	5.79186E+15	Tc-99m	4.21E+18	Pu-238	9.76E+15
Sb-127	2.58888E+17	Am-241	4.07E+14	Pu-239	8.99E+14
Sb-129	7.9281E+17	Cm-242	1.47E+17	Pu-240	1.51E+15
Te-127	2.54955E+17	Cm-244	1.38E+16	Pu-241	4.26E+17
Te-127m	4.34493E+16	La-140	4.59E+18	Ba-139	4.55E+18
Te-129	7.4313E+17	La-141	4.1E+18	Ba-140	4.39E+18

fission products and its release fraction according the accident. First, ORIGEN-S was used for the core inventory calculation. To be conservative in the calculation, we assumed that one year is 1 cycle and the fuel burns up continuously during 1095 days, 3 cycles without the cooling term. Also it was assumed that the core thermal power is 37.5 MWth per UO_2-ton, the total amount of UO_2 is 78 ton, and the uranium enrichment is 3.5w/o U-235. Table 2 shows the accident core inventory of Ulchin unit 3&4 nuclear power plants as the ORIGEN-S provides.

3.2 Modeling of Containment Leak

The source terms in the containment are released either through leak or ruptured area of containment. In our calculation this is simulated by assuming that the source terms are released either by 0.1 vol%/day leak rate assumed in integral leak rate test (ILRT) or through the ruptured area of containment at the given rupture pressure.

The rupture area at high pressure comes from the structure analysis of the containment. According to the structure analysis using ABACUS for Ulchin plant, the liner plate tearing near equipment hatch occurs first at the median pressure of 169 psig. The rupture area at this pressure is 6.0 in^2. Also the lower limit with 5% probability of this rupture pressure could be calculated from $P_m \exp(-1.65\beta u)$ and the value is 132 psig. Thus with conservatism, it is right to assume that a rupture of area 6.0 in^2 occurs at 132 psig.

The leak through containment is simulated assuming that at the design basis pressure P_{DBA} of containment, the 0.1 vol%/day leak is occurring. During an Integrated Leak Rate Test (ILRT), the containment structure is maintained at test pressure with most penetrations isolated. The leak rate test performed on the containment by simulating some of the conditions (e.g., penetrations vented, drained, flooded, or in operation) that exist during a design-basis accident (DBA) which results in the maximum primary containment internal peak pressure and in fission product release to the containment atmosphere.

In case of Ulchin plant, the calculated peak containment internal pressure related to design-basis loss of coolant accident, P_{DBA}, is 57psig. And the maximum allowable Type A test leakage rate at P_{DBA}, La (%/24 hours), is expressed in terms of percent weight per day, specified in the Technical Specifications as 0.1 percent by weight/day of containment air.

Using the atmospheric condition at P_{DBA} (57 Psig), the corresponding total dry air mass escaping from a leak or leaks can be determined directly utilizing the Ideal Gas Law (PV = mRT). The rate of change of air mass shall be converted to the leakage rate in units of percent per day. Assuming the steady condition (constant P and T), the change rate of volume is equal to the change rate of mass. Therefore, the 0.1 vol %/day can be expressed as 0.1 mass %/day and the equivalent mass flow rate is 0.0011 kg/sec where, V(Containment Volume) is 7.76E4 m^3, density of air is 1.2 kg/m^3 .

In MELCOR modeling, the flow path for containment leakage is modeled by a single flow path, which is located in annual compartment region at an elevation of 11m from ground level. The containment leakage area used in MELCOR when the containment pressure reaches P_{DBA} is determined as 1.0E-5 m^2 . The calculation is performed using the following equations taking into account the rate of mass leakage and pressure difference between containment and environment.

$$\dot{m} = \rho A v \,,\ A = \frac{\dot{m}}{\rho v},\ v = \sqrt{\frac{2(P_2 - P_1)}{\rho}}\,,\ A = \frac{\dot{m}}{\sqrt{2\rho(P_2 - P_1)}}$$

The leak is modeled to occur directly to the environment without filtration. The actual leakage rate of mass through flow path at P_{DBA} is 0.002 kg/sec in the MELCOR simulation. Considering that the density in annual compartment of containment is higher by about 2 times than that of dry air, MELCOR model for containment leak at design pressure, P_{DBA} is reasonable.

3.4 MACCS2 Calculation

The offsite consequence analysis was performed using MACCS2 code when assuming a hypothetical severe accident of Ulchin unit 3&4 plant. The core inventory and release fraction that were calculated previously was used as source terms and in other input parameter case, site specific data were used referencing the final safety analysis report of the Ulchin units 3&4 plant. The models in MACCS2 are implemented in three modules:ATMOS, EARLY, and CHRONIC. Figure 3 shows the structure of a MACCS2 consequence calculation.

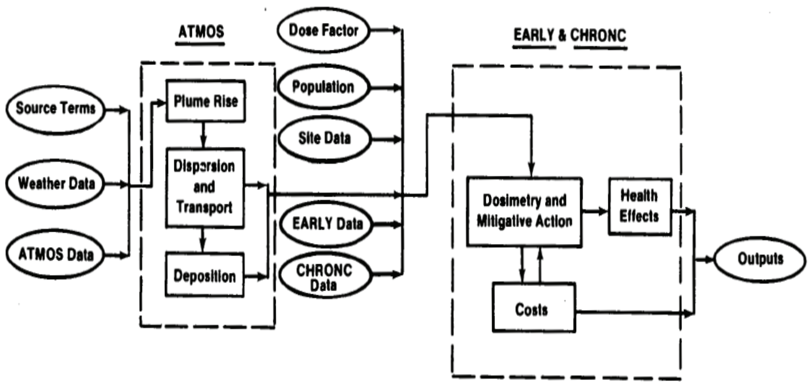

Figure 3. Structure of a MACCS2 consequence calculation

4. Evaluation of QHO and Target of Severe Accident Management

With the above MELCOR and MACCS2 inputs developed for Ulchin plant, we have calculated the early and cancer fatalities and tried to compare with the QHO value. As explained in section 2.1 the early and cancer fatality risks deduced from 24 year national statistics are 6.9E-7 and 1.1E-6, respectively.

As is well known, the MACCS2 code has a lot of limitations in simulating the source term release from containment in a sense that it can model only 4 plumes and also that the release duration is limited to 24 hours. Our experience of sensitivity analyses shows that the calculated values are highly dependent on what values we give for an energy of the plume, the location of the plume released and also the input options of the code. So in our evaluation of QHO, we will not give much credit to the calculated values itself. We will try to derive qualitative information from the analyses and thus verify our idea on determining whether our accident management could satisfy the QHO and also what should be the target of severe accident management. This evaluation process should be taken as so called "Gedanken experiment".

4.1 Initial Evaluation of Fatalities for Different Accidents

To get a rough estimation on the fatalities for different accident scenarios, we have selected the LBLOCA, SBLOCA, and SBO as the representative hypothetical severe accident event referring to the probabilistic safety assessment report [(6)] for the Ulchin unit 3&4 plant. We are assuming that after the initial event, no engineered safety features are available. This means that all the pumps or valves or what so ever are not available after initiation of the accident. These accident sequences were computed using

MELCOR code for the selected representative event. When the containment pressur reaches P_{DBA} then the source terms and energy of radioactive nuclide are released through 6.0 in^2 (3.9E-3 m^2) flow area out of the containment. Then the release data are used as inputs for MACCS2 code. The fatalities calculated are given in Figure 4 below. The results show that the LBLOCA only contributes to the early fatality, but the SBLOCA and SBO have higher probability of occurrence according to the UC N PSA. Neither early nor cancer fatalities could satisfy the QHO in these cases.

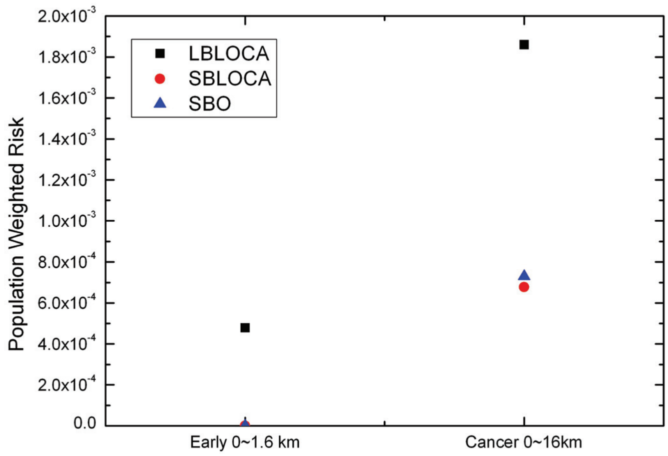

Figure 4. Population Weighted Risk of Severe Accident

4.2 Sensitivity Evaluation for SBLOCA Accident

Next we performed somewhat more detailed and realistic evaluation for SBLOCA accident. The detailed accident scenario is (SBLOCA * Rx Trip *HPSI Injection *AFW *MS ADV /HPSI Recirculation *RCS Depressurization Using Aux.Feed /LPSI Recirculation /Spray Recirculation) where * means the function works successfully and / means the function fails. This accident scenario makes reactor vessel to fail at 1.66E5 seconds and the containment pressure increases since then. The source term release from containment is modeled by one plume segment. There is no evacuation of public and also the relocation is suppressed by choosing the MACCS2 option parameter SRDOSHOT001 to be 1.0E10.

Table 3. Average Individual Risk for Different Cases

Cases	Average Individual Risk
1. leak through 0.1 ft^2 (9E-3 m^2) at 132 psig and sprayed at 10 hrs later	CAN FAT. / 0-1.6 km 5.54E-2 CAN FAT./ 64-80 km 3.13E-5
2. leak through 0.01 ft^2 (9E-4 m^2) at 1.82E5 sec. (pressure reaches P_{DBA} at 2.3E5 sec)	CAN FAT. / 0-1.6 km 4.45E-3 CAN FAT./ 64-80 km 1.59E-6
3. leak at the rate of 0.1 vol %/day (1E-5 m^2) at P_{DBA} and sprayed at 5.E5 sec. (8000 seconds before containment failure)	CAN FAT. / 0-1.6 km 2.64E-3 CAN FAT./ 64-80 km 2.53E-7

The average individual risks for different cases are compared in table 3 above. In MACCS2 calculation, the rate of release of sensible heat in plume segment, PLHEAT, is determined as a mean value of the amount of sensible heat in the plume segment by the duration of the calculation time (period). The duration time of the plume segment, PLUDUR, is specified by 1 day (86400.0seconds) which is maximum value allowed in MACCS2.

Remembering that the domestic QHO of cancer fatality is 1.115E-6 and comparing this QHO with the results of table 3, it becomes clear that there is no way to satisfy the current QHO. The case 2 simulates a venting strategy to prevent the containment failure and the pressure behaviour is shown in Figure 4 below.

In case 2, the venting with $0.01ft^2$ area was started at 1.82E5 second before the containment pressure reaches P_{DBA}. The containment pressure does not increase up to 132 psig until 7.0E5 second, so this venting capacity might be effective to control the containment pressure and in preventing the containment failure when other actions like spray actuation becomes available in appropriate time.

Sensitivity calculation using 0.05 ft^2 leak area shows that the pressure is controlled. Thus to control the containment pressure, venting of containment through certain size of leakage could be effective , but the above calculation clearly shows that in case the pressure rises higher than P_{DBA} , the QHO could not be satisfied.

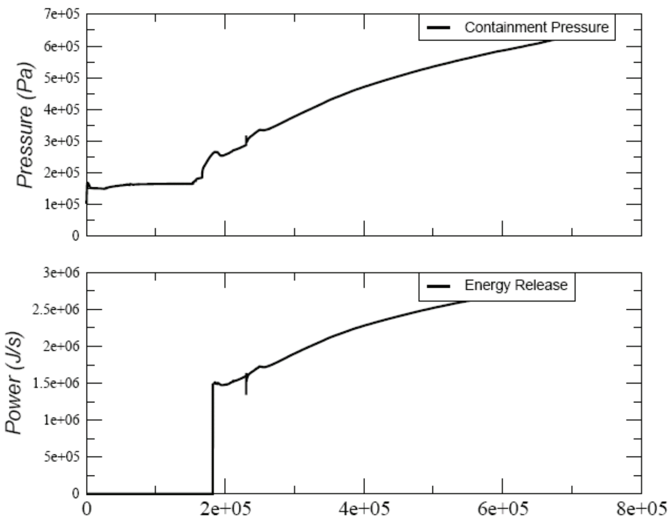

Figure 5. Containment Pressure for Case 2 Calculation

Now we have performed more sensitivity calculation using MACCS2 for case 3 above. In case 3-1, the duration of plume release is kept on until the spray is actuated at 5.E5 sec. When the spray is actuated successfully, the containment pressure might be reduced enough to stop the release of plume. Considering the spray actuation timing, the duration of plume release is modeled by 12 hours and 3 hours for case 3-2 and 3-3, respectively.

Table 4 shows that the result of case 3-3 approaches the QHO of 1.115E-6. The insights we could derive from these analyses are that the only way we could satisfy the QHO is to activate the spray in 3 hours after the containment reaches P_{DBA}. This insight could be paraphrased such that the containment pressure should be maintained below the P_{DBA} even during severe accident and this should be the target of our accident management viewed under the current QHO. Or the QHO should be revised.

Table 4. Sensitivity Analysis for Case 3

Cases	Average Individual Risk
3-1 leak at the rate of 0.1 vol %/day (1E-5 m^2) at P_{DBA} (2.3E5 sec.) and sprayed at 5.E5 sec. (containment pressure increases to 132 psig at 5.8E5 sec.)	CAN FAT. / 0-1.6 km 1.67E-3 CAN FAT./ 64-80 km 2.53E-7
3-2 leak at the rate of 0.1 vol %/day at P_{DBA} and sprayed at 2.732E5 sec(12 hr after leak begins).	CAN FAT. / 0-1.6 km 1.57E-5 CAN FAT./ 64-80 km 2.12E-9
3-3 leak at the rate of 0.1 vol %/day at P_{DBA} and sprayed at 2.408E5 sec (3 hr after leak begins).	CAN FAT. / 0-1.6 km 6.63E-6 CAN FAT./ 64-80 km 9.27E-10

5. Conclusion

We have evaluated whether risks from severe accident could satisfy the current QHO in a deterministic way. We have used MELCOR 1.8.5 and MACCS2 codes to simulate the accident progression and the individual risk caused by release of source terms for Ulchin unit 3&4 plant in Korea. Some insights we could derive from these analyses are the following.

1) The QHO has been derived from the Korean statistical data of prompt and cancer fatalities. The 24 year averaged value gives that the early and cancer risks should be 6.94E-4 and 1.115E-6, respectively.
2) Venting strategy using certain size of venting area could be effective in controlling the pressure to prevent the containment failure, but results cannot satisfy the QHO in case it is not a filtered one.
3) The uncertainties of MACCS2 code are too high in using quantitative value, but in case the containment pressure remains below P_{DBA}, then we could have a chance of satisfying the QHO even though we assume that leakage at the rate of 0.1 vol%/day occurs at P_{DBA} .
4) The conclusion of 3) could be paraphrased such that the containment pressure should be maintained below P_{DBA} even during severe accident. Thus target of our severe accident management activities should be this one when we want our plant to satisfy the current QHO.
5) What the accident management activities should be to satisfy this target might be different for different accident scenarios and for different plants. But having a quantitative target of accident management satisfying the QHO could provide more logical framework to develop strategies and also to convince public.

In conclusion, we can satisfy the current QHO only in case we could control the containment pressure below P_{DBA} during severe accident. Thus for a successful accident management viewed under the current QHO, we should have a way to control the pressure. Otherwise, we need to revise the current QHO which is determined partly from the public acceptance point of view as far as we understand.

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Verification of the SAMG for Paks NPP with MAAP Code Calculations

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1. Introduction

Paks NPP implemented a severe accident management program for the VVER-440/213 units. This program includes plant modifications and development of procedures.

The project for the development of Severe Accident Management Guidelines (SAMG)[1] was launched by Paks NPP with the lead of Westinghouse Electric Belgium Co. As a complementary effort, a domestic project on the verification of the SAMG[2] was initiated. The goal of this project was to check and support the development of SAMG.

The verification of the SAMG was performed with severe accident analyses using the MAAP4/VVER code[3]. The examined accident management procedures involved:

- depressurisation of the primary circuit,
- water injection into the primary system,
- in-vessel melt retention by external cooling of the vessel,
- preventing excessive vacuum in the containment,

[1] T. Rensonnet, B.Bognár, F. Medgyesy: Paks Nuclear Power Plant Severe Accident Management Guidelines Strategy Report, WENX/06/13, April 2008

[2] G. Lajtha, Z. Techy: Verification of the SAMG for Paks NPP with computer simulations, VEIKI 21.22-270, December 2008

[3] M. Van haesendonck, R. Prior: MAAP4/VVER User Guide, WENX 93/25, March 1996

- preventing containment overpressure,

- decreasing fission product release using the ventilation systems.

MAAP4/VVER code calculations were performed for severe accident sequences with assumption of SAMG actions.

2. Depressurization of the primary system

Depressurization of the primary system by means of the pressurizer safety valves and the relief valve (PORV) is the last step of the Emergency Operating Procedures (EOP) as a response to inadequate core cooling. When this procedure is not successful, and the core outlet temperature keeps increasing, then the operators leave the EOP and enter into the transition to the SAMG. Then primary system depressurization is tried again within the SAMG. If all previous operator actions have failed, then the personnel is directed to open all other reactor coolant system (RCS) vent paths to the containment, and to depressurize all intact steam generators to atmospheric pressure. The SAG 1 guideline contains instructions for evaluating the availability of equipment necessary to depressurize the RCS.

In the SAMG the procedure for the primary system depressurization will be performed when the primary pressure is higher than a certain setpoint. The current severe accident guideline (SAG 1) is entered from the Diagnostic Flow Chart (DFC).

Table 1. Time of the vessel failure and corresponding primary system pressure for the PDS_05C sequence

Time delay from the signal of core exit temperature 550 °C	1 PORV	1 PSV	1 PORV +1 PSV	2 PSV	3 valves of pressur.
+10 min		32373 s **23,1 bar**	54040 s **2.6 bar**	48594 s **2.1 bar**	41871 s **2,1 bar**
+20 min	28987 s **31 bar**	29730 s **25,6 bar**	54736 s **2.2 bar**	44252 s **2,2 bar**	43084 s **2,3 bar**
+40 min			59138 s **3,4 bar**	48594 s **2,1**	
+80 min			60574 **2,1 bar**		
+100 min			63669 s **2.2 bar**	50180 **2.5 bar**	49362 s **2 bar**
+120 min				24357 s **132 bar**	24357 s **132 bar**

Green indicates the successful, red the unsuccessful range

In the verification study the SAG 1 guideline was checked. The purposes of SAG 1 are:

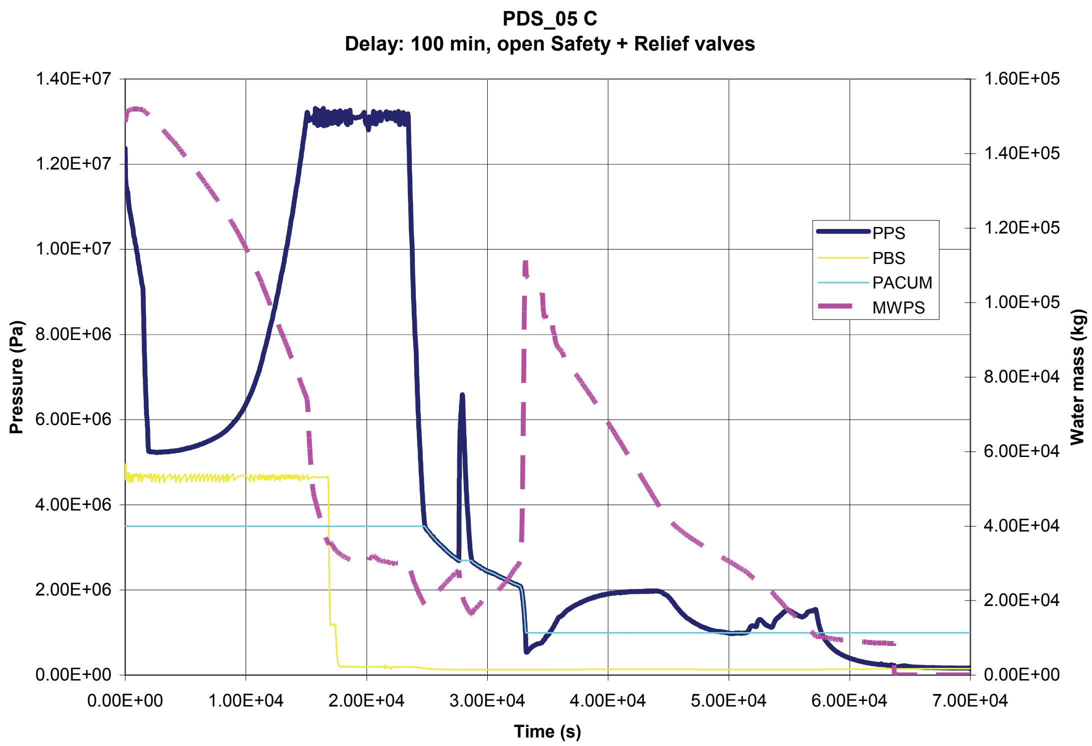
Decreasing the potential of a high pressure melt ejection (HPME) and creep rupture of steam generator tube.

Making available injection sources into the primary system at lower system pressure.

Initial events with primary break size larger than 40 mm do not lead to HPME or steam generator tube creep rupture. The primary depressurization is important for very small break LOCAs without secondary side cooling, therefore only these sequences were examined.

The break size for base case sequence was selected according to the Level 2 probabilistic safety analysis. The base case is a sequence with 11 mm equivalent break size without ECC and secondary heat removal (PDS_05C according to Level 2 PSA[4] classification), parameters of this case are shown on Fig.1.

Figure 1. Primary system pressure (PPS) and water mass (MWPS) in the vessel due to primary depressurization at 100 min after the severe accident signal



Timing of primary depressurization and the different vent paths (depressurization area) were selected for the verification study. The two main purposes of the primary system depressurization were examined. Table 1 shows that opening minimum 2 valves of the pressurizer before 100 minutes after the severe

[4] Level 2 Probabilistic Safety Assessment of the Paks Nuclear Power Plant, Final Report, August 2000- December 2003, Budapest, December 2003, AEKI-PSA2-2003-778-04-11, VEIKI 20.11-214

accident signal is sufficient to avoid high pressure vessel failure and this intervention is enough to get a chance for water injection by low pressure injection system.

Other possible depressurization paths were also checked. The effect of opening of different numbers of gas letdown valves is shown in Table 2. The opening time is 1 hour after the severe accident signal, which is a realistic assumption for the intervention time. It was found that a vent area with about 20 mm diameter is necessary to avoid the high pressure melt ejection and more a vent area larger than 40 mm diameter on the primary system should be open for the successful actuation of the low pressure injection.

Advantages and shortcomings of the interventions were also checked. For example the increased hydrogen production can be seen in Table 2. The findings confirmed the suggestions of the SAG 1 guideline. The primary system depressurization is one of the most important accident management actions.

Table 2. Different interventions in case the PDS_05C sequence

60 minutes delay from severe accident signal						
			Primary system pressure			
	Letdown cross section	Equivalent diameter (mm)	Lower gridplate failure	Vessel failure	Hydrogen mass kg	Time of the reactor vessel failure (s)
1	1.5386E-04	14	129	109	327	25876
2	3.0772E-04	19.79898987	113	82	330	26530
3	4.6158E-04	24.24871131	97	37	332	27711
4	6.1544E-04	28	84	34	326	28108
5	7.6930E-04	31.30495168	71	34	327	28192
6	9.2316E-04	34.2928564	63	32.6	327	28340
7	1.0770E-03	37.04051835	47	31.5	325	28943
8	1.2309E-03	39.59797975	35	16	324	60826
9	1.3847E-03	42	34	16	350	58405
10	1.5386E-03	44.27188724	32.8	1.7	361	84963
11	1.6925E-03	46.43274706	32.8	1.7	361	80404
12	1.8463E-03	48.49742261	32.25	1.5	360	55583
14	2.0002E-03	52.38320341	30	1.5	343	53660

green: effective primary system depressurization, red: the letdown cross section is not sufficient to reach low pressure

3. Water injection into the primary system

SAG 3 is a guideline for water injection after the depressurization of the primary system and it can be used only after the core heat up. MAAP4/VVER code calculations were performed until the lower support plate failure. It should be taken into the account that the uncertainty of the code calculations is increasing with the progression of core degradation.

The calculated sequences were selected as the dominant sequences of the Level 2 PSA study –PDS_05C (11 mm LOCA with loss of ECC and secondary heat removal) and PDS_02A (20 mm LOCA with ECC recirculation failure and loss of secondary heat removal). The selected 2 sequences cover about 80% of contribution leading to core melt. In addition a LBLOCA sequence was also studied as the fastest sequence leading to core melt.

Calculations were done with and without other accident management processes. The following interventions were taken into account: primary system depressurization, cavity flooding and vessel cooling from outside. These accident management processes can influence the advantages and shortcomings of the water injection into the primary system.

In the course of the verification calculations the SAG 3 guideline was followed step by step. Engineering judgement and assumptions were used for the timing of the procedure steps. In some cases MAAP calculations were performed to determine the time needed for the different steps. Timing of the water injection was assumed at different core states:

after core heat up but before melt down,

after melting but before the lower support plate failure or

when the core debris was relocated into the bottom of the vessel.

Table 3. Different interventions in case of PDS_05C sequence

LPIS restoration time (h)	Lower support plate failure (h)	Hydrogen production until the support plate failure and at the end of the calculation (kg)	Debris mass in the bottom of the reactor vessel (t)
7	-	(246)	-
8	12,7	177 (252)	20
9	9,4	220 (248)	40
10	9,4	220 (251)	40
11	9,4	220 (274)	40
12	9,4	220 (302)	47
12,7	9,4 (Vessel failure:12,7)	220(282)	80

The possibility of water injection and the effect of the different flow rates of injected water into the reactor vessel were examined for different core states.

Two sets of results are highlighted from the huge number of MAAP4/VVER calculations. Table 3 represents the effect of the recovery time of the low pressure injection system (LPIS). The produced hydrogen mass is strongly influenced by the starting time of the water injection.

The conclusion is that LPIS recovery should be started as soon as possible to avoid the increase of hydrogen production.

In another set of calculations the effect of the injected water mass flow rate was examined in the LBLOCA case (Table 4). It was found that injecting water with lower mass flow rates did not increase significantly the hydrogen production. Another result was that vessel failure could be avoided with lower water injection flow rates, than the amount necessary for the decay heat removal. Heat loss of the primary system and the fission product relocation from the core explain this result.

It was found in the study that the negative impact of water injection into the primary circuit was not as critical, as it was suggested in the SAG 3. If less water is injected, than the necessary amount to cool down the core, then the additional hydrogen production is just around 10%.

Table 4. Effect of water injection on hydrogen production and timing of events for a LBLOCA sequence

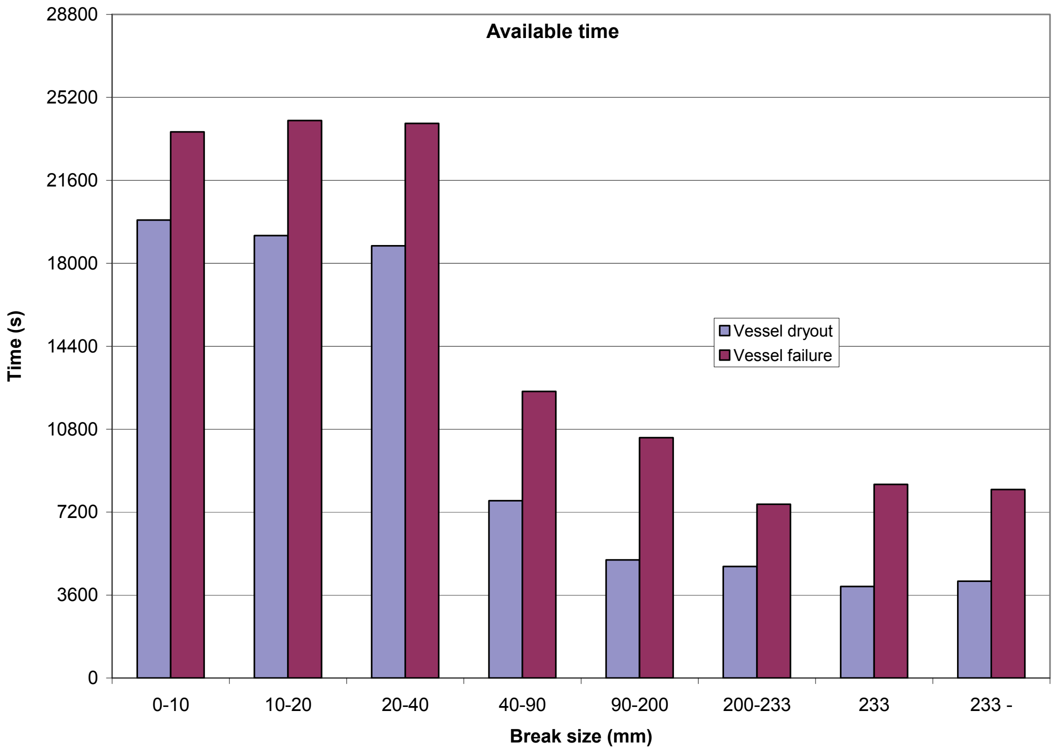
LBLOCA Event	**Water injection rate**			
	0 t/h	**6t/h**	**12 t/h**	**18 t/h**
Core uncovery:	21 s			
Core uncovery II:	1962 s			
$T_{gas\ at\ core\ exit} > 643$ K	2107 s			
$T_{gas\ at\ core\ exit} > 825$ K	2500 s			
Core melt starts	2938 s			
Water injection starts	No injection	3098 s	3098 s	3098 s
Lower plate failure	6031 s	6000 s	4927 s	3988 s
Vessel failure	10842 s	10975 s	12038 s	No failure
Hydrogen production				
At lower plate failure	209 kg	207 kg	213 kg	238 kg
At vessel failure	240 kg	239 kg	268 kg	243 kg

4. In-vessel melt retention (IVR) by external cooling of the vessel

The main points of this severe accident procedure (SAG 2) are the water letdown from the bubbler condenser into the sump, then from the sump into the cavity. First the calculated severe accident sequences were selected. Two representative sequences of the most probable vessel failure event (PDS_05C and PDS_02A) and two LBLOCA type sequences - with 500 mm and 200 mm break sizes – were modeled with the MAAP code. Calculations were performed with different assumptions. First the sequences were calculated without operator intervention called base cases. Then the sequences were calculated with different operator intervention timing for flooding the cavity.

The main examined parameters in the calculations are the available time until the vessel flooding, the pressure in the vessel, the corium mass and decay heat in the bottom of the vessel. The available time frame is the time from the opening signal for the bubbler condenser letdown valves to the vessel wall heat up. Vessel wall dry-out and heat-up was modeled with the MAAP code. The time delay between the vessel dry-out and vessel failure is considered as the safety margin of the intervention. There is quite a large safety margin, because the vessel failure without cooling occurs about an hour later from the time of the dry-out. The calculated safety margins were between 45 minutes to 60 minutes depending on the sequence. In some cases the time margin is quite useful, as the water letdown from the bubbler condenser to the reactor cavity also takes a certain time.

Figure 2. Available time for operator intervention until vessel dry-out and vessel failure



The first calculated sequence was a large break LOCA with 500 mm diameter, representing a special case, because the water flows back from the bubbler condenser to the sump due to the negative pressure difference. In this case the operator should open just the valves between the sump and the cavity. This action should be initiated after entering into SAG 2. The time between the severe accident signal and entering into SAMG is short. The primary pressure is low, therefore the first action of the Technical Support Center (TSC) is to perform SAG 2, i.e. the cavity flooding. The water is available in the sump in a few minutes after the initial event, therefore just the decision-making is necessary. The severe accident signal occurs in an hour after the initial event. This sequence points to the importance of the availability of the TSC team in an hour into the accident. The calculation shows that the vessel can be cooled from outside even in case of the LBLOCA with largest break size, if cavity flooding has been initiated at appropriate time.

The second calculated sequence was a 200 mm diameter LOCA case. In this case the water cannot flow back from the bubbler condenser, therefore the letdown valves should be opened to drain the water inventory to the sump. The lessons learned from the result of these calculations were that the letdown valves had to be opened by the operators before entering into the SAMG, as a last step of the EOP.

Fig. 2 shows the available time window to flood the cavity in case of LOCA accidents with different break sizes.

The most probable sequences according to the Level 2 PSA are those associated with a very small LOCA. For these sequences the available time for flooding the cavity before the vessel dryout is more than 18000 s, or 5 hours, so there is ample time to perform the action.

5. Preventing excessive vacuum in the containment

The VVER-440/213 containment has a special feature, i.e. the localization system with bubbler condenser and air locks. When the pressure increases in the primary system compartments of the containment, then the mixture passes through the bubbler condenser and the non-condensable gases enter into the airlocks. When the steam becomes condensed in the primary system compartments, then depression develops there. Steam condensation occurs on the walls of the containment and it can be caused by the operation of the spray system or the ventilation system.

It was studied with MAAP calculations, if the vacuum could represent a challenge for the containment integrity. For the study those sequences were screened, for which the containment vacuum was larger than 200 mbar. It was found that vacuum could develop only for large break LOCA cases with 3 operating spray systems and various additional assumptions.

According to the calculations, 3 operating trains of the spray system may cause excessive containment depression, if they start immediately after a hydrogen burn. However, this event is unlikely.

A simultaneous starting operation of the filtered vent release and the spray system can cause the largest vacuum in the containment. However, the prevention of the excessive vacuum is easy, namely the spray systems should be switched off. The conclusion of the analyses for the severe accident guideline is that excessive vacuum can be prevented by stopping one or two trains of the spray system, when containment pressure reaches the level of around 1 bar overpressure.

6. Preventing containment overpressure

According the SCG 2 guideline, a filtered venting procedure will be started to prevent containment overpressure. The procedure is expected to start, when containment pressure reaches 3.3 bar (absolute). This pressure level is expected to occur no sooner than 24 h after the start of a severe accident. Upon reaching the setpoint, there are another 50 minutes available for starting the filtered venting procedure. The designed filtered venting procedure will decrease containment pressure even with conservative assumptions, therefore excessive overpressure will be prevented.

Fission products will be retained in a dedicated severe accident filter with a filtering efficiency of 99.9 %, thus Cs and I release will be limited to 0.01 %.

Upon decreasing the containment pressure the venting line valve is expected to close. Even a failure to close the venting line would not lead to excessive release of fission products into the environment according to the analysis.

A spurious early opening of the venting line between 2.5 bar and 3.3 bar does not lead to excessive radioactive release threatening the environment. Although the noble gas release will increase in this case, the filter will retain most of the additional Cs and I release. The venting line is designed with a rupture disc with a 2.5 bar setpoint, therefore spurious opening below 2.5 bar has not been considered.

Hydrogen burns are not expected to occur in either the venting line or the filter, because oxygen is not available in sufficient amount.

Steam will condense in both the venting line and the filter, therefore an appropriate condensate removal system is needed. Condensate can be cooled and redirected into the containment, thus a loss of containment water inventory through the filtered venting system can be prevented.

7. Decreasing fission product release using the ventilation systems

There are 6 different ventilation systems available for cooling and removing aerosols from the containment atmosphere. MAAP calculations were performed with assumptions of different systems in operation, different boundary conditions as starting time of operation, number of working trains and inlet water temperature of the heat exchanger (Table 5.). The calculated sequence was selected as the dominant sequence (PDS-05C) according to the Level 2 PSA.

Table 5. Operation of the TL 01 recirculation ventilation system on containment pressure and release to the environment based on MAAP calculations

	3 trains $T_{water\ in}$=5 ºC	3 trains $T_{water\ in}$=25 ºC	1 train $T_{water\ in}$=25 ºC	1 train no cooling
Containment pressure at vessel failure time	1,34 bar	1,36 bar	1,4 bar	1,43 bar
at 120000 s	1,3 bar	1,42 bar	1,7 bar	1,8 bar
at 258000 s	1,9 bar	2,0 bar	2,85 bar	3,4 bar
CsI release at 120000 s	0,0138 kg	0,0139 kg	0,0143 kg	0,0145 kg
Condensed water mass at vessel failure time	12500 kg	9710 kg	3310 kg	0 kg
120000 s	49800 kg	38600 kg	15300 kg	0 kg

The effectiveness of the ventilation system is much lower, than that of the spray system, but they can be used to decrease the radioactive release by a few percent. For comparison, the spray system is capable to decrease the radioactive release by an order of magnitude.

Containment pressurization can also be slowed down by the ventilation systems. Time frames for reaching certain containment pressure levels for different configurations of the TL 01 ventilation system is presented in Table 6. Time can be gained for the intervention of the filtered vent operation. However, the best option for the containment pressure reduction is just the restoration of the spray system operation.

Table 6. Containment pressure versus time for different configurations of the TL 01 ventilation system

	3,3 bar	**4,5 bar**
Base case without operation	1.7 days	3,5 days
1 train in operation	2 days	4,5 days
2 trains in operation	2.4 days	6,3 days
3 trains in operation	3 days	10 days

8. Summary

MAAP calculations were performed for the verification of the SAMG for Paks NPP. SAMG actions were assumed in the code calculations with different options concerning the accident sequences, availability of systems and timing of accident management actions.

The verification study leads to the conclusion that fine-tuning and some modification of the existing severe accident guidelines are needed to meet the specific challenges represented by severe accidents at the Paks NPP.

Treatment of Accident Mitigation Measures in State-of-the-Art Reactor Consequence Analyses

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Extended Abstract

This paper discusses the evaluation of mitigation measures, including measures developed as part of security assessments, in conjunction with the U.S. Nuclear Regulatory Commission (US NRC) program entitled, State-of-the-Art Reactor Consequence Analyses (SOARCA).

The US NRC has undertaken the SOARCA program to perform an updated realistic evaluation of severe reactor accidents and their offsite consequences. It is the intent that these analyses reflect the accumulated improved understanding of severe accident behavior and potential consequences developed through the considerable research conducted by the US NRC, the industry, and the international research community over the last 25 years and that the analyses would provide a body of knowledge on the more likely outcomes of such remote events. This information would be the basis for communicating that aspect of nuclear safety to government authorities, licensees, and the general public. It is also an objective that SOARCA would update and replace quantification of offsite consequences documented in earlier studies, that in some cases was based on overly conservative assumptions and simple bounding analyses to the extent the earlier results are unnecessarily conservative and can be misleading.

The approach used in SOARCA has been to (1) use state-of-the-art analytical tools for accident progression and consequence analyses; (2) credit the use of Severe Accident Management Guidelines (SAMGs) and other new plant procedures, such as mitigative measures resulting from security related assessments and other like programs; and (3) use realistic site-specific evacuation scenarios and emergency planning modeling along with updated population and meteorological data. The focus of SOARCA is the application of detailed, realistic scenario-specific and consistent modeling using an integrated severe accident code, MELCOR and the offsite consequence code MACCS2. Analyses have been completed for two pilot plants, the Surry plant, a pressurized water reactor design, and the Peach Bottom plant, a boiling water reactor design. Accident scenarios adopted for analyses were selected by their potential severity for offsite consequences together with the frequency of occurrence.

In preparation for the detailed, realistic modeling of accident progression and offsite consequences, the US NRC had extensive cooperation from the licensees to (1) develop high-fidelity plant systems models; (2) define operator actions, including the most recently developed mitigative actions; and (3) develop models for simulation of site-specific and scenario-specific emergency planning. Moreover, in addition to input for model development, licensees provided information from their own probabilistic risk assessment (PRA) on accident scenarios through tabletop exercises (with senior reactor operators, PRA

analysts, and other licensee staff) of the selected scenarios. We received input on the timing and nature of the operator actions to mitigate the selected scenarios.

The assessment of the mitigation measures was undertaken with support from integrated accident progression analyses using the MELCOR code, NRC's detailed mechanistic model that incorporates our best understanding of plant response and severe accident phenomenology. MELCOR analyses were used to both confirm the timing available to take mitigation measures and to confirm that those measures, once taken, were adequate to prevent core damage or significantly reduce radiological releases. In other instances, MELCOR analyses using only installed equipment revealed that PRA success criteria were overly conservative, indicating core damage where MELCOR analysis indicated no core damage. These insights are being factored into future PRA assessments.

Best Practices Applied to Deterministic Severe Accident and Source Term Analyses for PSA Level 2 for German NPPs

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Abstract

Deterministic severe accident analyses are required as a basis for Probabilistic Safety Analyses (PSA) level 2 and for the development of Severe Accident Management (SAM) measures. GRS performed such studies in the past for different German nuclear power plants (NPP) and used at that time mainly the integral code MELCOR. The integral code ASTEC jointly developed by GRS and IRSN will be the tool for future analyses.

The calculations performed for German BWR and PWR have shown that typical severe accident scenarios are characterized by several phases and that the consideration of best estimate data and plant specific features are important e. g. to obtain realistic and reliable source term data.

Together with the compilation of the detailed integral code input decks, the qualification of the nodalisation schemes has always been pursued with comparative calculations with detailed GRS codes ATHLET/ATHLET-CD and COCOSYS. The results of these comparative analyses showed a good agreement of essential parameters and of the general plant behaviour during the accident progression. The greater level of detail of the NPP nodalisation schemes developed for the integral code application contributes significantly to this good agreement with detailed code results and is needed to meet the requirements from the point of view of PSA level 2 or SAM analyses.

The methods applied for the integral deterministic severe accident analyses are described in the paper. Information is provided along with the description of relevant phenomena of severe accidents in German NPPs. The topics are necessary for an appropriate nodalization of NPPs in integral severe accident codes like MELCOR or ASTEC, the appropriate modelling of relevant fission product release paths from the NPP buildings into the environment and methods used for the qualification of the nodalisation schemes utilized for the integral codes. The experiences gained in applying MELCOR in the past are beneficiary for the use of ASTEC in the future.

1. Introduction

GRS has been investigating the various possibilities and means to influence the progression of severe accidents by Severe Accident Management (SAM) measures since 1985. The research performed at GRS in different SAM projects in the past has resulted in proposals for the development of several SAM measures, e. g. to prevent Reactor Pressure Vessel (RPV) failure under high pressure by bleed and feed procedures or to prevent long term containment overpressure failure by a filtered venting system or to prevent containment challenges due to global hydrogen combustions by a Passive Auto-catalytic Recombiner (PAR) concept or by N_2 inertisation of the containment like in BWR type 69 NPPs. Based on such detailed integral code analyses a systematic development of possible Severe Accident Management Guidance (SAMG) as additional measure to support the work of the NPPs crisis teams in case of severe accidents seems to be possible, but has not yet been performed by GRS.

The plant specific deterministic calculations needed for the development of SAM measures and for PSA level 2 studies have been mainly performed with the integral code MELCOR[1] developed at Sandia NL, US, in the past supported by detailed code analyses with ATHLET/ATHLET-CD[2] and COCOSYS[3] developed at GRS.

The MELCOR application at GRS for BWRs started in 1990 and for PWRs in 1993. Many different sequences have been calculated for both reactor types and some detailed nodalisation studies[4] have been prepared in the past, to prove the developed core, reactor circuit and containment nodalisation[5]. Together with the compilation of the MELCOR data set, the qualification of the nodalisation has been pursued with comparative calculations with detailed GRS codes ATHLET/ATHLET-CD and COCOSYS. The results of these comparative analyses showed in most of the areas a good agreement of essential parameters and of the general plant behaviour during the severe accident progression[6]. The main reason for this good agreement between integral and detailed code results is the greater detail of the German NPP plant nodalisation used for integral codes like MELCOR and ASTEC calculations at GRS.

2. German PSA level 2 guidance

Construction and operation licenses of German NPPs have been granted in the past, based on purely deterministic analyses. Every ten years, a Periodic Safety Review (PSR) has been performed by the licensees mostly on a voluntary basis. PSA Level 1 has been part of the PSR for many years. Performing PSR at ten years intervals including a plant specific PSA is mandatory required now by the recent amendment of the Atomic Energy Act (2002).

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[1] R. O. Gauntt, J.E. Cash, R. K. Cole, C. M. Erickson, L.L. Humphries, S. B. Rodriguez, and M. F. Young, MELCOR Computer Code Manual, Vol. 1: Primer and Users' Guide, Vol. 2: Reference Manuals, Version 1.8.6 September 2005, Sandia National Laboratories, USA, Albuquerque, SAND2005-5713

[2] K. Trambauer, Coupling Methods of Thermal–Hydraulic Models with Core Degradation Models in ATHLET–CD, 6th Int. Conference on Nuclear Engineering (ICONE), San Diego CA, USA, May 10–15, 1998

[3] W. Klein-Heßling, S. Arndt, H.-J. Allelein, Current status of the COCOSYS development, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH - Eurosafe Forum 2001

[4] M. Sonnenkalb, MELCOR 1.8.4 Sensitivity Study of Core Melt Behaviour of PWR, Paper presented at 6. MCAP Meeting, Bethesda, April 29 - May 01, 1998

[5] M. Sonnenkalb, Summary of MELCOR Applications to German NPPs, Paper presented at MCAP-Meeting, Albuquerque, September 20-23, 2005

[6] M. Sonnenkalb, Experience and Results of MELCOR Application for German PWRs, Paper presented at SARJ-98, Workshop on Severe Accident Research held in Japan, Tokyo, November 4-6, 1998

PSA Level 2 has been performed in the past in Germany within R&D projects by the GRS for typical PWR and BWR units, exploring PSA Level 2 methodology. PSA Level 2 recently has become part of the periodic safety review in Germany. Therefore PSA guidance was developed and published in 2005. There are two volumes of the German PSA guidance, representing the status of knowledge:

The volume on “Methods for PSA”[7] deals with:

– PSA Level 1/2 interface (core damage state properties),

– quality requirements for integral deterministic accident and source term analysis,

– accident progression event tree (APET) issues to be considered,

– definition of release categories (source term) and

– handling of uncertainties.

The volume on “Data for PSA”[8] gives advice how to:

– quantify branching probabilities in the APET for complicated issues and

– specify for which branching probabilities generic or plant-type specific or plant specific numbers need to be used.

The experiences gained from the integral code applications was one of the basis for the development of this PSA guidance related to deterministic severe accident analyses.

3. Severe accident phenomena typically for German PWR

Deterministic severe accident integral code calculations have been performed within SAM and PSA projects at the GRS. The results have shown that typical severe accident scenarios in German PWRs can be characterised by several phases[9]. These are the in-vessel phase until global RPV failure (RPV without penetrations), typically at low pressure if SAM measures to depressurize the RCS are used, followed by a dry MCCI phase of some hours until sump water ingression into the reactor cavity takes place and finally the filtered containment venting to prevent a long term over pressure failure. Spray systems are not installed in German PWRs.

The analyses showed that the total amount of H_2 generated during the in-vessel phase is equivalent to an oxidation rate of 35-60 % of the total Zr amount of the reactor. The higher amounts of H_2 have been calculated for scenarios with late start and longer duration of the core melt process and for high pressure core melt scenarios. As a result of the MCCI reaction after RPV failure there is an ongoing release of e.g. H_2, CO and CO_2. The rates are strongly dependent on the concrete composition.

The amount of energy released from the primary circuit into the containment before RPV and the size and location of the release location has a great influence on the pressure, the convection flows, the local gas concentration and the long term behaviour of the containment and especially on the operation of the PARs and the filtered containment venting. Due to the design of the containment, e.g. of PWR of the Konvoi type, the convection flow regime inside the containment is determined by the initial

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[7] Bundesamt für Strahlenschutz, Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke, BfS-SCHR 37/05, August 2005

[8] Bundesamt für Strahlenschutz, Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke, BfS-SCHR 38/05, August 2005

[9] M. Sonnenkalb, Kernschmelzablauf In- und Ex-Vessel, Paper presented at Fachtag der KTG-Fachgruppen Reaktorsicherheit und Thermo- und Fluiddynamik: „Fortschritte bei der Beherrschung und Begrenzung der Folgen auslegungsüberschreitender Ereignisse“, Forschungszentrum Karlsruhe, 25./26. September 2003

event of a sequence and the release location. One or two main convection flow pattern may exist as shown in Fig. 1. The reason for this behaviour is the different pressure peak caused by different break sizes during an accident. The resulting pressure difference between the inner and outer containment part determines the number of burst membranes, which will be destroyed on the top of both SG compartments and determines, if openings (e. g. burst membranes or doors) inside the missile protection cylinder exist, which will be opened too. Dependent on the convection flows, the homogenisation process of gases (e. g. steam, H_2) and aerosols released into the containment is slower or faster.

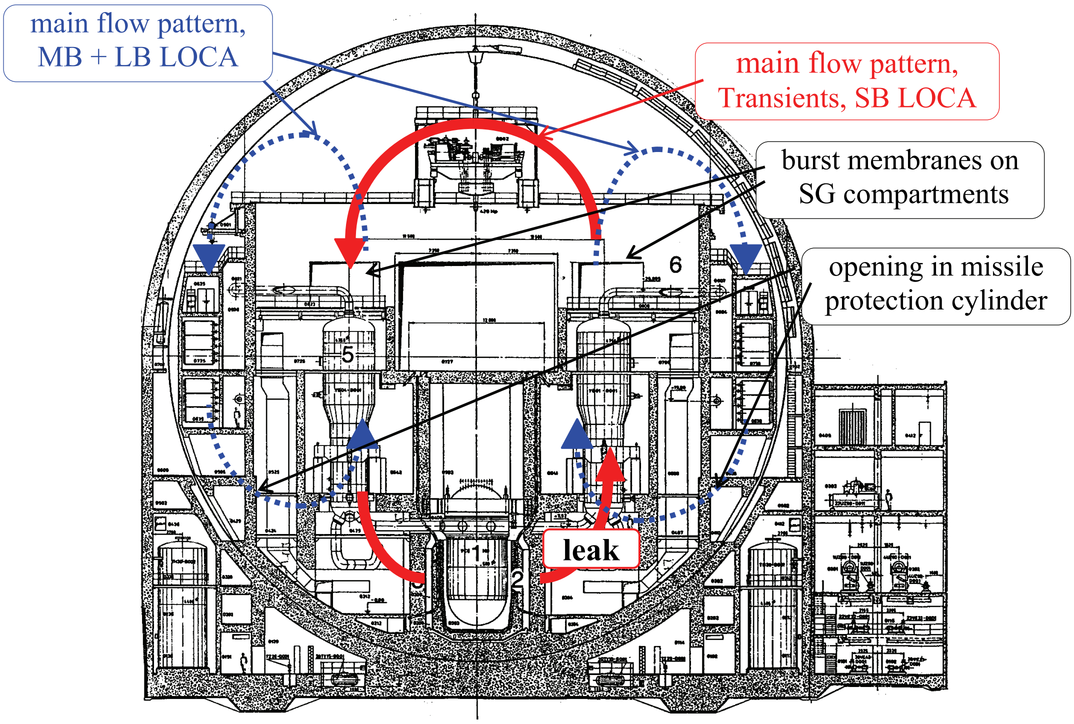

Fig. 1 Main convection flow pattern in German PWR of the Konvoi type NPP

The differences in the convention flow behaviour are of great importance and have also been considered for the qualification of a PAR concept and the definition of the PAR locations as one of our important SAM measure applied to PWR. The calculations have been performed with the GRS containment code COCOSYS and a very detailed PWR containment scheme. The results of selected MELCOR calculations, e. g. release rates and system heat from the reactor circuit into the containment, have been used as input for the COCOSYS analyses. The final concept for the large dry German PWR containments consists of about 65 PARs for a typically PWR of the Konvoi type.

4. Severe accident phenomena typically for German BWR

Two PSA level 2 projects have been completed in the last years at GRS - one for each type of a German BWR. In both projects severe accident calculations have been performed with the integral code MELCOR, while the development of the integral code ASTEC for BWR application is a topic for the near future.

The latest analyses of a BWR type 69 NPP also included an extensive model of the reactor building and the turbine hall, while the older analysis for a BWR type 72 (Fig. 2) NPP has been focused on the containment behaviour only. Based on these analyses typical severe accident scenarios in German BWRs consists of an in-vessel phase until local RPV failure (RPV with multiple penetrations) typically at low pressure due to the automatic depressurisation of the RPV, followed by an ex-vessel phase either inside the containment (BWR type 72) or partly outside the containment in the lining room of the reactor building (BWR type 69). The situation inside the containment during severe accidents strongly depends on the scenario especially on the release from the RPV which goes either into the wetwell or the drywell in case of a break at the main steam or feed water line.

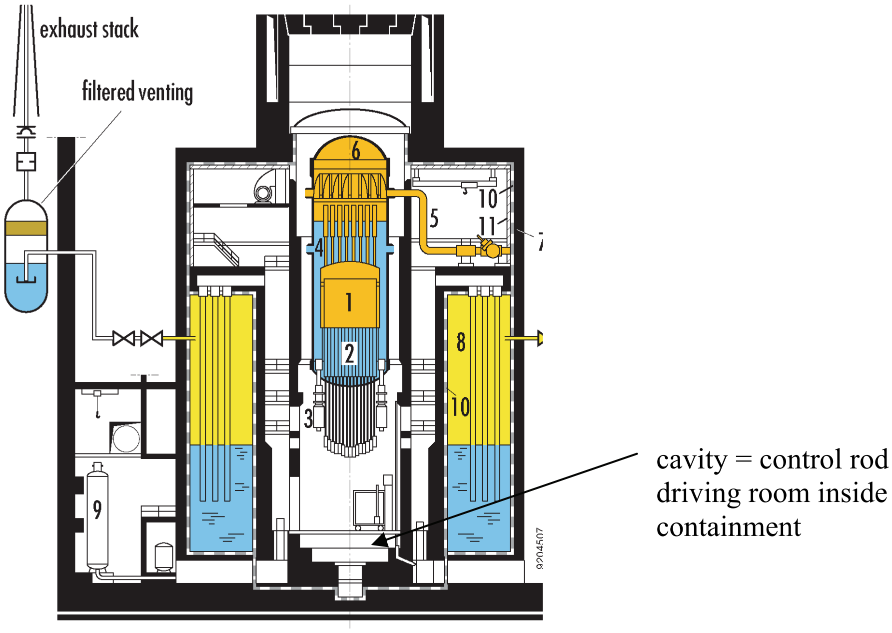

Fig. 2 Scheme of a German BWR type 72 containment including part of reactor building

The total amount of H_2 generated during the in-vessel phase is equivalent to an oxidation rate of 15 - 70 % of the total Zr amount of the reactor, which leads to a quick increase of the hydrogen concentration because of the lower containment volume in comparison to PWR containments. The higher amounts of H_2 have been calculated for high pressure core melt scenarios and scenarios with long lasting evaporation of the water in the core after overfeeding of the RPV in the early phase. The lower values are typically for low pressure core melt cases after automatic pressure reduction ADE which is often under steam starved conditions. Due to the automatic depressurization of the RPV the probability of high pressure cases is much less compared to the other cases.

Following the local RPV failure in a BWR type 72 NPPs the melt is released into the reactor cavity which is called control rod driving room and which is part of the containment. A dry or wet MCCI phase follows and H_2, CO and CO_2 are released from the concrete erosion. In most cases there is already water in the cavity at the time of RPV failure so that there is an ongoing steam production as well. The wetwell of the containment is N_2 inerted while in the drywell PARs are installed as SAM measures to prevent global hydrogen combustions. Another SAM measure installed is the filtered containment venting which is used one or more times dependent on the scenario before or after RPV failure to prevent an over pressure containment failure. A spray system located in the upper drywell may be used as well to limit the pressure increase or to decrease the airborne content of aerosols inside the containment. The "Notfallhandbuch" contains some information of how to use the systems. Such systems are typically used within SAMG as well which are not yet systematically developed. Detailed severe accident analyses could show the benefit of the use of such systems.

In case of a BWR type 69 NPP (Fig. 3) the situation after local RPV failure is quite different. The wetwell and the drywell of the containment are N_2 inerted as a SAM measure to prevent any hydrogen combustion. So, all the H_2 is retained in the containment as long as the venting system is not used.

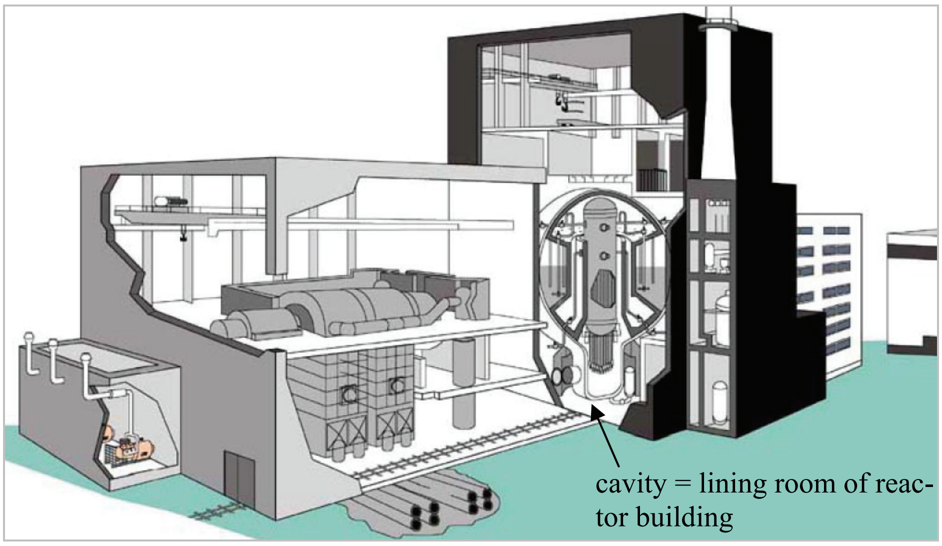

Fig. 3 Scheme of a German BWR type 69 NPP

The melt is released as well first into the so called control rod driving room of the containment. As the containment consists of a steel shell which is not designed to withstand any melt attack after RPV failure for a longer period of time it may fail soon after the melt release into the control rod driving room. This first part of the ex-vessel phase is quite short and the containment will fail mostly under elevated pressure if no manual action is considered. To use the containment venting system and/or the spray system installed in the upper drywell to reduce the containment pressure would probably be one of the major recommendations of a SAMG handbook, if systematically developed once. Another SAMG action which seems to be possible is the flooding of the control rood driving room to try to cool the RPV from outside. Even in this case plant specific analyses are required to prove the efficiency of the measure which exists already in some plant specific "Notfallhandbuch".

At the time of the containment failure probably a significant amount of H_2 accumulated in the containment (drywell and wetwell) is released into the lower reactor building rooms together with some steam and N_2. This may lead in addition to the "blow down" process to combustions outside the containment in the reactor building or the turbine hall. As a consequence further damages are not to be excluded. Thereafter MCCI will take place in the lining room of the reactor building mainly under dry but sometimes under wet conditions as well and H_2, CO and CO_2 are continuously released into the reactor building from the concrete erosion.

The weakness of the containment to withstand any core melt attack was one significant contribution to early high releases of fission products into the environment in the PSA level 2 for a BWR type 69 NPP. Again, such existing systems as described above are typically used within SAMG, though not systematically developed yet.

5. Best practices applied to deterministic integral severe accident analyses

In the following best practices derived from the different integral code applications are summarised and are based on a large number of different sequences that have been calculated for both reactor types in different projects.

- **Verification of the input deck**

As mentioned already in the introduction the verification of the developed integral code NPP models is very important and has been done by comparative calculations with detailed GRS codes ATHELT/ATHLET-CD and COCOSYS for selected phases of severe accidents. The plant models developed for such detailed codes have in general a much larger detail of the used nodalisation compared to the integral code so that the main topic of the comparisons was to achieve a good agreement of essential parameters and of the general sequence timing and calculated plant behaviour during the severe accident progression. The main reason for the good agreement achieved between integral and detailed code results in our applications is the greater detail of the German NPP integral code nodalisation used compared to published data decks from the literature.

Furthermore with all the datasets we developed we achieved very good steady state calculation results at nominal power without any well known user tricks or available special code features like time dependent constant pressure in volumes. This is one very important point to obtain later good results of the accident behaviour and is a result of the consideration of most of the plant specific systems even in the integral code input deck.

- **Detail of the nodalisation used for the RCS input deck**

To meet the recommendations from of the German PSA guidance the integral code nodalisation schemes developed for the German PWR 1300 Konvoi have a large detail compared to other applications. Contributions to these good results are the following facts, based on a RCS input developed for a PWR:

- detailed double or four loop nodalisation of the four loops of the RCS,
- detailed model of the steam generator (SG) secondary side,
- detailed model of the pressuriser (minimum 3 nodes: bottom - sub-cooled, middle - saturated, top - steam) and its relief tank,
- detailed modelling of plant specific (volume control and extra borating system, pressuriser spray and heaters) and safety injection systems (accumulators, ECCS injection systems).

For the latest BWR type 69 application we used as well a detailed hydraulic model of the reactor which consists of 15 volumes and 25 junctions. Here a realistic representation of the void fraction and the coolant flow behaviour in the core, the steam separation in the separator and steam dryer region and the water level determination is important for appropriate results. The later one is important as most of the reactor protection signals are dependent on the RPV water level.

- **Detail of the nodalisation used for the reactor core input deck**

One other important topic is the appropriateness of the reactor core model. In general the core input has to be checked very carefully, as the number and possible options of the models in the integral codes are large and are sometimes changed in later code version. Typically a detailed reactor core model for a PWR consists of 5 to 6 non-uniform radial rings and a minimum of 10 axial levels for the active core (~30 cm per axial core level). The subdivision is often done in accordance with the axial and radial power profile and the volume respectively the number of fuel elements considered in each core cell in one level should be increasing towards the periphery of the core. This is important as well to achieve realistically results of the core degradation and oxidation process.

For a BWR the number of radial rings in the core model is similar to the PWR while the number of axial levels is typically a little bit larger, as the total core height varies. Here as well an appropriate modelling of the fuel element canisters and the core bypass (the gap between them) is important. This influences the core degradation process and as well the relocation mechanisms of core melt into the lower plenum. Different to the PWR an early core material relocation into the lower plenum was often calculated, caused by the “open” design of the lower core support structures and the early failure of the control rods, located in the core bypass.

For both reactor types cases as well plant specific data of the initial fission product inventory are to be provided, typically at the end of the fuel cycle to get the highest amount of fission products. Often the default data provided by the code developers are based on older core designs with lower burn-up.

- **Detail of the nodalisation used for the containment input deck**

Besides the core and the RCS respectively RPV nodalisation the nodalisation of the containment and adjacent buildings together with the consideration of many plant specific details and relevant fission product release paths into the environment is another major challenge to the integral code applicant. Two examples of containment input decks used for integral codes are given in Fig. 4 - one for a German BWR type 72 and one for a PWR of the Konvoi type. The scheme used for the BWR type 69 is very similar with regard to the number of nodes to the BWR type 72 scheme shown below.

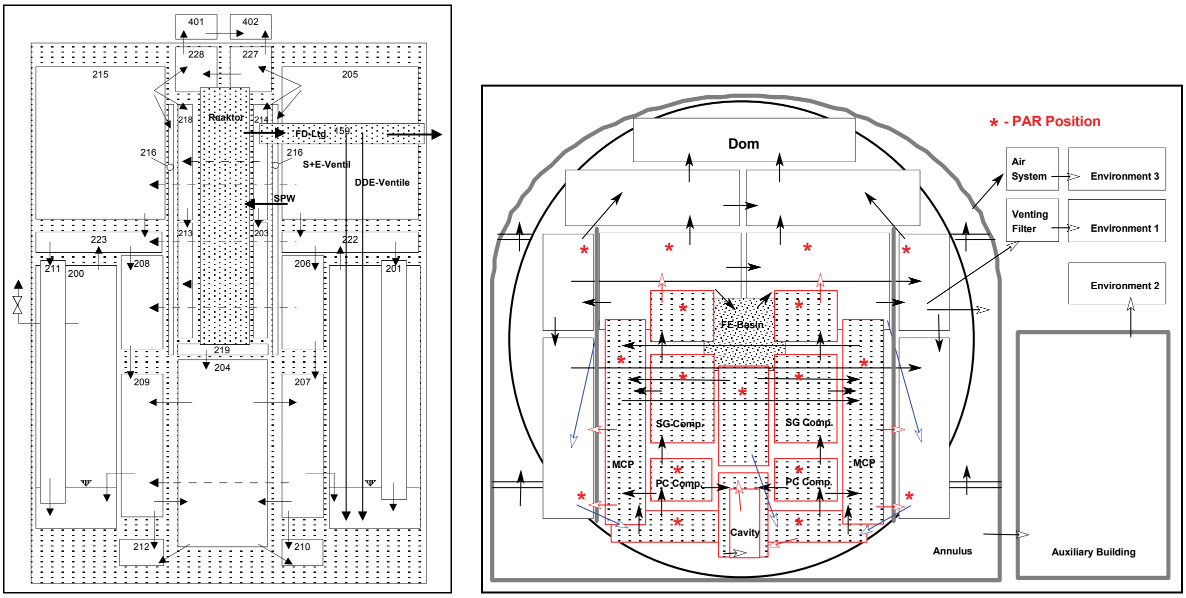

Fig. 4 Containment nodalisation of a German BWR type 72 and a PWR of Konvoi type

In principle the BWR containment was subdivided into two halves with exception of the wetwell and the cavity (control rod driving room). In both cases it was done to allow the calculation of temperature and gas distributions and main convection flows, which are relevant as well for aerosol processes. Most of the separate compartments of the containment are modelled as a single volume (minimum). Especially those which dominate the possible convection flow regime are modelled separately; some other small ones are lumped together. In this example a single cavity model was used for BWR type 72 while for the other BWR type 69 model three cavities have been defined, two of them outside of the containment in the reactor building.

The principles used to develop the containment input deck for the PWR of the Konvoi type are similar. The containment is subdivided into two halves. Most of the large open compartments, e.g. the dome region of the containment, which dominates the possible convection flow regime, are modelled by several control volumes[10]. This was done to allow the calculation of temperature and gas distribution and stratification to some extent especially in the dome region (see Fig. 4). Also all relevant flow pathes and especially pressure dependent openings have been modelled in detail. Finally it should be mentioned that the modelling of connections between volumes with large open areas needs special attention. The definition of the area according to the calculated large values for the opening may lead to a unrealistically high mass transfer and convection.

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[10] M. Tiltmann, M. Sonnenkalb, Requirements for Modelling Severe Accident Conditions inside Large dry PWR Containments, Paper presented at International Conference on Nuclear Containment, Robinson College, University of Cambridge, September 23-25, 1996

Plant specific concrete data have been used to get appropriate data from the MCCI reaction in the cavity. A difficulty still exists regarding the question/modelling of the melt transfer/spreading into adjacent rooms if the erosion process inside the cavity reaches the ventilation ducts inside the basemat.

One example which shows the benefit of detailed integral code analysis was the procedure of how the design of the German PAR concept was made. Detailed MELCOR analyses have been the basis for the conceptual design analyses made by COCOSYS[11].

- **Detail of the nodalisation used for the containment input deck**

The results of the latest PSA level 2 study made for a BWR type 69 showed that a detailed model of the reactor building and the turbine hall is important as well, as the containment may fail early in case of a severe accident. The number of rooms and its connecting flow paths modelled in the buildings have been selected depending on possible release paths for radio nuclides starting from different main release locations. Here it has been mainly the cavity (lining room) in the reactor building below the containment (see Fig. 3).

Especially the reactor building of German BWRs consists of many rooms located on up to 10 different floors which are interconnected by many doors, air ventilation system channels and in selected locations by burst membranes and flaps. The nodalisation of the reactor building and the turbine hall of the BWR type 69 input consist of more than 50 volumes in both buildings and takes into account that "death end" rooms are to be avoided in the model, if not existing in the real plant.

For the other two applications the reactor building was simulated in a simplified manner taking into account only the main important design details.

- **Consideration of plant specific systems**

Plant inspections showed that many different doors exist inside the containment of a PWR and the buildings of BWR which are often not leak tight. In the input deck such small gaps are simulated by remaining opening fractions of the relevant flow path. It simplifies the pressure balance inside the building during normal plant operation. The failure of the doors is dependent on a different Δp according to the door opening direction and its design. No failure of a door may be possible in case of a high water level on a floor especially if the doors are not leak tight. In addition a re-closure of doors after its failure in case of stronger reverse flow may happen and was modelled in the latest input decks assuming a 10 % remaining opening fraction of a failed door. Sensitivity analyses showed a significant influence on the radio nuclide behaviour and release to the environment. A realistic/appropriate definition of flow path (door) opening heights was important as well to allow a realistic modelling of water drainage between different rooms of the buildings parallel to the gas convection.

Air ventilation systems installed in the containment of a NPP as well as in most of the buildings are switched on during normal plant operation. The systems are designed to remove heat released from the RCS into the containment as well as from other components into the buildings and to keep a small sub-pressure to avoid leakages from the plant into the environment. Such systems have been modelled in a simplified manner in all our integral code input decks. We found it to be important as the availability respectively operability of such systems during accidents and severe accidents depends on the plant design. Some of the systems are in operation especially in German BWR even under accident conditions. It applies as well for the PWR e. g. for transient scenarios without any releases into the containment within the first hours of an accident. Nevertheless even if the systems may go out of operation during an accident, not all ducts of air ventilation systems which connect several rooms are

[11] J. Rohde, B. Schwinges, M. Sonnenkalb, Implementation of PAR Systems in German LWRs, Paper presented at the OECD-CSNI workshop on the implementation of severe accident management measures, PSI Villigen, Switzerland, September 10-13, 2001

closed. So the flow through the ducts contributes to convection processes between different rooms and should be modelled in an appropriate manner.

From our latest BWR type 69 application we learned that the stack connected by an off-gas line with one of the buildings of the NPP significantly contributed to the releases into the environment. As already mentioned the sub-pressure in this building is kept during normal operation by an air ventilation system which is switched off at the latest at the time of containment failure. But here the off-gas line stays open and the buoyancy force driven mass flow through the stack during long term phase contributes significantly to a small sub-pressure build up in buildings and a reverse mass flow direction into the buildings through e.g. leaks and open doors caused by the containment failure. This is important not only for the source term calculation.

6. Conclusion

The physical and chemical processes governing the progression of severe accidents are very complex, often involving many simultaneous phenomenological interactions, for which detailed experimental information is not available in all cases. Therefore, mathematical modelling and computer simulation of these phenomenological processes are influenced by various uncertainties. The codes MELCOR and ASTEC are still under development and not all models are completed or are on the same level of detail. However most of the integral code results derived from the NPP applications have been satisfactory to us.

Integral codes and many other codes as well provide the user with a great deal of flexibility. For instance the number of nodes to be used for the reactor circuit and the containment and the adjacent buildings is not limited. Furthermore different models for core degradation and radio nuclide release, transport and behaviour exist. The application of these models requires adequate and sufficient information on plant specifics and design and advanced knowledge of the code user on severe accident phenomena in general. A sufficient detail of the nodalisation schemes for all components and buildings and an appropriate verification of the developed input decks ensure good analyses results. The consideration of plant specific details in the input decks developed is a very important factor to achieve realistic analysis results and to meet the requirements defined e.g. in the German PSA guidance documents.

It would be inappropriate to assume that all of the models in integral codes like MELCOR or ASTEC as well as in other codes have been validated with very good results only. The validation process of such codes with a very broad spectrum of modelled phenomena is a still ongoing process mainly in parallel to the development work. Nevertheless the validation of the basic models in MELCOR and ASTEC has been done some time ago and the current code versions available are applicable to many phenomena and analyses in PSA Level 2 and SAM projects.

Use of a code is best approached from the issue perspective – what is the user trying to demonstrate from a safety point of view. The adequacy of a code and the specific plant model developed can then be judged from this perspective. This may point to the need of supplementary calculations e.g. in the areas of iodine behaviour and containment leakages or for the qualification of integral code input decks. Single runs of a code should not to be used for safety submissions. There is no requirement to be conservative in PSA level 2 studies as well as in studies supporting SAM program development, but it is desirable to demonstrate robustness of results to modelling uncertainties.

Many of the findings described in the paper derived from the MELCOR applications to German NPPs done in the past, supported by newer ASTEC applications for PWR and are valid as well for the application of other codes used to calculate severe accidents. High quality deterministic severe accident analyses performed e.g. in PSA level 2 projects allows its use for the development of SAMG. This was already done in many countries and is an ongoing topic for German plants.

Session 5

Severe Core Damage Accident Analysis for a CANDU Plant

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1. Introduction

Severe core damage (SCD) accidents in a CANDU®[1] reactor are postulated very low frequency reactor accidents that lead to the loss of core geometry, which is beyond the design basis accident realm. In order to minimize the health risk to the public or station staff, the progression of SCD accidents must be arrested using carefully selected severe accident management measures.

In general, the progression of a severe core damage accident in a CANDU reactor is slow because the fuel is surrounded by large quantities of light and heavy water, which act as heat sinks to remove decay heat [2,3,4]. Atomic Energy of Canada Ltd. (AECL) and the CANDU nuclear utilities selected the MAAP4-CANDU code as the primary tool for modelling the CANDU station response to an SCD accident. This paper provides an overview of the MAAP4-CANDU code, which includes the code development, code capabilities, and its current status. The application of the code to Level 2 PSA for a CANDU 6 plant is presented here. Based on the results obtained from the analyses, severe accident management measures (SAM) were incorporated to ensure containment integrity.

2. General Design Features of a CANDU 6

The core of a CANDU 6 reactor consists of 380 horizontal fuel channels. Each channel is about 6 m long and contains 12 fuel bundles inside a pressure tube filled with heavy water coolant. Each pressure tube is surrounded by a calandria tube. The fuel channels are inside a horizontal stainless steel cylindrical calandria vessel that is filled with a low-pressure heavy water moderator. The calandria vessel is housed in and supported by a light water-filled steel-lined concrete reactor vault, which provides thermal shielding. Figure 1 shows a schematic of the CANDU 6 reactor. The significant inventories of the heavy and light water surrounding the fuel and calandria vessel are highlighted; these act as heat sinks to remove the decay heat after reactor shutdown.

3. Severe Core Damage Accidents

An SCD accident typically requires a significant loss of moderator, which would otherwise act as a heat sink for voided fuel channels. Postulated SCD accidents begin as design basis accidents (DBA), but are combined with additional loss of safety or process systems (i.e., loss of heat sinks). Examples of SCD accidents include (a) a loss-of-coolant accident (LOCA) plus loss-of-emergency-core-coolant

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[1] CANDU is a registered trademark of Atomic Energy of Canada Ltd.

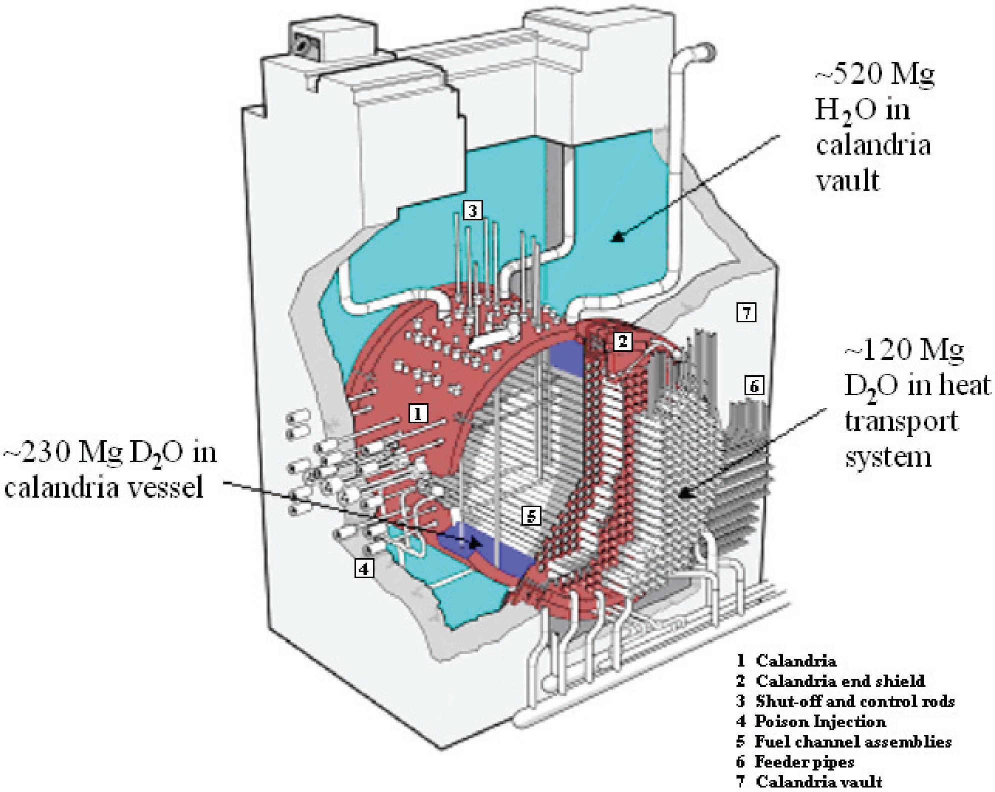
[2] Snell, V.G., Bonechi, M. and Kupferschmidt, W.C.H., "Advances in Nuclear Safety", Proceedings of Pacific Basin Nuclear Conference, Seoul, Korea, October 29-November 2, 2000

[3] Meneley, D.A., Blahnik, C., Rogers, J.T., Snell, V.G. and Nijhawan, S., "Coolability of Severely Degraded CANDU Cores", International Seminar on Mass and Heat Transfer in Severe Reactor Accidents, Cesme, Turkey, May 22-26, 1995

[4] Mathew, P.M., Kupferschmidt, W.C.H. and Bonechi, M., "Application of PSA to CANDU Design and Licensing", Proceedings of Pacific Basin Nuclear Conference, Shenzhen, China, October 21-25, 2002

(LOECC) and a loss of moderator cooling, (b) a station blackout (SBO) scenario, and (c) a multiple steam generator tube rupture with an LOECC, loss of steam generator feedwater and loss of moderator cooling. Severe accident phenomena occurring during core damage progression include: fuel bundle heat up and disassembly; fuel channel heat up, sagging, perforation and melt-through; fuel and fuel channel debris separation (disassembly) from the remaining channel and the formation of a suspended debris bed; suspended debris heatup, core collapse; terminal debris formation within the calandria vessel; calandria vessel failure; interaction of core debris with the concrete calandria vault; and failure of the containment structure. A severe core damage accident can be halted by accident management measures, such as reflooding the calandria vessel (in-vessel cooling) and maintaining cooling on the outside of the calandria vessel (ex-vessel cooling).

Figure 1 Schematic of a CANDU 6 Reactor Core



Atomic Energy of Canada Ltd. (AECL) and the Canadian CANDU utilities selected MAAP4-CANDU as the primary tool for modelling the CANDU station response to a Severe Core Damage accident.

4. General Description of MAAP4-CANDU Code

MAAP-CANDU is the adaptation of the MAAP (Modular Accident Analysis Program) code, specifically designed for integrated CANDU reactor station SCD accident simulation [5]. MAAP is an integral nuclear plant analysis code, for modelling SCD accidents; it was developed for pressurized and boiling light water reactors (LWR) by Fauske and Associates Incorporated (FAI), and is owned by the Electric Power Research Institute (EPRI). The change from simulating a pressure vessel reactor to the CANDU reactor required several new modules, developed by Ontario Power Generation (OPG), to

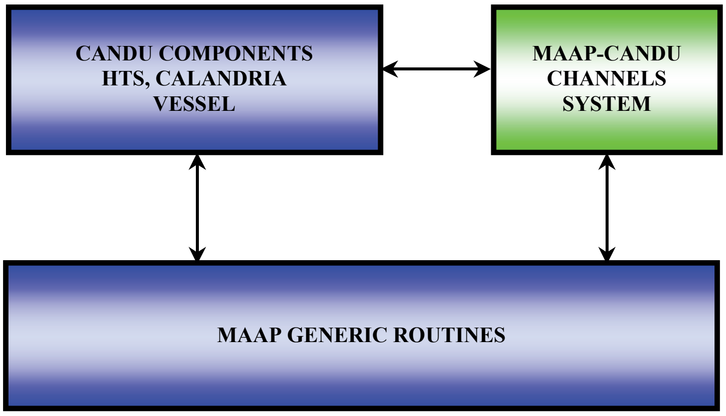
[5] P.M. Mathew, S.M. Petoukhov, M.J. Brown and B. Awadh, "An Overview of MAAP4-CANDU Code", 28th Annual Conference of the Canadian Nuclear Society, Saint John, NB, 2007 June 3-6

simulate the CANDU reactor core, calandria vessel, and debris formation and relocation. The development of MAAP-CANDU also required the adaptation of many non-core MAAP models to CANDU station designs. OPG is the MAAP-CANDU code licensee (code holder), and AECL holds a sub-license from OPG.

The MAAP4-CANDU code links the significant CANDU reactor systems such as Primary Heat Transport System (PHTS), safety systems, containment, fuel in an integrated fashion. MAAP4-CANDU models the core disassembly and debris behaviour, fission product release and containment response. The models are mechanistic where possible, using correlations or theoretical models. Other models employ failure criteria, some hard-coded and others reliant upon user input, for phenomena that either lack mechanistic models or have been simplified.

MAAP4-CANDU is a combination of "generic" MAAP4-LWR models, CANDU-specific component models, and the Channels System suite of CANDU core models (Figure 2).

Figure 2 The basic architecture of MAAP4-CANDU code



The generic models in the MAAP4-LWR code include subroutines and functions for fission product behaviour, thermal properties, containment behaviour, steam generators, etc. Some subroutines have been modified to adapt them to CANDU design features, but these models within the subroutines are essentially unchanged from MAAP4-LWR. The CANDU component models are specific to the CANDU design, and include models for the calandria vessel, pressure and inventory control system, and some of the engineered safety systems.

The following are some of the phenomena modelled by the MAAP4-CANDU code:

- PHTS coolant circulation, phase separation and blow-down;
- Temperature excursion of fuel and fuel channels, including Zircaloy-steam reaction;
- Thermal mechanical behaviour of fuel and fuel channel failures;
- Fuel channel disassembly and suspended debris bed formation;
- Solid and molten debris relocation and debris jet particulation in calandria vessel and containment compartments;
- Core debris interaction with coolant, steam, molten corium-concrete interaction and hydrogen burning and steam explosion;
- Containment iodine chemistry and fission product release, transport and deposition models.

The following is a list of the systems modeled in the MAAP4-CANDU code:

- Two-loop or one-loop PHTS including piping, pumps, reactor inlet and outlet headers and feeders;
- Pressurizer and pressure and inventory control system;
- CANDU reactor core; Steam generators - primary and secondary sides;
- Containment building, rooms, calandria vessel and reactor vault;
- Shield cooling, moderator cooling and shutdown cooling systems;
- Emergency core cooling system (high, medium and low pressure components);
- Containment dousing spray system and local air coolers;
- Containment ventilation system, hydrogen igniters and recombiners; and
- Power operated and passive (spring loaded) relief valves.

The most distinguishing feature of MAAP4-CANDU, compared with the MAAP4 PWR/BWR code, is the channels system model of the CANDU reactor core: fuel bundles inside pressure tubes, and surrounded by the calandria tubes inside the heavy water filled calandria vessel. The Channels System contains models of the core components between the inlet and outlet headers, and models the behaviour of these components within the calandria vessel volume as the fuel channels disassemble into suspended debris. The debris behaviour is modelled by the Channels System subroutines until the debris form the terminal debris bed within the calandria vessel. Because of their importance to the MAAP4-CANDU channel system, the following models are described below: fuel, fuel channel, core and debris models.

4.1 Fuel and Fuel Channel Models

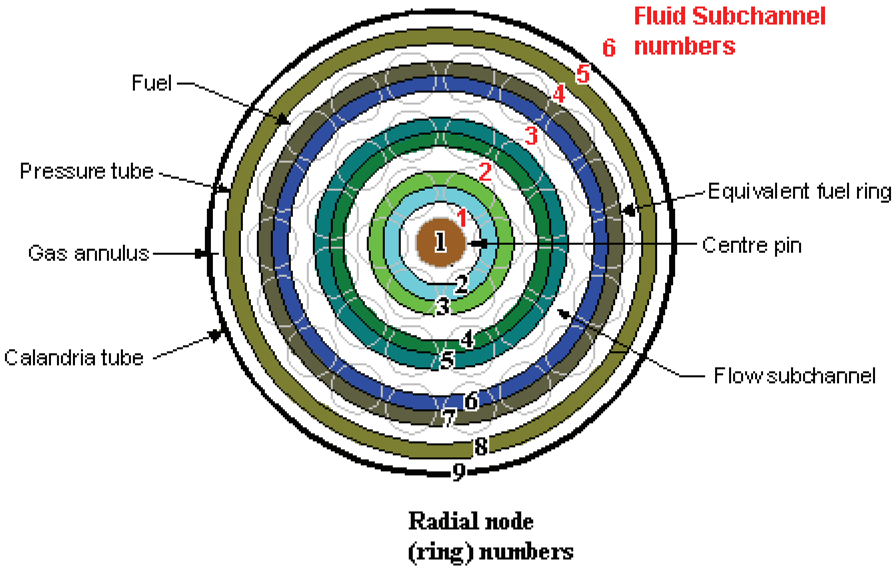
The CANDU 6 fuel and fuel channels are modelled in MAAP4-CANDU as a set of nine concentric rings that are continuous over the length of the channel (Figure 3). All the zirconium in the fuel bundle is incorporated into the fuel ring model, and the Zr from the in-core devices is incorporated into the calandria tubes. Each fuel element ring, except the centre element of a CANDU 6 bundle, is represented by two adjacent, contacting model rings.

The fuel channel is subdivided into 12 axial nodes, each representing a fuel bundle and the surrounding channel section. The fuel bundle rings represent a UO_2 and Zr mixture with a uniform temperature for each ring and axial channel node. The zirconium oxidation and hydrogen production rate depend upon the ring temperature, oxide thickness, and steam concentration. The fuel decay heat is calculated as a function of time and of the axial and radial position within its fuel channel. Calculations of the fuel and fuel channel temperatures in MAAP4-CANDU begin only when the channel is dry; prior to that a global energy balance is used to account for the heat transferred to the surroundings.

