Unclassified **NEA/CSNI/R(2010)10/PART2**

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/PART2
Unclassified

Implementation of Severe Accident Management Measures, ISAMM 2009

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

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

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 33 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

This work is published on the responsibility of the Secretary-General of the OECD. The opinions expressed and arguments employed herein do not necessarily reflect the official views of the Organisation or of the governments of its member countries.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 28 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Portugal, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

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.

Corrigenda to OECD publications may be found online at: *www.oecd.org/publishing/corrigenda*.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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.

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PSI Report Nr.: 10y-
NEA/CSNI/R(2010)10

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

OECD

Implementation of Severe Accident Management Measures, ISAMM 2009

Workshop Proceedings
Schloss Böttstein, Switzerland
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in collaboration with
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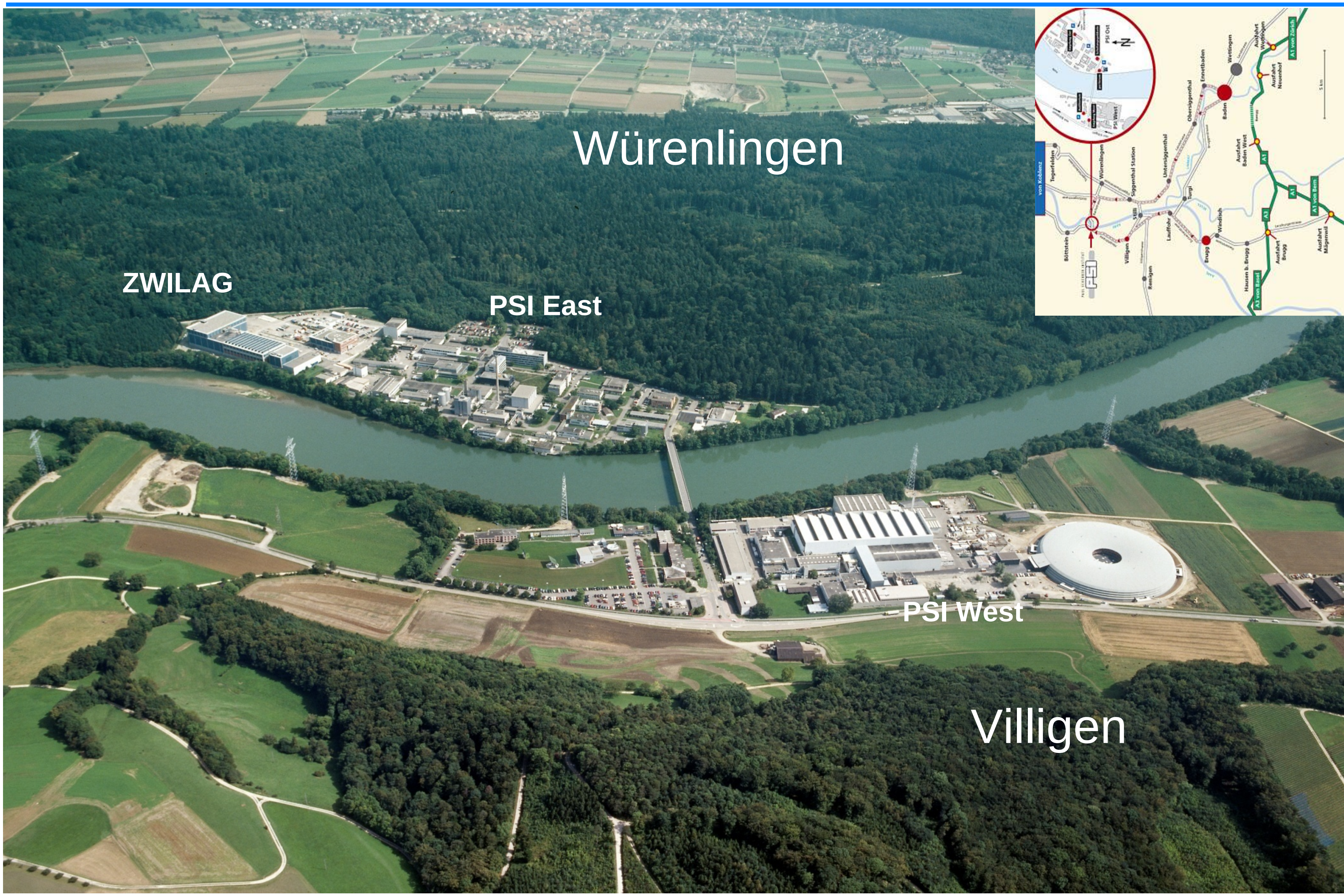
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ZWILAG
PSI East
PSI West
Villigen

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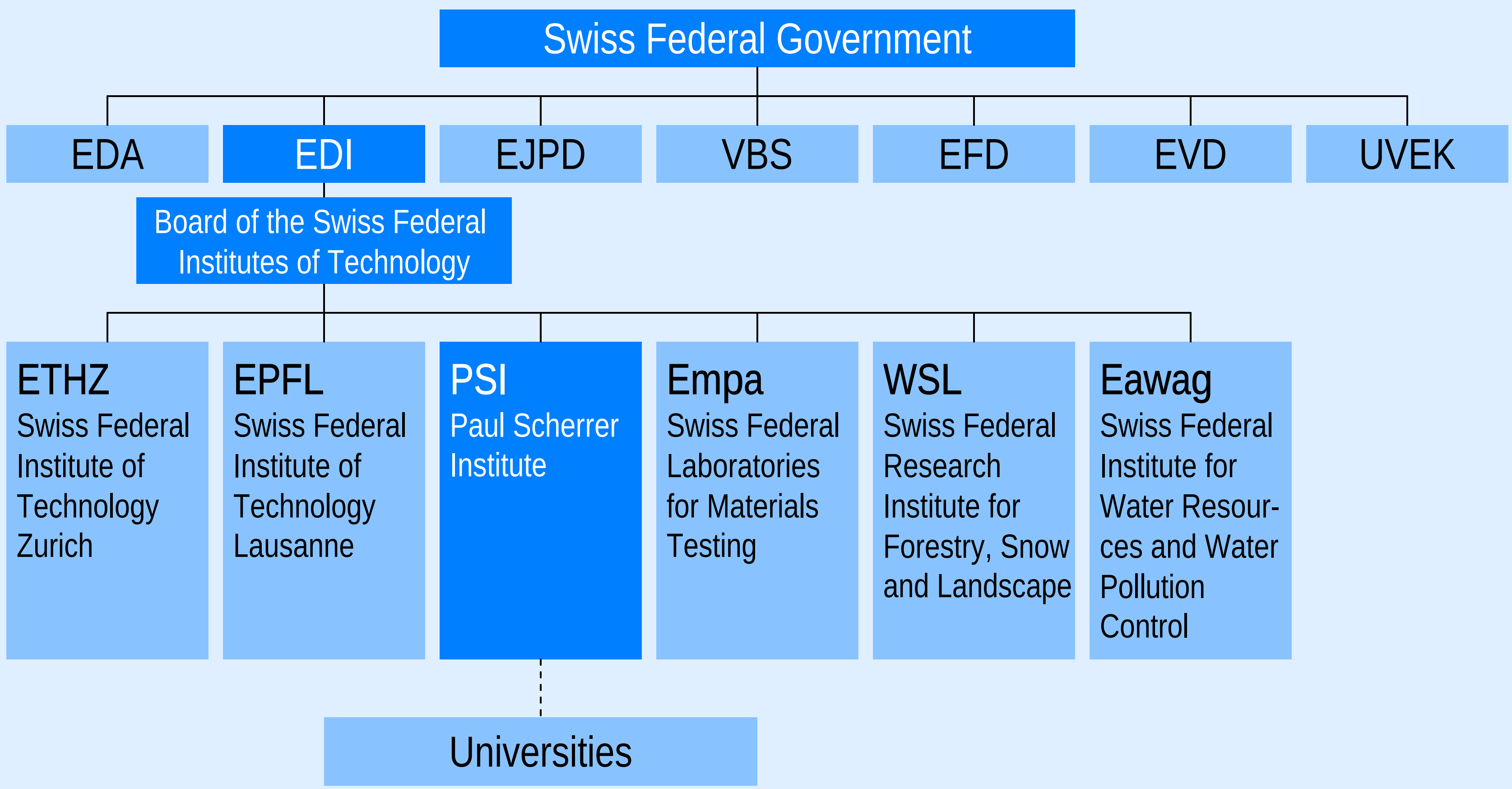
Embedding



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Mission

- To play a leading role on an international level in
  – physics of condensed matter and materials sciences
  – structural biology
  – radiochemistry, radiopharmacy and proton radiation therapy
  – particle physics

  by using large-scale facilities
  (SLS, SINQ, SμS, particle beams)

- To be a UserLab for external science community

- Energy research, primarily using complex facilities, towards an efficient, environmentally friendly and reliable energy supply

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Key Figures 2009

PSI funds (global budget)	244	MCHF
External funding	55	MCHF
Staff / FTE	~ 1350	PJ
Of which externally financed	~ 320	PJ
Doctoral students	~ 300	
Apprentices	80	
External users	~ 2000	
Number of scientific publications	~ 900	
PSI-employees with teaching duties at ETH and universities	~ 75	

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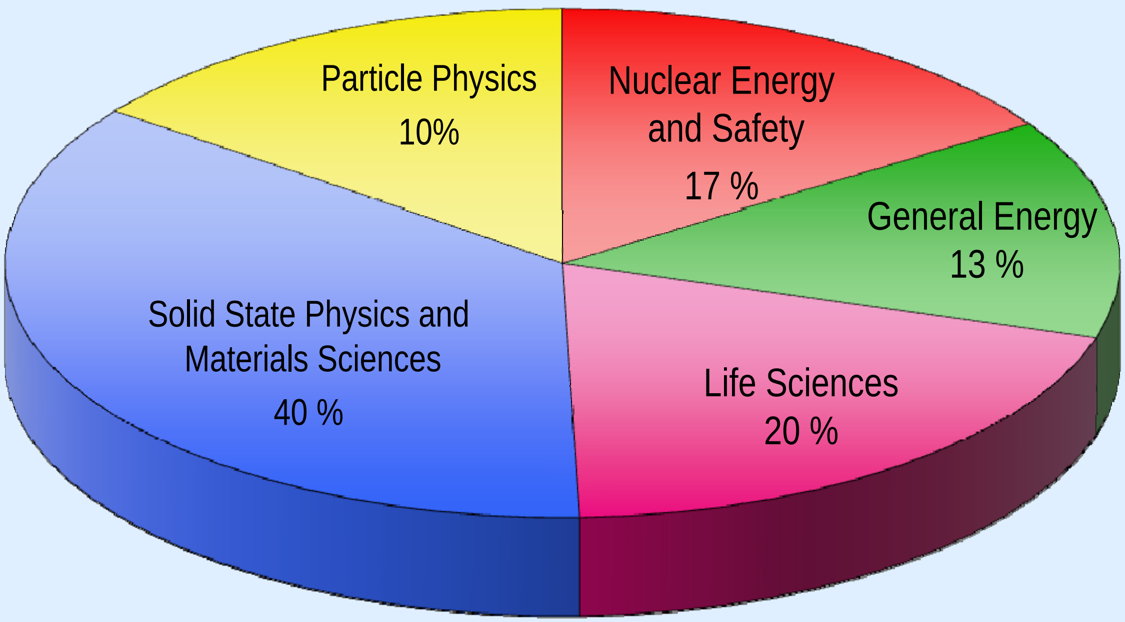
JM01 2009

Ressource Distribution 2009

ca. 300 MCHF (incl. external funding)



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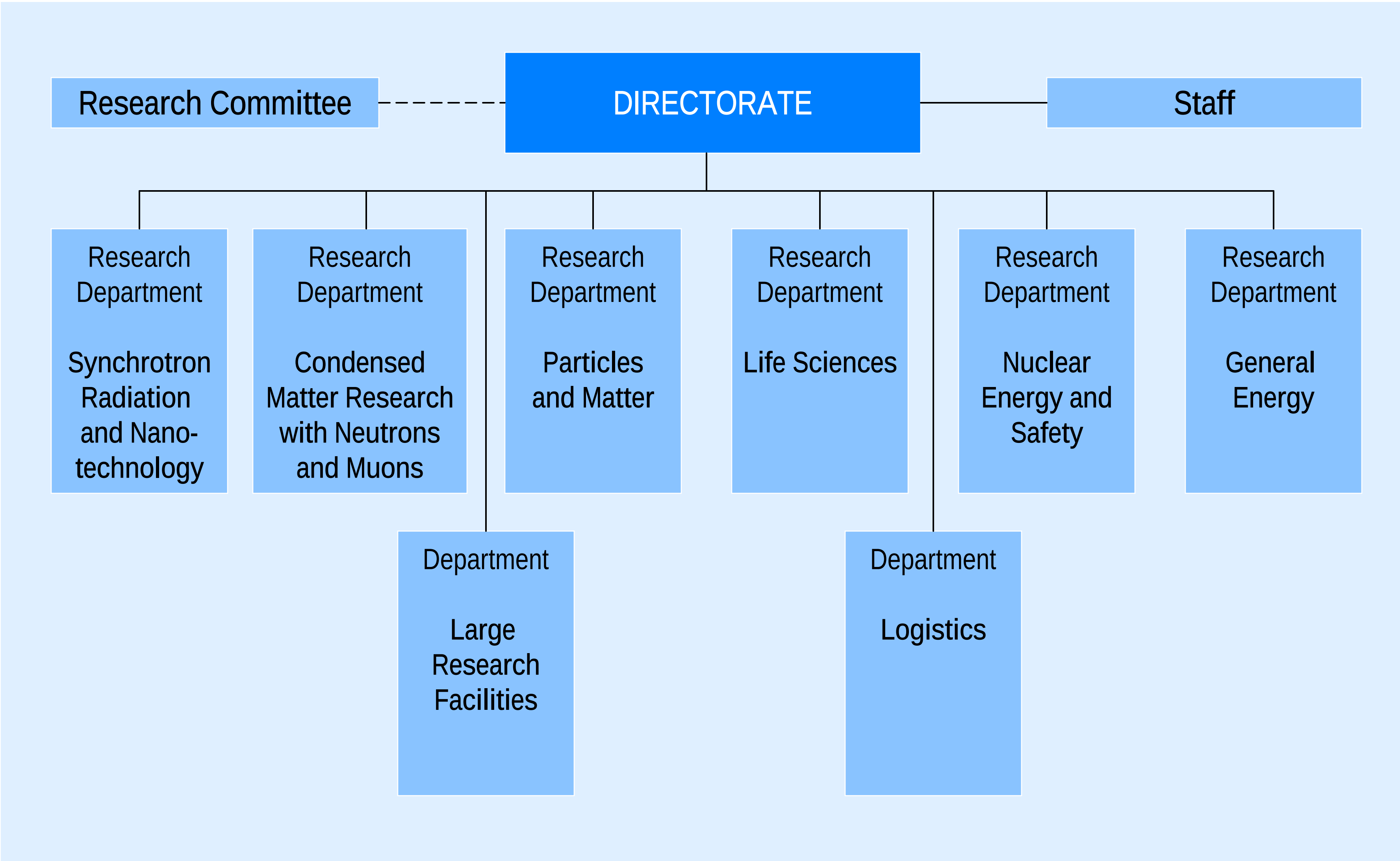