MAAP4-CANDU also models fuel-sheath melting and relocation, radial heat transfer from fuel to the calandria vessel liquid or gas phase, and fuel bundle slumping into contact with the pressure tube. When the calandria tube becomes perforated, due to sagging at low pressure, steam from the calandria vessel enters the annulus between the calandria tube and the pressure tube, increasing the rate of fuel sheath oxidation and hydrogen production. This exothermic reaction results in higher temperatures, thus accelerating fuel channel disassembly and the formation of debris. A sagging channel preferentially stretches at the ends of adjacent fuel bundles, and perforates when the fuel channel longitudinal strain exceeds a user-defined input strain.

A high-pressure fuel channel rupture model is used in MAAP4-CANDU. This model compares the pressure-tube hoop stress with the maximum sustainable hoop stress at that temperature, based on experiments performed for isothermal Zr-2.5%Nb tubes. If the calculated stress is greater, the pressure tube balloons into contact with its calandria tube; if the hoop stress of the combined tubes is greater than the maximum sustainable hoop stress, the fuel channel ruptures.

Figure 3 MAAP4-CANDU nodalization of a fuel bundle and fuel channel into a fuel ring model (37-pin fuel bundle)



4.2 Core and Debris Models

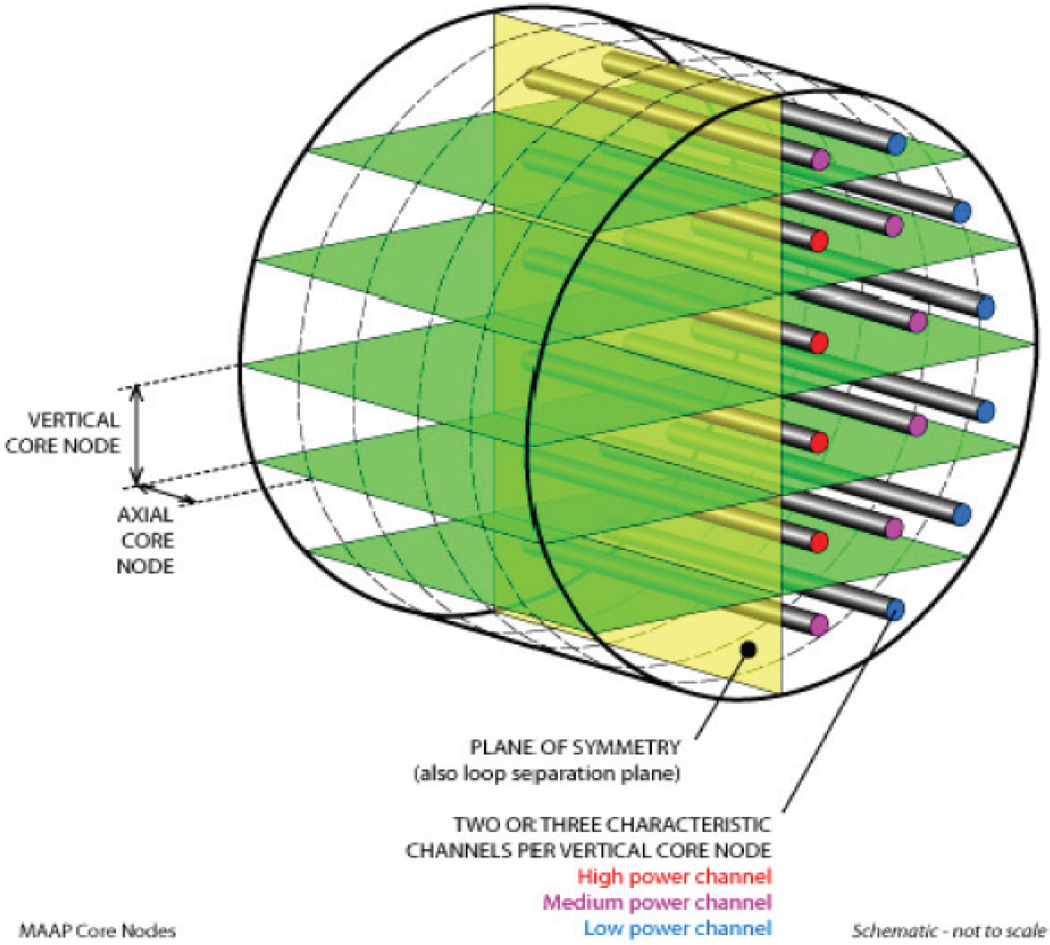
The CANDU core model uses two nodalization schemes: i) an axial channel nodalization to calculate the fuel and fuel channel heat up and channel disassembly to core debris, and ii) a calandria vessel nodalization to calculate the calandria vessel heat transfer conditions and the formation, heat up and relocation of core debris.

MAAP4-CANDU uses *characteristic channels*, each representing one or more actual fuel channels, called associated channels. Each characteristic channel is modeled as an average of all its associated channels. There are three characteristic channels - each belonging to a high, medium or low decay heat group - for each vertical core node of the calandria vessel and for each primary heat transport (PHTS) loop. A two-loop CANDU 6 with 380 channels is modelled with six vertical core nodes, which corresponds to 18 characteristic channels in each loop.

The calandria vessel volume is nodalized by vertical nodes, axial core nodes, and by PHTS loop. The CANDU 6 design has two PHTS loops, which are symmetrical about the centre vertical plane. MAAP4-CANDU uses six vertical core nodes and five axial core nodes for each loop (Figure 4).

The code uses a failure criterion, based on the pressure and calandria tube temperatures, for the disassembly of an axial fuel channel node and subsequent formation of fuel and fuel channel debris. When the pressure and calandria tube temperatures exceed the melting point of oxygenated zirconium, or a user input temperature, the fuel and fuel channel portion of that channel node disassemble from the original channel to become suspended core debris. The debris are held up by colder, and hence stronger, underlying submerged channels. The steam/hydrogen flow rate through the suspended core debris is calculated, and Zr oxidation and fission product release calculations continue, if the debris is above the moderator level and exposed to steam. The UO_2–Zr interaction can occur with the formation of molten material, which can trickle down to the moderator below to become part of the terminal debris bed at the bottom of the calandria vessel. The core collapses into the bottom of calandria vessel when the underlying supporting channels can no longer support the overlying debris and sagging channels. The resulting terminal debris bed at the bottom of the calandria vessel is cooled by the remaining moderator.

Figure 4 Nodalization of a two-PHTS-loop CANDU 6 reactor core, with six vertical and five axial core nodes for each PHTS loop and the distribution of the characteristic channels in a single PHTS loop



5. Application of MAAP4-CANDU Code to Point Lepreau Plant Refurbishment

A level 2 Probabilistic Safety Assessment (PSA) was performed by AECL for the Point Lepreau Generating Station (PLGS, a CANDU 6 reactor) by the Point Lepreau Refurbishment (PLR) project. The MAAP4-CANDU v4.0.5A+ code was used to perform the consequence analysis to support the level 2 PSA activity. An overview of Point Lepreau operations and refurbishment activities was written [6], and more details of the plant nodalization, analysis assumptions and results are provided elsewhere [7].

For the PLR Project MAAP4-CANDU v4.0.5A+ was used to estimate:

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[6] R. Eagles, "Point Lepreau Generating Station Refurbishment Project - Update and Status", Proceedings of the 28th Annual Conference of the Canadian Nuclear Society, Saint John, New Brunswick, Canada, 2007 June 3-6.

[7] S.M. Petoukhov, M.J. Brown and P.M. Mathew, "MAAP4-CANDU Application to the PSA Level 2 for the Point Lepreau Nuclear Generating Station Refurbishment Project", Proceedings of the 30th Annual Conference of the Canadian Nuclear Society, Calgary, Alberta, Canada, 2009 May 31 - June 3.

- The timing of the accident progression and accompanying thermo-physical and thermo-chemical phenomena,
- The effect of safety and normal operational system availabilities,
- Source terms for combustible gases, the resulting hydrogen and carbon monoxide concentrations in containment, and whether burning occurs,
- Fission product transport and retention within containment,
- The magnitude and nature of fission product releases from containment or directly to the environment,
- The timing and duration of challenges to containment integrity, and
- The effect of operator actions in mitigating severe accident consequences (reducing challenges to containment integrity and reducing fission product releases to the environment).

Five severe accident scenarios were selected for the PLR project:

1. Station blackout (SBO), with the loss of all cooling systems due to loss of electrical power to Group 1 and Group 2 equipment. In some cases moderator drained through failed channel bellows;
2. Small loss-of-coolant accident (SLOCA, 2.5% reactor inlet header break), with loss of Emergency Core Cooling (LOECC), loss of moderator cooling, and loss of other safety-related systems;
3. Stagnation feeder break (SFB) LOCA, with LOECC, loss of moderator cooling. The moderator draining was credited for all cases (fuel channel bellows at both ends of the fuel channel rupture), for one case (B1) the moderator drain was delayed.
4. Steam generator tube rupture (SGTR) with a LOECC and loss of the moderator cooling system; and
5. Shutdown state accident (SSA). The initiating event was a leak from the bearing seal of the shutdown cooling system (SDCS) pumps, with a simultaneous loss of the shutdown cooling system. The PHTS eventually drained almost down to the reactor header level, and this was combined with a LOECC and a loss of the moderator cooling system, shield cooling system and other safety-related systems.

In the analysis of each of the above accident scenarios, the reference case assumed no operator interventions and credited only a limited number of safety-related systems. The reference case was followed by a series of sensitivity cases assuming certain system availabilities, to assess their effects on accident progression. A total of more than 50 cases were analyzed. The timing of major events and fission product releases to the environment for five scenarios, which were considered representative cases having the highest frequency, are summarized here.

5.1 General Analysis Assumptions

For all level 2 PSA cases the following assumptions were common to all analyses:

- The reactor was shut down soon after accident initiation (except the SSA, where the reactor was shutdown before the initiating event);
- Main feed water, moderator, shutdown cooling and emergency water supply systems were not credited; but auxiliary feed water system, Emergency Power Supply (EPS), shield cooling system and Emergency Water Supply (EWS) to ECC HX (Heat Exchanger) were credited in some cases.
- Containment dousing system was credited;
- Passive autocatalytic hydrogen recombiners and containment isolation were credited;
- Containment leakage was modelled;
- The containment ventilation system was not modelled;
- Local air coolers (LACs) and operator interventions were credited in some cases;
- In all cases analyzed some type of ECC impairment was assumed.

For containment failure, it was assumed that both the inner and outer containment airlock seals of the doors (connecting reactor building and service building) failed together. An airlock seal blow-out, connecting the containment to the environment, was assumed to occur when the containment pressure reached 334.4 kPa (a).

5.2 Representative Cases (Cases with Highest Frequencies)

The representative sequences with the highest frequencies were simulations SBO-D1, SLOCA-E, SFB-C, SGTR-B and SSA-A. For these sequences there were differences in the assumptions of the system availability as shown in Table 1.

Table 1 Availability of various systems credited in representative sequences

Scenario/Case	SBO-D1	SLOCA-E	SFB-C	SGTR-B	SSA-A
Class III Power	No	Yes	Yes	Yes	Yes
Class IV Power	No	Yes	Yes	Yes	Yes
High pressure ECC (HP ECC)	Yes	Yes	No	No	No
Medium pressure ECC (MP ECC)	Yes	Yes	No	No	No
Low pressure ECC (LP ECC)	Yes	No	No	Yes	No
Loop Isolation	No	Yes	Yes	Yes	No
Emergency power supply (EPS) availability	72 h	Yes	No	Yes	No
Emergency core cooling heat exchanger (ECC HX)	No	No	No	No	No
SG Crash Cool Down	Yes	Yes	Yes	No	Not Applicable
Moderator Drain	4.2 kg/s	No	30 kg/s	No	No
Shield Cooling	No	No	No	No	No
Auxiliary Feed Water (AFW)	No	Yes	Yes	Yes	No
Main Steam Safety Valve	available	available	available	available	locked open

The SBO sequence was initiated by a loss of Class IV power followed by a loss of Class III power. When a fuel channel ruptured, both PHTS loops depressurized because loop isolation did not function with a loss of electrical power to the isolation valves. It was assumed that the channel rupture also failed the channel bellows, allowing moderator to drain out.

The SLOCA was initiated by a 2.5% break in the reactor inlet header (RIH), with a total break area of 0.00537 m^2.

The initiating event for the SFB scenario was an inlet feeder pipe break in PHTS Loop 1. The break was assumed be of a small size (normally from 2 to 17 cm^2) such that the coolant flow in the fuel channel stagnated, followed by the heat up and failure of the pressure tube. It was assumed that when the pressure tube fails the annulus between the pressure tube and the calandria tube becomes pressurized, resulting in the rupture of the bellows at both ends of the fuel channel and failure of the calandria tube. Through the break in the bellows the PHTS coolant and the moderator was assumed to leak into the containment. The leak rate was assumed to be 30 kg/s as shown in Table 1.

For the SGTR case, the initiating event was a single steam generator (SG) tube rupture with the break located in the “cold” SG leg, just above the top of the SG tube sheet.

The initiating event for the SSA scenario was a leak from one of the shutdown cooling system pump seals, which began 6 h after reactor shutdown, resulting in a coolant loss from both PHTS loops. This eventually led to moderator heatup, boil off, and progressed to severe core damage.

5.3 Major Results

5.3.1 Analysis Results for Five Representative Cases

Table 2 shows a summary of the MAAP4-CANDU calculations of the timing of major events and the fractional release to the environment (up to 500,000 s) of the initial core inventory of the active isotopes of Cs, Rb and I.

Table 2 Summary of significant event timings for the most the representative cases

Event/Case	SBO-D1	SLOCA-E	SFB-C	SGTR-B	SSA-A
SG dry (h)	0.8	2	33	10.7	138
PT/CT rupture (h)	3.8	41	38	13.1	N/A
Core disassembly starts (h)	76	17	1.4	52	13.2
Containment fails (h)	23	47	38	37	37.6
CV fails (h)	not applicable	81	54.5	120	66
MCCI begins (h)	not applicable	92	63	not applicable	78
Calandria Vault floor failure (h)	not applicable	not applicable	137	not applicable	not applicable
Percentage of initial inventory of the active isotopes (Cs+Rb+I) released to environment at 500,000 s	3.2%	2.7%	6.8%	12.8%	0.55%

The earliest containment failure was predicted for SBO-D1 at 23 h, due to over-pressurization as a result of steaming in containment; the core disassembly began later at 76 h. The containment failure time was longest at 47 h for the SLOCA-E sequence with core disassembly beginning at 17 h. The earliest core disassembly was predicted at 1.4 h in SFB Case C, since the moderator drained from the calandria vessel at a fast rate of 30 kg/s and no ECC was credited to cool the fuel in the fuel channels.

The SGTR-B case showed the greatest release of the Cs+Rb+I fission products to the environment at ~13%. This was primarily due to the core disassembly starting relatively late (~52 h), when the containment had already failed (at ~37 h). The volatile fission products were thus not deposited on the containment walls to the same degree as in other cases, but instead were released to the environment through the failed containment airlock seal.

In the SSA-A case, the containment failed well after the vast majority of the Cs+Rb+I fission products had been released from the fuel. The wet containment environment absorbed much of these isotopes so, when the containment failed, they were largely trapped within the containment.

The analysis of the cases performed also showed that the most efficient systems to delay core disassembly are ECC LP and SG AFW; the most efficient systems to delay containment failure are LACs and ECC LP; and the most efficient system to prevent calandria vessel failure is shield cooling.

5.3.2 Additional Results on Severe Accident Management Measures for PLR Project

As a result of the level 2 PSA activities the PLR Project recommended and evaluated the implementation of various severe accident management measures. Two new systems are being installed as part of the Point Lepreau refurbishment. One of the SAM measures was to add water with a flowrate of ~3 kg/s from an external source to the reactor vault 24 h after accident initiation when no other system to prevent containment failure was available. This SAM measure was initiated by the operator on low water level in the reactor vault. The second SAM measure was to use a containment filtered venting system. This system would discharge steam/gas mixture from the containment to the environment based on high containment pressure set points when no other system to prevent containment failure was available.

Several cases were analyzed to explore these SAM measures. Results of an additional case (SFB-D2), which does not credit any other system to prevent containment failure, showed that it was possible to keep the pressure in containment below 320 kPa (a) by venting the containment at relatively low pressure set-points (i.e., 180/150 kPa (a) for the system on/off pressures). Make-up water was added to the reactor vault after 24 hrs from accident initiation to prevent calandria vessel failure.

Similar to SFB Case D2, results of a SLOCA Case A2G, which does not credit any other system to prevent containment failure, indicated that it was possible to keep the pressure in containment below 300 kPa (a) by venting the containment at relatively low pressure set-points (i.e., 180/150 kPa (a) venting on/off), if make-up water was added to the reactor vault beginning 24 hrs after accident initiation.

Analysis results showed that water make-up to reactor vault could prevent calandria vessel failure and molten corium-concrete interaction.

6. Summary

The major features of MAAP4-CANDU v4.0.5A+ were described in this paper. More than 50 severe core damage accident sequences were analyzed to support PSA Level 2 activities for the Point Lepreau Refurbishment Project:

- Timings of significant events for the scenarios in a Point Lepreau station, as well as the estimated fission product releases, were obtained;

- From the most representative cases analyzed, the earliest core disassembly (1.4 h) was predicted in SFB Case C, because of moderator drain;

- From the most representative cases analyzed, the earliest containment failure was predicted in SBO Case D1 (23 h), due to steaming from the calandria vessel;

- From the most representative cases analyzed, the earliest calandria vessel failure (55 h) was predicted in SFB Case C;

- From the most representative cases analyzed, the largest fission product release to the environment (~13% of the Cs+Rb+I inventory) was predicted in SGTR case B;

- The most effective systems to delay core disassembly are ECC LP and SG AFW;

- The most effective systems to delay containment failure is LACs and ECC LP;

- The most effective systems to prevent calandria vessel failure is shield cooling

- Analysis results to support SAM measures showed that it was possible to keep the pressure in containment below the containment failure pressure by venting the containment at relatively low pressure set-points, provided that make-up water was added to the reactor vault after 24 hrs from accident initiation to prevent calandria vessel failure.

Time Window for Steam Generator Secondary Side Reflooding to Mitigate Large Early Release Following SBO-Induced SGTR Accidents

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Abstract

Steam generator secondary side reflooding has been implemented in some PWR power plants as a practical severe accident management measure to mitigate fission product release for spontaneous steam generator tube rupture severe accidents. The current work focuses on station black-out induced SGTR accidents, which would progress much faster with the reactor uncovered and fission products released much earlier than spontaneous SGTR accidents. Therefore, the plant staff would have a much shorter time in response to SBO-induced SGTR. The time window available for the plant staff to prepare mobile pump and fire water for injection of water into the SG secondary side is a critical parameter governing if this SAM measure could be successfully achieved. The current work uses the MELCOR severe accident analysis code to analyze the SBO induced SGTR accident progression and to characterize the boundary conditions for fission product retention onto the SG secondary side. The inherent plant safety features, such as establishment of a pool on SG secondary side by accumulator injection and aerosol retention by inertial impaction as well as turbulence deposition on SG tube and structure surfaces, were found to postpone large early release by a number of hours, making time for accident management to refill SG to probably avoid large early release of fission product to environment.

1. Station black-out induced steam generator tube rupture accidents

Severe accident induced steam generator tube rupture (SGTR) is a concern because the steam generator (SG) tubes are parts of the reactor coolant pressure boundary (RCPB) and failure of the SG tubes may lead to fission products bypassing the containment. The SG tube integrity may be challenged by high temperature and high pressure conditions and may have a potential to fail due to creep rupture in a broad category of station blackout (SBO) severe accident scenarios represented by the TMLB' sequence. In the TMLB' sequence, the primary side pressure is maintained high by the repeated cycling of the pressurizer power operated relief valve (PORV), the secondary side is depleted of coolant inventory and becomes dry due to loss of feed water, and the secondary side pressure is governed by the repeated cycling of the steam relief valve. Once the steam relief valve fails to reclose during the repeated cycling, the SG tubes would be subjected to the most severe temperature and pressure challenges.

Once the reactor core starts to heatup during the hypothetical TMLB' station blackout severe accident, the hot leg counter-current gas natural circulation would be established (Figure 1) if the cold leg loop seal is plugged with water, transferring heat from the reactor core to the pipings of the hot leg, surge line and steam generator. The hot leg counter-current natural circulation consists of an outbound flow and a return flow. The outbound flow starts from the reactor vessel upper plenum to the SG outlet plenum, carrying the relatively hot gas in the hot leg upper part and in a fraction of the SG tubes (hot tubes). The return flow carries the relatively cold gas in the remaining SG tubes (cold tubes) and in the hot leg lower part. The SG inlet plenum is where the outbound flow mixing with the return flow takes place. The hot stream forms a rising plume in the SG inlet plenum once exiting the hot leg upper part. The cold stream returning from the SG outlet plenum cools the gas surrounding the hot plume. The hot plume entrains a large fraction of the return cold flow when rising and therefore its temperature decreases. As a result of SG inlet plenum mixing, the SG tube temperature is lower than the hot leg temperature and the SG tubes experience less severe thermal challenges than the hot leg and surge line.

The potential of severe accident induced SGTR under such conditions was recognized early in the U.S. NRC Severe Accident Risks report[1], in which the likelihood of severe accident thermally-induced SGTR assessed by an expert panel could be very small for tubes which were flaw free, but could be a concern if flaws pre-existed in SG tubes. For tubes which were flaw free, the later detailed thermal-hydraulic analysis[2] also concluded that the first RCPB failure would be the surge line or hot leg, thus eliminating the potential for severe accident induced SGTR. For the case of pre-existing flaws in SG tubes, a creep rupture model has been developed and validated by tests for the degraded SG tubes under severe accident conditions[3], and the U.S. NRC[4] has developed a general methodology to assess the severe accident induced SGTR probability, which appeared significant at least for severely degraded steam generators.

Depending on the initial SG tube flaw size, the SG tube creep rupture failure may lead to a leak only in the flawed tube or a catastrophic rupture not only in the flawed tube but probably also in the adjacent tubes due to cascading failures. Both experimental and analytical evidence[5] have shown that if the initial flaw size is sufficiently small, jet impingement erosion from the induced tube leakage could not damage the adjacent tubes and therefore leaking from a single tube could not depressurize the primary system. In such a single tube failure case with a high primary pressure maintained even after the SG tube failure, the surge line or hot leg will be expected to fail also within minutes following the SG tube failure, and therefore the fission product release into the SG secondary side will be very limited.

On the other hand, large early release of fission product into the environment may be assumed if a catastrophic rupture is induced in the degraded tube with a sufficiently large flaw size and the rupture subsequently causes cascading failures in the adjacent tubes, and if fission product retention on the secondary side and accident management are not taken into account. The dynamic behavior of the SG tube axial crack after creep rupture failure was investigated to show that the crack opening rate increases sharply with increase of the crack length and gas temperature[5]. With a sufficiently large rupture flow, the hot leg counter-current natural circulation and steam generator inlet plenum mixing would be disrupted and therefore the SG tube temperature would increase to a level closer to that of the hot leg. The SG tube temperature elevation further increases the crack opening rate and consequently the effective velocity of the jet impinging the adjacent tubes. Moreover, the augmented rupture flow momentum would likely cause tube-to-tube contact and even damage, as evidenced by the Mihama SGTR event[6].

This work focuses on the conditional probability of SBO induced SGTR accidents and accident progression of the hypothetical fission product bypass event involving SG tube cascading failure. The event chronology and fission product release boundary conditions in several phases were studied using the MELCOR severe

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[1] U.S. NRC, Severe accident risks: An assessment of five U.S. nuclear power plants, U.S. Nuclear Regulatory Commission Report, NUREG-1150, 1990.

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[6] MacDonald, P.E., Shah, V.N., Ward, L.W., Ellison, P.G., Steam generator tube failures, U.S. Nuclear Regulatory Commission Report, NUREG/CR-6365, 1996.

accident analysis code. The time window and accident management to mitigate large release were suggested as a result of the analysis.

2. Probability of SBO induced SGTR accidents

Although the SG tubes which are flaw free would not fail before the surge line or hot leg, the degraded SG tubes might have a potential to fail first in hot leg counter-current natural circulation. The conditional probability of SG tube failure mainly depends on the tube flaw characterization and the thermal challenge to the SG tubes relative to those to the surge line and hot leg. In this section, the tube flaw characterized by the probability density functions about the flaw frequency, size and location is introduced, then the thermal-hydraulic responses of the SG tubes, surge line and hot leg predicted by MELCOR and CFD analyses are presented, and finally these two are integrated to assess the probability of SBO induced SGTR accidents[7].

As of 2007, around 43% of the US PWR power plants used thermally treated Alloy 690 steam generator tubes, and 25% used thermally treated Alloy 600 tubes, with the remaining balanced by power plants using mill annealed Alloy 600 tubes. Unlike mill annealed Alloy 600 tubes, for which stress corrosion cracking is the major inservice degradation mechanism[6], wear due to foreign object and tube support is the major inservice degradation mechanism for thermally treated Alloy 600 and 690 tubes[8,9].

There has been no occurrence of tube rupture for the U.S. power plants using thermally treated Alloy 690 or 600 tubes, but a few forced outages have occurred due to primary to secondary leakage. Altogether there were only three forced outages due to tube leakage for power plants using thermally treated Alloy 690 tubes: two leakages were caused by foreign object wear and one was caused by a fabrication flaw. There were only three forced outages due to tube leakage for power plants using thermally treated Alloy 600 tubes: all leakages were caused by foreign object wear. Therefore the steam generator operating experience showed that foreign object wear was a relatively important inservice degradation mechanism for thermally treated Alloy 690 and 600 tubes and would cause severe tube degradation. An additional reason to consider the tube flaw caused by foreign object wear in SBO induced SGTR accidents is that foreign objects tend to accumulate at the top of tubesheet, where the thermal challenge to SG tubes is most severe in the whole tube bundle during hot leg counter-current natural circulation.

The severity of the SG tube degradation is indicated by the number of flaws existing between two steam generator inservice inspections (the flaw frequency), the flaw through-wall depth and the flaw length. The power plant periodic steam generator inservice inspection reports contain the number of flaw indications detected and the flaw size measured using the eddy current testing technique. Also reported are the flaw locations with respect to the tube length and the tube's tubesheet position. A survey of numerous steam generator inservice inspection reports helped establish statistically the flaw distributions. The survey focused on those flaws caused by foreign object wear due to its relative importance for the new generation steam generators. As a result of the survey, a database of 445 SG tube flaws caused by foreign object wear has been set up with the observed flaw frequency, size and location reported in the steam generator inservice inspection reports from power plants using the thermally treated Alloy 690 or 600 new generation tubing materials. Only those steam generators that have ever experienced foreign object wear were considered in the database. Altogether the database consists of about 121 steam generator years (SG Years) of operating experience.

The database has been used to derive the statistical distribution of the SG tube flaw frequency, depth, length and location[7]. In hot leg counter-current natural circulation, the gas recirculation along the SG tubes heats up

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7 Liao,Y., Guentay, S., Potential steam generator tube rupture in the presence of severe accident thermal challenge and tube flaws due to foreign object wear, Nuclear Engineering and Design 239, 1128-1135, 2009.

8 Karwoksi, K.J., U.S. operating experience with thermally-treated Alloy 600 steam generator tubes, U.S. Nuclear Regulatory Commission Report, NUREG-1771, 2003.

9 Karwoski, K.J., Makar, G. L., Yoder, M.G., U.S. operating experience with thermally-treated Alloy 690 steam generator tubes, U.S. Nuclear Regulatory Commission Report, NUREG-1841, 2007.

the tube structure and loses its internal energy. The tube structure temperature attains the highest temperature at the SG entrance, and the temperature gradually decreases along the gas flow direction. At the SG entrance, the tube-to-tube temperature difference is expected due to the three-dimensional gas mixing phenomenon in the SG inlet plenum. Since the SG tube thermal response is site dependent, the tube flaw characterization must be coupled to the local thermal response to assess the failure probability. With regard to the SG entrance, Figure 2 shows the distribution of temperature and degraded tubes, with the temperature prescribed by the CFD analysis and the location of degraded tubes reported by SG inservice inspection reports[7].

The thermal response history for the heat structures at the SG entrance and the surge line is shown in Figure 3. MELCOR was used to predict the average temperature for the SG tubes in the hot and cold regions as well as the surge line, with regard to a Westinghouse 4-loop nuclear power plant. The detailed accident progression predicted by MELCOR will be discussed later. The MELCOR average temperature for the hot region was used to estimate the temperature for the hottest region with a temperature peaking factor inferred from the CFD analysis[10].

With the established tube flaw distribution and local thermal-hydraulic response, the creep rupture model with the life fraction rule was used to evaluate the failure sequence among the SG tubes, surge line and hot leg. With the same thermal-hydraulic response, the lifetime of the degraded tube is shorter for a more severe degradation. This fact is accounted for using a penalty factor in the SG tube creep rupture model which is a function of the flaw depth and length.

The methodology to assess the SG tube failure probability using the Monte Carlo random walk technique can be summarized now. First, sample the relevant parameters (flaw frequency, location, depth and length, temperature as well as pressure) according to their established probability density functions. Second, for a specified flaw location, the local thermal-hydraulic response is used with the Larson Miller creep rupture model to identify the critical penalty factor (or stress magnification factor, which quantifies the severity of degradation and depends on the flaw size) above which the SG tube would fail before the surge line and hot leg. Then the critical penalty factor is used with the flaw size probability density function to infer the SG tube failure probability. This process is repeated for a number of times prescribed by the sampled flaw frequency. Figure 4 shows the cumulative density function of the SG tube failure probability after 10,000 Monte Carlo random walks, with the mean and range factor equal to 0.025 and 3.8, respectively[7].

3. Accident progression of SBO induced SGTR accidents

The TMLB' station blackout accident was simulated using MELCOR for a 4-loop Westinghouse nuclear power plant. In the MELCOR nodalization, the loop with the pressurizer is modeled as a single loop while the other three loops are lumped together and modeled as one. The thermal-hydraulic nodalization of the core is a 5-ring, 4-level geometry. The reactor core nodalization is represented by as 5-ring, 12-level cell structure with three core cells per thermal-hydraulic level.

The event begins with the loss of A/C power and steam generator feedwater supply. The turbine and reactor trip immediately after the accident initiation. The steam relief valves on the pressurizer loop steam generator are assumed to fail upon first challenge and remain stuck open thereafter. The initial water inventory in the steam generator secondary side boils away, resulting in a loss of heat sink. The secondary side for the pressurizer loop becomes dry in a time earlier than those for the other three loops, since the steam relief valves for the other three loops are assumed working normally in the repeated cycling mode.

As the primary side pressure increases due to loss of secondary heat sink, the pressurizer PORVs open, and thereafter the primary pressure is maintained high by the repeated cycling of PORVs. This constitutes a loss of coolant accident which will lead to decrease of primary water inventory and eventual core uncovery.

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[10] Boyd, C.F, Helton, D.M., Hardesty, K., CFD analysis of full-scale steam generator inlet plenum mixing during a PWR severe accident, U.S. Nuclear Regulatory Commission Report, NUREG-1788, 2004.

Hot leg counter-current natural circulation is predicted to begin at about 9,300s when the primary water level has decreased below the top of the core. As the reactor core heats up, gas natural circulation transfers a portion of the decay and oxidation heat to the wall structures in the hot leg, surge line and SG tubes. Heatup is accelerated as more and more fuel cladding is oxidized. The thermal response during this heatup phase for the surge line and SG tubes is shown in Figure 3 and was discussed in the previous section. The heatup for the surge line is further accelerated after the pressurizer dries completely at about 12,800s. Fuel cladding oxidation starts at about 12,000s and temperature excursion starts at about 14,000s, after which significant core melt begins with fission products released from the fuel. Due to the very hot gas coming out of the reactor vessel, there is a potential for excessive heatup of the surge line, hot leg and SG tubes, along with the potential failure of these reactor coolant pressure boundaries. The probability of the SG degraded tube to fail first under these conditions was assessed in the previous section.

Depending on the initial crack size, the degraded tube would be induced to leak or rupture once creep failure occurs. Leak from a sufficiently small crack would not cause damage to the adjacent SG tubes and therefore the primary pressure would be maintained at a high level until the other pressure boundaries also fail. Once the pressure boundaries of the surge line or hot leg fail, the primary side depressurizes to substantially reduce the bypass flow through leakage of the failed SG tube. Only limited fission products would bypass the containment in this scenario.

On the other hand, leak from a sufficiently large crack would cause the crack to open widely in a short time. The crack opening rate is larger with a longer crack length and a higher gas temperature[5]. Leakage from a widening crack would comprise hot leg counter-current natural circulation and SG inlet plenum mixing, resulting in a higher temperature of the gas eroding the crack surface. Therefore, leak from a sufficiently large crack would develop into rupture in a short time. High temperature gas jet from a large crack or rupture of the failed SG tube would impinge the adjacent tubes with a very high velocity, sufficient to cause damage in the adjacent tubes within a few minutes[4] and result in the cascading tube failure. The jet momentum may also cause tube-to-tube contact and damage, as evidenced by the Mihama SGTR event[6].

The current work proposes a scoping analysis assuming that two rings of the adjacent tubes surrounding the failed tube (a total of 25 tubes) would be damaged in the cascading tube failure. The cascading failure is considered at the top of the tubesheet where the thermal challenge to SG tubes is most severe and the foreign object is most likely to accumulate and cause tube degradation. In the MELCOR analysis, the dynamic behavior of crack opening and jet impingement erosion was not modeled, but a total of 25 SG tubes were assumed to fail completely and immediately once the degraded tube induced to fail for simplicity of analysis. The overall accident progression would not be affected significantly by the simplification since this dynamic process is expected to complete in a few minutes. The flow area of 25 SG tubes is almost equivalent to a 6 inch break in the primary side, which would depressurize the primary side in a few minutes, rendering a lot more SG tubes besides the assumed 25 tubes to be damaged very unlikely.

From this point at about 14,900s, the primary side depressurizes quickly due to the cascading tube failure until the accumulator starts to inject coolant into the reactor core. Since the overall system opens at the SG steam relief valve level, no accumulator liquid inventory would bypass the system, unlike that in the conventional loss-of-coolant accidents initiated from a cold leg break. The accumulator coolant immediately covers the reactor core and the primary water level rises to the hot leg level. Afterwards, the reactor core starts to boil again and generates a large void inside the reactor vessel. The void generation expels a fraction of the reactor vessel liquid into the secondary side through the ruptured tubes. As more and more steam generated from pool boiling in the core, more liquid is carried by steam to the secondary side through the ruptured tubes, establishing a water pool on the top of tubesheet similar to that in the pressurizer when the PORVs at the top of the pressurizer open. Figure 5 shows such behaviour of the secondary side water level predicted using MELCOR, where the top of tubesheet is at the level of about 10m and the bottom of the reactor vessel lower head is at the level of 0m. The lifetime of the secondary side pool spans about 2 hours. Its water level starts to decrease after attaining a maximum water level of about 7m above the top of tubesheet.

Figure 6 shows the evolution of the secondary side average gas temperature. Prior to induced tube rupture, the secondary side temperature is in phase with the heatup of the SG tubes shown in Figure 3. After induced tube rupture, it is in phase with the accumulator injection to decrease and then maintains at a relatively low level during the lifetime of the secondary pool established by the accumulator injection. If no accident management is implemented to refill the SG secondary side, even liquid inventory in the lower head would be lost and severe core damage would occur with molten debris relocating to the lower head at about 21,000s. Thereafter, the secondary side temperature increases again and the increase would last until the lower head fails. The MELCOR simulation terminates when the lower head temperature increases to 1273K.

4. Suggestions for accident management to mitigate fission product release

The induced SGTR accident progresses much faster than the spontaneous SGTR accident. Furthermore, unavailability of power and disabled engineered systems make the plant staff more difficult to control the accident progression in such a hypothetical scenario. Once the reactor exit temperature reaches 923K at about 13,100s, the plant staff begins to monior the plant status and manages accident according to the severe accident management guidance. Afterwards, efficient accident management must be carried out within about half an hour, otherwise the degraded SG tube would be induced to rupture. Within this half an hour period, the plant staff might be able to depressurize the primary side using PORVs provided that power is available. If power is still unavailable, it would be very difficult to efficiently refill the SG secondary side using the mobile pumps and fire water within such a short time. Therefore, power recovery must be successful within half an hour starting from entry into the severe accident domain, otherwise fission product release from the primary to secondary side is deemed to take place.

While the plant staff continues to monitor the plant status and prepares to inject water into the steam generator after fission product starts to release to the secondary side through the ruptured tubes, the plant inherent safety features would retain most of the fission product onto the secondary side, which may avoid catastrophic environmental release. These plant inherent safety features include the pool on the secondary side self-established by the accumulator injection and the substantial SG tube and structure surfaces available for fission product deposition in case the self-established pool becomes dry later. The mechanisms for these inherent safety features to retain fission products have been and are under investigation in the ARTIST project performed at Paul Scherrer Institute[11]. This analytical work sets up preliminarily the boundary conditions for the self-established pool and the secondary side temperature, such as the pool level and life time, as well as the aerosol size inferred from the fluid temperature, which will be used with the final outcome of the ARTIST project to assess the fission product retention capability of these inherent safety features.

As a result of this work, Figure 7 illustrates an example showing effect of the inherent safety features and accident management to mitigate fission product release into environment. Though the mitigated release is not based on actual data, the release curves capture the basic retention characteristics in such hypothetical scenarios. The time window for release mitigation consists of three phases, with each phase of mitigation governed by a different retention approach. The first phase starts from the accumulator injection and ends when the self-established pool becomes dry. The fission product retention is characterized by pool scrubbing enhanced by internal structures. The second phase continues from a dry secondary side till accident management is executed to refill steam generator. The retention is characterized by fission product vapor condensation and aerosol deposition on the SG tube and structure surfaces by inertial impaction and turbulence deposition. The third phase corresponds to successful accident management to refill steam generator, with fission products retained by pool scrubbing enhanced by internal structure. As shown, though a large fraction of fission product is released from the primary side, the plant inherent safety features would retain most of the release onto the SG secondary side for a number of hours following tube rupture, and thus makes time for accident management execution to probably avoid large early release into environment.

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[11] Guentay, S., Suckow, D., Dehbi, A., Kapulla, R., ARTIST: introduction and first results, Nuclear Engineering and Design 231, 109-120, 2004.

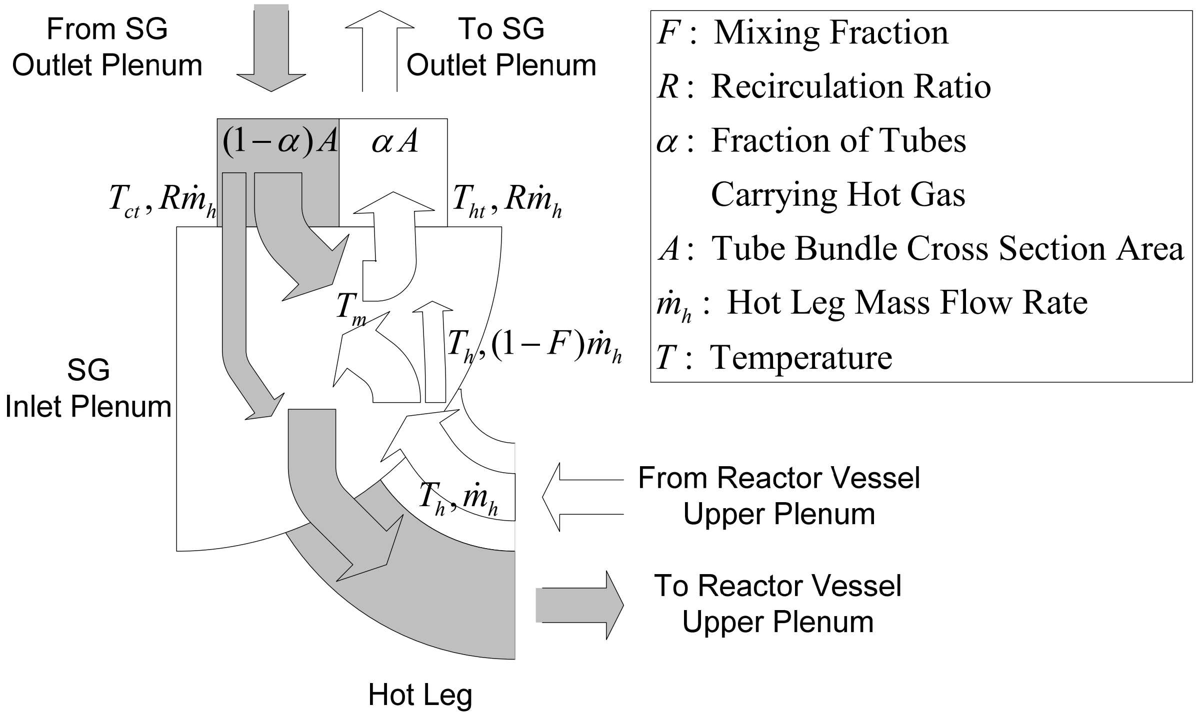

Figure 1: Steam generator inlet plenum mixing during the hot leg counter-current natural circulation

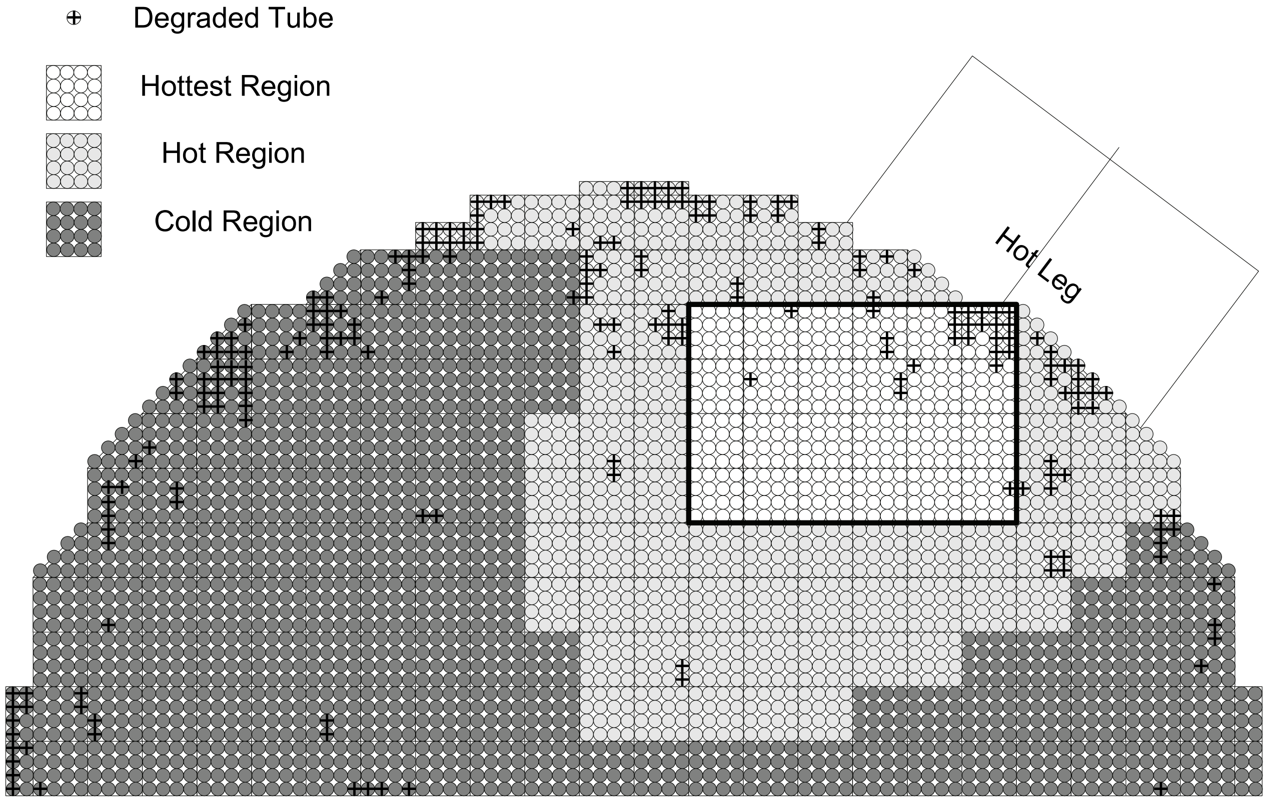

Figure 2: Distribution of temperature and degraded tubes at the steam generator entrance

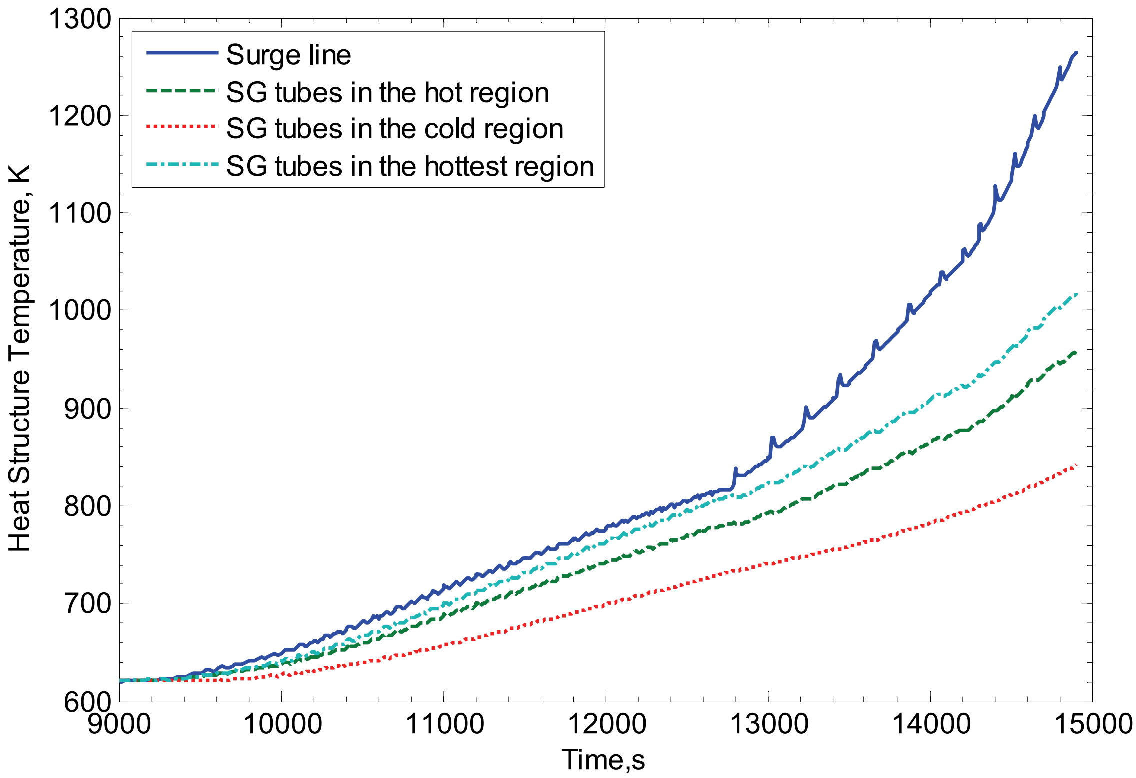

Figure 3: Heat structure temperatures at the surge line and SG entrance

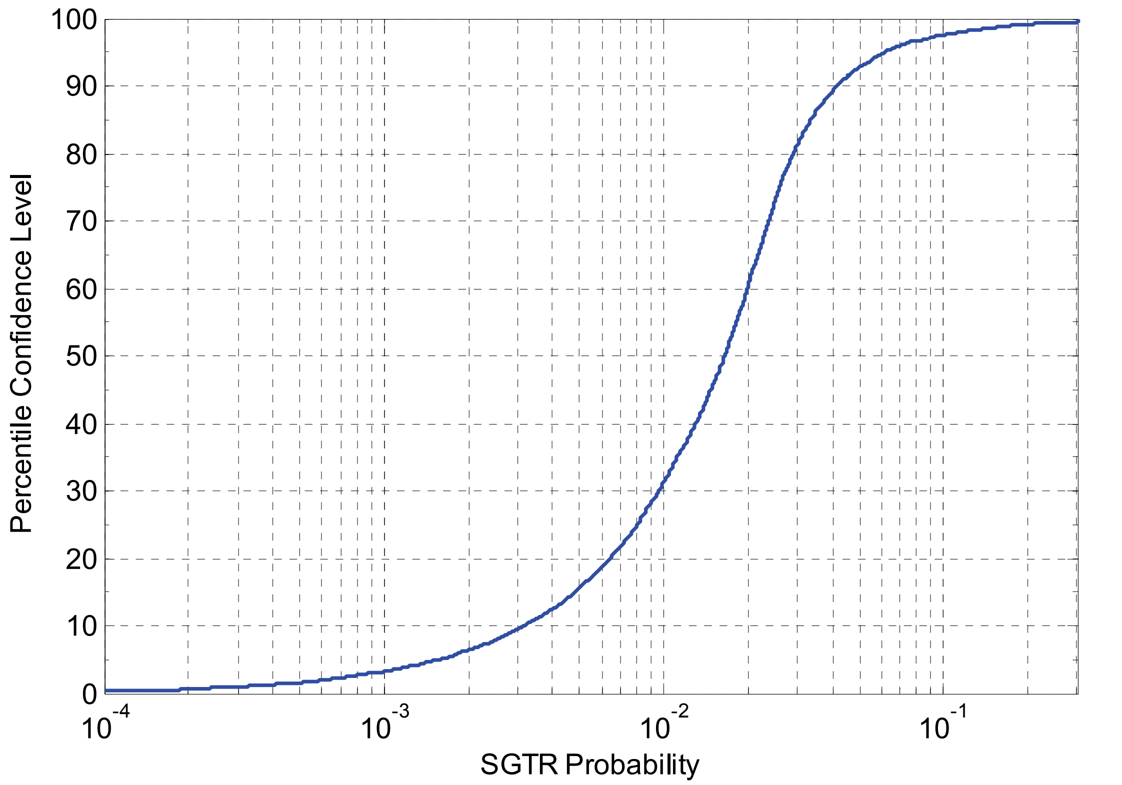

Figure 4: Distribution of SGTR probability induced by severe accident thermal challenge

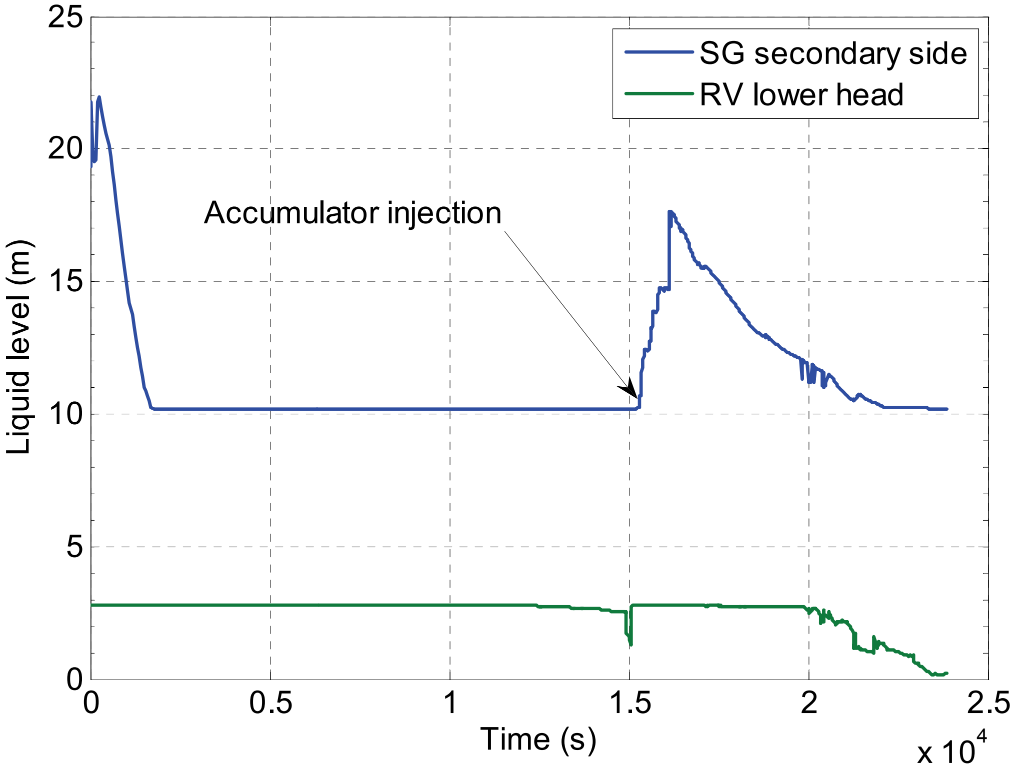

Figure 5: Accident progression with the water levels in SG secondary side and lower head

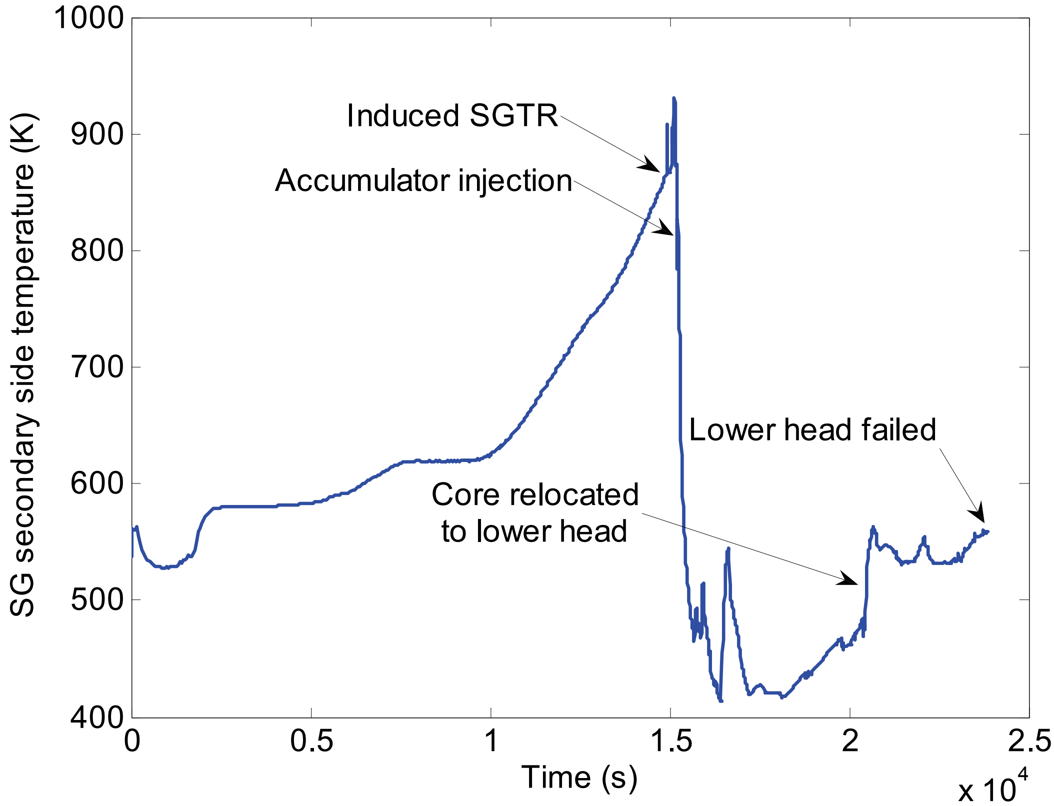

Figure 6: Accident progression with the SG secondary side temperature

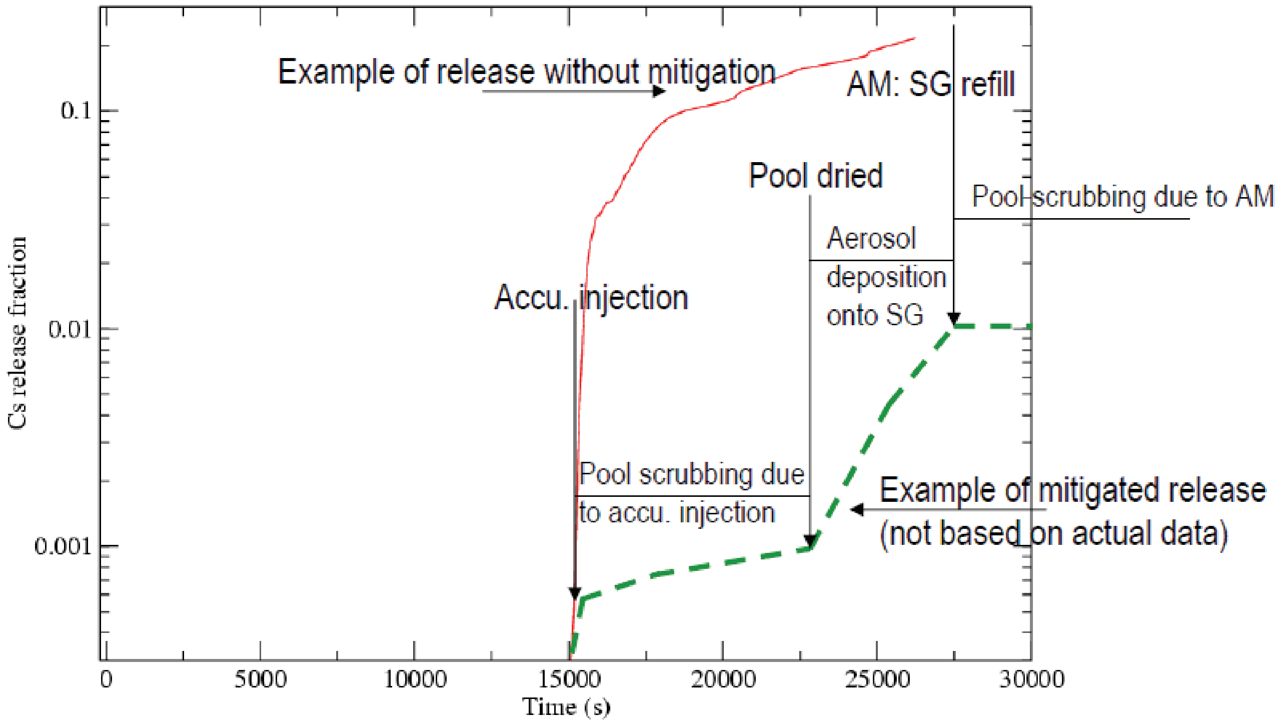

Figure 7: An example showing effect of fission product release mitigation

On the Effectiveness of CRGT Cooling as a Severe Accident Management Measure for BWRs

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Abstract – To quantify the effectiveness of control rod guide tubes (CRGT) cooling as a potential severe accident management (SAM) measure in BWRs, two analysis methods are adopted at KTH: a lumped-parameter analysis of accident sequences performed by the MELCOR code at system level; and a mechanistic multi-dimensional analysis of heat transfer in the corium-filled lower head by the Effective Convective Model (ECM) developed for modeling of corium behavior in a complex geometry as the BWR's lower plenum with a forest of CRGTs. The dual approach leverages on the strength of the two methods (MELCOR/ECM), and therefore increases the reliability of the assessment. The current paper is focused on the MELCOR calculations. The first-cut results show that the nominal flowrate (10.5kg/s) of CRGT cooling is sufficient to maintain the integrity of the vessel in a BWR of 3900 MWth, if the water injection is activated no later than 1 hour after scram. For late recovery of the CRGT cooling (later than 1 hour after scram), a higher flowrate rather than the nominal is needed to contain the melt in the vessel. For instance, if water injection through CRGTs is activated after 2 hours following scram, much higher flowrate (~40kg/s) is required for in-vessel retention (IVR).

Keywords: severe accident, in-vessel retention, coolability

1. Introduction

This paper is concerned with in-vessel retention (IVR) of corium in a hypothetical severe accident of boiling water reactor (BWR), assuming that coolant flow through the control rod guide tubes (CRGTs) remains available by severe accident management (SAM).

The IVR is preferred in nuclear power safety, since it arrests the radioactive materials in the reactor vessel, relieving the requirements on other mitigative/protective measures to ensure containment integrity. Although it has been implemented in few pressurized water reactors [1] (PWRs) by applying external cooling of the vessel, the IVR remains a challenge for most of the existing PWRs (e.g., the ones larger than 600MWe), due to the limiting heat flux (CHF) and surface area of the lower head for decay heat removal.

The IVR has not been applied to any BWR so far, but the BWRs have more penitential for IRV in term of their external cooling area of the lower heads which are much larger than those of PWRs. More importantly, the CRGT cooling system of a BWR in operation can be adapted as another/additional avenue for the IVR through severe accident management (SAM). The consideration is due to three folds: i) the modification will be minimal by capitalizing on the existing cooling system; ii) the forest of CRGTs provides large area for heat transfer from corium to coolant; iii) the flowrate of the CRGT cooling (~10kg/s) is small so that it can be ensured by introducing a battery-driven pump.

However, it is not straightforward to quantify the efficiency of the CRGT cooling, because of high Rayleigh number of melt pool convection, long transient of accident progression, and the presence of phase changes. The difficulty is further increased in a BWR whose lower plenum contains a forest of penetrations (CRGTs, IGTs) which significantly complicates the geometry, heat transfer, and fluid flow patterns induced by cooled CRGTs. Any experimental study on such a prototypical melt pool is impractical. Heat transfer correlations were developed to describe turbulent natural convection of melt pools, based on small-scale experiments with simplified geometries and corium simulants. The correlations are not directly applicable to the task at hand. Computational Fluid Dynamics (CFD) has also been employed to study turbulent natural convection in volumetrically heated liquid pools and fluid layers under small scales [2-3], but CFD simulations at high Rayleigh numbers are computationally expensive even for a steady-state problem. There is a clear need to have an analysis method which is sufficiently-accurate and computationally-efficient for simulation of melt pool heat transfer in the lower head. Upon the stringency of high-fidelity prediction, Effective Convective Model (ECM) and Phase-change Effective Convective Model (PECM) were developed at KTH [4-7], which incorporate the advantages of the modern CFD method and the available correlation-based method, being able to simulate melt pool behavior in a complex geometry as the BWR's lower plenum with a forest of CRGTs.

The ECM and PECM were applied to simulation and analysis of melt pool heat transfer in the lower plenum during a severe reactor accident of a BWR of 2500 MWth. The key findings are as follows [8]: In case of formation of melt pool with a thickness (height, or depth) less than 0.7 m in the BWR lower plenum, the CRGT cooling at nominal water flow rate, i.e. 62.5 g/sec per CRGT, is sufficient to remove the decay heat generated in the melt pool, and protect the vessel wall from thermal attack. In the case of a melt pool with depth higher than 0.7 m, the CRGT cooling is insufficient, and the vessel wall is predicted to fail in the section connected to the uppermost region of the melt pool; Additional cooling measure (e.g. external cooling, and/or increase of CRGT water flow rate) is needed for protection of the vessel from failure. The ECM/PECM simulation results and key findings suggest that CRGT cooling possesses a high potential as an effective and reliable mechanism to remove the decay heat from a melt pool formed in the BWR lower plenum. Thus, the CRGT cooling presents a credible candidate for implementation as a SAM measure in BWRs.

While the ECM/PECM method provides an efficient and computationally affordable tool for mechanistic multi-dimensional analysis of melt pool heat transfer in the lower head, the effects of system dynamics and accident progression on melt pool heat transfer were not directly considered in the analysis. The influence of coolant injection on other phenomena was also neglected. This is because ECM/PECM simulations were only focused on the melt pool in the lower head, without any link and interactions with other systems and phenomena. For instance, the core degradation and relocation are important for resulting melt pool in the lower plenum. But this cannot be captured in the ECM/PECM method; instead the boundary conditions (e.g., melt mass, compositions and initial temperatures) have to been assumed in the analysis. To lift this limitation, the whole accident progression (from core degradation, to melt relocation, and to melt pool formation in the lower head) must be simulated. Such a systematic analysis is realized here by using MELCOR code for the severe accident simulation. The dual approach leverages on the strength of the two methods (MELCOR and ECM/PECM), and therefore increases the reliability of the assessment. For instance, the MELCOR analysis provides the melt conditions for the PECM simulation, while the ECM/PECM calculations in turn verify the MELCOR prediction.

The present study places the focus on the MELCOR calculations to assess the capacity of the CRGT cooling in severe accident scenarios. The reactor chosen as reference is a boiling water reactor of 3900 MWth. More details of the reference reactor and the results are presented in the following sections. The melt conditions obtained in the lower plenum will be used in ECM/PECM simulation for further quantification of CRGT cooling as a potential SAM measure in BWRs.

2. Reference reactor and MELCOR nodalization

The reference reactor chosen in the study is a boiling water reactor whose main technical data are as shown in Table.1.): As shown in Figure 1, the BWR feathers a lower head which accommodates internal recirculation pump and numerous penetrations which hold control rod guide tubes (CRGTs) and instrument guide tubes (IGTs). The internal pump design eliminates the external recirculation loops and can avoid the risk of pipe breach. During normal operation, coolant a rate of 62.5 g/s per rod is flowing through the CRGTs to cool the drive mechanism of the control rods. The vessel of a boiling water reactor is larger than that of a pressurized water reactor. This is because it has a larger core (and the mass inventory of fuel) as well as the internal recirculation pumps, steam separators and steam driers. The fuel assembly is surrounded by a square canister to form a channel for coolant. The gaps between the canisters are called bypass of coolant flow. The CRGTs are the primary support of the core, and the core plate of BWR does not bear the weight of the fuel and its canisters. The great number of CRGTs does not only affect progress of melt relocation, but also significantly contributes to the percentage of metal mass in corium. The boiling water reactor has an inerted containment which is much smaller than that of PWR. Due to the smaller size, the containment could not only depend on its volume to constrain pressure increase during abnormal events (e.g., a large break LOCA). The condensation pool (wetwell) is designed to suppress the pressure during any abnormal event which releases steam. During severe accident, the reactor cavity (lower drywell) will be filled with water from the wetwell. The cavity flooding is the corner stone of the severe accident management (SAM) strategy in Swedish BWRs. The idea is to promote melt fragmentation and quenching, and form a coolable debris bed on the drywell floor so as to avoid containment failure due to molten corium concrete interaction (MCCI). This SAM measure increases the risk of steam explosion, while we still need answer whether the corium debris is coolable and stabilized in the "wet core catcher". These topics (ex-vessel coolability and steam explosion energetics) are investigated in other projects at KTH.

The present study is concerned with in-vessel coolability with the CRGT cooling in place. The integrated code MELCOR 1.8.5 for severe accident simulation in light water reactors [9] was employed in the calculations. The code can model *(i)* thermal-hydraulic response of the reactor cooling system and containment, *(ii)* noncondensable gas transport, *(iii)* thermal response of plant structures, (*iv)* combustible gas generation and deflagration (H_2 from zircaloy and steel oxidation, CO from B_4C and carbonaceous concrete), *(v)* core degradation and fission product release, *(vi)* aerosol/vapor release and behavior, *(vii)* ex-vessel molten core/concrete interactions (MCCI), *(viii)* relocation of decay heat sources, *(ix)* high pressure melt ejection and direct containment heating, and (*x)* performance and impact of engineered safety systems.

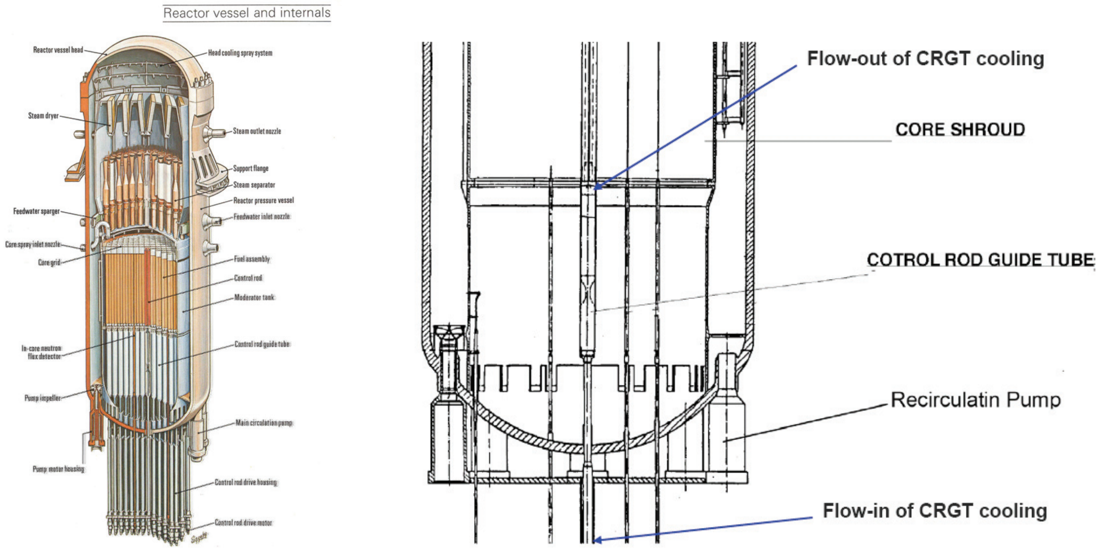

Fig. 1: The rector internals and lower plenum.

Table 1. Data of the reference BWR

Thermal power, MWt	3900
Operating pressure in vessel, bar	70
Reactor vessel outside height, m	21
Internal vessel diameter, m	6.4
Vessel wall thickness, m	0.198
Effective core height, m	3.68
Number of CRGTs	169
Nominal flow rate per CRGT, g/s	62.5
Nominal flow rate in entire CRGT, kg/s	10.5
Initial UO2 mass, kg	146000
Initial *Zr* mass, kg	52680
Initial steel mass in the core, kg	100400

Due to space constraint, only the MELCOR nodalization directly related to the CRGT cooling is presented here. Figure 2 shows the hydrodynamic nodalization (control volumes and flow paths) of the reactor vessel, the internals and CRGTs. All channels of the reactor core are grouped into one control volume named Channel and the bypass is represented by the control volume of BP. The other in-vessel control volumes are corresponding to the lower plenum, the upper plenum, the risers and separators, the steam dryer and dome, and the downcomer. The coolant of the CRGT cooling system is flowing through an annulus between the control rod and its guide tube. The annuluses for all the control rods are grouped into a single channel which is then axially divided into 7 even control volumes; see the control volumes of CV100-CV106 in Figure 2. The cooling coolant flows in CV100 from an external water source, and flows out of CV106 to the top of the lower plenum. The external water temperature is assumed 20^{o}C. The wall of the CRGTs is considered as heat structures to simulate heat transfer between the CRGTs and the lower plenum. It should be noted that heat transfer from the debris to the CRGTs is not directly calculated in the present study. But the impact of the debris is reflected by heating the fluids (pool and atmosphere) in the lower plenum, and the heated fluids transfer heat to the CRGTs. If the debris is quenched (which is the case here), such treatment is acceptable since the radiation heat transfer from the debris to the CRGTs is negligible at coolant saturated temperature.

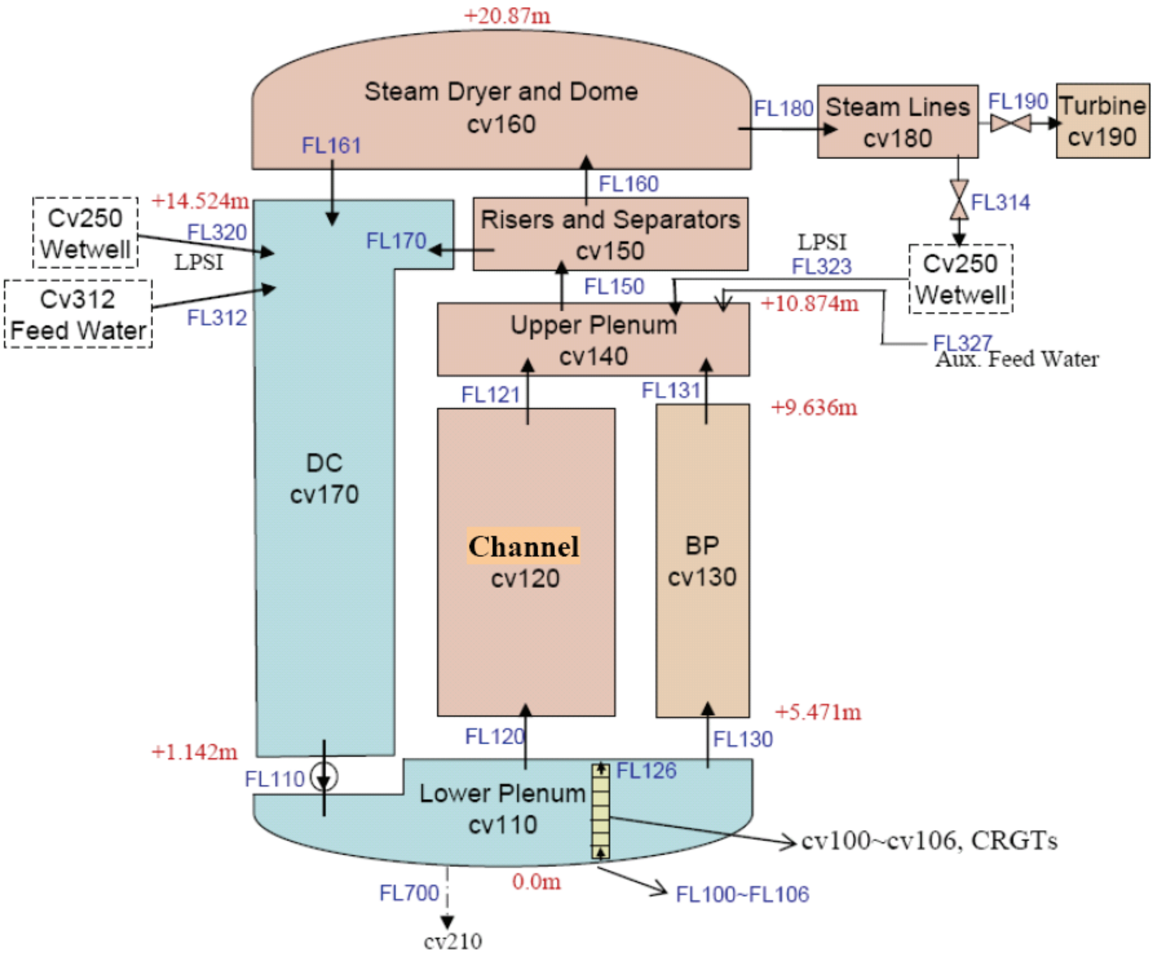

Fig. 2: Nodalization of the reactor vessel.

To examine the efficiency of the CRGT cooling as a potential SAM measure, total 6 accident scenarios related to station blackout (SBO) are calculated here, as shown in Table 2. This means all the emergency cooling systems, including the low pressure coolant injection system, auxiliary feed-water system, feed-water lines and shut-down cooling system, are unavailable. However, it is assumed that the containment spray system to the upper drywell is activated after 8 hours. The battery life time is 2 hours. The automatic depressurization system (ADS) operates 10 minutes after the water level in the downcomer is 0.5m above the core. The valves stay stuck open after ADS actuation. It is also assumed that the instrument tubes are plugged, and thus the modeling of penetration failure is disabled.

Table 2. Calculated accident scenarios

Scenario-1	Station blackout (SBO) ***without CRGT cooling***
Scenario-2	*SBO + CRGT cooling at* ***10.5kg/s from time 0***
Scenario-3	*SBO + CRGT cooling at* ***10.5kg/s from time 1 hr***
Scenario-4	*SBO + CRGT cooling at* ***10.5kg/s from time 2 hrs***
Scenario-5	*SBO + CRGT cooling at* ***42kg/s from time 2 hrs***

3. Results and discussion

Scenario-1 is a standard station blackout (blackout) accident without any emergency cooling system available. The CRGT cooling is not applied either. Fig. 3 shows the pressure evolution in both the vessel and the containment. At ~26 minutes, the core is uncovered and the ADS is activated, leading to rapid decrease in primary pressure, from 70bar to 10bar in 4 minutes, and to 3bar in 8 minutes. Meanwhile, the pressure in the containment is increasing gradually due to the blow-down of steam, and at ~4 hours it is identical to the primary pressure due to vessel failure. At 4 hours the pressure and temperature starts decreasing because of the containment spray.

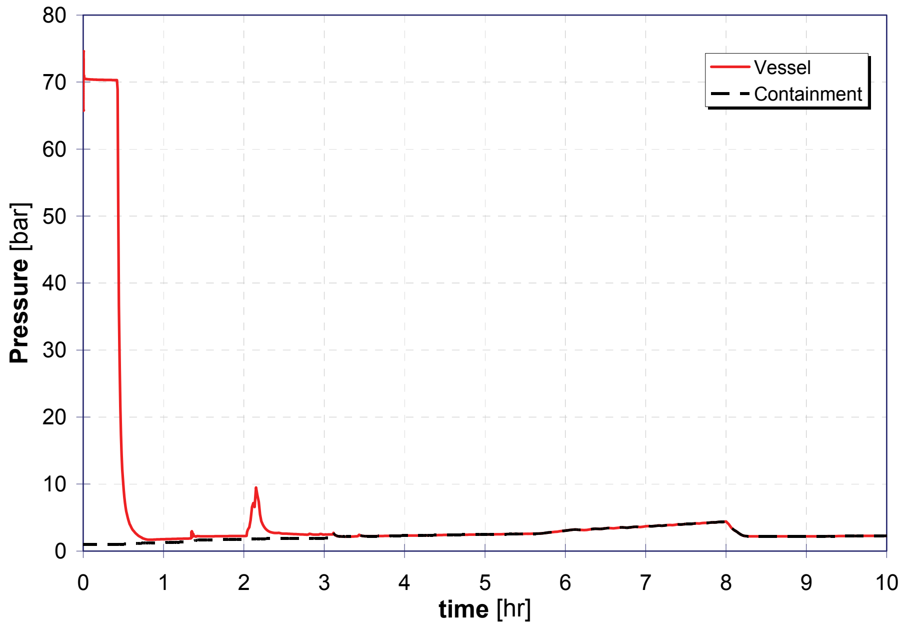

Fig. 3: Pressures in the vessel and the containment for Scenario-1.

Figure 4 shows the mass distributions of melt compositions in the lower plenum, and the melt ejection mass after vessel failure. Prior to vessel failure (at 3.11 hrs), around 15% of the total Zr inventory (52680 kg) is oxidized, and ~55% of the total Zr mass is relocated to the lower plenum. This scenario is used as reference case for the following scenarios, to be compared with the scenarios having CRGT cooling at various timing and flowrate.

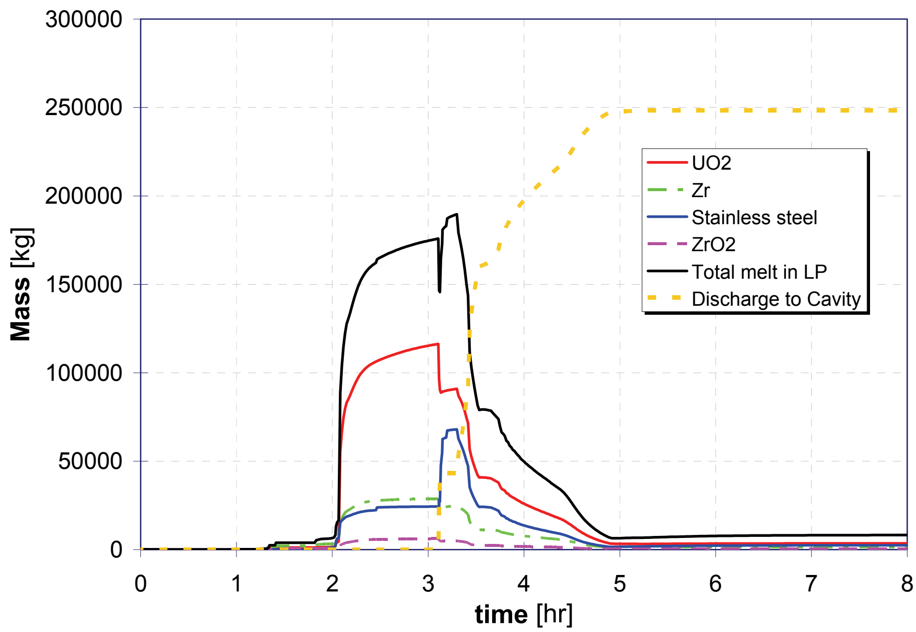

Fig. 4: Melt mass in the lower plenum and melt ejection for Scenario-1.

Scenario-2 is similar to Scenario-1, except that the CRGT cooling system is intact during the entire sequence. The flowrate of the CRGT cooling is maintained at minimal level (10.5 kg/s). Technically, this can be realized by an independent coolant supply system driven by battery or other backup power. In this case, the consequences are very different.

As illustrated in Figure 5, the core is uncovered and the ADS is activated at ~38 minutes, which is delayed compared with Scenario-1. The water in the core is boiled off at 1 hour or so, and the core stays uncovered during a period of ~40 minutes. The core is heated up and disrupted. The resulting melt (debris) is relocated to the lower plenum, as shown in Figure 6. However, the core degradation is terminated due to decreasing decay heat and steam cooling. Later on, the damaged core is quenched and re-flooded by water injection from the CRGT cooling system. This process takes more than 4 hours, but finally the core is covered again; see Figure 5. The corium relocated to the lower plenum is less than 1500 kg, and the vessel integrity is secured.

Figure 7 shows the coolant temperatures in the CRGTs and in the lower plenum. The coolant in the CRGTs is heated up by the saturated fluids (water/steam) in the lower plenum, and its temperature is elevated by more than 150 ^{o} at the very beginning. During the depressurization in the primary system, the fluid temperature in the lower plenum is also rapidly decreased. As a result, the temperature gradient between the CRGT and the lower plenum is reduced. This is why both the temperature itself and the temperature drop along the CRGTs declines after the depressurization. Interestingly, the trend of temperature variation is reversed after 6 hours, as shown in Figure 7, namely, the temperatures start rising rather than dropping. This is caused by natural circulation (see Fig. 8) between the channel (hot leg) and the bypass (cold leg), which transport the hotter fluids from the core to the lower plenum. The natural circulation does not take place until the water level in the core raises to the top (cf. Figures 5 and 8).

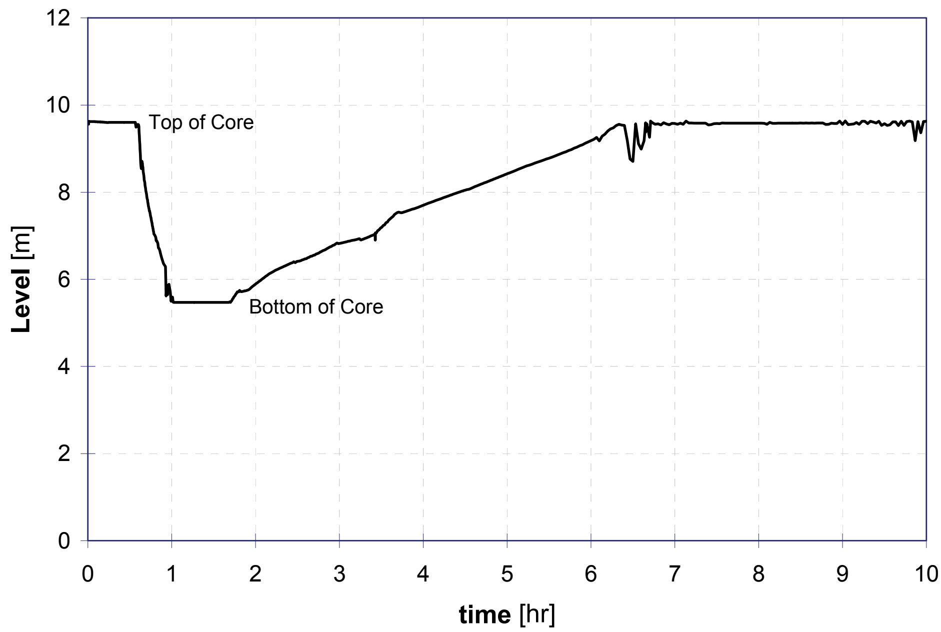

Fig. 5: Water level in the core for Scenario-2 (*CRGT flow 10.5 kg/s at time 0)*.

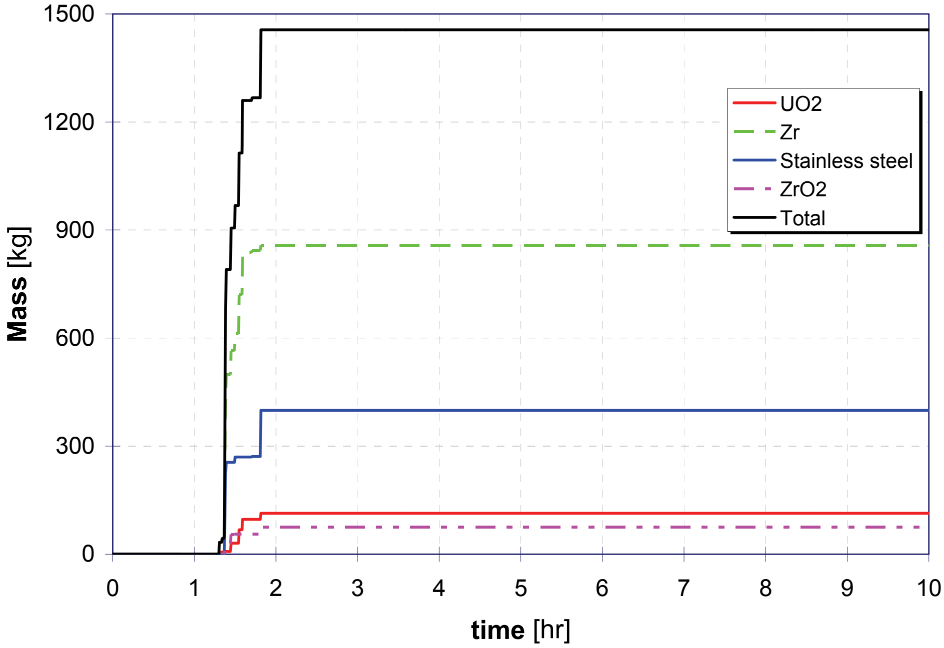

Fig. 6: Melt mass in the lower plenum for Scenario-2 (*CRGT flow 10.5 kg/s at time 0*).