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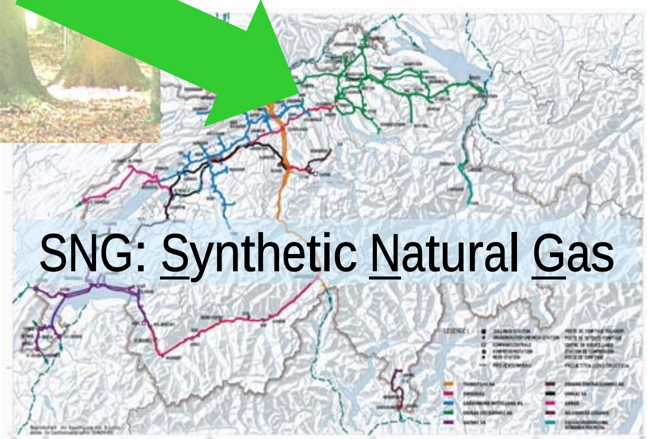
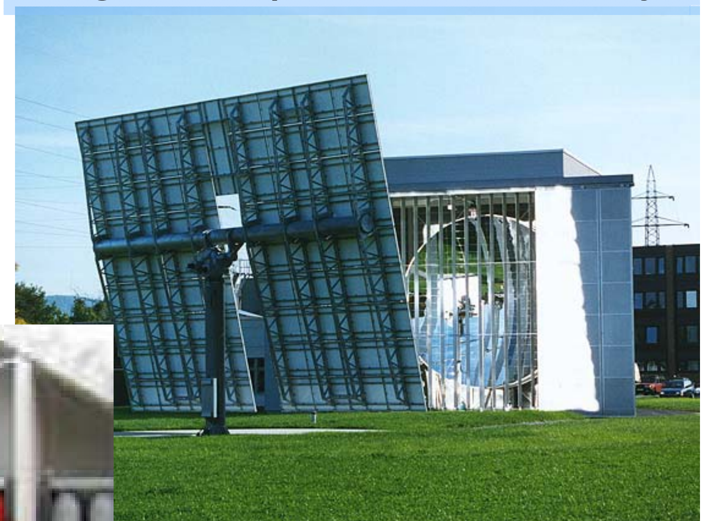
Non-nuclear Energy Research

Use of Biomass



SNG: Synthetic Natural Gas



High Temp Solarchemistry

Hydrogen

Electrochemistry
Fuel cells, Batteries

Combustion

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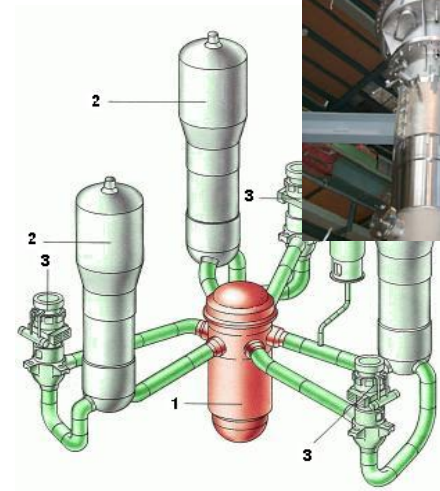
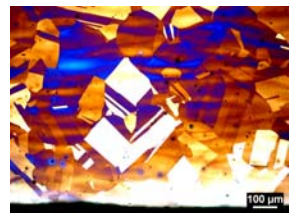
Nuclear Energy Research

Safety

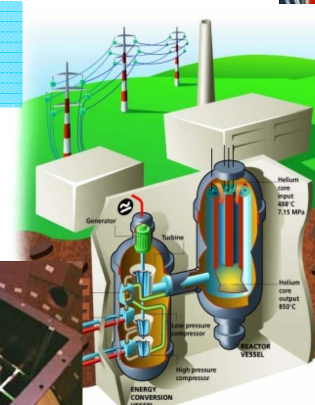
- Transient analysis
- Severe accidents
- Materials testing

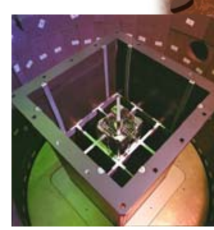




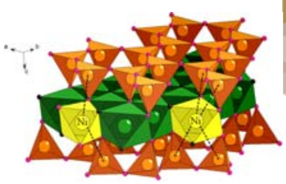
Generation IV

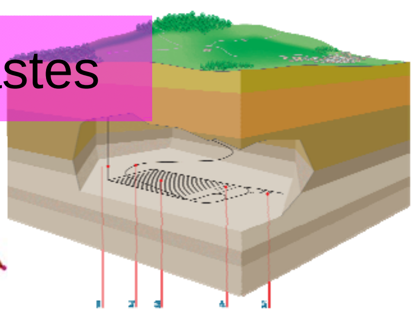
- Fuel cycles optimisation
- HT materials





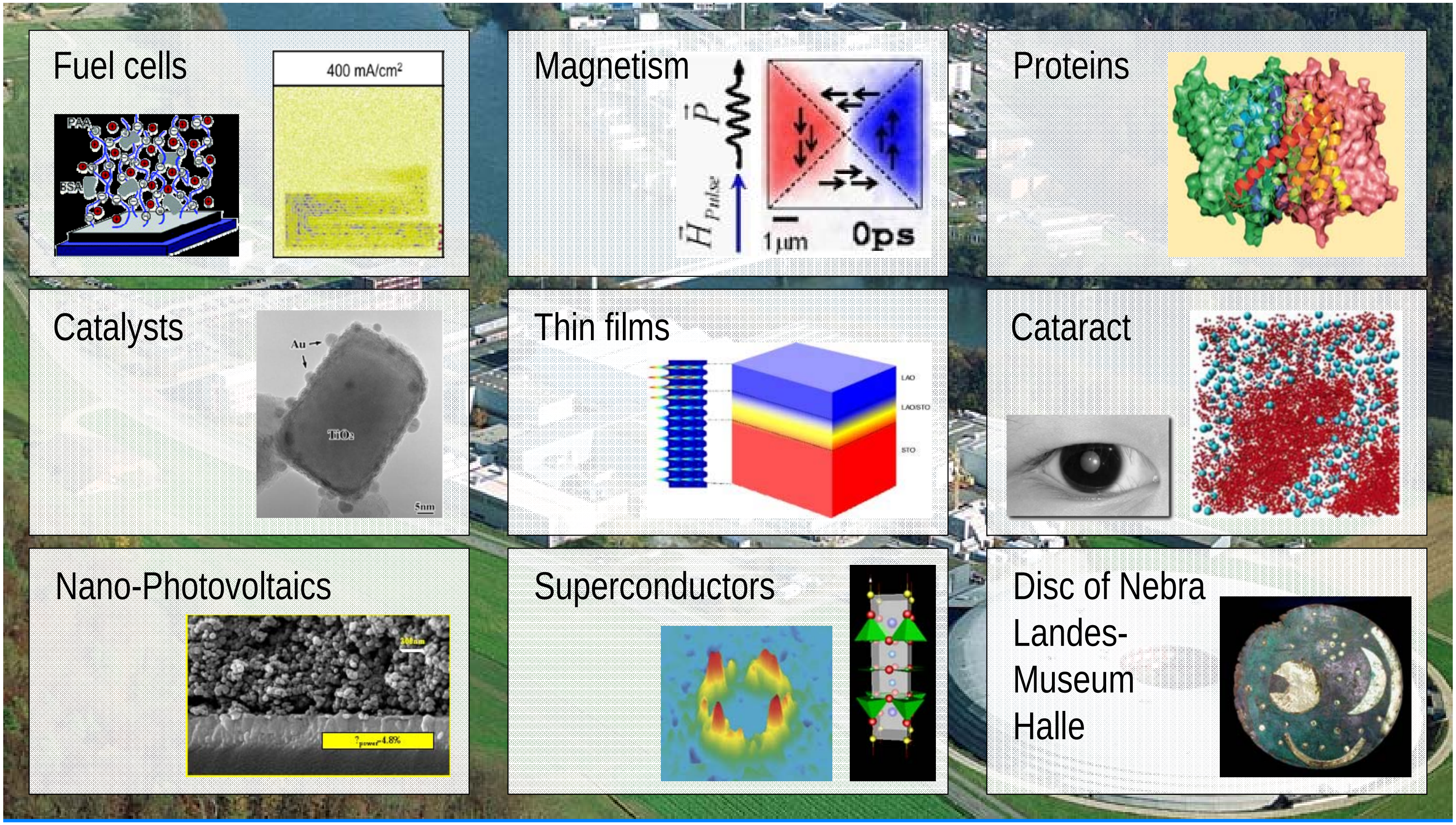
Nuclear wastes



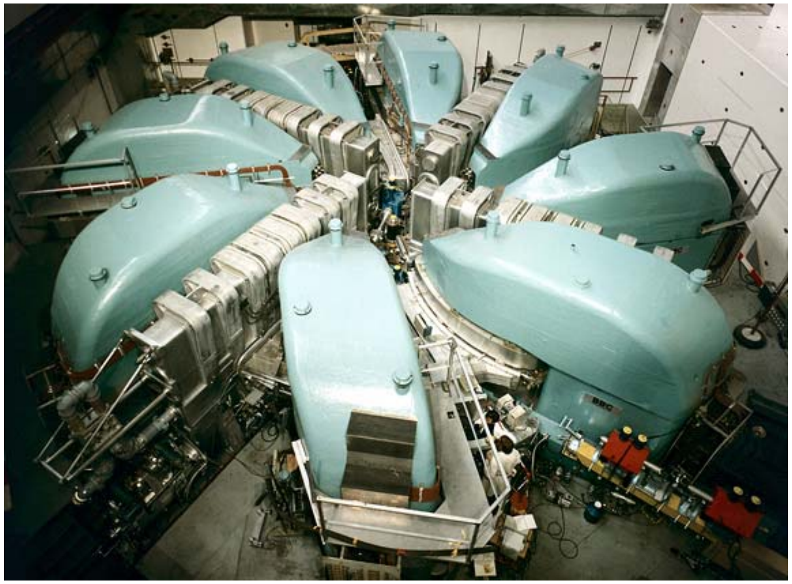


- Interactions of radionuclei in the near and far field
- Safety assessments of repositories

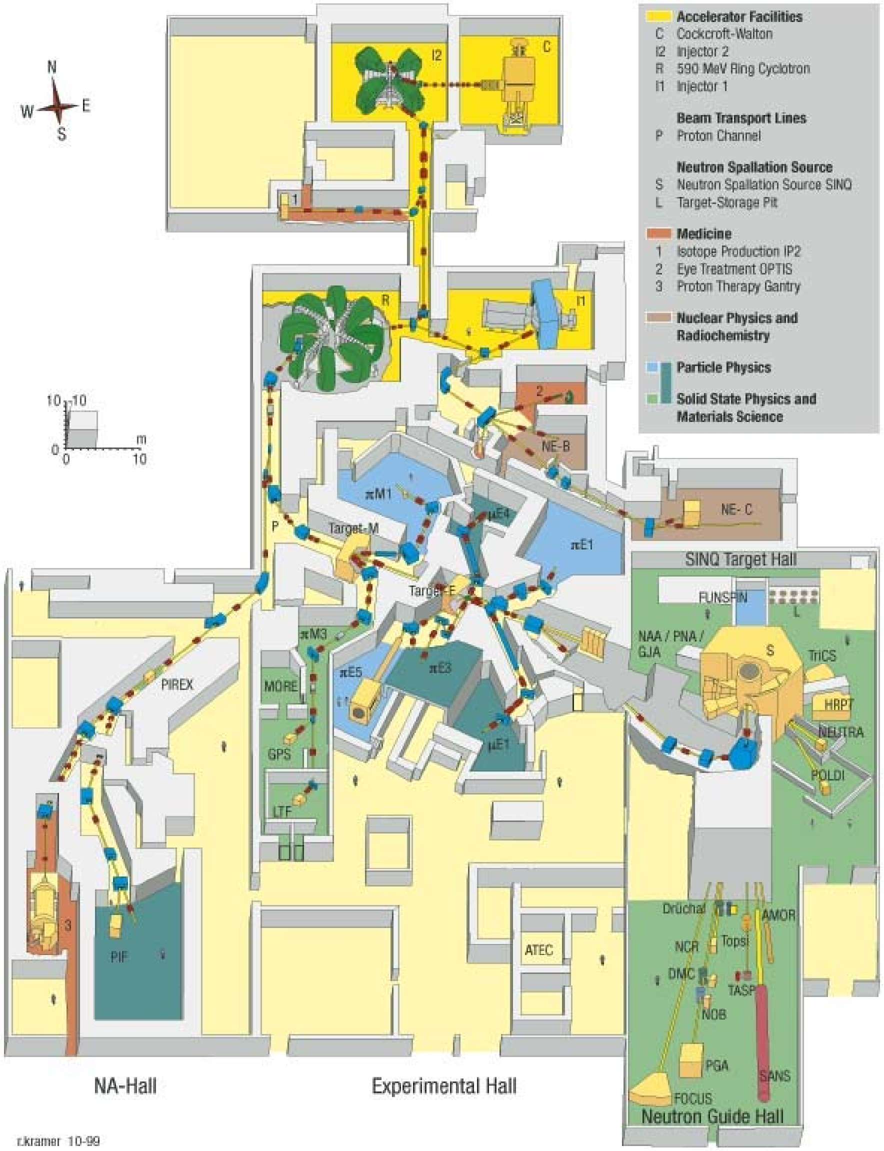
Use of Muon-, Neutron-, Photon-beams

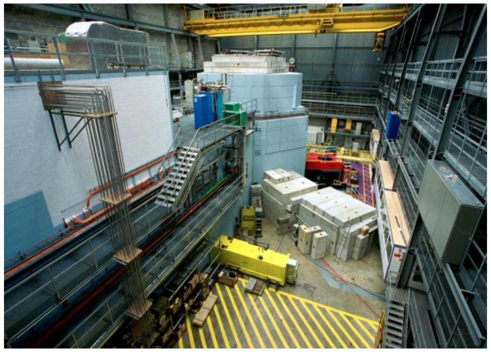


Proton-accelerator

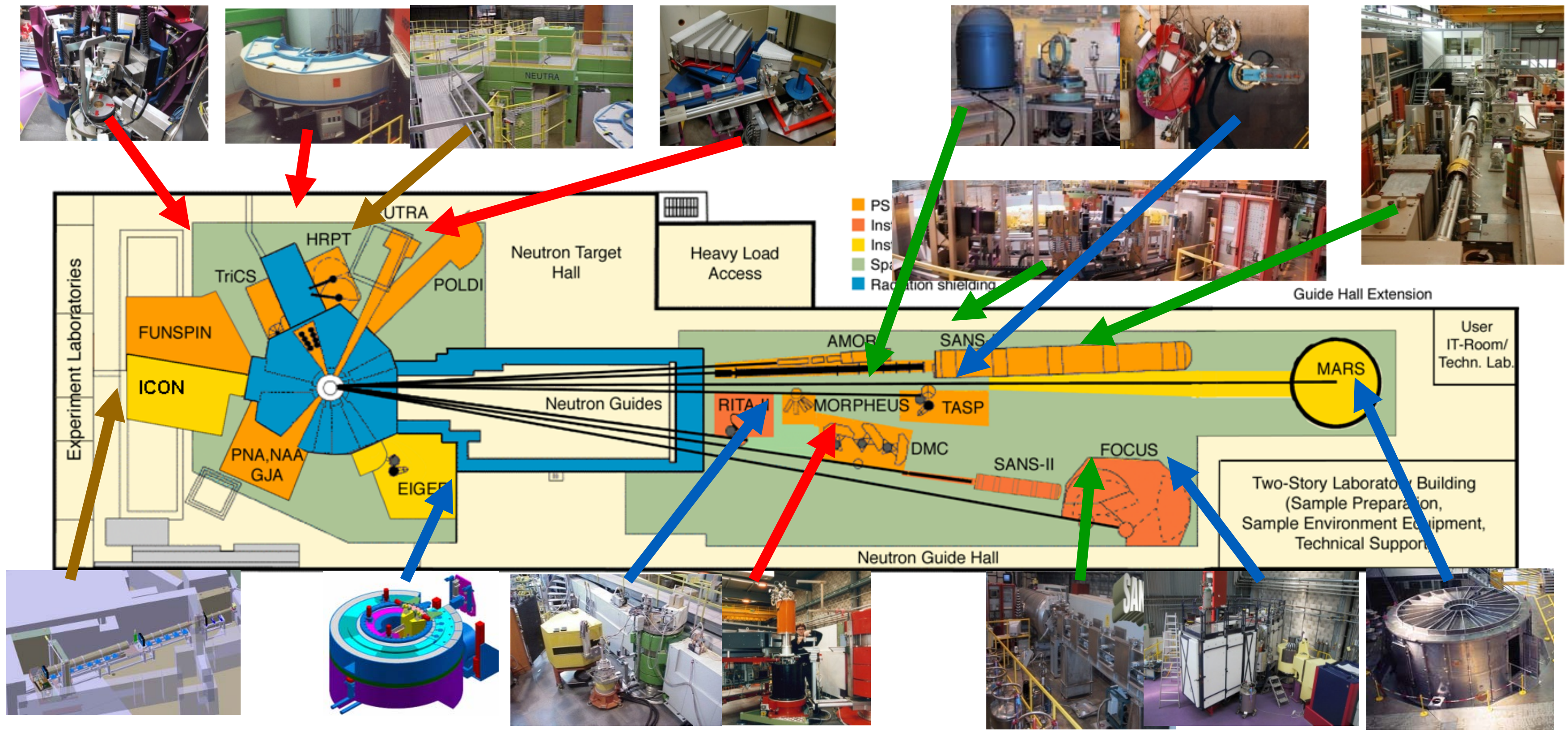


World most powerful accelerator of this type

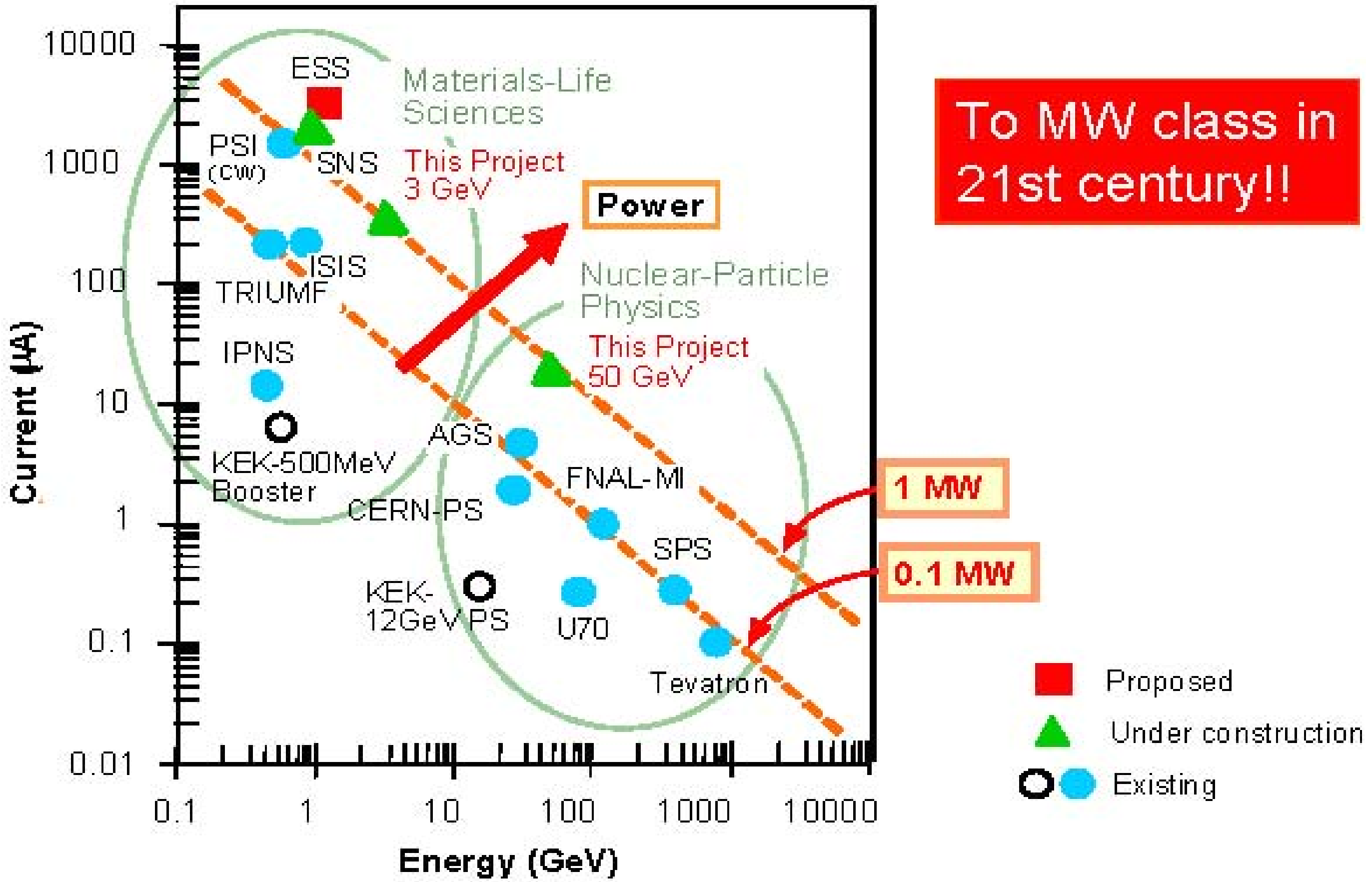




Spallation Neutron Source SINQ & Instrumentation

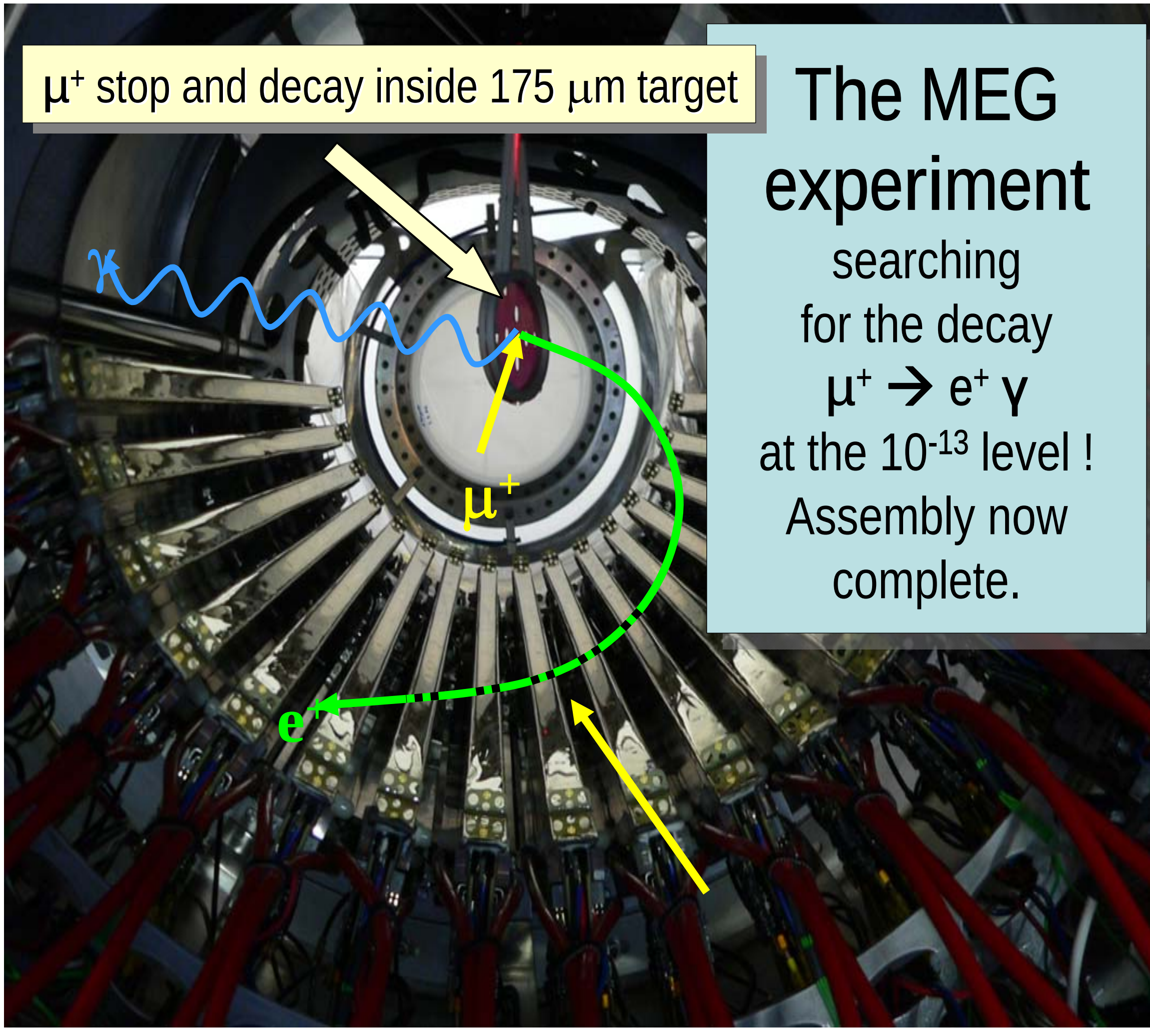


World's Proton Accelerators



To MW class in 21st century!!