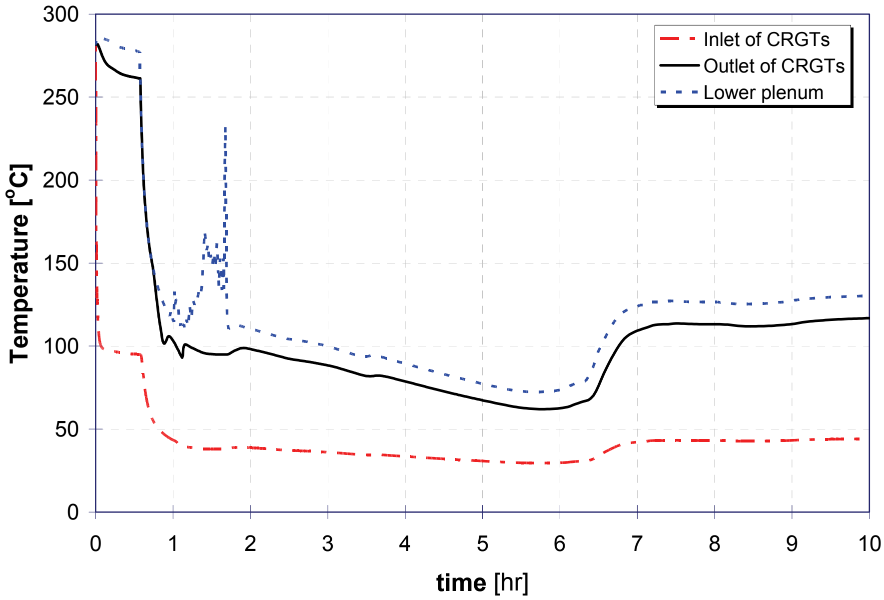

Fig. 7: Coolant temperatures in CRGTs and the lower plenum for Scenario-2 (*CRGT flow 10.5 kg/s at time 0*).

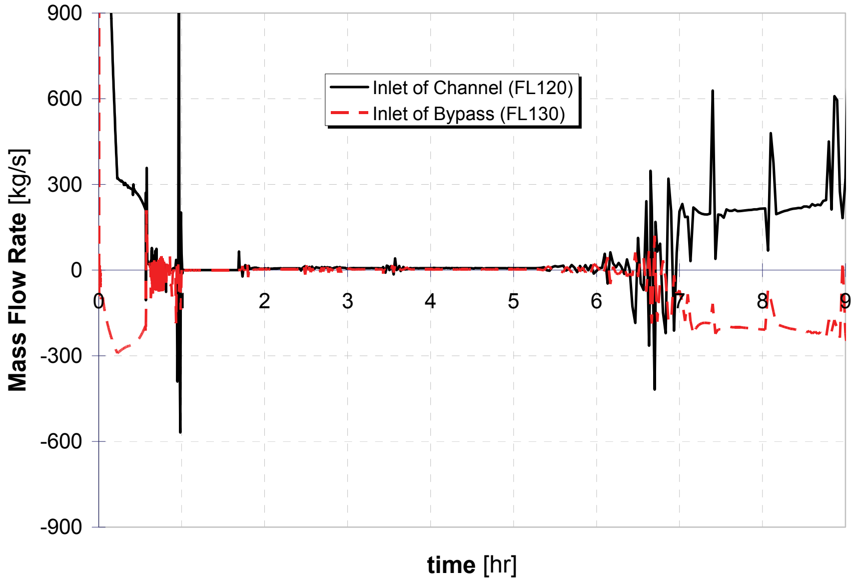

Fig. 8: Flows between the core and the lower plenum for Scenario-2 (*CRGT flow 10.5 kg/s at time 0*).

Scenario-3 is similar to Scenario-2, except that the CRGT cooling system is activated at 1 hour after initiation of the accident. The flow in the CRGT cooling system is at minimal rate (10.5 kg/s). This accident sequence can be considered as early recovery of the CRGT cooling by severe accident management. In this case, the core damage is quite pronounced, and more melt mass is relocated to the lower plenum, as shown in Fig. 9. Nevertheless, the fuel relocation is bounded to less than 5000 kg. This means that the core degradation is finally stopped by the coolant injection from the CRGT cooling system, although it takes much longer time (>9 hrs to re-flood the core) than Scenario-2. Most of the corium is stabilized in the core region.

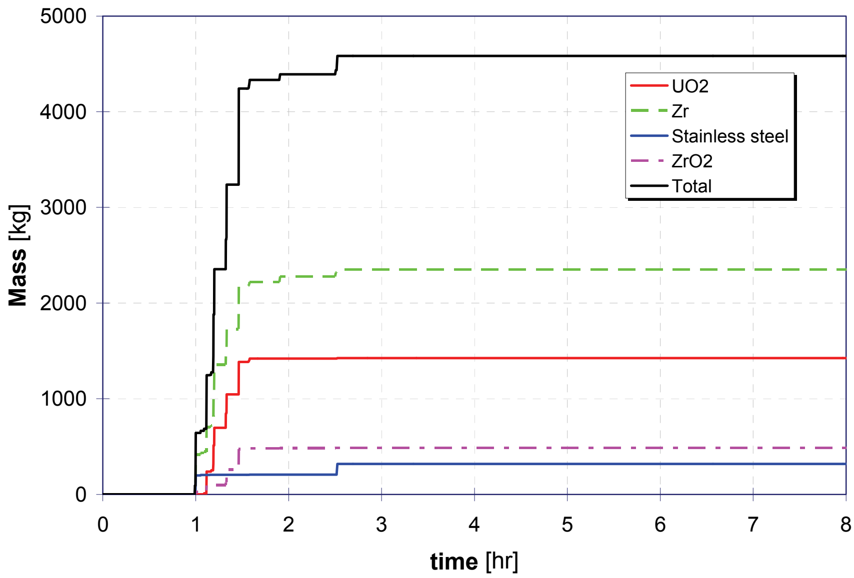

Fig. 9: Melt mass in the lower plenum for Scenario-3 (*CRGT flow 10.5 kg/s at I hr*).

If the recovery of the CRGT cooling is further delayed, say, to 2 hours after the beginning of the accident, is it still possible to arrest the melt in the vessel? The answer is No, and the melt will fail the vessel. This is exactly what Scenario-4 is intended to examine. For this scenario Figure 10 depicts melt relocation to the lower plenum and its final discharge to the cavity after the vessel failure.

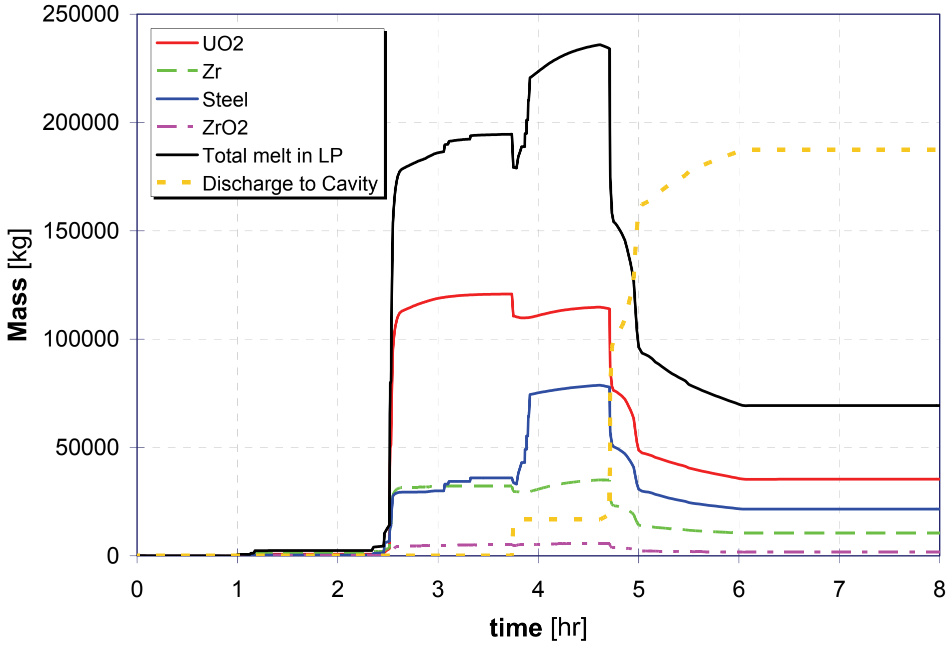

Fig. 10: Melt mass in the lower plenum for Scenario-4 (*CRGT flow 10.5 kg/s at 2 hr)*.

The consequence of Scenario-4 implies that either the coolant injection into the CRGTs is too late or the nominal flowrate is too low to prevent the core from melt-down. For such a late recovery of CRGT cooling, an increase in the water injection rate may help. As in Scenario-5, if the flowrate of the CRGT cooling is elevated to four times of its normal value, it is predicted that the corium can be contained in the lower plenum, although ~168 tons (around 67% of the total core materials) of the core melt is relocated there.

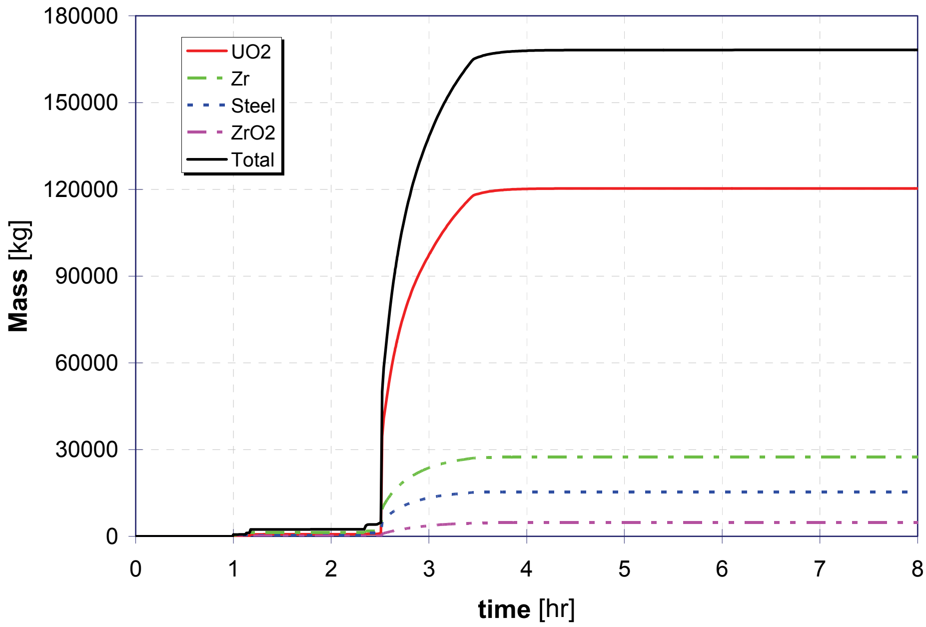

Fig. 11: Melt mass in the lower plenum for Scenario-5 (*CRGT flow 42 kg/s at 2 hr)*.

It should be noted that the MELCOR 1.8.5 does not have the capabilities to simulate melt pool convection in the lower plenum. Also, the heat transfers of the CRGTs with the lower head (conduction) and with the debris (radiation) are not modeled in the present study. These may underestimate the efficacy of the CRGT cooling. Such modeling deficiencies can be overcome by complementary/confirmatory analysis of ECM/PECM. New release (version 1.8.6 and 2.x) of the MELCOR code may also improve the fidelity of the simulations.

Another point that should be mentioned is that all the CRGTs are assumed to survive during the accident. This may not be true, especially for late recovery of the cooling system. Partial survival of the CRGTs is possible and has to be investigated.

4. Concluding remarks

This paper presents an assessment of control rod guide tubes (CRGT) cooling as a potential severe accident management (SAM) measure in BWRs, by the calculations of MELCOR 1.8.5 code. Based on the simulation results, the following points can be concluded.

- The nominal flowrate (~10kg/s) of CRGT cooling is sufficient to maintain the integrity of the vessel in a BWR of 3900 MWth, if the water injection is activated no later than 1 hour after scram.
- For late recovery of CRGT cooling (later than 1 hour after scram), a higher flowrate than the nominal is needed to contain the melt in the vessel. In case the water injection to the CRGTs is activated in 2 hours, for instance, much higher flowrate (~40kg/s) is required for in-vessel coolability and retention.

Although we are aware of the limitation of MELCOR modeling for melt pool heat transfer in the lower plenum, the first-cut analysis in the present study highlights the importance of timing and flowrate of the CRGT cooling when chosen as a SAM measure for IVR. For early actuation of the CRGT cooling, there is little corium relocated to the lower plenum for melt pool formation and the prediction by MELCOR code at system level is therefore more credible. For late recovery of the CRGT cooling, new version of MELCOR with better lower head modeling will be applied, but a mechanistic multi-dimensional analysis of heat transfer in the corium-filled lower plenum is necessary for confirmation and quantification. This is why we at KTH developed the Effective Convective Model (ECM) for simulation of corium behavior in a complex geometry like the BWR's lower plenum with a forest of CRGTs. The next step is to complete methodology formulation for information transfer from MELCOR output to the ECM method, and to perform ECM analysis for the late injection scenarios. The dual approach leverages on the strength of the two methods (MELCOR/ECM), and therefore increases the reliability of the assessment.

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[2] T.N. Dinh and R.R. Nourgaliev, Turbulence modeling for large volumetrically heated liquid pools, *Nuclear Engineering and Design*, **169**, pp.131-150, 1997.

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[8] C.T. Tran, The Effective Convectivity Model for simulation and analysis of melt pool heat transfer in a light water reactor pressure vessel lower head, Doctoral Thesis in Energy Technology, Royal Institute of Technology, Stockholm, Sweden 2009.

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Ex-Vessel Corium Management for the VVER-1000 Reactor

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1. Introduction

The VVER-1000 containment is modern PWR design. It is large, full pressure and exceptionally tight building (see fig.1). However from the point of view of severe accident it has got some inconvenient design features. The containment is built on non-hermetic lower part of reactor building and the thickness of containment basement slab is only 2.4 m. That's why there is real threat, in the course of severe accident, of corium melting through of the containment basement slab, penetration of fission products into non-hermetic lower part of the reactor building and finally into environment. On the other hand there is a free room on the containment floor for corium spreading out of the reactor cavity. Therefore several strategies for ex-vessel corium management in the reactor pit were proposed.

The major objectives of the study were to assess the effectiveness of ex-vessel corium management measures in the VVER-1000 containment and to estimate corium behaviour in the course and after melting through of containment basement slab.

2. Ex-vessel corium management strategies

Two strategies were proposed for ex-vessel corium management in the reactor pit. The first one consists in the corium spreading out of the reactor cavity on containment floor. Total corium flooded area was about 100 m^2 and it includes the bottom of the reactor pit, neighbour room, connecting tunnel and the part of containment corridor (see fig.2). To prevent the damage to tube penetrations and transport shaft cover corium spreading should be limited by portable barriers made of fire-resistant material. Corium pool cooling with water pouring from above was the second proposed strategy. To assess the effectiveness of the strategies and their combination following four scenarios derived from basic blackout severe accident were analyzed:

- TB2: basic reference case – no remedial measures were applied,
- TB3: corium was allowed to spread out of the cavity and to flood area over 100 m^2 on containment floor, no water cooling was applied,
- TB4: no corium spreading was allowed (melt pool is captured in the cavity) but immediately after the reactor pressure vessel failure low pressure ECC system was recovered and corium was cooled by water from above,
- TB5: corium was allowed to spread out of the cavity and to flood area over 100 m^2 on containment floor and corium pool was cooled by water poured from above.

.3. Assessment of corium management strategies

The corium behaviour and effectiveness of corium management strategies were studied during blackout scenario. Minimization of vertical and horizontal corium penetration depths and total mass of ablated concrete were the major criteria for the evaluation of the effectiveness of ex-vessel corium management strategies. Integral calculation of TB scenario was performed with the MELCOR version 1.8.5 within 48 hour time interval. Integral analysis of TB scenario showed that the development of the sequence towards severe accident was very fast, RPV failure was observed as early as 4 ½ hour after the start of the accident and immediately after that the melt pool was formed in the reactor

cavity. The major objective of integral MELCOR analysis was to provide initial and boundary conditions for the MEDICIS and the CORCON codes.
To increase the credibility of the results all four scenarios were calculated with both of the MEDICIS/ASTEC and the CORCON/MELCOR codes. Only phenomena in the reactor cavity were analyzed. Homogeneous corium pool was assumed. Initial mass of corium pool was about 193 ton and initial corium temperature was 2903 K. Initial corium composition is presented in following table.

Table 1: Initial corium composition

Constituent	Mass [kg]
UO_2	89723.
ZrO_2	32931.
FeO	133.
Fe	52893.
Cr	11889.
Ni	5297.
Zr	0.5
Total	192866.5

Decay heat power released in melt pool was also taken from integral MELCOR calculation. The pure siliceous concrete (without any carbonates) including steel rebars with density about 2400 kg/m^3 was assumed for calculation. The CORCON code version 3.01h (part of MELCOR 1.8.5[1]) and the MEDICIS version 1.3.2 (part of ASTEC 1.3 code[2]) were used for the analysis. Default input parameters were used for the CORCON and recommended models and input parameters were chosen for the MEDICIS. No adjustment of MEDICIS input parameters and models to match CORCON results was done. All scenarios: TB2, TB3, TB4 and TB5 were analyzed within time period of 24 hours.

4. Results of calculation

The major results for all four scenarios calculated with the MEDICIS and the CORCON are presented in following table.

Table2 : Comparison of the MEDICIS and the CORCON results at the end of scenarios

Scenario	Code	Vertical penetration depths (m)	Horizontal penetration depths (m)	Ablated concrete mass (ton)
TB2	CORCON	1.802	1.962	471.0
	MEDICIS	1.803	1.804	334.3
TB3	CORCON	0.780	1.366	398.1
	MEDICIS	0.940	0.921	313.3
TB4	CORCON	1.797	1.833	439.7
	MEDICIS	1.792	1.793	329.7
TB5	CORCON	0.805	0.729	285.7
	MEDICIS	0.823	0.804	266.3

[1] Summers, R.M., a.j.: MELCOR Computer Code Manuals (Version 1.8.5) NUREG/CR-6119, SAND93-2185, Rev.2 (May 2000)
[2] R. Fabianelli: MEDICIS User's Guide, Rev. 0, Technical Note DPAM/SEMIC 2004/34, June 2004.

Cavity profiles for the TB2 and TB5 scenarios are shown in fig.3, shape of corium flooded area (see fig. 2) was modified into circular one.

On the basis of the CORCON and MEDICIS results following findings can be summarized:

Corium spreading out of the reactor cavity is effective strategy to reduce corium penetration into concrete. Vertical penetration depth was reduced by factor 0.43 – 0.52, horizontal penetration depth was reduced by factor 0.51 – 0.68 and ablated concrete mass was reduced by factor 0.85 – 0.93.

Water cooling of corium pool in standard cavity case resulted only in negligible decrease in vertical and horizontal penetration depth and in ablated concrete mass, reduction factor for all three quantities was only 0.93 – 1.0. So that the effectiveness of water cooling in this case is not significant. High value of corium layer thickness (1.5 – 3.5 m) is probably the cause of it.

The calculations confirmed that the combination of both of the strategies: corium spreading and water cooling was the best solution of ex-vessel corium problem. The application of this corium management procedure resulted in reduction of

- vertical erosion depth by factor 0.45 – 0.46,
- horizontal erosion depth by factor 0.37 – 0.45 and
- mass of ablated concrete by factor 0.61 – 0.79 compared with the reference basic case.

Nevertheless even this best procedure is not able to terminate corium-concrete interaction definitely. Ex-vessel corium management is able only to slow down corium penetration through containment basement. The similar finding as for difficult water cooling down of melt pool in the case of siliceous concrete was derived on the basis of the OECD MCCI project results[3]. That's why the study of corium behaviour during and after containment basement melting through was performed.

5. Study of containment basement melting through

Understanding of corium behaviour in the course and after melting through of containment basement slab was the goal of the analysis. Corium behaviour in reactor building was studied with the MELCOR 1.8.5 code. Two integral scenarios were analyzed:

1) Reference TB scenario (marked REF) was calculated without containment basement failure. Homogeneous melt pool was assumed and no remedial measures were supposed.

2) Modified TB scenario (marked MOD) was analyzed taking into account melting through and break down of containment basement slab. Corium behaviour after failure of containment basement was studied during this scenario.

Following assumptions were taken for modified scenario:

- Extended model of VVER-1000 unit including the model of non-hermetic lower part of reactor building and model of the second melt pool located on reactor building basement was used.
- Containment basement slab broke down when residual thickness of the containment basement slab fell below 1 m which is quite conservative assumption.

Up to containment basement slab failure the course of both of the scenarios was of course identical. The containment basement slab failure was observed 20.5 hours after the start of scenario. Then corium transfer into the lower part of reactor building and formation of melt pool on the reactor building basement was initiated. Corium-concrete interaction proceeded in two "cavities". Containment pressure at the basement slab failure time was about 200 kPa. So that overpressure forced door leading to environment in the lower part of the reactor building which resulted in massive fission products leak into environment. Increase of fission products leak was in the range of 1 to 3 orders of magnitude (depending on fission product class) compared with reference scenario. Corium spreading in lower part of reactor building can be described as succession of following steps:

- After failure of containment basement corium penetrates into etage at level +6.6 m and floods the room located just under the reactor cavity.
- Corium melts thin cover on the floor that hides square opening with area about 2 m^2 and penetrates into etage at level 0.0 m.

[3]] M.T.FARMER et all: OECD MCCI Project, Final Report, February 28, 2006

- At this level corium melts two thin lids covering two square openings on the floor with total area of $2\ m^2$ and penetrates into etage at level –4.2 m which is final destination of melted material.
- Corium forms a pool on the reactor building basement slab and corium concrete interaction starts in the “second cavity”.

The corium movement in the reactor building can be seen in fig.1 where red arrows mark the location of corium penetrations. The figure 4 shows the development of corium mass in “reference” cavity and in both of the cavities formed during modified scenario. The figure 5 illustrates vertical corium penetration depths for all three cavities.

6. Ultimate corium management strategy

The major objective of the strategy is to prevent massive leakage of fission products into environment in the case of melting through and break down of containment basement slab. The strategy consists of following remedial measures:

- reinforcement and additional sealing of seven doors leading from lower part of the reactor building into environment,
- removal of the cover and lids on the floors of storeys +6.6m and ± 0.0m to facilitate corium transfer to the final destination i.e. reactor building basement slab, this should be done during accident before containment basement failure,
- containment depressurization before basement slab failure,
- assuring of long term heat removal from containment/reactor building and
- prevention of hydrogen detonation.

7. Conclusions

Two strategies were proposed for ex-vessel corium management in the VVER-1000 reactor cavity: corium spreading out of the cavity on containment floor and water cooling of melt pool. Extensive calculations with the CORCON and the MEDICIS codes showed that the most effective measure is the combination of both the strategies which can reduce the vertical corium penetration depth, which is the most relevant parameter, by factor 0.45. Nevertheless even this best procedure is not able to terminate corium-concrete interaction definitely. Ex-vessel corium management can only slow down corium penetration through containment basement.

The MELCOR integral calculation confirmed that the melting through and break down of containment basement slab could be expected in one or a few days from the start of the accident. This results in massive fission products release into environment. Increase of fission products leak depended on fission product class and it was in the range of 1 to 3 orders of magnitude in comparison with scenario without containment failure. That is why the proposal of ultimate strategy was presented that consists of following remedial measures: reinforcement and additional sealing of seven doors leading from lower part of the reactor building into environment, removal of cover and lids on the floors of storeys +6.6 m and ± 0.0 m to facilitate corium transfer to the final destination, containment depressurization before containment basement slab failure, assuring long term heat removal from containment/reactor building and prevention of hydrogen detonation.

Fig.1: VVER-1000 reactor building elevation view

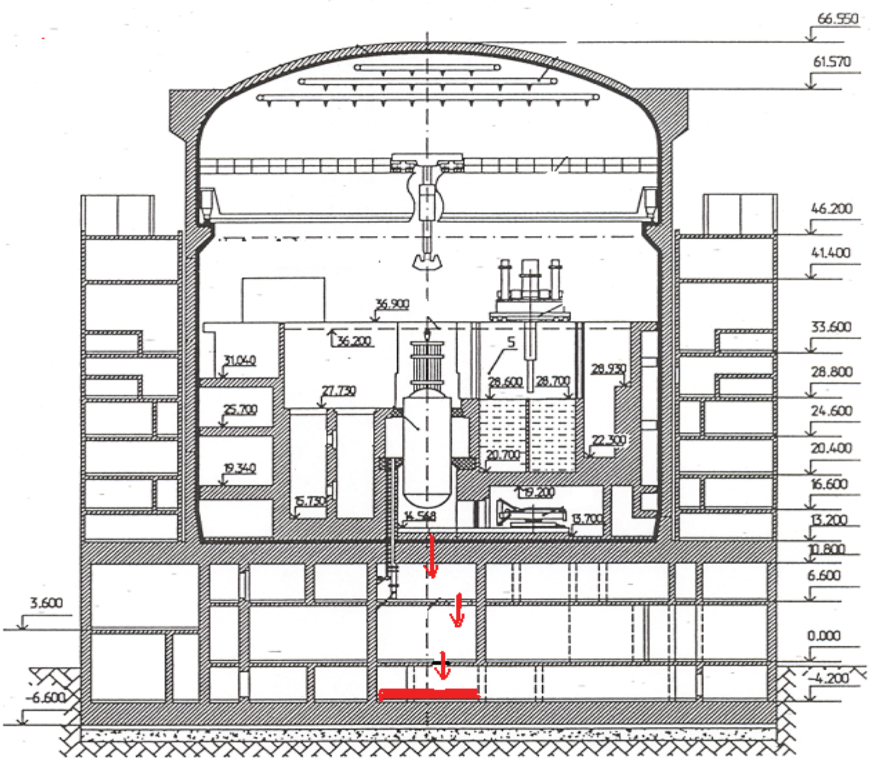


Fig.2: Shape of corium flooded area

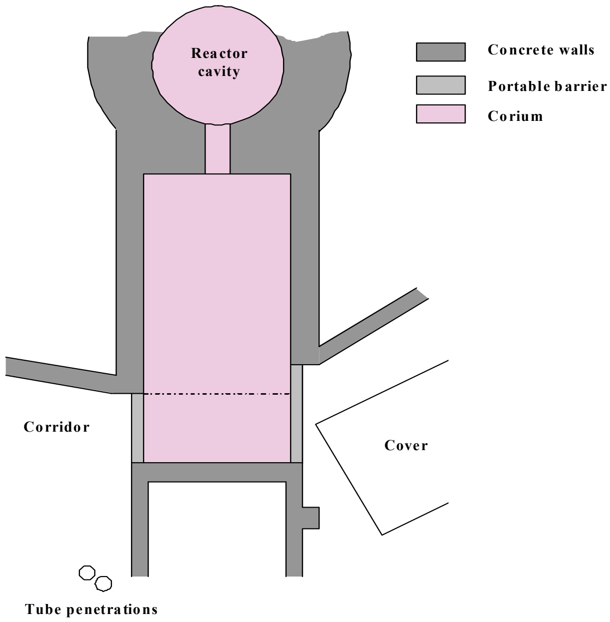


Fig. 3: Cavity shape

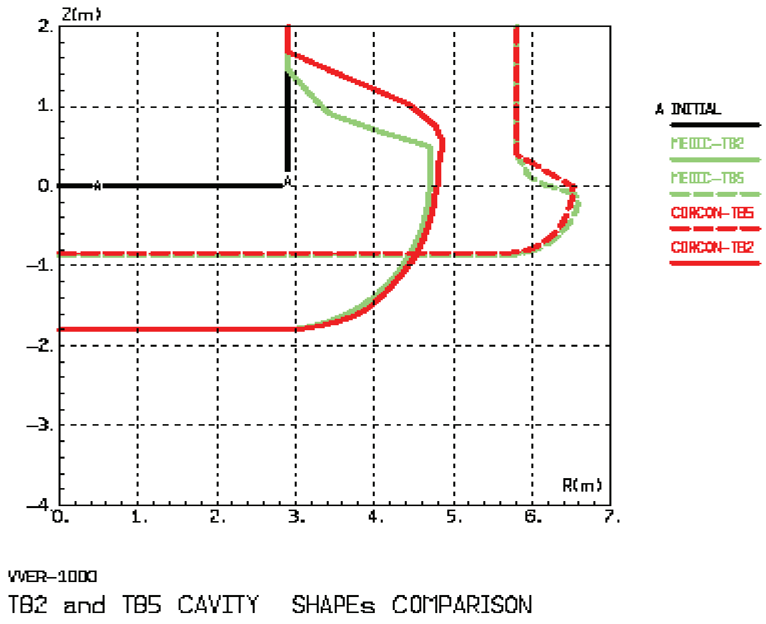


Fig. 4: Corium mass in cavities

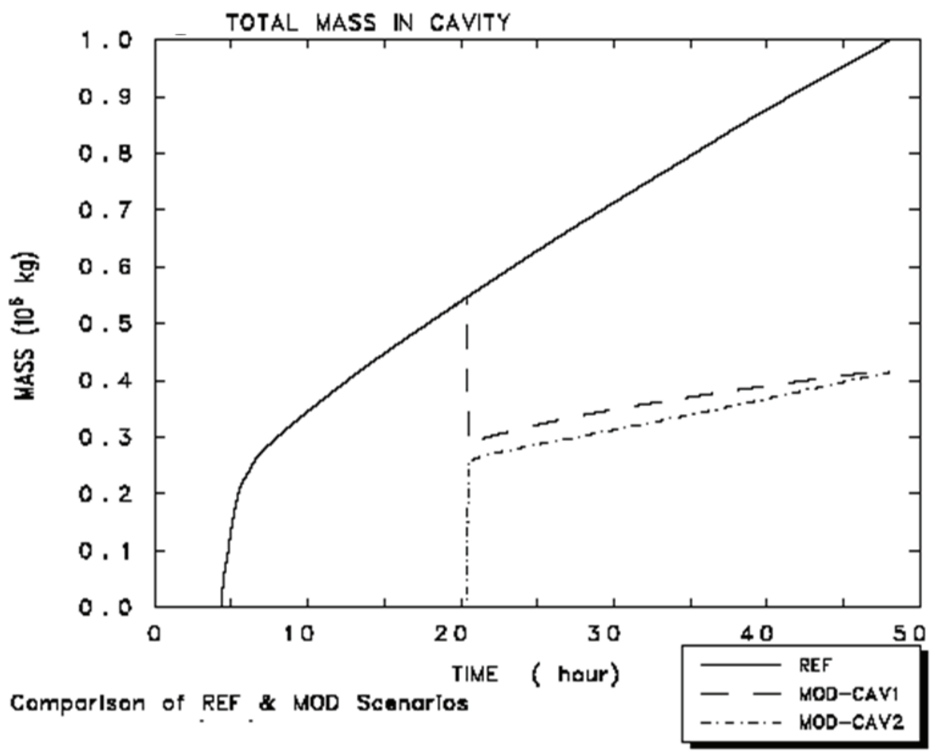
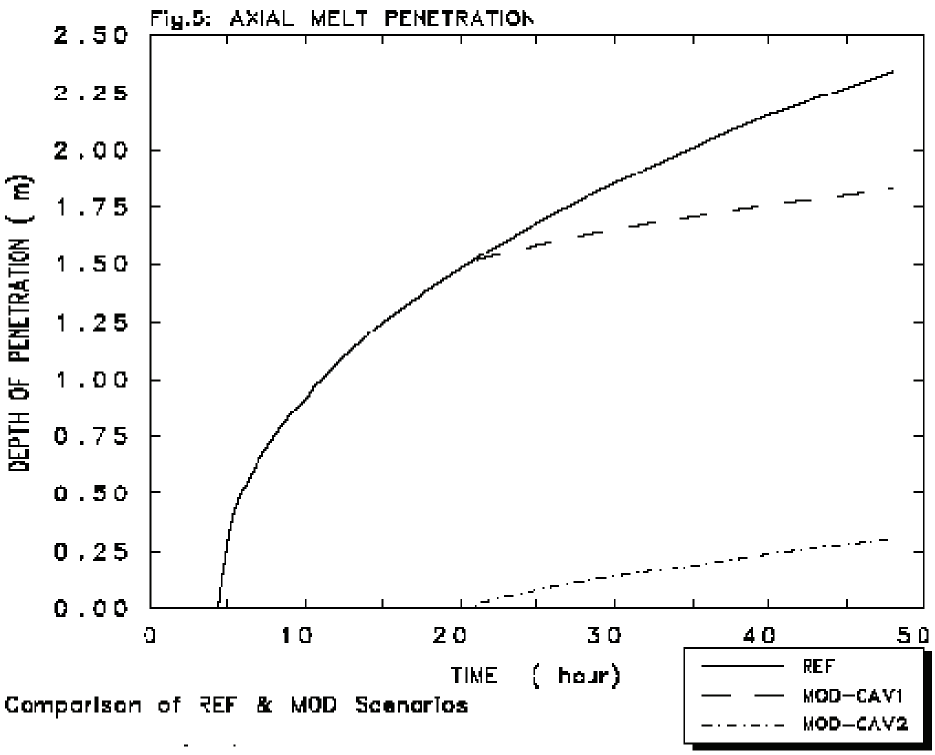


Fig. 5: Vertical penetration depths in cavities



Session 6

Criteria for the Transition to Severe Accident Management

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1. Introduction

Most Accident Management approaches distinguish between actions required and their priorities in the pre- and post-core damage phases of an accident. This is due to the need to focus efforts on protecting the core and preserving fuel integrity while this is still possible; thereafter, focus shifts to containing fission product releases. These two phases of Accident Management are often termed ‘preventive’ and ‘mitigative’, and are often included in Emergency Operating Procedures (EOP) and Severe Accident Management Guidance (SAMG) respectively.

This paper discusses the transition between these two phases. In practice, this requires that a symptom be used which identifies the onset or imminent onset of core damage. A ‘symptom’ in this context refers to a measurable plant parameter. The choice of this symptom and its setpoint(s) is important due to the change in priority of actions that will result. An indication is needed which:

- is unambiguous,
- is easily used,
- is representative in a known way of the conditions in the core being characterized, and
- provides for a timely transition to SAMG (neither too early or too late).

While various possible plant parameters have been considered in the past (this is discussed further below), most PWR and VVER type reactors (though not all) have used the core exit coolant temperature, as measured by the core exit thermocouple system, to perform this function.

Although most use this instrument, there is at first sight a large range of temperature values used, depending on the approach. By comparing a number of transition criteria used by widely applied approaches to SAMG, this paper discusses the influence of various other factors which can impact the choice of transition criterion and thereby explains why there is such an apparently wide range.

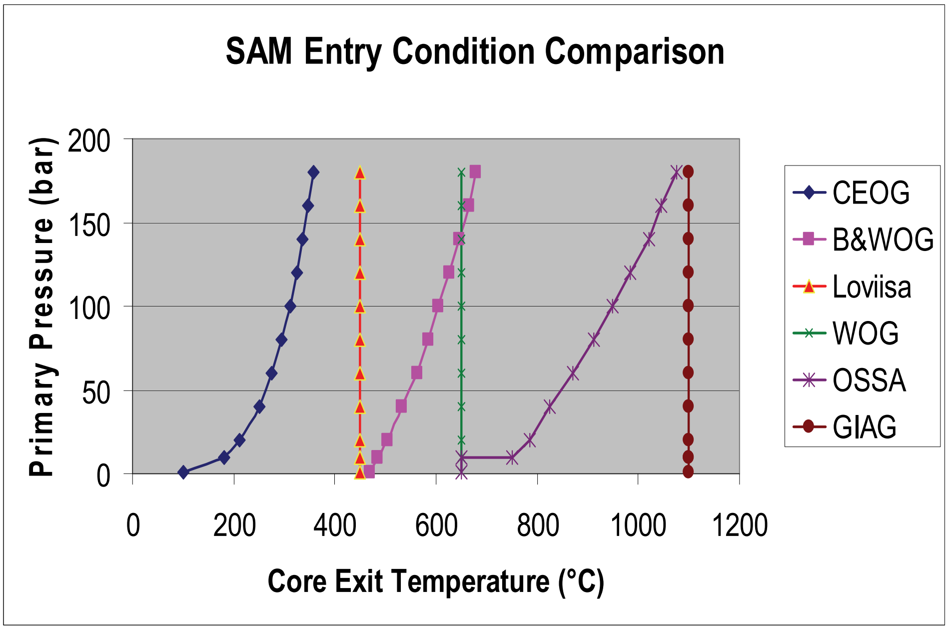
The issue of selection of a transition condition is often extensively discussed by SAM developers. It is hoped that the identification and discussion of these influencing factors will assist those developing such transition criteria.

2. Comparison of Some Transition Criteria

Figure 1 shows a comparison of EOP-SAMG transition criteria used by a number of widely applied SAMG approaches. In principle, once the core exit temperature shown is exceeded, the transition is made away from EOPs and into SAMG. It is important to note that the comparison is simplified. It does not for example list secondary setpoints or alternative or backup instrumentation which may be used. It is also not exhaustive, having chosen a set of representative approaches as a basis for the discussion. The values quoted are also optimized / changed during plant specific implementation. (Thus different plants applying the same vendor generic SAM approach may choose different setpoints). Such differences are not usually large though, and are not significant for the purposes of this discussion.

Figure 1 – A Comparison of SAMG Entry Criteria for Some Approaches

which use Core Exit Temeprature



In spite of these cautions/limitations, the comparison at first sight does indicate a surprisingly wide range of temperature values selected in the different approaches, for what is essentially always the transition from preventive to mitigative accident management space. Why is there such a divergence of entry conditions? What factors play a role in defining the entry condition? To some extent, the comparison presented is over-simplified. It does not consider these other important factors, which are discussed below.

3. Factors Influencing the Choice of Transition Criterion

Core Physical Condition

All criteria are attempting to characterize some core damage state. However, not all approaches use the same physical core state to define the 'onset of core damage' (or its approach). Thus, even if it were possible to know unambiguously the condition of the core based on core exit thermocouple readings (which it is not), we would expect to see different setpoint values depending on the approach used.

One of three physical 'states' is often considered:

- The core is uncovered. A liquid or mixture level below the top of the core indicates a serious absence of cooling situation. However, in general fuel damage and even serious cladding failures may not yet have occurred (although they will if the condition is not corrected). This condition indicates a severe loss of cooling, but there is still some remaining time before damage occurs. This condition is also, in principle, relatively easy to detect, since in a PWR/VVER configuration, superheat at the core exit cannot occur unless the core has been uncovered. Thus the symptom of positive superheat at core exit can be used if detection of this condition is required. The CEOG criterion shown in figure 1 uses this approach (the curve being therefore essentially a saturation curve, raised by a small margin to ensure that superheat has been detected).

- The core is deeply uncovered. This type of criterion accepts that core uncovery occurred some time previously, that significant superheat exists at the core exit, and that damage to the fuel is either in progress or very imminent. Typically, this type of condition represents the loss of high confidence that a subsequent reflood will succeed in quenching and re-establishing long term cooling to the overheated core (or a certain time available until such condition). Generally, a setpoint which is a fixed temperature (indicating a significant superheat under all conditions) or a superheat value will be used. The criteria used by WOG and OSSA (figure 1) are of this type.

- Severe damage is in progress – usually significant fission product release from fuel pellets has begun; releases into the primary system are therefore already occurring at significant level. The setpoint will also normally be a temperature measured at core exit, but significantly higher than one used to indicate the previous two conditions. The EDF GIAG criterion (figure 1) is of this type.

The choice of the core state, and of the symptom and setpoint used to detect it, depends very much on the other characteristics of the approach, and in particular the scope of EOP and SAMG actions, as discussed below.

Structure and Scope of EOP and SAMG

Two factors are most important:

- Whether simultaneous usage of Emergency Procedures and Severe Accident Management Guidance is permitted/intended, or whether at transition, EOP use is terminated and SAMG are used alone.

- Scope of Coverage

One of the key aspects of selecting the transition is that it is very important not to change priorities from core cooling to containment of fission products too early. All possible attempts to re-establish core cooling (within EOPs) should be prioritized as long as possible. Thus the choice of transition is a balance between maximizing time available to restore core cooling, and ensuring that fission products are retained once releases have begun.

Some approaches (notably CEOG's) allow (in fact intend that) EOPs are continued to be used after SAMG implementation has begun. In this approach, the transition itself does not terminate EOP usage. The advantage of this approach is that there is less 'pressure' to wait until the lates possible time before making the transition – the transition criterion can be chosen at a relatively 'low' value, since it does not (in principle) lead to a reduction in efforts to restore core cooling. This is seen in the CEOG criterion on figure 1, which is the lowest of all those shown. This approach has the disadvantage though, that since EOP and SAMG actions are not the same, conflicts are likely to arise if both are used in parallel. Then it is necessary with this approach to foresee and deal with potential conflicts (by assigning priorities to actions/instructions) within the EOP/SAMG package.

Most approaches do not allow the simultaneous application of EOP and SAMG. For these, there remains an incentive to choose a transition criterion which does not cause unnecessarily early transition. The temperatures used by these approaches are therefore higher, as indicated in the figure.

The other important aspect here is the scope of coverage, particularly of EOPs. For example, how how far into the severe accident regime do the EOPs go? And, at what point after entry to SAMG would one really do something different (than what was already being attempted in EOPs)? Severe accident phenomena such as hydrogen accumulation and combustion, high pressure melt ejection and creep/induced structural failures are not addressed in EOPs since they just do not occur in the regime of EOP applicability. But, the use of an 'ultimate' EOP may allow coverage of some SA phenomena which occur in the early stages of a SA, and therefore allow a later formal and complete transition. This approach is adopted by the GIAG (figure 1), and this explains to a large extent the apparently much higher transition criterion than most.

Open-ended versus specific criterion

If severe accident analysis is used to understand the timing of the events or plant conditions being discussed here, it is found (and understood) that once core heatup begins, the metal-water oxidation reactions which occur in the core add significant reaction heat, and contribute to an acceleration of heatup. The time windows available between different core states become shorter as the heatup progresses due to the autocatalytic effect of the oxidation reaction.

Some SAMG approaches will allow the entry to be made based on a criterion after which certain actions must be attempted as a 'last attempt' to restore core cooling. This tends to results in a less precise (in the sense of timing) transition, but ensures that all possible means to recover core cooling have been attempted before transition occurs. There is also the possibility of 'getting stuck' trying to perform the last actions and delaying the entry significantly. (This has been observed in exercises). Some plants have modified this type of entry condition to have two setpoints: one above which the last actions must be attempted, and if unsuccessful, transition made, and one above which transition must be made regardless of other conditions or attempts at recovery. There is a parallel here with systems which allow simultaneous use of EOP and SAMG.

Procedurizing Entry

Most SAMG approaches include specific steps within the appropriate emergency procedures to instruct transition to SAMG on reaching the chosen criterion. Since EOPs are opersting procedures, it is therefore the responsibility of the operations staff to monitor and detect the transition condition and to make the transition once this is reached. Most approaches procedurise the transition in this form, which ensures timely transition once they do. But some require authorization of Emergency Management (or other management staff). In practice, this is (at the time in question) usually a member of the operations team anyway. But it can induce delay.

'Margins'

Margin is often applied to the selected setpoint value. This may be done for various reasons, including:

- the wish to use a simple criterion, perhaps a single value of temperature, to cover a range of potential conditions,
- the wish to allow for inaccuracy in the instrumentation,

- the wish to account for unforeseen possible delay mechanisms in the response of the instrumentation.

The first of these is a decision involving a trade-off between the ease of use of a simple criterion, against the better characterization of a given fuel condition which can be obtained by using a more complex algorithm. For example, the underlying concern is the fuel integrity – probably best characterized by the cladding temperature, which is not a measurable parameter, and is therefore an unsuitable symptom. Since the CET represent the most direct measurement from which cladding conditions can be inferred, the setpoint selected must take account of the relationship between the core exit coolant channel temperature and the cladding temperature. This relationship is well understood, but is not linear; in particular, it is a strong function of system pressure. Clearly, a given superheat will be a strong function of pressure when expressed as a temperature alone, but in addition, higher steam density at high pressure increases rod-coolant heat transfer coefficients and therefore reduces the clad-coolant temperature difference. The net effect is that the expected core exit temperature for a given cladding temperature varies significantly with pressure. Some criteria (CE, B&W and OSSA, figure 1) incorporate this dependence into the transition criterion, resulting in an entry temperature which is a function of pressure. Others, (e.g. WOG) do not, and choose a single value of temperature which envelopes all pressure values. The advantage of the former approach is that the core (clad) conditions at the time of reaching the criterion will be (more or less) independent of the coolant conditions (and hence the event), thus representing a more consistent core condition under all circumstance of transition. The advantage of the latter approach is its simplicity and ease of use, even though it may (under certain circumstances) contain considerable margin to the conditions of concern. Thus the time in EOP can be increased by not including unnecessary margin in the criterion, but this usually comes at the price of increased complexity. This choice is therefore a balance, and different approaches have selected different solutions.

Instrumentation Issues

There are numerous other issues associated with the instrumentation used to detect the transition symptom. For example, survivability and qualification issues may impose limitations, though this is more likely to be the case in new designs which have addressed severe accidents and their management in the design.

Another important issue, which is not discussed in detail here but is the subject of a separate paper, alternative or backup parameters have been considered, and in some cases used. The most commonly considered parameters are vessel water level, containment radiation and containment hydrogen. Assessments of the suitability of these alternates must consider such aspects as survivability, accuracy, time response etc. Except in a few cases, such assessments have led to the CET being the preferred choice. It is noted though that the design of the thermocouple system in most PWRs is such that it is unavailable during certain shutdown plant states – which ‘forces’ the choice of an alternative for these situations.

Finally, the instrument response in transient situations is important. As noted above, allowing for delays or inaccuracies probably adds margin and therefore reduces the time available in EOP. However, inadequately allowing for any delays may result in a late transition. This is a topic which has been under investigation by a working group under the GAMA program, which is briefly discussed in the following section.

4. The WGAMA Task Group on CET

Up to now, there has been little discussion here of transient or timing issues associated with the transition. It was noted above that once core uncovery occurs and heatup begins, the process is self catalyzing – and therefore accelerates. As heatup proceeds, the time windows to make decisions and take actions become shorter. It is therefore important that the transition point, when it occurs, is not ‘discussed’ (something observed in exercises) – that action is taken without delay. In addition, any other transient effects, for example any effect which leads to a delayed response of instrumentation, can become important. It has been seen that developers often add ‘margin’ to the transition criterion, to account for example for the instrumentation accuracy, to ensure coverage of a range of possible accident situations, and sometimes to allow the choice of a simple and easy to use transition criterion.

Some recent experimental data, in particular the results of OECD/NEA ROSA project Test 6.1, have shown unexpected delays in the response of the core exit thermocouple readings during heatup situations. Preliminary analysis of the test results seemed to indicate that the observed delay between rod surface temperature and CETs readings evolution could have had a significant impact on test results. This concern drove WGAMA to propose to CSNI an activity which has been finally carried out by the Task Group (WGAMA Task Group on CET). The group was tasked with reviewing the experimental results in question, and also assembling and reviewing results from other relevant experimental programs, with a particular emphasis on the potential for delays in response of the CET in Accident Management applications. In addition, the group gathered information from member states on the use and application of CET in Accident Management, and in particular the technical bases for selecting setpoints based on the CET. The objective was to conclude as to the importance of the apparently unexpected response of the thermocouples.

The issue is complex since numerous physical effects can play a role in the observed response, such as radial temperature profiles (both in the core and above the core) or cooling effect of the unheated structures; backflow from the hot legs during core heat-up (steam condensation in SG tubes or pressurizer water-pool drain-down) is another important phenomenon affecting CET delays.

In addition to ROSA test 6.1, experiments performed in various other programs were reviewed in detail, including LOFT, PKL, PSB-VVER, and also twelve other ROSA-LSTF experiments. The delays found in most of revised cases are consistent with the conclusions previously drawn on this topic, and seem not to affect to the reliability of the AM actions which are initiated based on CET readings. However there were some specific experimental cases, as Top Head and Lower Head RV breaks, which presented doubts about the effectiveness of AM actions based on CET readings.

In general, member countries reported a generalized use of CET in EOP (preventive AM), in the transition from EOP to SAMG, in SAMG (mitigative AM) and, in some cases, in Emergency Planning. In relation to the Technical Bases for CET use on AM, not all the bases have been clearly identified, but criteria based on subcooling, saturation, onset of superheating and/or significant superheating, were reported by most of the surveyed organizations. Another important topic investigated in the survey was the relationship between CET Readings and maximum cladding temperature. It has been noted that a significant fraction of the responses indicated that specific analysis had been performed to address this issue, but some of them felt the model validation was not fully adequate. Consistently to that, some of the responses expressed that either “delayed response” or “accuracy” was a concern.

Final conclusions of the working group are not yet available, but a detailed report is in preparation and will be submitted to GAMA this year. This report will provide much more detail on both the work performed in reviewing uses of CET in accident management, reviewing of applicable experimental results, and will provide conclusions and recommendations regarding the transient effects of measuring the core exit temperature using the CET.

5. Conclusions

This paper has provided a review of the factors influencing the choice of entry condition for SAMG. By considering each of these factors, the apparent wide variation in choice of temperature setpoint between AM approaches can be explained. This has also helped to identify important concerns which must be addressed during the development and implementation of a severe accident management program.

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Use of the Software Module SPRINT in the Netherlands for Prediction of the Source Term

OECD/NEA Workshop on Implementation of Severe Accident Management Measures

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Extended Abstract

Development of tools for use in rapid accident diagnosis and subsequent radiological source term forecasting at Nuclear Power Plants (NPPs) is increasingly desired by off-site emergency planning and response personnel. The source term (in this context) is the quantity and characteristics of the release of radioactivity to the environment through available release paths.

The SPRINT (**S**ystem for the **PR**obabilistic **I**nference of **N**uclear power plant **T**ransients) software module estimates the source term in case of an accident at a nuclear power plant. The fast running software module has been developed within the Euratom Framework FP4, FP5 and FP6.

SPRINT uses information on NPP plant status that is deduced from key plant observations using a probabilistic model, known as Bayesian Belief Networks (BNN) analysis. The software module is based on manual input of plant observations and judgments by an operator or analyst, from which the possible final plant states is deduced by probabilistic inference in the BBN. The probabilistic element of the method can also overcome unknown or missing information by resorting to prior probabilities determined by plant experts who set up the model (e.g. by the use of PSA level 1 and 2 information). One of the major benefits of using a probabilistic model (rather than a deterministic model) is that it alerts the user to the existence of alternative possible plant states based on the known and unknown plant observations. Thus the outcome is typically a number of possible alternative plant states each with an associated environmental source term and probability ranking.

A specific model for the Borssele NPP in the Netherlands has been developed. The Borssele NPP, in operation since 1973, is a Siemens designed 2-loop PWR. The plant is using EOPs and SAMGs based on Westinghouse standards.

This paper addresses the establishment of the SPRINT model for the Borssele NPP and the use of SPRINT within the Emergency Response Organisation (ERO) and by the authorities in the Netherlands. For preparation of the SPRINT model of the plant amongst other PSA level 1 and level 2 information has been used. Severe Accident Management measures are also implemented in the model. The most important features and some limitations for use of the SPRINT software module are described.

SPRINT was used during SAMG exercises (including the use of Accident Management measures), EOP exercises and a full scale national emergency exercise. The source term prediction and training of the different responsibilities within the Emergency Response Organisation are part of these exercises.

The use of SPRINT has shown that the group responsible for prediction of the source term, is alerted at an early stage of the accident on the existence of a final source term with a low probability, but severe consequences. For this purpose SPRINT is well suited and better than the earlier used method, a decision tree on paper. Some findings for improvement of SPRINT and training of staff are discussed.

Furthermore it pointed out that SPRINT predictions are useful in the communication of the possible source term from the ERO at the plant to the authorities and the planning for the authorities of possible emergency measures (such as evacuation), especially at the early stage of the accident.

1. Introduction

Development of tools for use in rapid accident diagnosis and subsequent radiological source term forecasting at Nuclear Power Plants (NPPs) is increasingly desired by off-site emergency planning and response personnel. Availability of such analytical tools would enhance the efficiency in preparing accident response options and result in a more appropriate off-site response. The source term (in this context) is the quantity and characteristics of the release of radioactivity to the environment through available release paths.

Recent developments in Decision Support Systems for emergency response, within the European Union and elsewhere, have been predominantly concerned with improvements of models for dispersion, radiological consequence assessment or countermeasures planning. An area which is not so well developed is the assessment of the plant status and the associated source term. Of immediate concern to the emergency response team and the incident controller is a timely estimate of the likely release of radioactivity to the environment.

In plant analysis, the source term is calculated on a deterministic basis by assuming the parameters that define the input conditions (based on initiating events and assumed systems response) and using the calculated event progression based on the accident phenomenology in the analytical models. When using the deterministic approach to source term prediction for emergency response, the same constraints apply; i.e. a single, deterministic scenario to describe the plant status must be made before any predictive source term calculation can start. This is appropriate where the plant status can be positively and uniquely identified, based on instrument readings, as is typically the case for design basis faults. However, and particularly in Beyond Design Basis conditions, this is not always the case: instrumentation may be operating beyond its designated range (and so provide unreliable readings) or it may fail altogether. In order to make source term predictions in this situation, decisions about the reliability of conflicting instrument readings and suitable substitutions for any missing information must be made in “real time” before any calculation can be carried out.

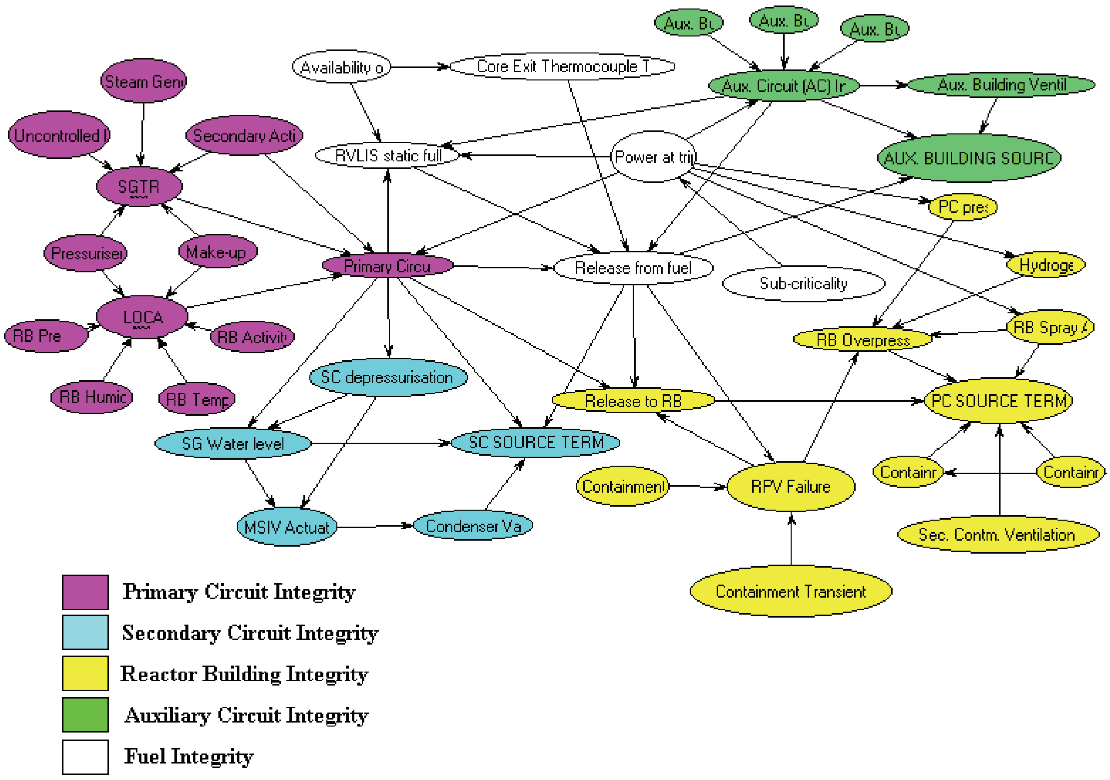
This paper describes the development of the SPRINT (**S**ystem for the **PR**obabilistic **I**nference of **N**uclear power plant **T**ransients) software module and the use for the Borssele NPP in the Netherlands. The module estimates the source term in case of an accident at a nuclear power plant. The module is based on information on plant status deduced from plant key observations using a probabilistic based model. The module has been developed within the Euratom Framework FP4 [1], FP5 [2] and FP6 [3]. The FP5 partners which were involved in the SPRINT development were AMEC (UK), Enconet

Consulting (A), ERI Consulting and Co (CH), GRS mbH (DE), KTH (S), NRG (NL), Silesian University of Technology (PL), VEIKI (HU) and VUJE (SK).

2. SPRINT Approach

SPRINT uses information on NPP plant status that is deduced from key plant observations using a probabilistic model (Bayesian Belief Networks (BNN) analysis) [4]. Some of the nodes of the model represent plant parameters / conditions that are observable, at least in principle, while others represent conditions that are intrinsically unobservable. The SPRINT approach uses the observations from the NPP to interrogate a database of pre-calculated source terms, typically compiled from existing plant specific analyses performed as part of a Level 2 PSA study. The basis for this interrogation is the belief network of plant behaviour. The end points of this logic model are then mapped onto the database of pre-calculated source terms. An example of a belief network is shown in figure 1 [5]. In a belief network logical relationships between various key plant parameters are represented by a graphical network that consists of nodes and directed links between the nodes.

Figure 1. An example of a belief network



For the belief network analyses the Netica software package from Norsys is used [6]. The software module is based on guided manual input of approximately 30 plant observations and judgments by an operator or analyst, from which a corresponding plant state is deduced by inference in the BBN. The probabilistic inference is used to determine the implications of the guided manual input on the possible

final plant states. The outcome is typically a number of possible alternative plant states each with an associated environmental source term and probability ranking.
An example of an input screen is shown in fig. 2. The user has only to answer the questions. For each question the user can select one of the choices. It is also possible to change the answers at any time or to omit some questions. After answering the questions the source term results are shown within a few seconds. An example of the results is shown in the figures 3 and 4. This information provides an indication of the most likely source term (see figure 3, example of a SGTR) as well as the spread of the possible source terms (for each source term the release rate and total released activity for each nuclide, see fig. 4 for one of the source terms). Note that the user has further nothing to do with the belief network itself. The user does even not see the network.

Figure 2. An example of an input screen

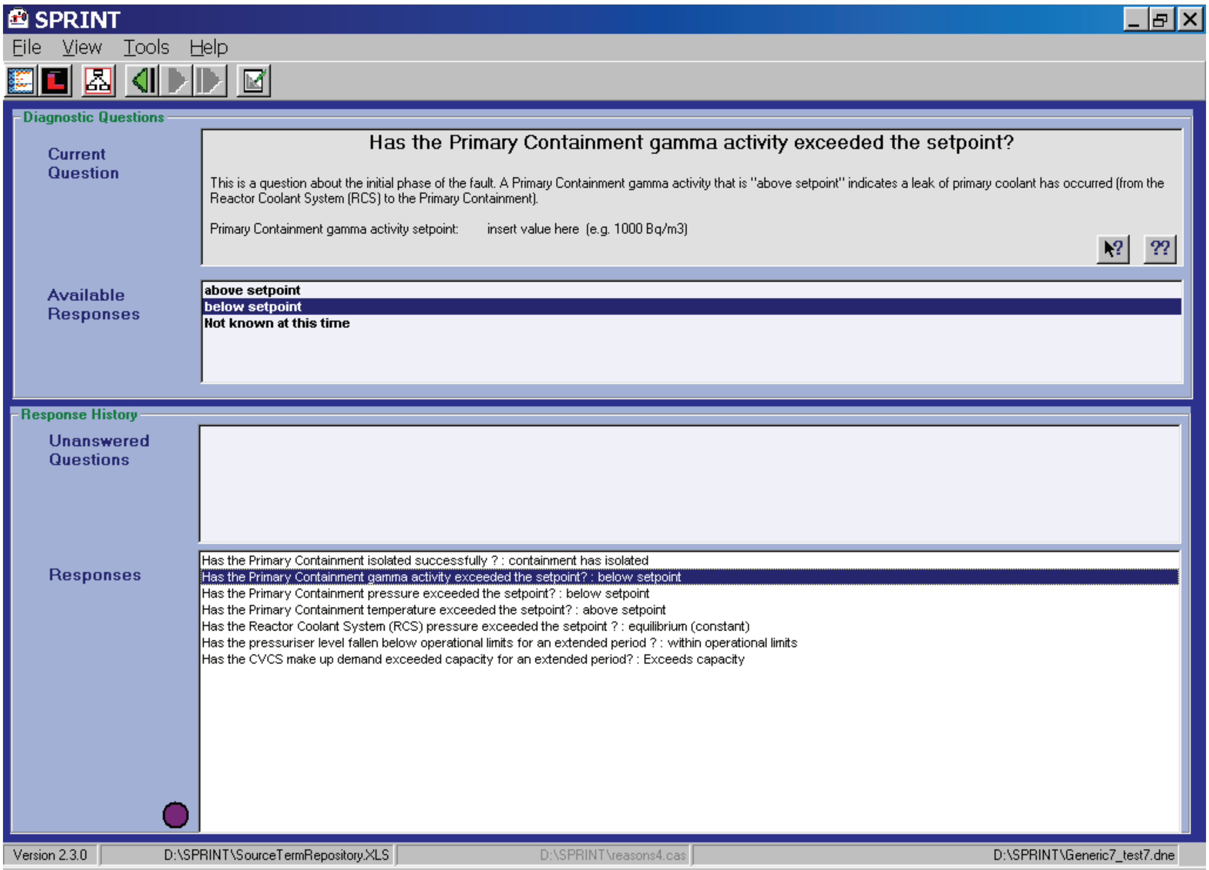


Figure 3. An example of the source term probabilities

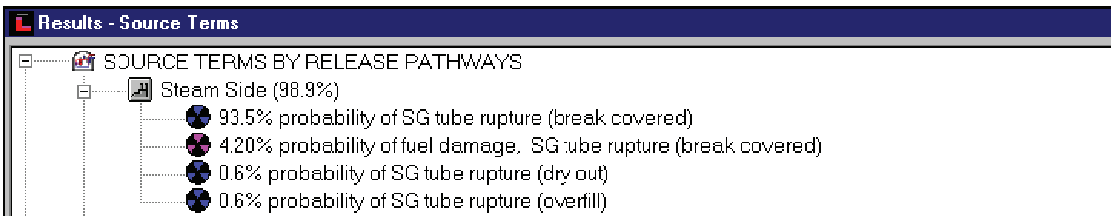
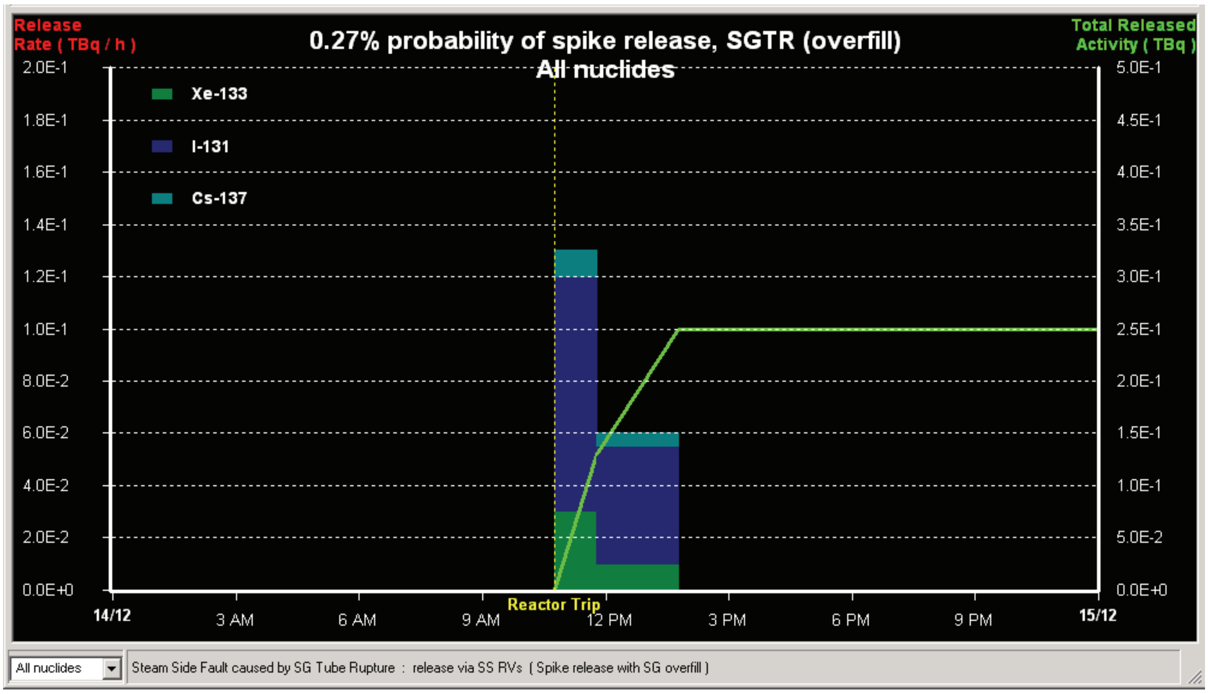


Figure 4. An example of one of the source terms



The probabilistic element of the method can also overcome unknown or missing information by resorting to prior probabilities determined by plant experts who set up the model (e.g. by the use of PSA level 1 and 2 information). This overcomes the need for making data substitutions in a "real time" stressful environment.

One of the major benefits of using a probabilistic model (rather than a deterministic model) is that it alerts the user to the existence of alternative possible plant states based on the known and unknown plant observations. Unknown information can be of 3 basic types:

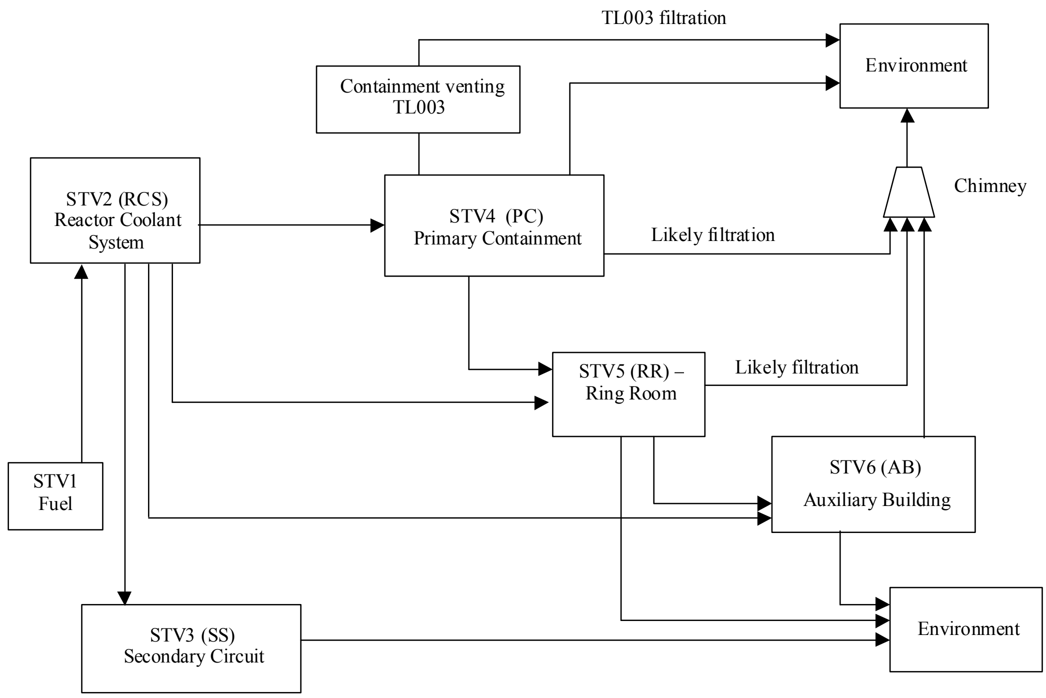
1. information that in principle is available but is missing for some reason e.g. due to instrument failure or a time lag in communicating information,
2. information that is not known because there is no practical way to observe it directly e.g. RPV status,
3. information that is inherently unknowable as it relates to a stochastic event e.g. hydrogen combustion.

3. Establishment of the SPRINT Model for the Borssele NPP

The Borssele NPP in the Netherlands, in operation since 1973, is a Siemens designed 2-loop PWR. The plant is using EOPs and SAMGs based on Westinghouse standards. Once conditions indicating a severe accident is in progress have been detected, use of the EOPs is terminated, and a transition to the SAMGs is made.

The key plant parameters for inclusion in the probabilistic model were identified through a systematic consideration of fission product transport and retention phenomena in the identified plant compartments. The physical volumes and potential fission product transport pathways of the Borssele NPP were documented and are depicted in figure 5.

Figure 5. The physical volumes and potential fission product transport pathways of the Borssele NPP



By using this scheme a description was made of the logical relationships between various key plant parameters. These relationships are represented by a graphical network that consists of nodes and directed links between the nodes. In this case, the directed links usually represent a causal connection between two nodes. Besides the network structure itself, the conditional probability tables supporting the network contain the data which determine the strength of influence of a parent node on a daughter node.

Plant systems, which are available for the mitigation of accidents, and the most important accident management strategies were also considered in the SPRINT network. The basic knowledge for the plant specific network comes from the Borssele plant system descriptions and system response, the PSA level 1 and level 2, and several thermal hydraulic- and severe accident analyses.

The plant specific SPRINT model of the Borssele NPP was establishment by NRG with consultation with the NPP staff.
The conditional probability tables represent level 1 issues (i.e. issues between the initiating event and core damage) and level 2 issues (i.e. from core damage until environmental releases). The values of the conditional probability tables were determined based on the following information/sources:

- plant system descriptions
- PSA level 1 information
- PSA level 2 information
- Thermal hydraulic analyses and severe accident analyses for the Borssele NPP
- Expert judgement.

3.1 Verification of the SPRINT Model

The Verification of the SPRINT model has been performed according to the following procedure:
- Basic checks
- Comparison with best estimate event progression analysis
- Comparison with PSA Level 2 results
- Representation of high consequence/low probability events.

Each part of the verification and of the results is described below.

Basic checks
During the development of the SPRINT network, continuously simple checks were performed, mostly to determine the correct dependencies between a few nodes of the network. One of the basic checks for severe accidents is the prediction of the initiating event. The basic checks for the Borssele NPP showed reasonable SPRINT results.

Comparison with best estimate event progression analysis
It was checked whether the SPRINT network assigns a high probability to those event sequences which are considered as best estimate event progression analysis (results of accident analyses with the codes MAAP or MELCOR).
Typically this kind of comparison mainly consist of the input list of SPRINT and of a list of the interesting results, e.g. for initiating events or core damage states or containment failure modes. A comparison is made between the probabilities determined by the SPRINT network and the pertinent information from the accident analysis. The results which have been determined with best estimate accident analysis should have the highest probabilities in the SPRINT network results.
For the Borssele NPP it showed that the probabilities determined by SPRINT are in reasonable agreement with the pertinent information from the accident analysis. The SPRINT results showed high probabilities for the results of the accident analyses.

Comparison with PSA Level 2 results
The result of a PSA level 2 consists of a set of different accident evolutions, intermediate characteristic events (e. g. containment failure) and source terms together with their relative probabilities. This result can be given separately for different initiating events. For the Borssele NPP it showed that the probabilities calculated by SPRINT are within the error bands of the PSA results.

Representation of high consequence/low probability events
For accident sequences with high consequences, but generally have a low probability and/or complicated accident progression it was shown that the network produces probabilities for these sequences which are compatible with PSA level 2 results. Examples of large releases for the Borssele NPP are severe accident sequences (occurence of core melt) with an early containment rupture, an early containment leak, an early release from a dry SGTR without SG isolation or an early release from a dry SGTR with SG isolation.

4. Use of SPRINT During Exercises

The use of SPRINT during exercises and the experiences are described below.

4.1 Organization of Emergency Response

An overview of the emergency-organisation is depicted in figure 6. The follwing responsibilities within the emergency-organisation are recognised:

- BOC “Bedrijfs Ondersteunings Coordinator”. This person is responsible for actions in the plant, e.g. restore a pump
- BT “Beleids Team”. This includes the Site Emergency Director, the MB, the MOD and the MSB
- MB “Manager Bedrijfsvoering”. This is the person to which the TAG reports
- MSB “Manager Stralingsbescherming”. This is the person responsible for radiation protection and off-site simulations
- S Control room shift personnel
- SED Site Emergency Director. This person is the head of the ERO and takes the decisions
- SM Control room shift manager
- TAG “Technische Analyse Groep”. This is the group responsible for EOP/SAMG evaluations in the shelter, amongst others for prediction of source term

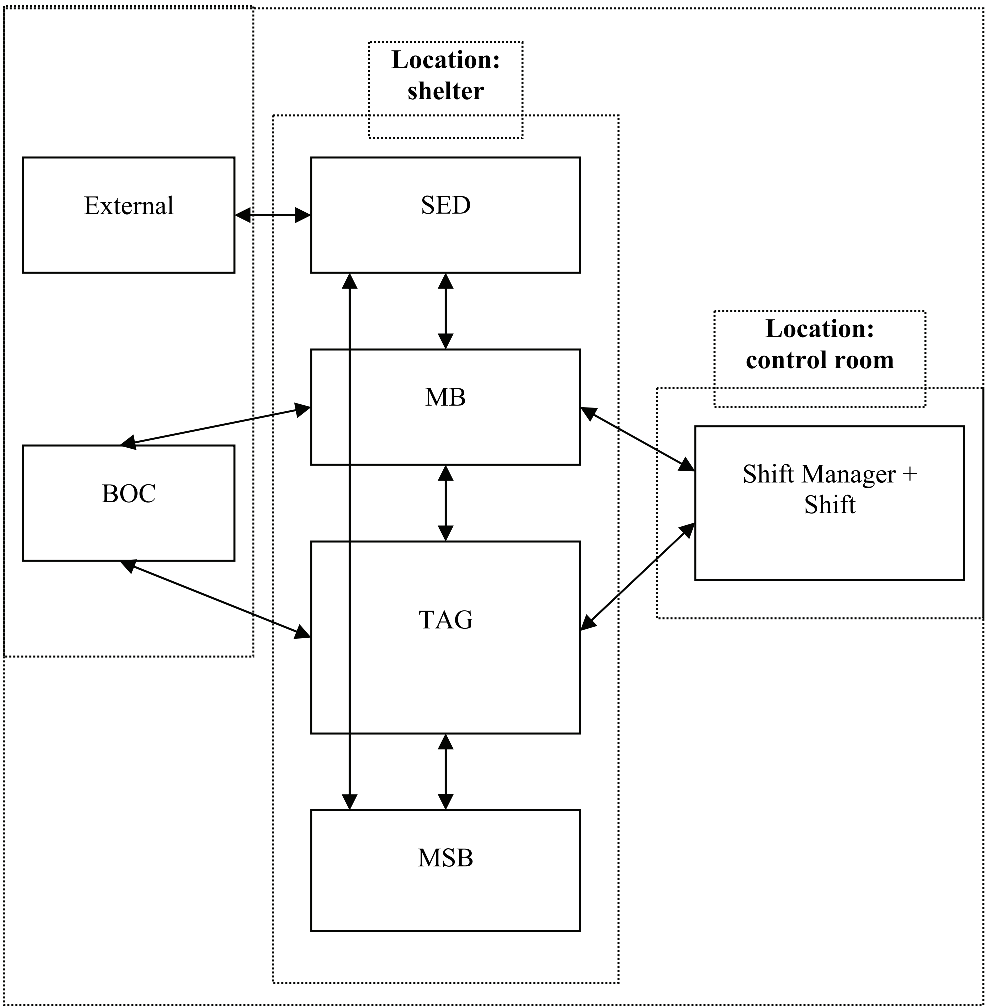
The determination of the source term is the responsibility of the TAG group at the plant Technical Support Centre. Therefore SPRINT is used by the TAG group. For the training of the SPRINT users a separate User Manual was written [7]. This manual contains a/o the user instructions, all SPRINT questions, and an explanation of the questions.

The source term predictions determined by the TAG group are communicated via the Site Emergency Director to the authorities (KFD) and are also communicated to the Source Term Manager (MSB) at the plant Technical Support Centre.

Before using SPRINT the source term was determined at the Borssele NPP based on the pre-calculated data for the selected accident sequences. The selection of the most appropriate sequence was made by the use of a decision tree on paper (1 page with 19 source terms), which is based on the source term tree from PSA level 2. The endpoints of this tree represent the Source Term Groups. The specific source term data are available (pre-calculated) for each Source Term Group based on accident simulations (using o/a the MAAP code). The decision tree on paper is still available for the TAG group, because the tree on paper gives an overview of all pre-calculated source terms groups for the Borssele NPP on one page and can be used as a back-up if the SPRINT program does not run (e.g. due to a failure of the PC).

The source term is provided in the format adjusted to the input requirements of the off-site simulation code WINREM. The data include the information on the beginning, magnitude, and duration of release for each nuclide group, elevation of the release point, temperature and energy of release.

Figure 6. An overview of the emergency-organisation for SAM



4.2 Exercises

SPRINT was amongst others used during the following exercises:

- EOP exercises
- SAMG exercises
- a full-scale national emergency exercise.

EOP exercises and SAMG exercises

These exercises are focused on training the operator in application of the EOPs and/or SAMGs within the Emergency Response Organisation. The source term prediction and training of the different responsibilities within the Emergency Response Organisation are part of these exercises as well. The duration of these exercises is approximately 3 hours exercise and 1 hour debriefing/evaluation. The exercise participants are operating in the shelter (the work location of the Technical Support Centre).

The participants to these exercises are:

- scenario-leader
- members of the TAG (typically 3 members)
- MB (for SAMG exercises optionally)
- optionally a control room shift leader or his 2nd in command
- optionally the national safety authority KFD
- optionally observers.

Full scale National emergency exercise

A full-scale national emergency exercise is focussed on training of the complete Emergency Response Organisation, including the authorities (regional and national) and other organisations which are involved in case of a nuclear accident (fire brigades, police etc.). The exercise participants of the Technical Support Centre are operating in the shelter. The duration of the exercise is approximately 10 up to 15 hours.

The participants of these exercises include:
- a full Beleidsteam (SED, MB, MOD and MSB)
- the shift personnel (on the full scope simulator)
- the TAG
- the BOC
- the radiation protection group
- all relevant local, regional and national authorities.

4.3 Experiences of the Exercises

The most important experiences during the exercises with respect to the use of SPRINT were:
- The use of SPRINT can enhance the source estimation significantly due to the power of an early prediction of the timing of a (large) source term to the environment, in case the recovery actions in the plant will turn out to be not successful. The Emergency Response Organization is alerted at an early stage of the accident on the existence of a final source term with a low probability, but severe consequences. For this purpose SPRINT is well suited.
- It is important to start the use of SPRINT at an early stage of the accident. The SPRINT program is very useful for predicting source terms and possible accident evaluations in the early stage of the accident. The SPRINT program predicts also the source term if a part of the questions can't be answered due to the unavailability of some measurements and/or lack of information.
- There were no serious user problems experienced with operating SPRINT during exercises. The SPRINT questions were formulated in cooperation with potential users to avoid misunderstandings. However, for the use of SPRINT it is recommended that the users have had some training in the use of SPRINT, among others some background of the questions and interpretation of the SPRINT results.
- The most important accident management measures are included in SPRINT. The influence of some specific accident management measures on the source term is not included in SPRINT, because there is no separate modelling in the network. This can be improved by additional modelling in the network.
- It is important to have the source term tree on paper available as well (one page with pre-calculated data for the selected accident sequences), because the tree on paper gives an

overview of all pre-calculated source terms groups for the Borssele NPP on one page and can be used as a back-up if the SPRINT program does not run (e.g. due to a failure of the PC), and as a check of the SPRINT results (e.g. to counter specific accident management measures which are not included in SPRINT).

- It is important for the authorities that their staff has some training in understanding SPRINT results (different predicted source terms with each a probability in stead of only 1 predicted source term).
- It is recommended to establish a suitable strategy for the Emergency Response Organisation how to deal with uncertainties and probabilities of the source term prediction. Before using SPRINT only 1 predicted source term was used by the authorities on the status form for communication and for the planning of possible emergency measures. After using SPRINT it should be possible to use more than 1 predicted source term on the status form for communication and for the planning of possible emergency measures.
- The performance of SPRINT is less accurate if the actual accident progression changes due to a **temporarily** restoration of some systems or some **temporarily** accident management measures which influence the timing of the source term. This is caused by the difficulty of determining the influence of all possible temporary measures on the source term predictions.

5. Conclusions

A specific SPRINT model for the Borssele NPP in the Netherlands has been developed. The SPRINT model has been used within the Emergency Response Organisation and by the authorities in the Netherlands during SAMG exercises (including the use of Accident Management measures), EOP exercises and a full scale national emergency exercise. Experiences during the exercises with respect to the use of SPRINT and some findings for improvement of SPRINT have been discussed.

The exercises showed that the use of SPRINT can enhance the source estimation significantly due to the power of an early prediction of the timing of a (large) source term to the environment, in case the recovery actions in the plant will turn out to be not successful. The Emergency Response Organization is alerted at an early stage of the accident on the existence of a final source term with a low probability, but severe consequences. For this purpose SPRINT is well suited.

SPRINT predictions are useful in the communication of the possible source term from the ERO at the plant to the authorities and the planning for the authorities of possible emergency measures (such as evacuation). It is important for the authorities that their staff has some training in understanding SPRINT results.

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List of abbreviations

BNN	Bayesian Belief Networks
EOP	Emergency Operating Procedures
ERO	Emergency Response Organisation
NPP	Nuclear Power Plant
PSA	Probabilistic Safety Analysis
SAMG	Severe Accident Management Guideline
SGTR	Steam Generator Tube Rupture
SPRINT	**S**ystem for the **PR**obabilistic **I**nference of **N**uclear power plant **T**ransients
STV	Source Term Volume

Safety Goals and Safety Targets for Severe Accidents in View of IAEA Recommendations

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1. Introduction

In June 2009 the Council of the European Union has published a directive [1] regarding the safety of nuclear installations, in which the original and still unfulfilled demands made by the EURATOM in 1957 [2] are reiterated:
In order to perform its task, the Community shall, as provided in this Treaty:
(b) establish uniform safety standards to protect the health of workers and of the general public and ensure that they are applied…. In particular, these standards are discussed in articles 30 through 32 of the treaty:

This call for the establishment of common safety standards has been renewed in 1996 [3]. While standards about operational and design basis accidents radiation limits have been established for a long time, only some limited action has been initiated with respect to limits for severe accidents also covered in Article 30, and the fundamental work is pending. In particular, the recent directive [1] requests the following:
Member States should assess, where appropriate, the relevant fundamental safety principles set by the International Atomic Energy Agency [IAEA Safety Fundamentals: Fundamental safety principles, IAEA Safety Standard Series No SF-1 (2006)] which should constitute a framework of practices that Member States should have regard to when implementing this Directive.

The IAEA SF-1 [4] is a high level document, and the specific reference to safety is found in the following principles:

Principle 5: Optimization of protection
Protection must be optimized to provide the highest level of safety that can reasonably be achieved.
3.22. To determine whether radiation risks are as low as reasonably achievable, all such risks, …, must be assessed …

1 Council of the European Union, Brussels, 23 June 2009, 10667/09, Interinstitutional File: 2008/0231
2 Treaty establishing the European Atomic Energy Community (Euratom), Governments of Belgium, Federal Republic of Germany, France, Italy, Luxembourg and the Netherlands, 1957
3 Council Directive 96/29/Euratom of 13 May 1996 laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation, Official Journal L 159 , 29/06/1996 P. 0001 - 0114
4 IAEA Safety Standards for protecting people and the environment, Fundamental Safety Principles, Safety Fundamentals, No. SF-1, IAEA, Vienna, 2006

3.23. The optimization of protection requires judgments to be made about the relative significance of various factors, including:
—The number of people (workers and the public) who may be exposed to radiation;
—The likelihood of their incurring exposures;
—The magnitude and distribution of radiation doses received;
—Radiation risks arising from foreseeable events;
—Economic, social and environmental factors.

Principle 6: Limitation of risks to individuals

Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.

This paper is meant to initiate the response to the EU Directive's call, and parts of it will be discussed within the EU project ASAMPSA2 [20] for inclusion in common EU PSA Level 2 guidelines. The paper first gives a brief summary of the current status of the most recent findings of the numerous working groups, which have investigated safety objectives and goals. The most widely accepted definition of goals is based on the concept of Large Early Release Frequencies (**LERF**) and its derivatives, a surrogate concept derived from results of Probabilistic Safety Assessments (PSAs) which was first introduced in the USA almost twenty years ago and later accepted by the USNRC for risk informed decision making (but not for safety demonstrations). Other types of Safety Goals have been adopted by some nuclear authorities, but the main drawback of most current definitions is that they may apply only to LWRs.

The paper, as recommended in the EU directive [1], will then summarize some of the specific IAEA recommendations and definitions about safety in the nuclear industry, and safety goals in particular, with a special view to the INES scale of accidents and incidents. Next, a brief discussion will be given on the implications and technical bases of the various safety goals with respect to consequences (and risks) on the civilian population, society and the environment.

It will be shown that the currently accepted definitions do not completely conform to these recommendations. Thus an attempt is made to arrive at a definition of a set of Risk Targets for severe accidents in NPPs, consistent with the IAEA definitions and having a technical basis, which can be adopted without modifications for Generation IV power plants.

Finally, the paper discusses some possible applications and especially implications of these Risk Targets with respect to Severe Accident Management, because the proposed scheme may be used to provide a bridge between PSA and Severe Accident Management activities, which so far have been largely based only on deterministic considerations.