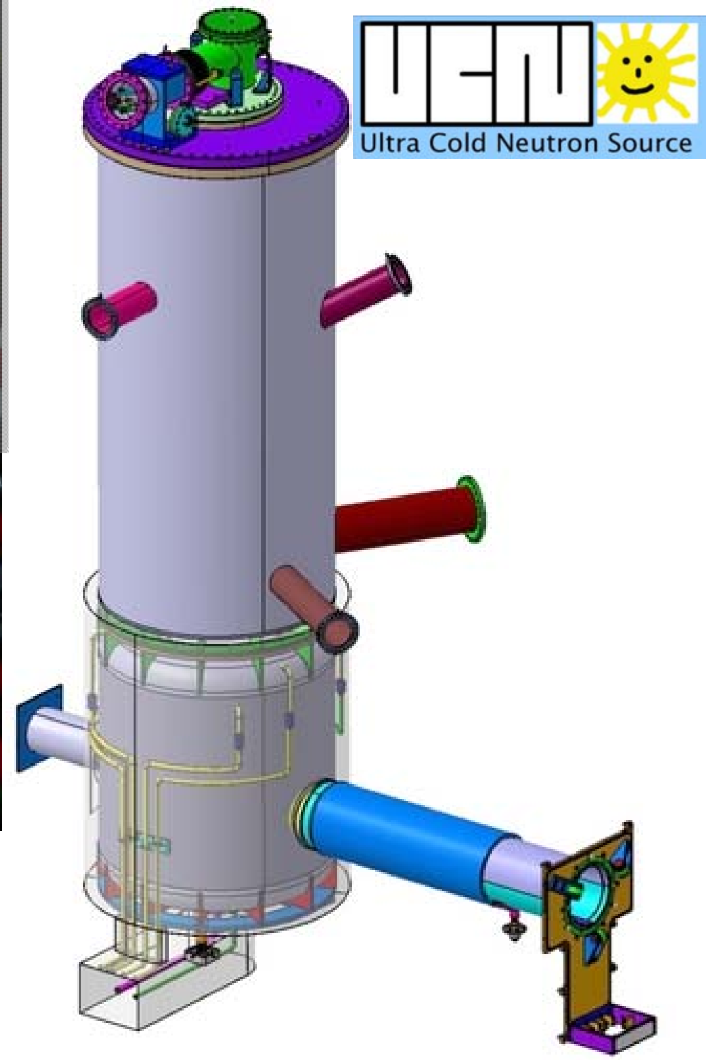
Particle Physics



The MEG experiment

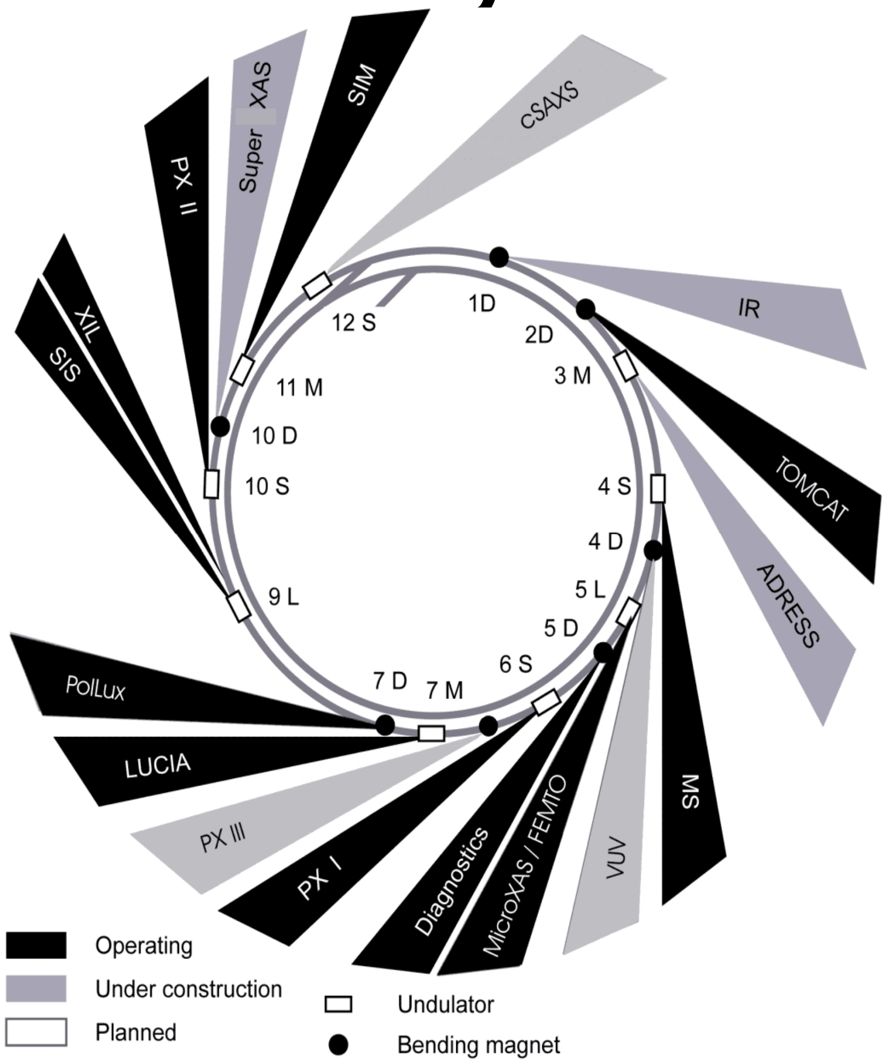
searching for the decay $\mu^+ \rightarrow e^+ \gamma$ at the 10^{-13} level ! Assembly now complete.

e+ are measured in drift chambers built by PSI's Detector Group

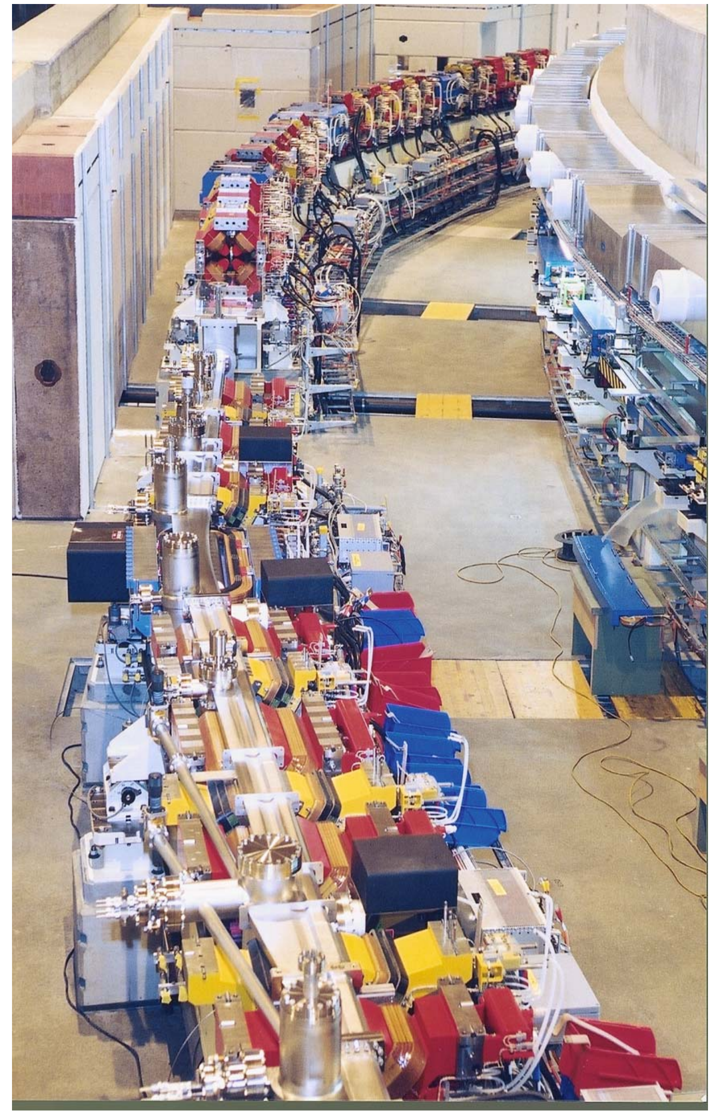


EDM Experiment

SLS - Layout



SIM
Super XAS
PX II
cSAXS
XIL
SIS
IR
TOMCAT
ADRESS
PolLux
LUCIA
PX III
PX I
Diagnostics
MicroXAS / FEMTO
VUV
MS
1D
2D
3 M
4 S
4 D
5 L
5 D
6 S
7 M
7 D
9 L
10 S
10 D
11 M
12 S
Operating
Under construction
Planned
Undulator
Bending magnet



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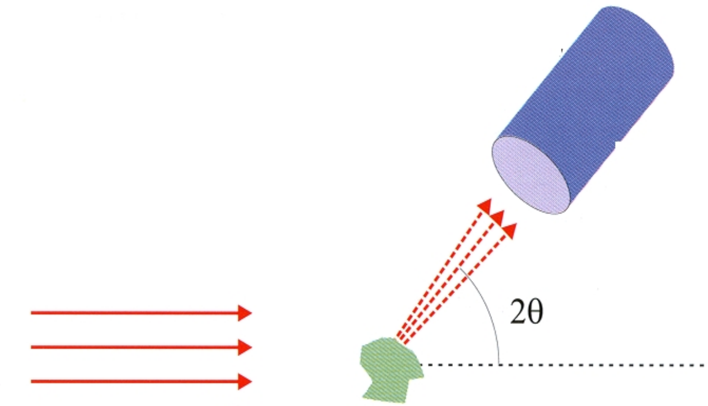
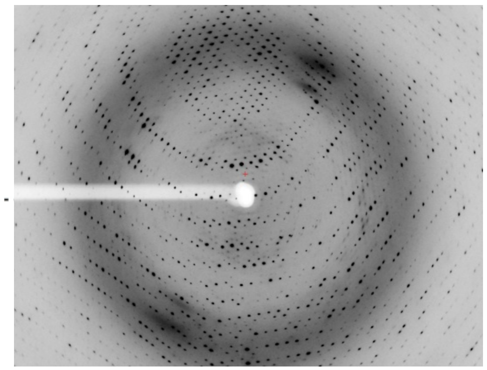
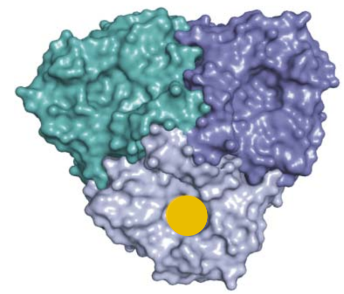
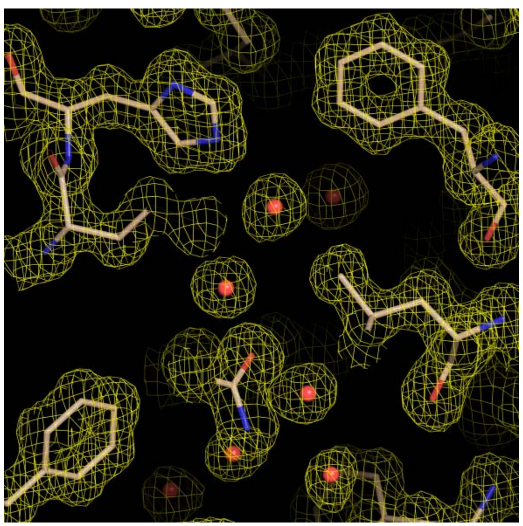
Structure of Proteins measured with Synchrotron Light

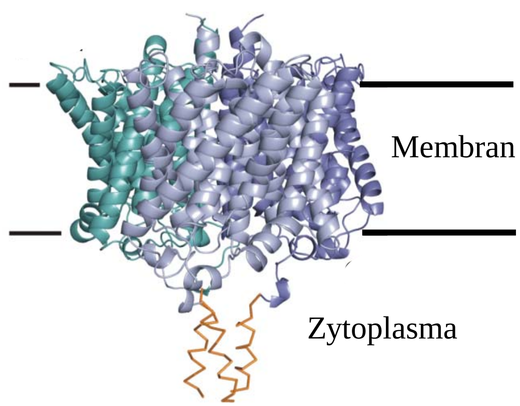


Diffraction pattern

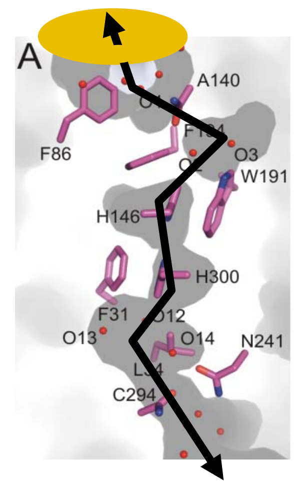








Rhesus Protein

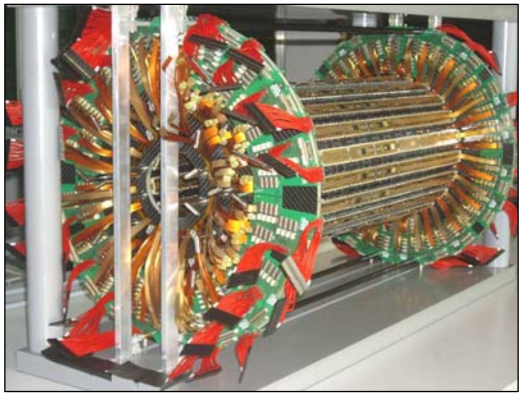


Amonium transport

Pixel Detector for CMS at 14TeV LHC (CERN)

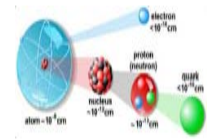
CMS experiment searches for Higgs, SUSY

Pixel Detector detects beauty, charm and tau-jets



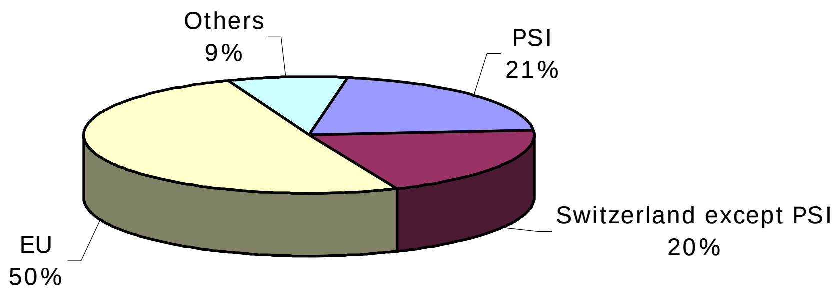
PILATUS 2M
pixel detector from
spin-off DECTRIS

PSI user laboratory key numbers 2008 inc LTP

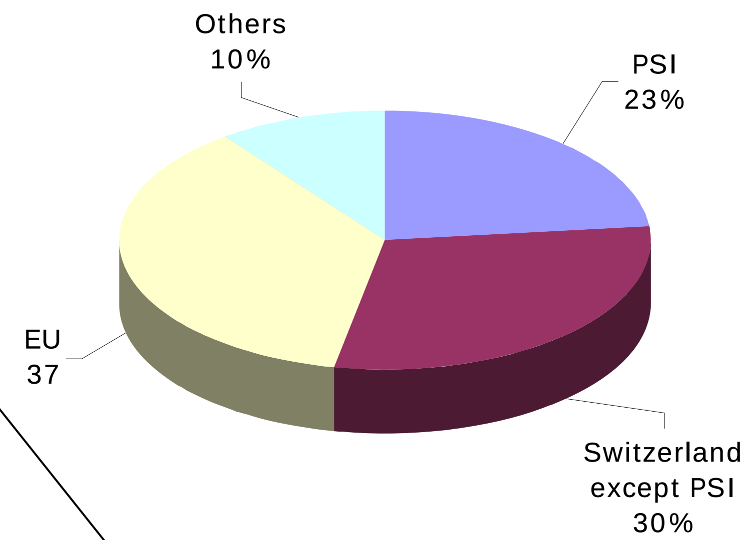


2008	SLS	SINQ	SμS	LTP	PSI total
Beamlines	14	13	6	7	40
Instrument Days	1657	1895	655	660	4867
Experiments	1036	446	168	8	1658
User Visits	2912	677	185	180	3954
Individual Users	1616	447	151	120	2334
New Proposals	656	275	156	1	1088

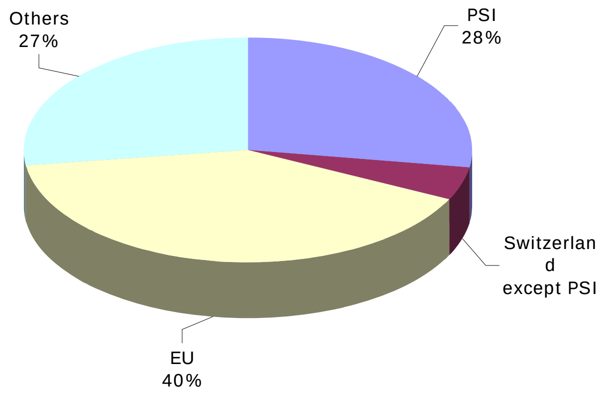
Geographic distribution SLS users
2008,



Geographic distribution of SINQ users 2008



Geographic distribution of SμS users



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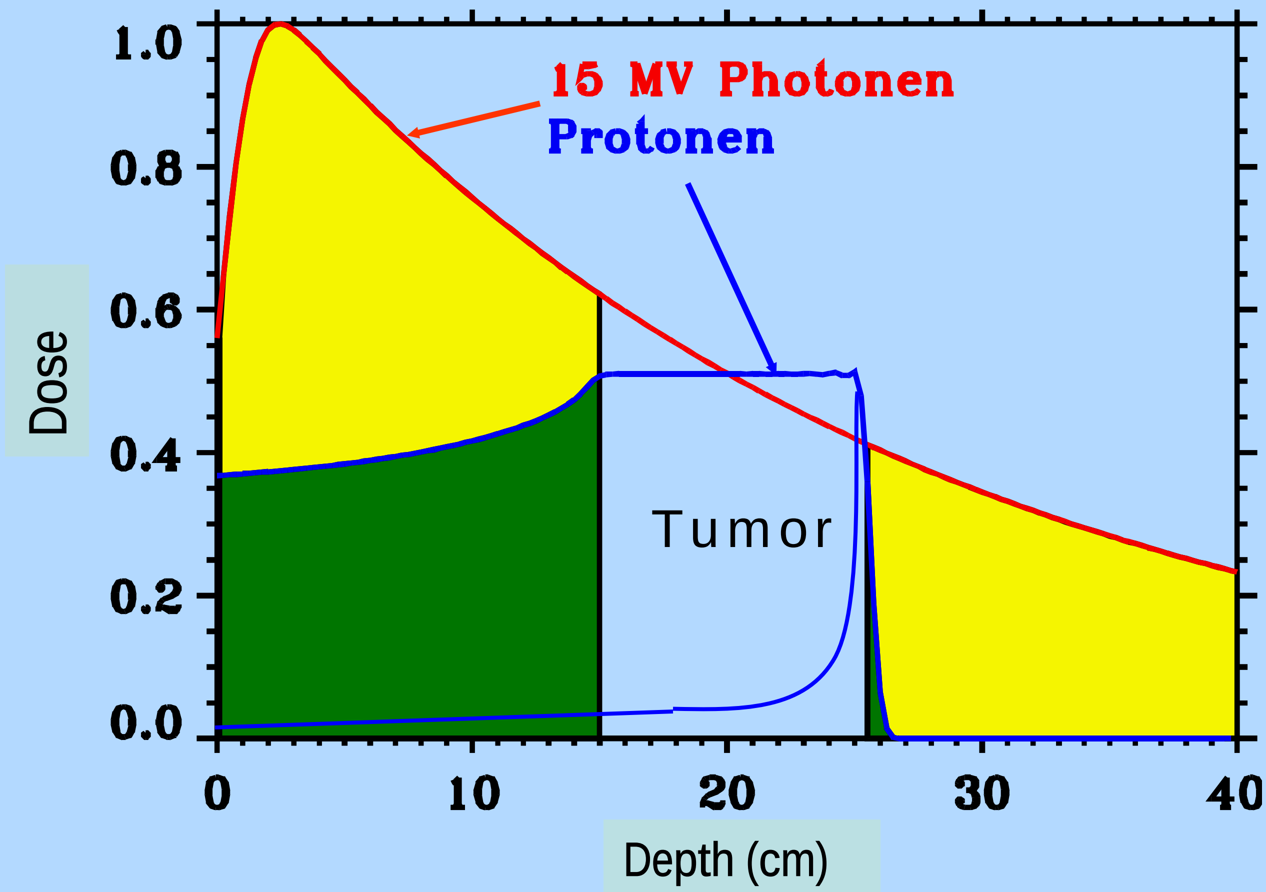
Comparison of Characteristics of Photons und Protons for Radiation Therapy



JM01 2009

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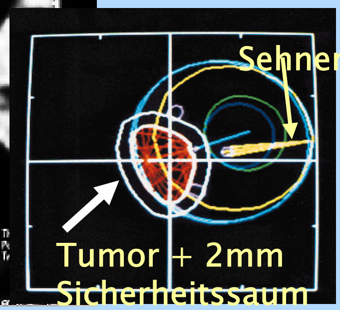
OPTIS: Proton Therapy for Eye Melanomas (since 1984)



Cooperation with Univ. Lausanne

A.G., 5 years old, retinoblastoma, left eye, 18 month after treatment

>5100 patients treated
>98 % tumor control

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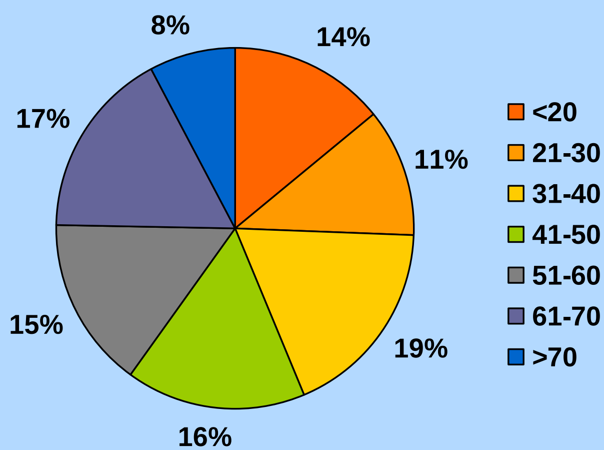
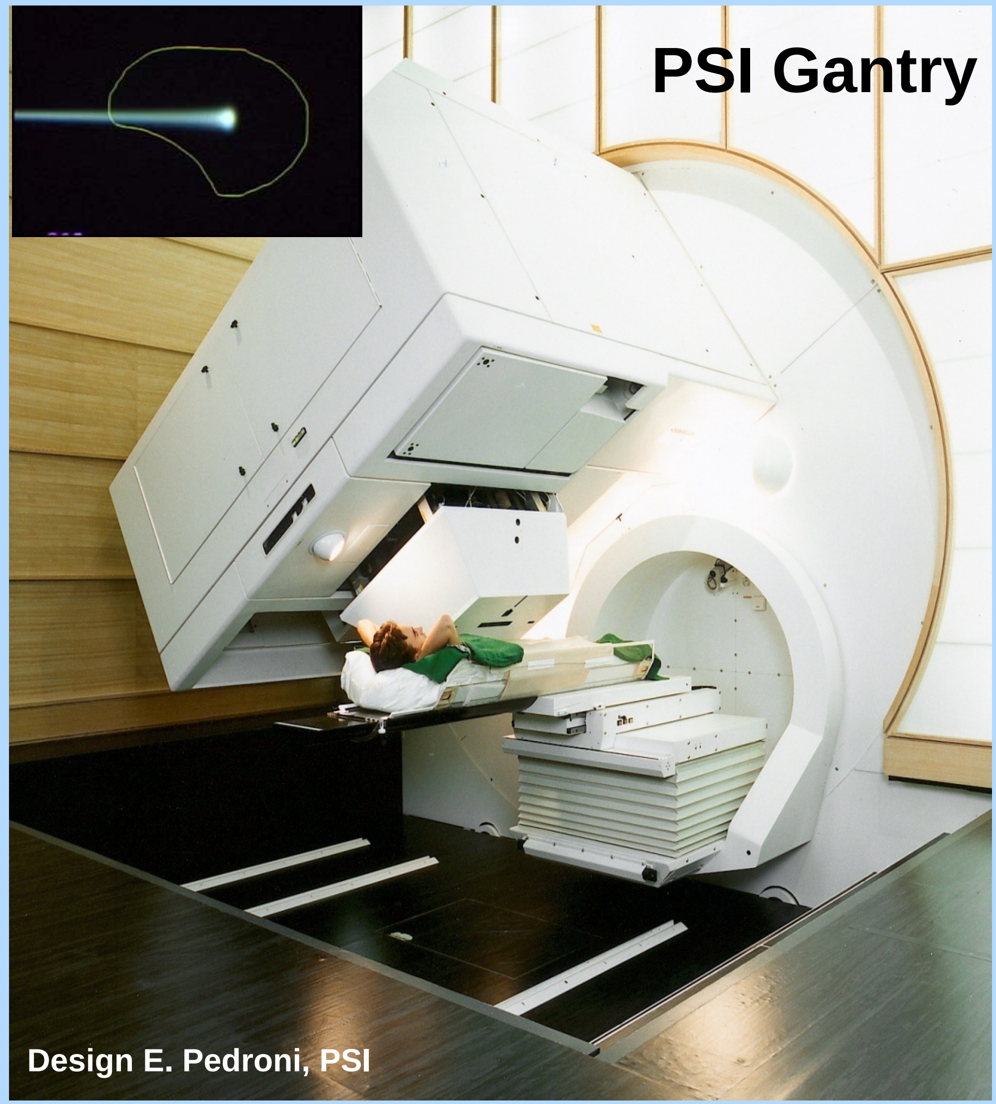
Proton Therapy

> 500 patients treated with deep-seated tumors

> 40 % of patients are below 40 years old

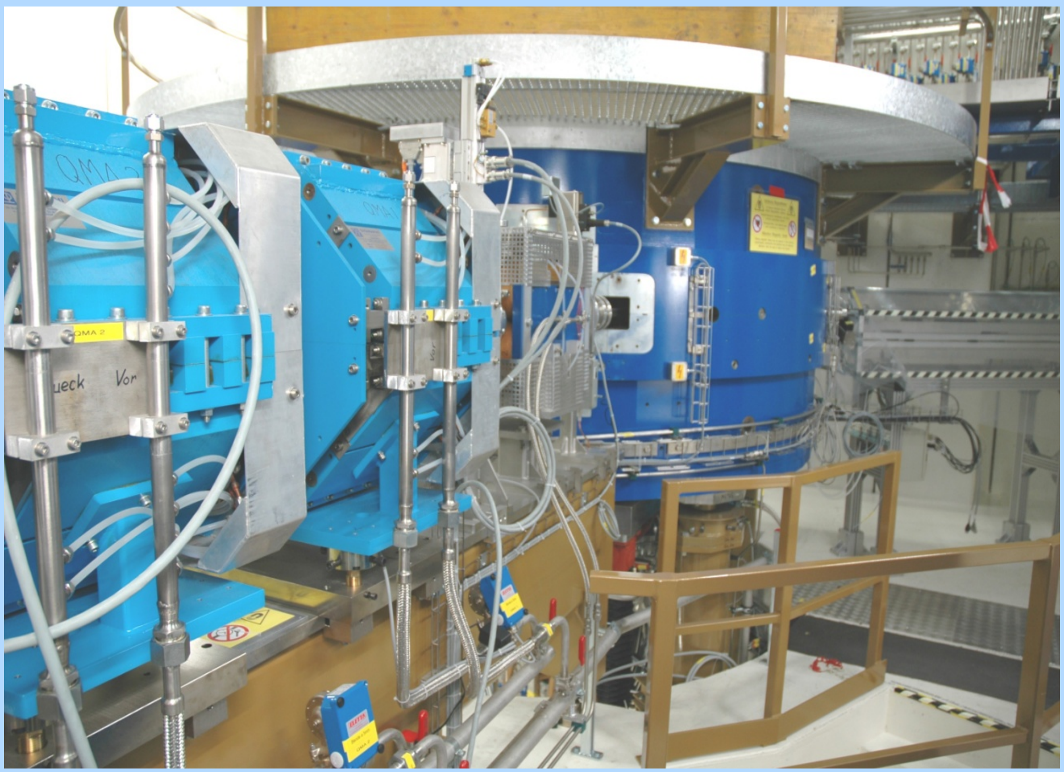




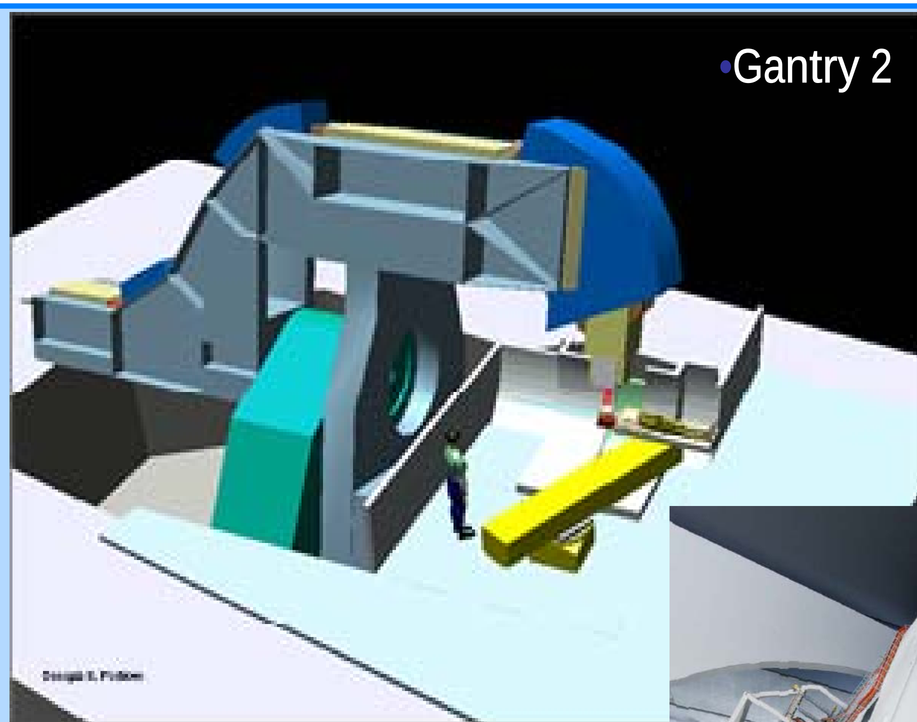
Treatment of pediatric tumors

Collaboration with
Children Hospital ZH
(for anesthesia)