2. Current Understanding of severe accidents Safety Criteria

An excellent summary accompanied by appropriate discussion on safety criteria for severe accidents (goals, targets, objectives) has been produced by the NEA (OECD) in [5]. Additional work is under way in the Nordic Countries PSA Group (NPSAG) on this subject [7]. The NEA work has been recently supplemented and in some way completed by the results of answers to a questionnaire on safety criteria [6]. A variety of definitions (both of terminology and criteria) is used in the community, and in all works undertaken on the subject of safety "goals" there seems to be a certain reluctance to discuss the technical bases of the individual criteria. In general, one should distinguish between "limits" and "objectives", in that limits are numerical values that should not be exceeded, no matter what the circumstances, while objectives may be defined with a metric or with surrogates as

5 Council of the European Union, Brussels, 23 June 2009, 10667/09, Interinstitutional File: 2008/0231
6 Untitled, WGRISK Task (2006)-2, Probabilistic Risk Criteria, Draft, OECD-NEA 2009
7 Nordic PSA Group (NPSAG) Newsletter, December 2008

levels to which one should strive for but which may never be achieved. As [5] states, “The most common metrics used are core damage frequency (CDF) and large release frequency (LRF) or large early release frequency (LERF). In some cases these criteria have been defined as surrogates for higher level metrics and in some cases they have been defined in their own right.” In addition, and not to be found in either [5] or [6], the Spanish CSN [10] has approached this issue a few years ago with a view also to the LERF philosophy. It is interesting to note that [5] states that “it is unclear from the information provided what is meant by “early” [in LERF]”.

The concept of LERF was first introduced in the early 1990s, and accepted by the USNRC in 1998, albeit apparently only for risk informed decision making [8]. In [8] the precise definition is:
LERF is being used as a surrogate for the early fatality Quantitative Health Objective (QHO). It is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects.

This concept obviously has little meaning when transported to localities where evacuation planning is not mandatory. Despite this, the same metric is in fact used, without the same context, almost everywhere. For the remainder of this work, the term LERF will be used for all criteria, which have as objective the “early” individual risk of death. This, however, is only one component of “early” individual risks, another being the risk of injuries and delayed cancer fatalities due to early exposure.

The problem of context was recognized in some countries, and a more precise metric is defined there (e.g., UK, Japan, Canada, Holland, Finland). For the most part, the metric has as basis the wish to avoid individual or individual and/or societal risks (specifically, one acute fatality in the immediate aftermath of an accident, or an excessive number of fatalities due to cancer).

For societal risks, however, it is not clear what is meant by “excessive risk of fatalities”, because the relative comparative value (risk of deaths from all other causes) is not given, and therefore the value of the choices (10 or 100 late deaths) cannot be judged in this context. Moreover most of the “other risks” to be compared with may be related to voluntary activities, while the risk from an NPP is essentially imposed on the population.

It is interesting to note that a limit for total core damage frequency (CDF) is also used in conjunction with these criteria. In general, the CDF limit is one order of magnitude higher than the frequency defined for LERF (limit or objective), i.e., it is assumed that for a “balanced” plant the “large” releases make up 10% of the total CDF. This may be deceiving, because it would make one think that the remainder of the CDF results in “harmless” or no releases. The risks of consequences resulting from “smaller” releases are thus not considered.

Until now a technical basis for most of all the definitions and values has not been presented because it can not been found in the literature.

3. Safety Goals and IAEA Recommendations

3.1. The Role of the IAEA

According to definition, the IAEA is the world’s center of cooperation in the nuclear field. The Agency works with its Member States and multiple partners worldwide to promote safe, secure and peaceful nuclear technologies. Three main areas of work underpin the IAEA´s mission: Safety and Security; Science and Technology; and Safeguards and Verification. Within the mission “Safety and Security” the IAEA helps countries to upgrade nuclear safety and security, and to prepare for and

5 Council of the European Union, Brussels, 23 June 2009, 10667/09, Interinstitutional File: 2008/0231
6 Untitled, WGRISK Task (2006)-2, Probabilistic Risk Criteria, Draft, OECD-NEA 2009
8 USNRC, Regulatory Guide 1.174 - An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1, November 2002
10 M. Khatib-Rahbar, et al., “An Approach to Definition of Large Release”, PSAM 7, ESREL ’04, Berlin, June 2004

respond to emergencies. The main aim is to protect people and the environment from harmful radiation exposure. The Agency has put forward the vision of a global safety regime that provides for the protection of people and environment from effects of ionizing radiation. The Department of Nuclear Safety and Security issues safety standards and other publications relating to nuclear, radiation, transport and waste safety in accordance with its programme.

3.2. IAEA definitions related to safety of NPPs

From [11]:

First and foremost, each Member State bears full responsibility for the safety of its nuclear facilities. States can be advised, but they cannot be relieved of this responsibility. Secondly, much can be gained by exchanging experience; lessons learned can prevent accidents. Finally, the image of nuclear safety is international; a serious accident anywhere affects the public's view of nuclear power everywhere. …the means for ensuring the safety of nuclear power plants have improved over the years, and it is believed that commonly shared principles for ensuring a very high level of safety can now be stated for all nuclear power plants; …The international consequences of the Chernobyl accident in 1986 have underlined the <u>need for common safety principles</u> for all countries and all types of nuclear power plants. The comparison of risks due to nuclear plants with other industrial risks to which people and the environment are exposed makes it necessary to use calculational models in risk analysis. To make full use of these techniques and to support implementation of this general nuclear safety objective, it is important that quantitative targets, 'safety goals', be formulated.

1. First of all there is an incontestable need for acceptance criteria related to the basic safety principles, as it results from [11], paragraphs 4.29 and 4.30:

   The acceptance criteria should be defined for the deterministic assessment and the PSA. These normally reflect the criteria used by the designers or operators and are consistent with the requirements of the regulatory body.
   The criteria should be sufficient to meet:
   General nuclear safety objective
   Radiation protection objective
   Technical safety objective

2. The objectives [11] are defined as follows:

   a) General nuclear safety objective
   • To protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard.
   • In the statement of the general nuclear safety objective, radiological hazard means adverse health effects of radiation on both plant workers and the public, and radioactive contamination of land, air, water or food products.
   • The protection system is effective as stated in the objective if it <u>prevents</u> significant addition either to the risk to health or to the risk of other damage to which individuals, society and the environment are exposed as a consequence of industrial activity already accepted. In this application, the risk associated with an accident or an event is defined as the arithmetic product of the probability of that accident or event and the adverse effect it would produce. The overall risk would then be obtained by considering the entire set of potential events and summing the products of their respective probabilities and consequences.

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11 BASIC SAFETY PRINCIPLES FOR NUCLEAR POWER PLANTS 75-INSAG-3 Rev. 1, INSAG-12, 1999

b) Radiation protection objective
To ensure in normal operation that radiation exposure within the plant and due to any release of radioactive material from the plant is as low as reasonably achievable [ALARA], economic and social factors being taken into account, and below prescribed limits, and to ensure mitigation of the extent of radiation exposure due to accidents.
c) Technical safety objective
To prevent with high confidence accidents in nuclear plants; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor; and to ensure that the likelihood of severe accidents with serious radiological consequences is extremely small.

3. The acceptance criteria for severe accidents are usually formulated in terms of risk criteria (probabilistic safety criteria) [12]:

- Large off-site release of radioactive material: A large release of radioactive material, which would have severe implications for society and would require the offsite emergency arrangements to be implemented, can be specified in a number of ways including the following:
- As absolute quantities (in Bq) of the most significant nuclides released,
- As a fraction of the inventory of the core,
- As a specified dose to the most exposed person off the site,
- As a release giving ‘unacceptable consequences’.
- Probabilistic safety criteria have also been proposed by INSAG for a large radioactive release. The following objectives are given:
- 10^{-5} per reactor-year for existing plants,
- 10^{-6} per reactor-year for future plants

4. There is uncontestable need of international consensus on the risk criteria [11], as presented already 17 years ago:

A large off-site release of radionuclides can have severe societal consequences
There is at present no international consensus on the most appropriate measure of what constitutes a large off-site release. However, Member States should give serious consideration to establishing their position on a criterion for large off-site release.
Until such time as an international consensus has been reached, it is suggested that the target frequency for a large off-site release should be 10-6/Ry. A large off-site release is defined as one that has severe social implication.

3.3. PSA scope

Probabilistic risk assessment (probabilistic safety assessment) is a systematic methodology to evaluate risks associated with a complex engineered technological entity. PRA/PSA is characterized by two quantities:
- the magnitude (severity) of the possible adverse consequences and
- the likelihood/probability/frequency of occurrence of each consequence

11 BASIC SAFETY PRINCIPLES FOR NUCLEAR POWER PLANTS 75-INSAG-3 Rev. 1, INSAG-12, 1999
12 IAEA SAFETY STANDARDS SERIES, Safety Assessment and Verification for Nuclear Power plants, Safety Guide No.NS-G-1.2, IAEA, Vienna, 2001
13 The Role of Probabilistic Safety Assessment and Probabilistic Safety Criteria in Nuclear Power Plant Safety, Safety Series No. 108, Safety Reports, IAEA, Vienna, 1992

Consequences are expressed numerically (e.g., the number of people potentially injured or killed) and the likelihoods of occurrence are expressed as probabilities or frequencies (i.e., the number of occurrences or the probability of occurrence per unit time). The stress is on evaluation of risks as the product of frequencies and consequences and this is repeated in several IAEA documents, e.g.: Probabilistic Safety Assessment is a comprehensive, structured approach … for deriving numerical estimates of risk [13]. Probabilistic analysis is used to estimate risk and especially to identify the importance of any possible weakness in design or operation or during potential accident sequences that contribute to risk. [11] The PSA should set out to determine all significant contributors to risk from the plant… PSA should address the contributions to risk arising from all the modes of operation of the plant. [12]

3.4. IAEA INES scale

Originally introduced in March 1990 jointly by IAEA and OECD/NEA, the aim of the International Nuclear Event Scale (INES) [14] is to consistently communicate the severity of reported nuclear and radiological incidents and accidents. It was revised in 2007 to become a more versatile and informative tool. It is taken in the context of this work because it may provide a good bridge between the more abstract definitions shown in the previous section, and quantification of goals. The INES scale follows the ALARA principle, as explicitly stated in the IAEA quotes shown.

The following figure, from [14], provide a summary of the definitions in the INES scale.

11 BASIC SAFETY PRINCIPLES FOR NUCLEAR POWER PLANTS 75-INSAG-3 Rev. 1, INSAG-12, 1999

12 IAEA SAFETY STANDARDS SERIES, Safety Assessment and Verification for Nuclear Power plants, Safety Guide No.NS-G-1.2, IAEA, Vienna, 2001

13 The Role of Probabilistic Safety Assessment and Probabilistic Safety Criteria in Nuclear Power Plant Safety, Safety Series No. 108, Safety Reports, IAEA, Vienna, 1992

14 The International Nuclear and Radiological Event Scale, User's Manual, 2008 Edition, Co-sponsored by the IAEA and OECD/NEA

LEVEL/ DESCRIPTOR	NATURE OF THE EVENTS	EXAMPLES
7 MAJOR ACCIDENT	• External release of a large fraction of the radioactive material in a large facility (e.g. the core of a power reactor). This would typically involve mixture of short and long lived radioactive fission products (in quantities radiologically equivalent to more than tens of thousands of terabecquerels of ^{131}I). Such a release would result in the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long term environmental consequences.	Chernobyl nuclear power plant, USSR (now in Ukraine), 1986
6 SERIOUS ACCIDENT	• External release of radioactive material (in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of ^{131}I). Such a release would be likely. to result in full implementation of countermeasures covered by local emergency plans to limit serious health effects.	Kyshtym Reprocessing Plant, USSR (now in Russian Federation), 1957
5 ACCIDENT WITH OFF-SITE RISK	• External release of radioactive material (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of ^{131}I). Such a release would be likely to result in partial implementation of countermeasures covered by emergency plans to lessen the likelihood of health effects. • Severe damage to the installation. This may involve severe damage to a large fraction of the core of a power reactor, a major criticality accident or a major fire or explosion releasing large quantities of radioactivity within the installation.	Windscale Pile, UK, 1957 Three Mile Island nuclear power plant, USA, 1979
4 ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK	• External release of radioactivity resulting in a dose to the critical group of the order of a few millisieverts.[a] With such a release the need for off-site protective actions would be generally unlikely except possibly for local food control. • Significant damage to the installation. Such an accident might include damage leading to major on-site recovery problems such as partial core melt in a power reactor and comparable events at non-reactor installations. • Irradiation of one or more workers resulting in an overexposure where a high probability of early death occurs.	Windscale Reprocessing Plant, UK, 1973 Saint Laurent nuclear power plant, France, 1980 Buenos Aires Critical Assembly, Argentina, 1983
3 SERIOUS INCIDENT	• External release of radioactivity resulting in a dose to the critical group of the order of tenths of millisieverts.[a] With such a release, off-site protective measures may not be needed. • On-site events resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination for example a few thousand terabecquerels of activity released in a secondary containment where the material can be returned to a satisfactory storage area. • Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur.	Vandellos nuclear power plant, Spain, 1989
2 INCIDENT	• Incidents with significant failure in safety provisions but with sufficient defence in depth remaining to cope with additional failures. These include events where the actual failures would be rated at level 1, but which reveal significant additional organizational inadequacies or safety culture deficiencies. • An event resulting in a dose to a worker exceeding a statutory annual dose limit and/or an event which leads to the presence of significant quantities of radioactivity in the installation in areas not expected by design and which require corrective action.	
1 ANOMALY	• Anomaly beyond the authorized regime, but with significant defence in depth remaining. This may be due to equipment failure, human error or procedural inadequacies and may occur in any area covered by the scale, e.g. plant operation, transport of radioactive material, fuel handling, and waste storage. Examples include: breaches of technical specifications or transport regulations, incidents without direct safety consequences that reveal inadequacies in the organizational system or safety culture, minor defects in pipework beyond the expectations of the surveillance programme.	
DEVIATION 0	• Deviations where operational limits and conditions are not exceeded and which are properly managed in accordance with adequate procedures. Examples include: a single random failure in a redundant system discovered during periodic inspections or tests, a planned reactor trip proceeding normally, spurious initiation of protection systems without significant consequences, leakages within the operational limits, minor spreads of contamination within controlled areas without wider implications for safety culture..	

[a] The doses are expressed in terms of effective dose equivalent (whole dose body). Those criteria, where appropriate, can also be expressed in terms of corresponding annual effluent discharge limits authorized by national authorities.

3.5. Current Safety Goals/Objectives and consequent Quantitative Targets: implications on Offsite Consequences

According to what was briefly discussed in Section 2, the safety objectives and safety goals should be consistent with the IAEA documentation and comprehensive and consistent from the point of view of PSA scope to assess the safety of the nuclear installations. Following the IAEA definition of the Technical Safety Objective, the following points are generally accepted:

- minor (if any) consequences stem from DBAs
- there is an extremely small likelihood of severe accidents with serious radiological consequences.

However, both the IAEA documents as well as PSA philosophy deal with terminology as “large release”, “small likelihood”, “severe”, “serious”, “minor”, etc., without exact definitions of the terms. This is in contradiction with the requirements of quantitative targets, which in turn would provide for credibility and wide acceptance of PSA. Thus the issue is: does the INES scale have technical bases, and do the currently accepted definitions conform to these bases?

As noted, the most widely used semi-quantitative target is LERF. The genesis of the concept of LERF was discussed in Section 1. It is understood as the frequency of “large early” radioactive release, but neither large nor early is exactly defined. The consequences to which it points (i.e., half of the measure of risk) are also not clearly defined, because they involve plant-, site- and offsite countermeasures-dependent aspects. But in general, and especially when a more precise metric is not used, it would seem to take into consideration just the releases of I_{131}, and specifically only the “early fatalities” component, i.e., the extreme consequences that would be induced in humans through inhalation during the passage of a radioactive cloud.

To put into perspective the various definitions of limits and objectives, in terms of offsite consequences, Table 1 shows the results of several MACCS2 [15] calculations. The calculations were performed for a plant located in Central Europe, with a relatively low population density around the plant (the first large settlement located at about 20 km), with Central European weather data. The radioactive release has the characteristics of an early containment failure. The results shown in the Table 1 are for the 95th percentile confidence level (i.e., consequences are not expected to exceed the values shown, no matter what the weather pattern will be). With these assumptions in mind, the results, given the population density, may be said to be optimistic for an “average” European site with larger population density.

Absolute consequences are shown as examples. Similar considerations may be shown with respect to expected doses and relative consequences of different releases. Expected doses are independent of population data, and thus are less site sensitive than the calculated absolute consequences. Table 2 summarizes some of the known or deducible facts about health effects related to exposure to radiation. In order to arrive at some of the data, other calculations have been performed using MACCS2 [15], with several different site specific inputs. Some data is directly deduced from the probabilistic models for chronic mortality (such as mortality rates as a function of TEDE - Total Estimated Dose Equivalent).

One of the problems of estimating absolute consequences is that the effects cannot be calculated deterministically. For example, it is accepted with certainty that a person will die within a short time if the thyroid is exposed even briefly to about 10^7 Bq of I_{131} (from the ICRP dose conversion factors), since this activity would result in about 3 Sv thyroid dose, which is mortal (see [15]). However, the exposure depends on a person’s breathing rate and air concentration of I_{131} at ground level. Assuming a breathing rate between repose and slight activity (between 3 and 6 m^3 / s), the air concentration must be between 3 and 6 x 10^{11} Bq. In order to relate this concentration to a release, it must be determined how far the person is located from the point of release, hence the actual potentially mortal release is a

15 D. Chanin et al., „Code Manual for MACCS2,“ NUREG/CR-6613, SAND97-0594, Sandia National Laboratory, May 1998

Table 1 Consequences of an "early" release corresponding to some of the accepted safety objectives/limits*

Country	Metric	INES	equivalent		consequences				
			I_{131} [TBq]	Cs_{137} [TBq]	Early Fatalities (distance in km)	Early injuries	Late cancer fatalities	Permanent or temporary loss of Land (km^2)	Number of person relocated temporarily or permanently
US + others	**LERF** (Minimize early cont. failure, cont. bypass, isolation failure, SGTR)	**7**	$20e^4$ to $> 100\ e^4$	$2e^4$ to $> 10\ e^4$	0 to > 2 (0.2 to > 5)	2 to > 300	8,700 to > 18,000	800 to > 20,000	57,000 to > 2,000,000
Canada	**Limit** 1% of Cs_{137} **core inventory**								
UK	**Limit**, 10,000 TBq I_{131}	**6**	$1\ e^4$	$< 0.1\ e^4$	0 (0.1)	1	900	1,000	37,000
	Objective, 200 TBq Cs_{137}	**6**	$0.2\ e^4$	200	0 (0)	0	180	200	8,000
Sweden	0.1 % of **core inventory**	**5-6**	$> 0.1\ e^4$	> 100	0 (0)	0	150	>100	> 5,000
Finland	**Limit** 100 TBq Cs_{137}	**5-6**	$> 0.1\ e^4$	100	0 (0)	0	< 100	100	4,000
Canada	**Objective** (new plants) 100 TBq Cs_{137}								
		5 lower limit	**200**	**20**	**0**	**0**	**20**	**< 20**	**<< 800**

* 1 - consequences shown are for a site with low population density (<< 150 person per km^2)
2 - only long-term countermeasures (relocation) are considered
3 - the consequences will not exceed the values shown with a 95% confidence

function of site boundaries, in case it is prescribed that "no person at the site boundaries should be exposed to a mortal dose", or that "there will be no potential for acute death to the public".

Hence, the limit is also determined by weather conditions, and other release characteristic (such as release elevation and energy). Figure 1 shows exploratory calculations to determine mortality distance as a function of different releases and release characteristics. Clearly, the potential is shown to correspond to releases much smaller than what is contemplated in some LERF definitions.

Figure 2 on the other hand shows that the release limit based on 100 TBq release of Cs_{137} assures that no acute health effects will occur, and that long term effects, including land contamination, are very small beyond site boundaries. In the figure are shown in addition the expected doses for one LERF assumption (Large being defined as a release of 3% I_{131} per 1000 MWTh), and for the projected TMI2 accident source term.) Finally, in the figure is also shown a set of calculations performed for a DBA LLOCA accident for a VVER-440 reactor with technical specifications confinement leak of > 10% volume per day.

From the results shown here it is obvious that the wording "potential for early health effects" has been re-interpreted to mean "the potential for measurable or observable early health effects". From the deterministic point of view, potential should mean that an effect is possible no matter what attenuating conditions are conceivable. Measurable, on the other hand, means that, given a site population and weather patterns and effective countermeasures, the most likely outcome (death) is greater than one. With the second interpretation a release of 10^{17} Bq of I_{131} can be shown to have no potential for acute deaths in this exploratory study. However, the result is dependent on the population around the plant, and site-specific weather patterns, and thus cannot be extrapolated with confidence to other plants. Since acute effects are in the deterministic domain, they should be dealt with on a deterministic basis.

Table 2 Radiological consequences

Dose (Sv)	**Type**	**Release of**	**Health effect to exposure**	**Comments**
15 - 25	TEDE lifetime, person-Sv		One cancer death in 60 years	Cumulative low doses; depends on cut-off criteria for mortality. Deducible from MACCS calculations and IAEA Chernobyl projections. A societal effect
7-8	**TEDE acute**	**10^{16} Bq I-131**	**One acute death**	Effect deducible from MACCS calculations. Individual, deterministic. From inhalation and contact only. All radionuclides contribute. THIS LEVEL CORRESPONDS IN EFFECT APPROXIMATELY TO **INES7** LIMIT
3	**Thyroid acute**	**10^{15} Bq I-131**	**One acute death**	Effect deducible from MACCS calculations. Deterministic but site dependent (where are the site boundaries?). THIS LEVEL CORRESPONDS IN EFFECT APPROXIMATELY TO **INES6** LIMIT
0.6 - 1.5	TEDE acute		~One cancer death in ten exposed	MACCS calculations. Deterministic; 1.5 Sv is effective cut-off for acute fatalities.
0.25	TEDE acute	10^{15} Bq Cs-137 (probabilistic)	~One cancer death in 100 exposed	Deterministic "measured" lower limit for contracting some type of cancer. From bomb data and occupational exposure. Threshold for relocation in USA (~4000 kBq/m^2 contamination). Deducible probabilistically from MACCS calculations, figure 2.
0.15 - 0.25	TEDE acute/lifetime		~One death in 1,000 within one year	Gray area where deterministic data on exposures of nuclear workers show some chance of contracting acute cancers. Data disputed by nuclear industry.

Dose (Sv)	Type	Release of	Health effect to exposure	Comments
0.07	TEDE lifetime		~One death in 2,000 in 60 y.	Relocation around Chernobyl. > 1000 kBq/m^2 of Cs-137 contamination
0.05	**TEDE Lifetime**	**< 3x10^{14} Bq Cs-137 (deterministic)**	**~One death in 10,000 in 60 years**	Corresponds to > 740 kBq/m^2 contamination of Cs-137. Fulfils STUK requirements. THIS LEVEL IS IN EFFECT APPROXIMATELY **INES5** LIMIT
0.02	Thyroid acute	< 5x10^{13} Bq I-131 (deterministic)	~One death in 30,000	KI tablet prophylaxis is still needed.
0.0025	TEDE Lifetime	10^{13} Bq Cs-137	< one death in 100,000	Background radiation. Data on fatality rate disputed by industry
0.00025	TEDE Lifetime	10^{12} Bq Cs-137, < 10^{13} Bq I-131	< one death in 1,000,000	EPA 1972 proposed limit for evacuation. Ground contamination 7.4x10^3 Bq/m^2 of Cs-137; **effective cut-off in MACCS2 calculations.**

Figure 1 Probability of mortality versus distance and release of I-131

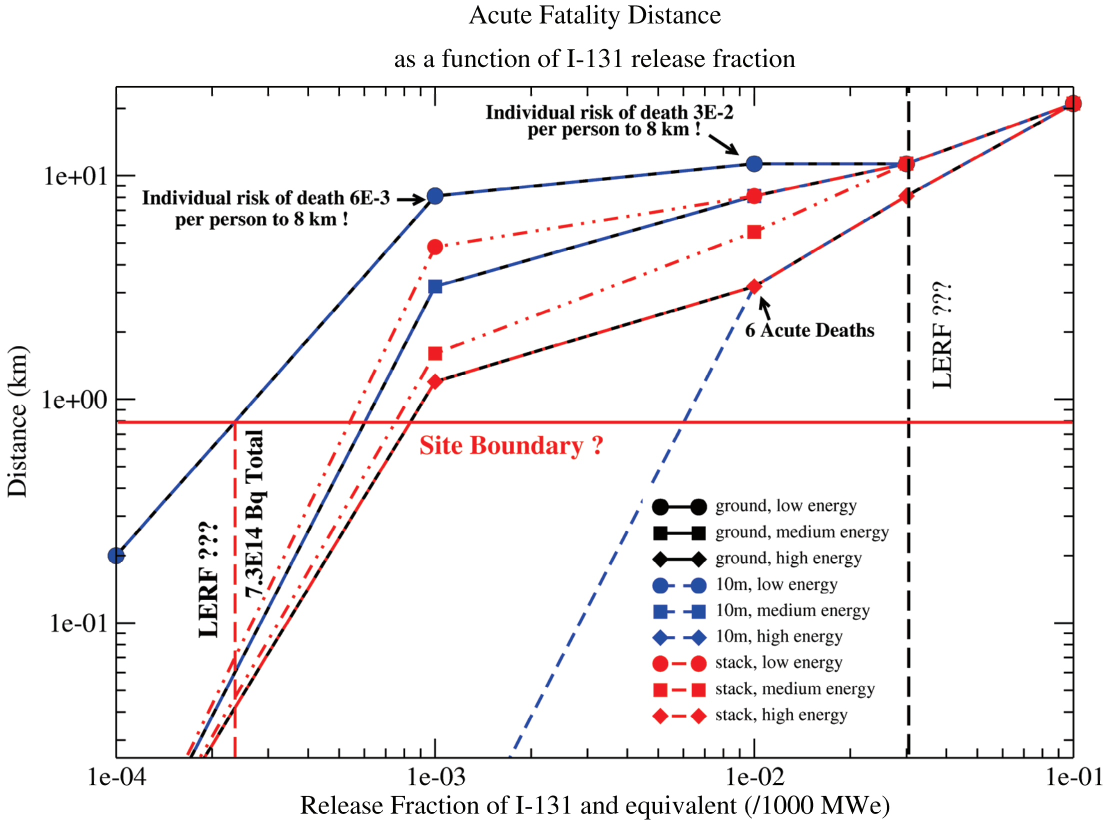
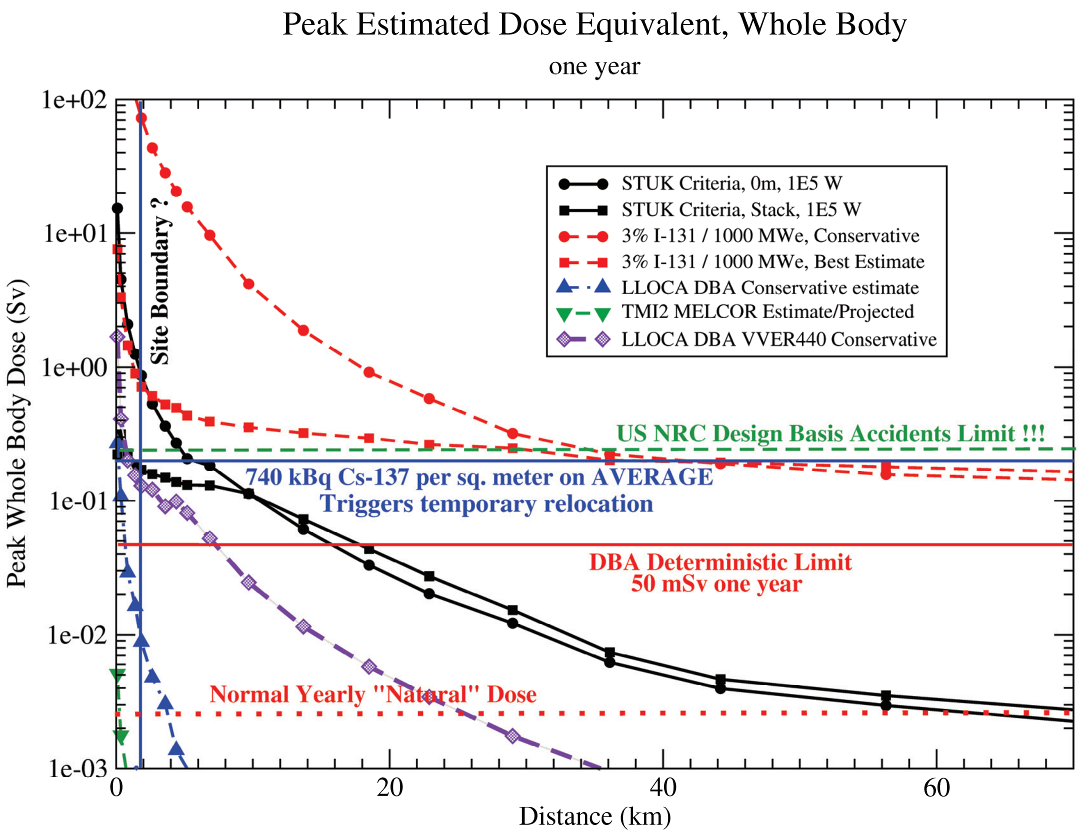


Figure 2 Dose versus distance for various release assumptions



4. A Set of Risk Targets Consistent with IAEA Recommendations

The supra-national consequences of the Chernobyl accident have underlined the need for common safety principles for all countries and all types of nuclear power plants.[11] Currently each country defines its own rules for operation of nuclear power plants, including safety objectives (if any) with respect to severe accidents. Thus safety in general is perceived as a subjective concept and is allowed to be individually defined. There is no question that even the Chernobyl plant fulfilled the requirements of the local authority at the time when the accident happened. The weaknesses of RBMK reactors were already widely known but ignored especially with respect to severe accidents risks. This proves the urgent need for commonly defined and understood risk targets.

Safety goal should be parameters defining the limits, beyond which events are unacceptably dangerous with respect to ALL consequences. According to PSA scope and risk definition, the objective of the safety evaluation should be to demonstrate a Risk Target rather than a Safety Goal, because of two reasons:

- Risk is an exactly expressible value - multiplication of consequences and frequency, whereby consequences and frequency are concrete numbers (definition of technical safety objective, [11]) and not vague terms.
- Target - something to strive for to all extents, and which should be achieved, else the endeavor should be abandoned.

Only this way the IAEA requirements of quantitative targets/criteria [11] are fulfilled, as well as the requirement of risk assessment ([11], [12], and [13].

11 BASIC SAFETY PRINCIPLES FOR NUCLEAR POWER PLANTS 75-INSAG-3 Rev. 1, INSAG-12, 1999
12 IAEA SAFETY STANDARDS SERIES, Safety Assessment and Verification for Nuclear Power plants, Safety Guide No.NS-G-1.2, IAEA, Vienna, 2001
13 The Role of Probabilistic Safety Assessment and Probabilistic Safety Criteria in Nuclear Power Plant Safety, Safety Series No. 108, Safety Reports, IAEA, Vienna, 1992

4.1. Frequency

CDF is used together with LERF and in general the CDF limit is one order of magnitude higher than the frequency defined for LERF. Both are used to demonstrate the level of safety of the plant, assuming that if both frequencies for a given plant are below or equal to the limits, the plant is in general safe enough.

4.1.1. CDF and LERF

It follows that, if the plant's assessed CDF is approximately the same as the LERF limit or lower, large releases will automatically occur with lower frequency than required, even if all sequences leading to core damage lead to LER. And so in this case, according to the currently accepted "frequency" philosophy the plant would not need a containment or the performance of a level 2 PSA to demonstrate that it is safe enough. For this reason the "frequency"philosophy is not sufficient, since it does not guarantee the minimization of consequences stated in the technical safety objective [11].

4.1.2 Risk contribution

The overall risk is obtained by considering the entire set of potential events and summing the products of their respective probabilities and consequences [11]. One of the IAEA requirements is: … there should be no excessive contribution of any sequences to the total risk of the plant.[12] The current semi-qualitative evaluation of LERF may not allow to identify or rank the "significant" contributors to risk, since LERF is frequency oriented and consequences are considered only qualitatively, and thus the risk as the multiplication of the two cannot be calculated (i.e., only the frequency contributors are identified). This is a second reason, why the "frequency" approach of LERF is not sufficient to evaluate the safety of a nuclear power plant.

4.1.3 Real frequency of severe accidents

It must be emphasized, that one of the PSA requirements is the use of data preferably based on the real statistics about equipment, operations, and experience. With respect to this it should not be forgotten that the approach based on the real data should be required in all areas of PSA, and not just in Level 1. From the point of view of real frequency it must be said, that the Chernobyl accident, as representing REAL large early releases (LER), occurred with a frequency of approximately 10^{-4}/year (assuming all reactor-years of operating plants in the whole world) or > 10^{-3}/year (considering only RBMK experience) and is thus 10 to 100 times higher than the defined frequency limit of LER. In this context to fulfill the IAEA safety requirements (including no environmental impacts) the "consequence" part of risk should be reduced to remain within the risk bounds defined by LERF.

This is a third reason, why the "frequency" approach of LERF is not sufficient to evaluate safety. If we consider all severe accidents that have been reported in operating power plants, i.e., six or more events with severe core damage and/or measured release of radioactivity to the environment, the actual frequency is about 10^{-3}/year for all types of plant, i.e., one order of magnitude higher than the accepted frequency limits. The estimated prior frequency for a given plant would be much higher if the Bayesian update approach were to be used.

4.2. Consequences

LERF focuses on early large releases related to "immediate" fatalities after an early containment failure, containment bypass, isolation failure or steam generator tube rupture. However, possible consequences of an accident include also early injuries and delayed fatalities due to short term exposure (two additional components of individual risks), late fatalities and late cancers (both components of societal risks), loss of land/infrastructures and population involved in permanent relocation (components

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11 BASIC SAFETY PRINCIPLES FOR NUCLEAR POWER PLANTS 75-INSAG-3 Rev. 1, INSAG-12, 1999

of societal and environmental risks). The IAEA requirements ask for minimization of consequences [general safety objective] in all three areas, individual, societal, and environmental without any time constraints. So LERF by definition ignores most of the IAEA requirements, even though as is shown in Table 1, some of the metrics used seem to attempt to fulfill such requirements, but this is not specifically stated, because the technical bases are not provided.

According to the other definitions of LERF (serious social and environmental impact) we can assume, that large means minimum of INES7 consequences – i.e. ~10 000 TBq of I_{131}, which means that the currently used metric of safety corresponds to a Chernobyl type accident, i.e. the worst possible case.. In fact by the current definitions of LERF as a safety goal we measure safety ONLY by "serious social impact" but not by "**no** radiological hazards". This is in contradiction to the IAEA requirement of general safety objective [11]: To protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard - radiological hazard means adverse health effects of radiation on both plant workers and the public, and radioactive contamination of land, air, water or food products.

This is the fourth reason why LERF as metric of safety is not sufficient.

4.3. Risk Target as a common safety principle

Risk is expressed as a multiplication of frequency and consequences, i.e.:

RT = f x c, where

RT is Risk Target
f is frequency
c are consequences

The frequency of ALL sequences leading to core damage is expressed by CDF for each plant. CDF limit as one of the risk measures is already defined by IAEA and set to 10^{-4} for operating NPPs and 10^{-5} for future plants. If this frequency is reached, the safety of the plant at the level PSAL1 is satisfied.

Some of these sequences (in the worst case all of them) might lead to large early releases loading the environment by radiological contamination having the effect on health or other damage to which individuals, society and environment are exposed.

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According to the definition of the risk, the total risk foe any risk measure can be expressed as follows:

Sequence/	**frequency**	**conseq.**	**Risk**
s_1	f_1	c_1	r_1
s_2	f_2	c_2	r_2
s_3	f_3	c_3	r_3
…			
s_n	f_n	c_n	r_n
Total risk			***sum*** $(r_1 : r_n)$

Therefore, taking into account ALL IAEA requirements related to risk and consequences it could be said that the minimum INES5 limit in releases of 200 TBq I_{131} equivalent is the NECESSARY condition to fulfill the IAEA safety objectives regarding individual, societal and environmental consequences, while the limit in frequency of 10^{-4} / Ry is the SUFFICIENT condition to minimize ALL risks.

11 BASIC SAFETY PRINCIPLES FOR NUCLEAR POWER PLANTS 75-INSAG-3 Rev. 1, INSAG-12, 1999

Thus, the global Risk Target can be expressed in the following way:

RT = CDF_{lim} x $INES5_{low}$ where

CDF_{lim} is the accepted limit of CDF
$INES5_{low}$ is the lower limit of INES5 scale consequences (release of I_{131} equivalent)

Therefore:

$$\boxed{\mathbf{RT` = 200 \times 10^{-4}\ TBq\ I_{131} / Ry}}$$

If for a given plant

sum (r_1 : r_n) **<= CDF_{lim} x $INES5_{low}$** and
there is no excessive contributor to the risks from among r_1 through r_n,
Then the statement that "the plant is safe enough" is true.

This risk target follows all IAEA recommendations and requirements, and also follows the ALARA principle. This would accommodate the accepted and widely used CDF limit value AND the quantification suggested in the INES scale (although the scale is currently used just as a tool for evaluation of operational events).

4.4. Defence in depth - containment leak tightness

PSA is the tool for risk evaluation and should determine if the safety systems contain an adequate level of redundancy and diversity, i.e., if there is sufficient defence in depth.[12]. The strategy of defence in depth [6] includes several levels of accident prevention and accident mitigation:

Level 1 Prevention of deviation from abnormal operation
Level 2 Control of abnormal operation
Level 3 Control of accidents within Engineered safety features: design basis
Level 4 Control of severe plant conditions, including prevention and accident management; Complementary measures of accident progression and mitigation of the consequences of severe accidents
Level 5 Mitigation of radiological consequences: Off-site emergency response, consequences of significant releases of radioactive materials

The confinement as a last physical barrier - usually in the form of the containment - should provide the tool to keep all excessive releases from reaching the environment. It is known that a weak point of some VVER reactors is the poor leak tightness of the confinement, resulting practically in a large leak for any accident, even when no containment failure occurs (see Figure 2). For this reason, beside the Common Risk Target, also a Common Leak Tightness Limit should be defined, as the necessary technical condition for the safe operation of a plant and as the complement to the Common Risk Target.

5. Further observations on risk targets

5.1. Core inventory

The extent of source terms and possible releases depends on the core inventory. It should be noted that with respect to the frequency as a part of the risk target the historical frequency of a severe accident

6 Untitled, WGRISK Task (2006)-2, Probabilistic Risk Criteria, Draft, OECD-NEA 2009
12 IAEA SAFETY STANDARDS SERIES, Safety Assessment and Verification for Nuclear Power plants, Safety Guide No.NS-G-1.2, IAEA, Vienna, 2001

is almost one order of magnitude higher than the currently accepted CDF limit. This means that the community of safety analysts should be aware, that the corresponding consequence part of risk in the Common Risk Target relation should be even lower than the INES5 lower level to preserve the IAEA "no health effects and no environmental impact" requirements. The only assured way to reduce the consequence part is to reduce core inventories. This is in contradiction with the trend in the evolution of nuclear power plants of the newer generations, for which an increasing power level is typical.
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5.2. ALARA and ALARP principles

The ALARA principle means "as low as reasonably achievable" and it is one of the requirements of IAEA documents ([11], radiation safety objective). On the other hand, the ALARP principle has been adopted most of the time with respect to nuclear safety. ALARP means "as low as reasonably practical", It means that each operator/authority might define his own level of "practicality". Thus, the ALARP principle used in conjunction with LERF, which ignores some IAEA requirements related to safety objectives is only qualitative, does not represent an adequate tool for safety evaluation, since safety is a concept, which is objective, and so it should not be treated subjectively.

5.3. Metric based on I_{131}

A metric that uses a single value, and this in terms of I_{131}, cannot capture the full spectrum of consequences and risks. This mainly for two reasons:

- The teratogenic effects of I_{131} (inhalation dose) are minimal once the radioactive release cloud has passed over the population, and consequent land contamination due to I_{131} deposition is essentially inconsequential.
- In order to induce large enough doses for non-stochastic effects, the air concentration at ground level of I_{131} must exceed about 1 Ci (~ 4 x 10^{10} Bq) per m^3 for a period of one full hour. This is almost impossible regardless of the magnitude of the release, unless a person is located very close to the release point and if the release is concentrated into a short period of time.

Therefore, risk metrics based on I_{131} alone are very insensitive to any type of release, except the catastrophic ones. The INES scale speaks of I_{131} isotope equivalent, and this should not be overlooked or forgotten and so additional INES tables of equivalencies to that radionuclide should be used to account for the other major components of the release. Moreover, the definition based on the single I_{131} isotope should be considered only as a surrogate to summarize all released radioactivity, regardless of the expected effects.

5.4. Applications

5.4.1 Generation IV reactors

The INES scale based on the concept of I_{131} equivalency has provided the basis for a definition of risk targets. Since the reference radionuclide may be of less relevance to Generation IV reactors, there could be two ways to define the "equivalency" to I_{131}.

The first method is based on the ratio of dose conversion factors defined by the ICRP, and the IAEA has provided [16] reference information at least for the most relevant isotopes. However the list of radioisotopes is very limited.

The second method is based on the equivalency of offsite consequences of isotope groups, and has been used for estimates of risks from a variety of power plants, including Generation IV reactors ([17]

11 BASIC SAFETY PRINCIPLES FOR NUCLEAR POWER PLANTS 75-INSAG-3 Rev. 1, INSAG-12, 1999
16 IAEA publication for equivalency of isotopes to I131
17 E.G.Cazzoli and J.Vitázková, "Risk assessments on future nuclear industry development in Switzerland (or alternative nuclear options)", Phases 1 and 2, Summary Executive Report, prepared for GaBe/PSI, September 1st, 2006
18 C.C. Stoker, F. Reitsma, "PBMR Fuel Sphere Source Terms", 2nd International Topical Meeting on HTR Technology, Beijing, China, September 22-24, 2004, Paper C15

Development, Validation and Training of Severe Accident Management Measures

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Summary

The paper describes the current practice in SAMM development, validation and training. It discusses the requirements for an integrated solution for SAMM development, validation and training and outlines a practical approach to accomplish these requirements. A realization of these requirements is demonstrated.

1. Introduction

Severe Accident Management Measures (SAMMs) have been and are being introduced in a number of countries to provide the plant operators and the technical support center or accident management team guidance and training in recognizing and managing a severe accident. The implementation of these SAMMs in terms of development, validation and training has differed from country to country, and more significantly when compared to the validation and training for the plant emergency procedures which prepare the plant operators and the technical support center team in the handling of plant transients and design basis accidents. This paper explores the reasons for these differences and suggests ways to more closely align the processes.

2. Historical Perspective

The introduction of SAMMs was triggered by the Three Mile Island accident where it became recognized that beyond design basis conditions could develop from relatively benign initiating events. SAMMs tend to be symptom oriented rather than checklist oriented, meaning that instructions tend to be of the type: “Is the pressure in the vessel less than xx bar?”, with different paths taken if the answer is Yes and NO. Most SAMMs consist of a set of flowcharts that display the symptom based branching decisions and action instructions, backed up by detailed step by step instructions, explanations and checklists. An example of a SAMM Chart is shown in Figure 1 and was discussed in an earlier paper.

In the United States the Owner’s groups, supported by the plant manufacturers, took the lead in developing the Severe Accident Management Guidelines (SAMGs) for their respective plant designs.

These SAMGs have been implemented on a plant-specific basis at the individual plants. The US took the "new look" approach, where the SAMGs direct the Technical Support Center (TSC) staff to take over the management of an accident when it progresses beyond the plant design basis, and the TSC starts again from the ground up, re-examining the symptoms and evidence of the accident evolution.

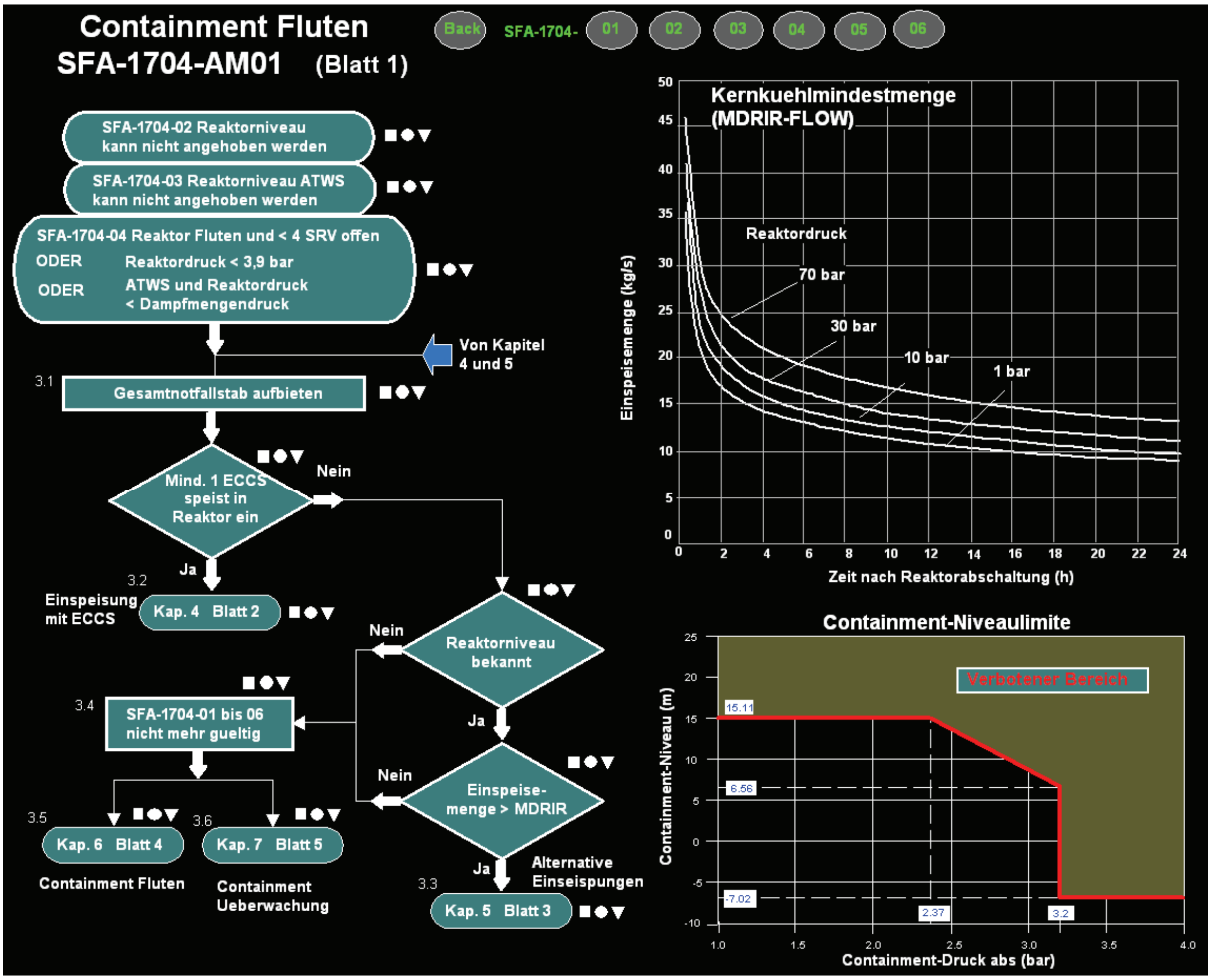


Figure 1: Example of a SAMM Flowchart

In contrast, in Europe SAMMs are currently being developed either by the individual plant operators or by the manufacturers. Here the preferred approach is the "continuity" approach, where the accident management continues at the Shift Technical Advisor level and the TSC comes in as an advisory function to the accident manager, although with decision making authority. In the European approach the SAMMs link to and they continue from the Design Basis Emergency Procedures (DBEPs). The DBEPs have transfers to the SAMMs, and there is no complete re-evaluation of what transpired before the TSC becomes involved. In some cases the SAMMs are fully integrated as an extension of the DBEPs.

3. SAMM Development Approach

In most cases in Europe the SAMMs are developed by the plant technical staff supported by experts in the severe accident phenomenology and severe accident analysis. The plant technical staff is most familiar with both the plant-specific circumstances and the plant-specific implementation of the DBEPs. Through the plant-specific PSA the plant technical staff is familiar with the plant-specific major issues and important aspects of beyond design basis accidents. This knowledge is complemented by the more broadly based knowledge of severe accident phenomenology as well as the modeling and analysis capability of severe accidents provided by experts in the field.

Decisions and actions in the SAMMs are based on the physical conditions encountered in the evolution of an accident. On the one hand this provides a new and independent look at the current situation in an accident that differs from the procedure based approach in the DBEPs. On the other hand it is important to recognize that the physical conditions can change during an accident, so what constitutes a “YES” answer now may constitute a “NO” answer 10 minutes later.

The SAMM development is supported by a limited set of severe accident progression analyses using either the MELCOR or the MAAP code, mostly to develop accident time lines and SAMM graphs. Examples of SAMM graphs are also shown in Figure 1. The MAAP code was first developed by the US nuclear power industry under the Industry Degraded Core Rulemaking program (IDCOR) and is now maintained and licensed by the Electric Power Research Institute (EPRI). MAAP is used by the majority of US based nuclear power plants. The newest version released just recently is MAAP5. It includes more refined models as well as new models for the Balance of Plant (BOP) systems and it is intended for use in analyzing accidents both within the design basis as well as severe accidents. In Europe on the other hand most plants and organizations rely on the use of the MELCOR code developed by the USNRC which is available free of charge in countries that participate in the USNRC sponsored code development and validation program. At some point in the future the ASTEC code developed by a European cooperation is also expected to become a candidate code.

SAMMs are introduced at a plant by training the TSC staff both in the systematic use of the SAMMs and in the technical background knowledge of the severe accident phenomenology and in the accident behavior of their plant. Pre-calculated accident scenarios are used to illustrate both. SAMM drills are carried out in regular intervals of one to two years. The drills may involve a control room shift at the simulator to train the communication between the TSC and the operators via the shift supervisor. In most cases the drill instructor prepares the drill scenario with the support of MAAP or MELCOR analyses of candidate drill scenarios and possibly of variant scenarios with different corrective actions or additional failures. The results of the accident analyses are used to prepare slide plots of the relevant parameters for the accident evolution and the conditions that are discussed in the drill. Slide plots hide the display for times greater than the time frame under discussion so that the TSC staff in a drill does not see the downstream behavior. A SAMM Simulator tool is only used in a few plants.

4. Limitations of Current SAMM Development Approach

The development and implementation approach for SAMMs differs significantly from the traditional approach for DBEPs. The DBEPs are routinely trained in real-time on the plant simulator. The DBEPs are validated continuously through their use in the training exercises, backed up by transient analyses with first principle design basis analysis codes such as RELAP, RETRAN and COCOSYS. The DBEP validation is only limited by the fidelity of the plant simulator.

For SAMMs on the other hand there is in general only a limited validation through the severe accident analyses performed in support of the SAMM development and in support of SAMM drills and the PSA. There are no real-time training exercises or simulations for the spectrum of hardware failures and operator responses beyond the design basis regime. The plant simulator can only be used until the severe accident phenomena, typically is the onset of significant hydrogen production or fuel damage, begin. Beyond that point SAMM training usually consists of a discussion of the accident symptoms, of the expected plant behavior and of the measures that are to be taken according to the SAMMs, without sequence specific real-time input to the Severe Accident Management Team with respect to the time dependent values of the symptom parameters on which the SAMMs are based. So a question can be raised as to whether a) the development of SAMMs should include simulator-like real time validation exercises, and b) if so, whether the plant simulator should be expanded to incorporate the modeling of severe accidents.

The plant simulator is for the training of the plant operators, the shift crews in the Main Control Room (MCR). It is in most cases a full fidelity replication of the MCR driven by the simulator software. SAMM training on the other hand is mostly focused on the TSC staff, although it can involve the shift crew at the plant simulator or in the MCR for communication. The TSC uses different displays for the plant status often referred to as Safety Parameter Display Screens (SPDS) which are more suitable for the TSC to gain an overview of the plant safety status. The TSC training should be, and is, conducted in the environment where the TSC would perform its SAMM function in a real emergency, namely the accident management room or crisis center with some form of a Safety Parameter Display System (SPDS) to indicate the plant status to the TSC. Therefore, for TSC training an extended plat simulator with severe accident capability is not required or useful, but the SPDS must cover the severe accident domain. A secondary consideration is the fact that at many plants the plant simulator is overburdened by the traditional needs for operator and shift training. Whether a real time validation of the SAMMs with means other than an expanded training simulator should be part of the SAMM development depends on the objectives set for the SAMMs and will be addressed in the following.

5. SAMM Validation and Training: The Current Practice

Once the SAMM flowcharts and instructions are developed, they must be tested, validated and trained. The flowcharts should be complete and self-sufficient, so that for at least the first one to two hours of any incident the shift crew only has to follow the charts without consulting the backup instructions, because during this time the accident may be evolving rapidly and the shift must be capable of acting independently until the TSC is assembled and fully up to speed on the plant condition.

Validating the flowcharts is not a straightforward matter because severe accidents can evolve rapidly or over many hours and it is not a priory known what path a given accident will take through the SAMM flowcharts, and therefore it is not a priory known what SAM actions will be called for and in what sequence and time frame. Furthermore, validating the SAMMs in isolation is not practical for the European “continuity” approach to the SAMMs because the SAMMs are entered from the DBEPs and therefore they are linked to the DBEPs. For this reason the joint DBEP and SAMM package must be validated and trained as a unit.

In order to discuss what level of SAMM validation is appropriate we must recall what the fundamental goal of the SAMMs is. The definition preferred by this author is that the SAMMs must lead the operators and the TSC staff to take the right corrective actions in time to either (1) keep the core protected, or (2) to mitigate the consequences to the environment if the prevention of core damage does not succeed. The SAMMs must accomplish their goal regardless of when an accident initiates, during on a workday shift when all plant resources are available immediately or at 3 AM on New Year's day when the shift is initially on its own. What does this mean in practice? Consider the following:

- If only one train of one ECCS system works as designed, no Operator Actions to protect the core are needed because the available ECCS train will be started automatically by the plant protection system and on a realistic basis one train of one ECCS system is sufficient to protect the core even for the limiting DBDA.
- Operator Actions to protect the core are only needed if:
  - (1) some ECCS/safety systems fail mechanically and for all remaining systems and trains the automatic actuation signal fails. In this case the operators have to get through the DBEPs and SAMMs to where the first system with the failed actuation train is started and they have to manually start the train. The limiting case in this scenario is where all ECCS/ safety systems and trains are failed mechanically except for the last train in the last system to be actuated in the DBEPs and in that train the actuation signal is failed and the pump alignment has to be done locally. The least limiting scenario is where the first train of the first system to be actuated by the DBEPs is the one with the failed actuation signal and it can be started manually from the MCR.
  - (2) all trains of all ECCS/safety systems fail mechanically (i. e. not due to signal failures). In this case the operators have to get through the DBEPs and SAMMs to where the primary system is depressurized and auxiliary systems, such as firewater injection, can be started manually. These systems are usually not considered in the DBEPs and they are considered last in the SAMMs. Therefore one can expect that the timing of getting through the DBEPs and SAMMs in the trained systematic manner is most critical.
  - (3) Even if the prevention of core damage is not successful, a measure of SAMM success is still possible if the operators can successfully implement SAMMs that mitigate the off-site consequences of the accident, i.e. the source term release to the environment, by measures such as the local manual isolation of any open containment penetrations or the flooding of the containment.

Operators are trained to systematically follow the DBEPs and SAMMs, they are trained not to make ad hoc decisions. Each step requires its time, sometimes checking or taking actions locally, away from the control room. The time required to reach the point in the DBEP/SAMMs where the first corrective operator action can be implemented depends to a significant extent on which systems and trains are not mechanically failed. This is why timing is critical, but the minimization of the time to go through the DBEP/SAMMs may not have been an explicit goal of the DBEP/SAMM development.

SAMM validation is important because the goal is for the SAMMs to be successful for as many accident sequences as possible. Upon accident initiation, the Operators first follow the DBEPs. They connect to the SAMMs when a SAMM entry condition is reached in the DBEPs. The SAMM entry condition can differ from sequence to sequence. The actions required by the operators to protect the core tend to come at or near the end of the combined DBEPs and SAMMs. Therefore the time required to execute every step through the combined DBEPs and SAMMs is critical, and it varies from sequence to sequence. In some cases different teams or operators pursue different tracks in the DBEPs and SAMMs, such as pressure control, level control or power control with the potential for conflicting decisions and actions.

Today two principal tools are available at virtually any nuclear power plant to validate the DBEP/SAMMs, namely the training simulator and the MELCOR or MAAP severe accident model. The simulator is not a practical tool for DBEP/SAMM validation because (a) it does not include the severe accident phenomenology, and (b) even if it did, the simulator would have to be run in real time so that the DBEP/SAMM decisions and actions can be executed by the operators at the simulator in the correct time frame. This is not feasible for a systematic validation.

Validating the DBEP/SAMMs using the plant-specific MELCOR or MAAP model is inherently an iterative process whereby a specific accident calculation is first allowed to proceed without SAMM actions. The results of this unmitigated analysis are analyzed to determine what the first DBEP/SAMM action would be and the time when the operators would reach this action in the process of going through the DBEP/SAMM charts is assessed. Now the action is programmed at the time when the operators get to this action in the charts. The calculation is repeated or continued with the first SAMM action modeled. This process is repeated for each SAMM action. To validate one severe accident sequence therefore can involve as many restarts as there are SAMM actions called for. Considering that according to PSAs there can be hundreds of distinct severe accident functional sequences, this type of manual validation is not practical, because each accident sequence may have to be run for many times before it is known whether the SAM actions were taken in a timely manner to protect the core. Because of the link to the time it takes the operators to go through the Charts these repeated calculations cannot be automated in the MELCOR or MAAP model and for the same reasons the normal practice to model the actions to be taken when the physical condition arises in the accident calculation may not be valid.

This raises the obvious question: Is the current generation of SAMMs adequately validated? The answer is that we do not know because there has been no broadly implemented methodology or tool to validate the combined DBEPs and SAMMs focused on the scenarios where operator intervention to protect the core or mitigate the consequences is both possible and necessary. If on the other hand the question is “Do current SAMM implementations assure a high degree of confidence that the SAMM actions will be successful?” the answer would have to be that we do not have that evidence at this time. Again the main reason is that no methodology or tool to provide such an assurance has been broadly applied. The next section will examine what the features or requirements for such a methodology or tool would be.

6. Elements of a Reasonably Complete Validation

The goal of a validation would be to verify that the DBEPs and SAMMs lead the operators to the correct actions in time to protect the core and/or mitigate the consequences for a broad spectrum of accidents. To estimate the scope for a reasonably complete validation the first step would be to define a validation matrix of scenarios where operator actions are both possible and necessary.

The main elements of a validation matrix would be:

- Define the functional sequences that require manual operator action
- Consider timing variations: Actions in MCR vs. remote actions. Failures at t=0 vs. staggered or delayed failures, longer or shorter times for the operators to execute the DBEP/SAMM chart steps.
- Check against the important PSA sequences

Analyzing the Validation Matrix sequences involves the following considerations:

- A Validation Matrix may have many sequences. Because of running time considerations with MELCOR it may be necessary to prioritize the validation sequences. The MAAP code is much faster running and it is feasible to analyze a large number of sequences.
- Executing the Validation Matrix sequences manually in real time as described in the previous section is not practical. A separate computer model of the DBEP/SAMMs logic is needed.
- The DBEP/SAMM logic model is linked to the running MELCOR/MAAP model to execute the DBEP/SAMM decisions and actions automatically. Only in this way is it possible to take advantage of the fact that computers can execute logic and actions much faster than operators and therefore it is no longer required to run validation sequences in real time.
- Because a computer model can execute actions much faster than operators it is necessary to model the time the operators require to execute the DBEP/SAMM decisions and actions. These time delays must be realistic and they must come from the Operations staff.
- Now the Validation Matrix sequences can be run in a batch mode. An automatic extraction of the key data to determine success (i. e. core protected) or partial success (i.e. consequences mitigated) or failure (i. e. core damaged and consequences not mitigated) can substantially reduce the time required to evaluate the results of the Validation Matrix calculations.
- For SAMM training the same system can now be used in a real time manual mode where all decisions and actions are taken manually (interactively) by TSC/MCR Staff in real time.

The described Validation features were implemented in the MELSIM/MAAPSIM ActiveCharts system. To date this system has been used to mini-validate a few plant-specific DBEP/SAMM implementations for BWRs. In all cases the DBEP/SAMM instructions performed reasonably well, but all applications showed sequences which did result in core damage. The main causes were:

- The extreme nature of sequences that require manual operator intervention

- The time required to execute DBEP/SAMM instructions, particularly remote actions. In the development and training of DBEP/SAMMs increased attention to the actual times required to execute DBEP/SAMM instructions would be beneficial.
- Conflicting timing of actions in parallel DBEP/SAMM tracks

The main insights gained to date from these limited applications include:

- No two functional sequences behave the same or have the same progression through the DBEP/SAMM charts. Generalized conclusions are of limited value
- The indicated improvements are in many cases self-evident once they are identified. Many have to do with the chart completeness questions.
- Optimization of the DBEP/SAMM charts for optimal execution speed would be expected to bring additional success sequences.

The remainder of the presentation consists of a demonstration of a MAAPSIM ActiveChart implementation.

among other works). As an example for Generation IV, Table 6 shows a summary of results for decay heat calculated with ORIGEN-JUeI [18] for each sphere in a pebble bed reactor (PBR), compared to reference PWR core inventories (MACCS2, 3412 MWTh plant, [15]). Some of the most important radioisotopes are shown in the table.

Table 3 Radioactive inventory per sphere in a PBR, compared to a PWR core inventory

Isotope	Half Life (Years)	Inventory per sphere, average (Bq)	Fraction of total Activity (%)	PWR Inventory, whole core (Bq)	Fraction of total PWR core activity (%)
Sr-90	28.1	8.04e10	0.3	1.94e17	0.1
Ru-103	0.11	6.28e11	2.0	4.54e18	2.0
I-131	0.02	1.77e11	0.6	3.21e18	1.4
I-132	< 0.01	2.02e11	0.7	4.73e18	2.0
Cs-134	2.06	1.20e11	0.4	4.32e17	0.2
Cs-137	30.1	9.71e10	0.3	2.42e17	0.1
La-140	< 0.01	4.52e11	1.5	6.35e18	2.7
Total		3.08e13	100.0	2.32e20	100.0

As an example for the second method of assessing consequences, the comparison given in the table shows that inventories per MW(Th) are about a factor of three higher for isotopes with half lives greater than 1 year, and about a factor of two lower for isotopes with half lives smaller than 1 year (with the exception of Ru). On this basis, assuming a 100% inventory of a reference PWR, source terms for an equivalent HTGR can be adjusted by multiplying by three the release fractions of the groups Cs, Sr, La, and Ce, by keeping the same estimated release fraction for Ru, and dividing by two the release fractions of Xe, I, and Te.

5.4.2 Severe Accident Management

The application of definitions of safety targets based on the LERF concept is well established [8]. The application, however, is limited to risk reduction only for large releases, i.e., only for the releases that would result in risk of individual immediate death, unless the definition of "large" is much more restrictive.

The definition of targets based on the INES scale, on the other hand, provides a more powerful tool, which can also be used for applications to Severe Accident Management (SAM). An example is given here for prioritization of interventions or measures to be implemented. The example is based on work performed for the Swiss HSK (now ENSI) [19] many years ago, and is now outdated, because the plant in question has undergone modifications, the PSA has been revised, and proper plant SAM guidelines have been developed. Therefore, the discussion given here should be regarded as purely theoretical. The data shown, however, was valid at the time when the work was performed.

The issue for the plant in question was that a filtered containment venting had been recently installed, and the PSA showed a very marginal risk reduction due to this system. One reason was the large uncertainty connected with hydrogen combustion at the time of venting, especially related to a potential for detonation in the venting tank or at the exit of the system. This would have resulted in relatively large source terms due to failure of the filtration system. At the time, only sensitivity analyses were performed to calculate the risk reduction, and if the LERF concept had been applied, the results would not have shown any advantage for venting because the sequences needing venting would not have fallen into the class of "Large".

On the other hand, if the concept proposed in this work had been applied, a clearer response would

8 USNRC, Regulatory Guide 1.174 - An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1, November 2002

15 D. Chanin et al., „Code Manual for MACCS2," NUREG/CR-6613, SAND97-0594, Sandia National Laboratory, May 1998

19 E. G. Cazzoli, „Risk Reduction through Severe Accident Management Measures at KKB Power Plant", OECD Workshop on SAM, Stockholm, Jun2 1994

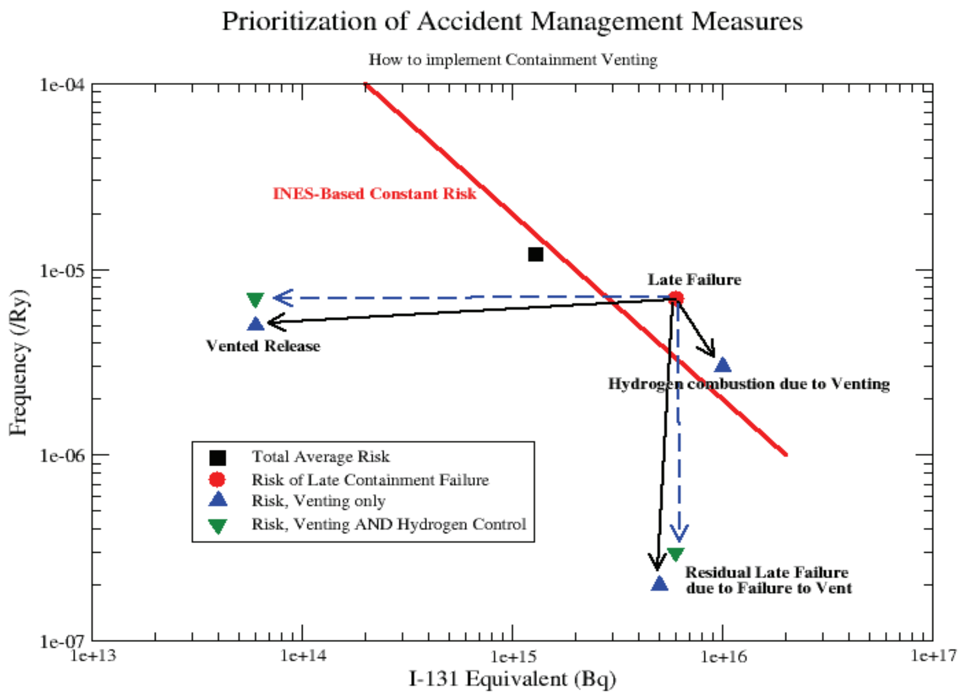
have been given. This is illustrated in Figure 3. The safety targets here defined can be displayed in a line of constant risk (the red line in the figure). When a data point lies to the right of this curve, the result can be considered "unsafe". The red circle represents the risk contribution of late containment failure without the venting system. After the venting system is implemented, three outcomes are possible.

Blue triangle 1: Release due to venting
Blue triangle 2: Failure of the venting system and late containment failure.
Blue triangle 3: Hydrogen combustion due to venting and failure of the venting system.

It can be seen that when the venting option alone is implemented, there is some risk reduction because the risk from accidents with late containment failure (LCF) diminishes. However, one component of risk ("Hydrogen combustion due to venting") remains to the right of the INES line, and therefore some residual risk is still present.

On the other hand, if an effective hydrogen reduction method is provided (e.g., igniters or recombiners) - green triangle, then the risk component due to hydrogen combustion would be eliminated and then the implementation of a venting would have been recognized as fail safe. There would be no discussion with respect to the advantages of installing hydrogen reduction systems at the plant in conjunction with a containment venting system. Note that since this safety targets related technique was not available at the time, discussions on hydrogen control in relation to a venting strategy continued for several years after the PSA was concluded. Similar examples could be given to show effectiveness of SAM measures, and also for prioritization of actions in the actual SAM guidelines.

Figure 3 Example of the possible use of INES-Based safety targets for prioritization of SAM actions



6. Conclusions

The present paper has introduced the concept and definition of Safety Targets for Severe Accidents in NPPs. The proposed general definition is:

$RT = CDF_{lim}$ x $INES5_{low}$

A limit in releases of 200 TBq I_{131} equivalent is the NECESSARY condition to fulfill the IAEA safety objectives regarding individual, societal and environmental consequences, while a limit in frequency of 10^{-4} / Ry is the SUFFICIENT condition to minimize ALL risks.

The definition is consistent with repeated IAEA recommendations (dating from 1990 to the present) that have never been fulfilled. It is general enough to be used for current and future power plants, and has implications for ensuring that the currently adopted or planned SAM measures are effective and necessary.

Part of this work will be used for discussions within the ASAMPSA2 [20] community, for the development of common EU PSA Level 2 Guidelines, in response to the recent EU directive [1] about the development of common safety goals.

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1 Council of the European Union, Brussels, 23 June 2009, 10667/09, Interinstitutional File: 2008/0231

20 SEVENTH FRAMEWORK PROGRAM THEME - Fission-2007-2.13, ASAMPSA2 Project, Advanced Safety Assessment Methodologies: Level 2 PSA, www.asampsa2.eu

Severe Accidents Training in Spain: Experiences and Relevant Features

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1. Introduction

This paper is aimed to the description of the main activities accomplished by Spanish NPPs in the severe accident framework, since the Severe Accident Management Guidance (SAMG) official implementation (2001-2002) until now.

Within this period, a significant experience on features related to the training of plant operators and Technical Support Centre (TSC) members has been gathered, being Tecnatom the responsible of these training activities. The other relevant line of activity has been the updating and improvement of the plant specific SAMGs, with a feedback of new mitigation strategies or changes to the former ones.

SAMG program is officially implemented in Spanish NPPs since 31 December 2000 for American technology NPPs (Almaraz, Garoña, Cofrentes, Ascó and Vandellós) and since 2002 for Trillo NPP (German technology), see Table 1 for more detail.

Programs development was based on the application of the corresponding Owners Group generic guides to specific features of each plant, generating a complete specific technical documentation, including methodology manuals, verification and validation plan, training modules covering different aspects, etc.

An exhaustive process of verification and validation was fulfilled to assess guides usefulness in the decision making process and management feasibility. Validation scenarios were developed taking the NPP Probabilistic Safety Analysis database and calculations carried out with MAAP code as reference.

The complete programs of SAMG implementation have been subjected to technical audits (official revision) by the Spanish Regulatory Body.

Table 1. SAMG Implementation Program

Spanish NPP	Concept	Electric Output (MWe)	Startup Date	SAMG implementation date
Santa Mª de Garoña	GE BWR/3 Mark I	465	1971	December, 2000
Almaraz I, II	W PWR 3-L	980 x 2	1981, 1983	December, 2000
Ascó I, II	W PWR 3-L	1025 x 2	1983, 1985	February 2001
Cofrentes	GE BWR/6 Mark III	1080	1984	December, 2000
Vandellós II	W PWR 3-L	1080	1987	December, 2000
Trillo	KWU PWR 3-L	1065	1988	2002

2. Training activities

Training activities are based on knowledge maintenance and updating related to two basic areas: accident physical phenomenology inside and outside the reactor vessel and scenario proper management using SAMGs.

Training Program designed during the implementation process includes a series of modules covering the following aspects[1,2]:

- Phenomenology and sequence of events associated to severe accident evolution, such as: core damage and relocation, hydrogen flammability, vessel failure, core melt ejection, direct containment heating, corium-concrete interaction, containment failure, etc.

- Technical Basis of SAMGs and Computational Aids with practical applications addressed to cover the correct guidance use.

- High-level actions to be carried out by assigned personnel and performance analysis of the instrumentation involved in severe accident management.

- Training exercises developed from PSA relevant calculations with MAAP code, introducing operator actions contemplated in SAMG.

Specific training programs are addressed for different personnel profiles: operators and TSC members, including Emergency Director. In the case of Trillo NPP, training is extended to operation auxiliaries and instrumentation groups, due to special issues involved in the Severe Accident Manual instructions.

[1] "Implementation of the Severe Accident Management Guidance (SAMG) in Spanish PWR NPPs". M.T. Otero et al. (A.N. Ascó-Vandellós II). PSI-Villigen Workshop, September 2001

[2] "Implementation of the Severe Accidents Program in Garoña NPP". J.M. de Blas et al. (Garoña NPP, Nuclenor) PSI-Villigen Workshop, September 2001

After the plant implementation of SAMG specific programs, the Spanish NPPs set about the retraining program of the technical people involved in their application. The Nuclear Safety Council (CSN, Spanish regulatory body) requirements include carrying out an annual retraining program, mainly focused to TSC team, with the development of severe accidents management drills using and following the SAMG, besides individual emergency exercises for the different groups included in the Internal Emergency Plan framework.

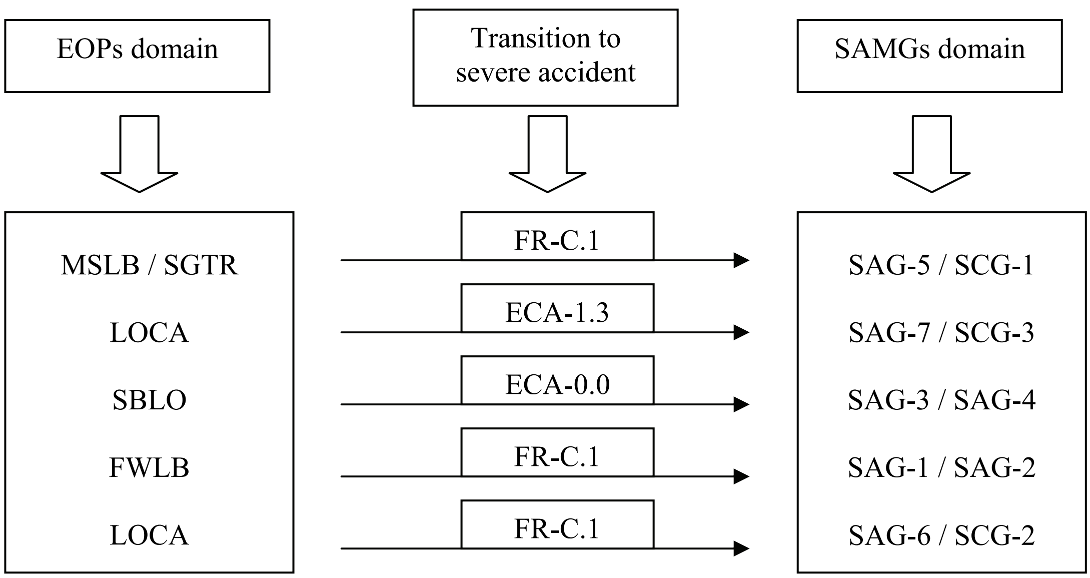
The SAMG retraining to TSC members applies to: Radiological Control and Evaluation Groups, Emergency Director and some members of the Main Control Room crew, in the case of plants with more than one Unit (Shift Manager and Supervisor of the intact Unit). This training process is included in the TSC exercises and drills concerning the NPP Emergency Plan. SAMG retraining for Control Room personnel is based on Severe Accident Control Room Guidelines (SACRG) exercises and transition points from EOPs.

In the case of Trillo NPP, because of its German technology origin, the annual retraining is extended to TSC, Control Room, operation auxiliaries and instrumentation and control groups. For BWR plants (Garoña), retraining program applies to operation personnel, Emergency Director, SAM team and engineering personnel.

The global objective of accidents retraining is the maintenance and upgrading of the knowledge related to the phenomenology and management of their, as well as the performance of the different plant staff groups. This objective takes shape in two basic issues: to remind the use rules of guides and procedures involved in the accident management and to evaluate the simulated scenarios.

During the retraining development, a general view of the SAMGs scope is addressed, and a particular physical aspect is reinforced to a greater detail each year: hydrogen features, fission products release, high pressure in containment, …. Figure 1 shows an example of this type of planning, extended to a 5 years cycle (Almaraz case) where the basic scenario, EOP transition and guidelines are represented.

Figure 1. SAMGs retraining planning



During the annual re-training, the SAMGs basic scope is not only satisfied but also the different transits from the Emergency Operation Procedures (EOP) as a previous situation before the guidelines entry.

In the same line, Trillo applies a 6 years cycle for the retraining in the complete Severe Accidents Manual for all involved groups.

Supporting scenarios for the re-training activities are obtained from accident sequences previously modelled in the full scope simulators, combined with calculations of plant PSA sequences. It makes possible to show the close relation existing between PSA Level 2 conditions and the proposed strategies in the accident management guidelines.

The TSC training schedule is composed of several items and, considering Almaraz NPP as reference[3], is divided in two phases: design basis and severe accident conditions, covering the complete response to an emergency scenario eventually leading to a severe accident condition. Retraining activities in the rest of Spanish NPPs are similar, covering the same scope but in separate time frames. Main items included in the retraining course are:

- Summary of the main physical and operational features of the scenario to be treated.
- Scenario evaluation in the Emergency Plan framework (initial event identification and emergency classification).
- Accident management according to DBA (Design Basis Accident) conditions, resolving the most relevant questions to TSC from EOPs domain.
- Summary of the main physical features of the severe accident with special emphasis on the particular aspect considered in the specific training course.
- Strategies related to the degraded scenario and contemplated in the appropriate SAMGs.
- Practical use of Severe Accident Guidelines (SAGs) and Severe Challenge Guidelines (SCGs), diagnostic diagrams (DFC and SCST) and transition from EOPs.
- Practical applications of the diagnostic diagrams and computational aids required in this accident sequence.
- Discussion about the possible or real modifications carried out in the SAMGs current version.
- Evaluation and Control Radiological Groups performance to response to emergency situation.
- Training drills for the emergency management, before and after severe accident threshold.

[3] "TSC Retraining of Almaraz NPP (2008)". R. Martínez et al. (Tecnatom) INF-9410-01, April 2009

Using simple computer tools for training purposes by TSC members is a very important feature to properly carry out accident management. Nowadays, Spanish NPPs are accomplishing an upgrading plan of the software/hardware tools existing in the TSC of each plant, and training is carried out using these tools[4].

Auxiliary aids have been developed to facilitate the SAMG application and understanding:

- Multi-media software was used as a supporting tool to improve physical phenomena and basic strategies fundamentals understanding.

- Hydrogen curves developed during the process, where the correspondence between the "dry" and "wet" measures for hydrogen concentrations is showed for each plant. This item contributes to clarify hydrogen concentrations and setpoints. These curves can be used also in the EOPs domain.

- Modifications in some Safety Parameters Display System (SPDS).

- Tecnatom is currently developing a TSC aid computer tool for SAMGs use with, basically, a training purpose. This development is being carried out according to the programs for improving and updating the different tools existing in Spanish TSCs.

  This improvement involves software and hardware issues, from redesigning of TSC places in the plant up to introduction of new simple tools. These tools include different dynamic screens to be used by operation and radiological groups, as well as the tasks and responsibilities of the different team members, according to the procedures and guides to be used in the course of an emergency.

  This computer tool is being already used in Almaraz and Garoña NPPs and is in development process for Trillo NPP.

3. SAMG Revision

Revisions of the specific SAMGs (PWR) and EPG/SAG (BWR) has been carried out by the Spanish NPPs, introducing the applicable changes included in the last generic revision of the respective Owners Group[5]. The objectives to be accomplished in the guidelines updating are the following ones:

- To introduce the changes package included in the generic guidelines revision, identifying previously its applicability to plant specific SAMGs.

- To include the results and experiences of the training courses and exercises carried out during last years (6-7 years of real experience).

- To generate a methodology to make change implementation easier for future updating. For this reason, Maintenance Control Sheets have been developed, remarking the modification cause: design change, generic guidelines identification or improvement due to training experience feedback.

[4] "Continuous Training in Severe Accident Management of Almaraz NPP". F. González et al. (Tecnatom) SNE Meeting, October 2005

[5] "WOG SAMG Revision 1 (MUHP-2315), October 2001

- To improve the specific guidelines increasing their applicability and efficiency (e. g., results of PSA revisions, plant design modifications).

- To identify errors, discrepancies and comments to generic SAMG for feedback to Owners Groups.

Nowadays, the situation related to SAMG revision in Spanish NPPs is the following:

- Official implementation of SAMG Revision 1: Almaraz (July, 2008), Ascó and Vandellós (foreseen December 2009).

- Official implementation of SAG Revision 2A: Garoña (2007) and Cofrentes (2008), based on EPG/SAG Revision 2.

From a general point of view, available instrumentation and resources are considered adequate to measure parameters and associated ranges in SAMG (hydrogen concentration, containment pressure, external releases) and relevant instrumentation changes is not considered necessary.

Regarding this matter, containment hydrogen concentration measurement system has been changed by a continuous measurement system with several sampling points spatially distributed in the containment. The pressure measurement range in containment was modified for the correct application of the guideline SCG-4 “Control Containment Vacuum”.

During the SAMG development process some Spanish plants identified a series of areas where it would be interesting to evaluate possible design modifications improving the severe accident management possibilities. Ascó and Vandellós NPPs summarised these possible analyses in the following items:

- Study of other possible recombination systems such as Passive Autocatalytic Recombiners (PAR).

- Analysis to improve the filling capability of the reactor cavity by active or passive ways (dry cavity).

- Analysis of RWST fast filling capability from other site water sources.

- Analysis of the feasibility for establishing fire protection system supply to SGs.

- Analysis of the potential use of other venting paths (different penetrations to containment) considering the venting products transportation to filtering systems already existing in plant.

Concerning to severe accident management, Trillo NPP[6] has undertaken actions on the following areas:

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[6] “Severe Accidents Management in Trillo NPP”. J. de Santiago et al. (Almaraz-Trillo NPPs) PSI-Villigen Workshop, September 2001

- DC Batteries Capacity
- Containment Isolation
- Control Room Air Filtering
- Primary and secondary "feed and bleed"
- External Energy Supply
- Containment Hydrogen Control (introduction of Passive Autocatalytic Recombiners (PAR).

4. Simulator models development

Finally, regarding simulator models development, Tecnatom has carried out the implementation of a severe accident module for the full scope simulator of Laguna Verde NPP (Mexico) (2003-2005).

Laguna Verde Unit 2 is a General Electric design Boiling Water Reactor BWR/5, owned by CFE (Elecricity Federal Commission) and located on the coast of the Gulf of Mexico, in the state of Veracruz in Mexico.

To accomplish the scope, reliability and accuracy requirememnts of the simulator specification, TECNATOM´s TRAC-RT thermalhydraulic code has been utilized for the reactor coolant and main steam system parameter calculations, and the neutronic program NEMO as core neutronics and instrumentation modelling tool.

A MAAP-4 based severe accident module has been incorporated to the simulator modelling package, in order to extend its modelling scope to beyond design accident conditions and to increase the training range making possible severe accident conditions consideration[7].

This module, called 'Containment Advanced Model' (MAC), is integrated with the plant models and enhances the simulator scope to cover the severe accident associated phenomenology, both in the NSSS and the containment.

The approach for the severe accident module implementation is based upon a special set of initial conditions allowing the switching of the core, reactor recirculation, pressure vessel thermohydraulics, and containment models to the MAAP simulation model interfaces. Although MAAP code does not completely model core neutronics and thermohydraulics, it provides sufficient modeling to initiate an accident sequence from full power with subsequent core degradation with very good results for degraded core, containment, and radioactivity parameters.

In this way, simulator capabilities are enhanced by allowing it to be used for the evaluation and validation of the plant specific severe accident management guidelines (SAMG), and to support related training sessions for the involved personnel (operators, Technical Support Centre and managers).

[7] "Laguna Verde Simulator: A New TRAC-RT Based Application". A. Tanarro et al. (Tecnatom) SCS New Orleáns, 2005

This is a supporting tool for: definition and evaluation of severe accident mitigation strategies and analysis of available or alternative instrumentation.

5. Conclusions

From the experience obtained in severe accident retraining in Spanish NPPs, the foreseen objectives in the corresponding programs have been widely covered. The main results and conclusions during the annual retraining are analysed and considered in the future work and the guidelines improvement. Gained experiences mainly show the increasing importance of the dynamic exercises with an increasing participation degree of the involved personnel.

Relevant results of the severe accident training during the last years in Spanish NPPs have been:

- The measurement of an appreciable improvement and familiarisation with the SAMGs use by the emergency team. Practical exercises with application of individual aspects referred to SAMG have been very useful and appreciated by the training groups.

- Technical features feedback related to strategies, response on severe conditions, unusual alignments, working teams actuation in accident management, etc. have been identified and analysed.

- Increasing of the plant participation degree in the “severe accident culture”. This is a very remarkable aspect in the plant actuation indicators on safety features. Organisation aspects such as efficient communication between Technical Support Centre and Control Room, compromising of different plant organisations, decision making process, have been also identified from the SAMG implementation.

- Introduction of simple tools supporting the TSC members actuation supply an improvement in the training process, increasing the interactivity between the participants and making more dynamic the training activity.

- Feedback of obtained experience and participant suggestions to future retraining courses.

- Extension of the PSA to different groups in the plant, not only engineering personnel. Feedback of the PSA and SAMG: data and experience.

- Improvement of SPDS (Garoña NPP).

- Experience interchange between these plants and with similar ones has been an important consequence of the general process.

***Session** 7*

A Novel Process for Efficient Retention of Volatile Iodine Species in Aqueous Solutions during Reactor Accidents

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Abstract

In severe accidents elemental iodine and organic iodides are the main gaseous iodine species in the containment atmosphere. Their contributions vary along the transient depending on many different complex processes. Iodine paint reactions, as well as the reaction of iodine with organic residuals dissolved in sump water are the main source of generation of high volatile organic iodides. Potential release of large quantities of gaseous elemental iodine from the reactor coolant system at certain conditions, or its radiolytic generation in the sump at acidic conditions and subsequent transfer into the containment atmosphere constitute the source of iodine in containment atmosphere. Release of the gaseous iodine species in sufficient quantities through the containment leaks or from the containment venting filter system generates the risk for public health.