COMET:
superconducting cyclotron
250 MeV proton beam
3.5 m diameter
90 t



Entwicklung einer neuen Gantry zur Behandlung beweglicher Tumoren



- Isozentrisch, Radius 3,6 m
- 2-dimensionales, schnelles Scanning (mit ‚re-painting‘)
- Intensitätsmodulation
- Permanent begehbarer Boden

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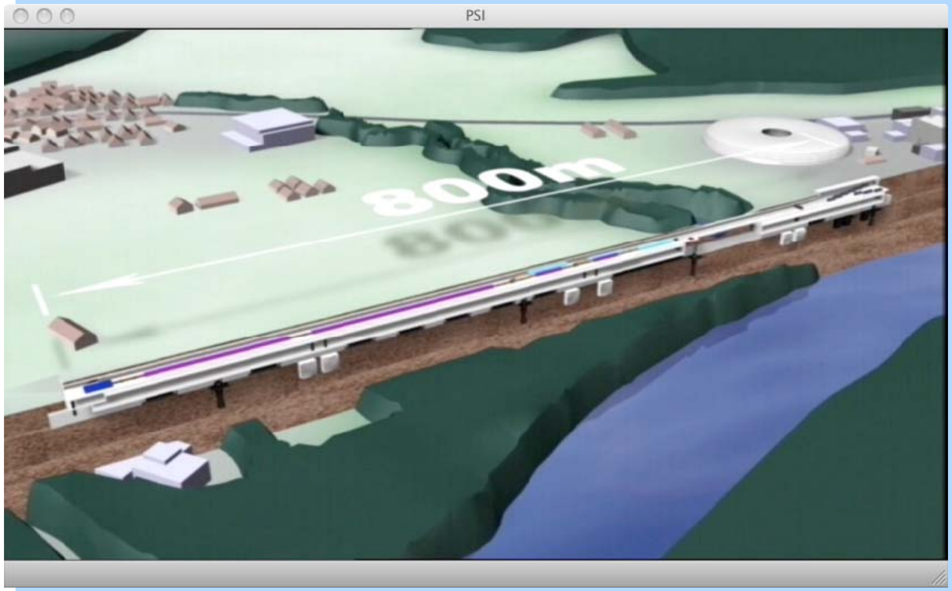
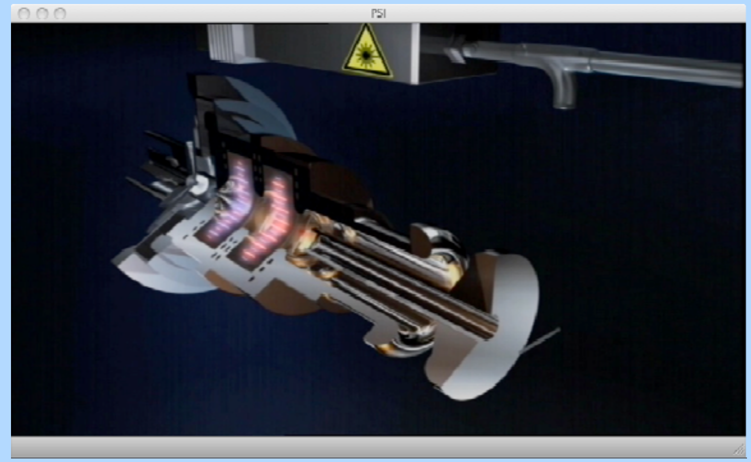
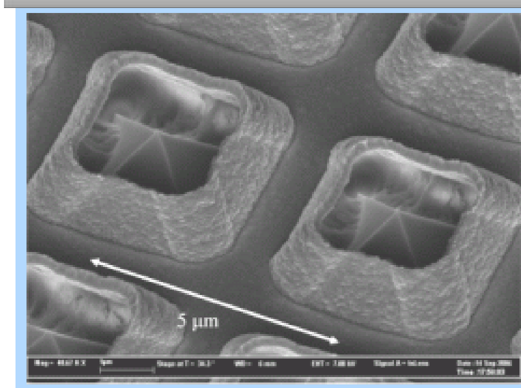
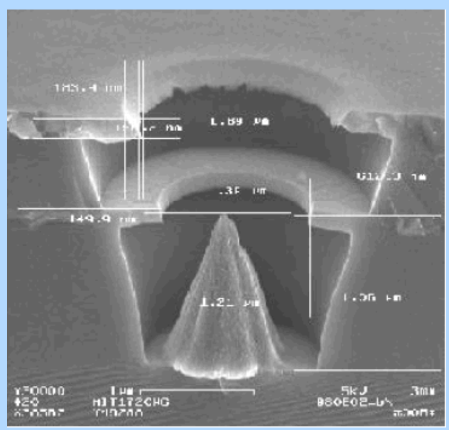
Strategic Initiative: SwissFEL (XFEL)









Low-Emittance Gun

High-Voltage Pulser

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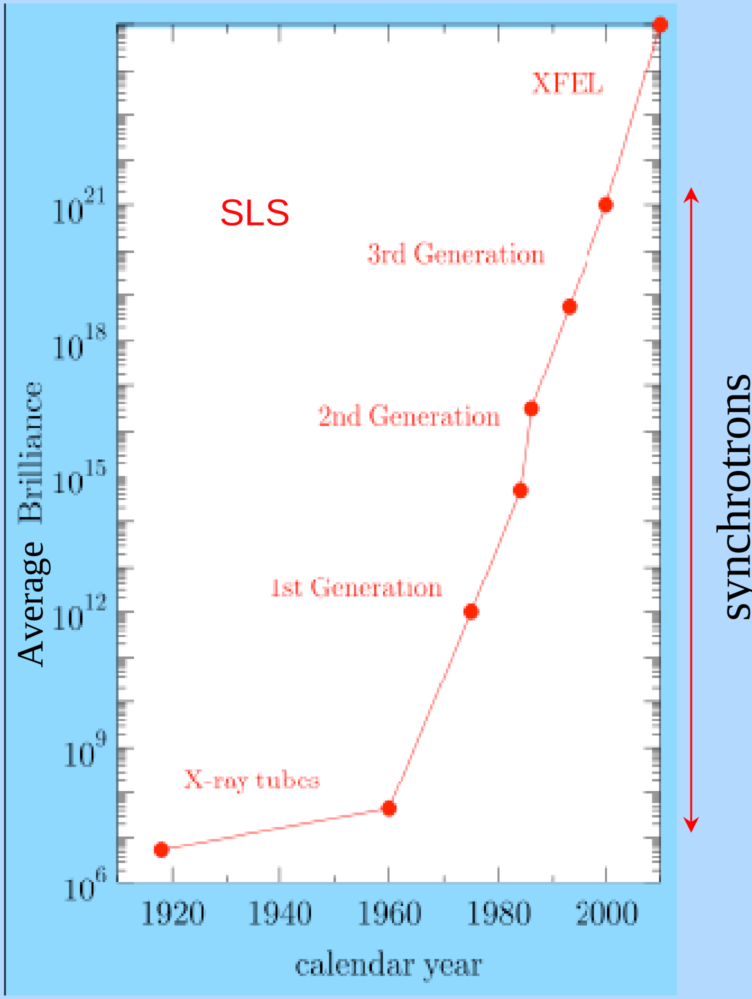
SwissFEL performance



- <u>Gain in brilliance</u>

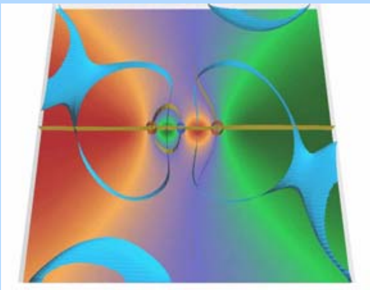
  10^4 (Mittel) - 10^{10} (Peak)

  comparable to EU, US, JP X-FEL

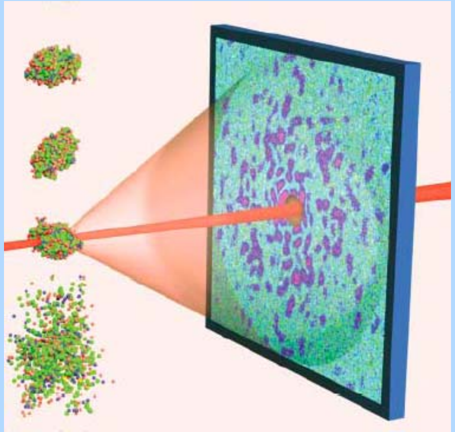
- Investments ≈ 250-280 MCHF

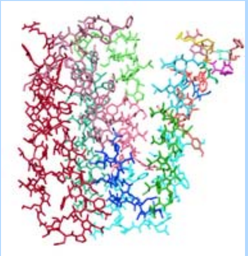
SwissFEL: what for?

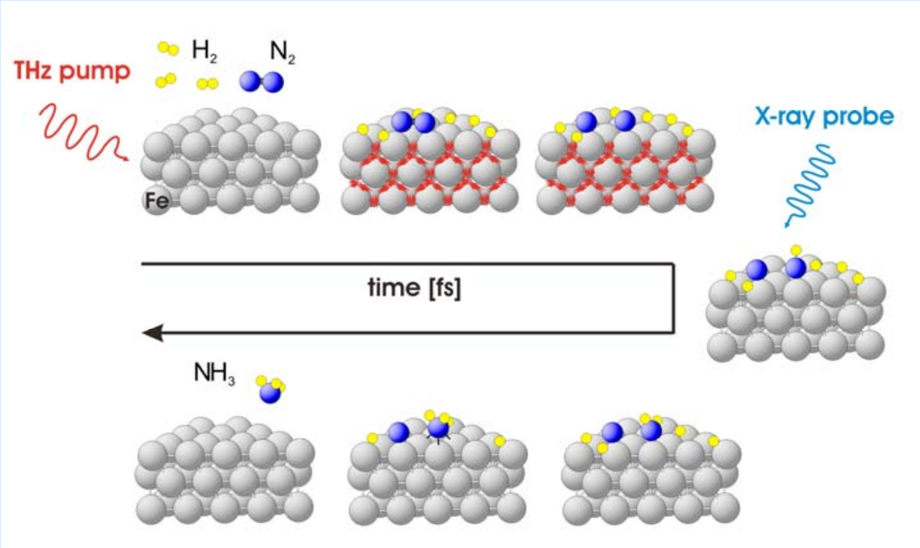
How fast and small can magnetic writting be ?



Catalytic reactions (time)







Determination of protein structures and interactions

http://fel.web.psi.ch

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Thank you for the attention

Welcoming Speech of Dr. U. Weidmann, Director, NPP-Beznau is not available

Laboratory for Thermal Hydraulics
Nuclear Energy and Safety
Severe Accident Research Group, SACRE

OECD/NEA Workshop on
"Implementation of Severe Accident Management (SAM) Measures"
ISAMM2009
Schloss Böttstein Switzerland
October 26-28, 2009

Severe Accident Management and OECD
An Introduction and Welcome

Salih Güntay

Working Group on the Analysis and Management of Accidents (WGAMA)

Severe Accident Management: The main purpose

as stated in WGRisk Report “PSA-Level 2 and SAMG” (NEA/CSNI/R(1997)11)

Provide a logical and structured guidance to identify the actions needed to stabilise the plant and return it to a controlled state following a multitude of potential accidents involving core damage

Severe Accident Management: key steps for development of SAM capability

(i) SAM development and assessment.

(ii) Assessment of plant vulnerabilities and capabilities.

(ii) Identification of guidance and strategies.

(iii) Investigation of information needs and instrumentation.

(iv) Assessment of SAM strategies/ measures.

Ref.: WGRisk Report “PSA-Level 2 and SAMG” (NEA/CSNI/R(1997)11)

Severe Accident Management Programmes: Key steps in Implementation

- Development of symptom based guidance/ procedures supported by technical assessment of strategies and plant specific capabilities.
- Plant organisation and decision making process which involves staff from the technical support centre interacting with the main control room staff.
- Validation of guidance and procedures to ensure their usability, technical accuracy, scope and function.
- Training is of special importance to overcome the degradation of human performance during stressful situations.
- Periodic exercises necessary to ensure maintenance of the capability and guidance usability.

Ref.: WGRisk Report “PSA-Level 2 and SAMG” (NEA/CSNI/R(1997)11)

OECD Role in Reactor Safety

- Organizes preparation of Status and State of Art Reports on Scientific Issues of common interest to Member States, in which:
    - o Different practices are displayed
    - o Issues in common for further development/Investigations are highlighted
- Organizes creation of new scientific data through OECD research projects as well as assessment of computer tools through ISPs
- Fosters cooperation among member states by organizing workshops/specialist meetings

Severe Accident Management: development to implementation

An account of OECD SAM Workshops on SAM

- Specialist meeting on Severe Accident Management Programme Development Rome, Italy, Sept. 23-25, 1991
- 1. Specialist Meeting on Instrumentation to Manage Severe Accidents, Köln, Germany, March 16-17, 1992
- 1.Specialist Meeting on Operator Aids for Severe Accident Management and Operator Training (SAMOA-1), Halden, Norway, June 8-10, 1993
- Specialist Meeting on Severe Accident Management Implementation, Niantic, Connecticut, USA, June 12-14, 1995
- Workshop on Hydrogen mitigation techniques, Winnipeg, Manitoba, Canada, May 13-15, 1996
- 2.Specialist Meeting on Operator Aids for Severe Accident Management and Operator Training (SAMOA-2), Lyon France, September 8-10, 1997

An account of OECD SAM Workshops on SAM

- Workshop on Iodine Aspects of Severe Accident Management, Vantaa, Finland, May 18-20, 1999
- Workshop on Severe Accident Management - Operator Training and Instrumentation Capabilities, Lyon France, April 12-14, 2001
- Workshop on Implementation of severe Accident Management Measures, Villigen Switzerland, September 10-13, 2001
- Workshop on Implementation of Severe Accident Management Measures, Schloss Böttstein Switzerland, October 26-28, 2009

Welcome to ISAMM2009, Schloss Böttstein

As the general chair of the workshop it is my pleasure to welcome you to ISAMM2009 on the behalf of OECD/NEA/CSNI/WGAMA+WRisk, organized in collaboration with PSI and co-sponsored by PSI and Swiss Nuclear Power Plants Beznau, Leibstadt, Gösgen and Mühleberg

ISAMM2009 realized following the proposal from US-NRC, as built in the Working Programme of WGAMA, CSNI endorsed the organization in 2008 as a joint activity of WGAMA and WGRisk.