Development of a process leading to a fast, comprehensive and reliable retention of volatile iodine species in a containment of a nuclear reactor during a severe accident has been subject to a research project in the recent years at Paul Scherrer Institute.

The process developed utilizes simultaneous use of two customary chemical additives in an aqueous solution. The results of the experimental program have demonstrated achievement of desired fast and reliable destruction of high volatile organic iodine species in the solutions and efficient mitigation of the formation of gaseous elemental iodine from iodide ions under conditions covering a broad range of anticipated severe accident conditions in the containment.

The final phase of the PSI research project focused on the application of the novel process to the nuclear reactors and developed a passive add-on to existing containment venting filter systems in order to dope the containment venting filter water with the chemicals. The necessary hardware may be an integral part of new containment venting filter designs.

This paper introduces a brief state of art on the iodine issues, and then describes the process developed. The hardware modification of existing containment venting filter systems is also described for the implementation of the process. The safety benefits for the implementation of the novel process using an example are highlighted.

1. Iodine issue

Iodine with its nine oxidation stages from minus one to plus seven is perhaps the most reactive fission product in the spectrum of the whole fission products generated and released into the primary coolant system and eventually into the containment during a severe accident. Many different gas and liquid phase chemical reactions taking place in the atmosphere and sump water. They are extremely complex and dependent on a large number of parameters – not only temperature and pressure but concentrations of iodine and other chemical species that iodine may undergo reactions, pH value, radiation dose rates, radical reactions, redox conditions. Surface reactions – adsorption, desorption, chemical reactions – on surfaces with different natures, and mass transfer of gaseous iodine species between the aqueous and gas phases produce additional complexities. Therefore, such complex physical and chemical system make the understanding and hence prediction of the iodine behavior in the containment extremely difficult.

Many small and large scale separate effect tests have been conducted in the last several decades to understand the chemistry and underlying processes and parameters, however under so called 'clean' laboratory conditions. In-pile integral tests , e.g., Phebus FP[1], provided the complexity of the iodine behavior in various phases of the simulated severe accidents; starting from the release of iodine and other fission products and structural materials from the melting fuel bundle with AgInCd or B_4C control rod to the early phase of the transient in the containment when the fission products were transported in the primary coolant piping and further into the containment where aerosol particles, include those containing iodine, largely settle, and to the late phase of the transient when the iodine behavior is basically dominated by the chemistry in the sump water, surface reactions and mass transfer between the sump water and the containment atmosphere. The Phebus test FPT3[2] involving B_4C as the control rod showed an unexpected behavior. The use of B_4C control rod instead of AgInCd provided a large amount of gaseous iodine species entry into the containment, much more than any anticipation based on the past research and modeling.

Empirical and mechanistic computer programs (e.g., IMPAIR, IOD, LIRIC, INSPECT, PSIODINE, etc.) model the iodine behavior based on the understanding gained from the large efforts spent in the experimental programs, however, they present the only practical means of correlating a wide range of possible scenarios, but without providing a confident uncertainty range in the speciation and concentrations of the gaseous iodine species in the containment. The stochastic nature of the development of an accident, operator interventions and many other processes, which affect directly or indirectly the iodine behavior, are also the factors contributing to large uncertainties in predicting the transient concentrations of the gaseous iodine species. The time dependent speciation and the concentration of the gaseous species are the main parameters needed to determine the environmental source term, especially if the containment develops a large leak rate or even fails.

The physical speciation of iodine is traditionally treated as gaseous and particulate form. The main gaseous forms under the containment atmospheric conditions are either elemental iodine or organic iodides. Most volatile form of the organic iodides is methyl iodide in a large spectrum of organic iodides that can be generated. As one of the constituents of airborne aerosol clusters appearing in the containment iodine is mostly in metallic iodides, such as CsI, AgI, etc.

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[1] M. Schwarz, G. Hache, P. von der Hardt, Phébus FP: A Severe Accident Research Programme for Current and Advanced Light Water Reactors, Nucl. Eng. Des. 187, 1999.

[2] B. Biard, Y. Garnier, J. Guillot, C. Manenc, P. March, F. Payot, FPT3 Preliminary Report, Phébus PF IP/06/569, DPAM/CPEX-2007-047, 2007

The recent OECD state of art report (SOAR) on iodine chemistry[3] provides a summary of the achieved state in the iodine chemistry and its modeling. OECD work on the insights into control of release of iodine, cesium, strontium and other fission products in the containment by severe accident management[4] provided a review on the international practices regarding management of iodine in containments. OECD workshop on iodine aspects of severe accident management[5] was organized to address the role of iodine in severe accident management, the needs of the utilities and how research could fulfill these needs.

Although a great progress in the understanding and modeling of several basic aspects of iodine chemistry has been achieved, there is still a deficit in the scientific understanding of the underlying processes which ultimately determine the gaseous iodine species in the containment; elemental iodine and organic iodides and their relative concentrations. Therefore, further international efforts (OECD-BIP, ISTP) are currently spent to understand and generate data on generation of organic iodides, surface reactions, etc.

Long-term research has, unfortunately, not led to a consensus within the international research community on the generation mechanisms of highly volatile organic iodides. At the same time numerous dedicated research projects[6], which were mainly completed in the 1970s, did not lead to effective measures to provide a sufficiently good retention of highly volatile organic iodides after their thermal and radiolytic generation in the containment. Therefore, necessity for qualified and effective iodine management was not achieved, although it was much desired.

The Phébus-tests[1], carried out from 1993 to 2006, have clearly demonstrated the presence of gaseous elemental iodine and highly volatile organic iodides in sufficiently high concentrations persisting in the containment atmosphere. Presence of such concentrations of volatile iodine species in a real accident potentially produces serious consequences if their releases into the environment are not mitigated. The necessity for a proven iodine management is again confirmed by the outcome of the Phébus tests. This fact has imposed a well-known safety deficiency in the management of consequences of severe accidents in NPPs.

This deficit comes from the fact that no proven means (reagents and methods) have been found, which offer a fast and effective decomposition of highly volatile organic iodides and suppression of elemental iodine formed by radiolytic oxidation of generated iodide ions under the severe accident conditions, such as, high temperatures and radiation fields, etc. Difficulties to analyse, identify and quantitatively monitor reduction or oxidation reactions, which generate volatile and non-volatile iodine species, have also contributed to this deficiency.

Filtered containment venting is an attempt to avoid containment failure at high pressure by manual initiation of the venting. Some designs have been equipped with a rupture disc designed to allow automatic initiation of the venting when the pressure reaches an absolute maximum. Venting strategy may vary from plant to plant. The likelihood of need for containment venting is also dependent on the containment fragility and the accident scenarios leading to the need for venting are determined by their PSAs.

Containment venting filters already being installed in nuclear power plants, especially the ones using wet scrubber techniques, were already demonstrated for high retention of the particulates, including metallic iodides. However, demonstration of the high retention of volatile gaseous iodine species was either not secured or not systematically made.

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[3] B. Clement., et. al., State of the Art Report on Iodine Chemistry, NEA/CSNI/R(2007)1, 2007

[4] Insights into control of release of iodine, cesium strontium and other fission products in the containment by severe accident management, NEA/CSNI/R(2000)9

[5] OECD workshop on iodine aspects of severe accident management, CSNI/NEA/R(2000)12

[6] L. F. Parsly, "Chemical and physical properties of methyl iodide and its occurrence under reactor accident conditions (A summary and annotated bibliography)", ORNL-NSIC-82, (1971).

The sump pH, paint surface reactions, and many different radiolytic reactions are controlling the partitioning of the gaseous iodine between elemental iodine and organic iodides. Depending on the venting time and power plant type, accident progression and sump pH control, one of elemental iodine and organic iodides is the main species to be given more focus regarding its higher contribution to the total gaseous iodine concentration in the containment atmosphere. However, available data and model calculations generally suggest that organic iodides are the major contributor of the gaseous iodine in the containment atmosphere in the long term[7].

Therefore, methods must be sought to minimize iodine volatilization in the aqueous systems: reactor sump, containment spray and venting filter scrubber solutions. Fast and efficient conversion of high volatile iodine species to non-volatile iodide ions should be the first step in the process. Furthermore, their possible radiolytic oxidation should be greatly suppressed under any operational conditions, such as at low pH.

PSI has chosen a different direction in managing the gaseous iodine during a severe accident irrespective of how it is generated and (independent of type and the origin of generated iodine species) without knowing with deemed accuracy its magnitude, especially for those power plants equipped with wet containment venting filters. The aim is to suppress iodine release from a containment venting filter system at all feasible conditions of the filter unit defined by temperature, pH, activity levels and other conditions, i.e, presence of other ions, which otherwise might promote the iodine release from the filter system.

2. Effective quantitative iodine retention as an severe accident management measure

In order to manage iodine retention during a severe accident a new efficient technical process has been developed as a result of a dedicated research and development project carried out at PSI since 2002. PSI initiated the research and development project by performing basic research devoted to generation of an easy and effective preparation of labelled organic iodide aqueous solution, iodine speciation analysis techniques[8,9,10]. An experimental programme studying the basic decomposition mechanism of CH_3I by water hydrolysis and radiolysis was conducted as a part of the contribution to the European efforts[11] aiming at investigation of new methods enabling mitigation of iodine release from the containment. The results of the experiments using the methyl iodide demonstrated the repeatability of the literature results and extended the database to other conditions, such as, in-situ β-irradiation effects. The effects of different additives were investigated with the aim to increase the CH_3I decomposition rate. Although, the EU-ICHEMM project related work[12] created new data on decomposition processes of CH_3I, no break through could be attained for a sufficiently high decomposition rate of aqueous CH_3I of interest to reactor safety.

A dedicated project was then conducted between 2002 and 2003, which yielded a big step forward by establishing a fast and effective process leading to very high decomposition rates of CH_3I in aqueous solutions under a variety of boundary conditions. A number of chemical reagents, either singly or in

[7] K. H. Neeb, Radiochemistry of Nuclear Power Plants with Light Water Reactors, Walter de Gruyter Verlag, ISBN 978-3-11-013242-7, 1997.
[8] H. Bruchertseifer, R. Cripps, S. Guentay, B. Jaeckel, Analysis of iodine species in aqueous solutions, 12th Euroanalysis, Dortmund, Germany, p.617, 8-13 September 2002, Anal Bioanal Chem (2003) 375: 1107-1110
[9] R. Cripps, L. Venuat, H. Bruchertseifer, Quick Analytical Method for the Determination of Iodide and Iodate Ions in Aqueous Solutions, Short communication, Journal of Radioanalytical and Nuclear Chemistry, Vol. 256, No. 2 (2003) 357.360
[10] S. Guentay, H. Bruchertseifer, R. Cripps, B. Jäckel, Radiochemical Studies of the Retention of Volatile Iodine in Aqueous Solutions, Proceedings of the 1st International Nuclear Chemistry Congress (1st-INCC), Kuşadası, Turkey, 22-29 May 2005
[11] S. Dickinson, H.E. Sims, F. Funke, S. Guentay, H. Bruchertseifer, J-O Liljenzin, H. Gänneskog, M.P. Kissane, L. Cantrel, E. Krausmann, A. Rydl, Iodine Chemistry and Mitigation Mechanisms (ICHEMM), FISA-2003 / EU Research in Reactor Safety, 10-13 November 2003, EC Luxembourg, http://cordis.europa.eu/fp5-euratom/src/ev-fisa2003.htm
[12] H. Bruchertseifer, R. Cripps, S. Güntay, B. Jäckel, Experiments on the Retention of the Fission Product Iodine in Nuclear Reactor Accidents, PSI Annual Report 2003, Annex 4, 2004

combination, were systematically tested. In particular, efforts were focussed on one which was developed in the last decades and successfully used in the spent fuel reprocessing and metal extraction processes. This reagent, known as Aliquat336®[13](ALI), a quaternary long chain amine used as a phase transfer catalyst and ion exchanger, was investigated in carefully designed chemical processes and through the use of newly-developed analytical techniques. Results indicate the feasibility of not only achieving high decomposition rates of CH_3I in aqueous solution but also of obtaining an effective process to bind the CH_3I decomposition product iodide and thus suppressing its subsequent thermal and radiolytic oxidation to volatile elemental iodine.

The goal of the project conducted afterwards at PSI (2003-2007) consisted of establishing a process involving a phase transfer catalyst and a reducing chemical reagent and to produce a large database for the process characterization covering a wide range of boundary conditions and all possible scenarios in the interests of reactor safety.

As a further prerequisite, the new process should not be difficult to implement in an existing nuclear safety system in order to fulfil its primary goal of achieving a significant reduction of the released volatile organic iodides and gaseous elemental iodine into the environment. In addition to any requirements for the implementation in engineered systems, the process to be established should also clearly require:

- Robustness with respect to reaction parameter variations and operational conditions,
- Demonstration of the guaranteed effectiveness under operational conditions of the existing Containment Filtered Venting Systems (CFVS) and Containment Spray Systems (CSS),
- Long term sustained effectiveness in the presence of other possible constituents in the solution of CFVS and CSS which might also react with CH_3I decomposition products and /or with any one or both additives, especially under radiation fields,
- Demonstration of non-interference with the existing systems, which were already validated for removal of aerosol particles and to certain extent gaseous iodine.

The process developed[14] is basically utilization of an aqueous solution mixture composed of two additives, for use as a chemical reagent to dope containment spray and /or venting filter systems, not only effectively reduces gaseous elemental iodine generated in the aqueous solutions during a severe accident but it also binds iodide ions and hence substantially suppresses its re-volatilization under favourable oxidising conditions. Hence methyl iodide if generated or transferred in the aqueous solution is efficiently retained in solution. The invented process therefore closes a gap long overdue in the severe accident management strategy.

The engineered use of the solution mixture in existing containment venting filter systems in NPPs, e.g., in Swiss NPPs, offers, for the first time, an internationally acceptable[14] and reliable means for effective suppression of the environmental release of volatile iodine species from a containment venting filter system.

3. Outcome of the PSI research and development project

The results of the PSI research project have confirmed the conclusion of past research[6,7] on the use of alkaline thiosulphate solution, which facilitated an effective reduction of elemental iodine and CH_3I into non-volatile iodide ions. However, dynamic boundary conditions, for example, changing mass transfer rates, such that might be expected to occur in a containment venting filter system, have produced unsatisfactory, undefined and ineffective retention. Furthermore, the known reduced

[13] Trademark, COGNIS, www.cognis.com

[14] S. Güntay and H. Bruchertseifer, Fast Reduction of Iodine Species to Iodide, European patent application, 2005 and International patent application PCT, 2005.

effectiveness of aqueous thiosulphate solution at low pH, which might be caused by acidification due to other chemical reagents generated during the progression of the severe accident, might provide favourable conditions for radiolytic re-oxidation of iodide ions into volatile elemental iodine. The PSI research demonstrated that the concurrent use of a phase transfer catalyst, specifically, Aliquat336®[9] eliminates these problems.

The experiments conducted in the dedicated research and development programme demonstrated the suitability of Aliquat336 for NPP safety systems. It was characterized as a versatile chemical additive, since:

- Aliquat336 is a technical product, which is successfully already applied to nuclear technological processes, such as, spent fuel reprocessing and other metallurgical processes for metal extraction from ores. Its high stability to ionising radiation was demonstrated by the spent fuel reprocessing industry,
- As a co-additive to alkaline thiosulphate solutions (THS) commonly used in the existing containment venting filter solutions, it increases the thermal decomposition rate of CH_3I (Figure 1 and Table 1) and additionally binds the iodide ions formed from the decomposition process. This latter process is its most important aspect. That is, it effectively suppresses the oxidation of iodide ions in uncontrollable boundary conditions anticipated to occur in the system,
- A wet scrubber system doped with these additives could be the basis of a potential filtration system of the containment for normal operation and could eliminate many disadvantages of the currently used HEPA/active charcoal filter systems,
- A comparison of the results of experiments containing typical THS solution concentrations used in containment venting filters of some NPPs as well as using those developed from our research results clearly demonstrate (Figure 2) an increased rate of the methyl iodide decomposition by about 3 orders of magnitude, which translates into a decomposition rate of 97% per 5 minutes instead of 0.1%.

The new procedure for the retention of all volatile iodine species is already patented[14].

The outcome of the experimental program, consisted of over 1000 experiments, can be summarized as:

- The high retention efficiency (decomposition rate) of methyl iodide is valid over a very wide range of CH_3I concentrations and covering all anticipated concentrations (Figure 2, Table 1). It is independent of the pH in the range of 4 to 14 (Figure 3) and is even very effective at very strong acidic conditions. The decomposition rate rapidly increases with increasing solution temperature (Figure 1 and Figure 4) although the true decomposition rate at high temperature should be even higher than the measured values due to the difficulties in measuring the extremely fast process with the measurement techniques used,
- Release of elemental iodine is suppressed under irradiation (Figure 5) and continuous gas (N_2O) sparging,
- The decomposition rate and binding of iodide ions are not affected by the presence of the selected ions (Figure 6) in the experimental programme,
- High retention is secured even if a potential reduction in the pH of the solution by acidification due to acid inflow or generation (HCl, HNO_3, H_2CO_3) in the aqueous solution happens
- The efficiency of the containment venting filter for removing aerosol particles and iodine is not impaired when doped with the additive mixture.

The dynamic system behaviour by using non-condensable sparging gas containing CH_3I through a water column provided decontamination factors at water temperatures greater than 30 ^oC from several hundreds to several thousands depending on submergence. At cold water conditions, due to excessive steam condensation extremely high decontamination factors are expected.

Finally, long term experiments (aging of solutions up to a year and pre-irradiated additive solutions indicated that the thiosulphate and Aliquat336 aqueous solutions are stable for long-term operation and over a wide range of pH range of and whilst being subject to high radiation in the aqueous system which guarantees practically a complete retention of all volatile iodine species at any operational temperature.

4. Implementation of PSI process in NPP containment venting filter systems

The implementation, Figure 7, foresees a complete autonomous operation requiring no operator intervention for the operation and no external energy source. The implementation, although dependent on the utility requirements, foresees either one or two pairs of relatively small tanks to store separately alkali thiosulphate and Aliquat336 solutions. The first pair of tanks will automatically discharge both solutions directly into the water pool of the containment venting filter tank using the higher pressure in the connection line between the containment and the venting filter tank as the means for pumping, once the venting is initiated. The second pair of tanks will automatically inject the solutions later especially if the venting is to be done for a second time. The second pair of tanks is also foreseen for desired refilling and manual operation at any post accident time. The gas space of the tanks is foreseen to be filled with nitrogen to eliminate any question about the degradation of the solutions by oxygen diffusion along the years that the system stays in standby condition and will hopefully never be in operation.

The dimensioning of the tanks, the conditioning of the venting filter water with the solutions as well as specific conditions regarding the filter operation are to be performed for specific containment venting filter design and the power plant characteristics including the venting strategies.

5. Anticipated global safety benefits

The exact global safety benefit of using the novel system[14] developed at PSI for the iodine management as a part of the containment venting system is very much dependent on the core damage frequency of the nuclear power plant in question and the fractional distribution of the accident scenarios leading to high pressure in the containment challenging its integrity. As an example if a PWR, based on its PSA, has the following very rough distribution of initiating events:

- 50% due to the fires and earthquake, each of which leads to a station black-out (SBO) scenario,
- 25% due to the loss of feed water (LOFW) transients and
- 25% due to the small breaks (SB) loss of coolant accidents.

Then one may very roughly expect based on the general experience that about 50% of the SBO, 40% of LOFW and 60% of SB transients would lead to the pressurization of the containment challenging its integrity, especially under assumption that the containment remains isolated and the leak rates stay very small. This assumption will lead to then approximately 50% of the whole code damage frequency involving scenarios resulting in containment venting, if equipped, via the venting filter. This means, if the core damage frequency (CDF) is roughly $7\cdot10^{-6}$ y^{-1} it means that the venting frequency is roughly 4.10^{-6} y^{-1}. Again one should remember that actual numbers are to be established using the real figures for a real power plant in question. The reduction of the source term to the environment is then to be considered using the information as depicted in Table 1. Therefore, a substantial safety benefit regarding the reduction in iodine source term and hence associated risk is to be expected by implementing the PSI iodine management system.

6. Conclusions

Even after many decades of research there are still missing gaps in the understanding and modeling of some key issues of iodine behavior, such as formation of organic iodides, possibility of existence of inacceptable high containment iodine concentrations during the core melting phase, especially from the cores with B_4C control rods. The current understanding of the iodine behavior is that unlike the airborne aerosol, some gaseous iodine species will persist to exist at a certain concentration in the containment atmosphere, however, high enough to cause health concern, if released into the environment by large leaks or containment failure. The PSI research has concentrated on finding and establishing a novel process to suppress the release of gaseous iodine species from aqueous solutions, independent of the kind of their formation. The process enables not only fast and efficient destruction of organic iodides into non-volatile iodide ions but also fixation of iodide ions so that their subsequent radiolytic and thermal oxidation is suppressed. The process is demonstrated to be effective at a large range of pH, dose, temperature, in the presence of other ions and under dynamic systems, in which volatile iodine species are transferred from the gas into the aqueous phase during a sparging application such as in a containment venting filter operation.

Back-fitting existing wet containment venting filters or a complete new filter system incorporated with the necessary hardware for the implementation of the iodine management system is the anticipated near future applications to improve the safety of the nuclear power plants. Although the real benefit of implementing the PSI's iodine management system is dependent on the safety level of a NPP, regarding the CDF and relative importance of the initiating events leading to the pressurization of the containment challenging its integrity and hence to cause potentially high iodine release into the environment. An example has been provided for a fictitious plant with assumed fractional distribution of initiating events and fraction of such events leading to the venting to provide an idea about the safety benefit of implementing the PSI novel system for iodine management.

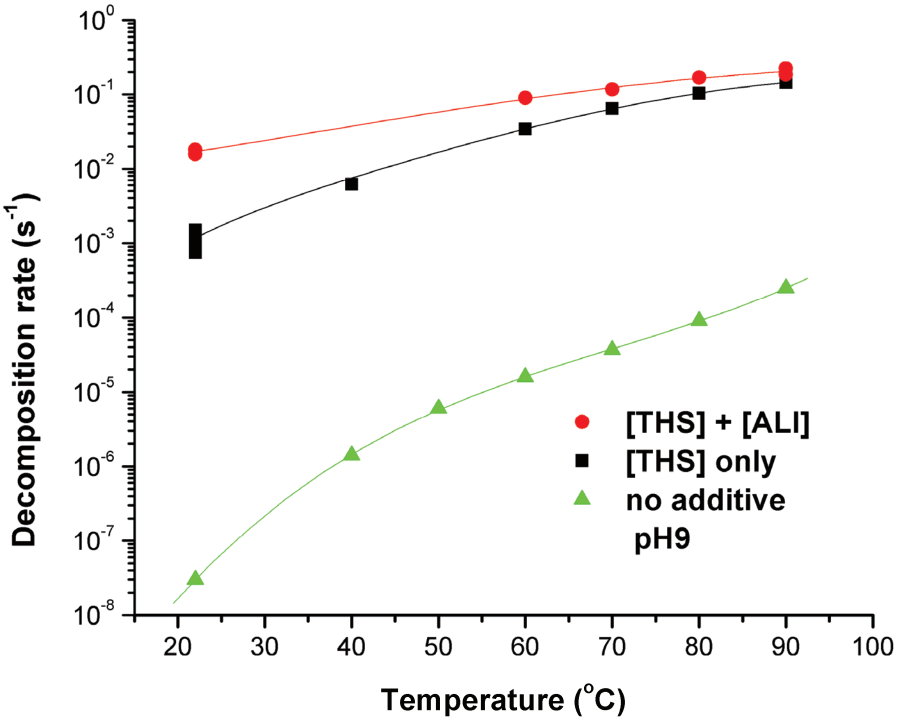

Figure 1: Enhanced decomposition of CH_3I by the use of Aliquat336 (ALI)

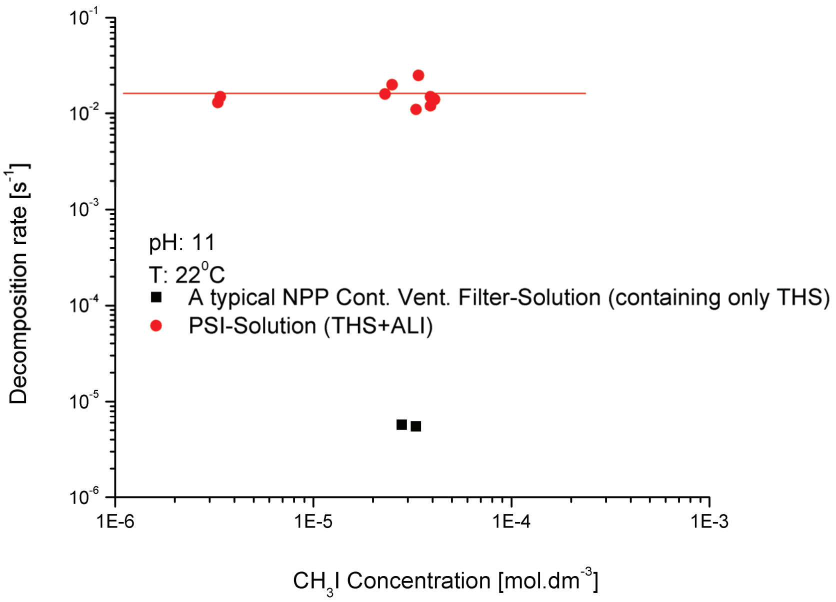

Figure 2: Enhanced methyl iodide decomposition by the use of PSI process

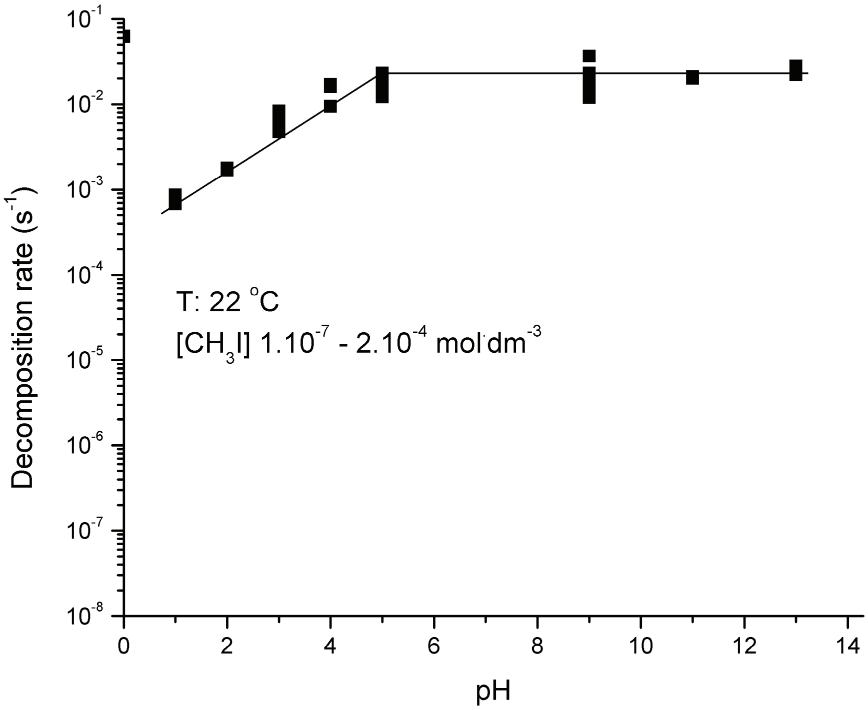

Figure 3: Effective methyl iodide decomposition in a large range of pH

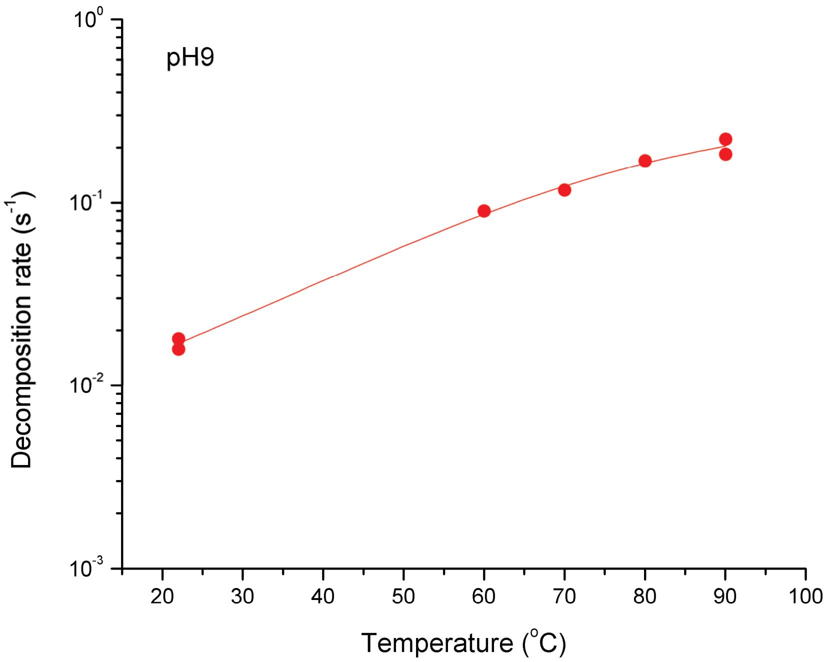

Figure 4: Increased decomposition at higher temperature with PSI solution mixture

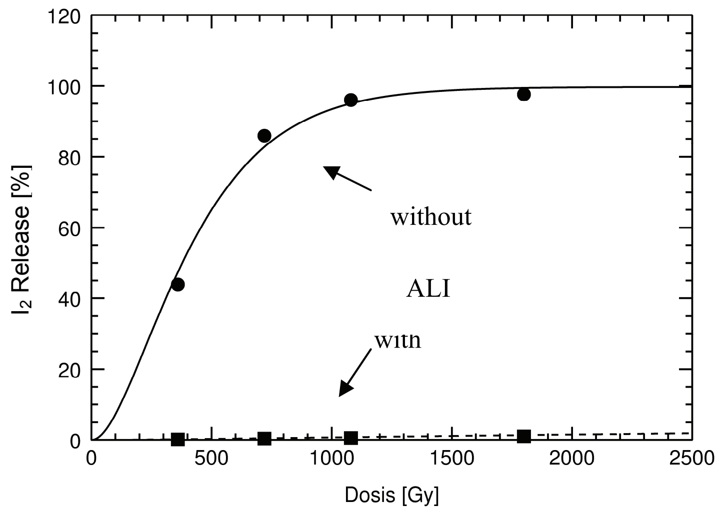

Figure 5: Suppression of elemental iodine release under irradiation

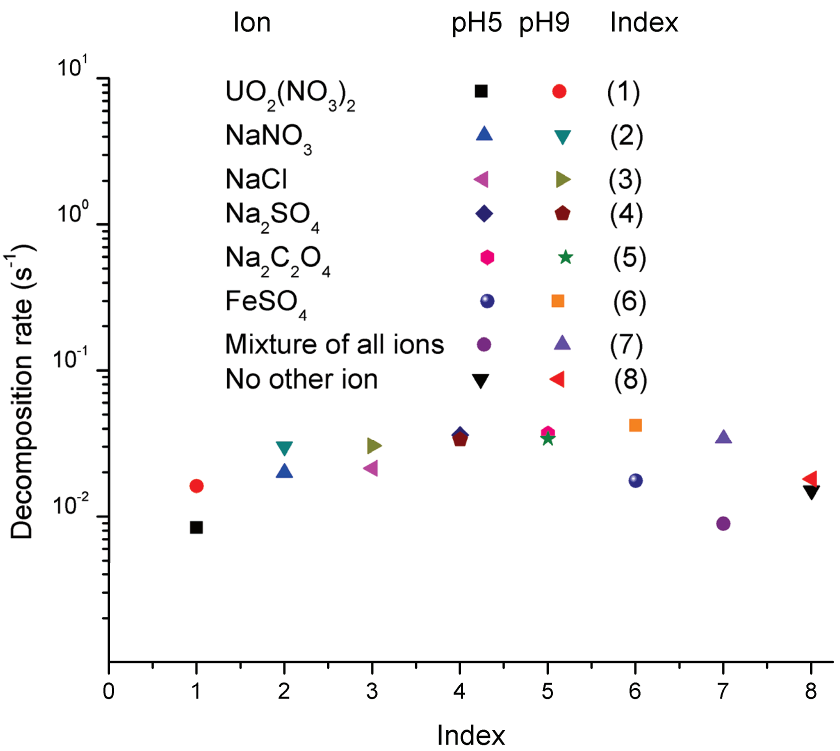

Figure 6: CH_3I decomposition rates vs. selected ions on interference on the process

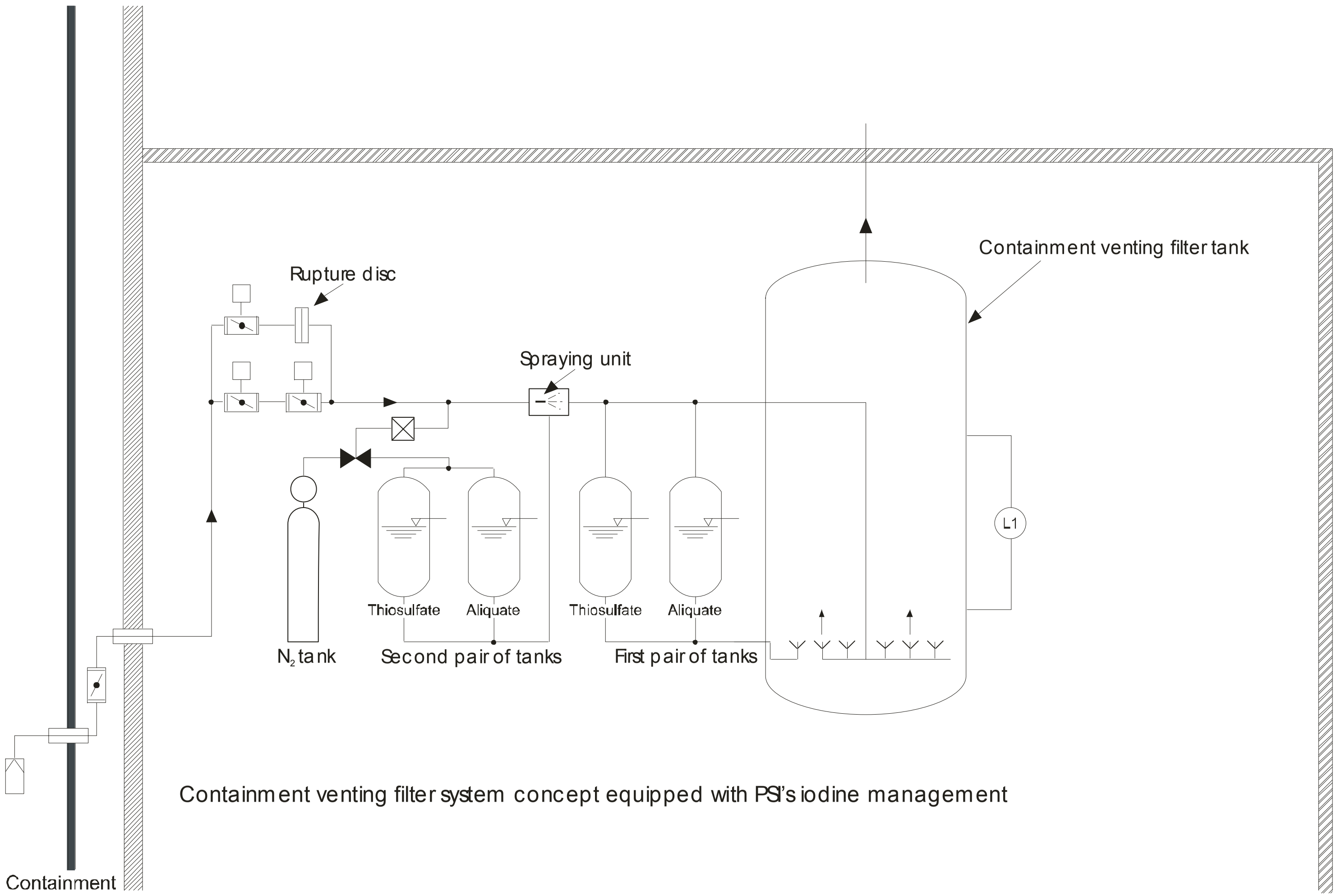

Figure 7: A typical back fitting of a containment venting filter system for the implementation of the PSI process

Reaction mechanism	Enhancement factor in reaction rate with respect to that at 25^{o}C or 80^{o}C	
	25 ^{o}C	80 ^{o}C
No additives (hydrolysis alone)	1	$1.4 \cdot 10^3$
Radiolysis + Hydrolysis	$11 \cdot 10^3$	$12 \cdot 10^3$
Hydrolysis + THS alone	$15 \cdot 10^3$	$1200 \cdot 10^3$
Hydrolysis + THS+ALI	$200 \cdot 10^3$	$>2000 \cdot 10^{3*}$
Hydrolysis + Radiolysis + THS+ALI	$210 \cdot 10^3$	$>2000 \cdot 10^{3*}$

*actual factors must be higher due to the limitation of the measurement technique used to determine very fast decomposition rate at high temperatures

Table 1: Enhancement of CH_3I Decomposition Rate

DEVELOPMENT OF LEIBSTADT NPP SEVERE ACCIDENT MANAGEMENT GUIDELINES FOR SHUTDOWN CONDITIONS (SSAMG)

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Kernkraftwerk Leibstadt AG, Switzerland

1. Abstract

In response to the request of the Swiss Nuclear Safety Inspectorate (ENSI), the Leibstadt NPP introduced 2004 the current Severe Accident Management Guideline (SAMG) based on the BWR Owners Group (BWROG) Emergency Procedure and Severe Accident Guidelines, Rev. 2. The existing procedures cover all plant operating states from full to low power with the exception of shutdown conditions. The important strategy and main procedure for severe accident management of the Leibstadt BWR-6/MARK-III boiling water reactor is containment flooding.

Following the completion of the Leibstadt Shutdown PSA, the Swiss authority requires the development of additional Shutdown SAMG (SSAMG) by the end of 2009. As a basis for managing core degradation situations, an understanding of plant-specific severe accident conditions needed to be developed. Based on the Shutdown PSA results a set of eight scenarios was defined and analyzed using the MELCOR/MELSIM code.

The Leibstadt specific insights are currently used to expand the existing EOP and SAMG procedures. The procedures will be developed in accordance with the new ENSI Regulatory Guideline B12 “Notfallschutz in Kernanlagen”.

2. Brief Description of Leibstadt NPP

The Leibstadt NPP is located on the south side of the Rhine river in the northern part of Switzerland. Leibstadt is a BWR-6 type NPP rated at 3600 MW (thermal) with Mark-III containment. It is the most recent and largest plant in Switzerland with a net electrical output of 1165 MW.

The Emergency Core Cooling System (ECCS) of the Leibstadt plant is composed of seven subsystems: one High Pressure Core Spray (HPCS), one Low Pressure Core Spray (LPCS), three Low Pressure Core Injections and two Special Emergency Heat Removal (SEHR) systems. HPCS is the backup system for the steam driven Reactor Core Isolation Cooling System (RCIC). Once the RPV level cannot be maintained by the high pressure injections systems (feed water, HPCS, RCIC) or a failure of these sytems occurs, the Automatic Depressurization System (ADS) will activate in order to allow low pressure systems (LPCS, LPCI) inject into the RPV. LPCI is one of the Residual Heat Removal (RHR) modes. The other two important RHR modes for accident mitigation are Suppression Pool and Fuel Storage Pool Cooling.

The Leibstadt NPP has a MARK-III type containment. The Mark-III design incorporates the concept of the pressure-suppression feature with a dry containment configuration above the suppression pool as illustrated in Fig. 1. The primary containment is composed of containment, drywell, upper pool, suppression pool and vent system. One of the functions of the upper pool is to supply water inventory to the suppression pool in the event of a LOCA. The suppression pool is an annular pool located between the drywell weir wall and the containment wall. The Mark-III arrangement uses horizontal vents to direct steam from the drywell into the suppression pool in a LOCA event. A build-up of pressure in the drywell will force water down into the vent annulus and uncover the vent holes.

During the refueling outage the drywell head cavity is drained. The drywell head (separating drywell and upper pool), the RPV head, steam dryer, and steam separator are removed. Afterwards the cavity is flooded again and the gate (isolating the upper fuel storage pool and the transfer area) is removed for refueling. At the end of refueling the spent fuel is transferred to the fuel handling building with the Inclined Fuel Transfer System (IFTS). The containment and drywell equipment hatches are opened to the secondary containment for maintenance purpose. At that time the containment pressure suppression capability is defeated.

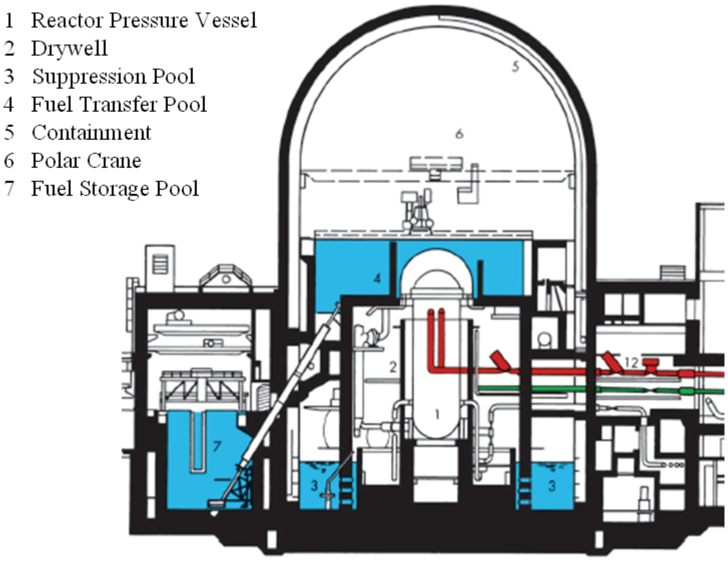


Figure 1 Mark-III Containment

3. Leibstadt Operating Procedures

The safe and reliable operation of a nuclear power plant requires comprehensive operation guidelines for all possible plant operating states including normal power operation, start-up, and shutdown and plant outage/refueling.

Leibstadt has a structured system of procedures for both normal and off normal operating conditions.

The „Technical Specification Leibstadt", TSL includes, as the central document, all binding requirements for safety systems and equipment necessary for the safe operation of the plant. Compliance with the TSL requirements is verified with the System and Instrumentation Functional Test procedures (SFT/IFT).

The specified operating limits defined in the TSL are ensured with the aid of the Station Operating Procedures (GFV), the System Operating Procedures (SFV) and Checklists (URC), the Alarm Response Procedures and the Event Oriented Emergency Procedures (SFA). During abnormal and emergency conditions mainly symptom oriented Emergency Procedures (SFA (EOP)) come into use. The existing Emergency Operating Procedures have been expanded by Severe Accident Management Guidelines (SFA-AM (SAMG)) to prevent or mitigate the consequences of accidents jeopardizing the core integrity.

The Station Emergency Procedures (NFA) are a set of administrative directions, covering the necessary organisational aspects of an emergency, including alerting, information and evacuation.

<table>
<tr><th colspan="2">Responsibility</th><th colspan="3">Operations</th><th colspan="3">Emergency Organization</th></tr>
<tr><td colspan="2">Safety Layers</td><td colspan="3">All Operating Modes</td><td>> 93 °C</td><td>< 93°C</td><td rowspan="6">NFA</td></tr>
<tr><td></td><td>Normal Operation</td><td rowspan="3">TSL</td><td rowspan="4">GFV
SFV</td><td></td><td rowspan="3"></td><td rowspan="3"></td></tr>
<tr><td>SE2</td><td>Operational Transients (control - and safety provisions required)</td><td rowspan="3">ASA
SFA</td></tr>
<tr><td>SE3</td><td>Design Bases Accidents (safety systems and provisions required)</td></tr>
<tr><td>SE4</td><td>Severe Accidents Severe Accident Management</td><td></td><td rowspan="2">AM01
(SAMG)</td><td rowspan="2">AM02
(SSAMG)</td></tr>
<tr><td>SE5</td><td>Core Melting / Radiation Release, Severe Accident Management</td><td colspan="3"></td></tr>
</table>

Table 1 KKL Operating Procedures

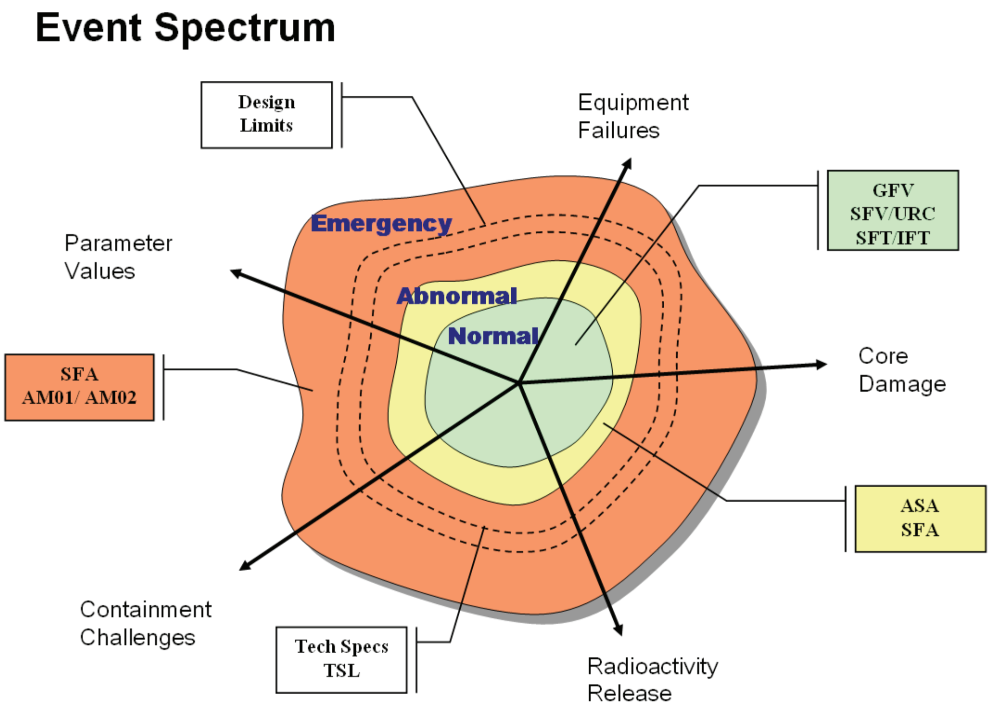


Figure 2 Event Spectrum

3.1. SAMG History and Regulatory Requirements for the Leibstadt NPP

1990	Introduction of symptom oriented SFA (EOP) based on the BWROG EPG Rev. 4 Accident management measures are integrated in the SFA
1998	HSK (ENSI) requires the introduction of (at-power) SAMG in all Swiss NPPs
2000	HSK defines the Requirements (HSK-AN-3674) for the development and introduction of SAMG
2001	Implementation of BWROG generic EPG/SAG Rev. 2. The EPGs/SAGs are divided into Emergency Procedure Guidelines and Severe Accident Guidelines. The document forms the bases for the development of the Leibstadt SAMG.
2004	Introduction of the new Leibstadt SAMG
2005	Enactment of the new Nuclear Energy Law in Switzerland. The ordinance requires written decision guidance for severe accident management as a basic technical document.
2008	HSK requests the development of Leibstadt specific Shutdown-SAMG (SSAMG)
2009	Development of new Shutdown SAMG
2009	Enactment of the ENSI regulatory guideline ENSI-B12 “Emergency Response”. The guideline defines the requirements for SAMG including the coverage of all relevant operational modes.
2010	Update of the existing (At-Power) SAMG

Table 2 SAMG History

4. PSA

In 1991 HSK required the utilities to develop plant-specific low power and shutdown PSAs. The plant specific PSAs include internal events as well as external events such as fires, flooding, earthquakes, aircraft impacts and high winds.

The Leibstadt analysis deals with all expected operating conditions not covered by the At-Power studies (power reduction, hot/cold shutdown, refueling, start-up and power increase to full power).

The objectives of the Shutdown PSA are as follows:

a) Understand and quantify the risk during shutdown in a manner which allows a comparison with the risk during normal operation.

b) Enable KKL to adjust operating and maintenance strategies to ensure an even risk profile throughout an operating cycle (At-Power and shutdown) which contains no unforeseen or unplanned risk peaks.

c) Understand and quantify shutdown-specific risks which may not be relevant for a and b but which are important for determining and reducing absolute risk during shutdown.

Event Type	Description	CDF At-Power	FDF Shutdown	Overall CDF
Internal Events	All LOCA Events	1.04E-07	3.34E-08	1.37E-07
	Transients and special initiators	3.26E-07	5.92E-09	3.32E-07
	Total:	**4.30E-07**	**3.94E-08**	**4.69E-07**
External Events	Earthquakes	2.14E-06	3.01E-07	2.44E-06
	High winds and tornadoes	6.47E-08	1.21E-08	7.68E-08
	Airplane crash	1.34E-08	5.66E-10	1.40E-08
	Weir failure	3.21E-14	1.96E-13	2.28E-13
	Total:	**2.22E-06**	**3.14E-07**	**2.53E-06**
Area Events	Fire	7.59E-07	4.07E-07	1.17E-06
	Flood	5.02E-07	5.71E-07	1.07E-06
	Turbine Missile	-	-	-
	Total:	**1.26E-06**	**9.78E-07**	**2.24E-06**
	Grand Total:	**3.91E-06**	**1.33E-06**	**5.24E-06**

Table 3 Overview Leibstadt PSA Results

The overall Core Damage Frequency (CDF) is estimated to be 5.24E-06 per year. Of this total, approximately 25% is contributed at reduced load or shutdown.

The main contribution to the Core Damage Frequency during At-Power operation is caused by earthquakes (55%) and internal fires (19%). At reduced load or shutdown the main contributors to the Fuel Damage Frequency (FDF) are flooding events (43%) and internal fires (31%).

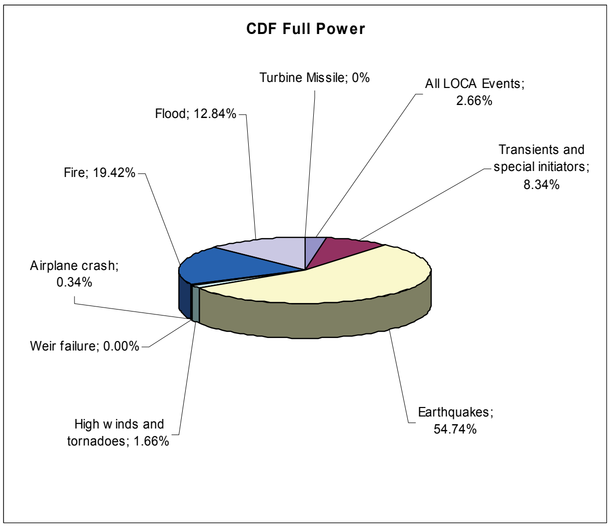


Figure 3 CDF At-Power

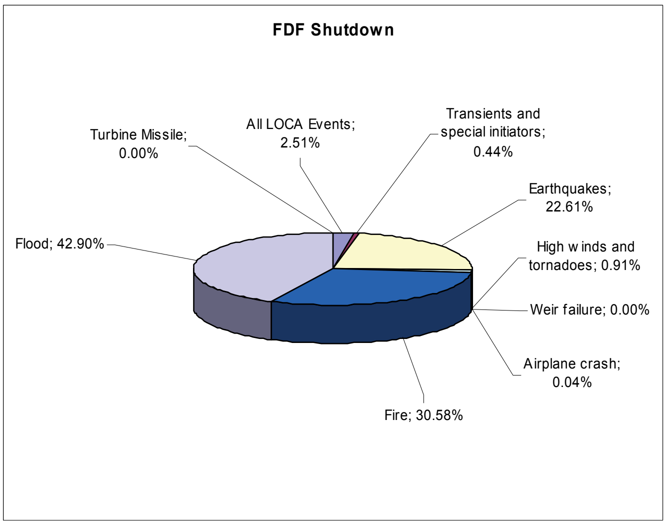


Figure 4 FDF Shutdown

5. SSAMG Development Strategy

The process of developing new SSAMG procedures is based on several individual steps.

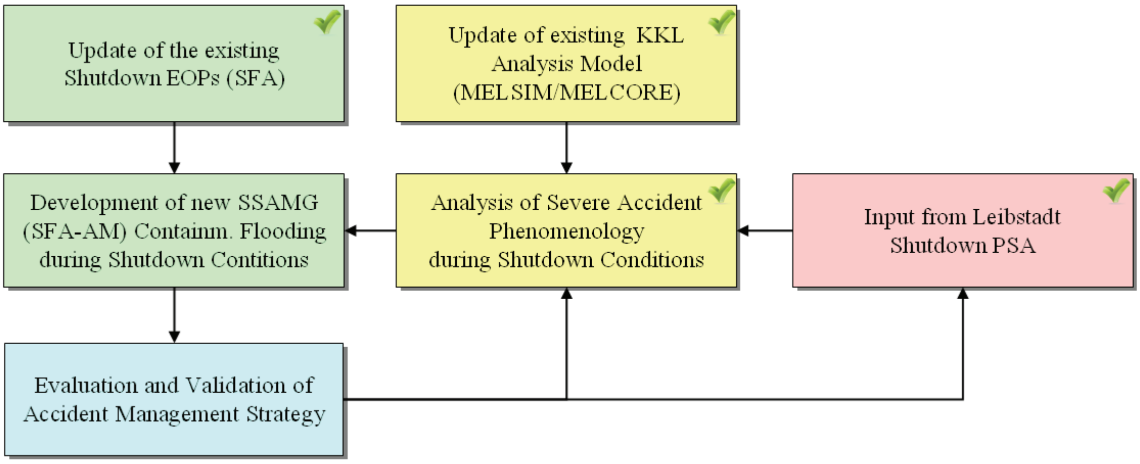
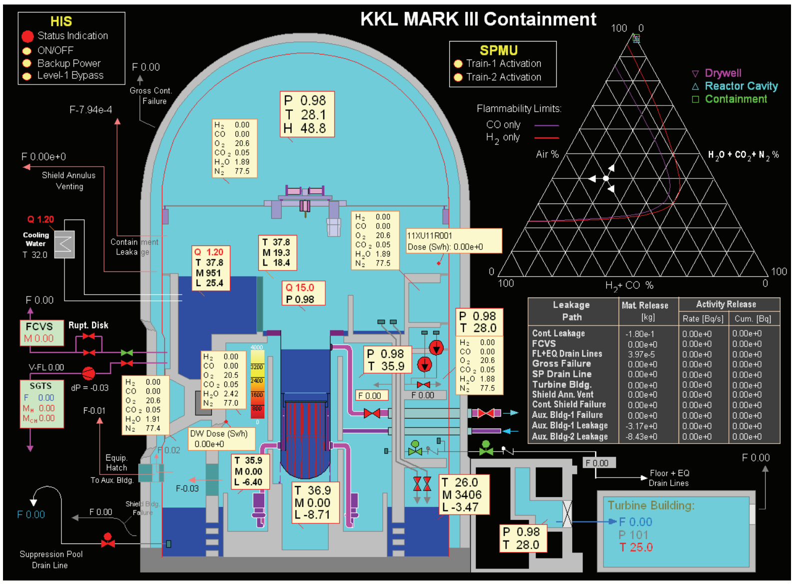


Figure 5 Overview SSAMG Development Strategies

5.1. Update of the KKL Analysis Model

In order to support PSA Level 2 analysis, an interactive plant specific simulation model, based on the MELCOR severe accident codes, was developed. The simulation environment is called MELSIM and offers the possibility to gain plant specific insights on the progression and phenomenology of severe accident core melt scenarios. The model has been used to define and validate effective accident management measures. To validate the existing full-power SAMG decision diagrams, Leibstadt's symptom oriented accident procedures were integrated into the model. The feature provides both a manual and automatic mode. In automatic mode the simulator initiates the requested actions based on plant parameters (symptom) and assumed operator response times.



The model has been updated to analyse shutdown scenarios up to the point in time where spent fuel is unloaded from the core, with the containment equipment hatch, drywell hatch, drywell head, and reactor vessel head either on or off, and with the upper vessel internals either in or out. Accident scenarios with the core partially or fully unloaded can be simulated up to the point of fuel uncovery with a user specified decay heat feature and a user specified spent fuel pool heat load feature. To allow the progression calculation of radio nuclides in an open drywell and containment scenario, the room model had to be expanded by the simulation of the secondary containment, the reactor auxiliary building and the turbine building.

5.2. Input from Leibstadt NPP Shutdown PSA

Based on the results of the Shutdown PSA, eight shutdown sequences were defined. It is believed that the corrective actions for the chosen scenarios will envelop all sequences covered by the Leibstadt NPP Shutdown PSA.

5.3. Analysis of Leibstadt NPP Shutdown Scenarios

The accident scenarios initiated during refueling shutdown were analysed using the new MELCOR 1.8.6 based KKL Shutdown Model. The purpose of the analyses is to evaluate the behaviour and timing of the selected sequences in order to determine the time available for corrective actions.

5.4. Update of the existing Shutdown EOPs (SFA)

Prior to the development of the new Shutdown SAMG the already existing Shutdown EOPs were revised and optimized. While the objective of the EOPs is to prevent a potential severe accident condition, the objective of Shutdown SAMGs is to mitigate core melting and the effects of a vessel break through.

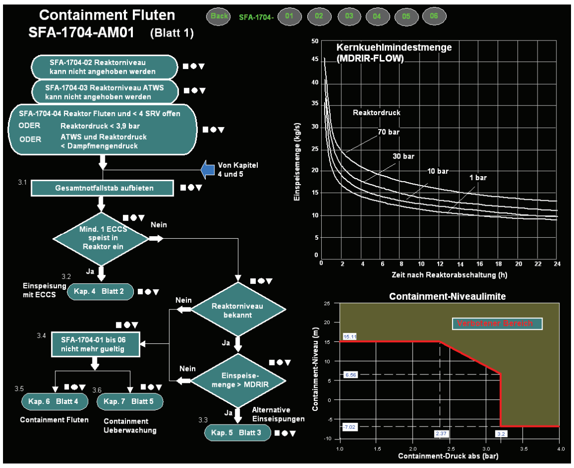
Shutdown EOPs (SFA)
(prevention)

- Loss of Shutdown Cooling (RHR/SEHR during Shutdown
- Loss of Secondary Containment during Shutdown
- Loss of Power Supply during Shutdown
- Loss of Coolant during Shutdown
- Inclined fuel Transfer Tube - Stuck Fuel Bundle
- Loss of Fuel Pool Cooling

Shutdown SAMG (SFA-AM)
(mitigation)

- Containment Flooding during Refueling and Shutdown Conditions

5.5. Verification and Validation of the Leibstadt NPP Shutdown SAMG



As for the At-Power SAMG, the effectiveness of the accident management measures defined in the Shutdown SAMG will be verified using the new MELSIM shutdown model. The new decision diagrams will be integrated. The accident management strategy shall be tested in manual and automatic mode. In automatic mode the simulator initiates the requested actions based on plant parameters (symptoms) and plausible operator response times. The latter will allow an unattended run and a later analysis of a large number of possible scenarios and run times. If necessary the chosen strategy shall be optimised.

Following a successful verification, the introduced accident mitigation measures will be validated with the Leibstadt Shutdown PSA.

6. Insights gained from the Shutdown Scenario Analysis

6.1. Summary of Results

The purpose of the analyses is to evaluate the behaviour and timing of the selected sequences in order to determine the time available for corrective actions. Therefore, the cases are analysed under the extremely conservative assumptions with no injection available and the loss of all AC power (Station Blackout). No corrective operator actions are considered.

The individual scenarios are listed below:

- ☞ Station Blackout, reactor subcritical, reactor vessel head closed, containment/drywell open, residual heat removed using the RHR in shutdown-cooling mode and pressure less than 9.3 bar:

Scenario 1	Loss of RHR by the Station Blackout. The shutdown line is manually isolated when the RCS pressure reaches 9.5 bar.
Scenario 2	Loss of RHR cooling caused by the Station Blackout. The shutdown line is not isolated.
Scenario 3	Station Blackout combined with a break in the common section of the RHR shutdown line. The shutdown line is manually isolated 30 min after transient initiation.
Scenario 4	Station Blackout combined with a break in the common section of the RHR shutdown line. The shutdown line is not isolated.

- ☞ Station Blackout, reactor subcritical; reactor vessel head removed, containment/drywell open:

Scenario 5	Loss of RHR cooling caused by the Station Blackout. RCS temperature is 60 °C.
Scenario 6	Station Blackout combined with a leakage through a removed control rod drive. The reactor well is flooded and the fuel gate is removed.
Scenario 7	Station Blackout combined with a leakage through a removed control rod drive. The RV level is at the Steam Lines and the fuel gate is installed.
Scenario 8	Station Blackout combined with leakage through the removed RWCU valve 10TC11S001.

Table 4 shows a summery of the timing of the key core damage related events for all eight shutdown scenarios.

Scenario	1	2	3	4	5	6	7	8
Entry frequency (appr.) [1/calendar year]	1E-6	1E-7	<1E-8	<1E-8	1E-6	<1E-8*	<1E-8*	<1E-8*
	Scenario Time [hour]							
Core Uncovery (TAF)	8.25	3.79	1	0.77	14.29	12.5	1.64	1.3
Gap release	9.63	4.8	2.29	1.93	15.77	13.2	2.36	1.64
Start Release to the Environment	10.3	5.32	2.96	2.08	16.45	14.2	3.82	2.44
Vessel Breach	19.54	16.2	15.66	9.68	38.68	20.6	7.45	18.1
Start of Core Concrete Interaction	20.42	16.42	16.76	9.74	38.72	20.8	8.26	18.8
* Combined with station blackout or loss of ECCS								

Table 4 Summary of Analysis Results

The initiating scenarios are summarized below:

Scenario 1 The loss of RHR cooling leads to re-heating of the reactor vessel coolant inventory and consequent re-pressurization of the RCS. The boiled off coolant is released to the suppression pool via the SRVs.

Scenario 2 The loss of RHR cooling leads to re-heating of the reactor vessel coolant inventory and consequent re-pressurisation of the RCS. The pressure increase is limited to about 36 bar by the RHR Relief Valves (RVs), discharging to the suppression pool. The coolant inventory is lost faster than in a case releasing via the SRVs, because the RHR RVs are connected to the reactor vessel through the RHR shutdown line and the recirculation loop, and thus they are first discharging water.

Scenario 3 The break of the RHR shutdown line leads to a quick loss of the RCS coolant inventory. The in-shroud water level is down to the level of the jet pump throats in less than 5 minutes, with the downcomer and the recirculation loops empty. After the line isolation, the RCS reheats and re-pressurises and the boiled off coolant is discharged to the suppression pool via the SRVs.

Scenario 4 The break of the RHR shutdown line leads to a quick loss of the RCS coolant inventory. The in-shroud water level decreases to the level of the jet pump throats in less than 5 minutes, with the downcomer and the recirculation loops empty. Because the break is not isolated, the RCS pressure remains low and the boiled off coolant continues to be released via the broken shutdown line to the Pipe Tunnel (containment bypass).

Scenario 5 The loss of RHR cooling leads to re-heating of the RCS coolant inventory and consequent boil off. The inventory is boiled off at low pressure (~1 bar) and the steam is released to the containment via the removed reactor vessel head. With no injection, the core uncovers and heats up, resulting in severe core degradation with debris relocation to the lower plenum, vessel breach and Core-Concrete Interactions (CCI).

Scenario 6	The leakage through the removed Control Rod Drive drains a large part of the upper pool water inventory and the RCS coolant inventory into the reactor cavity/drywell. The available water inventory is discharged in about 14 hours, leaving the reactor core and the lower plenum dry (in-shroud water level ~1.2 m), while the downcomer and the recirculation loops are filled up to the level of the jet pump throats. The loss of the Spent Fuel Pool inventory above the bottom of the refueling gate leads to an early boil off and uncovery of the spent fuel.
Scenario 7	The leakage through the removed Control Rod Drive drains the RCS coolant inventory into the reactor cavity/drywell. The available water inventory is discharged in about 2.7 hours, leaving the reactor core and the lower plenum dry (in-shroud water level ~1.2 m), while the downcomer and the recirculation loops are filled up to the level of the jet pump throats.
Scenario 8	The leakage through the open RWCU line drains the RCS coolant inventory into the reactor cavity/drywell. The available water inventory is discharged in about 45 minutes, leaving the reactor core flooded up the top of the active fuel, while the downcomer and the recirculation loops are mostly empty.

The resulting release path is common to all scenarios. The containment pressure remains low, because the Equipment Hatch is open to the annulus and secondary containment.

The secondary containment fails early after the first core damage, from an overpressure induced by the hydrogen burning in the containment. The blow-out panels between the ECCS rooms and the pipe tunnel, between the pipe tunnel and the turbine Building and between the turbine building and the ruptured windows to the Environment fail open from the gradual pressurization of the ECCS rooms, creating a sizeable release path from the annulus to the environment. This path is the major vent path to the environment.

The radionuclide release to the environment is high for noble gases and moderately high for volatile radionuclide groups as well as for other radionuclide groups with significant release from the fuel or CCI debris.

6.2. Discussion

General

All analyzed cases lead to severe core damage, with relocation of debris to the lower plenum, progressing to vessel breach and ejection of debris to the Reactor Cavity. This behaviour is expected because no makeup or emergency injection systems are in operation due to the total loss of power and no operator actions such as the potential line-up of geodetic injection systems or portable fire pumps are considered.

Timing

The time to core uncovery (refer to Table 4) varies considerably from case to case. These differences are caused mainly by different coolant inventories available in the reactor vessel for the boil-off. Cases with the short core uncovery times have a significant loss of the coolant inventory due to a break or leak. The time between the core uncovery and the gap release (first core damage) is more consistent. The two cases with the shortest times are those where the in-shroud coolant inventory is drained via the removed CRD and thus the core is already dry from the beginning of the heatup. The time between the core uncovery and the vessel breach also shows some considerable variation. Again the cases with removed CRD have the shortest times, for the same reason as stated above.

RCS Behaviour

The RCS pressure remains low in all cases with RV open to the Containment and for the unisolated break of the RHR shutdown line. In the other cases the RCS pressure increases up to the opening pressure of the SRVs and for the unisolated break of the RHR shutdown line up to the opening pressure of the RHR Relief Valves.

Containment Behaviour

The containment pressure remains low, because the Equipment Hatch is open to the Annulus and to the Secondary Containment, and a sizeable venting path from the Annulus to the environment opens early during the Containment pressurization via the ECCS rooms, Pipe Tunnel and Turbine Building. Another venting path, the Secondary Containment failure, opens later from pressurization caused by the hydrogen burning.

Hydrogen Burning

Combustion of the hydrogen and CO occurs in the rooms of the Containment, the Annulus and the Secondary Containment (mostly ECCS rooms and the Pipe Tunnel). The burning location as well as the amount of hydrogen/CO burned varies from case to case.

Radionuclide Release

The radionuclide release to the environment is large for the Noble Gases and for the aerosols as well. Small release fractions are seen only for those radio nuclides, which have not been released from the fuel or CCI debris in large quantities.

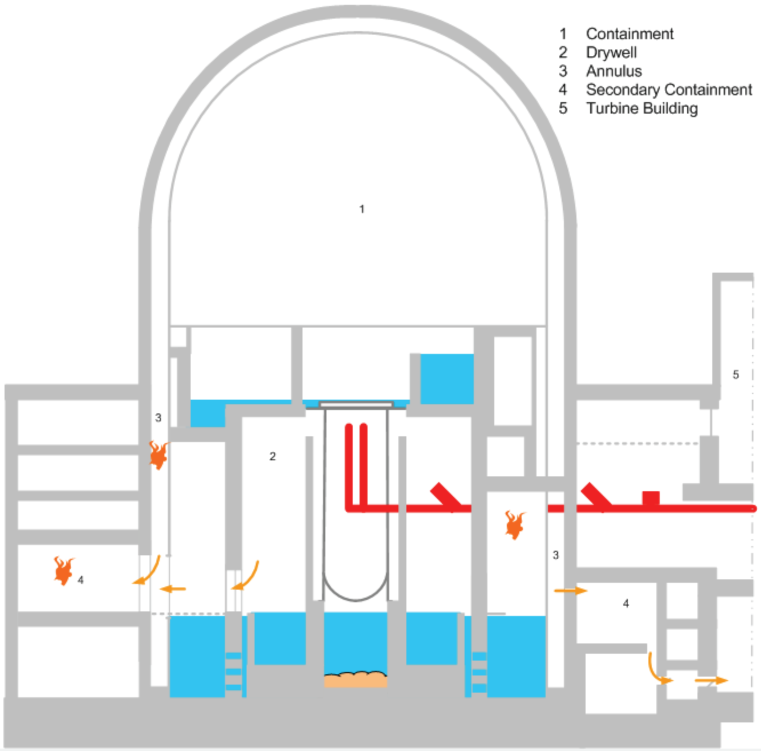


Figure 6 Containment Release Path

7. Development of BWR Severe Accident Management Guidance (SSAMG)

7.1. Scope of SSAMG

SSAMG need to cover a wide range of plant configurations during shutdown conditions, defined mainly by the status of the barrier integrity of containment, drywell and reactor pressure vessel. As long as the integrity of all barriers remains intact, the At-Power SAMG applies.

An exception is caused by the Drywell Leak Rate test after closure of the Drywell Equipment Hatch. While performing the test, instrument air inside containment and drywell is isolated, preventing a set point controlled opening of the SRV in case of pressurisation of the reactor pressure vessel due to loss of the RHR system.

State	Operating Modes	Reactor Temp.	Reactor Cavity flooded	RPV Integrity	Drywell Integrity	Contain-ment Integrity	Fuel above TAF	Gates Open
A	4	< 93 °C	F	intact	Leak rate test	intact	--	yes
B	4	< 93 °C	--	intact	intact	open	--	no
C	4	< 93 °C	--	intact	open	open	--	no
D	5	< 60 °C	--	open	open	open	--	no
E	5	< 60 °C	F	open	open	open	--	no
F	5	< 60 °C	F	open	open	open	yes/no	yes

Table 5 Plant Conditions covered by SSAMG

As long as the preventive actions defined in the Shutdown EOP's are successful, fuel will not be jeopardized. As the conditions degrade SSAMG need to be entered. Depending on the integrity of the RPV and Containment, different decisions have to be made.

As the RPV head is removed (states D, E, F), enough coolant has to be injected with ECCS or alternate injection systems to maintain the debris submerged by maintaining the RPV water level above the bottom of the active fuel (BAF) or maintaining the RPV injection rate greater than the Minimum Debris Retention Injection Rate (MDRIR). As long as the total injection rate is greater than MDRIR, the decay heat can be removed continuously and the core debris can be retained in the vessel. The RPV head is removed approximately 80% of the time between shutdown and ready for start-up.

If the RPV head is closed (states A, B, C) the vessel pressurizes as RHR fails. Inventory is lost by opening of the SRVs blowing steam into the suppression pool. To inject with low pressure ECCS or alternate injection systems the RPV needs to be depressurised under consideration of the Pressure Suppression Pressure (PSP) and the SRV Tail Pipe Level Limit. In this case containment flooding, provided the Containment Equipment Hatch is closed, has to be limited to the weir wall height or consequently containment pressure or reactor pressure have to be decreased.

7.2. BWR Containment Flooding Strategy

As during power operation, containment flooding remains the basic strategy to cope with core melting. The objectives of primary containment flooding are consequently identical.

1) Re-establish core cooling
2) Remove heat from the RPV
3) Retain core debris in the RPV
4) Quench debris outside the RPV
5) Preserve containment integrity
6) Scrub fission products
7) Minimize core-concrete interaction
8) Facilitate long-term recovery

The following objectives are shutdown specific and highly dependent on the status of the containment barriers.

9) Re-establish containment integrity to prevent an early release path due to the failed secondary containment
10) Restore SRV instrument air supply
11) Restore Filtered Containment Venting System (FCVS)

7.3. SSAMG Overview

7.3.1. Event Handling Strategy

The analysed scenarios are reduced to two key event paths:

1. Loss of Shutdown Cooling during shutdown conditions
   (initiating events: loss of power supply, component failure, human error)
2. Loss off RPV Coolant inventory during shutdown conditions due to inadvertent draining
   (initiating events: human error, component failure)

A concurrent Loss of RPV Coolant and Station Blackout are not considered.

In both cases the corresponding EOPs are entered based on entry conditions. If the event cannot be handled within the scope of the EOPs the SSAMG are entered trough defined entry conditions (refer to figure 7).

The transient will initially be handled by the control room staff based on the applicable shutdown EOPs. Due to a loss of core cooling or loss of reactor coolant the emergency state will be declared.

The responsibility moves to the emergency organization.

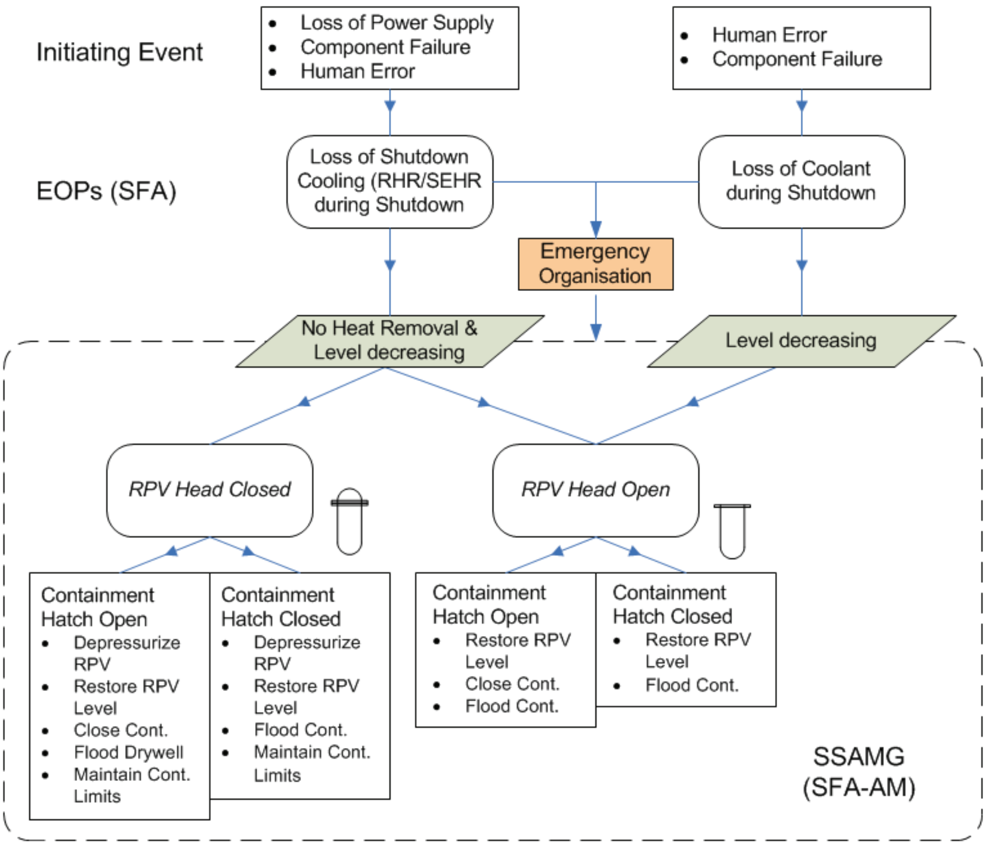


Figure 7 Event Handling Strategy

7.3.2. Containment Recovery Strategy

Is the RPV head closed?

no

- ☞ Close Containment Equipment Hatch
- ☞ Verify secondary containment integrity
- ☞ Establish primary containment integrity to the extend possible
  1. close at least one (inboard or outboard) isolation valve
  2. close test valves between isolation Valves

yes

- ☞ Close Drywell Equipment Hatch (without shield blocs)
- ☞ Close Containment Equipment Hatch
- ☞ Verify secondary containment integrity (all doors closed)
- ☞ Establish primary containment integrity to the extend possible
  1. close at least one (inboard or outboard) isolation valve
  2. close test valves between isolation Valves
- ☞ Establish drywell integrity
  1. close at least one (inboard or outboard)isolation valve

7.3.3. Fundamental SSAMG Logic

Has core debris breached the RPV?

yes →

- Pressure suppression no longer required
- Flood drywell/containment at least to the Minimum Debris Submergence Level (between 4 feet above floor or top of weir wall)
- Limit containment water level to -32 cm if containment hatch is not installed

Priorities:

1. Maximize RPV injection from outside containment (Containment Hatch closed)
2. Maximize Containment injection from external sources (Containment Hatch closed)
3. Maximize RPV injection from suppression pool (Containment Hatch open)

no ↓

Can RPV water level be restored and maintained above top of active fuel (TAF)?

yes →

- Restore and maintain RPV water level > TAF
- Use external sources and in-shroud injection only if required
- Limit containment water level to -32 cm if Containment Hatch is not installed

no ↓

Can RPV water level be restored and maintained above bottom of active fuel (BAF)?

yes →

Debris expected to remain in RPV

- Restore and maintain water level > BAF

Priorities:

1. Operate core spray
2. Maximize injection of external sources

- Re-establish containment integrity
- Restore essential systems
- Flood drywell to the Minimum Debris Submergence Level (above the top of the weir wall)
- Limit containment water level to -32 cm if containment hatch is not installed

no ↓

Can RPV injection be restored and maintained above the Minimum Debris Retention Injection Rate (MDRIR)?

yes →

Debris expected to remain in RPV

- Restore and maintain water level > MRDIR
- Maximise injection of external sources to the RPV
- Re-establish containment integrity
- Restore essential systems
- Flood drywell to the Minimum Debris Submergence Level (above the top of the weir wall)

no ↓

Containment is inside Pressure Suppression capability?	yes	Debris may melt through RPV ☞ Restore and maintain injection flow > MDRIR ☞ Maximise injection into the RPV from external sources ☞ Maintain pressure suppression by RPV venting ☞ Limit containment level to top of weir wall
no		
Containment is outside Pressure Suppression capability?	yes	Debris may melt through RPV Containment / secondary Containment may fail ☞ Maximise total RPV injection ☞ Maximise containment injection if RPV injection will not be reduce

8. Shutdown specific Challenges

The requirements for safety systems and equipment necessary for the safe operation of the plant in Shutdown mode are defined by the Technical Specifications Leibstadt (TSL). In addition the TSL is supplemented by the KKL guideline VO/262 „Planning Guideline for Refueling and Maintenance under the Consideration of Plant Safety“. The latter document requires an even more restrictive system configuration. It defines the systems that may be taken out of standby/operational mode for a given period of time in order to conduct required maintenance.

Nevertheless certain systems or equipment required under severe accident considerations (outside of the design bases) will be not available throughout the shutdown period. The systems or equipment need to be identified in advance and special provisions have to be made for a timely restoration.

8.1. Drywell/Containment Equipment Hatch

Because of the open Annulus and Containment Equipment Hatch the Secondary Containment fails due to hydrogen burning, causing a sizeable release path to the environment. This emphasizes the importance of an early closure of at least the Containment Equipment Hatch.

The closure of the Containment and Drywell Equipment Hatch is of vital importance for the containment flooding strategy. Each hatch takes two hours to close under normal conditions.
Considering the available time frame between start of a transient and the beginning of fuel failure, the decision to restore the containment integrity has to be made early.

With an open Containment Equipment Hatch the suppression pool level cannot be raised to the weir wall elevation (refer to Figures 8 and 10). For initial drywell flooding, water needs to be injected inside the weir wall.

The following SSAMG actions are required:

☞ Define the entry condition for containment closure

☞ Line up a mobile power supply in case of a Station Blackout to close the hatches

☞ Line-up water directly into drywell flooding up to the Minimum Debris Submergence Level (above the top of the weir wall)

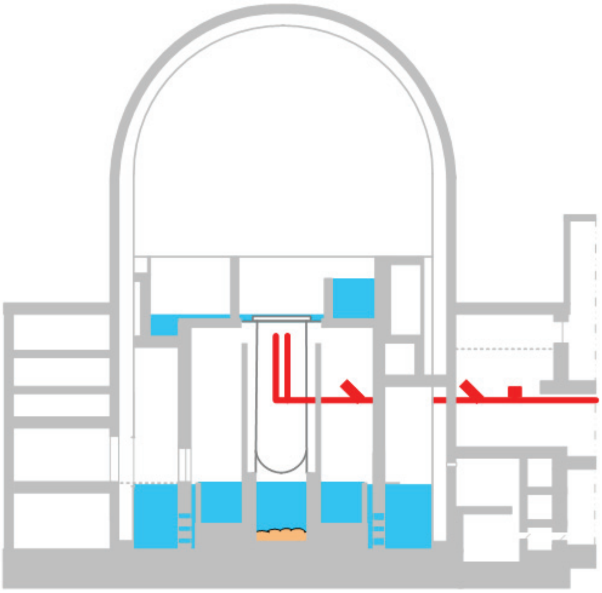


Figure 8 Flooding with open Cont. Hatch

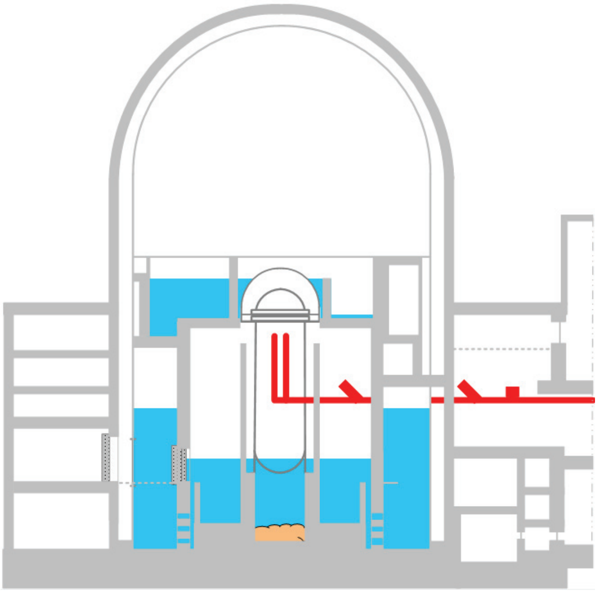


Figure 9 Flooding with closed Cont. Hatch

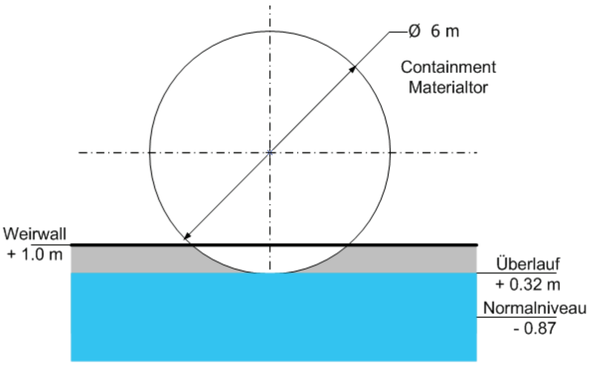


Figure 10 Suppression Pool Level Restriction

Because of the open containment equipment hatch during refueling, the suppression pool level cannot be raised up the top of the weir wall unless that the water overflows into the annulus and finally into the secondary containment.

Consequently the drywell cavity has to be flooded by injecting water to the inside of the weir wall to establish the minimum Debris Submergence Level (MDSL)

8.2. Restore unavailable Systems

Depending on the current maintenance schedule and operational readiness, unavailable systems need to be restored. These may include instrument air, ECCS, Suppression Pool Cooling, Filtered Containment Venting System, electrical power supply, feed water system, main condenser, etc.

9. Conclusion

The shutdown specific risks are identified based on the Leibstadt Shutdown PSA. With the use of the plant specific MELSIM/MELCOR simulator a set of accident scenarios has been analysed to gain a better understanding and the necessary insights for the development of Shutdown SAMG.

As during power operation containment flooding remains the key strategy to master severe accident progressions. The necessary mitigation measures however need to be adjusted to the status of the RPV and containment barriers.

Design Modifications of the Mochovce Units 3 & 4 Dedicated to Mitigation of Severe Accident Consequences, Providing Conditions for Effective SAM

Milan CVAN
VUJE, Inc., Slovakia

Dušan Šiko
SE, a.s., Slovakia

1. Introduction

Initiation of activities dedicated to enhancement of Slovak nuclear units regarding severe accident mitigation is dated to around 2005. At that time, there were four units of VVER440/V213 in operation in Slovakia, operated by Slovenské elektrárne (SE, a.s.). Additionally two other units (Mochovce units 3 and 4), being already in advanced state of construction, were candidates for completion in near future. Initiated by the NPP operator, couple of projects started with focus onto mitigation of severe accident consequences, up to development of draft SAMGs (Severe Accident Management Guidelines).

In the area of design modifications, the main purpose and aim of those activities was focused onto creation of a technological basis for introduction of draft SAMGs for units in operation. But, after decision to complete Mochovce 3 and 4 units (MO34), specification of the relevant design modifications of these units was the first priority. This paper presents and summarizes outputs of this activity.

Based on summarization of requirements and suggestions of IAEA (International Atomic Energy Agency) and EUR (European Utility Requirements), detailed identification of deficiencies of the original design was the initial step. Initial set of potential equipment candidates for modifications or for extension of existing systems was proposed. Through consideration of both impact of realization of individual modifications to design (costs, technical solutions, feasibility in the existing status of unit construction) and of its contribution to mitigation of consequences of severe accidents, the final set of measures to be included into the design of the units has been selected.

Major design modifications are: complex of measures to flood the reactor cavity for external reactor vessel cooling and extended hydrogen recombiner and igniter system. To cope with specific threats some additional measures were also included, e.g. dedicated depressurization system of primary circuit, additional external source of coolant and monitoring system for severe accident control.

The MO34 units with the described design upgrade shall be put into operation in 2012/2013. Within the completion phase, the full scope SAMGs will be developed, tested and included into basic set of operation procedures. With support of the available technical measures, the SAM shall provide effective mitigation and control of severe accidents even for past century design of VVER440/V213 units.

2. Main deficiencies of the MO34 original design regarding mitigation of severe accident consequences

Identification of main candidates for design modification of the MO34 units was based on summarization of all available analytical results (both probabilistic and deterministic) and understanding of initiators, process evolution and plant response to severe accidents. Relevant suggestions and requirements of IAEA, EUR and national regulator authority were taken into account.

Measures to cope with ex-vessel phase of severe accidents were not included into the considerations. There were two main reasons for this limitation. One reason followed from assumption that a sufficiently effective prevention of reactor pressure vessel failure will be available, the second one reflected the fact that the already completed structures of MO34 units (reactor cavity, containment) limit installation of corresponding large scope modifications (as e.g. core catcher).

The identified candidates for modifications were evaluated from the point of view of possible technical solution, required functionality and feasibility. Each area had been split into several parts to allow preliminary assessment of diverse aspects, as contribution to limitation of large early releases of radionuclides, significance in the complex of dedicated measures, status and availability of technical solution, feasibility in specific conditions of MO34 units, interference with other safety features (no or negligible interference with design safety) and legislation requirements at and after the start up of the units. Also, preliminary assessment of expected price of each modification had been derived. Integral evaluation resulted in initial proposal of the set of modifications, including also specification of the basic design of the measures.

During the initial phase of MO34 units completion project, all involved parties (mainly technical support organization VUJE (Slovakia), design company EGP (Czech republic) and MO34 owner and operator) reconsidered all aspects of the modifications dedicated to mitigation of severe accident consequences. This resulted into agreed final set of design measures to be incorporated into the basic design, as outlined in the following section.

3. Modifications incorporated into the basic design of MO34 units

Final set of modifications for extension of design basis of the MO34 into the severe accident mitigation consists of several interdependent solutions. As it consists of relatively large number of measures, they are only listed below, in groups in accordance with the grouping in [2].

Compared to the initial set of areas for consideration (mentioned in section 2 above), there were no specific measures included into the basic design to cope with containment bypass (this shall be covered by procedural measures or later by some technical solutions). Also, no dedicated system is included for removal of fission products from containment atmosphere or filtered venting system, as these features would be mostly redundant to existing spray system functions and would result in increase of the cost, with potential to interfere with existing systems.

3.1 Management of containment atmosphere

<u>Group of measures to manage hydrogen concentration inside containment</u>

Addition of the system for monitoring of the containment atmosphere composition in selected rooms (monitoring of hydrogen concentrations).

Installation of passive autocatalytic recombiners with severe accident capacity.

Installation of igniters.

Vacuum breaker (addition of system for containment deep subpressure prevention)

Modification of existing pipelines leading from the air traps (addition of branches before closing valves).

Installation of electrically locked flaps, controlled by absolute pressure or subpressure inside containment, electrically charged from the first category of the emergency grade electric power supply and included into the ESFAS structure (engineered safety features actuation system).

Modification of the pipeline and flaps to prevent damage of neighbouring structures and systems by dynamic forces if the system is initiated.

3.2 In-vessel retention of corium

Modification of shielding at the bottom of the reactor pressure vessel

Enlargement of the gap between RPV (reactor pressure vessel) wall and bottom shielding structures.

Creation of a central opening in the shielding, which will be equipped with buoyancy driven system to mediate passive opening, derived from increasing coolant level inside reactor cavity, and locking of the device in open position.

Relocation of openings (used for RPV examination) in the shielding to preserve original function of the openings.

Reinforcement of the shielding to provide sufficient firmness in the mode of flooded cavity and long term external cooling.

Modification of the manipulation platform, to provide sufficiently free access of coolant to the RPV wall.

Addition of filtration grid constructions at the inlet of coolant into the reactor cavity.

Modification of penetrations leading into the reactor cavity, to prevent permanent loss of coolant from the flooded cavity and to enable access of hydraulic control lines of a special valve at the inlet into the cavity drain line.

Modification of the cavity access door, to reach sufficient resistance against pressure, thermal and radiation loads in the cavity, to prevent permanent coolant loss.

Provision of sufficient coolant inventory and circulation of coolant in the channel along the RPV wall

Modifications of the drain system of the bubble tower trays to enable their drain down to steam generator boxes floor.

Creation of inlet opening for coolant into the existing ventilation system pipeline below the floor of the connecting corridor and installation of closing valve, including control, monitoring and filtration of impurities.

Installation of U-tube (siphon) to ventilation system pipelines to prevent loss of coolant into the ventilation system room at both legs of the ventilation system (both corridors).

Partial reconstruction of the structures around the reactor pressure vessel nozzles, to minimize pressure losses by flow of steam from the reactor cavity into the steam generator boxes, possibly with installation of overpressure flaps and optimization of components in the area.

Modification of the drain line from the reactor cavity (closing device)

Addition of new closing valve inside the reactor cavity at the inlet into the drain line, with a system of its (hydraulic) opening from the neighbouring room. The valve will be permanently closed during normal operation.

3.3 Management of open reactor severe accidents

Addition of delivery pump for supply of coolant into the spent fuel pool or into the open reactor applicable during severe accident.

Installation of delivery pipeline from tanks into the pipeline of the spent fuel pool and into the low pressure ECCS (emergency core cooling system) system.

Provision of sufficient boric acid solution inventory for this purpose (see External sources of coolant below).

Installation of necessary pipelines, valves and control of the devices.

3.4 External sources of coolant

Installation of a system of three tanks, together with all necessary auxiliary systems for mixing of the solution (boric acid + water), heating, draining and operation.

Installation of appropriate pipelines from the tanks outside of the containment and connections into the low pressure ECCS system, spray system and into the pipeline of the spent fuel pool cooling.

Addition of corresponding valves and their control.

3.5 Additional measures for mitigation of evolution and consequences of severe accidents

<u>Controlled depressurization of primary circuit during severe accident</u>

Construction of additional branch (leading from existing pipeline to steam generator box) of existing pipeline from pressurizer into the steam generator boxes.

Installation of two closing valves at the above-mentioned pipeline, with measurement of pressure between the valves, as well as a drain system of the section. The pipeline should be designed in such a way to prevent heat up of the valves prior required action of the valves.

<u>Ultimate heat sink (long term heat removal from containment)</u>

Modifications limited to procedures for revisions and operative maintenance of the pumps during severe accidents to enable permanent operation of the system.

<u>Electricity supply for the systems for severe accidents mitigation</u>

All systems for severe accident mitigation connected to non-emergency sources.

Corresponding sections of this source were modified in order to provide reliable source of energy for these selected systems, especially in severe accident conditions.

The non-emergency source has been extended with additional diesel generator, which will cover in case of a need (of total loss of power supply) all power supply of relevant equipment.

Additionally the most important systems are powered from backup sources (accumulators) of the DC power.

Locking device of the vacuum breaker is powered also from emergency source of the first category of electric power supply, as this system is safety important also from the point of view of design basis accidents.

Pumps of the system for emergency delivery of coolant from the new external source of coolant are connected directly to the dedicated diesel generator.

3.6 Monitoring of parameters needed for control of severe accidents

Requalification (replacement) of original temperature measurement at core outlet.

Requalification (replacement) of original pressure sensors inside reactor pressure vessel.

New measurement of coolant level inside reactor cavity.

New measurement of coolant level inside steam generator boxes.

Replacement of original steam generator boxes pressure (containment) monitoring system.

Replacement of containment temperature sensors and measurement chains.

New measurements of hydrogen concentration at different rooms of containment.

New measurement of pressure inside individual air traps.

New measurement of atmosphere temperature inside individual air traps.

Requalification (replacement) of original pressure sensors of pressure difference between primary and secondary circuit.

Installation of radioactivity sensors throughout the containment.

Modification of monitoring system of the coolant level inside steam generators.

Modification of monitoring system of feed water flow into the steam generators.

Modification of monitoring system of the pressure inside hydroaccumulators.

4. PROPOSED APPLICATION OF THE SYSTEMS, DEDICATED TO SEVERE ACCIDENT MITIGATION

The above listed modifications of the original design of MO34 units are dedicated to mitigation of severe accidents, to be included into the SAMGs (with the exception of open reactor coolant delivery, see below). The SAMGs for the MO34 units are still to be developed. Nevertheless as a part of basic design, initial concept of SAMG approach with utilization of new features had been assumed.

Most actions and characteristics of the dedicated systems are based on passive components, being designed to affect exclusively evolution of severe accidents. Nevertheless there are some active interventions of the operator required. By relevant operation guidelines and procedures it shall be prevented to initiate the system before such a state of the unit is reached, in which it is obvious that the accident tends to evolve into severe one. Transition phase from EOPs (emergency operating procedures) to SAMGs is expected to be based on measured temperature above the core, with the setpoint specified later.

In case of station blackout by both closed and open reactor, there is clear indication of unavailability of the electrical supply already during earlier phases of accident. Therefore at the earliest, possibly before severe accident conditions establish, the dedicated diesel generator shall be put into operation, to supply pumps of the external sources of coolant. For this reason, this additional diesel generator, whose fundamental function is mitigative, can be also employed in the EOPs as a measure for prevention of core damage in cases, not interfering with its mitigative function.

Having the power supply for the external coolant source pump, the delivery into the spray pipeline shall be initiated. Flooding of the pipeline (if needed) will be done before unit enters severe accident. This shall be conditioned by brief evaluation of conditions in the containment (pressure, content of hydrogen, status of normal containment spray system) and coupled with symptoms of a specific type of accident.

Another potential operator manual action is to drain down the bubble tower trays. Although it is possible to drain all the trays in most cases just at the beginning of SAMG actions, modification of drain down scheme may be applied (drain down of only limited number of trays), based on the e.g. coolant level in steam generator boxes, on the delivery of coolant from external source, etc.

Basic active intervention of operator is opening of the valves at the inlets into the ventilation lines for flooding the reactor cavity. In a case of fast evolving scenarios, it will follow immediately after opening of the drain lines of the bubble tower trays. Once the coolant level in steam generator box reaches overflow level (after opening of the valves at the inlet) it starts to flood the cavity passively. Increasing level of coolant in reactor cavity causes passive opening of the inlet opening in lower shielding of the reactor bottom. This results in full opening of the cooling loop for the external cooling of the reactor pressure vessel.

Hydrogen will be controlled by passive components. It will be continuously recombined (oxidized in recombiners). In some of containment rooms, where accumulation of hydrogen exceeds recombination capacity and its concentration reaches ignition levels, hydrogen combustion is initiated by igniters to combust hydrogen before detonation level is reached.

Prior or parallel to the above actions, based on indication of high pressure in the primary circuit, operator (being unsuccessful in depressurization using means of the EOPs) opens the depressurization line, dedicated to severe accident.

For the next at least 12 hours, if there is no operator action to control the accident evolution, the characteristics of the proposed dedicated systems shall maintain heat removal from the core (via external cooling) to prevent reactor pressure vessel failure. It is assumed that it will be necessary to control operation of the spray system to create favourable conditions for hydrogen combustion initiation by igniters and for management of containment pressure to minimize radionuclide releases. The capacity of the external source of coolant shall cover the needs of this period for most of scenarios without exceeding acceptance criteria of containment failure and fission product releases.

Passing 12 hours from the beginning of the severe accident phase of the accident, the containment spray system is assumed to overtake the role of heat removal from the containment. It reduces pressure in containment to subpressure and keeps it in continuous operation.

5. Summary

The modifications of the original design of the Mochovce 3 and 4 units, dedicated to severe accident mitigation will provide sound basis for effective SAMGs. The modifications were proposed and included into the basic design of the units. Their extent and composition were affected by several limitations as existing structures and buildings, economical views, complexity of phenomena, original design basis etc. Nevertheless they shall present important contribution to enhancement of safety of these units and to promote their acceptance for the entire lifetime.

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Session 8

Experimental Investigation of Melt Debris Agglomeration with High Melting Temperature Simulant Materials

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1. Introduction

Reactor cavity flooding is the cornerstone of severe accident management strategy adopted in Swedish and Finnish BWRs in case of a hypothetical severe accident with core melting and reactor vessel melt through. It is assumed that the melt ejected into a deep water pool will fragment and form a coolable porous debris bed. If coolability of the debris bed can not be provided then corium debris will reheat, remelt and attack containment base-mat, threatening plant's containment integrity. Coolability of the debris bed depends on its properties. Agglomerated debris present considerable obstacle for a coolant flow and thus may negatively affect coolability of the debris bed. Although agglomeration of debris and formation of "cake" were observed in previous fuel-coolant interaction (FCI) experiments with prototypic corium mixtures (e.g. in FARO[1] and CCM[2] tests) and with corium simulant materials (e.g. in DEFOR-E[3] and DEFOR-S[4] tests), there is a lack of systematic data and understanding of the governing physical phenomena.

Present work is a part of the DEFOR (Debris Bed Formation) research program[5,6,7] initiated at the Division of Nuclear Power Safety (NPS) Royal Institute of Technology (KTH). The aim of the DEFOR program is understanding and quantification of phenomena that govern formation of the debris bed in different scenarios of corium melt release into a deep water pool. Results of the previous DEFOR-S experimental campaign suggest that porosity of the debris bed formed in the process of melt-coolant interaction is 50-70% which is much higher than previously assumed 30-40%. Debris agglomeration and even "cake" formation were observed in the DEFOR-S tests when subcooling of water was lower than 30˚C and water pool depth was not sufficient to provide complete solidification of the molten material. Debris agglomerates were observed as "soldered" together groups of particles (Figure 1a). "Cake" (Figure 1b) is formed when water depth is smaller than jet breakup length and there is enough liquid melt to glue together most of solid particles formed in the upper part of the jet.

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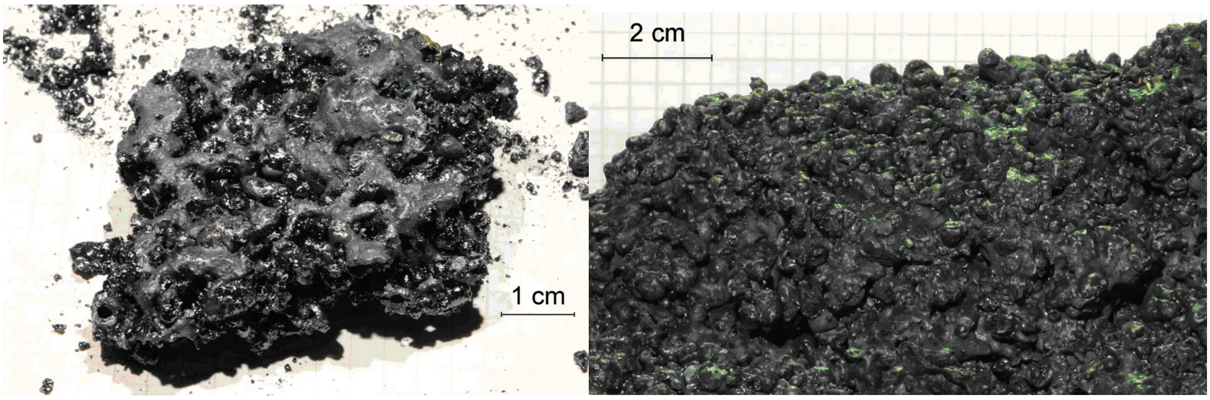
The goal of the present work is systematic experimental study of the debris agglomeration in the process of melt jet pouring into a water pool under well-defined conditions. The paper disuses experimental results of the first series of DEFOR-A (Debris Bed Formation and Agglomeration) tests. Specifically, the aim is to provide quantitative data about the influences of pool depth, water subcooling, melt jet diameter, jet free fall height and initial melt superheat on the mass fraction of agglomerated debris and occurrence of different modes of agglomeration (cake formation and agglomeration of the debris). Such data is necessary for development of new models and validation of simulation codes[8].



a) b)

Figure 1. Agglomeration modes: a) Fragile agglomerates (DEFOR-S5); b) "Cake" (DEFOR-S8)

2. DEFOR-A Experimental Installation

The DEFOR-A tests are performed in the DEFOR facility[4] which is composed of a 45kW medium-frequency (up to 30 kHZ) Induction Furnace (IF) for melt generation, a melt delivery funnel, and a coolant tank with glass windows for visual imaging of transient melt-coolant mixing and debris formation[4] (Figure 2). The simulant-material melt is generated in a SiC crucible. The liquid melt is delivered to the funnel by tilting the crucible. The delivery funnel is conical with a replaceable discharge nozzle up to 20 mm in diameter. The test section is an open to the atmosphere of the lab rectangular tank (2 m tall, with cross section 0.5x0.5 m). Distance from the bottom of the test section to the nozzle outlet is 1.7 m (Figure 2).

Several DEFOR-S tests as well as several material tests with small amount of melt were performed[4] to identify the best corium simulant material for experiments on the debris bed formation. Among the tested materials were WO_3-CaO (DEFOR-E7)[3], MnO-TiO_2, WO_3-TiO_2, Bi_2O_3-TiO_2, Bi_2O_3-CaO, and Bi_2O_3-WO_3. The mixture Bi_2O_3-WO_3 was favored because of its high density and exhibition of a broad range of fragmentation behavior reported in the previous FCI experiments with ceramic melts including corium. In addition, this material is easy to work with over a wide range of chemical compositions for which the melting temperature remains accessible by the inductive heating technology and SiC crucible used in the DEFOR facility[4].

In the DEFOR-A experiment up to 3 liters of melt simulant materials are poured in a 1.5 m deep water pool. Debris catchers were installed at different elevations (Table 1) in the test section to collect debris and agglomerates during the melt pouring process. Analysis of the debris collected in the catchers provides data about mass fraction of agglomerates as a function of water pool depth at given conditions of a test.

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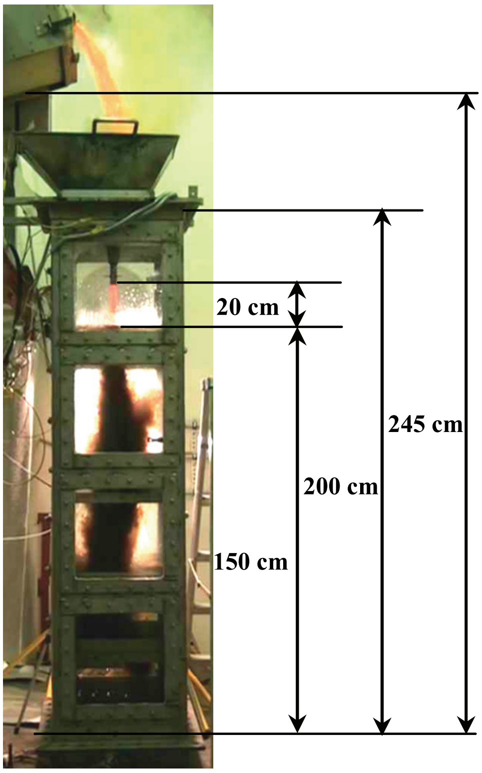


Figure 2. Test section of the DEFOR-A experimental facility

Table 1. DEFOR-A Debris catchers.

Catcher	Depth measured from water surface, m	Elevation from the pool bottom, m
Catcher-1	0.6	0.9
Catcher-2	0.9	0.6
Catcher-3	1.2	0.3
Catcher-4	1.5	0.0

A set of K-type thermocouples was used for the measurements of the transient temperatures in the pool and inside the debris bed during experiments. Ten thermocouples are installed at the bottom plate, 5 thermocouples are installed in the pool at elevations 1.0 m, 0.9 m, 0.8 m, 0.7 m, 0.6 m from the bottom of the test section. Six thermocouples are installed in the Catchers 1-3 (2 thermocouples in each catcher). Temperature of the crucible is monitored during the melt preparation by K-type thermocouples (for experiments with melting temperature lower than 1200˚C) or by B-type thermocouples (for higher melting temperature melts). Melt initial temperature is measured during the delivery by a K-type or B-type thermocouple positioned directly in the nozzle.

3. Test Conditions and Experimental Results

In the present work we discuss results of three DEFOR-A tests namely A2, A5 and A6. Conditions of the tests are presented in Table 2. Eutectic mixture of Bi_2O_3–WO_3 was used in all tests. Influence of water subcooling was studied in the range from 2˚C to 27˚C. Nozzle diameters used in the test are 10mm, 12mm and 20mm. Melt superheat was varied from 100˚C to 136˚C. Other parameters were kept the same in all experiments (Table 2). Observed increase of water temperature during melt pouring is 5˚C.

Photo images of the debris beds obtained in DEFOR-A2, A5 and A6 are presented in Figure 3, Figure 4 and Figure 5 respectively. Melt material is distributed more or less equally between catchers (Figure 3, Figure 4, Figure 5) giving quite enough material for measurements of mass averaged quantities such as mass fraction of agglomerates on each catcher. Spatial distribution of the debris depends on water subcooling and jet diameter. In DEFOR-A2 tests with low subcooling and bigger jet particles are spread over the catchers more uniformly than in DEFOR-A6 test with smaller jet and higher subcooling of water (Figure 3 and Figure 5).

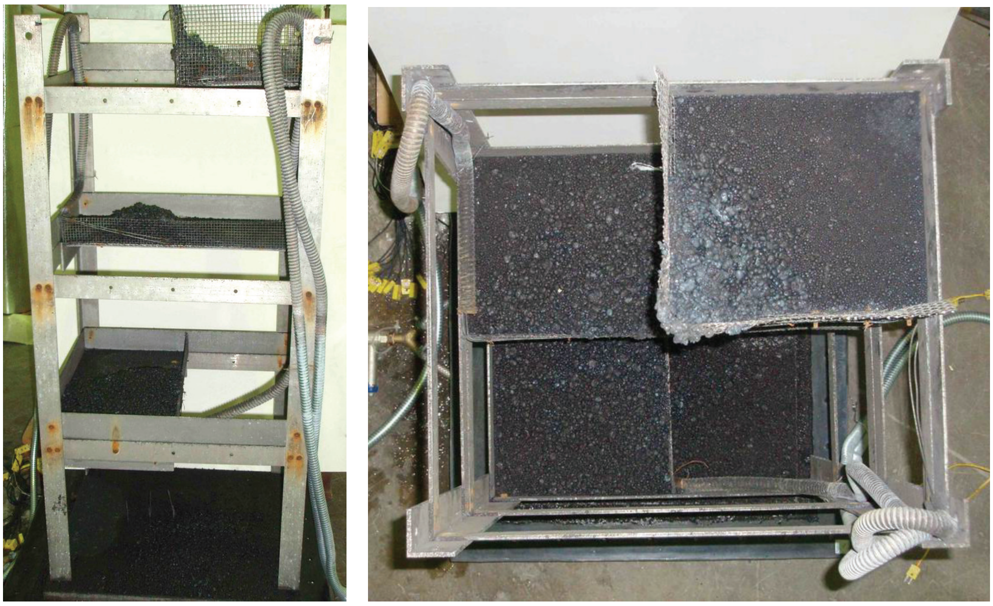
Table 2. DEFOR-A test conditions

N	Parameter	A2	A5	A6
1	Component 1	Bi_2O_3	Bi_2O_3	Bi_2O_3
2	Component 2	WO_3	WO_3	WO_3
3	Component 1 molar fraction, %	27%	27%	27%
4	Component 2 molar fraction, %	73%	73%	73%
5	Eutectic mixture	Yes	Yes	Yes
6	Density of the mixture, kg/m^3	7811	7811	7811
7	Melt volume, liter	3.0	3.0	3.0
8	Melt mass, kg	23.43	23.43	23.43
9	Melting temperature of the melt, ˚C	870	870	870
10	Maximum temperature in the funnel, ˚C	973	972	1006
11	Water temperature before melt pouring, ˚C	94	91	73
12	Water temperature after melt pouring, ˚C	98	96	78
13	Water pool depth, m	1.52	1.52	1.52
14	Jet free fall height, m	0.18	0.18	0.18
15	Jet diameter, mm	20	10	12
16	Maximum melt pool depth in the funnel, m	0.15	0.15	0.15

a)

b)

Figure 3. Debris catchers with debris in DEFOR-A2 test:
a) side view; b) view from the top (bottom Catcher-4 is not shown)



a) b)

Figure 4. Debris catchers with debris in DEFOR-A5 test:
a) side view; b) view from the top

a) b)

Figure 5. Debris catchers with debris in DEFOR-A6 test:
a) side view; b) view from the top

Debris cakes obtained in Catcher-1 in DEFOR-A2, A5 and A6 are presented in Figure 6, Figure 8 and Figure 9 respectively. The cakes are different in size, morphology and location in the bed. In DEFOR-A2 and A5 tests the cakes were smaller and located at the bottom of the bed, covered on top with a layer of fragmented debris particles. In DEFOR-A6 the cake is considerably bigger and sitting on top of a thin layer of fragmented debris.

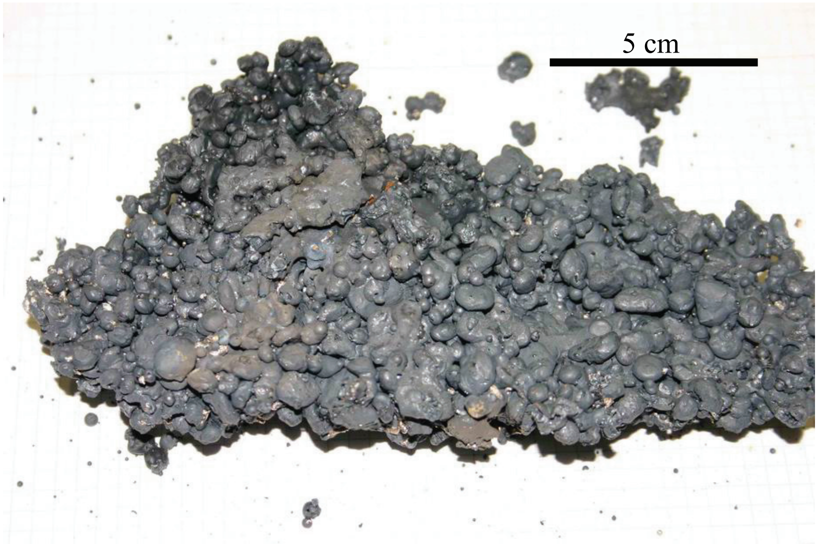


Figure 6. DEFOR-A2 test: cake in Catcher-1

Figure 7. DEFOR-A5 test: Top layer of fragmented debris in Catcher-1

Figure 8. DEFOR-A5 test: cake in Catcher-1

Figure 9. DEFOR-A6 test: cake in Catcher-1

Figure 10 summarizes measured dependency of mass fraction of agglomerates on water depth obtained in DEFOR-A2, A5, A6, DEFOR-S5, S8 and S10 tests. Data presented in Figure 10 suggests that fraction of agglomerated debris decreases rapidly as the depth of the coolant increases. Debris collected at Catcher 4 (1.5 m deep), are completely fragmented in all DEFOR-A experiments. It worth mentioning that data on fraction of agglomerated debris from the DEFOR-A tests agrees well with the data from the DEFOR-S experiments (Figure 10) where smaller amount of melt (~1 liter) was used.

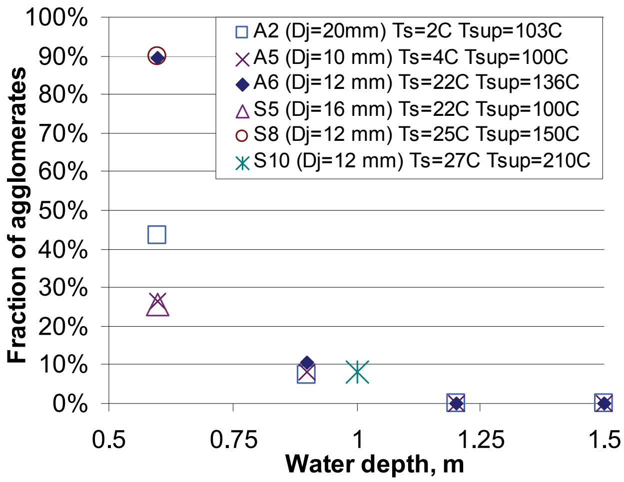


Figure 10. Dependence of agglomeration fraction on water pool depth

Despite considerable variation of mass fraction of agglomerates (mostly cakes) observed in Catcher-1 (0.6 m deep) in different experiments some common trends in the influence of water subcooling, jet diameter and melt superheat can be identified. First of all, in two experiments (DEFOR-A6 and DEFOR-S8) with similar conditions, namely relatively small jet (12 mm), moderate subcooling of water (25˚C - 27˚C), and melt superheat around 136˚C - 150˚C, the highest mass fraction of agglomerates is obtained. In the tests with lower subcooling of water (DEFOR-A2, A5) and with bigger jets (DEFOR-A2 and DEFOR-S5) mass fraction of agglomerates is smaller than in DEFOR-A6 and DEFOR-S8. Some preliminary hypotheses about the mechanisms which are responsible for such behavior of the agglomeration are presented in[8]. Namely analysis performed in[8] suggests that steam production (which decreases with reduction of jet diameter and with increasing of water subcooling) may considerably affect mass fraction of agglomerates. This phenomenon has to be further investigated in experiment and analysis. The question about the role of melt superheat is also to be studied in new series of experiments.

Cumulative size distribution of completely fragmented debris obtained in DEFOR-A2 experiment in Catcher 4 is presented in Figure 11. Particle size distribution is important for code validation purposes[8]. Rather small deviations from the size distribution presented in Figure 11 were observed in the other DEFOR-A experiments. Data about particle size distributions obtained in the FARO test[1] are also presented for comparison in Figure 11. Overall there is a good agreement between particle size distributions observed in the FARO and in the DEFOR-A experiments. That confirms that Bi_2O_3–WO_3, as a simulant material, reasonably well represents corium fragmentation behavior.

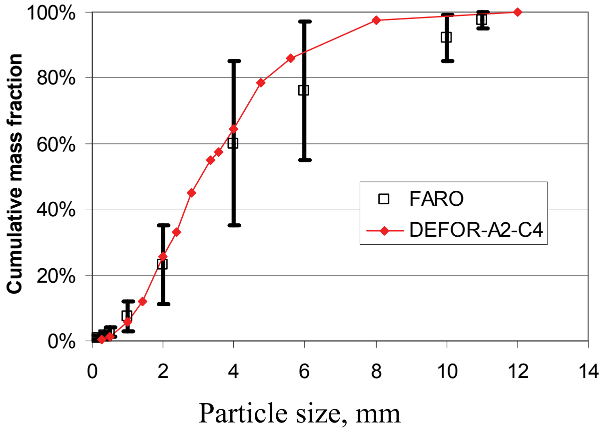


Figure 11. Particle size distribution (red solid line) measured in DEFOR-A experiment and in FARO[1]

5. Conclusions

First of a kind systematic experimental data on the mass fraction of agglomerated debris as a function of water pool depth was obtained in the DEFOR-A experiments with 3 liters of high melting temperature simulant materials. Observed particle size distribution is in a good agreement with the data from the FARO fuel-coolant interaction experiments with corium, which confirms that the simulant material well represents corium fragmentation behavior.

Increase of water temperature during melt pouring was 5˚C. Also data on fraction of agglomerated debris from the DEFOR-A tests agrees well with the data from the DEFOR-S experiments with smaller amount of melt (~1 liter). Thus DEFOR-A data is valuable for a separate effect code validation and model development.

Main finding of the DEFOR-A tests is that fraction of agglomerated debris decreases rapidly as the depth of the coolant is increasing. Debris collected in Catcher-4 (1.5m deep) in all DEFOR-A experiments are completely fragmented. The highest mass fractions of agglomerates were obtained in experiments with relatively small jets at moderate water subcooling and with high melt superheat. Further investigation of the physical mechanisms which are responsible for such behavior of agglomerated debris is necessary. Preliminary analysis[8] suggest that steam production rate may significantly affect fraction of agglomerated debris.

Acknowledgement

This work is supported by the Swedish Nuclear Radiation Protection Authority (SSM), Swedish Power Companies, European Commission (SARNET), Nordic Nuclear Safety Program (NKS), Swiss Federal Nuclear Safety Inspectorate (ENSI) under the APRI- MSWI program at the Royal Institute of Technology (KTH), Stockholm, Sweden.

Approach to Prediction of Melt Debris Agglomeration Modes in a LWR Severe Accident

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1. Introduction

Severe accident management strategy adopted in Swedish and Finish BWRs relies on reactor cavity flooding as a means for termination of ex-vessel accident progression. It is assumed that the melt ejected into a deep water pool will fragment quench and form a debris bed heated by decay heat and cooled by natural circulation. If the debris bed is not coolable then debris will reheat, remelt and attack the containment base-mat threatening plant's containment integrity.

Debris agglomeration and "cake" formation may lead to degradation of the debris bed coolability margin. Until recently, debris agglomeration and cake formation have not been studied systematically, although cake has been observed in a number of experiments with prototypical corium mixtures (e.g. in FARO[1] and CCM[2] tests), as well as in test with corium simulant materials.

There are significant epistemic uncertainties in physical phenomena and lack of mechanistic models for prediction of cake formation and debris agglomeration. In our previous work[3] we proposed concept of "agglomeration mode map" which defines domains of different agglomeration modes in the space of severe accident scenario parameters. Thermo-physical state of the debris immediately before deposition on top of the debris bed (pre-deposition state) is considered as the initial conditions for onset of different agglomeration modes. High sensitivity of pre-deposition state of the debris to the parameters of melt-coolant interaction and especially to jet breakup mode was identified in[3]. Epistemic uncertainty in pre-deposition state of corium debris due to the influence of different modes of jet breakup was addressed with conservative-mechanistic approach, while mass fraction of agglomerated debris was not estimated in the previous work[3].

The goal of the present paper is development of conservative-mechanistic approach for prediction of mass fraction of agglomerated debris and its application to prediction of ex-vessel debris bed formation in prototypical plant accident conditions. The VAPEX-P[4,5,9] code is used in the present

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work as computational vehicle for mechanistic simulations of the molten fuel-coolant interaction (FCI) phenomena. Present work is a part of the DEFOR (Debris Bed Formation) research program[3,6,7,8,9,10,11] initiated at the Division of Nuclear Power Safety (NPS) Royal Institute of Technology (KTH). The aim of the DEFOR program is understanding and quantification of phenomena that govern formation of the debris bed in different scenarios of corium melt release into a deep water pool. First series of the DEFOR-A[11] (Agglomeration) experiments has been performed in order to study systematically influence of melt jet diameter, melt superheat, water pool depth and subcooling on mass fraction of agglomerated debris. In the DEFOR-A experiment up to 3 liters of high density, high melting temperature oxides mixture simulating corium were poured in a test section filled with water. Four debris catchers were installed at different depths in the test section. In each DEFOR-A experiment dependency of mass fraction of agglomerated debris on water pool depth was obtained at well defined conditions, thus eliminating possible uncertainties due to variations in test conditions.

In the present paper we use DEFOR-A experimental data in order to provide necessary information for a semi-empirical closure for prediction of mass fraction of agglomerated debris and then for demonstration of conservatism in the developed models. Application of the developed approach for plant conditions is also discussed in the paper.

2. Approach to Estimation of Agglomerated Debris Fraction

In the previous work[3] we introduced three different characteristic modes of agglomeration according to their potential impact on coolability:

1) No agglomeration. The bed consists of completely fragmented debris.

2) Agglomeration. The bed consists of debris which are partially agglomerated and connected by fragile inter-particle bonds.

3) Cake. The bed represents a chunk of solidified melt. No debris particles are distinguishable. No open porosity on the outer surface of the bed for coolant ingression into the cake interior.

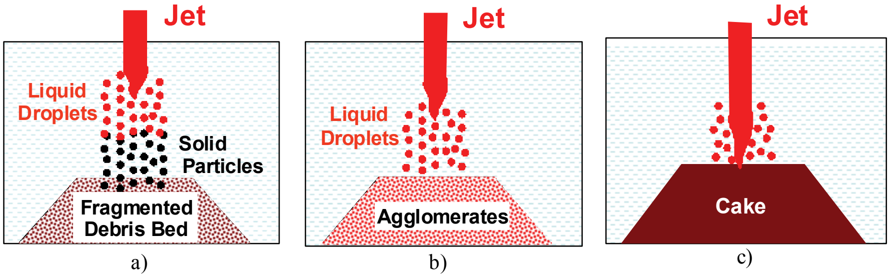

Figure 1. Modes of debris agglomeration: a) no agglomeration; b) agglomeration; c) cake.

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We used conservative assumptions about onset of different debris agglomeration modes. Namely it is assumed that agglomeration modes are defined by the pre-deposition state of the debris as it is shown in the Figure 1. No-agglomeration or completely fragmented debris bed (first mode, Figure 1a) occurs when debris particles solidify completely before deposition on the bed. Partial agglomeration (second mode, Figure 1b) occurs when some fraction of the debris may reach the top of the debris bed in liquid state. A cake is formed when jet breakup length is bigger than depth of water pool (third mode, Figure 1c).

In the present work we propose an approach for prediction of the mass fraction of agglomerated debris. We assume that agglomeration is a result of physical processes which occur at particle scale when liquid droplets interact with each other and with neighboring solid particles immediately after their deposition on top of the debris bed. The approach is based on hypothesis that jet breakup related phenomena do not affect agglomeration related phenomena in any other way except providing "initial conditions" for agglomeration. Namely, melt fragmentation and solidification define relative crust thickness distribution for the melt particles as initial conditions for the onset of debris agglomeration and mass fraction of agglomerated debris is completely defined by these initial conditions (pre-deposition state of the debris). We further assume that mass fraction of agglomerates is proportional to the total mass fraction of completely liquid droplets and thin-crust particles (we will name it fraction of "glue" or "liquid" particles)

$$m_{aggl} = \alpha \cdot m_{liq} \tag{1}$$

where m_{aggl} is mass fraction of agglomerated debris, m_{liq} is mass fraction of liquid particles, $\alpha = f\left(m_{liq}\right)$ is coefficient of agglomeration, which also can be a function of mass fraction of liquid particles.

Formula (1) is simple and has pretty obvious interpretation: agglomerates are produced by "gluing" of liquid particles with each other and with neighboring solid particles. Unfortunately it is not possible to obtain pure empirical closure for the coefficient of agglomeration α. While it is easy to measure m_{aggl} after the test, it is very difficult, if at all possible, to measure during the experiment pre-deposition state of the debris in terms of crust thickness distribution to determine the fraction of liquid particles m_{liq}. On the other hand, even if such measurements would be feasible, one can expect significant variations of values of agglomeration coefficient α due to various sources of uncertainties which are in plenty both in experiment and in plant accident scenario. Therefore instead of attempting to provide best estimation for the coefficient of agglomeration α we aim to make a conservative-mechanistic assessment for it. The goal is to ensure conservative but still physically reasonable prediction of fraction of agglomerated debris with taking into account intrinsic epistemic uncertainties of agglomeration phenomena. The approach to conservative-mechanistic estimation for the coefficient of agglomeration is based on combined use of mechanistic simulation tool VAPEX and experimental data from DEFOR-A tests. In the next sections the approach is discussed in detail.

3. Sensitivity Study and Influence of Water Subcooling and Jet Diameter

First of all, we recognize and respect uncertainties in the code prediction results and in the experimental data. Therefore we start with study of liquid particles fraction sensitivity to different options in the code modeling and to different test conditions. The goal of such study is to identify "bounding scenarios" which give lowest and highest mass fraction of the liquid particles necessary to provide an estimation for α. We also use sensitivity study to explain phenomena observed in the DEFOR-A experiment.

In the simulations we use computational domain and conditions as in the DEFOR-A experimental facility[11] (water pool depth 1.5 m, cross section 0.25 m^2, atmospheric pressure above water level). Particle crust thickness distributions were calculated at the same depths at which debris catchers were installed in the DEFOR-A test section (Catcher 1: 0.6 m; Catcher 2: 0.9 m; Catcher 3: 1.2 m; Catcher 4: 1.5 m). On the base of preliminary calculations we selected jet diameter and water subcooling as most influential parameters for the fraction of the liquid particles. Four baseline cases were selected for simulations to cover diapasons of DEFOR-A experimental conditions:

Table 1. Baseline cases for sensitivity study

Case	Melt jet diameter	Coolant state
Case-1	Dj=10 mm	Subcooling
Case-2	Dj=10 mm	Saturation
Case-3	Dj=20 mm	Subcooling
Case-4	Dj=20 mm	Saturation

Calculated baseline cases give us useful insights into dynamics of the debris solidification in the test section during melt poring. Namely it was found that water subcooling significantly affects steam production rate, which, in turn, has significant effect on particle sedimentation and solidification dynamics. Accurate prediction of steam production due to FCI and condensation in subcooled water pool is a challenging problem for numerical methods and is a big source of uncertainty in best estimate codes. Therefore we used bounding approach for the modeling of cases with subcooled water. Namely it was assumed that all generated steam condenses immediately and all heat released by the melt is spent only on heating up the water. Such assumption can be thought of as the limiting case of very big subcooling.

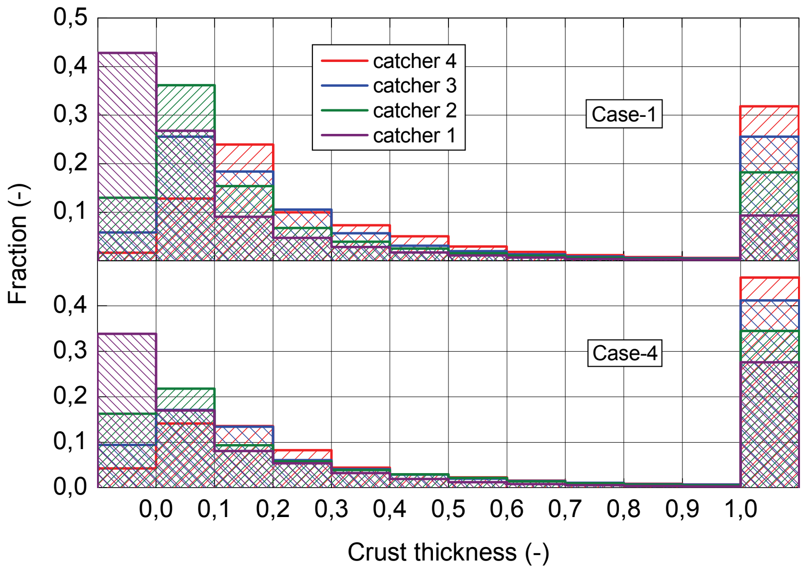


Figure 2. Mass fraction distribution of particles with different relative crust thicknesses.
Catcher 1: 0.6 m; Catcher 2: 0.9 m; Catcher 3: 1.2 m; Catcher 4: 1.5 m.
Baseline Case-1 and Case-4

In the process of sensitivity study it was found that mass fraction of liquid and thin-crust particles ($\delta_{rel} = \delta_{crust} / R_{drop} \le 0.1$, where δ_{crust} – absolute crust thickness, R_{drop} – radius of the particle) in Case-1 with subcooled coolant and 10 mm melt jet diameter is bigger than in Case-4 with saturated water and melt jet diameter 20 mm (Figure 2). This observation, which may seem surprising at first glance, is also confirmed by the DEFOR-A data[11] on the debris agglomeration. A physical explanation for such phenomenon can be found in Figure 3 where volume fraction and mass averaged velocities of the debris are presented.

Intensive heat transfer from relatively big melt to water in Case-4 with saturated coolant leads to very intensive steam generation and appearance of strong upward stream of water-steam mixture. Such upward flow is slowing melt droplet sedimentation and can even push the droplets upward (see vectors of particle velocities in Figure 3). As a result, droplets of the melt levitate for longer time in coolant and have better chance to be cooled down and solidified before deposition on top of the debris bed. Similar effect of steam upward flow rate on fraction of liquid particle at pre-deposition state was identified in our previous work[3] in case of plant scale problem.

In case of subcooled water and relatively small jet (Case-1) there is no significant steam production and no considerable upward motion of the coolant. Thus melt droplets are falling down much faster and are much hotter at the time when they are reaching top of the debris bed.

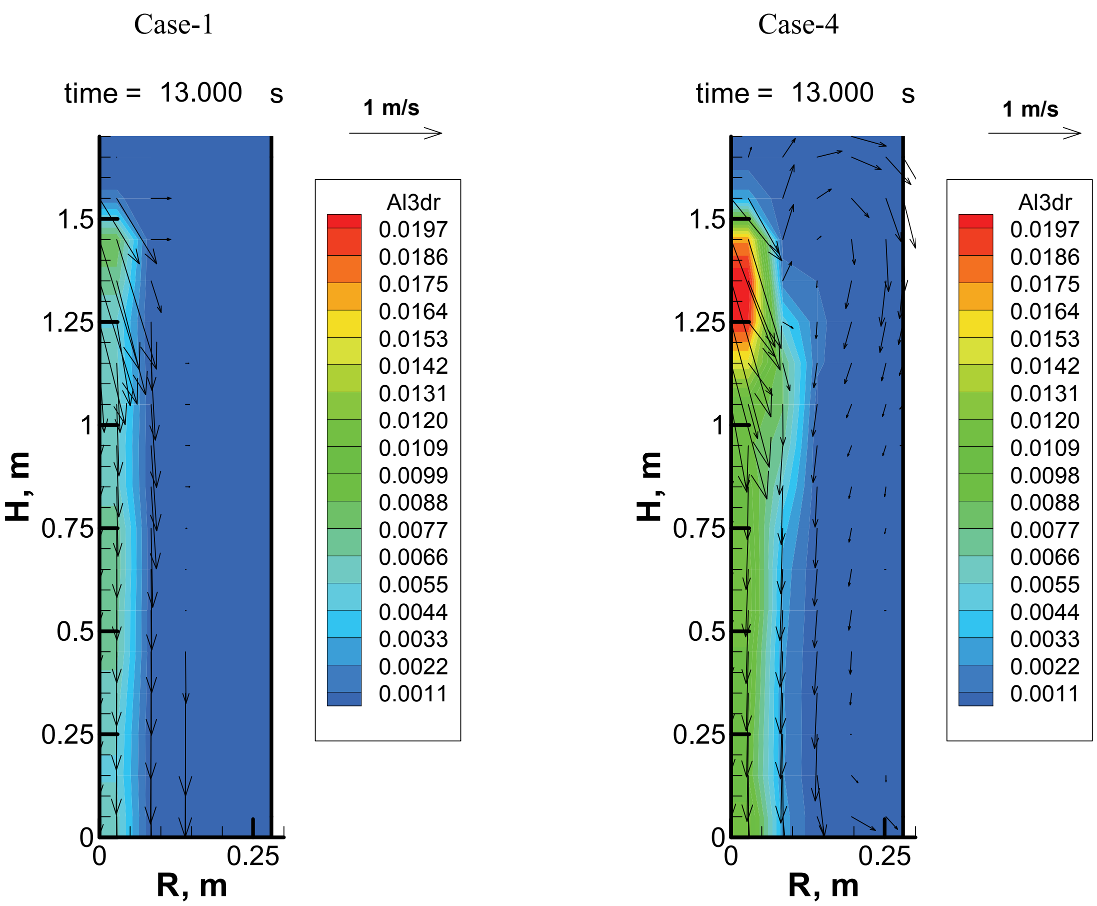


Figure 3. Comparison of volume fractions and mass averaged velocities of the debris. Baseline Case-1 and Case-4

4. Development of Conservative Closure for Coefficient of Agglomeration

The goal of this section is to develop a conservative-mechanistic approach for estimation of maximum physically reasonable value of the coefficient of agglomeration as a function of liquid particle fraction. For such estimation we combine data on mechanistically predicted fractions of liquid particles obtained in the set of selected baseline cases with experimental data on fractions of agglomerated debris in the DEFOR-A and DEFOR-S experiments. As a result we obtain a set of semi-empirical dependencies $\alpha_{ij} = m^j_{agl} / m^i_{liq}$ each of those is calculated by combining one data set from mechanistic simulations for m^j_{liq} with one experimental data set for m^i_{agl}. Then we define conservative-mechanistic estimation of α as a curve which envelops the domain covered by the set of α_{ij} dependencies. The results of calculations for α_{ij} are presented in Figure 4. For sensitivity analysis purposes we selected two bounding estimations for α represented by formulas (2) and (3) respectively.

$$\alpha(m_{liq}) = \begin{cases} 4 \cdot m_{liq}, & m_{liq} \le 0.5 \\ 1/m_{liq}, & m_{liq} > 0.5 \end{cases} \tag{2}$$

$$\alpha(m_{liq}) = \begin{cases} 25/4 \cdot m_{liq}, & m_{liq} \le 0.4 \\ 1/m_{liq}, & m_{liq} > 0.4 \end{cases} \tag{3}$$

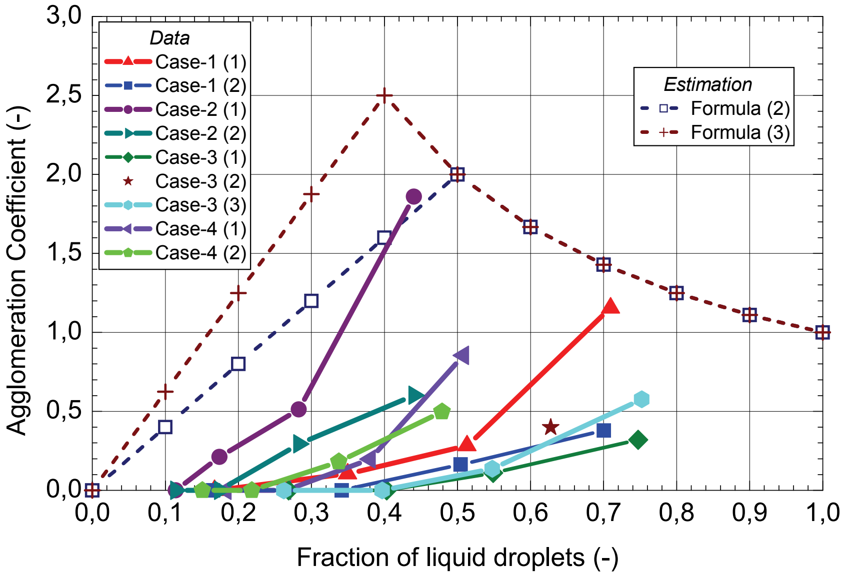


Figure 4. Coefficient of agglomeration as a function of liquid particle fraction. Case-*i*(*j*) indicates a combination of *i* base-line simulation case for m^i_{liq} and *j* experimental data for m^j_{agl}

Formulas (2) and (3) are based on the following considerations and assumptions:

(i) Typical mass fraction distributions of particles with different relative crust thicknesses at different depths in the pool were presented in (Figure 2). We consider particles as contributing to the fraction of "liquid" or "glue" particles (m_{liq}) if relative crust thickness is less than 0.1 (first two groups of particles in Figure 2).

(ii) In the limiting case of large mass fraction of liquid particles all solid particles will be glued with and eventually devoured by the liquid particles. Formulas (1), (2) and (3) gives $m_{aggl} = 1$ if m_{liq}

is larger than threshold (0.5 in (2) and 0.4 in (3)). Solid cake is formed when mass fraction of agglomerated debris is equal to 1. We made assumption that the threshold for transition to the cake bed is around 0.4 – 0.5 based on the following consideration. If we assume that cake is produced by filling of empty spaces between solid particles with liquid melt then fraction of liquid particles has to be more than 50%. Results of simulations for α_{ij} (Figure 4) are suggesting that estimation used in formula (2) for the α at $m_{liq} > 0.5$ is conservative. The threshold $m_{liq} > 0.4$ in formula (3) is rather over conservative and was selected to assess sensitivity of the agglomeration mode map to the selected threshold.

(iii) For $m_{liq} \leq 0.5$ there is big epistemic uncertainty in physical phenomena of formation of the debris agglomerates. There are also considerable uncertainty in the code prediction and quite high sensitivity of the predicted liquid particle fraction to FCI conditions. We require that estimation of α should be conservative with taking into account these uncertainties. Particularly we conbine experimental data on m_{agl} from DEFOR-A experiments with the simulations results obtained with same jet diameter to calculate set of dependencies for α_{ij} as it shown in Figure 4. Coolant conditions and jet breakup mode (as most uncertain and most influential parameters) were varied to obtain m_{liq} as function of the pool depth. According to formula (1) the lowest values of m_{liq} will result in highest values for α at the same experimental value of m_{agl}. In the present work the lowest values for m_{liq} (highest for α) where obtained with assumption about Kelvin-Helmholtz jet breakup mode (see also[3]). Figure 4 demonstrates that formula (2) provides enveloping estimation even for the Case-2(1) where m_{liq} was calculated for saturated coolant conditions (which results in m_{liq} around 0.45 on the Catcher 1) and experimental data taken from the case with subcooled water and more around 90% of agglomerated debris fraction in the Catcher 1.

(iv) To reduce uncertainties related to the influence of liquid droplet breakup models, we use particle size distribution obtained in the DEFOR-A experiments (Figure 5) for estimation of enveloping values of α.

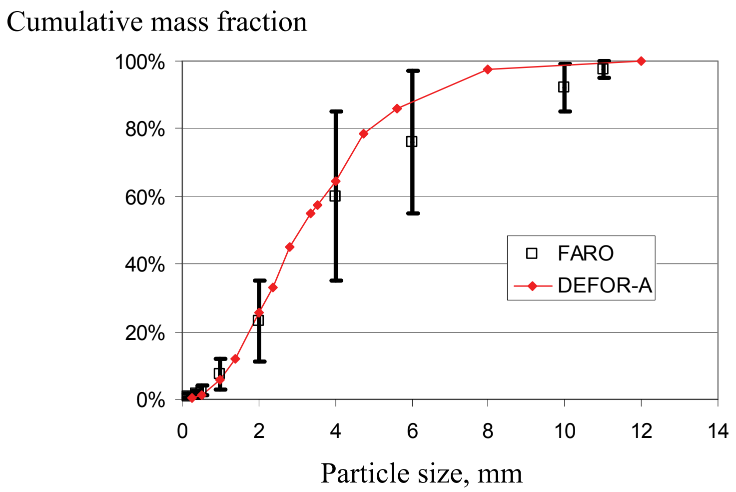


Figure 5. Particle size distribution (red line) measured in DEFOR-A experiment and used in the present simulations. Diapasons of particle-size distribution from FARO tests are shown for reference

(v) Some sources of uncertainty exist in the DEFOR-A experimental data itself. First of all, not all of the thermo-physical properties of the corium melt simulant material used in the DEFOR-A are well known at the moment. Second, measured melt temperature (superheat) was varying during the melt pouring process. It worth mentioning that melt composition (properties) and superheat are also intrinsically uncertain elements in the plant accident scenario. Therefore sensitivity study was performed to assess potential effect of these uncertainties on the prediction of liquid particle fraction and α. In the simulations melt thermo-physical properties (heat capacity C_p, heat of fusion H_{fus}, thermal conductivity λ) and melt superheat (T_{sup}) were varied according to Table 2. Results of sensitivity study for coefficient of agglomeration to thermo-physical properties and melt superheat are presented in Figure 6, which also confirms that formula (2) provides bounding estimate for α.

Table 2. Variations of melt thermo-physical properties for sensitivity study

Case	Melt thermo-physical properties			
	C_p, J/(kg·K)	H_{fus}, J/kg	λ, W/(m·K)	T_{sup}, C
Baseline Case-4	**280**	**83**	**5.3**	**100**
"C_p case"	**200**	83	5.3	100
"H_{fus} case"	280	**25**	5.3	100
"λ case"	280	83	**3.0**	100
"T_{sup} min case"	280	83	5.3	**70**
"T_{sup} max case"	280	83	5.3	**150**

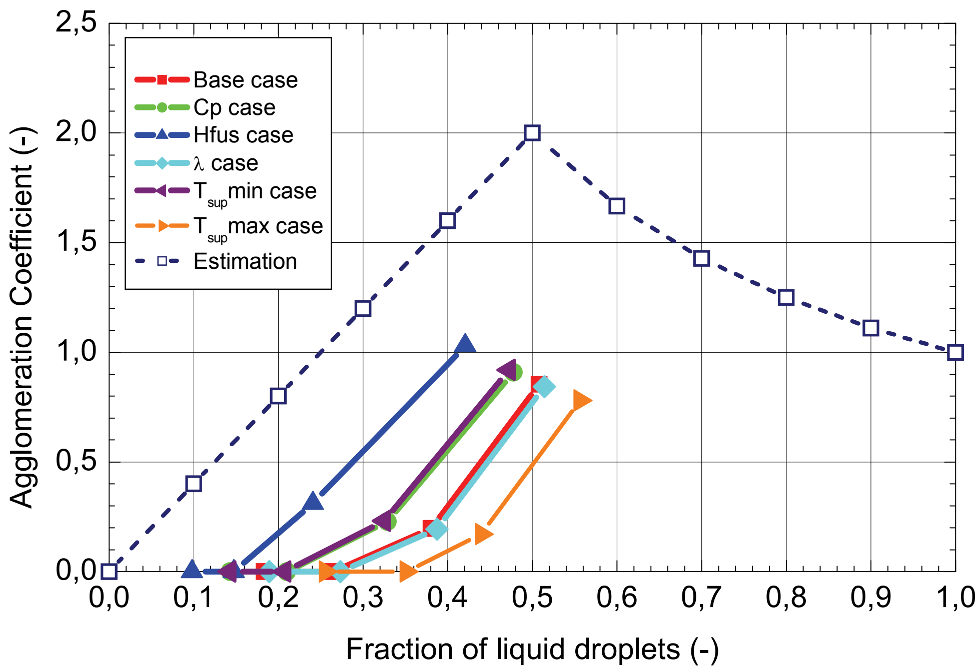


Figure 6. Sensitivity study for coefficient of agglomeration to thermo-physical properties and melt superheat. Red line is baseline Case-4

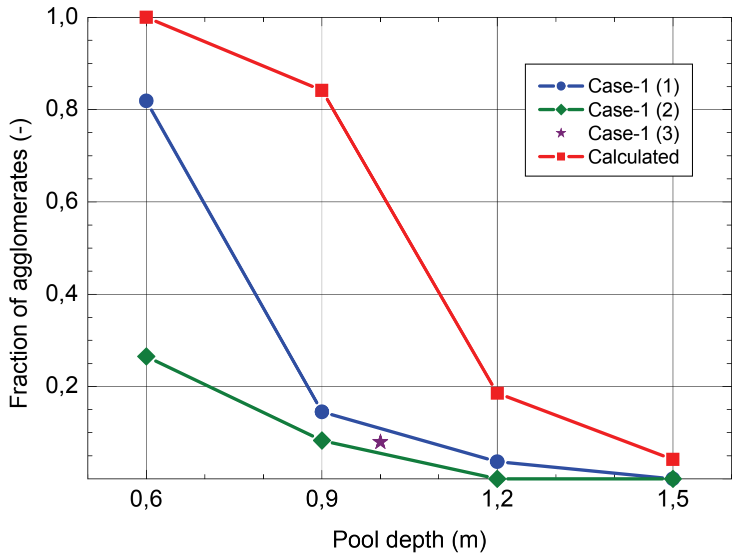


Figure 7. Comparison of simulation and experimental data for mass fraction of agglomerated debris as a function of water pool depth. Melt jet diameter 10 mm

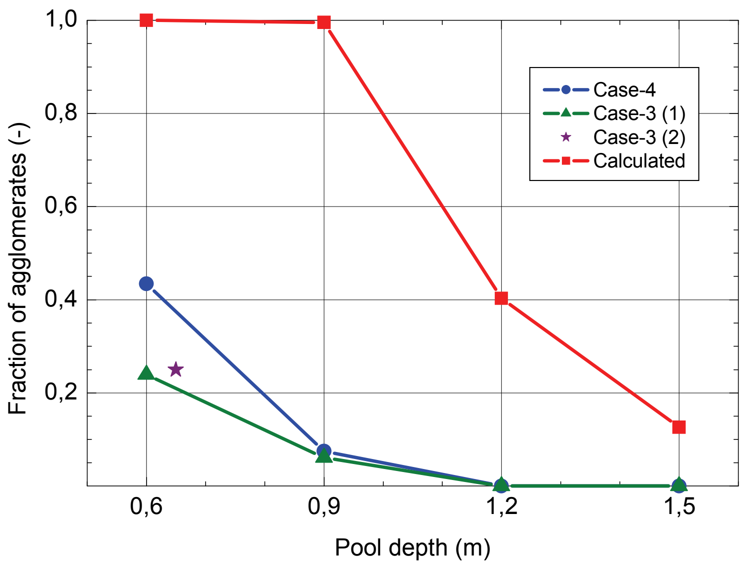


Figure 8. Comparison of simulation and experimental data for mass fraction of agglomerated debris as a function of water pool depth. Melt jet diameter 20 mm

(vi) Degree of conservatism in the prediction of fraction of agglomerated debris is demonstrated by comparison of the DEFOR-A and DEFOR-S experimental data with results of the VAPEX-P code simulations obtained with conservative-mechanistic estimation of α provided by formula (2). Also conservative assumptions about other parameters and models such as jet breakup model and water subcooling were used in simulation. Namely water was assumed to be at subcooled conditions and jet breakup mode was assumed to be leading edge Rayleigh-Taylor instability. The fraction of liquid particles is expected to be largest at such assumptions as it has been shown in the present and in the previous work[3]. Results for two cases with 10 mm and 20 mm melt jet diameter are presented in Figure 7 and Figure 8 respectively. Predicted values of the mass fraction of agglomerates are well above the experimentally measured ones. This is a demonstration of conservatism in modeling and resultant margin sufficient to cover possible uncertainties in the code prediction or in the scenario parameters. On the other hand, predicted dependencies are in good qualitative agreement with the experimental ones. Namely predicted fraction of the agglomerated debris decreases rapidly with increasing of the water pool depth, as it is also observed in the experiments. This is important advantage of conservative-mechanistic approach which provides both necessary margin, and, at the same time, takes into account mechanistic limiting mechanisms in the system behavior.

5. Application to Plant Accident Analysis

In this section we apply developed model for prediction of mass fraction of agglomerated debris for development of the agglomeration mode map at plant accident conditions. Parameters used in simulations are presented in Table 3. As in the previous work[3] we applied conservative assumption that jet breakup mode is Rayleigh-Taylor (R-T) instability. Results of simulation for the agglomeration mode map are presented in Figure 9. Red line represents the boundary of “cake” domain. Other lines in the figure represent cases of partially agglomerated debris with 5%, 10% и 20% mass fraction of agglomerates.

Results of study of the agglomeration mode map sensitivity to the α closure are presented in Figure 10. Relative difference between results obtained with formulas (2) and (3) is no more than 3% in terms of jet diameter at which $m_{aggl} = 5\%$. That shows that results of prediction are robust and insensitive to variations of bounding closure for α.

Table 3. Plant accident conditions for agglomeration mode map development

Parameter	Value
Pool parameters	
Diameter, m	9
Depth, m	7-12
Initial pressure, bar	1
Water temperature, K	373
Melt parameters	
Composition	Eutectic corium
Total mass, t	180
Initial temperature, K	3000
Initial met superheat, K	189
Jet diameter, mm	70-300
Jet release height, m	6

It is important to mention that predicted mass fraction of agglomerated debris reduces rapidly with increasing of the pool depth or decreasing of melt jet diameter (Figure 9), even with conservative estimation (2) for the coefficient of agglomeration. Safety implication of this finding is that main threat of agglomeration occurrence comes from incompletely disintegrated jet. Liquid droplets, once formed, solidify pretty fast. Thus no significant agglomeration is expected to occur at 1-2 meters below the leading edge of the melt jet. Therefore it is important in severe accident management strategy to avoid conditions at which jet breakup length can be larger or equal to the pool depth.

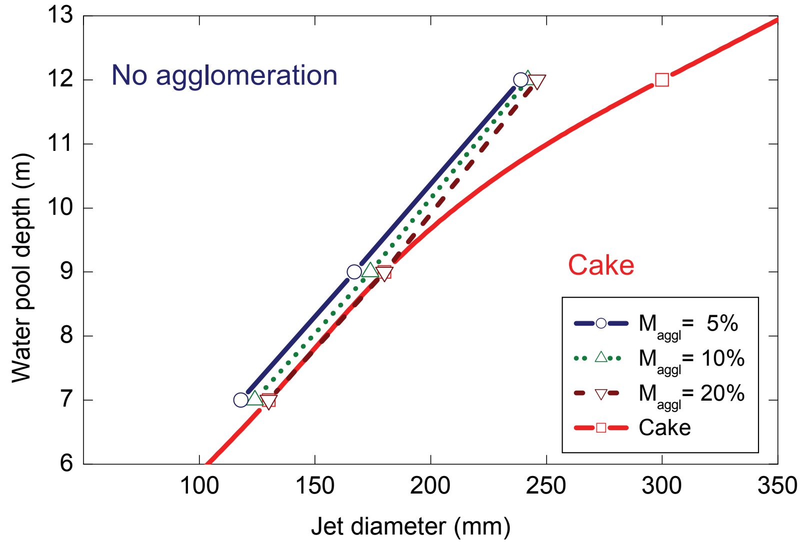


Figure 9. Agglomeration mode map for plant accident conditions with mass fractions of agglomerated debris

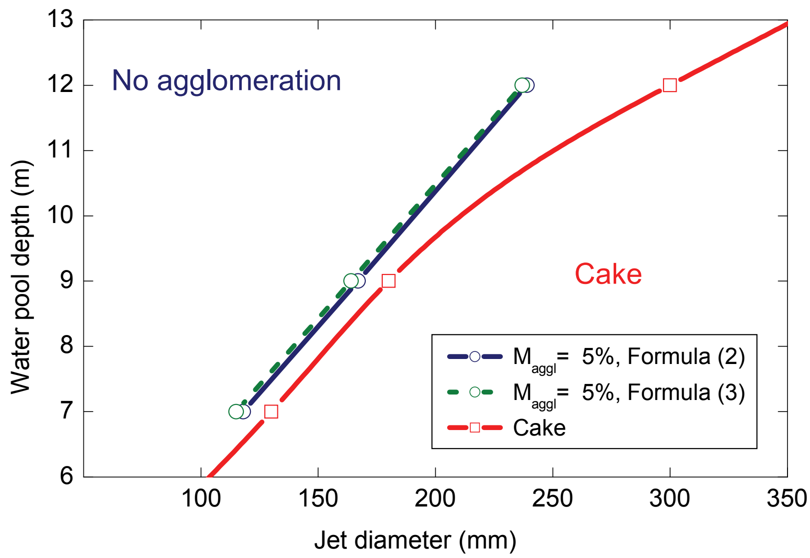


Figure 10. Sensitivity of agglomeration mode map to agglomeration coefficient

5. Conclusions and safety implications of results

One of the factors which may significantly affect ex-vessel debris bed coolability is debris agglomeration. There are considerable aleatory and epistemic uncertainties in scenarios and physical phenomena of the debris agglomeration and cake formation. In the present work we develop conservative-mechanistic approach for quantification of the debris agglomeration mode map. The approach is based on conservative assumptions in modeling by mechanistic FCI simulation tool (VAPEX-P code).

An approach for estimation of the fraction of agglomerated debris is proposed. Experimental data from the DEFOR-A experiments are used for development and validation of semi-empirical conservative-mechanistic closure. It is demonstrated that conservative treatment of epistemic uncertainties in agglomeration phenomena and aleatory uncertainties in scenario (melt properties and superheat) creates sufficient margin and simulation data are enveloping the set of mass fractions of agglomerated debris obtained at various experimental conditions.

Application of the developed models to the plant accident conditions allows us to quantify "partial agglomeration" domain on the agglomeration mode map[3] in terms of mass fraction of agglomerated debris. Plant scale analysis suggests that it is possible in principle to achieve completely fragmented debris bed within the present design of Swedish BWRs. Important and encouraging finding is that mass fraction of agglomerated debris reduces rapidly with increasing of the pool depth or decreasing melt jet diameter even if there is considerable degree of conservatism in analysis. No significant agglomeration is expected to occur at 1-2 meters below the leading edge of the melt jet.

In the next steps new data from the coming DEFOR-A experiments will be used for more rigorous validation of developed approach. Sensitivity study for location of boundaries between domains of the agglomeration mode map at different scenarios of melt release (initial melt superheat, composition, etc.) is to be performed.

Acknowledgement

This work is supported by the Swedish Nuclear Radiation Protection Authority (SSM), Swedish Power Companies, European Commission (SARNET), Nordic Nuclear Safety Program (NKS), Swiss Federal Nuclear Safety Inspectorate (ENSI) under the APRI- MSWI program at the Royal Institute of Technology (KTH).

OECD SERENA:

A Fuel Coolant Interaction Programme (FCI) devoted to reactor case.

P. Piluso
Commissariat à l'Énergie Atomique, Cadarache, France
S.W. Hong
Korea Atomic Energy Research Institute, Korea

1. Introduction

SERENA is an OECD programme on fuel-coolant interaction (FCI), which has the scope of making a status of the code capabilities to predict FCI induced dynamic loading of the reactor structures (Phase 1), and performing the complementary research possibly needed to increase the level of confidence of the predictions (Phase 2). Phase 1 has been completed. It consisted of comparative calculations by available tools of selected existing experiments and reactor cases, in order to identify those areas where lack of understanding induced large uncertainties in the predictions of the loads in reactors[1]. Phase 2 has the scope of carrying out the confirmatory analytical and experimental research needed to reduce these uncertainties to acceptable level for risk assessment. Phase 1 was the first comparative exercise undertaken since ISP-39, which however concerned premixing only[2].

Organisations participating in Phase 1 were Commissariat à l'Énergie Atomique (CEA) jointly with Institut de Radioprotection et de Sûreté Nucléaire (IRSN), France, Korea Atomic Energy Research Institute (KAERI) jointly with Korea Maritime University (KMU), Nuclear Regulatory Commission (NRC), USA, University of Wisconsin (UW) and University of California Santa Barbara (UCSB) sponsored by NRC, Institute für Kernenergetik und Energiesysteme (IKE) sponsored by Gesellschaft fuer Anlagen und Reaktorsicherheit (GRS), Germany, Forschungszentrum Karlsruhe (FZK), Germany, Electrogorsk Research and Engineering Centre (EREC), Russian Federation, Japan Atomic Energy Research Institute (JAERI), Japan and Nuclear Power Engineering Corporation (NUPEC), Japan.

The FCI codes used were ESPROSE-m (UCSB), IDEMO (IKE), IFCI (KINS), IKEMIX (IKE), JASMINE (JAERI), MATTINA (FZK), MC3D (CEA-IRSN, IKE), PM-ALPHA (UCSB), TEXAS-V (UW, KAERI), TRACER (KMU) and VESUVIUS (NUPEC), respectively.

[1] Magallon D., Bang K.-H., Basu S., Berthoud G., Buerger M., Corradini M.L., Jacobs H., Meignen R.,Melikhov O. Naitoh M., Moriyama K., Sairanen R., Song J-H., Suh N. and Theofanous T.G. 2003.OECD Programme SERENA (Steam Explosion Resolution for Nuclear Applications): Work Programme and First Results. NURETH-10, Seoul, Korea, October 5-9.
[2] Annunziato A., Addabbo C. and Leva G. 1997. OECD/CSNI International Standard problem No. 39 on FARO Test L-14. NEA/CSNI/R(1997)31.

As being a status of the code capabilities to calculate FCI in reactor situations, it was not within the scope of SERENA to establish whether or not a specific code was qualified to be in. It was let to the responsibility of each partner to judge whether his code or the code he is using had the required degree of qualification and verification to participate. Most data used for code validation and verification last decade came from the FARO and KROTOS programmes performed under international sponsorship at the Joint Research Centre of the European Commission, Ispra site (Italy). However, about half of the partners in SERENA were not involved in these programmes and had not full access to the detailed data prior to SERENA.

Consequently, calculating typical experiments prior to reactor application had the twofold objective of establishing a "setting to zero" of the codes and each participant starting with verification of their tools on a similar basis. It allowed partners to verify in which conditions of parameters and model options the codes were able to capture the essential features of experiments performed in so-called "realistic conditions", and set up model options and parameters for calculating the reactor situations. Integrating information coming from this variety of backgrounds allowed identifying the common areas where uncertainties are consequential to the estimate of the loads.

Organisations participating in Phase 2 are Commissariat à l'Énergie Atomique (CEA) jointly with the Institut de Radioprotection et de Sûreté Nucléaire; the Korea Atomic Energy Research Institute (KAERI), in association with Korea Institute of Nuclear Safety; Tractebel Engineering, a division of Suez-Tractebel S.A., Belgium; the Atomic Energy of Canada Limited; the Valtion Teknillinen Tutkimuskeskus, Finland; the Gesellschaft für Anlagen- und Reaktorsicherheit, Germany; the Japan Nuclear Energy Safety Organisation, Japan; the Jozef Stefan Institute, Slovenia; the Statens Kärnkraftinspektion1, Sweden; the Paul Scherrer Institute, Switzerland; the United States Nuclear Regulatory Commission, USA.

For ex-vessel steam explosion, calculated loads with the current models, although low, are partly above the capacity of typical cavity walls. The scatter of the results raises the problem of the quantification of the safety margin for ex-vessel FCI. OECD SERENA phase 2 has started at October 1, 2007 and will be finished September 30, 2011. The role of void (gas content and distribution) and corium melt properties on initial conditions (pre-mixing) and propagation of the explosion are the key issues to be resolved to reduce the scatter of the predictions to acceptable levels. OECD SERENA phase 2 is formulated to resolve the uncertainties on these issues by performing a limited number of well-designed tests with advanced instrumentation reflecting a large spectrum of ex-vessel melt compositions and conditions, and by the required analytical work to aim the code capabilities to a sufficient level for use in reactor case analyses. These goals will be achieved by using the complementary features of KROTOS (1D mock-up/CEA) and TROI (2D – mock-up/KAERI) corium facilities including fitness for purpose oriented analytical activities. A first validation of models on KROTOS (before validation on TROI) data and verification of code capabilities are performed with regard to the reactor case.

The scope of the paper is to give a general picture of the present FCI code capabilities to reproduce existing data, to summarise the conclusions that have been reached to tentatively explain the observed differences, and to deduce the consequences for reactor application in the frame of SERENA (phase 1). It is not intended to draw conclusions on which code performs better than another, or the best, if any. For these reasons, calculation results are presented without any reference to the codes. A comparative review of the codes and models has been performed in the frame of SERENA[3]. In the second part, SERENA (phase 2) will be presented, the tow test facilities (TROI and KROTOS) and the experimental grid.

2. Methodology-SERENA (phase 1)

First, generic situations corresponding to plausible melt relocation scenarios and capable to produce potentially damaging steam explosion were identified. For ex-vessel, large pour equivalent to some tens of centimetres in diameter of UO_2-ZrO_2-Zr-Steel melt into a cavity flooded with subcooled water was selected as a situation matching the above criteria. For in-vessel, multi-jets arriving offcentrein the lower head was considered the most challenging for the vessel. Then, existing experiments as far as possible in

[3] Meignen R., Magallon D., Bang K.-H., Basu S., Berthoud G., Buerger M., Corradini M.L., Jacobs H.,Melikhov O., Naitoh M., Moriyama K., Sairanen R., Song J-H., Suh N. and Theofanous T.G.2005. Comparative Review of FCI Computer Models Used in the OECD-SERENA Program.Proceedings of ICAPP-05, Seoul, KOREA, May 15-19, Paper 5087.

relation with these reactor situations were selected. Noting that no relevant multi-pour experiment exists, the best we could extract today from experimental database in relation to the SERENA Phase 1 objectives was found in the FARO, KROTOS and TROI programmes. Participants were given same sets of initial conditions and reference data. They translated these initial conditions into adequate inputs for their codes. Participants were let free to set model options and parameters as they used to. However, they were asked to provide at least one calculation with using standard parameters, to document their choices and possibly make sensitivity calculations.
Explosion phases of the experiments were calculated both for imposed and calculated pre-mixing whenever required. Comparison was made on a set of pre-established quantities, either for codes-todata comparison or code-to-code comparison. These quantities included nodalisation, pressure and impulse, vaporization/condensation rates, energy release, component fraction, debris characteristics.
The calculation work was divided into 3 tasks, namely, calculation of pre-mixing experiments, calculation of explosion experiments, reactor applications. In SERENA, phenomena were considered important as far as they induce large uncertainties on the loads calculated for reactor configurations. For this reason, conclusions drawn from code application to experiments about the importance of a given phenomena were considered as provisional until reactor application were performed.

3. Pre-mixing calculation-SERENA (phase 1)

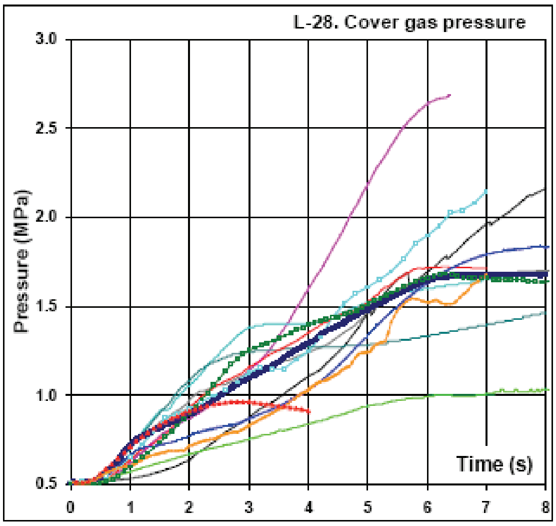
The comparison exercise for the premixing phase was performed essentially for two FARO experiments, namely, FARO L-28[4]which investigated the premixing phenomena only, and FARO L-33 [4]; which was an integral experiment covering both premixing and explosion. In these experiments quantities up to 175 kg of 80 wt% UO_2 -20 wt% ZrO_2 molten corium were poured into water at different pressures and subcooling levels. Table 1 summarises the specific conditions of L-28 and L-33 tests. Note that FARO L-28 was performed with saturated water typical of in-vessel conditions, and FARO L-33 with sub-cooled water typical of ex-vessel conditions. Participants were given two sets of initial and boundary conditions, one each for L-28 and L-33.
Figures 1 and 2 compare the various predictions for pressure and energy release with the experimental records for FARO L-28 and FARO L-33, respectively. Actually, experimental energy data were calculated by using the pressure and temperature records. Time zero corresponds to start of melt delivery to the water. In FARO L-28 melt delivery duration is approximately 6 s, and the melt front reaches the bottom of the test section after about 1 s. In FARO L-33, pre-mixing ceases at about 1.1 s when the explosion is triggered, corresponding approximately to melt-bottom contact. Figure 3 shows the global void fractions at 1s and 4s. Experimental values in Figure 3 have been calculated by using the level swell records. Figure 1 shows that the codes have difficulties to reproduce the pressure data both for the initial phase of the pressurisation up to melt-bottom contact and for the linear increase of the pressure, which roughly corresponds to a steady state phase. They tend to underestimate heat transfer (Figure 3) and overestimate void.

[4] Magallon D., Huhtiniemi I., Dietrich P., Berthoud G., Valette M., Schütz W., Jacobs H., Kolev N., Graziosi G., Sehgal R., Bürger M., Buck M., Berg E.V., Colombo G., Turland B., Dobson G. and Monhardt D. 2000. *MFCI Project—Final Report.* INV-MFC I(99)-P007, EUR 19567 EN.

Table 1: Conditions of experiments for premixing calculations

Experiment	FARO L-28 (Premixing test)	FARO L-33 (Integral test)
Melt and composition	80 wt% UO_2 20 wt% ZrO_2	80 wt% UO_2 20 wt% ZrO_2
Melt mass released	175 kg	100 kg (40 kg at trigger)
Melt temperature	3053 K	3070 K
Melt superheat	203 K	220 K
System pressure	0.51 MPa	0.41 MPa
Water temperature	424 K	294 K
Water subcooling	0 K	124 K
Release diameter	0.05 m	0.05 m
Δp melt delivery	Gravity	Gravity
Free fall in gas space	0.89 m	0.77 m
Water depth	1.44 m	1.62 m
Water pool diameter	0.71m	0.71 m
Free-board	Closed volume (3.5 m^3)	Closed volume (3.5 m^3)
Trigger	No	Yes (applied at bottom at melt-bottom contact)

Figure 1: Comparison of calculated vessel pressure and energy release with data (bold curves) for FARO L-28 (pre-mixing of 175 kg of corium melt in 1.5-m-deep saturated water pool; melt-bottom contact at ~1s)



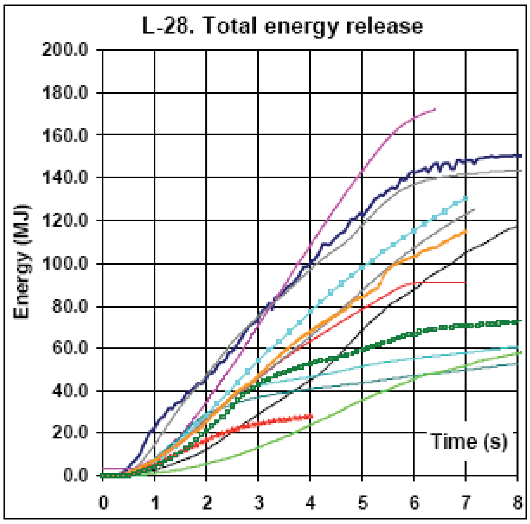


Figure 2: Comparison of calculated vessel pressure and energy release with data (bold curves) for FARO L-33(1.5-m-deep subcooled water pool; triggering at 1.1 s, i.e., approximately at melt bottom contact with 25 kg of corium melt in pre-mixing).

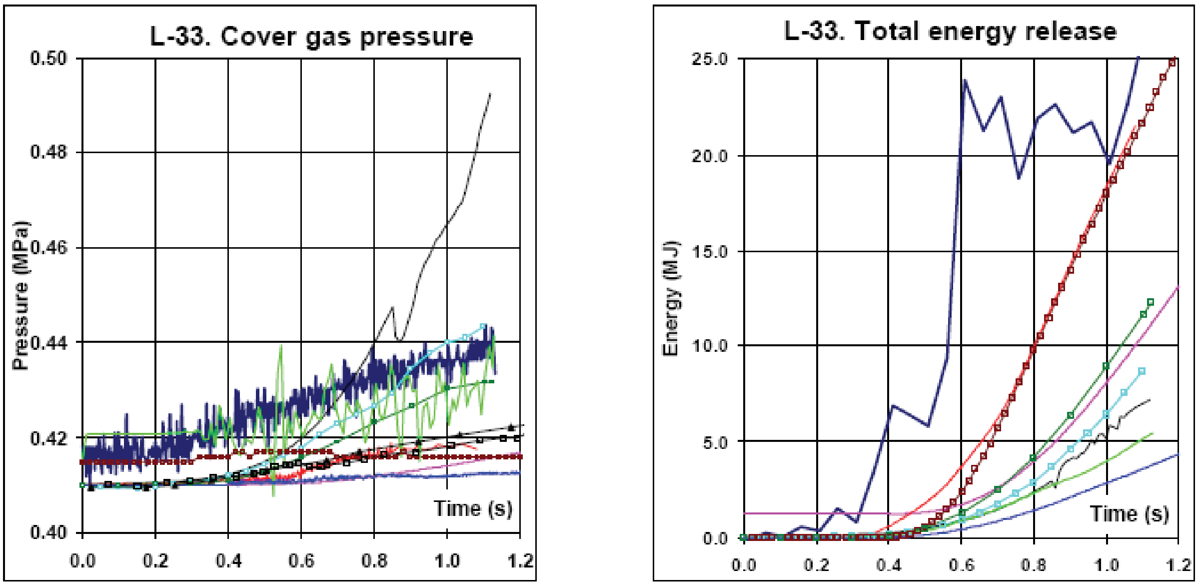
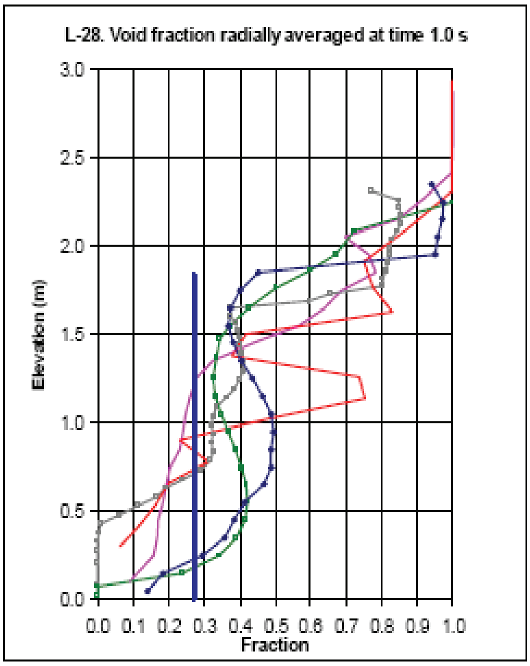
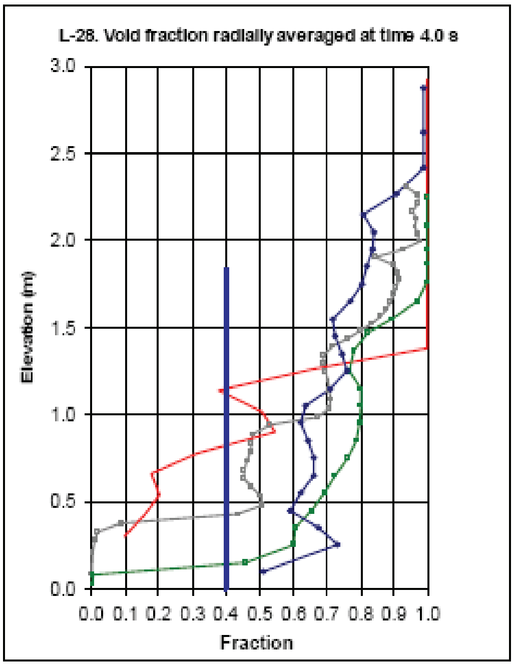


Figure 3: Comparison of radially-averaged void fraction calculated by the codes with global value calculated from experimental level swell (bold lines) for FARO L-28 at time 1.0 s and time 4.0 s.