ISAMM2009: Focuss

Objective of this workshop is to put balanced emphasis on both severe accident consequence analysis and risk assessment aspects such as

- The current status and insights related to SAM
- Issues of modeling SAM in PSA
- Code analysis supporting SAM development
- Decision-making tools, training, risk targets, and SAM entrance
- Design modifications for implementation of SAM
- Physical phenomena affecting SAM

ISAMM2009

- ❖ 43 papers on 6 main topics to be presented in 8 Sessions
- ❖ 2 Panel sessions
  - ➢ ISAMM2009 Highlights by Session chairs/co-chairs
  - ➢ Keynote speakers on:
    - Human and Organizational Aspects of SAM: their importance vs. technical issues by C. Huh (KINS, Korea)
    - Effectiveness of current SAMG implementation - How can consequence analyses be used to improve the effectiveness of SAM?, by Mark Leonard (Dycoda, US)

Thanks to:

- My Organizational Committee members
- All authors and participants
- My management at PSI and Swiss Nuclear Power Plants
- Ms Renate van Doesburg for local organization/administration

ISAMM2009: Administration

- All participants are kindly invited to Dinner

  Tuesday, October 27, 2009, 19:30 at the Kurhotel Restaurant, Zurzach

- Please contact Ms Renate van Doesburg (or me) for any assistance

Enjoy the workshop

Session 1

International Atomic Energy Agency

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

Presented by: Artur Lyubarskiy
A.Lyubarskiy@iaea.org

OECD/NEA Workshop on Implementation of Severe Accident Management Measures (ISAMMM-2009)

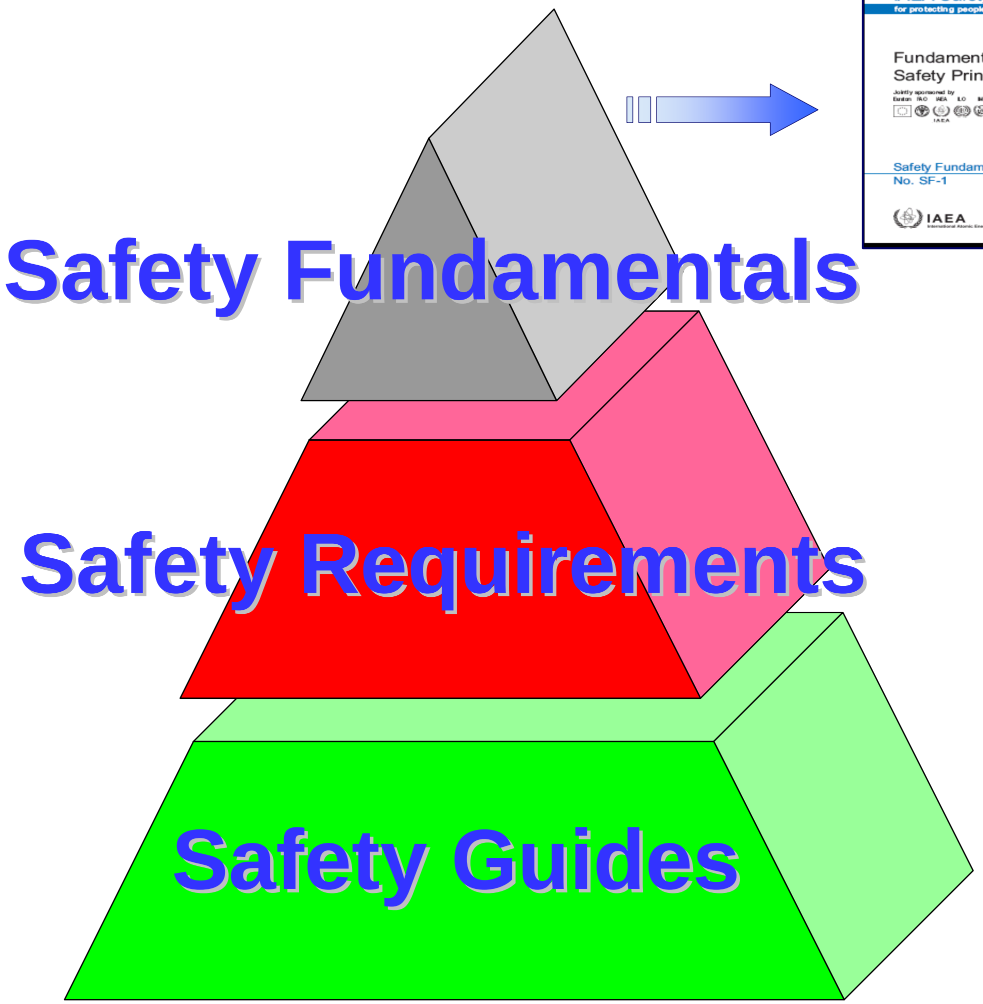
26-28 October, 2009

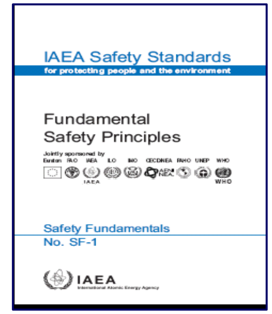
Bottstein, Switzerland

HIGHLIGHTS OF PRESENTATION

- Structure of IAEA Safety Standards
- New Safety Guide on Severe Accident Management
- New Safety Guides on PSA
- IAEA services
- Relevant recent activities

SAFETY STANDARDS HIERARCHY





Global Reference Point for the High Level of Nuclear Safety

The two Conventions

National Safety Regulations

National Regulatory Guides

STRUCTURE OF IAEA SAFETY STANDARDS

Publications categories:

Safety Fundamentals

Basic principles, objectives, and concepts of safety and protection in the development and application of atomic energy for peaceful purposes

- ***Safety Fundamentals are primary texts for a number of Safety Standards Series publications***

Safety Requirements

The requirements that must be met to ensure safety for particular activities or application areas

- ***Expressed as "shall" statements***
- ***Governed by the safety principles presented in the Safety Fundamentals***
- ***reflect an international consensus on what constitutes a high level of safety***

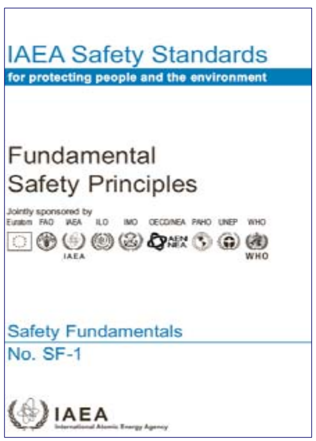
Safety Guides

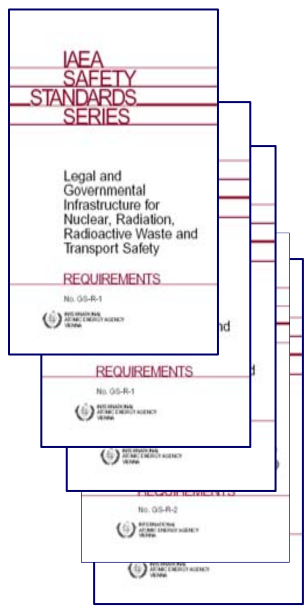
Actions, conditions or procedures for meeting safety requirements

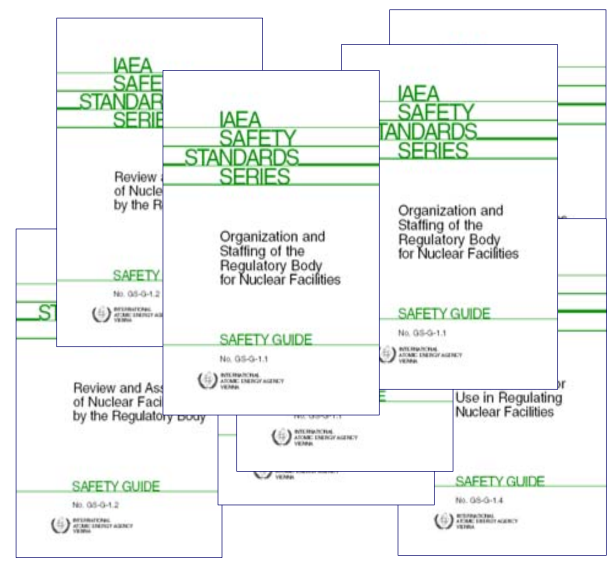
- ***Expressed as "should" statements***
- ***Implied that it is necessary to take the measures (recommended or equivalent alternative) to comply with the requirements***
- ***Reflect an international consensus on best practices***

New Structure of Safety Standards

SF	SR	SG
1	15	104







SAFETY STANDARDS RELATING TO NPPs

Safety Fundamentals (SF-1)

Safety Requirements
Safety of NPPs:
Design
No. NS-R-1

- **Safety Guide**
  Software for Computer Based Systems Important to Safety in NPPs
  No. NS-G-1.1
- **Safety Guide**
  Safety Assessment and Verification for NPPs
  No. NS-G-1.2
- . . .

Safety Requirements
Safety of NPPs:
Operation
No. NS-R-2

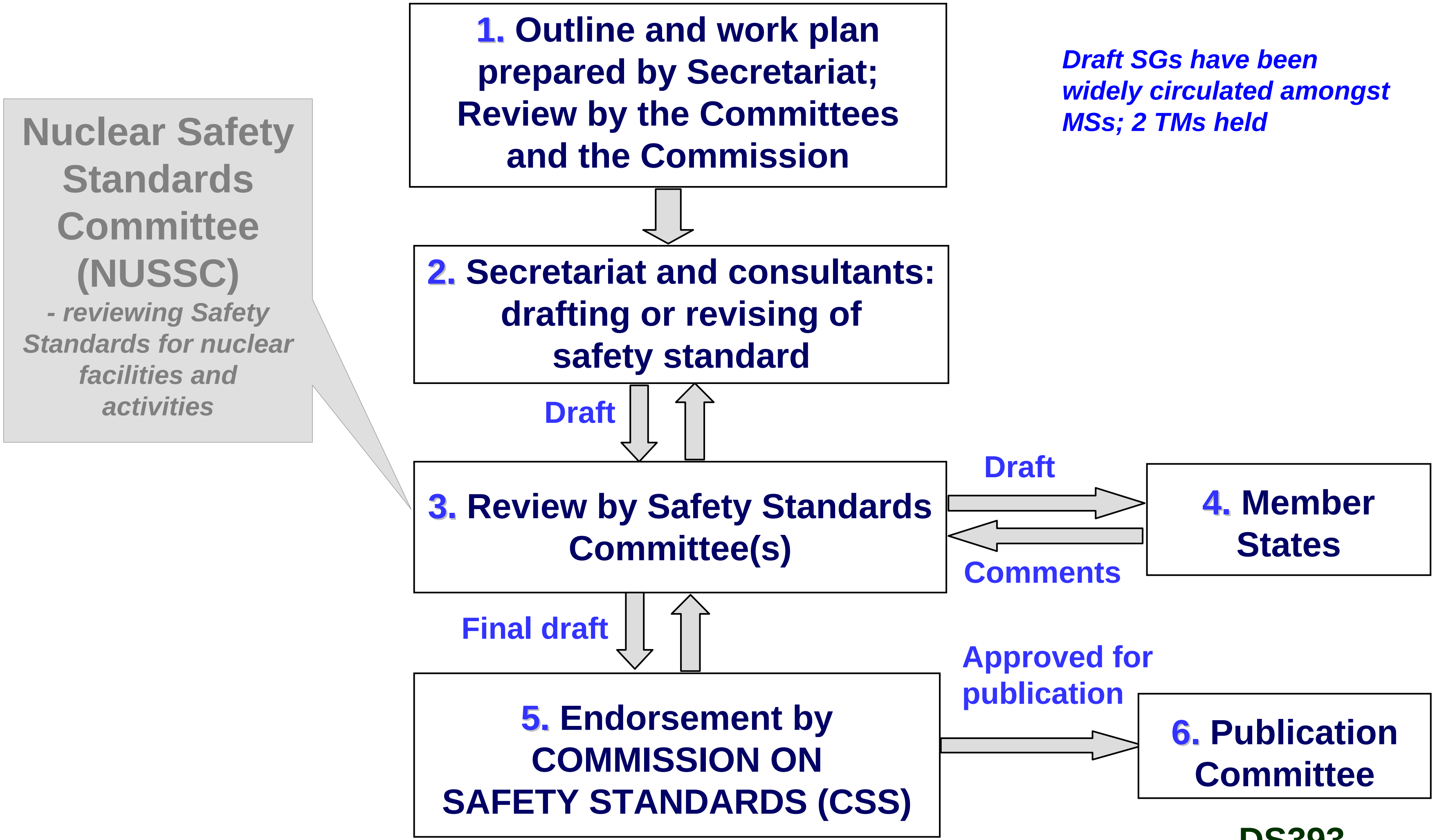
- **Safety Guide**
  Fire Safety in the Operation of NPPs
  No. NS-G-2.1
- **Safety Guide**
  Operational Limits and Conditions and Operating Procedures for NPPs No. NS-G-2.2
- . . .
- **Safety Guide**
  Periodic Safety Review of NPPs No. NS-G-2.10
- . . .

Safety Requirements
Safety Assessment for Facilities and Activities
(No. GS-R-4)

- **Safety Guide**
  Severe Accident Management Programmes for NPPs
  No. NS-G-2.15
- **Safety Guide**
  Development and Application of Level-1 PSA for NPPs
  (DS-394)
- **Safety Guide**
  Development and Application of Level-2 PSA for NPPs
  (DS-393)
- . . .

. . .