Sensitivity calculations showed that reducing significantly interfacial steam/water friction, or changing the transition range between water and steam continuous regimes allowed reducing void to values comparable to the data. Reducing the initial particle diameter or forcing heat exchange in continuous steam flow regime improved the pressurisation curve. However, no physical basis exists, which could justify one or another modification. In addition, large uncertainties affect experimental data in the absence of detailed information of the pre-mixing zone internals: experimental void fraction is global value retrieved form water level swell measurements, which does not allow to identify where the void is located actually. The reasons for the initial strong pressure increase in L-28 and the contribution of the debris cooling to the overall void level are not well understood. The major uncertainty on pre-mixing as can be deduced form application to the selected experiments stands in void (prediction and data). The major question in relation to the scope of SERENA is whether the differences in modelling and the scatter of the predictions are relevant for reactor applications. This question will be answered after analysis of reactor calculations. One can simply say for now that most codes overestimate void fraction with respect to that calculated from the integral experimental data, especially in saturated conditions. Confirmation that voiding is calculated properly and finding the same trends for the reactor cases would practically exclude steam explosion as an in-vessel issue with respect to dynamic loading.

4. Explosion calculation for SERENA (phase 1)

In addition to FARO L-33, two other experiments were selected for testing the code performance for the explosion phase, namely, KROTOS-44 with alumina melt [5](Huhtiniemi *et al.,* 1999) and TROI-13 with 70 wt% UO_2-30 wt% ZrO_2 melt[6]. Table 2 summarises the main conditions of these tests. An experiment with alumina melt was chosen because it was a well-characterised 1-D steam explosion with well defined external trigger, producing a very energetic interaction and thus well appropriate to test the explosion models. Then, TROI-13 dealing with a similar quantity of corium and 2-D geometry, represented an extension to more realistic conditions. Finally FARO L-33 with 25 kg of corium melt in water at the time of the trigger and 2-D geometry represented a step further in scale. Calculations were performed either for

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[5] Huhtiniemi I., Magallon D. and Hohmann H. 1999. *Results of recent KROTOS FCI tests: alumina versus corium tests.* Nucl. Eng. Des. **189,** 379-389.

[6] Song J.H., Park I.K., Shin Y.S., Kim J.H., Hong S.W., Min B.T. and Kim H.D. 2003. *Fuel coolant interaction experiments in TROI using a UO2/ZrO2 mixture.* Nucl. Eng. Des., **222**, 1-15.

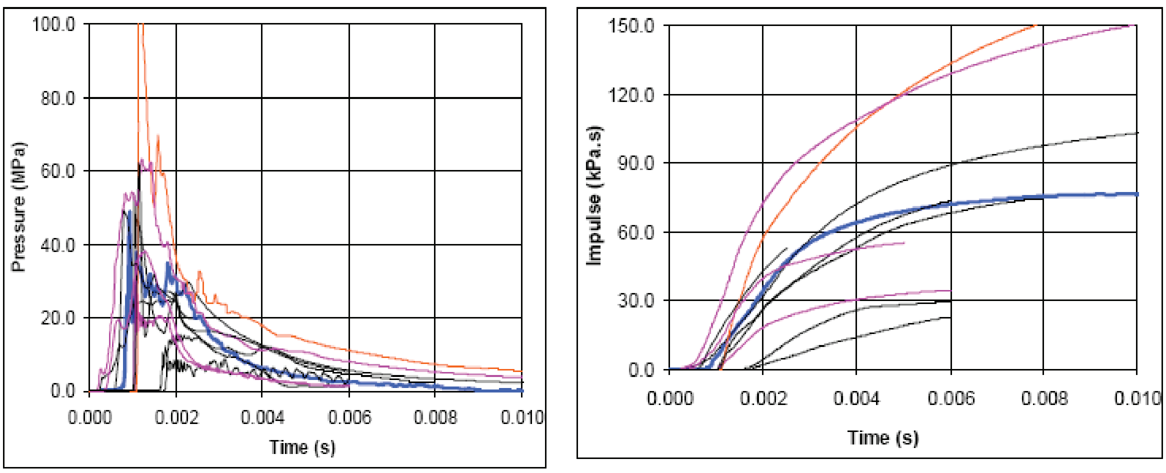
a given pre-mixture (K-44), or for both a given and a calculated pre-mixture (TROI-13, L-33). Note that a blind exercise was done also on a TROI test, subsequently performed as TROI-34. Results do not significantly differ from TROI-13.

Table 2: Conditions of KROTOS and TROI experiments used for explosion calculations

Experiment	TROI-13	KROTOS-44
Melt and composition	70 wt% UO_2-30 wt% ZrO_2	100 % Al_2O_3
Melt mass released	7.7 kg	1.5 kg
Interacting melt mass	1.14 kg	1.44 kg
Melt temperature	~3300 K	2673 K
Melt superheat	~500 K	359 K
System pressure	0.1 MPa	0.1 MPa
Water temperature	292 K	363 K
Water subcooling	81 K	10 K
Release diameter	0.02 m	0.03
Δp melt delivery	Gravity	Gravity after crucible impact
Free fall in gas space	3.9 m	0.43 m
Water depth	0.69 m	1.115 m
Water pool diameter	0.60 m	0.20 m
Free-board	Closed volume (8.03 m^3)	Closed volume (0.23 m^3)
Trigger	No	Yes (at bottom)

Figure 4 shows the explosion pressure and corresponding impulse for KROTOS K-44. It can be seen that most of the models applied to the same initial and boundary conditions were able to globally reproduce the strong event observed in the experiment even with differences in modelling of the key effects of fragmentation, and non-homogenous heat transfer to the coolant. Comparison with the other pressure records at different levels in the water shows a common interpretation of the experimental results as a propagating-escalating event. This is somehow not surprising since KROTOS data has been used as a basis to validate the models. In general, standard values of the parameters were used for the simulations. Note however that some predictions noticeably underestimate the loads.

Figure 4: Comparison of dynamic pressure and corresponding impulse calculated by the codes with experimental value (bold lines) at mid-height in the water pool for KROTOS-44.



For TROI-13 (Figure 5) and FAROL-33 (Figure 6 and 7) tests, the agreement seems of the same order than for KROTOS, despite the events in these experiments were significantly less energetic than in KROTOS K-44, with however a larger overestimate of the loads for TROI-13 than for L-33.

Actually, these loads have been obtained with reducing more or less arbitrarily key effects such as heat transfer and fragmentation. Possible physical explanations for the observed reduced explosion energetics are melt freezing and hydrogen production during pre-mixing. But differences in test and/or pre-mixing geometry (radially 2-D in FARO and TROI instead of 1-D in KROTOS alumina) may also have a educing effect because they allow venting during the explosion. Visualisation performed in KROTOS has shown that such differences in pre-mixing lateral extension exist between alumina and corium. It should be noted that the data concern two similar types of oxidic corium only, namely, 70 wt% UO_2-30 wt% ZrO_2 and 80 wt% UO_2-20 wt% ZrO_2. Therefore, it would be hazardous to extend the conclusion of “low explosivity” to other corium melt compositions before understanding the very reasons that led to mild explosions in the FARO, KROTOS and TROI corium experiments.

Figure 5: Comparison of dynamic pressure and corresponding impulse calculated by the codes with experimental value (bold lines) for TROI-13.

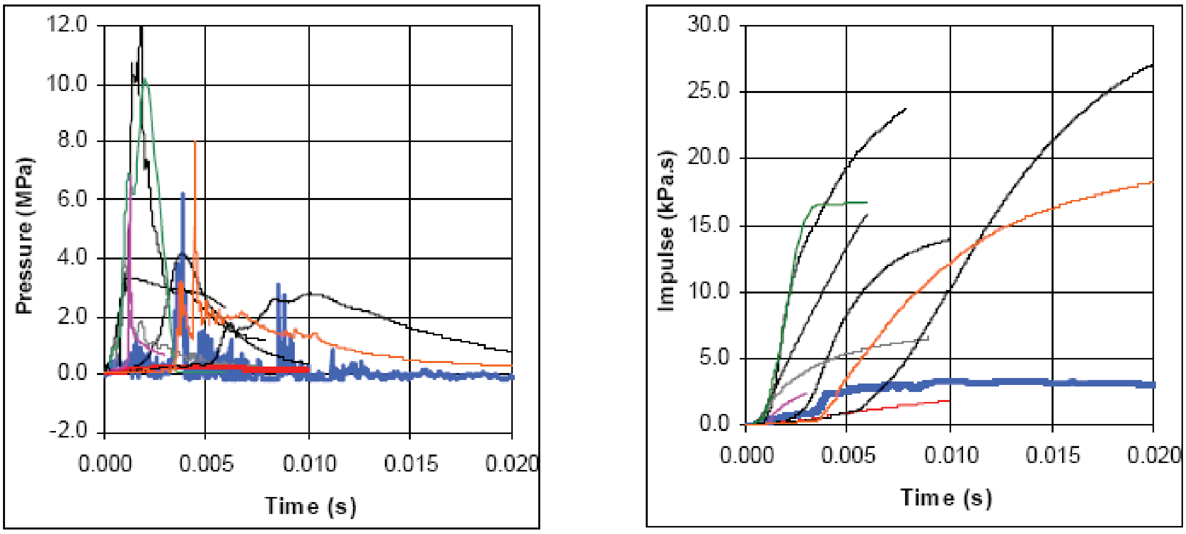


Figure 6: Comparison of dynamic pressure and corresponding impulse calculated by the codes with experimental value (bold lines) at level 1390 mm in the water pool for FARO L-33 (highest value measured). Full calculations premixing+explosion.

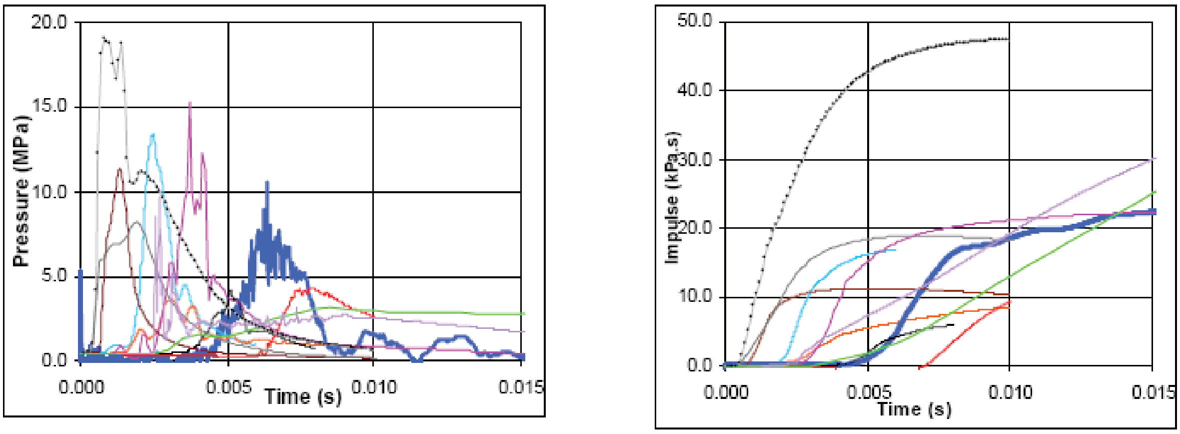
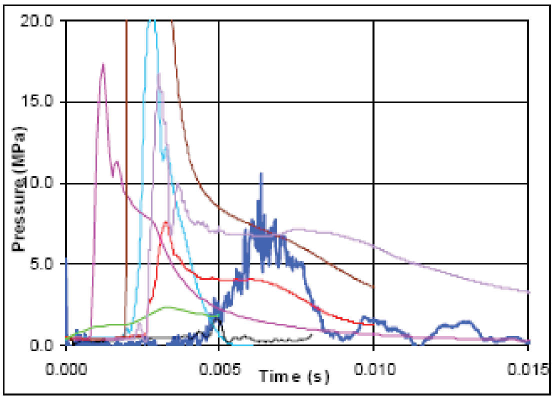
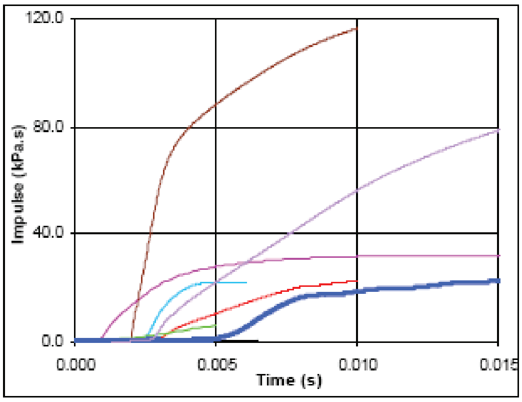


Figure 7: Comparison of dynamic pressure and corresponding impulse calculated by the codes with experimental value (bold lines) at level 1390 mm in the water pool for FARO L-33 (highest value measured). Common premixing conditions.





It should be noted also that for the two basic descriptions of the explosion used in the codes, namely the micro interaction and the non-equilibrium heat transfer models, a number of parameters have to be given for the fragmentation, heat release and partition between steam and water. Playing with these parameters allows in general finding back the order of magnitude of the data, but basic physical explanation is missing.

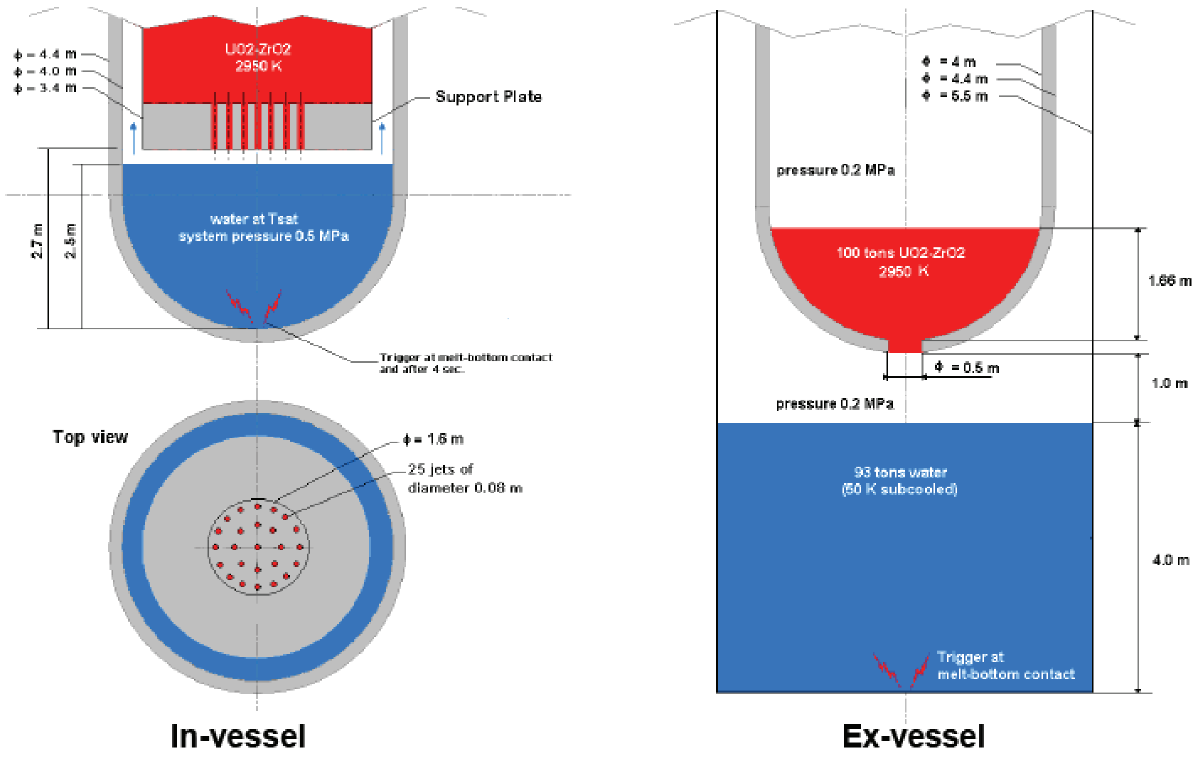
For FARO L-33, when comparing full calculations (pre-mixing and explosion calculated both) and explosions calculations performed with common pre-mixing conditions (Figures 6 and 7, respectively), no significant differences are observed on the level of the loads, except in two cases. Actually, in sub-cooled water pre-mixing void differences that may exist between codes and data are such that the flow regime does not change in the calculations, and thus, impact on the overall explosion behaviour is less significant than for saturated water. Concluding, one can say that the major uncertainties on explosion as can be deduced form application to the selected experiments stands in pre-mixing geometry and material behaviour during propagation of the explosion as a function if its state in pre-mixing at the time the explosion triggers (material effect). Again this is important relatively to the objective of SERENA as far as it is relevant for reactor estimates. This question is addressed in the next section.

5. Reactor case calculation -SERENA (phase 1)

Figures 8 summarises the initial and boundary conditions used for in- and ex-vessel cases, respectively, according to the "generic situations of most interest" established at the beginning of the programme . When looking at these conditions, one has to keep in mind that the scope was not to calculate a specific scenario in a reactor specific geometry and draw conclusions about the FCI risk for that geometry. The scope was to verify whether the codes used by the partners as their tools for FCI analysis are able to calculate plausible reactor situations, and to compare the results in order to identify the differences and the actions required to understand and reduce them. In both cases, a gravity pour was considered. For in-vessel, the multi-jet configuration is obviously fully 3-D, while most codes have to be run 2-D axi-symmetric even when applied to reactor situations. This was made deliberately in order to include in the simulation all the aspects related to code application to reactor cases, and, in particular, the simplifications that have to be made to reduce to 2-D the actual 3-D situations. Note that for the ex-vessel case, oxidic melt was also chosen not to introduce metal oxidation process that most codes are not modelling at present stage. The calculations were made for both the pre-mixing and the explosion phases. The choice of the parameters for explosion was not made consistently with respect to Task 3. Some partners used the reduced parameters. Some used the standard ones, as they were considered to be conservative. A trigger was applied when the melt front reached the bottom of the vessel or cavity. For the in-vessel case, this occurs approximately after 1 s. It was planned to perform a calculation with a trigger applied after 4 s. However, it was found that this case had little interest as practically no liquid water was present in the mixing zone at that time, and in any case negligible loads were calculated.

Figures 9 shows the dynamic pressure histories calculated at the wall for both the in-vessel and ex-vessel cases. For each code, it corresponds to the pressure history at the location where the maximum value was obtained at a time during the explosion. Figure 10 shows the corresponding impulses. In general those impulses were also the maximum obtained. In a few cases, the impulse was slightly higher at another location, but not such as to bias the conclusions that can be drawn from the exercise.

Figure 8: Calculated reactor situations

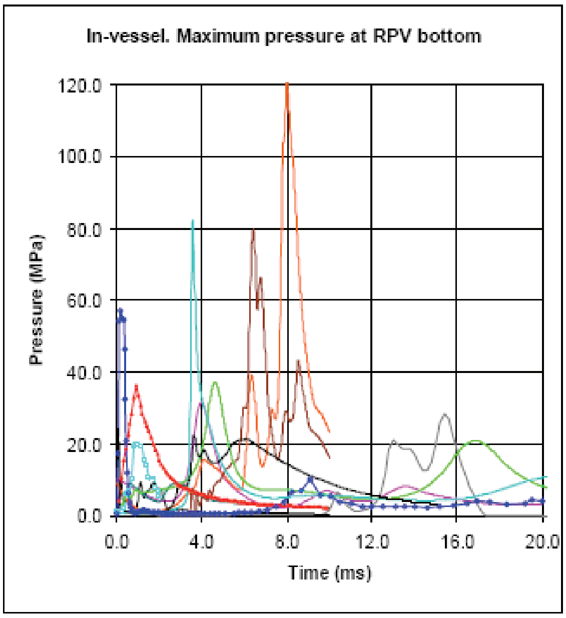


- The in-vessel results show noticeable differences in predicting the peak pressure at the RPV bottom (**ranging from ~10 MPa to ~120 MPa**) and a rather reduced prediction range for the impulse (from **some tens of kPa.s to ~200 kPa.s**). These loads are far below the capacity of the defined-model intact vessel and, therefore, the safety margin for in-vessel steam explosion may be considered as sufficient. This conclusion is challenged by the large scatter of the results and the uncertainties on the void predictions revealed by experiment calculations. Figure 11 shows that the level of the averaged void is high for both in- and ex-vessel cases. In addition, only one in-vessel case has been calculated that might not be the worst possible (This is somehow in contradiction with the choice of a multi-jet configuration which was supposed *a priori* to give the largest mass in pre-mixing, but which is compensated in part by the large voiding of the pre-mixing region as calculated).
- The ex-vessel results show noticeable differences in the predictions for both the explosion pressure and the impulse. The calculated maximum pressure loads at the cavity lateral wall vary from a few **MPa to ~40 MPa and** the impulses from **a few kPa.s to ~100 kPa.s** (except one case where the impulse is significantly higher due to the fact that the pressure level remains high for a long time).

These loads, even low, are above the capacity of cavity walls. The question of the safety margin for exvessel steam explosion already raises here prior to any further consideration related to the scatter of the results, the level of void (very high here too, see Figure 11), or the melt relocation scenario.
Therefore, besides reducing the uncertainties on void, it is important to increase the knowledge level of steam explosion behaviour of corium melts to be able to quantify the safety margin for ex-vessel steam explosion. This would certainly minimize the scatter of computer code predictions as well.
Despite the variety of the approaches and parameter setting philosophy, all codes calculate loads that are rather low, which might be due to relatively limited melt mass in pre-mixture, high voids and venting possibilities existing in large geometry. It is not clear which effect is dominant in the codes.

Answering this question would have required more sensitivity calculations, which could not be performed within the time frame of SERENA Phase 1. But the reactor calculations confirm that these effects have to be accounted for together with the material effects (not modelled in the codes) to analyse the reasons for the reduced energetics observed in corium experiments.

Figure 9: Calculated pressures for reactor



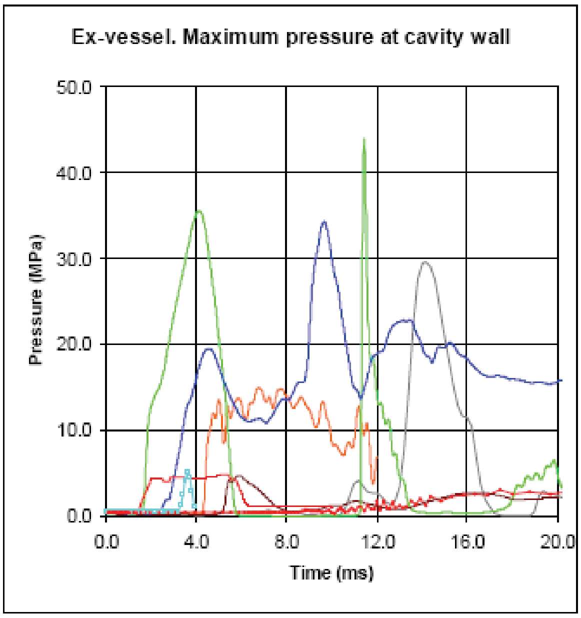
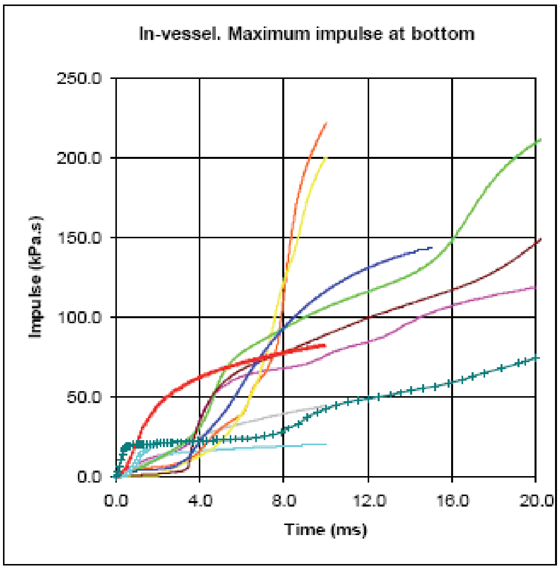


Figure 10: Calculated impulses for reactor



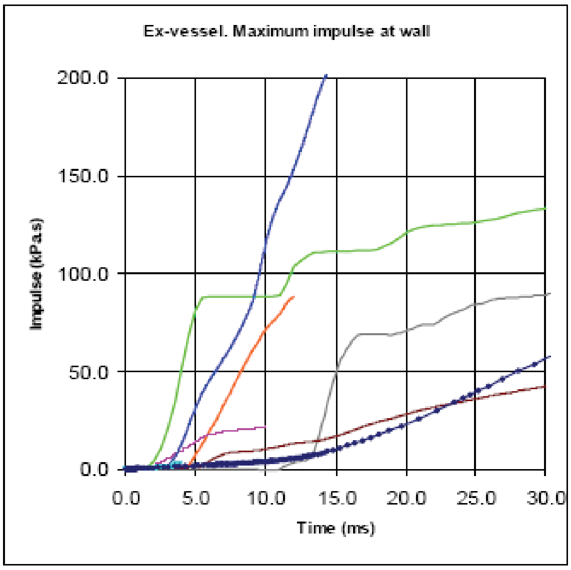
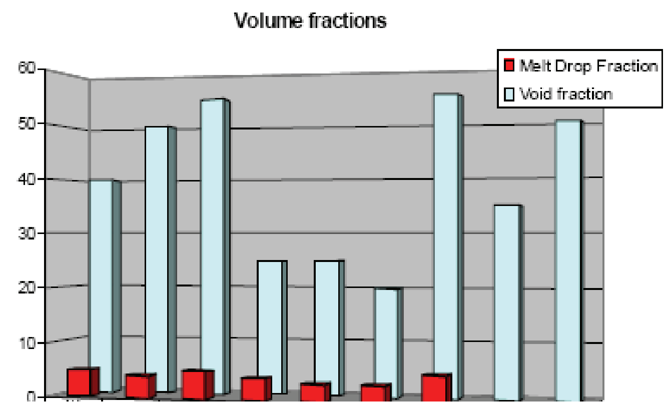
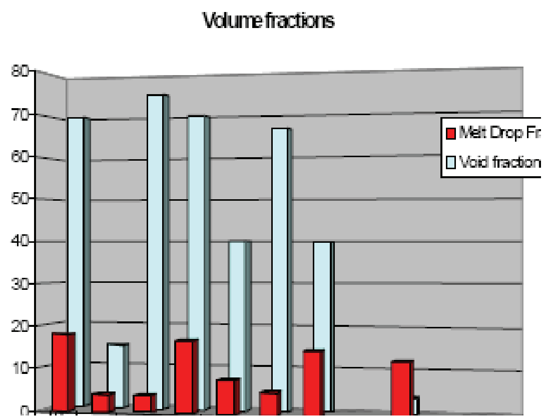


Figure 11: Global component fractions at melt-bottom contact averaged over a cylinder of height the water depth, and diameter 1 m. Ordinate: component fraction in %; Abscissa: codes. Left: In-vessel case; Right: Ex-vessel case.





5. Ex-vessel situations -SERENA (phase 2)

During Serena-phase-1, typical generic in- and ex-vessel FCI situations were simulated. The programme concluded that in-vessel FCI would not challenge the integrity of the containment whereas this cannot be excluded for ex-vessel FCI. However, the large scatter of the predictions indicated lack of understanding in some areas, which makes it difficult to quantify containment safety margins to ex-vessel steam explosion. The results clearly indicated that uncertainties on the role of void (gas content and distribution) and corium melt properties on initial conditions (pre-mixing) and propagation of the explosion were the key issues to be resolved to reduce the scatter of the predictions to acceptable levels. Past experimental data does not have the required level of details to answer the question. Concerning void content and effect, only global data are available on pre-mixing, which revealed not to be sufficient to explain the behaviour of different melts. Concerning material effect, and particularly the fact that prototypic corium melts would produce rather mild explosions, the limited number of geometrical configurations and corium compositions tested so far do not allow to generalise the conclusion neither to justify the use of specific parameters or models.
The present programme is formulated to resolve the uncertainties on these issues by performing a limited number of well-designed tests with advanced instrumentation reflecting a large spectrum of ex-vessel melt compositions and conditions, and the required analytical work to bring the code capabilities to a sufficient level for use in reactor case analyses.

The objective of the experimental programme SERENA is threefold:

1. Provide experimental data to clarify the explosion behaviour of prototypic corium melts,

2. Provide innovative experimental data for validation of explosion models for prototypic materials, including spatial distribution of fuel and void during the premixing and at the time of explosion, and explosion dynamics,

3. Provide experimental data for the steam explosion in more reactor-like situations to verify the geometrical extrapolation capabilities of the codes.

These goals will be achieved by using the complementary features of KROTOS (CEA) and TROI (KAERI) corium facilities including fitness for purpose oriented analytical activities. KROTOS is more suited for investigating the intrinsic FCI characteristics in an one-dimensional geometry. TROI is more suited for testing the FCI behaviour of these materials in reactor-like conditions by having more mass and multi-dimensional melt water interaction geometry. Validation of models on KROTOS data and verification of code capabilities to calculate more reactor-oriented situations simulated in TROI, will strengthen confidence in code applicability to reactor FCI scenarios.

5.1 Experimental facilities

TROI is a facility dedicated to steam explosion experiments located in KAERI (South Korea). KROTOS is currently being rebuilt in the CEA Centre of Cadarache (France) after having been operated for many years at JRC-Ispra by the European Commission.
Both facilities will implement technological features and advanced instrumentation to improve the quality of the test conditions and data. In particular, devices for a clean delivery of the melt and a detailed characterization of premixing (high energy X-ray radioscopy in KROTOS and electric tomography in TROI) will be implemented.
Both facilities have ability to use a wide variety of fuel materials and advanced capabilities for post-test physical and chemical analyses of the debris. Such analyses recently revealed important different physico-chemical behaviours between alumina and corium, which have to be investigated further to determine their possible contribution to the reduced energetics observed with corium melts with respect to alumina melts and help understanding the "material effect".

The KROTOS facility features rather one-dimensional behaviour of mixing and explosion propagation. This allows a clear characterisation of mixing behaviour (melt and void distribution) and escalation and propagation behaviour (given path starting from bottom triggering), with the respective possibilities of direct checking of code results. The wider test section in TROI allows more prototypic sideways spreading of the mixing region (void and melt) and multi-dimensional pressure wave propagation. This addresses questions of more extended, less voided (easier steam release) but also leaner mixtures as well as limitations in the explosion phase (explosion venting effects).
Six complementary tests in each facility are planned.

The effect of the fuel material properties will be investigated with the use of 4 different compositions. The basic oxidic corium will be 70%UO_2-30%ZrO_2, as it revealed to induce spontaneous explosions more energetic than with 80%UO_2-20%ZrO_2 in TROI conditions. Tests will be performed with standard ex-vessel conditions, i.e., a pressure of 0.2 MPa and a subcooling of 50 K.

KROTOS tests will be performed with the following common conditions (see Figure 12):

- Corium melt mass ~5 kg
- Pool depth ≤1.1 m
- Pool diameter 200 mm
- Free fall 50 cm
- New release mechanism and X-Ray radioscopy

TROI tests will be performed with the following common conditions (see Figure 13):

- Corium melt mass ~20 kg
- Pool depth 0.7 – 1.3m
- Pool diameter 600 mm
- Free fall 50 cm – 100 cm with an intermediate catcher.

Figure 12 : KROTOS facility test tube and test vessel section instrumentation

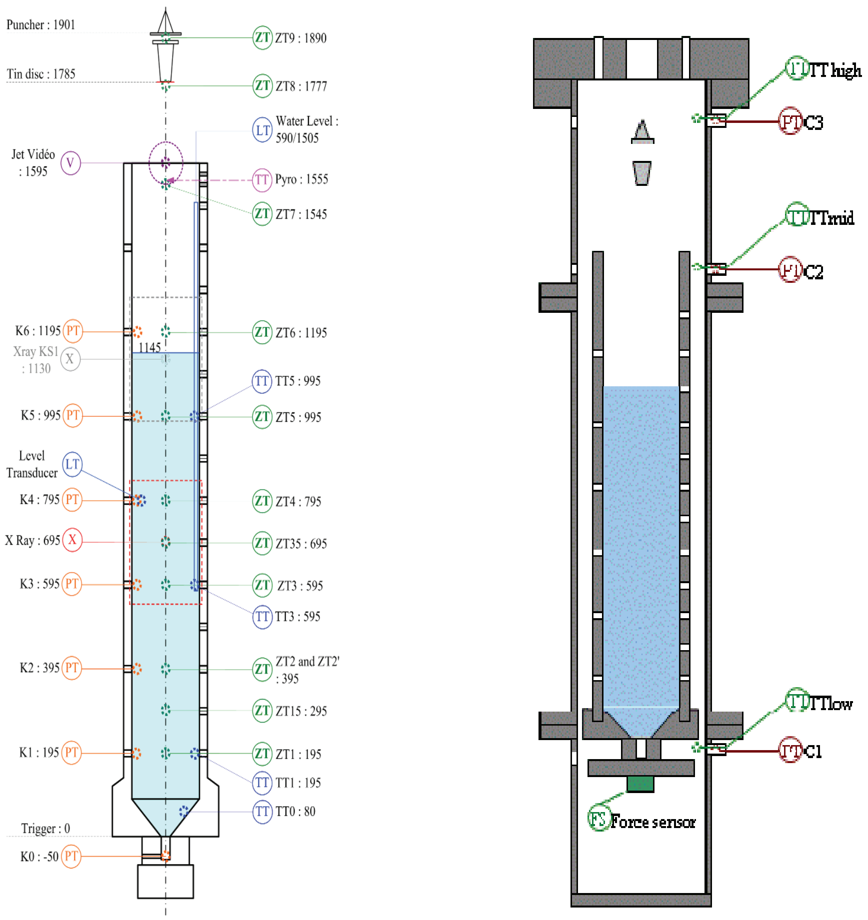
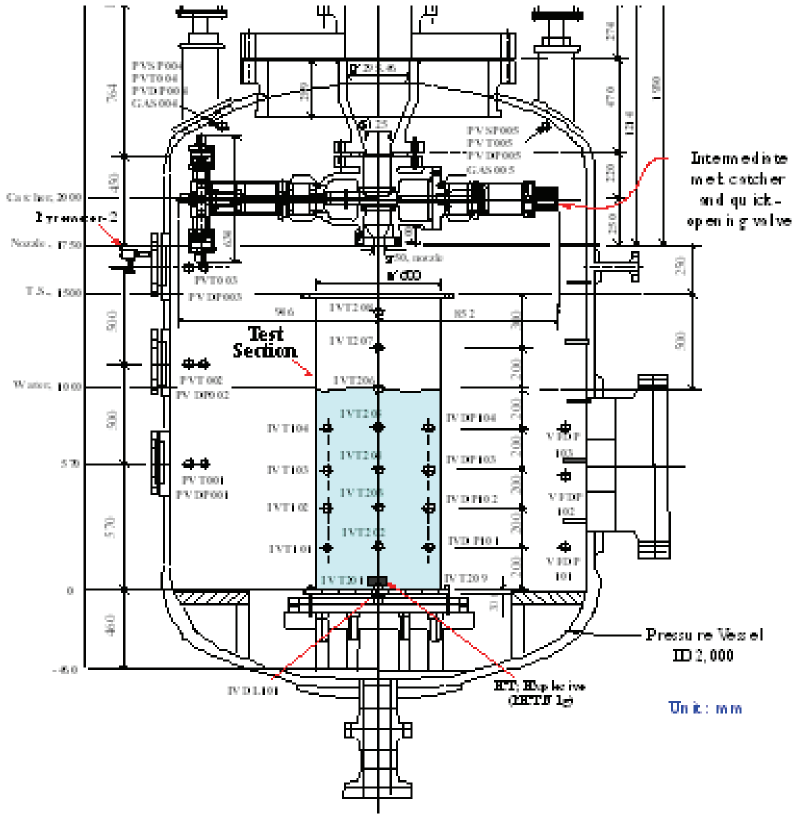


Figure 13 : TROI facility test tube and test vessel section instrumentation



5.2 Experimental test matrix

The proposed test matrix (table 3) is composed of 6 series of 2 complementary tests, one in KROTOS with 5 kg of melt and 1-D configuration, and one in TROI with 20 kg of melt and 3-D configuration. Test 1 will be performed first, but the order and final specifications of the further tests is decided as a function of the knowledge received in the previous ones. Krotos Tests 1 and 2 have been completed. Troi Tests 1 to 3 have been completed. Data analysis of the test are in progress.

Table 3 : test matrix experimental grid.

		KROTOS	TROI
	Challenging conditions (to be finalised through discussion with the partners)	Standard geometrical conditions High melt superheat High system pressure (0.5 MPa)	High system pressure (0.5 MPa) Reduced free fall (Melt jet velocity) and thick melt jet
		Mat: to be decided	
	Geometry effect Effect of geometry by comparison between KROTOS and TROI	Standard conditions: jet of diameter 3 cm	Large jet at penetration (5 cm)
		Mat 1: 70%UO_2-30%ZrO_2	
	Material effect Oxidic composition	Standard conditions	Large jet at penetration (5 cm)
		Mat 2: 80%UO_2-20%ZrO_2	
	Material effect Oxidation/composition	Standard conditions	Large jet at penetration (5 cm)
		Mat 3: 70%UO_2-30%ZrO_2 +steel +Zr	
	Material effect Large solidus/liquidus ΔT	Standard conditions. Effect of fission product: higher melt superheat	Large jet at penetration (5 cm). Failure at the bottom, considering layer inversion. *(2-5 cm)*
		Mat 4: 70%UO_2-30%ZrO_2 +FP+iron oxide+absorber materials	
	Reproducibility tests	Idem Test 3 or 4	Idem Test 3 or 4

5.3 Analytical activities

An analytical working group is established with the main aim of increasing the capabilities of the CI models/codes for use in reactor analyses by complementing the work performed in Phase-1 through ntegrating the results of the Phase-2 experimental programme. The work is oriented at fitting for purpose for safety analysis and elaboration of the major effects which reduce the explosion strength. The main tasks of the group are:

- Performing pre-, post-test calculations in support of test specification and analysis,
- Organizing a benchmark exercise with "blind pre-test" calculations for one test being performed in the second period of the project,
- Improving the common understanding of those key phenomena that are believed to have a major influence on the FCI process,
- Addressing the scaling effect and application to the reactor case,
- Giving specific attention to the link between FCI models/codes and general system codes (e.g. COCOSYS) or integral codes (e.g. ASTEC, MELCOR),
- Demonstrating the progress made in SERENA Phase-2 as compared with Phase-1, e.g., by repeating the “ex-vessel reactor exercise” (deviati ons shall be explained).

Acknowledgments: The authors gratefully thank OECD for the organization of the SERENA project and would like also express their gratitude to Daniel Magallon, Joint Research Center, EC for his very important contribution to this paper and to SERENA programme.

Improved Molten Core Cooling Strategy in a Severe Accident Management Guideline

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1. Introduction

As the number of nuclear power plants increases, the consequences of severe accidents at nuclear power plants draw more attention, though the probability of occurrence of severe accident is very low. It is to be noted that the risk to public health and safety is dominated by severe accidents while design bases accidents are not risk significant [1]. For these reasons, severe accident management guideline (SAMG) for the pressurized water reactor has been developed by utilities in USA [2, 3], France [4], Korea [5], and Finland [6]. In Korea, the accident management program has been implemented for the operating nuclear power plants between 2001 and 2008.

The basic philosophy of SAMG is to do one's best by using the available equipments to minimize the consequences of severe accidents under given circumstances. The experience from reviewing the SAMG by KINS (Korea Institute of Nuclear Safety) tells us that the present SAMG provides a reasonable guideline to cope with severe accidents in harsh conditions, but it is not clear whether these SAMGs could really contribute to mitigation of severe accidents or not [5].

A flooding of the reactor cavity is suggested as an accident management strategy. However, the success probability of a stabilization of the molten core is still subjective due to the complexity of the phenomena including the molten core concrete interaction and the energetic fuel and coolant interaction, which are still unresolved safety issues [7]. For these reason, a new engineering solution of a core catcher is provided for new reactors, such as VVER-1000 [8], EPR [9], and ESBWR [10] to stabilize and cool the molten core materials outside the reactor vessel.

In this paper, we would like to revisit various molten core cooling strategies implemented in a SAMG and evaluate their effectiveness considering the results of recent research results, which are accumulated more than 10 years after the birth of SAMGs. In section 2, we will briefly describe the SAMG implemented in operating plants and we will review the recent research results to examine the fundamental mechanisms behind the various strategies. In section 3, we will evaluate the effectiveness of current molten core cooling strategy and propose an improvement. Summary and conclusion are provided in section 4.

2. Implementation of a molten core cooling strategy in a SAMG

2.1 Korean Accident Management Guideline (KAMG)

As a first step, a generic severe accident management guideline called Korean Accident Management Guideline (KAMG) was developed. Then, a plant specific SAMG is developed for each nuclear power plant. As Korea has reactors originated from Westinghouse, Combustion Engineering and Framatome, the KAMG was developed by referencing the generic severe accident management strategies from Westinghouse Owner's Group (WOG)[2] and Combustion Engineering Group (CEOG) [3].

As shown in Table 1, KAMG has seven strategies. It consists of actions for monitoring the safety parameters, and actions for mitigation. Mitigating actions consist of various operator actions to restore the safety parameters to their stable ranges. Mitigating actions are described as 7 guidelines.

Table 1 Structure of the KAMG

Guideline /Strategy	Objective	Equipment
M-01: Inject into Steam Generator (S/G)	Secure RCS cooling source	S/G feed system
M-02: Depressurize the Reactor Coolant System (RCS)	Prevent high pressure melt ejection, direct containment heating, Secure core cooling	Pressurizer Pilot Operated Safety Relief Valve (POSRV)
M-03: Inject into Reactor Coolant System	Establish core cooling	Safety Injection, Charging pump
M-04: Inject into Containment (CV)	Establish in vessel retention, Prevent molten more concrete interaction	Gravity feed from Refueling water storage tank (RWT)
M-05: Mitigate Fission Product (FP) release	Reduce fission product release	Isolate the leakage Spray, Fan cooler
M-06: Control containment condition	Maintain containment integrity	Spray, Fan cooler
M-07: Control containment Hydrogen	Prevent hydrogen explosion	- H_2 recombiner - Deliberate ignition - Inerting

As an example of a plant specific SAMG, Ulchin 1&2 SAMG is illustrated in Fig. 1. Once the core exit temperature exceeds 650 ^oC , we consider that all the emergency operation procedure (EOP) actions have failed and a damage of reactor core has occurred. From that point on, the plant enters into severe accident management region and the operators will try to mitigate the accident progression by using SAMG. According to the flow chart of Fig. 1, the plant safety parameters will be monitored, and once each parameter exceeds its set-point, then the operators will refer to the corresponding guidelines and try to mitigate the accident progression.

For example, if the steam generator (SG) water level is higher than the 0.44 m Wide Range (WR), then the operator will check the next parameter, the RCS pressure. In the case when the level is lower than 0.44 m Wide Range (WR), the operator will initiate Mitigation-01 and try to increase the S/G water level based on the guidelines of M-01. Mitigation-01 describes what equipments are available to inject water into the steam generator, what will be the pros and cons of each operator action, and what parameters should be monitored to exit the guideline successfully.

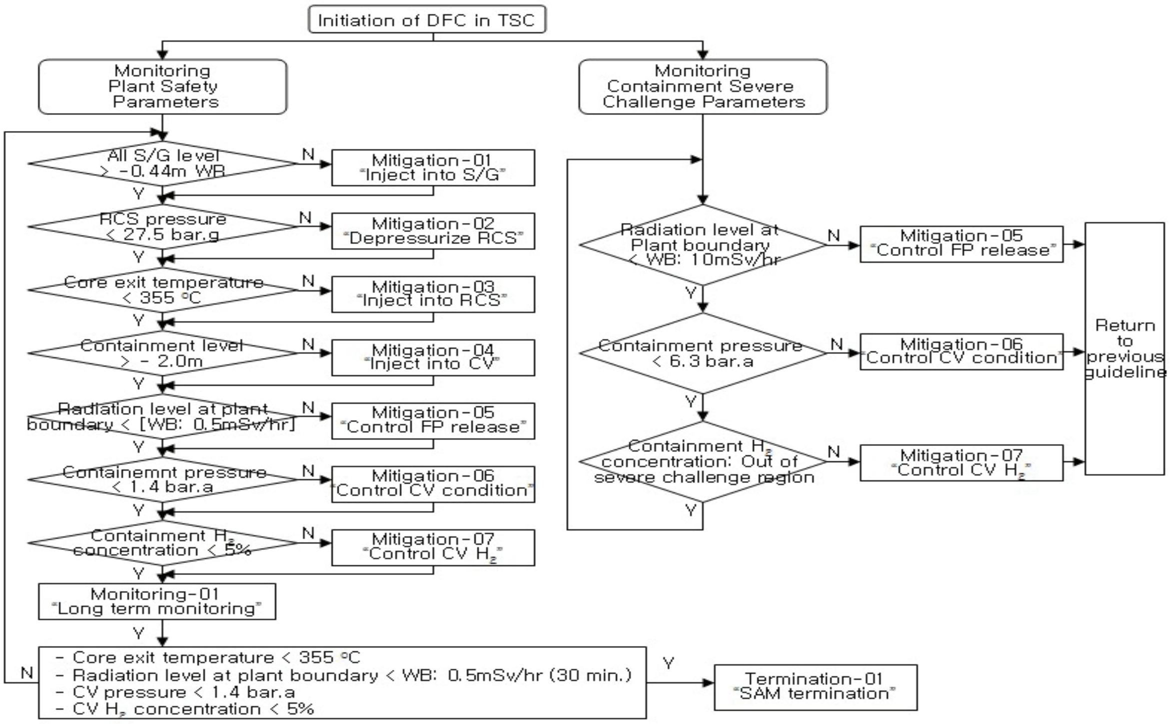

Fig. 1 Flow chart of SAMG for Ulchin 1 &2

Thus, managing a severe accident using the SAMG means that the operators will monitor 7 parameters, and in case that any parameter value becomes higher than its set-point, the corresponding mitigating guidelines will be implemented to control the safety parameters back to their safe level.

2.2 Evaluation of various molten core cooling strategies

Here, we revisit the fundamental mechanisms behind the molten core cooling strategies incorporated in the SAMG and evaluate their effectiveness considering the results of recent research which are accumulated more than 10 years after the birth of SAMGs.

2.2.1 In-Vessel Retention (IVR) by a pre-flooding of the reactor cavity

A pre-flooding of the reactor cavity is a strategy to cool the molten core from outside the reactor vessel. It is implemented for Loviisa [6], AP600 [11] and AP1000 [12] with accompanied design changes. The decay heat and sensible heat from the molten corium in the reactor vessel is transferred via the reactor vessel wall to the coolant outside the reactor vessel, which results in a boiling of water filled in a reactor cavity.

Often it is highlighted that the IVR strategy has an advantage that the molten corium is confined in a reactor vessel not inside the containment, which seems to be better in terms of public acceptance. However, it has to be noted that either confinement of molten corium inside the reactor vessel or inside the containment is technically the same as both approaches confine the radiological materials inside the containment.

Depending on the details of plant design, the success probability of the IVR strategy varies. As an example, Seiler et al. [13] indicated that the feasibility of the IVR strategy for high power reactors (i.e. above 1000 MW) is certainly questionable due to a focusing effect.

Also, some of the plants do not have water inventory to fill up the reactor cavity to submerge a substantial portion of the reactor vessel. If there is not enough water, the reactor cavity should not be filled with water before the reactor vessel failure. A wet cavity can lead to an energetic steam explosion which might challenge the structural integrity of the containment and can lead to a non-uniform spreading of the melt, which can result in a localized ablation of the reactor cavity floor.

2.2.2 Top Flooding

For ex-vessel accident sequences in which the reactor vessel fails and corium is released into the reactor cavity, an operator would flood water over the molten corium pool formed in the cavity. If a sufficient cooling is provided, the erosion of the base mat concrete would stop and it would prevent a further release of fission products into the environment. This is the basis for a provision of cavity flooding system and provision of a proper amount of spreading area in the reactor cavity. However, recent research results of OECD/MCCI project [14] indicate that stable cooling of the molten corium in the reactor cavity with top flooding even with the plant having a sufficient water supply and enough spreading area needs to be carefully evaluated.

The results of the OECD/MCCI tests were incorporated into an analytical model and the model was used to predict the coolability on a plant scale [14]. The model determined the total axial ablation depth at stabilization versus initial melt depth for various corium concrete contents. The results for the Limestone/Common Sand (LCS) concrete indicate that a melt stabilization can be achieved under one meter of axial ablation as long as the cavity is flooded before the melt concrete content exceeds 15% for initial melt depths ranging up to 40 cm. However, if the flooding is delayed past this point, the possibility of stabilization becomes more unlikely. Under the same set of conditions, the results for siliceous concrete indicate a much narrower coolability envelope.

2.2.3 Cooling of the debris bed in a pre-flooded cavity

We consider the case where the flooding of the cavity was before the reactor vessel failure. In this case, the molten corium will be discharged into the reactor cavity filled with probably sub-cooled water. This can lead to an energetic steam explosion [15] which might challenge the structural integrity of the containment in case the water pool is deep [16].

Keeping this fact in mind, a pre-flooding of the reactor cavity will be beneficial only in the case when there is a high success probability of in-vessel retention. If the water inventory is not sufficient to submerge significant amount of reactor vessel lower head, it will only increase the risk of damaging the containment. This would potentially increase the Large Early Release Fraction (LERF) because of higher probability of early containment failure due to an ex-vessel steam explosion.

3. Evaluation of molten core cooling strategies in the SAMG

We chose Kori unit 1 plant and Ulchin 1 &2 to evaluate the effectiveness of molten core cooling strategy implemented in a SAMG, as these two plants have a substantial contribution from the base-mat melt though (BMT) for the containment failure among various failure modes of the containment. The contribution from the BMT was 16.5 % and 30.8 % [17, 18] respectively. These plants have very limited capability of water supply to the reactor cavity and insufficient space for the spreading of molten corium in the reactor cavity.

The Kori-1 nuclear power plant is a two-loop Westinghouse pressurized water reactor (PWR) with 2 reactor coolant pumps and 2 steam generators at 1724 MWt. Ulchin 1&2 are 3-loop Framatome PWRs with 3 reactor coolant pumps and 3 steam generators at power of 2775MWt. The Ulchin 1&2 and Kori-1 SAMG were developed from the same framework as that of KAMG, while modifications to reflect the plant specific features were incorporated.

Among these 7 mitigating guidelines, M-04 is a guideline to handle the debris coolability. The purpose of this strategy is 1) to establish a means of a core cooling, 2) to minimize the amount of corium concrete interaction and 3) to reduce the fission product release. To achieve these objectives, operators will inject water into the containment cavity using either a spray or a gravity feeding from the refueling water storage tank (RWT).

In the following we evaluate the strategy to confirm whether the objectives are achievable for Kori-1 plant, and Ulchin 1&2 plant or not.

The first objective M-04 strategy is to delay the failure of the reactor vessel by in-vessel retention through an ex-reactor vessel cooling. For this strategy to be successful, the reactor cavity should be filled with water up to the level of a hot leg and a proper steam flow path should be established between the reactor vessel wall and the insulation structure.

We found that even all the water source available in the system having been poured into the cavity, it can barely reach the very low part of the reactor vessel both for Kori unit 1 and Ulchin 1-&2 nuclear power plant. The method of filling the reactor cavity is either by a spray or a gravity feeding from the RWT.

The second objective of the M-04 strategy is to cool the debris by injecting water into the cavity. This strategy is based on the idea that having water in the cavity would be better than having no water to cool the corium. But this expectation does not guarantee that having water in the cavity could cool the corium and thus mitigate the corium concrete interaction.

3.1 A quantitative plant analysis for the evaluation of M-04

As discussed in section 2, the success probability of a molten core cooling depends highly on the plant parameters, such as the initial core power, the amount of molten corium inventory, the cavity floor area, the type of concrete, and the containment pressure at the time of flooding. And the success probability of the in-vessel retention depends on the initial core power, the amount of available water to flood the cavity, and the formation of flow path around the reactor vessel. These points need to be elaborated to convince the operator that his actions will lead to a permanent stabilization of a plant. However, a quantitative analysis on a plant scale has hardly been performed before to support the effectiveness of SAMG.

In the following, we have performed an analysis at a plant scale using a severe accident analysis computer code MELCOR [19] and applied the recent OECD/MCCI results [14] to evaluate the effectiveness of SAMG for cooling the molten corium by top-flooding the reactor cavity. We chose Kori-1 nuclear power plant. MELCOR analysis models for the RCS and containment of Kori -1 is illustrated in Fig. 2. The accident scenario chosen was a Station Black Out (SBO) without any operator action.

According to the MELCOR analysis summarized in Table 2 and 3, the reactor vessel breach occurs at 23,650 seconds and after that time the corium is poured into the reactor cavity. The code provides information on the amount of corium in the cavity and also on the erosion process as a function of time. Using this information and geometric data of the reactor cavity, we can calculate the initial collapsed melt depth, the ratio of concrete content, and finally the ablation depth using the information from figure 6-9 of OECD Report [16]. Comparing the ablation depth with the intact base mat depth, we can calculate the margin before base mat melt through.

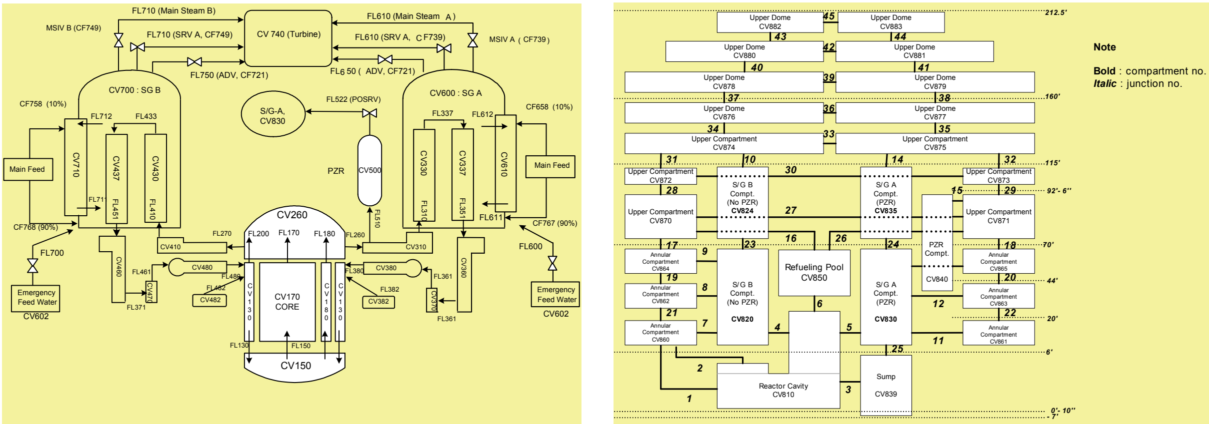

Fig.2 MELCOR analysis models for the RCS and containment of Kori -1

Table 2. SBO Sequence of Events for Kori-1

Time (sec)	Top Events
0	Reactor Trip
5,350	SG Dry out
9,852	Core Uncover
10,510	Core Dry out
14,530	Clad Melting
23,640	UO_2 Relocation to Lower Head
23,650	Lower Head Failure
40,209	Cavity Dry out

Table 3. MCCI Condition in the reactor cavity

	At 10 hrs after SBO	At 24 hrs after SBO
Corium Mass in Cavity	102.8 ton	166.5 ton
Concrete Mass Eroded	38.2 ton	122 ton
Ratio of Concrete Content	27%	42%
Melt Depth (by MELCOR	0.47m	1.17m
Remaining Base mat Depth	1.953m	1.333m

The type of concrete for Kori-1 cavity is Limestone Command Sand (LCS) and the average design depth of base mat concrete is 2.133 meters. Now suppose the operators have recognized that the vessel has breached and thus decide to implement the M-04 guideline at 36,000 seconds after initiation of SBO. At that time, the melt depth is 0.47m and the ratio of concrete content is 27% as is described in T~~t~~able 2. Fig. 6-9 of OECD report indicates that the total ablation depth at stabilization by a top flooding will be around 110 cm for the case with the initial collapsed melt depth of 40 cm and the concrete content of 20%. Therefore, we could say that we have 0.853m of margin before melt through. This would indicate that if the operators flooded the cavity at about 10 hours after SBO initiation, the melt in the reactor cavity can be stabilized.

On the other hand, at 24 hours after the initiation of SBO, the melt depth is 1.17m and the ratio of concrete content is 42%. Using the value for melt depth of 0.4m and the concrete content of 20% again, we can say that the margin to melt through is far less than 0.233m. This can be interpreted that even though the operators flood the reactor cavity, it cannot stabilize the melt.

We have uncertainties in this calculation, but it is clear that an operator action of top-flooding should be performed as soon as possible but not before the reactor vessel breach. Otherwise it can lead to an unexpected steam explosion or molten core cooling cannot be achieved. One of useful information from the above analysis is that we have reasonable amount of time for evacuation, as the containment failure does not occur within 24 hours, if operator would not flood the cavity before the reactor vessel breach. .

3.2 Evaluation of M-04 in conjunction with optimized M-02

In the RCS depressurization strategy M-02 of the current SAMG, it is recommended that an operator needs to depressurize rapidly RCS pressure below 2.75 MPa using all pressurizer (PZR) pilot operated safety relief valves (POSRVs) for high pressure accident like SBO.

However, it was pointed out that it can result in an excessive discharge of the Safety Injection Tank (SIT) water and thus can lead to a rapid core damage [20]. It triggered an effort to optimize the RCS depressurization strategy M-02. Here, we compared the current depressurization strategy and an optimized RCS depressurization strategy to look at the effect of optimized depressurization strategy on the molten core cooling performance.

A MELCOR analysis was performed for Ulchin 1&2 in a similar manner as that of Kori 1. According to the MELCOR calculation, the reactor vessel breach occurs at 6.2 hours when we open 3 POSRVs, while it breaches at 9.2 hours when we depressurize by optimum strategy. The pressure at the time of vessel breach is 0.39 MPa and 0.76 MPa respectively which are far below the DCH cut off pressure of 1.7MPa.

The code provides information on the amount of corium in the cavity and also on the erosion process as a function of time. Using this information and geometrical data of the reactor cavity, we can calculate the initial collapsed melt depth, the ratio of concrete content, and then the ablation depth using the information of Fig. 6-9 of OECD report [14]. Comparing the ablation depth with the intact base mat depth, we can calculate the margin before base mat melt through occurs.

The probability of power recovery at 13 hours is 98 %. So even if our depressurization strategy during SBO could not delay the vessel breach for longer than this time, but if we could show that the ex-vessel corium could be cooled by top-flooding at the time where the probability of power recovery is 98 %, our accident management is sufficiently reliable.

The type of concrete for Ulchin unit 1 is Limestone Command Sand (LCS) and the average design depth of base mat concrete is 3.0 m. Now suppose the power has recovered at 13 hours for the case of optimum discharge strategy. At that time, the melt depth is 0.472m, the ratio of concrete content is 24% and the remaining thickness of base mat is ~2.8m as is described in Table 4 below. Comparing the values with MCCI result [16], it is slightly out of range of experimental validity, but it is still within the marginal point of data and we can deduce from the data that the ablation depth to stabilization for this case is ~ 1.2 m. Comparing this value with the remaining thickness of 2.8 m, it is quite sure that the corium could be cooled in case top-flooding becomes possible at this time even if we recognize the uncertainties in our calculation.

The other case of depressurizing by opening 3 POSRVs is described also in the Table 4. But in this case, the values are outside the region of experimental validity and by extrapolating the data just to have a rough estimation, the ablation depth to stabilization for this case is larger than 2.0 m. Comparing the value with the remaining thickness of base mat ~2.67 m and taking into account the uncertainties inherent in the calculation, it is not possible to say that the debris coolability can be assured.

The above results clearly show that the current depressurization does not guarantee coolability of ex-vessel debris by top-flooding in case of Ulchin 1&2. On the other hand, the optimized depressurization delays the vessel breach time and the debris coolability is guaranteed when the power recovers in 13 hours, whose probability was estimated as 98 %. This justifies our efforts of developing an optimum depressurization strategy in conjunction with M-02.

Table 4 Summary of Top Flooding initiated at 13 hours

Depressurization using Current Strategy	Depressurization using Optimum Discharge
Initial Condition in Cavity corium mass in cavity ; 169 ton corium height ; 0.6m concrete content ; 32% remaining thickness of basemat ; ~2.67m	Initial Condition in Cavity corium mass in cavity ; 152 ton corium height ; 0.472 m concrete content ; 24% remaining thickness of basemat ; 2.8m
Out of range of data applicability, extrapolation gives rough estimation	within the marginal point of data
Results ablation depth to stabilization; >2.0m	Results ablation depth to stabilization ; ~ 1.2m
Uncertainty renders the coolability not guaranteed	Sufficient margin for the coolability guaranteed

3.3. Improvement suggested

The analysis above gives us three insights about what is lacking in the present strategy. The first one is we need an appropriate way to detect either the breach of the reactor vessel or discharge of corium into the reactor cavity. It would be very helpful if we can put in a proper instrumentation to detect a breach of a reactor vessel and the subsequent corium discharge into the reactor cavity.

The second one is the need for a calculation aid, which would give us a direction as to whether we should initiate a pre-flooding or a post-flooding. If there is little chance of delaying the failure of a reactor vessel by a pre-flooding, for example, due to a shortage of the water inventory or an inadequacy of the reactor vessel and the insulator geometry in providing steam flow path, there is no reason to pre-flood the cavity.

The last one is that when an operator wants to depressurize the reactor coolant system, the operator should carefully consider an optimal number of valves to be opened relying on a calculation aid, as it can affect the timing of reactor vessel breach and coolibility of the molten corium in a reactor cavity. Therefore, a calculation aid need to be provided for implementing the M-02 strategy to determine the optimal capacity of depressurization, as it can affect the timing of vessel breach and coolibility of the molten corium outside the reactor vessel.

3.4 Engineering solutions for the molten core cooling

As discussed above, the success probability of the severe accident management strategy depends highly on the severe accident scenarios and plant specific conditions. There are chances that the proposed strategy may not work. It is recommended to put in a dedicated engineering solution to stabilize the molten corium discharged into the containment for a new reactor.

As examples, EPR and VVER already took engineering solutions for the molten core cooling to ensure the coolability of molten core. In the EPR and VVER core catcher concept, the decay heat from the molten corium pool is removed indirectly either from the water flow channel provided along the surface of the reactor pit or core catcher vessel wall. Therefore, if we have a limited space available for the cooling channel in the reactor cavity, indirect cooling from the bottom may not be adequate for the cooling of molten corium pool, especially for the operating reactors.

There is another concept, which is not implemented in the real plant yet. COMET concept is based on water injection into the melt layer from the bottom. The water is forced up through the melt, the resulting evaporation process of the coolant water breaks up the melt and creates a porously solidified structure from which the heat is easily removed [21]. This mode of molten core cooling is expected to be adequate even in the case where limited space is available for melt spreading in the reactor cavity. Therefore, the bottom injection concept could be applicable for both the future reactors and operating reactors.

4. Summary and conclusion

From the evaluation of the molten core cooling strategies implemented in the SAMG of an operating plant, it was observed that the current SAMG has weak points in handling the cooling of the molten core either inside the reactor vessel or in side the reactor cavity. To improve the current SAMG, it was suggested that we need an appropriate way to detect either the breach of the reactor vessel or discharge of corium into the reactor cavity and we need a calculation aid which would give us a direction as to whether we should initiate a pre-flooding or a post-flooding. Also it was suggested that an optimal choice of depressurization capacity would delay the timing of the reactor vessel breach and increase the coolability margin of the molten corium in a reactor cavity, which can be easily implemented by a calculation aid.

Acknowledgment

This work was supported by Nuclear Research & Development Program of the Korea Science and Engineering Foundation (KOSEF) grant funded by the Korean government (MEST).(grant code: M20702040004-08M0204-00410)

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OECD/SARNET WORKSHOP ON IN-VESSEL COOLABILITY
October 12 – 14, 009

Main Outcomes

B. Clément (IRSN), J. Birchley (PSI), H. Löffler (GRS), W. Tromm (KIT),
A. Amri (OECD/NEA)

Note: please see the presentation. No paper is provided

Simulation of Ex-Vessel Debris Bed Formation and Coolability in a LWR Severe Accident

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1. Introduction

Severe accident management strategy for Swedish type BWRs adopts reactor cavity flooding for termination of ex-vessel accident progression. It is assumed that core melt materials ejected from the reactor vessel into a deep pool in the reactor cavity will be fragmented, quenched and will form a porous debris bed coolable by natural circulation.

A criterion generally accepted for successful long-term cooling of the porous medium with decay heat release is that the flow rate of coolant through the debris bed should be sufficiently high, so that no local dryout occurs. The possibility of local dryout occurrence in the natural circulation (gravity-driven) flow is determined by the distribution of coolant and vapor volume fractions which depend on debris bed geometry, as well as on the properties of the porous medium (particle size, porosity, homogeneity etc).

In the studies on debris bed coolability available so far, the heat-releasing porous medium was usually assumed to be homogeneous with some prescribed shape, porosity and effective particle size. An assumption of a flat layer[1] enabling the use of one-dimensional approach was used in the first works on coolability of heat releasing porous materials. In this case, counterflow conditions are established in the debris bed, with vapor moving upwards, and liquid coolant flowing downwards. The flooding limit condition, under which the balance between these two flows no longer can be achieved, was taken for the calculation of dryout heat flux (DHF), i.e., the maximum heat release rate per unit area of the debris bed which can be cooled by top flooding.

Further studies of debris bed coolability addressed the multidimensional effects, mostly in the 2D axisymmetric problem statement. Contrary to the case of 1D flat porous layer, there is a possibility that a natural circulation loop can be established in the 2D debris bed with regions of co-current flow of liquid coolant and vapor due to water ingress from the side surface of the bed. It was also found that in the presence of a downcomer which facilitates ingress of the coolant into the debris bed, substantial increase in the DHF is observed in comparison with the top flooding case.

Despite the progress in the experimental and theoretical study of the debris bed coolability, some important issues remain uncertain and require more comprehensive research. In all the coolability studies available so far, the shape of the debris bed was specified ad-hoc, e.g., a cylinder[2]

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(with a possible annular downcomer[3]), cone, or Gaussian-shaped heap[4]. However, to obtain a realistic debris bed shape for a particular scenario of melt release, the process of the debris bed formation has to be modeled, taking into account that sedimentation and packing of the debris particles is strongly affected by natural convection flows developing in the pool.

In the DEFOR-A experimental and analytical studies[5,6] possibility of a "cake" formation on top of the debris bed was shown to be important for plant accident scenarios. Also, experiments have shown that solid particles formed by freezing of the melt, can possess internal void space (encapsulated porosity). Due to the encapsulated porosity, the effective mass (and, therefore, effective density) of the particles is reduced. As a result, for the same mass of corium, the volume of debris bed increases, while the average heat release rate per unit volume of porous medium is decreased accordingly. The implications of these two factors on the debris bed coolability have not been studied so far[3].

In the current paper, numerical simulations of the process of ex-vessel debris bed formation and coolability in a water pool are presented. Two main problems are addressed: transient formation of debris bed upon gradual release of corium, and parametric studies of coolability of a fixed-shaped debris bed, including the effects of "cake" and encapsulated porosity. In the latter case, coolability limits of the heap-shaped debris bed are compared with those for a flat uniform top-flooded porous layer having the same mass of solid material, but covering the whole base-mat of the pool.

2. DECOSIM code

DECOSIM (Debris COolability SIMulator) is an axisymmetric computational code which enables coupled calculation of transient multiphase flows in the pool and in the debris bed, including sedimentation of dispersed particles, their accumulation on the pool bottom and growth of the debris bed[4,7]. Here, an overview of the modeling approach is presented.

Two distinct flow subregions are considered in DECOSIM, namely, the porous medium flow in the debris bed, and free turbulent flow outside it. Saturated conditions are assumed, and two-fluid model is applied to liquid coolant and vapor phases. In the porous medium, the flow is dominated by the balance between the pressure gradient and drag, while in the free flow full momentum equation is solved for each phase. To facilitate numerical implementation, the equations are formulated in a unified manner, with relevant terms set to zero depending on whether the current point is in the porous medium or in the ambient flow.

The system of governing equations of phase continuity and momentum takes the form

$$\frac{\partial(\varepsilon\rho_i\alpha_i)}{\partial t} + \nabla(\varepsilon\rho_i\alpha_i\mathbf{U}_i) = -\Gamma_i \tag{1}$$

$$\varepsilon\rho_i\alpha_i\left(\frac{\partial\mathbf{U}_i}{\partial t} + (\mathbf{U}_i\cdot\nabla)\mathbf{U}_i\right) = -\varepsilon\alpha_i\nabla P + \varepsilon\alpha_i\nabla\boldsymbol{\tau}_i + \varepsilon\alpha_i\rho_i\mathbf{g} - \mathbf{F}_{is} - \mathbf{F}_{ij} \tag{2}$$

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where ε is the porosity, ρ_i is the phase density, α_i is the volume fraction ($\alpha_l + \alpha_v = 1$), $\mathbf{U}_i$ is the phase velocity, Γ_i is the phase transition rate, P is pressure, $\mathbf{g}$ is gravity acceleration, $\boldsymbol{\tau}_i$ is the stress tensor due to turbulent viscosity, $\mathbf{F}_{is}$ is drag due to solid material, $\mathbf{F}_{ij}$ is the interphase drag, the subscripts $i = l, v$ denote the liquid and vapor phases, respectively.

In the porous medium, the left-hand side of Eq. 2 is set to zero, together with the second term on the right-hand side (viscous stress), $\mathbf{F}_{is}$ and $\mathbf{F}_{ij}$ are taken from the respective porous and interphase drag model. Namely, for the porous drag, Ergun's equation containing linear and quadratic terms in the phase velocity is used, with relative permeabilities and passabilities depending on phase volume fractions. The interphase drag is either omitted, or taken into account, depending on the choice of the model[2]. The evaporation rate $\Gamma = \dot{Q}/\Delta H_{ev}$ (with $\Gamma_l = \Gamma$ and $\Gamma_v = -\Gamma$) is evaluated from the heat release rate per unit volume of porous material $\dot{Q}$ divided by the latent heat of evaporation. As the volume fraction of liquid becomes low ($\alpha_l \le 0.05$), Γ is ramped to zero linearly with α_l, to account for the reduction in the particle area wetted by water for low liquid contents. Note that we assume the saturation conditions and energy equation for the solid material is not considered. The dryout criterion is, therefore, based on the void fraction evaluation: it is assumed that dryout occurs as soon as the maximum void fraction in the porous medium reaches the value of $\alpha_v \ge \alpha^* = 0.99$.

In the ambient flow we have $\varepsilon = 1$, the porous drag terms $\mathbf{F}_{is} = 0$, while the interphase drag terms $\mathbf{F}_{ij}$ are taken from the liquid-gas drag correlations accounting for different flow regime depending on the void fraction (see details in[4,5]). The turbulent viscosity stress tensor $\boldsymbol{\tau}_i$ is calculated only in the liquid phase, the turbulent viscosity is obtained from the $k-\varepsilon$ model of turbulence modified by the introduction of volume fraction α_l. The equations for the turbulent kinetic energy and its dissipation rate are solved throughout the computational domain, but in the porous medium all their transport and source terms are ramped to zero, so that both turbulent quantities just remain equal to some small background values.

Lagrangian model is used for the melt particles when DECOSIM is run in the mode where formation of debris bed is considered. Each *k*-th melt particle is characterized by its position vector $\mathbf{r}^k$, and velocity $\mathbf{U}_m^k$. For each melt particle, the following equations are solved:

$$\frac{d\mathbf{r}^k}{dt} = \mathbf{U}_m^k, \tag{3}$$

$$\rho_m \frac{d\mathbf{U}_m^k}{dt} = -\mathbf{F}_{lm} - \mathbf{F}_{vm} - (\rho_m - \rho_a)\mathbf{g} \tag{4}$$

where $\rho_a = (1-\alpha)\rho_l + \alpha\rho_v$ is the void fraction-weighted ambient density. The drag forces acting on the melt particle due to its interaction with *i*-th phase ($i = l, v$) are calculated using appropriate correlations for a spherical particle drag coefficient[4,5]. To account for the turbulent dispersion of particles, a random walk model is used in which the effects of turbulence are modeled by taking the effective velocity of liquid phase as a sum of its average value calculated from the momentum equation (2) and a fluctuating component:

$$\hat{\mathbf{U}}_l = \mathbf{U}_l + \mathbf{U}_l' \tag{5}$$

The fluctuating component $\mathbf{U}_l'$ is modeled by a random vector with uniform angular distribution; each of its three components has a Gaussian probability distribution with zero mean value and variance determined by the local turbulent kinetic energy. The frequency of picking up new fluctuating component value is determined by the turbulent time scale.

An important feature of the problem of debris bed formation is its multiscale nature. Namely, the characteristic time scale of a severe accident scenario with gradual corium melt release can be as long as several hours, while the characteristic time scales of the phenomena which affect the debris bed formation are orders of magnitude smaller. For example, the time taken by the particles to reach the pool bottom, or a characteristic time for the establishment of large scale natural convection flow in the water pool are of the order of dozens of seconds. Dispersion of particles by turbulent flow has a characteristic time scale less than a second.

The multiscale nature of the problem poses a computational challenge because full simulation with direct and simultaneous resolution of all time scales is impractical: to obtain a numerically stable solution for a single particle motion or for developing two-phase flow, the time steps have to be of the order of 10^{-2} sec, while the physical time of the whole process is of the order of 10^4 seconds. To overcome the challenge, a multiscale "Gap-Tooth" algorithm[7] was developed for the prediction of debris bed growth. According to this approach, particle sedimentation and flowfields are calculated only at several snapshot instants ("teeth"), separated by time intervals ("gaps") during which the debris bed growth rate is considered constant. Such an assumption is valid in the case of small mass flow rate of the melt, when the characteristic time for variation of the debris bed shape is considerably larger than the characteristic time for the establishment of natural circulation flow which drives the particle spreading. In this case, the flow in the pool develops in a quasi-steady regime, quickly adjusting to the slow changes in the debris bed state.

The main parameters of the debris bed are the total mass of corium *M*, porosity ε, and effective diameter D. It is assumed that debris bed particles are composed of corium with the specific heat release rate W (per unit mass of the material). In this study, the debris bed porosity $\varepsilon = 0.4$ was taken the same in all cases, including the cases of debris bed formation in the gradual release mode. The permeability and passability of porous material were evaluated from Ergun's law. In the case where the debris bed was formed gradually from falling particles, the melt particle diameter was used as the effective particle diameter in the debris bed. More sophisticated packaging models may be implemented in the future, especially for polydispersed particles with some prescribed size distribution.

If the particles possess some internal closed porosity ε_{int}, the effective material density is $\rho_p^{eff} = (1-\varepsilon_{\text{int}})\rho_c$, where ρ_c is the density of corium. For such particles, the effective diameter D and debris bed porosity accessible to fluid flow, ε, are assumed the same as for non-porous particles; however, with the increase in ε_{int}, the debris bed sizes increase, while the heat release rate per unit volume of debris bed is decreased accordingly. In the cases where the influence of "cake" on debris bed coolability was considered, both the permeability and passability were multiplied by the same reduction factor $\chi < 1$ in the domain occupied by the cake.

3. Debris bed formation upon gradual melt release

Consider first the simulation results obtained for the scenario of gradual melt release. The following parameters are assumed: total melt mass $M = 200$ t released over a period of 4 hours into a 9 m-diameter pool filled with water to the level of 8 m. The debris bed porosity was taken constant and equal to 0.4. Initially, a small Gaussian-shaped debris bed with the height of 0.3 m and characteristic radius of 1 m (mass of material 3.8 tones) was placed on the bottom plate of the water pool to initiate the convection. Three particle sizes were used in the calculations: $D = 3$, 5 and 10 mm, also, the specific heat release rate in corium took two values, $W = 25$ and 62.5 W/kg (which gives the heat release rates per unit volume of debris bed of 120 and 300 kW/m^3, respectively). Available combinations of these parameters allow us to study different convection intensity and degree of particle-flow interaction.

As the baseline case, consider formation of debris bed from 5 mm particles with the specific heat release rate of 62.5 W/kg. In Fig. 1, the flowfields, void fraction distributions, particle paths and instantaneous debris bed shapes are presented at four consecutive times, $t = 0.5$, 1, 2, and 4 hours. In each picture, void fraction distribution is presented by color map; the liquid velocity streamlines are shown by white color in the left part, while trajectories of ten representative particles are shown in the right side. The shape of debris bed is shown by the dashed yellow line.

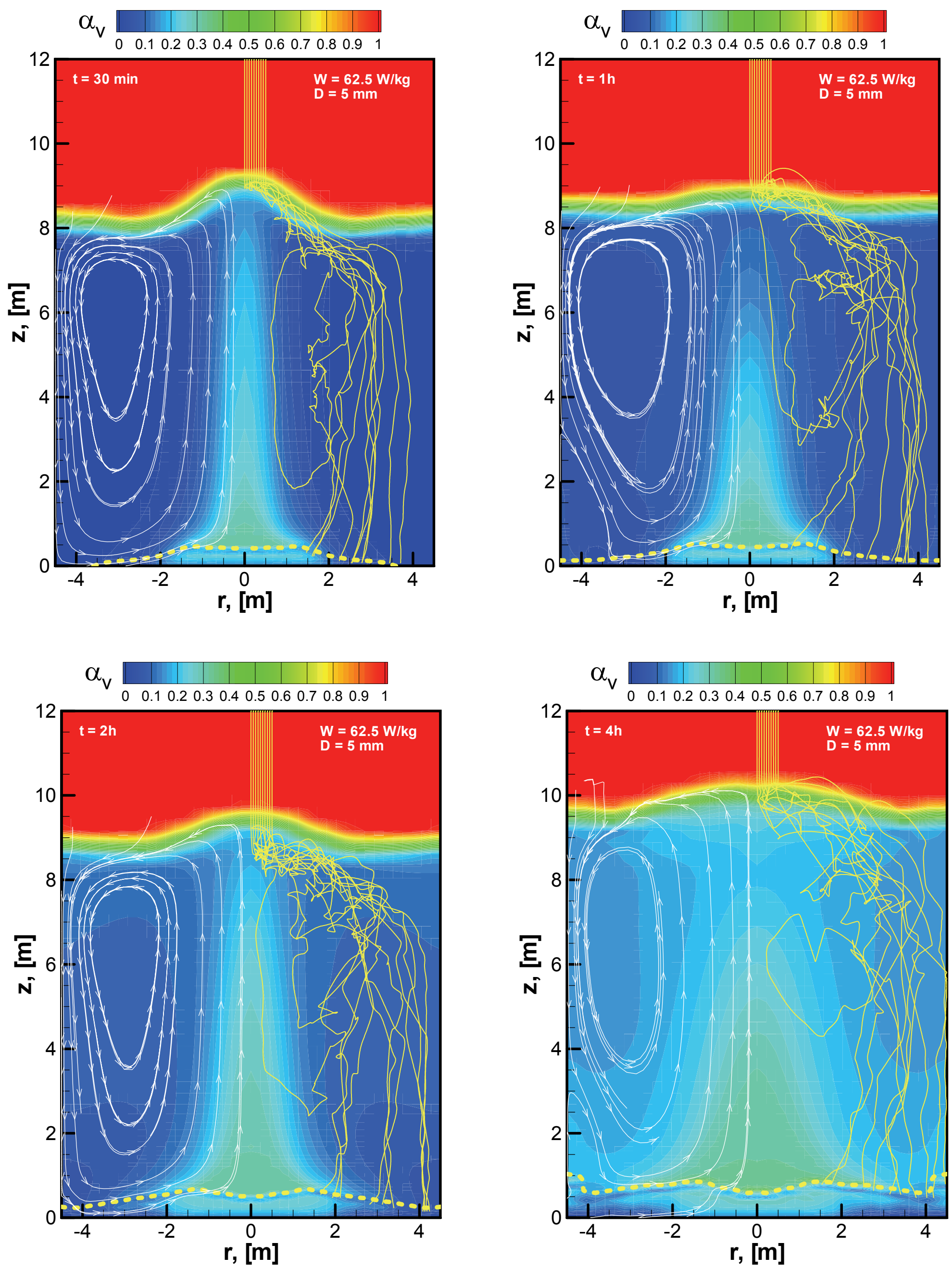

Fig. 1. Formation of debris bed from gradual release of corium melt: particle diameter 5 mm, specific heat release rate 62.5 W/kg

It is evident from Fig. 1 that particle sedimentation is strongly affected by the turbulent convective flow in the pool. Despite the fact that all particles are released from a relatively small source at the top boundary (the diameter of the source was 0.5 m), they reach the bottom plate at various locations, so that the dripping melt is spread over the pool bottom. On their way, particles tend to avoid the bubble plume generated by the already existing debris bed and land at the debris bed periphery, causing its lateral growth. Therefore, the process of debris bed growth features strong "self-organization" effects due to feedback between the available debris bed, natural convection and sedimentation of new particles affecting further growth of the bed.

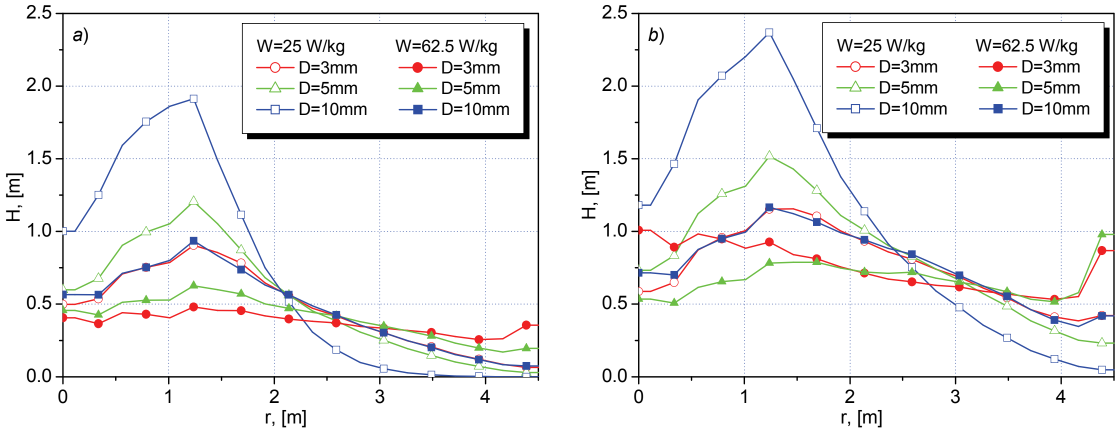

Fig. 2. Debris bed shapes at 2 hours (*a*) and 4 hours (*b*) for different particle diameters D and specific heat release rates W

To demonstrate the influence of particle diameter and specific heat release rate on the debris bed formation, in Fig. 2*a,b* the radial distributions of debris bed height are plotted at times $t = 1$ and 4 h, respectively. These graphs represent the intermediate and ultimate shapes of the debris bed after complete sedimentation of half and all melt particles. It can be seen that larger particles (cf. graphs for $D = 10$ mm) are weakly influenced by the flow and settle near the center of the pool bottom plate, although the maximum is reached at a distance of about 1.3 m, rather than at the axis, as would be the case if sedimentation was not affected by the flow. Therefore, for large particles the ultimate debris bed has a "crater"-type shape. Smaller particles (cf. $D = 3$ mm) trajectories are stronger affected by the flow and particles are carried farther away from to the side walls of the pool.

The graphs in Fig. 2 also demonstrate the effect of heat release rate on the debris bed shape. Higher heat release rate in the material leads to the development of a more intensive natural convection flow in the pool, which, in turn, results in more pronounced spreading of corium particles. The "self-organization" mechanism plays on the safety side, because it hinders accumulation of particles in narrow regions and promotes the leveling of corium melt over the pool bottom. Indeed, in all the calculations presented, the maximum void fractions in the debris bed were well below unity, i.e., conditions for dryout did not develop. Calculations of debris bed formation and coolability for higher rates of heat release (up to 200-300 W/kg) relevant to scenario with earlier reactor vessel failure are the subject for the future work; however, the tendency to the formation of a nearly flat debris bed allows us to assess the coolability using the results to be presented in the following section for an idealized 1D debris bed corresponding to uniform distribution of particles over the pool base-mat.

4. Parametric studies of debris bed coolability

In this section, we consider effects of different debris bed parameters on coolability of heat-releasing porous medium. The following set of parameters is taken as baseline: total mass of melt in the debris bed $M = 200$ t, debris bed porosity $\varepsilon = 0.4$, total volume of debris bed $V = M / \rho_c (1-\varepsilon) = 41.7\ \mathrm{m}^3$. The debris bed had a Gaussian shape with the maximum height at the axis of symmetry $H = 2.49$ m and characteristic width $R_0 = 2.3$ m. The pool has the diameter of 9 m and filled with water to the level of 8 m, the system pressure above the water surface was taken to be 3 bar. To save the computational efforts, a simplified model for the outer flow was used: the space beyond the debris bed was filled with passive (not releasing heat) fictitious porous medium with the porosity 0.9 and effective particle diameter 0.1 m (similar approach was used in the coolability studies[2]). The modified Tung-Dhir model[2] was used for the drag in the porous medium, including interphase drag.

The main variable parameters in the calculations presented below are the particle diameter D, heat release rate per unit mass of corium material W, internal porosity of particles $\varepsilon_{\mathrm{int}}$ and permeability reduction factor (only for calculations of "cake") χ. To specify the mean particle diameter D, debris size distributions obtained in DEFOR-A experiments were used. Different methods of averaging the size distribution gave the following effective diameters: mean mass diameter 3.92 mm, mean area diameter (Sauter) 2.65 mm, mean length diameter: 1.51 mm. Since it is so far unclear which averaging gives more appropriate effective diameter which must be used in the drag laws in the porous medium, two values were chosen for the particle diameter in the present DECOSIM coolability studies: D = 2 and 3 mm.

The heat release rate per unit mass of the material, W, was varied in the range between 50 and 300 W/kg, which is representative of corium with different contents of molten structural materials and different times of reactor pressure vessel failure after reactor shutdown. To find out the coolability limits, the specific heat release rate was increased sequentially with the step of 10 W/kg, and at each power level the calculation run until a steady state was reached (corresponding to a coolable debris bed), or the vapor volume fraction increased to unity (non-coolable debris bed).

Consider first the effect of heat release rate on the void fraction distributions in the debris bed (no internal porosity or cake in this case). In Fig. 3, the results obtained for the particle diameter $D = 2$ mm are presented for the specific heat release rates $W = 130$, 150 and 180 W/kg. One can see that the highest void fractions are observed in the top part of the debris bed, where dryout occurs for the heat release rates above 160 W/kg. Similar results obtained for the particle diameter $D = 3$ mm are presented in Fig. 4 for $W = 200$, 220 and 240 W/kg. Evidently, for larger particle diameter cooling is easier and dryout occurs at higher specific powers (above 230 W/kg).

To study the effect of debris bed shape on its coolability, calculations were carried out for a Gaussian-shaped debris bed having the height of 2 m and characteristic radius of 2.6 m. This debris bed has the same volume as the baseline one, but is lower by 0.5 m. Calculations have shown that this change in the debris bed shape have pronounced effect on its coolablity: for 2 mm particle diameter dryout occurred at the specific heat release rate $W = 220$ W/kg instead of 160 W/kg in the baseline case. For 3 mm particles, no dryout occurred in the 2 m high debris bed in the whole specific power range studied (up to 350 W/kg).

Even better coolability is observed when the porous material is distributed as a flat layer over the whole base-mat of water pool. Due to large diameter of the pool (9 m), the height of the flat debris bed is quite low (60 cm). Calculations showed that, despite the counterflow conditions for coolant and vapor, the flat layer is much more coolable than the equivalent heap-shaped debris bed: in the least favorable case of 2 mm particles the dryout occurred at the specific power of 370 W/kg, whereas for 3 mm particles dryout occurred at the power levels as high as 530 W/kg. These figures agree well with the experimental and theoretical results on top-flooded debris bed coolability (see review[2]), according to which the dryout heat flux for a debris bed of 2 mm particles is as high as 600-1000 kW/m^2. For the total pool bottom area of 63.6 m, this gives the heat release rates of about 190-320 W/kg of corium.

The results of above calculations are summarized in Fig. 5, where the maximum void fraction in the debris bed is plotted against the heat release rate in the material.

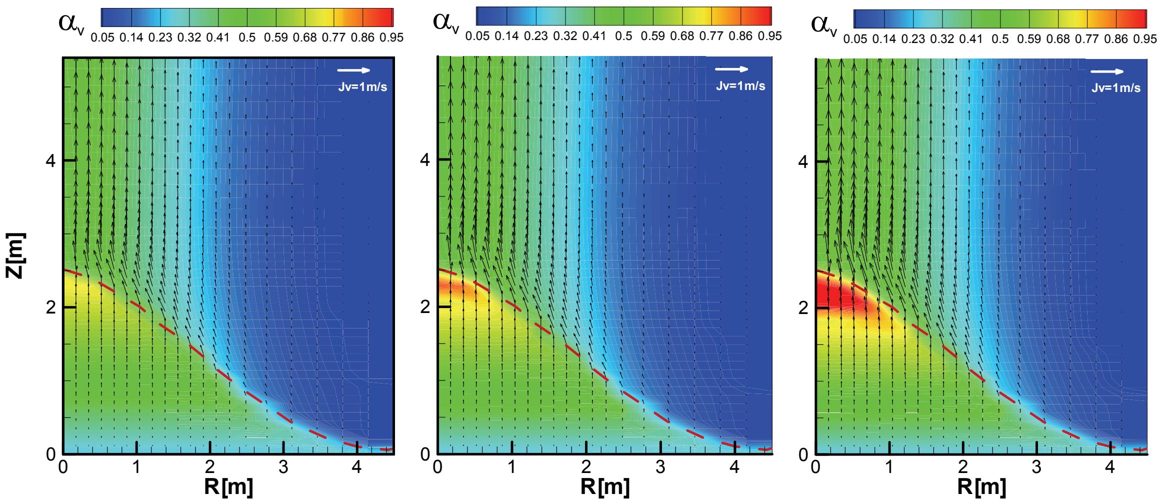

Fig. 3. Void fraction distributions in a Gaussian-shaped debris bed with particle diameter $D = 2$ mm at heat release rates $W = 130$, 150, and 180 W/kg (left to right); superficial gas velocities are shown by vector field, dashed line shows the boundary of debris bed

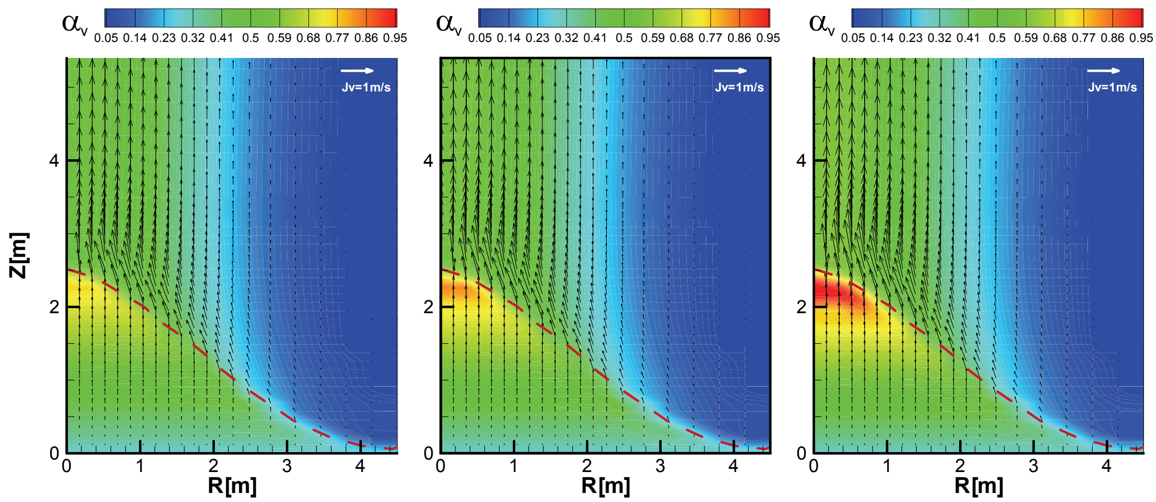

Fig. 4. Void fraction distributions in a Gaussian-shaped debris bed with particle diameter $D = 3$ mm at specific heat release rates $W = 200$, 220 and 250 W/kg (left to right); superficial gas velocities are shown by vector field, dashed line shows the boundary of debris bed

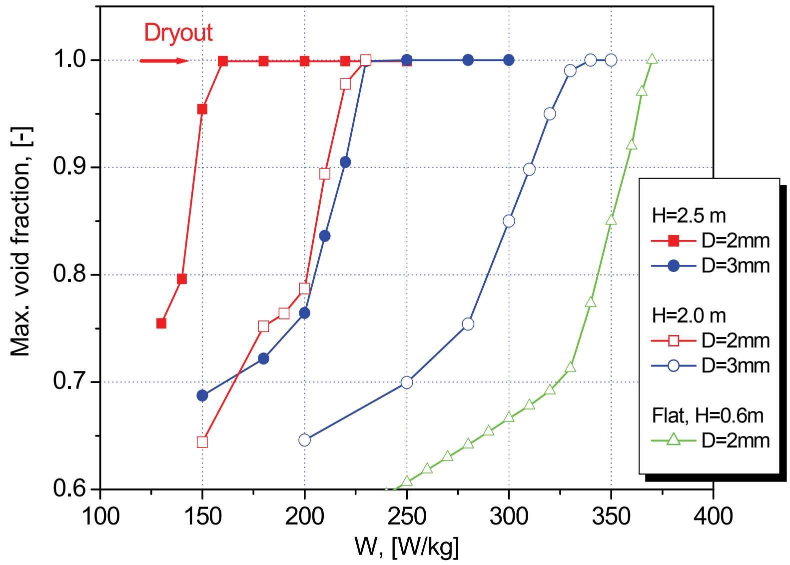

Fig. 5. Maximum void fraction in the debris bed vs specific heat release rate for different particle sizes and debris bed geometries

Simulations of a debris bed with a “cake” were carried out by decreasing the permeability and passability in a region occupying the top 0.9 m of the debris bed by a factor of $\chi = 0.5$ (moderate decrease in the permeabiity) and 0.2 (strong decrease). For this geometry, the volume of “cake” was equal to 5% of the total debris bed volume.

Calculations indicated that in the presence of “cake” coolability of the debris bed is deteriorated, especially in the case of strongly reduced permeability of porous medium. In Fig. 6, the distributions of void fraction in the debris bed are shown for the specific heat release rate $W = 65$ W/kg and different permeability reduction coefficients. The shape of the debris bed and lower boundary of the “cake” are shown by the dashed lines. One can see that, expectedly, dryout and vapor accumulation occur primarily in the “cake”.

In Fig. 7, the maximum void fractions in the “cake” are plotted against the specific heat release rate for particle diameters of 2 and 3 mm. Comparison of this data with those in Fig. 5 shows that for 2 mm particles dryout occurs at the specific powers of 70-90 W/kg; about the same critical value of specific power is obtained for 3 mm particles, while for moderately reduced permeability dryout occurs at about 150W/kg. Thus, except for the latter case, the “cake” can be considered as non-coolable at the heat release rates characteristic of reactor applications.

Finally, consider the effects of internal porosity of debris bed particles. In the calculations it was assumed that the overall (internal and accessible to fluids) volume fraction increased from 0.4 in the baseline case to 0.55. This gave the internal porosity of particles $\varepsilon_{\text{int}} = 0.25$, the total volume of debris bed increased by 33% to $V = 55.5$ m^3, whereas the effective density of the debris particle material ρ_c decreased from $8{\cdot}10^3$ to $6{\cdot}10^3$ kg/m^3. The linear parameters of the Gaussian shape of the debris bed were increased proportionately to accommodate the larger volume: the maximum height $H = 2.85$ m and characteristic width $R_0 = 2.5$ m.

The results obtained turned out to be dependent on the system pressure. In the preliminary calculations carried out at a low system pressure of 1 bar, coolability of the debris bed with encapsulated particle porosity was better than in the case of solid particles. In this case the reduction in the volumetric density of heat release rate due to the decrease in the effective density of corium material outweighs the increase in the linear sizes of the debris bed, i.e., plays on the safety side. However, calculations carried out at the system pressure of 3 bar give an opposite result: e.g., for the specific power of 130 W/kg, a debris bed with 2 mm solid particles turned out to be coolable, while an equivalent debris bed with porous particles was not coolable. Further research is necessary to find out the parameter ranges in which the coolability of debris bed is improved or deteriorated due to the encapsulated particle porosity.

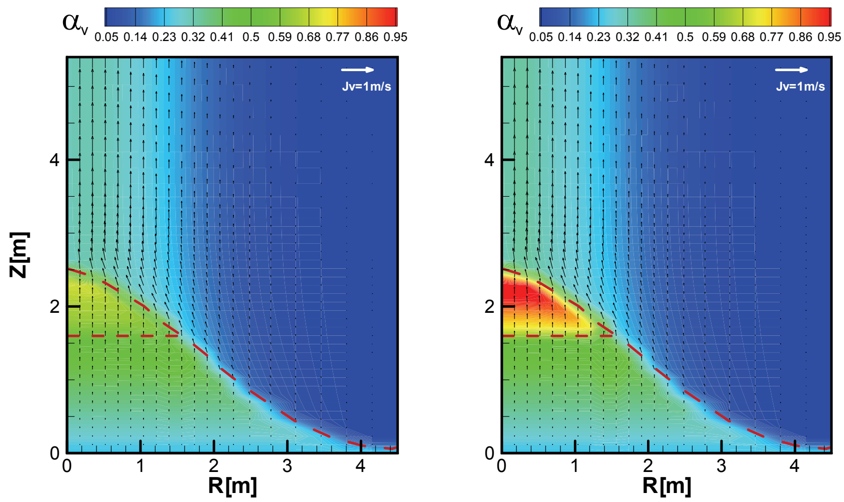

Fig. 6. Void fraction distributions in a Gaussian-shaped debris bed with a “cake” (particle diameter $D = 2$ mm) at heat release rate $W = 65$ W/kg: the permeability and passability in the “cake” are reduced by a factor of 0.5 (left) and 0.2 (right)

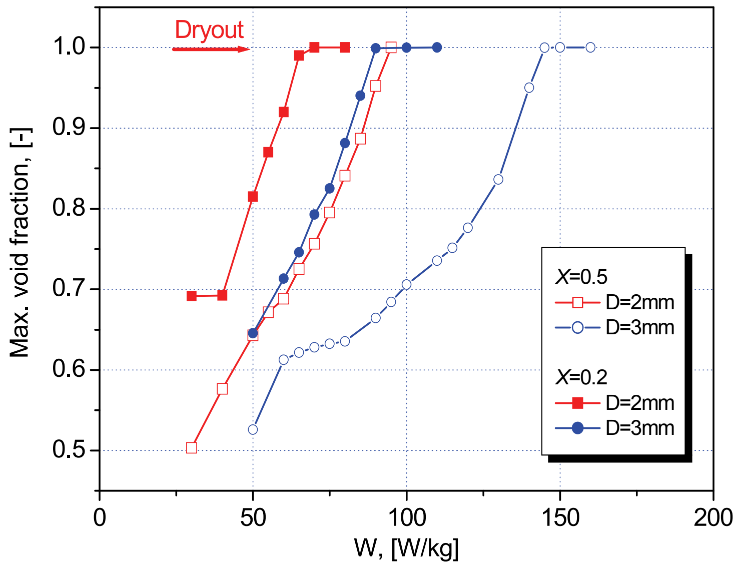

Fig. 7. Maximum void fraction in the debris bed with a “cake” vs specific heat release rate for different particle sizes and permeability reduction factors

5. Conclusions and safety implications of results

Numerical simulations performed by DECOSIM code have been performed for two scenarios of severe accident. In the case of gradual melt release, it is shown that “self-organization” plays an important role in particle sedimentation and debris bed formation. Generally, interaction of particles with the natural circulation flow results in lateral spreading of particles over the pool bottom, which prevents formation of a tall compact debris bed and improves coolability of debris. It was shown that smaller particles are more affected by the flow, so that their lateral spreading is more pronounced. A consequence of this is that debris bed expected to be non-homogeneous, both in vertical and horizontal directions in case of polydisperse particles. Implications of this effect for debris bed coolability have to be studied.

In the scenario of massive melt release, it is expected that the debris bed will have some heap-like shape, though, ad hoc specification of the shape is used at the moment in all coolability studies. Here, Gaussian-shaped debris bed is considered, for which the effects of particle diameter, internal porosity and “cake” are studied. It is shown that generally, dryout in the heap-like debris bed occurs more readily than in an equivalent flat layer, even despite the fact that side ingress of coolant is possible. Formation of a low-permeability “cake” on the top of debris bed has a pronounced negative effect. The effect of encapsulated particle porosity on the coolability of debris bed was found to be system pressure-dependent and requires more thorough analysis.

Acknowledgement

This work is supported by the Swedish Nuclear Radiation Protection Authority (SSM), Swedish Power Companies, European Commission, Nordic Nuclear Safety Program (NKS), Swiss Federal Nuclear Safety Inspectorate (ENSI) under the MSWI-APRI and SARNET program.

Substantiation of strategy of water supply recovery to steam generators at in-vessel severe accident phase for VVER-1000 Balakovo NPP

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Summary

The Severe Accident Management Guidance (SAMG) for Balakovo NPP, Unit 4 with VVER-1000/V-320 reactor has been developed in 2008. Brief description of SAMG development and current activities is given.

Results of PSA level 1 for the units of Balakovo NPP show that initial events and failures leading to dryout of steam generators in secondary circuit make a large contribution into frequency of core meltdown. In such scenarios the decay heat can not be removed from primary circuit and consequently the severe accident begins at high primary pressure.

In these scenarios it is necessary to consider possibility of SG tubing failure under impact of hot gases flowing from the core and also possibility of core melt release from reactor vessel at high primary pressure. These phenomena and their consequences with respect to severe accident progression are discussed.

Recovery of heat removal from primary circuit is possible if any way of water supply into steam generators is successful. At Balakovo NPP the strategy of water supply into steam generators from fire engines has been implemented. This strategy has been included into the SAMG of Balakovo NPP, Unit 4. Description of the strategy is presented.

To evaluate efficiency of this strategy depending on water flow rate and time of water supply beginning the computer analyses have been performed. The results of analyses are presented. The results show influence of water supply from fire engines on severe accident progression at in-vessel SA phase. The analyses have been fulfilled with MELCOR 1.8.5 code.

1. Introduction

The works on development of Severe Accident Management Guidance (SAMG) for the operating Russian VVER-1000 plants were started in 2001. The SAMG development working program was presented in paper[1].

The first project oriented on SAMG development for VVER-1000 plants was performed in the frame of cooperation between the Russian Minatom International Nuclear Safety Centre (RMINSC) and the US International Nuclear Safety Centre (US INSC) and Argonne National Laboratory (ANL). The results of this project included selection of severe accident management (SAM) strategies applicable for VVER-1000 plants, selection of instrumentation for plant diagnostics during severe accident and development of a set of simple SAM guidelines called later "revision 0 of generic VVER-1000 SAMG". The structure and components of the Westinghouse SAMG[2] were taken as a basis for the VVER-1000 SAMG development.

Further work on SAMG development for the operating Russian VVER-1000 plants was held under sponsorship of the Russian utility organization "Concern Energoatom" (former "Rosenergoatom"). In 2006 the revision 1 of generic SAMG for VVER-1000/V-320 NPPs was developed[3]. Based on revision 1 and comments form Balakovo and Kalinin plants the revision 2 of generic SAMG was prepared.

The generic SAMG was used as a basis for development of SAMG for the Balakovo NPP, Unit 4. The validation of the Unit 4 SAMG was held in beginning of 2009. The results of validation were used for improvement of SAMG and development of revision 1.

All the SAMG versions mentioned were developed by the specialists of the Institute for Nuclear Reactors of RRC "Kurchatov Institute" and RMINSC experts.

2. Description of Balakovo NPP, Unit 4 SAMG

2.1. Main components of the SAMG

The SAMG of the Balakovo NPP, Unit 4 consists of the following components:

- the set of SAMG guidelines and computational aids,
- the set of documents "Rules of accident management",
- the document "Executive volume".

For each SAMG guideline or computational aid the respective document "Rules of accident management" was developed. For SAMG guidelines this document includes the description of basic SAM strategies realized in the guideline and description of the guideline steps with explanation of step purposes. For computational aids (CA) the document includes description of the CA purpose,

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[1] V.Ignatov et al. Development of Severe Accident Management Guidance for VVER-1000 Plants. Workshop on the Implementation of Severe Accident Management Measures. Paul Scherrer Institut, Villigen-PSI, Switzerland, 10-13 September 2001.

[2] R.Prior. Westinghouse Owners Group. Severe Accident Management Guidance. SAM '99 Meeting, Obninsk, Russia, October 1999.

[3] L.Kabanov et al. Development of the First Version of Generic Severe Accident Management Guidelines for Operating VVER-1000/V-320 NPPs. Regional IAEA Workshop on Severe Accident Analysis and Accident Management Programme for NPPs. Kiev, Ukraine, 18-22 June 2007.

description of assumptions made and data used for the CA development and presentation of the CA itself.

The document "Executive volume" includes description of principles of severe accident management, the SAMG user manual, description of principles of instrumentation use for the unit diagnostics in the course of severe accident and presentation of principles of NPP personnel and specialists interaction in the process of severe accident management.

2.2. Composition of guidelines and CAs of the Balakovo NPP, Unit 4 SAMG

The composition of the SAMG of Balakovo NPP, Unit 4 corresponds in general to Westinghouse approach and is the following:

– Diagnostic Flow Chart (DFC), seven DFC guidelines,
– Severe Challenge Status Tree (SCST), four SCST guidelines,
– two guidelines for MCR,
– two severe accident exit guidelines,
– three auxiliary computational aids.

The following DFC guidelines are included into the SAMG:

– SAG-1, "Inject into the Steam Generators"
– SAG-2, "Depressurize the RCS"
– SAG-3, "Inject into the RCS"
– SAG-4, «Inject into the Containment"
– SAG-5, "Reduce Fission Product Releases"
– SAG-6, "Control Containment Conditions"
– SAG-7, "Reduce Containment Hydrogen"

The following SCST guidelines are included into the SAMG:

– SCG-1, "Mitigate Fission Product Releases"
– SCG-2, "Depressurize Containment"
– SCG-3, "Control Hydrogen Flammability"
– SCG-4, "Control Containment Vacuum"

Two guidelines for MCR are:

– SACRG-1, "Severe Accident Control Room Initial Response",
– SACRG-2, "Severe Accident Control Room Guideline for Transients After the TSC is Functional".

Two severe accident exit guidelines are:

– SAEG-1, "TSC Long Term Monitoring Activities",
– SAEG-2, "SAMG Termination".

Three auxiliary computational aids are:

– CA-1, "RCS Injection to Recover Core",
– CA-2, "Injection Rate for Long Term Decay Heat Removal",
– CA-3, "Hydrogen Flammability in Containment".

Specific features of some guidelines of the Balakovo NPP, Unit 4 SAMG should be noted. The guidelines associated with hydrogen management are based on the auxiliary computational aid CA-3 because at the Balakovo plant there are no hydrogen concentration measurement devices that could be available during accidents.

Unlike the Temelin VVER-1000 plant (Czech Republic) SAMG[4] the SAG-4 guideline of the Balakovo plant SAMG has been designed for application after core melt release from the reactor vessel. In operating Russian VVER-1000 plants there is no possibility to inject water directly into the reactor pit. Supply of non-borated water is possible into containment from some systems. In this case water supplied into the containment will be mixed with borated water of the containment sump. To avoid non-borated water delivery into primary circuit at the in-vessel severe accident phase (for instance, in case of power supply recovery) it was decided to start actions in the frame of the SAG-4 guideline according to criteria indicating that the core melt has been released from the reactor vessel and the hermetic door of the reactor pit has been knocked out by pressure difference.

Among different ways of feeding the secondary side of steam generators the actions on ensuring passive water delivery from the feedwater trains and on water supply from fire engines are implemented in the SAG-1 guideline. These actions and their analytical evaluation are presented below.

2.3. Verification and validation of the Balakovo NPP, Unit 4 SAMG

The SAMG verification was held mainly at the stage of the generic SAMG evaluation by specialists of Balakovo and Kalinin NPPs.

Validation of the Balakovo NPP, Unit 4 SAMG was held in January of 2009. The validation was preceded by training of the Balakovo NPP specialists in the field of severe accident management.

The training of the Balakovo NPP specialists included the following topics:

- Severe accidents, SA progression and phenomenology with respect to VVER-1000/V-320 reactors,
- Principles of severe accident management, international experience in SAMG development,
- General SAMG description;
- Elements of the Balakovo NPP, Unit 4 SAMG,
- General information on analytical support of SAMG development including information on the SA computer codes.

It was decided to concentrate in the course of SAMG validation on those SAMG elements that allow for mitigation of SA consequences during the in-vessel phase of severe accidents. The following scenarios were played in three validation exercises:

- Total loss of feedwater,
- SBLOCA Dn40 from cold leg with HPIS and LPIS failure,
- Station blackout.

Prior to validation exercises the computer analyses of these three scenarios were performed using the MELCOR 1.8.5 code. In station blackout scenario the assumption on possibility to recover some NPP systems after certain time was used.

2.4. Further activities associated with SAMG development for VVER-1000 plants

Now the set of documents of Balakovo NPP, Unit 4 SAMG is evaluated in “Atomenergoproekt” organization (The General Architect of VVER plants) and EDO “Gidropress” organization (the Main Designer of VVER reactor facilities). The comments of these organizations will be used for further improvement of the Balakovo NPP, Unit 4 SAMG.

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[4] J.Kubicek. Severe Accident Management at Temelin NPP. Regional IAEA Workshop on Severe Accident Analysis and Accident Management Programme for NPPs. Kiev, Ukraine, 18-22 June 2007.

In 2009 the works on SAMG development for Units 1 and 2 of the Kalinin NPP with VVER-1000/V-338 reactors have been started. The generic SAMG of VVER-1000/V-320 plants has been taken as a basis for SAMG development for Kalinin NPP, Unit 1 and 2.

3. The strategy "Inject into the steam generators"

The strategy "Inject into the steam generators" realized in the SAG-1 guideline is applied with the following purposes:

- ensure heat removal from the primary circuit and thus ensure primary circuit integrity,
- protect steam generator tubes from damage caused by the creep,
- scrub fission products which are transported into steam generators through leakages in SG tubes.

If this strategy is not applied the consequences are the following. In accidents with the core dryout and heatup the overheated steam and hot gases are transported from the core to hot legs and SG tubing. This can lead to induced hot leg and SG tubing failures due to creep. If damage of hot legs occurs the containment pressure rises quickly and the containment shell integrity is challenged. If SG tubing failure occurs due to creep then fission products together with gas-steam mixture come into steam lines and further into environment.

In VVER-1000/V-320 plants the water can supplied into the steam generators with three groups of feedwater pumps: main feedwater, auxiliary feedwater and emergency feedwater. If all these pumps are not available the following non-standard means can be used:

- passive feeding steam generators by water from feedwater trains and deaerators,
- feeding steam generators from mobile pumps (fire engines).

For passive feeding steam generators the depressurization of steam generators is needed after their dryout in order to have SG pressure below the pressure in the feedwater trains and the deaerators. This allows for passive water delivery from the feedwater trains and the deaerators. SG depressurization can be done by forced opening of BRU-As (steam dump to atmosphere).

Besides that in the Balakovo NPP units the water supply into steam generators from fire engines has been implemented. The equipment modernization needed was performed. The main element of the modernization was installation of special pipeline Dn100 into the feedwater pipeline system. In the Balakovo NPP the following pumps of ire engines are available for feeding steam generators: pumps with capacity of 40 kg/s and 110 kg/s at pressure below 1,18 MPa and also a pump with capacity of 30 kg/s at pressure below 5,88 MPa.

Basic uncertainty in case of the strategy implementation is associated with cooling of SG tubes when water is supplied into steam generators. Depending on the primary circuit state (respectively severe accident phase) the SG tube cooling can prevent the tube creep (moderate primary coolant heatup at oxidation phase of SA) or facilitate the SG tube damage in case of their strong heatup with hot gases leaving the core at the phase of severe core degradation. So the primary circuit depressurization is desirable for success of the SAM strategy discussed.

4. Computer investigation of recovery of water supply into steam generators from fire engines

4.1. Accident scenario and initial data

The total loss of feedwater accident is considered. It is assumed that safety systems are available and able to supply borated water into the primary circuit when primary pressure becomes low enough due to accident management (AM) measures.

The following AM measures are simulated: opening of BRU-As (steam dump to atmosphere) in 6500 seconds after initial event (Figure 2) together with water supply from fire engines with total flow rate of 40 kg/s (i.e. flow rate of 10 kg/s into each steam generator) in 7000 seconds after initial event (Figure 6). It is assumed that fire engine pumps supply water from the source of large enough volume.

4.2. Results of calculation

If the NPP personnel does not intervene the accident considered the first stage of the accident is dryout of steam generators due to absence of feedwater supply. When water inventory in steam generators (secondary circuit) becomes low enough the parameters of primary circuit begin to grow because heat removal to secondary circuit is lost. Primary pressure rises up to the pressurizer safety valve opening setpoint. Starting from this moment the primary coolant is discharged through the pressurizer safety valves. Loss of primary coolant leads to the core dryout and heatup. The accident comes to severe phase.

Time moment of beginning of AM measures in computer analysis (BRU-A opening) was taken at the phase of the core heatup (Figures 3, 4). With this selection of AM measures beginning they can not prevent transition of the accident into severe phase and determine the NSSS behaviour after beginning of the core meltdown.

Water supply into steam generators was simulated after beginning of the core meltdown when particulate debris are formed. The water supply recovers heat removal from primary circuit to secondary circuit that can be observed by decrease of primary pressure (Figure 1) and decrease of primary coolant temperature in the core inlet and outlet (Figure 5).

Primary pressure decrease leads to borated water supply by the HPIS pumps. After certain time period the primary pressure stabilizes.

Thus, water supply into steam generators from mobile pumps (pumps of fire engines) leads to cooling of the core melt inside the reactor vessel and prevents the transition of the accident into the ex-vessel stage.

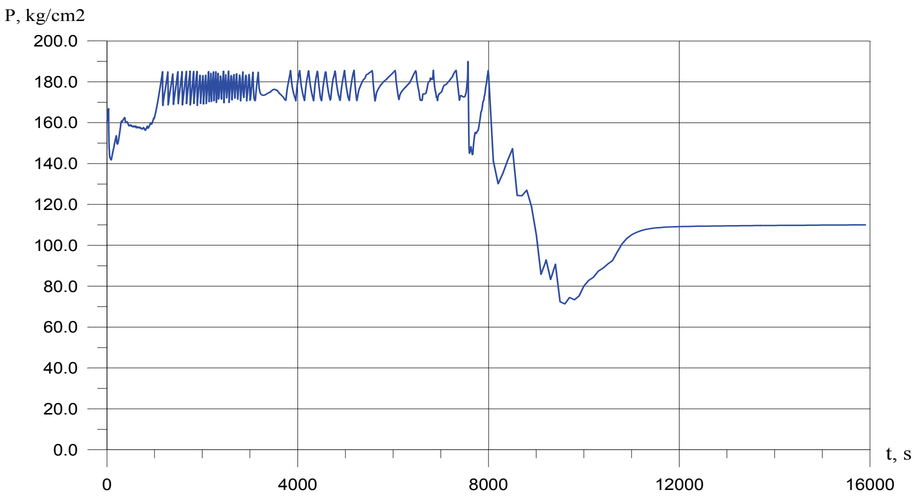

Figure 1. Primary pressure

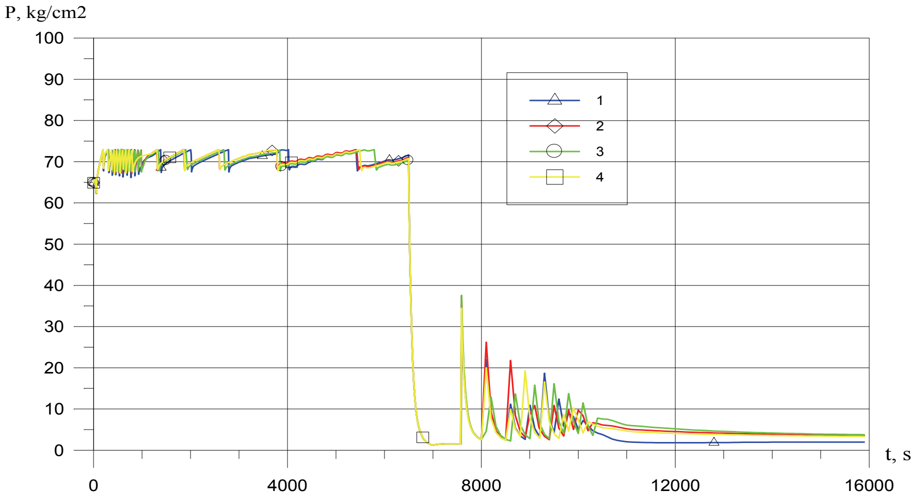

Figure 2. Pressure in steam generators

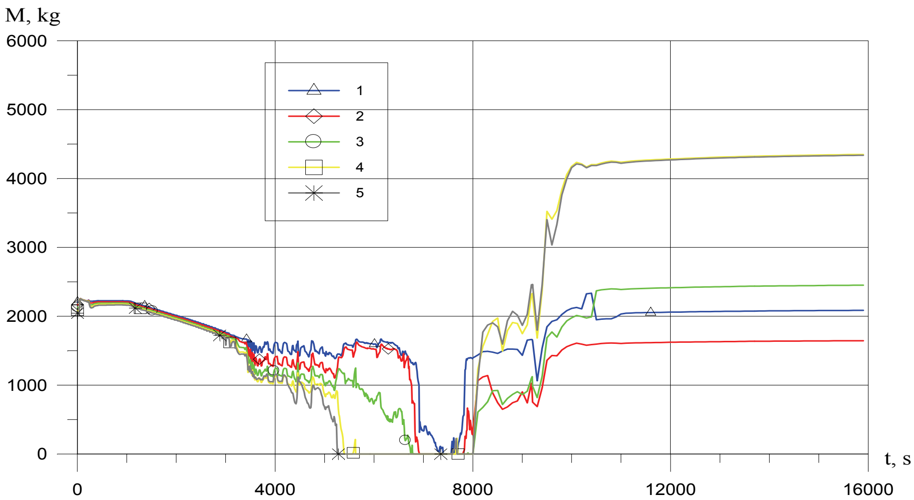

Figure 3. Coolant mass in the core axial nodes 1 to 5

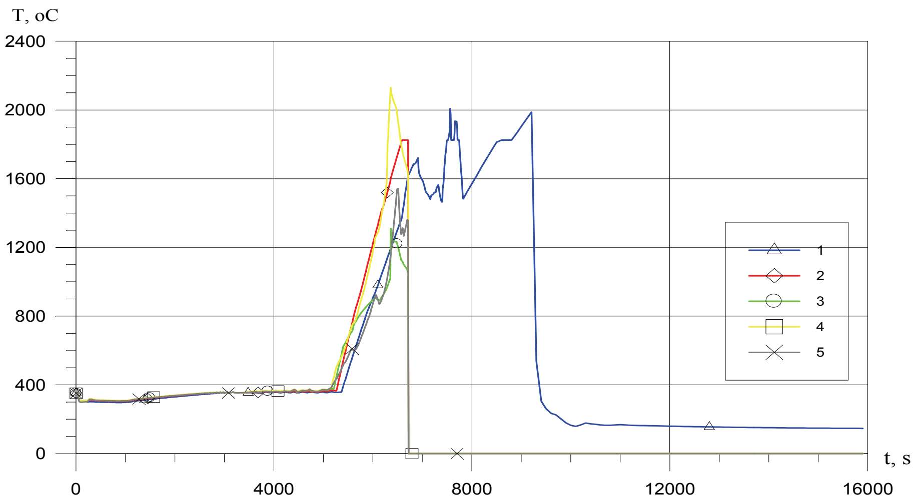

Figure 4. Fuel rod cladding temperatures in the core axial nodes 1 to 5

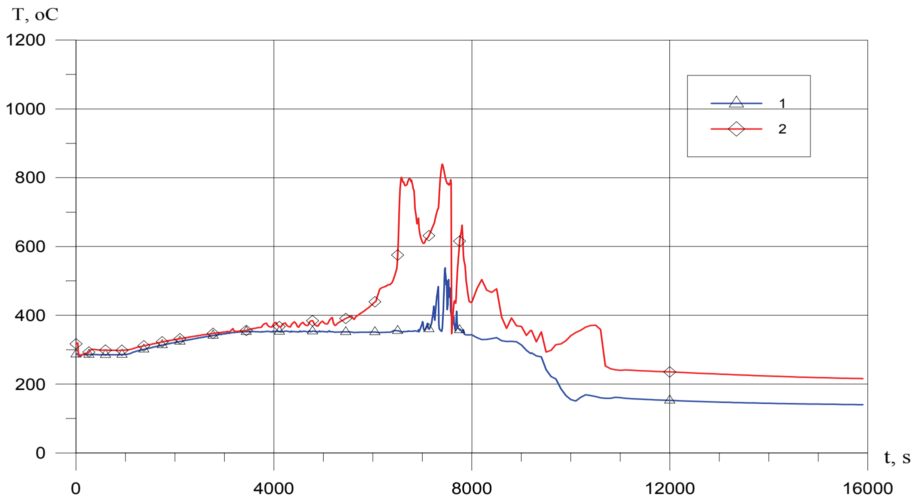

Figure 5. Temperature of coolant at core inlet and outlet

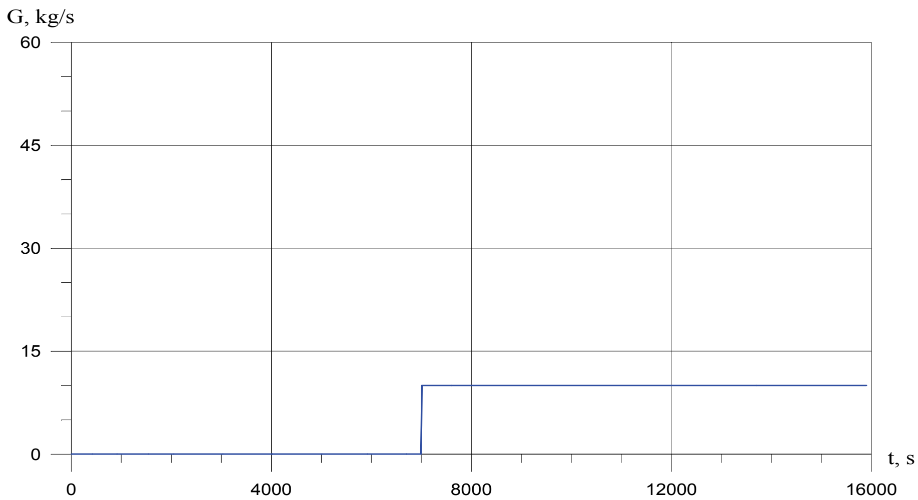

Figure 6. Flow rate from fire engines into each steam generator

Ambient Pressure-dependent Radionuclide Release from Fuel Observed in VEGA Tests under Severe Accident Condition and Influence on Source Term Evaluation

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1. Introduction

The radionuclide release from fuel during severe accidents is a primary issue for the source term evaluation. Although numbers of experiments[1-5] have been conducted in this domain of research in the world, information is still insufficient for the precise evaluation. For example, radionuclide release could mostly occur at high temperature under elevated pressure but very few studies have investigated the pressure effect so far due to difficulties in experimental operation. Thus, in the previous source term evaluation[6, 7], the radionuclide release correlations obtained from the tests at atmospheric pressure have been also applied to the case at elevated pressure. In order to obtain the release data under such high temperature and elevated pressure and to clarify the pressure effect and the release mechanisms, VEGA (Verification Experiments of radionuclide Gas/Aerosol release) program[8] was conducted at Japan Atomic Energy Agency (JAEA) from 1999 to 2005. The program especially focused on the behavior of low-volatile radionuclide and actinides released from high burn-up UO_2 or MOX fuel under high temperature above the fuel melting point and elevated pressure up to 1.0MPa.

This paper describes the dependency of radionuclide release on ambient pressure observed in VEGA tests, proposed release mechanisms and release model with pressure effect, limitations of VEGA tests and future issues, possible influences on source term evaluation and accident management measures.

2. VEGA apparatus and measurements

The VEGA apparatus (see **Fig. 1**) consists mainly of gas supply system, high frequency induction furnace, three sets of thermal gradient tubes (TGT), aerosol filters, condenser, noble gas trap and γ-ray measurement system. The fuel specimen of a few pellets set in the tungsten or ThO_2 crucible is inductively heated up to 3150K (max) in inert or oxidizing atmosphere. The bottom of crucible is measured by pyrometers located below the furnace which are adjusted so that the measured temperature might be equal to the fuel one within the error of ±50K. The facility can be pressurized up to 1.0MPa between the gas supply system and the valves just after the aerosol filters.

The temperatures of downstream TGT, filters, condenser and noble gas trap are always kept at the constant values during a test irrespective of the furnace temperature. The temperature gradually decreases as the distance from furnace becomes far. The low or semi-volatile radionuclide is trapped at the piping of furnace outlet at 1023K. The volatile radionuclide is collected at TGT of which temperature changes from 1023 to 473K and at filters (473K). A condenser (273 K) collects humidity and gaseous iodine species such as I_2. A noble gas trap collects Kr-85 by physical adsorption onto a cold charcoal at 210K. The behavior of radionuclide release and transport in the apparatus during a test is on-line measured by γ-ray from fuel specimen, filters, condenser and noble gas trap using the hyper-pure germanium semiconductor detectors.

After the heat-up test, the off-line γ-ray measurements are performed for fuel specimen, the piping between furnace and aerosol filters to evaluate the remained radionuclide in fuel or deposition distribution in the whole apparatus. The microphotographs of specimen before and after the heat up test are taken to know the change of fuel structure. A microanalysis with ICP-AES (Inductively Coupled Plasma - Atomic Emission Spectrometry) is also performed for the nitric acid solution leached from the VEGA apparatuses to evaluate the released and deposited masses for short-life or no γ-ray emitting radionuclide.

3. Reference tests with PWR, BWR and ATR/MOX fuels

At first, the results of reference heat-up tests with PWR-UO_2, BWR-UO_2 and ATR/MOX fuels performed under the same conditions are described to help understanding of the basic behavior of cesium (Cs) release from different fuels[9)]. The specifications of three different fuels used in the tests are shown in **Table 1**. The PWR and BWR fuel specimens were irradiated at Japanese commercial reactors. The MOX fuel irradiated at the Advanced Thermal Reactor (ATR) Fugen was fabricated at JAEA with the one-step process, similar to the short binderless route (SBR) [10)]. The area fraction of Pu-rich spots was 0.2% and the Pu concentration in MOX fuel was almost homogeneous[9)]. The pellet temperatures at center and peripheral regions were calculated from the average values of linear heat rates during reactor operation with the FRAPCON-2 code[11)].

The conditions for reference tests are shown in **Table 2**. Fuel specimens without cladding were set up in a tungsten crucible and heated up to 3130K at a rate of 1K/s in helium (He) atmosphere at 0.1MPa. Histories of Cs release during the three tests are shown in **Fig. 2** together with the fuel temperatures. In all the tests, fractional releases of Cs finally reached about 100% while a difference was found in release behavior below 1700K among them. The releases from BWR and MOX fuels started at about 1000K while that from PWR fuel did not start until the temperature increased above 1700K.

The reason for release enhancement below 1700K in BWR and MOX fuels is considered that the averaged linear heating rates of 26kW/m in BWR and 28kW/m in MOX during reactor operation were higher than 18kW/m in PWR. This would result in temperature increase up to 1500K - 1700K at pellet center that is higher by about 500K than that of PWR and could enhance the movement of Cs from center pellet to peripheral region[12)]. The distributions of Cs-137 in the diameter direction of PWR and MOX pellets before heat-up tests were separately measured by γ-ray detector (see **Fig. 3**). It can be seen that during reactor operation, most of Cs was moved to peripheral region in MOX while almost no movement in PWR. For BWR, a similar distribution as MOX but slightly less movement was confirmed in other test[13)]. It is considered that Cs at pellet center region would become vapor due to high temperature during reactor operation (boiling and melting points of CsI = 1553K and 899K, respectively) and have been transported to the low-temperature peripheral region and deposited as i.e., cesium iodide (CsI) at the grain boundary or open pores. During heat-up tests, this could be vaporized below 1700K before the diffusion in grains becomes significant at higher temperature.

4. Observed decrease in cesium release under elevated pressure

To investigate the effect of ambient pressure of radionuclide release, three different fuel specimens were heated up two times at 0.1and 1.0MPa, respectively. Their test conditions and fractional releases at the end of each test are shown in **Table 3**. In the PWR and MOX fuel tests, fuel specimens without cladding were heated up in He atmosphere under the same temperature histories except for ambient pressure. In the BWR fuel tests, fuel specimens were re-irradiated just before the heat-up test at Japan Research Reactor No.3 (JRR-3) of JAEA with the thermal neutron flux of about 1.0 x 10^{12} $cm^{-2}s^{-1}$ for 26 days to accumulate short-life radionuclide such as I-131, Te-132, Ba-140, Ru-103 and so on. Moreover, the specimens were heated up in steam atmosphere under the condition with Zr cladding.

The fuel temperatures and Cs fractional releases of the tests with PWR, MOX and BWR fuels are shown in **Figs. 4 - 6**, respectively. The tests with PWR fuel experimentally first showed that Cs fractional release at 1.0MPa decreased by about 30% compared with that at 0.1MPa[14]. On the other hand, similar pressure effect was not observed clearly in MOX and BWR fuel tests. In MOX fuel tests, Cs release below 1700K at 1.0MPa slightly became smaller than that at 0.1MPa while no large difference in Cs release above 1700K between 0.1 and 1.0KPa[12]. For BWR fuel tests, to the contrary, Cs fractional release at 1.0MPa slightly increased compared with that at 0.1MPa. The reason for difference in release behavior under elevated pressure among three fuels is discussed in **chapter 5**.

5. Discussions on test results and mechanisms for pressure effect

5.1 Mechanism for pressure effect

It is considered that the important process of radionuclide release from fuel would be described as the lattice diffusion of elemental radionuclide in UO_2 grain followed by the diffusion of gaseous radionuclide in open pores as shown in **Fig. 7**. The gaseous diffusion in pores depends on temperature and ambient pressure ($D_g \propto T^{1.5}/P$) while the diffusion in grains depends only on temperature. Since the diffusion coefficient in grains is much smaller by more than 10 orders of magnitude compared with that in pores, the diffusion in grain has been considered as the rate-determining step in previous studies[15), 16)]. However, there is no large difference in diffusion time between grains and open pores particularly in elevated pressure if the difference in diffusion length between grains and open pores, interconnected (open pores) porosity and total porosity of fuel are taken into account[14].

Expected distribution of radionuclide concentration in UO_2 grain and open pores in pellet at elevated pressure is shown in **Fig. 8**. The elevated pressure causes increase in gas density in open pores and decrease in gaseous diffusion velocity. These result in increase in radionuclide concentration in small-volume pores and at grain surface although the diffusion velocity in grain is much slower than that in open pores. The concentration increase at grain surface may suppress radionuclide release from grain because it decreases the concentration gradient in grain that is a driving force of diffusion. This proposed mechanism implies that the pressure effect would not appear before radionuclide release can be mainly described as the diffusion in UO_2 grain followed by the diffusion in open pores.

5.2 Observed difference in pressure effect among fuel type

(1) Effect of fuel temperature during normal operation

During normal operation, the temperature at center region of PWR pellet was lower by about 500K than that of BWR or MOX pellet as described in **chapter 3**. As a result, in PWR fuel, most of Cs at pellet center would have remained at original place while in BWR and MOX fuels, more than half of Cs inventory would have been moved from center to peripheral region and deposited at peripheral region. Therefore, in PWR fuel, the release was governed primarily by the diffusion in grains and the pressure effect easily appeared. On the other hand, in BWR and MOX fuels, the release mainly occurred by vaporization from the peripheral region and the pressure effect did not appear clearly. The reason for slight decrease in Cs release from MOX below 1700K at 1.0MPa is considered that the boiling point of CsI increased at elevated pressure[12)].

(2) Effect of fuel oxidation and eutectic reaction with cladding

In case of BWR fuel tests with Zr cladding under steam atmosphere, it is considered that fuel was oxidized below Zr melting point of 2130K and fuel liquefaction due to eutectic reaction with liquefied Zr occurred above 2130K[17)]. This could have enhanced the Cs release compared with PWR and MOX fuel tests. When the fuel is oxidized from UO_2 to UO_{2+x}, the concentration of vacancy or defect in the fuel grain matrix would increase and the diffusion of radionuclide through vacancy or defect could occur easier than the case without oxidation[18)]. Meanwhile, if the liquefied Zr touches fuel, the oxygen inside fuel begins to diffuse in liquefied Zr. As a result, fuel could be reduced and the melting point of fuel could decrease from 3123K (for UO_2) to 1405K (for Uranium) depending on the degree of reduction. Once the fuel liquefaction occurs, volatile radionuclide release could be enhanced due to disappearance of the grain matrix in which radionuclide is confined[19)].

One possible reason for larger Cs release from BWR at 1.0MPa compared with that at 0.1MPa is considered that the steam concentration at 1.0MPa was higher by a factor of 4 than that at 0.1MPa taking into account the mass flow rates of He and steam (see **Table 3**). As a result, the fuel oxidation at 1.0MPa could have enhanced compared with the test at 0.1MPa.

(3) Limitations of VEGA tests

The fuel oxidation depends strongly on the surface area of UO_2 exposed to steam or the ratio of Surface area to Volume (S/V)[1)]. In the VEGA tests under oxidizing condition in which both cross sections of top and bottom of two fuel pellets are exposed to steam, geometric S/V equal to about 100m^{-1} could become larger, that is, the effect of fuel oxidation could be overestimated compared with the reality in which only the limited area near the cladding rupture point would be oxidized. A similar attention has to be also paid for the eutectic reaction with cladding. The cross section of crucible after the heat up in other VEGA test using PWR-UO_2 with cladding under steam atmosphere is shown **Fig. 9**[17)]. The photograph shows that the intact UO_2 was not seen at 8mm high from the crucible bottom and above. Since the length of test fuel was originally 20mm, 60% to 75% of the original pellet was estimated to be liquefied during the test. This result also indicates that the eutectic reaction could proceed too much in the VEGA geometry compared with the reality without crucible that maintains liquefied materials.

(4) Pressure effect for low-volatile radionuclide release

The VEGA tests with re-irradiated BWR fuel further showed that release of low-volatile radionuclide with short-life was not enhanced by fuel liquefaction (see **Table 3**) but decreased at elevated pressure. This suggests that low volatile radionuclide would not move inside fuel pellet during reactor operation, the release would be governed by diffusions in grains and pores, and the form of low volatile radionuclide at time of release from grains could be vapor that subsequently diffuses in open pores.

5.3 Summary of pressure effect

Perspectives obtained form the pressure effect tests of VEGA program are summarized in **Table 4**. The pressure effect could appear when the radionuclide release is governed by diffusion in grains followed by diffusion in pores. The effect could appear easier in PWR fuel than in BWR or MOX fuel although it depends on the temperature history of fuel during reactor operation. The release of low-volatile radionuclide depends on neither the irradiation history nor fuel liquefaction while it decreases at elevated pressure because the form of low volatile radionuclide at time of release from grains could be vapor. It is expected that the relationship between the pressure effect on radionuclide release and irradiation history during reactor operation, fuel oxidation, the eutectic reaction with cladding be further examined in other future tests that simulate better the real conditions during severe accidents.

6. Release model with pressure effect and analyses of VEGA & SFD 1-4 tests

The observed pressure effect can be explained by a 2-stage diffusion model that considers the lattice diffusion in grains followed by the gaseous diffusion in open pores as described in **chapter 5.1** and previous studies[20]. Based on this model, a simplified model with the release rate coefficient multiplied by $1/\sqrt{P}$ (P>1 atm), that is, $1/\sqrt{P}$ CORSOR-M model was derived for the source term analysis[20]. The multiplier $1/\sqrt{P}$ comes from the pressure dependency of gaseous diffusion in open pores.

6.1 Analysis of VEGA tests with proposed release model

The Cs releases from PWR fuel at 0.1 and 1.0MPa calculated by proposed $1/\sqrt{P}$ CORSOR-M model are shown in **Fig. 10**. It is noted that the pre-exponential factor of the CORSOR-M model was modified in this study so that the calculation at 0.1MPa might agree with the measurement at 0.1MPa. The calculation with $1/\sqrt{P}$ CORSOR-M gave a reasonable agreement with the measurement at 1.0MPa although the calculation slightly underestimates the release evolution.

6.2 Analysis of SFD 1-4 test with proposed release model

In order to verify the effectiveness of the proposed model, the model was applied to other experiment performed under elevated pressure. The Severe Fuel Damage Test 1-4 (SFD 1-4) was conducted at the Power Burst Facility in USA in 1985 to obtain data mainly on the release, transport, and deposition of radionuclide[21]. A test bundle, comprised of 26 previously irradiated (36GWd/t) PWR type fuel rods, 2 fresh instrumented fuel rods, and 4 Ag-In-Cd control rods, was contained in a pressurized in-pile tube. During the test that simulated S_2D sequence, the test bundle was heated up at 6.95MPa and finally 18% of fuel was liquefied while the measured Cs fractional release at the end of the test was 51%. The best estimate analysis with CORSOR model predicted the Cs fractional release of 83%[20].

In this study, Cs release during the SFD1-4 test was analyzed with conventional CORSOR-M, ORNL-Booth and $1/\sqrt{P}$ CORSOR-M models. The spatial temperature distribution for SFD 1-4 bundle[21] used in the analyses are shown in **Fig. 11**. Calculated Cs release evolution and the fractional release at the end of test are shown in **Fig. 12** and **Table 5,** respectively. It can be seen that $1/\sqrt{P}$ CORSOR-M model gave more reasonable prediction compared with the conventional ones.

7. Influences on source terms and accident management measures

The decrease in radionuclide release under elevated pressure may affect, i.e., PWR source term evaluation, accident management (AM) measures such as intentional primary system depressurization by operators. The results of source term analyses[20] for TQUX sequence of BWR5 using THALES-2[6] with and without the pressure effect release model are shown in **Table 6**. Although the analyses were performed for BWR, the perspectives obtained from the analyses can be also applied to PWR. The analyses showed that the decrease in radionuclide release under elevated pressure could result in hastening the accident progression before RPV melt-through (see **Fig. 13** and **Table 6**) and in increase or decrease in the source terms depending on the accident sequence in particular on the timing of containment (CV) failure. In case of early CV failure, source terms could increase by about one order of magnitude due to release during molten core concrete interaction (see **Fig. 14**) because about 40% of CsI inventory still remains in the molten debris at Reactor Pressure Vessel (RPV) and CV failures.

It is expected that the intentional depressurization, to the contrary, would enhance radionuclide release to the primary system but could delay the accident progression before and after RPV melt-through due to reduction of decay heat from fuel or molten debris. Moreover, the source terms at time of early CV failure would be mitigated due to decrease in radionuclide release from molten debris to containment atmosphere. The advantages and disadvantages of intentional depressurization with considering the pressure effect needs to be further evaluated in detail.

8. Conclusions

In VEGA program, totally 10 tests were performed under the highest pressure and/or temperature conditions from 1999 to 2004. Tests with PWR fuel at 1.0MPa showed experimentally first that Cs release rate was suppressed by about 30% compared with that at 0.1MPa. Observed pressure effect could be explained by 2-stage diffusion in UO_2 grains & pores, and predicted by a simplified $1/\sqrt{P}$ CORSOR-M model. In BWR and MOX fuel tests, however, this effect was not observed clearly due to domination of vaporization from Cs deposited at peripheral pellet as a result of higher linear heating rate during reactor operation, differences in test conditions such as fuel oxidation and eutectic reaction with cladding. Relationship between the pressure effect and the factors described above is desirable to be further examined by other future tests considering the scale effect and irradiation history of fuel.

The decrease in radionuclide release under elevated pressure may affect PWR source terms, accident management such as intentional primary system depressurization. Present analyses suggested that the intentional depressurization has many advantages such as delay in accident progression and mitigation of the source terms at time of early CV failure in spite of increase in radionuclide release into primary system. The effect of pressure on consequences needs to be evaluated systematically for various combinations of accident sequences and AM measures considering their occurrence probabilities.

Acknowledgments

The PWR and BWR fuel specimens used in VEGA program were provided by Kansai Electric Power Company and Tokyo Electric Power Company, respectively. Special thanks are due to Mr. T. Kudo, who is currently on loan from Japan Atomic Energy Agency, for his leading the VEGA heat-up tests and the post-test analyses.

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Table 1 Specifications of PWR-UO_2, BWR-UO_2 and ATR/MOX fuels

	PWR-UO_2	BWR-UO_2	ATR/MOX
Pellet diameter (mm) [a)]	8.1	10.4	12.4
Density (% Theoretical density)	95	97	95
Burn up (GWd/t)	47	56	43
Estimated Pu content after irradiation (wt%)	1.1	1.2	2.9
Linear heat rate (average) (kW/m)	18	26	28
Estimated pellet temperature (K) [b)] (center/peripheral)	1000/660	1500/870	1700/900
Fission gas release during reactor irradiation (%)	0.4	12	20

a) Nominal values as fabricated
b) Average temperature during reactor irradiation

Table 2 Reference test conditions

Test No.	VEGA-3	VEGA-8	VEGA-M1
Test fuel	PWR-UO_2	BWR-UO_2	ATR-MOX
Pellet weight (g)	11	13	24
Max temperature (K)	3123		
Heat - up rate (K/s)	1		
Pressure (MPa)	0.1		
Carrier gas and flow rate (Nm^3/min)	Helium / 0.001		

Table 3 Conditions for pressure effect tests and fractional releases at the end of each test

	VEGA-1	VEGA-2	VEGA-M1	VEGA-M2	VEGA-6	VEGA-7
Test fuel	PWR-UO_2		ATR/MOX		BWR-UO_2	
Re-irradiation	No				Yes	
Cladding	No				Yes	
Max temperature (K)	2773		3123		2773	
Heat-up rate (K/s)	1				0.33	
Pressure (MPa)	0.1	1.0	0.1	1.0	0.1	1.0
Carrier gas	Helium				Steam + Helium	
Flow rate (Nm^3/min)	0.001	0.005	0.001	0.005	0.00075 +0.001	0.00075 +0.004
Release results (%) (Half life)						
Cs-137 (30 year)	86	61	97	98	93	98
Sb-125 (3 year)	89	68	95	96		83
I-131 (8 day)					97	96
Te-132 (3 day)					98	98
Ba-140 (13 day)					49	34
Ru-106 (1 year)	5	0	6	3	14	6
Ru-103 (39 day)					16	7
La-140 (2 day)					3	4

Table 4 Summary of pressure effect on release observed in VEGA tests

Fuel (center / peripheral temp. K)	Carrier gas	Temp.	Noble gas	Cs / iodine	Low volatile FP
PWR without cladding (1000/660)	He	□ 2300K	●	○	□
		□ 2300K	●	○	●
BWR with cladding (1500/870)	Steam + He	□ 2300K	×	×	□
		□ 2300K	×	×	○
MOX without cladding (1700/900)	He	□ 2300K	×	×	□
		□ 2300K	○	□	●

○ Effect measured by test, ● Not measured but mechanistically possible, □ Small effect measured, × No effect measured, – Mechanistically impossible (Not measured)

Table 5 Comparison of Cs release at the end of SFD 1-4 test between analysis and measurement

	SFD1-4 Measurement	CORSOR	CORSOR-M	ORNL-Booth	1/√ P CORSOR-M
Cs fractional release (%)	Cs-137: 51± 15 Cs-134: 39± 14	83	72	60	40

Table 6 Effect of decrease in radionuclide release under elevated pressure on timings of occurrence and source terms in TQUX sequence of BWR5

Events	TQUX (late CV failure)		TQUX (early CV failure)	
(min)	CORSOR-M	1/√ P CORSOR-M	CORSOR-M	1/√ P CORSOR-M
Core melt initiation	53	53	53	53
Core support	184	88	184	88
Core collapse	129	120	129	120
Vessel failure	313	260	313	260
Pedestal failure	768	745	765	742
Containment failure	1996	2038	313	260
Release fraction to environment (%)	17.8	14.8	1.73	19.6

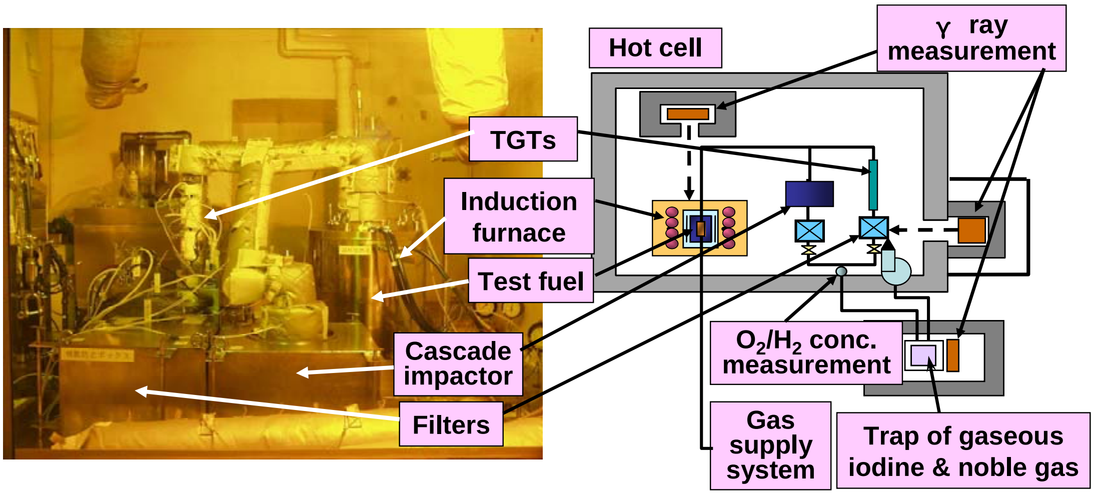

Fig. 1 Schematic of VEGA test apparatus

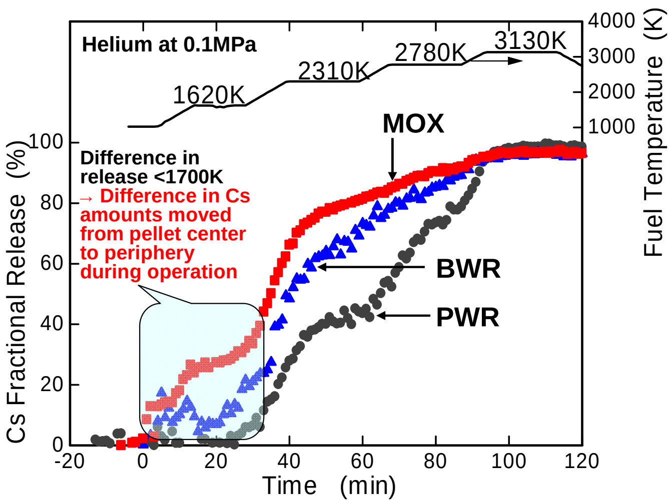

Fig. 2 Comparison of Cs fractional releases at 0.1MPa among PWR, BWR and ATR/MOX fuels

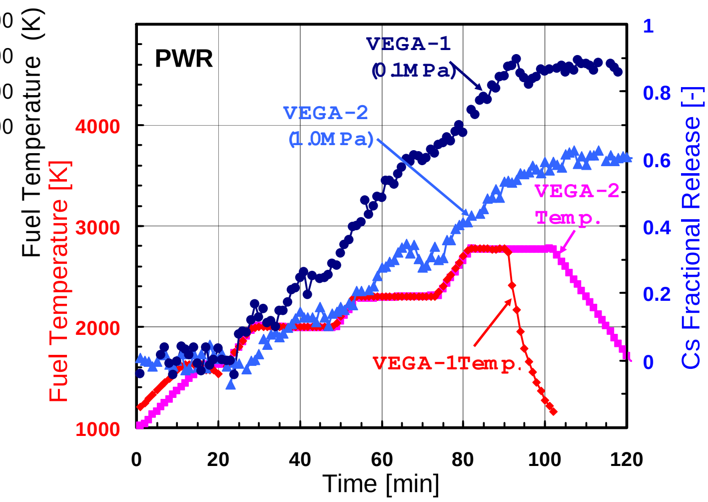

Fig. 4 Fuel temperatures and Cs fractional releases from PWR fuel at 0.1 and 1.0MPa

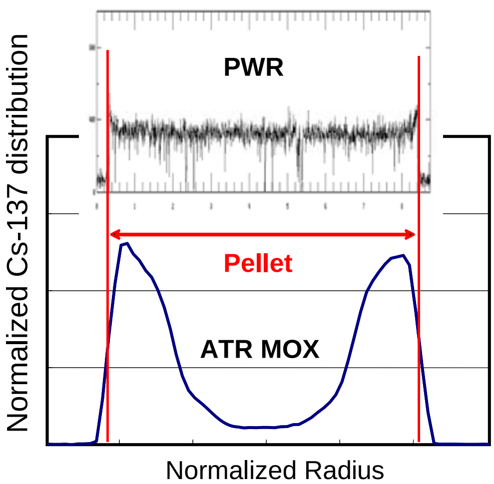

Fig. 3 Measured distribution of Cs-137 in diameter direction of PWR and ATR/MOX pellets

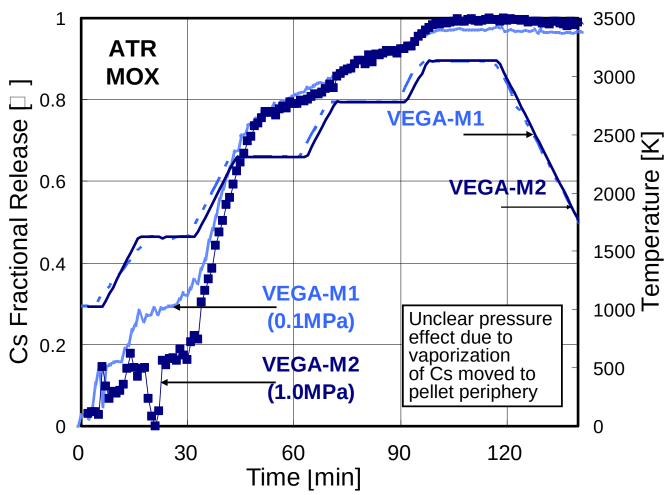

Fig. 5 Fuel temperatures and Cs fractional releases from ATR/MOX at 0.1 and 1.0MPa

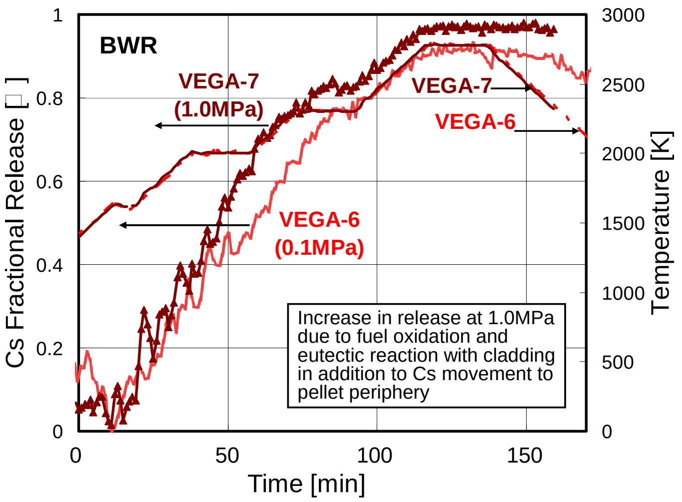

Fig. 6 Fuel temperatures and Cs fractional releases from BWR fuel at 0.1 and 1.0MPa

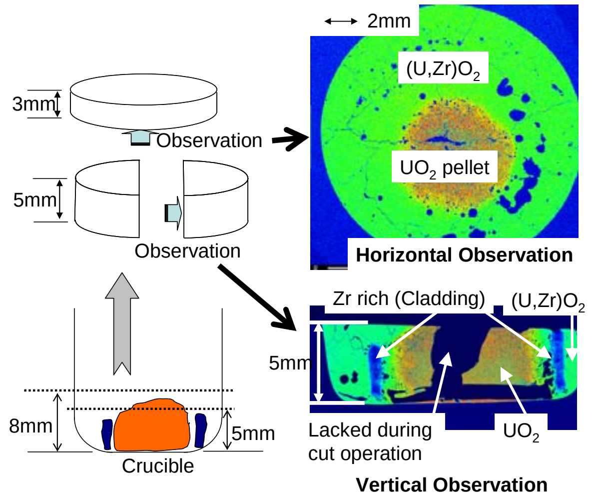

Fig. 9 Cross section of tested UO_2 with cladding under steam atmosphere

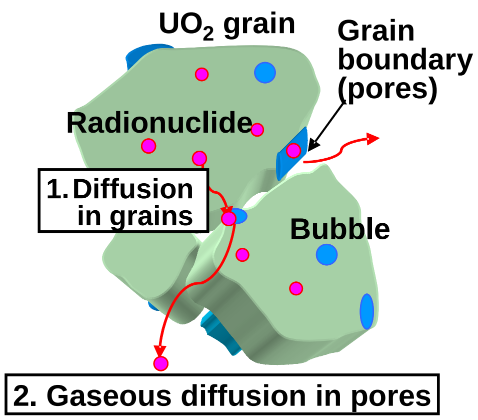

Fig. 7 Schematic of UO_2 grain & pores and process of radionuclide release

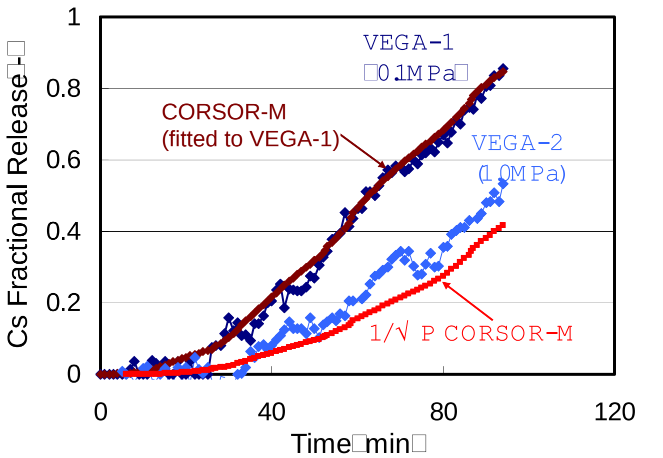

Fig. 10 Cs release from PWR fuel at 0.1 & 1.0MPa calculated by proposed release model

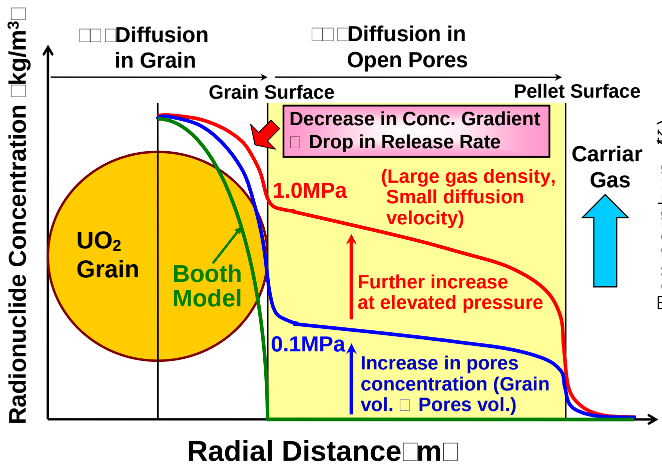

Fig. 8 Expected distribution of radionuclide concentration in pellet at elevated pressure

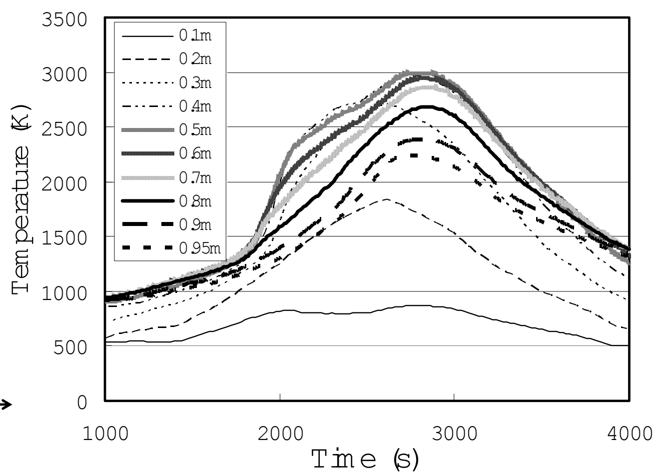

Fig. 11 Temperature distribution for SFD 1-4 bundle

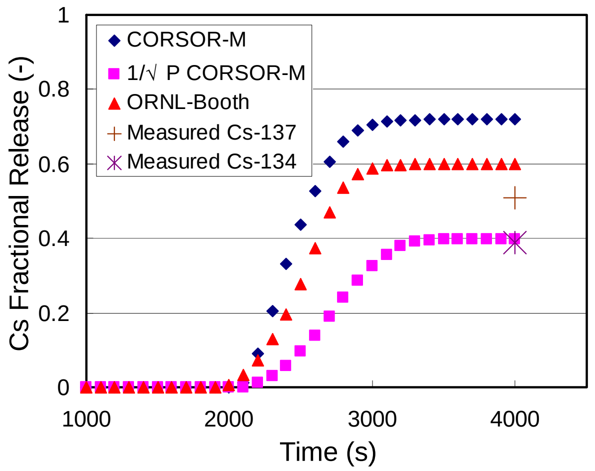

Fig. 12 Calculated Cs release evolution during SFD 1-4 test

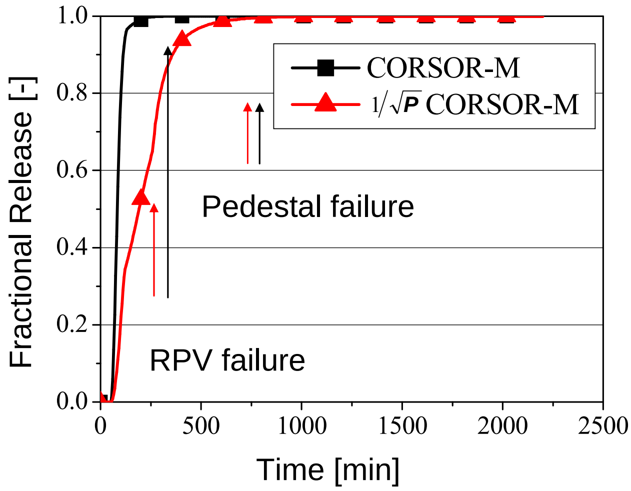

Fig. 13 Comparison of CsI release from fuel during TQUX between CORSOR-M and 1/ √ P CORSOR-M models

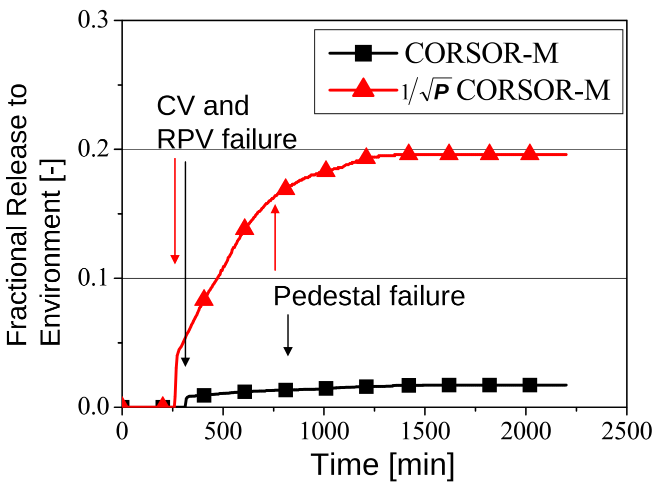

Fig. 14 Comparison of CsI release to environment during TQUX with early CV failure between CORSOR-M and 1/√P CORSOR-M models

Appendices

- **Workshop Programme**
- **Organising Committee and Session Chairpersons**
- **List of Participants**

OECD/NEA Workshop "Implementation of Severe Accident Management (SAM) Measures

ISAMM2009

Böttstein, Switzerland, October 26-28, 2009

Workshop Programme

Sunday, Oct. 25,2009	
Time	**Session / Event**
16:00-19:00	Registration/Apero (Kurhotel in Bad Zurzach)
18:00-19:00	Organizers' meeting
Monday Oct. 26,2009	
Time	**Welcome and Introduction** **Session chair/co-chair: S. Guentay/A. Amri** **Meeting Room: Festsaal**
8:30	M. Jermann , Vice Director, PSI
8:50	U. Weidmann, Director, NPP-Beznau,
9:10	S. Guentay, Vice Chair, CSNI/WGAMA
	Session 1: Current Status & Insights of SAM, Part 1 **Session chair/co-chair: N. Suh/A. Torri**
9:30	Recent IAEA Activities in the area of Severe Accident management and Level-2 PSA, *A. Lyubarskiy*
10:00	Technical Challenges in Applying SAMG Methodology to Operating CANDU Plants, *K. Dinnie, R. Henry, M. Chai*
10:30	Break
11:00	Accident Management in German NPPs: Status of Implementation and The Associated Role of PSA Level 2, *M. P. Scheib, M. K. Schneider*
11:30	Circumstances and Present Situation of Accident Management Implementation in Japan, *H. Fujimoto, K. Kondo, T. Ito, Y. Kasagawa, O. Kawabata, M. Ogino and M. Yamashita*
12:00	Progress in the Implementation of Severe Accident Measures on the Operated French PWRs – Some IRSN Views and Activities, *E. Raimond, G. Cenerino, N. Rahni, M. Dubreuil, F. Pichereau*
12:30-14:00	Lunch
Time	**Session 2: Current Status & Insights of SAM, Part 2** **Session chair/co-chair: H. Fujimoto/ M. Sonnenkalb** **Meeting Room: Festsaal**
14:00	Perspectives on Severe Accident Mitigation Alternatives for US Plant License Renewal, *T. Ghosh, R. Palla, D. Helton*
14:30	Effect of SAMG on the Level 2 PSA of Korean Standard Nuclear Power Plant, *Y. Jin, K.I. Ahn*
15:00	Insights from a full-scope Level 1/Level 2 all operational states PRA with respect to the efficiency of Severe Accident Management actions, *J.U. Klügel, S. B. Rao, T. Mikschl, D. Wakefield, A. Torri, V, Pokorny*

15:30	Break
16:00	PRA Level 2 Perspectives on the SAM during the Shutdown States at the Loviisa NPP, *S. Siltanen, T. Routamo, T. Purho, H. Tuomisto*
16:30	Development of the SAM strategy for PAKS NPP on the basis of Level 2 PSA, *J. Elter, G. Lajtha, É. Tóth, Z. Téchy*
17:00	Development of Technical Bases For Severe Accident Management in New Reactors, *E. L. Fuller, H. G. Hamzehee*
17:30	**End of day 1**

Tuesday Oct. 27, 2009		
Time	**Session / Event**	**Parallel Session**
	Session 3: PRA Modelling Issues Session chair/co-chair: J. Primet /V. Dang Meeting Room: Festsaal	**Session 8: Physical phenomena affecting SAM Part 1 Session chair/co-chair: F. Kappler/ M. D. Leteinturier Meeting Room: Bogenkammer**
9:00	Some international efforts to progress in the harmonization of L2 PSA development and their implication within OECD-CSNI, ANS, IAEA, EC (ASAMPSA2), *E. Raimond, S. Guentay, C. Bass, D. Helton, A. Lyubarskiy*	Experimental Investigation of Melt Debris Agglomeration with High Melting Temperature Simulant Materials, *P. Kudinov,A. Karbojian, C.-T. Tran*
9:30	Accident Management and Risk Evaluation of Shutdown Modes at Beznau NPP, *M. Richner, S. Zimmermann, J. Birchley, T. Haste, N. Dessars*	Approach to Prediction of Melt Debris Agglomeration Modes in a LWR Severe Accident, *P. Kudinov, M. Davydov*
10:00	The Role of Severe Accident Management in the Advancement of Level 2 PRA Modelling Techniques, *D. Helton, J. Chang, N. Siu, M. Leonard, K. Coyne,*	A Fuel Coolant Interaction Programme (FCI) devoted to reactor case., *P. Piluso, S. W. Hong*
10:30	Break	
11:00	Overview of the Modelling of Severe Accident Management in the Swiss Probabilistic Safety Analyses, *V.h N. Dang, G. Schoen, B. Reer*	Improved Molten Core Cooling Strategy in a Severe Accident Management Guideline, *J.H Song, N.D Suh, C.W Huh*
11:30	Extended Use of MERMOS to assess Human Failure Events in Level 2 PSA, *H. Pesme, P. Le Bot*	Summary and Outcome of OECD-Workshop on In-vessel Melt Pool Retention, *B. Clément*
12:00–13:15	Lunch	
	Session 4: Code Analysis for Supporting SAMGs Part 1 Session chair/co-chair: E. Raimond/ Y. Liao Meeting Room: Festsaal	**Session 8: Physical phenomena affecting SAM Part 2 Session chair/co-chair: F. Kappler/M. D. Leteinturier Meeting Room: Bogenkammer**
13:15	Best-Estimate Calculations of Unmitigated Severe Accidents in State-of-the-Art Reactor Consequence Analyses, *C. G. Tinkler, K.C. Wagne, M. T. Leonard, J. H. Schaperow*	Simulation of Ex-Vessel Debris Bed Formation and Coolability in a LWR Severe Accident, *S. Yakush, P. Kudinov*
13:45	Deterministic Evaluation of Quantitative Health Objective and Target of Severe Accident Management, *C.W. Huh, N.D. Suh, G.H. Jung*	Substantiation of strategy of water supply recovery to steam generators at in-vessel severe accident phase for VVER-1000 Balakovo NPP, *A. Suslov, V. Mitkin.*
14:15	Verification of the SAMG for PAKS NPP with MAAP Code Calculations, *G. Lajtha, Z. Téchy, J. Elter, É. Tóth,*	Ambient Pressure-Dependent Radionuclide Release from Fuel Observed In VEGA Tests under Severe Accident Condition and Influence on Source Term Evaluation,

14:45	Break	
15:15	Treatment of Accident Mitigation Measures in State-of-the-Art Reactor Consequence Analyses, *C. G. Tinkler, K.C. Wagner,* *M. T. Leonard, J. H. Schaperow*	
15:45	Best Practices Applied to Deterministic Severe Accident and Source Term Analyses for PSA Level 2 for German NPPs, *M. Sonnenkalb, N. Reinke, H. Nowack.*	
	Session 7: Design Modifications for Implementation of SAM **Session chair/co-chair:** **J. Primet/A. Lyubarskiy** **Meeting Room: Festsaal**	**Session 5: Code Analysis for Supporting SAMGs Part 2** **Session chair/co-chair:** **M. Leonard/ M. G. Cenerino** **Meeting Room: Bogenkammer**
16:30	A Novel Process for Efficient Retention of Volatile Iodine Species in Aqueous Solutions during Reactor Accidents, *S. Guentay, H. Bruchertseifer, H. Venz,* *F. Wallimann, B. Jaeckel*	Severe Core Damage Accident Analysis for a CANDU Plant**,** *P. Mani Mathew, S. M. Petoukhov,* *M. J. Brown and, B. Awadh.*
17:00	Development of Leibstadt NPP Severe Accident Management Guidelines for Shutdown Conditions (SSAMG), *W. Hoesel*	Time Window for Steam Generator Secondary Side Reflooding to Mitigate Large Early Release Following SBO-Induced SGTR Accidents, *Y. Liao, S. Guentay*
17:30	Design Modifications of the Mochovce Units 3 & 4 dedicated to Mitigation of Severe Accident Consequences, providing Conditions for Effective SAM, *M. Cvan, D. Šiko*	On the Effectiveness of CRGT Cooling as a Severe Accident Management Measure for BWRs, *M. Weimin, T. Chi-Thanh*
18:00	**End of day 2**	
19:30-22:00	**Dinner**	

Wednesday Oct. 28, 2009	
Time	**Session / Event** **Meeting Room: Festsaal**
	Session 5: Code Analysis for Supporting SAMGs Part 2 (Cont)
8:30	Ex-Vessel Corium Management for the VVER-1000 Reactor, *B. Kujal*
	Session 6: Decision making, Tools, Training, Risk Targets and Entrance to SAM **Session chair/co-chair: D. Helton/P. Le-Bot**
9:10	Criteria for the Transition to Severe Accident Management, *B. Prior*
9:40	Use of The Software Module Sprint in the Netherlands, *M. Slootman.*
10:10	Safety Goals and Risk Targets for Severe Accidents in View of IAEA Recommendations, *J. Vitázková, E. Cazzoli*
10:40	Break
11:10	Development, Validation and Training of Severe Accident Management Measures, *A. Torri, V. Pokorny, U. Lüttringhaus*
11:40	Severe Accident Training in Spain: Experiences and Relevant Features, *R. Martínez, J.Benavides, J. M. de Blas, M. A. Catena, I. Sol*
12:10-13:50	Lunch
13:50	**Panel Discussions** **Session chair/co-chair: S. Guentay/A. Amri** **Part I:** **Topic:** ISAMM2009 Highlights by Session chairs/co-chairs
15:05	**Part II:** **Topic 1:** Human and Organizational Aspects of SAM: their importance vs. technical issues by *C. Huh (KINS, Korea)* **Topic 2:** Effectiveness of current SAMG implementation - How can consequence analyses be used to improve the effectiveness of SAM?, by *Mark Leonard (Dycoda, US)*
16:10	Final remarks, *A. Amri (OECD/NEA)*
16:20	**Closure of Workshop ISAMM2009**

Thursday Oct. 29, 2009	9:00-17:00	**Organizers' meeting**
Friday Oct. 30, 2009	9:00-16:00	**Organizers' meeting**

OECD Workshop on the Implementation Of Severe Accident Management Measures (ISAMM-2009), Schloss Böttstein, 26-28 October 2009

Organizing Committee

Salih Güntay, Paul Scherrer Institute, Switzerland, General Chair

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François Kappler, Electricité de France, France

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Haruo Fujimoto, Japan Nuclear Energy Safety Organization, Japan

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Don Helton, US Nuclear Regulatory Commission, USA

Abdallah Amri, OECD Nuclear Energy Agency, France, NEA Secretariat

Session Chairs and Co-chairs		
Sessions	**Date/Time/Location**	**Chair/Co-chair**
Welcome&Introduction	26.10.2009/8:30-9:30/ Festsaal	**S. Guentay/A. Amri**
Session 1: Current Status & Insights of SAM Part 1	26.10.2009/9:30-12:30/ Festsaal	**N. Suh/A. Torri**
Session 2: Current Status & Insights of SAM Part 2	26.10.2009/14:00-17:00/ Festsaal	**H. Fujimoto/M. Sonnenkalb**
Session 3: PRA Modeling Issues	27.10.2009/8:30-12:00/ Festsaal	**J. Primet /V. Dang**
Session 4: Code Analysis for Supporting SAMGs Part 1	27.10.2009/13:15-15:45/ Festsaal	**E. Raimond/Y. Liao**
Session 5: Code Analysis for Supporting SAMGs Part 2	27.10.2009/16:30-18:00/ Bogenkammer 28.10.2009/8:30-9:00/ Festsaal	**M. Leonard/ M. G. Cenerino**
Session 6: Decision making, Tools, Training, Risk Targets and Entrance to SAM	28.10.2009/9:10-12:40/ Festsaal	**D. Helton/P. Le-Bot**
Session 7: Design Modifications for Implementation of SAM	27.10.2009/16:30-18:00/ Festsaal	**J. Primet/A. Lyubarskiy**
Session 8: Physical phenomena affecting SAM	27.10.2009/8:30-12:00/ Bogenkammer	**F. Kappler/M. D. Leteinturier**
Panel Discussions and Closure	28.10.2009/13:50-16:20/ Festsaal	**S. Guentay/A. Amri**

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