SAFETY STANDARDS DEVELOPMENT PROCESS



Draft SGs have been widely circulated amongst MSs; 2 TMs held

DS393
DS394

SAFETY GUIDE

Severe Accident Management Programmes for Nuclear Power Plants

(NS-G-2.15)

NS-G-2.15 - OBJECTIVE

- **To provide recommendations for the development and implementation of an accident management programme (including managing severe accidents)**
  - **Meeting the requirements that are established in NS-R-1, NS-R-2 and GS-R-4 for accident management**
  - **Intended primarily for use by operating organizations of nuclear power plants, utilities and their support organizations**
    - ✓May also be used by regulatory bodies to facilitate preparation of the relevant national regulatory requirements

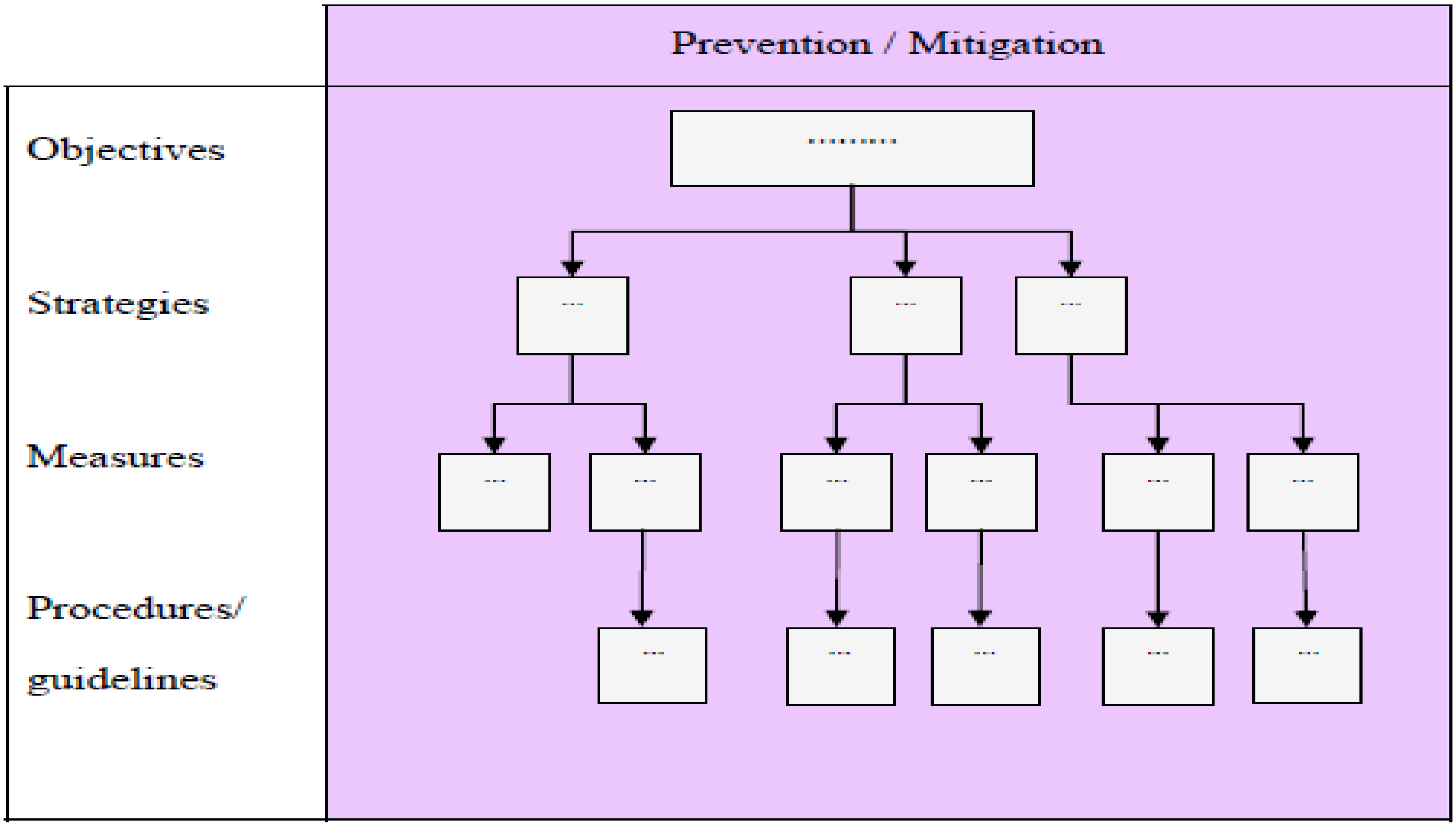
Basically two sections:

- **Concept of AM guidance (AMG)**
- **Detailed recommendations in process of development of AMG**
  - **Appendix:** Practical use of SAMG
  - **Annex:** Example of a categorization scheme for accident sequences
  - plus Annex on the use of the AMG

- **Major reference is IAEA Safety Series Report No. 32, ‘Implementation of AM Programs in NPPs**

2. CONCEPT OF AMP (1/2)

- ***Basic Concept:*** **Top Down**



2. CONCEPT OF AMP (2/2)

- **Develop AMG for ALL plants,** irrespective **of CDF/LERF**
- **Develop AMG for** all physically identifiable **challenge mechanisms for which guidance can be developed (limited probability considerations)**
  - **Identification of accident sequence *not needed***
- **Add/upgrade equipment for** *meaningful* **AMG**
  - **I.e., a program that indeed *reduces risk***
- **Determine type of AMG:** *overall* **or** *detailed* **guidance in an** iterative **process (via drills)**
  - **Overall: may have *too much latitude***
  - **Detailed: may be *too prescriptive***
- **Define roles and responsibilities, compatible with AMG**
- **Define adequate** transition **between preventive and mitigative domain, including transition of responsibilities / decision making authority**

3. DEVELOPMENT OF AN AMP

- **Contains detailed recommendations for all steps in developing the AMG**
- **Total: 113 recommendations; apart from items in the 'concept'**
  - **Analysis (a.o. plant vulnerabilities, plant capabilities)**
  - **What should be in the AMG**
    - ✓ e.g. initiation of actions, resources, cautions, throttling, monitoring response, termination
  - **Priorities between AM strategies**
  - **Consideration of positive and negative consequences of actions**
  - **Use of I&C and how to deal with missing information**
  - **Formation of AMG development team**

USEFULNESS OF SG NS-G-2.15

- **‘Severe accidents’ is a complex issue**
  - **Physics only *partly understood***
  - **Most plants *not designed* to such accidents**
    - ✓ but many plants have features that can be used to mitigate such accidents
  - **Plant status *partly unknown* (I&C often outside quality range)**
  - **Actions can have both** positive ***and*** negative **consequences**
    - ✓ **E.g. spraying the containment: reduces pressure, but de-inerts containment atmosphere**
  - **High uncertainty**
- **Guidance on meeting the requirements that are established in NS-R-1, NS-R-2 and GS-R-4 for severe accidents management is extremely needed**
  - **Intended primarily for use by operating organizations of nuclear power plants, utilities and their support organizations**
    - ✓ May also be used by regulatory bodies to facilitate preparation of the relevant national regulatory requirements

SAFETY GUIDES ON PSA (DS393 & DS394)

DS394- Development and Application of Level 1 PSA

DS393- Development and Application of Level 2 PSA

SAFETY GUIDES ON PSA (DS393 & DS394)

- **Objective:** to provide recommendations for performing or managing a PSA project for an NPP and using it to support the safe plant design and operation

  - 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 PSA and its various applications

PSA SCOPE COVERED IN SAFETY GUIDES

- **All plant operational conditions, i.e.:**
  - **(a) full power**
  - **(b) low power and shutdown**

- **All potential initiating events and hazards, i.e.:**
  - **(a) internal initiating events caused by random component failures and human errors;**
  - **(b) internal hazards (e.g. internal fires and floods, turbine missiles, etc.);**
  - **(c) external hazards, both natural (e.g. earthquake, high winds, external floods, etc.) and man-made (e.g. airplane crash, accidents at nearby industrial facilities, etc.)**

- **Radioactivity source: reactor core**

DS394: SAFETY GUIDE ON LEVEL 1 PSA

- **1.** INTRODUCTION
- **2.** GENERAL CONSIDERATIONS RELATING TO THE PERFORMANCE AND USE OF PSA
- **3.** PSA PROJECT MANAGEMENT AND ORGANIZATION
- **4.** FAMILIARIZATION WITH THE PLANT
- **5.** LEVEL 1 PSA FOR INTERNAL INITIATING EVENTS FOR FULL POWER CONDITIONS
- **6.** GENERAL METHODOLOGY FOR INTERNAL AND EXTERNAL HAZARDS PSA
- **7.** SPECIFICS OF INTERNAL HAZARDS PSA
- **8.** SPECIFICS OF EXTERNAL HAZARDS PSA
- **9.** LEVEL-1 PSA FOR LOW POWER AND SHUTDOWN MODES
- **10.** USE AND APPLICATIONS OF THE PSA

DS393: SAFETY GUIDE ON LEVEL 2 PSA (1/4)

1. Introduction

- **Discussion on**
  - ✓ General PSA classification
  - ✓ Connection of the Safety Guide to other Safety Standards publications
  - ✓ Scope, and objectives

2. PSA project management and organization

- **Specific recommendations relating to the management and organization of a Level-2 PSA project**

3. Familiarization with the plant and identification of design aspects important to severe accidents

- **Specific recommendations dealing with acquisition of information important to severe accident analysis**

4. Interface with Level-1 PSA:

- **Grouping of sequences:**
  - ✓ Addresses the analysis tasks covering the interface between Level-1 and Level-2 PSAs
  - ✓ Definition of plant damage states for all initiating events and hazards, and plant operational states

DS393: SAFETY GUIDE ON LEVEL 2 PSA (2/4)

5. Accident progression and containment analysis

- **Key recommendations regarding**
  - ✓ 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
  - ✓ Interpretation of containment event tree quantification results

6. Source terms for severe accidents

- **Key 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

DS393: SAFETY GUIDE ON LEVEL 2 PSA (3/4)

7. Documentation of the analysis:

- **Presentation and interpretation of results**
  - ✓ Discusses specific issues relating to the documentation of a Level-2 PSA

8. Specific needs and recommendations for applications of Level-2 PSA

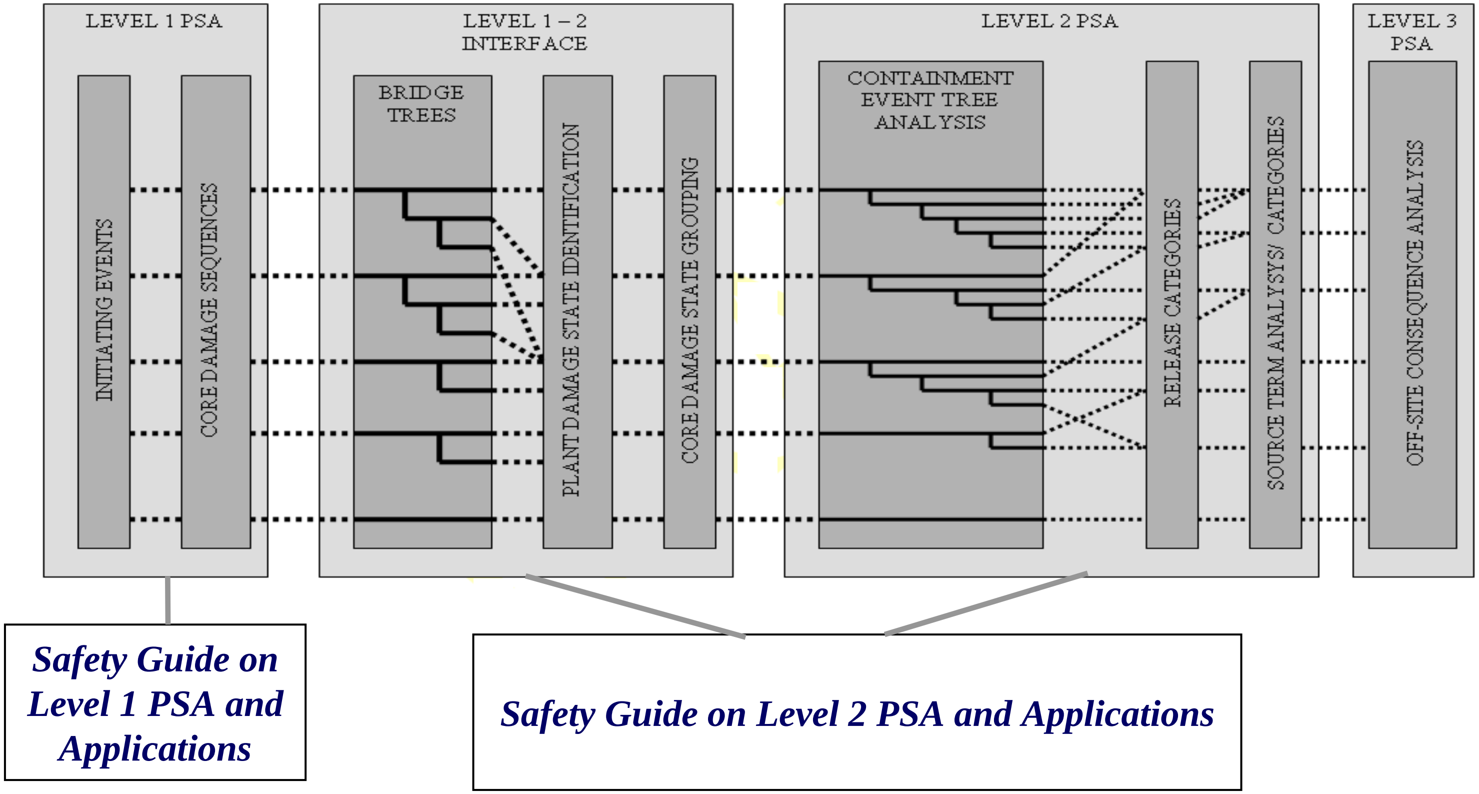
- **Recommendations for a number of Level-2 PSA applications**
  - ✓ Comparison with numerical criteria
  - ✓ Design evaluation
  - ✓ Severe accident management
  - ✓ Emergency planning
  - ✓ Off-site consequences
  - ✓ Prioritisation of research
  - ✓ Other PSA applications

- **Three annexes:**
  - **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**

DS393: APPLICATION OF LEVEL-2 PSA FOR SAM

- **The SG DS303 provides specific recommendations on the use of Level 2 PSA results for**
  - **The evaluation of the measures and actions that can be carried out to mitigate the effects of a severe accident**
    - ✓ To determine the effectiveness of the severe accident management measures that are described in the SAM guidelines or procedures
    - ✓ 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 the importance of consideration of interdependencies**
        - E.g. depressurization of the primary circuit may prevent high pressure melt ejection but might increase the probability of an in-vessel steam explosion
  - **The updates of the Level 2 PSA and updates of the SAMGs guidelines should be performed in an iterative manner to facilitate the progressive optimization of the severe accident management guidelines**
  - **Recommendations correspond to those, provided in NS-G-2.15**

INTERFACE BETWEEN SAFETY GUIDES ON PSA



➔ ***Level-3 PSA will be covered later***

PLANNED TECDOC ON IRIDM

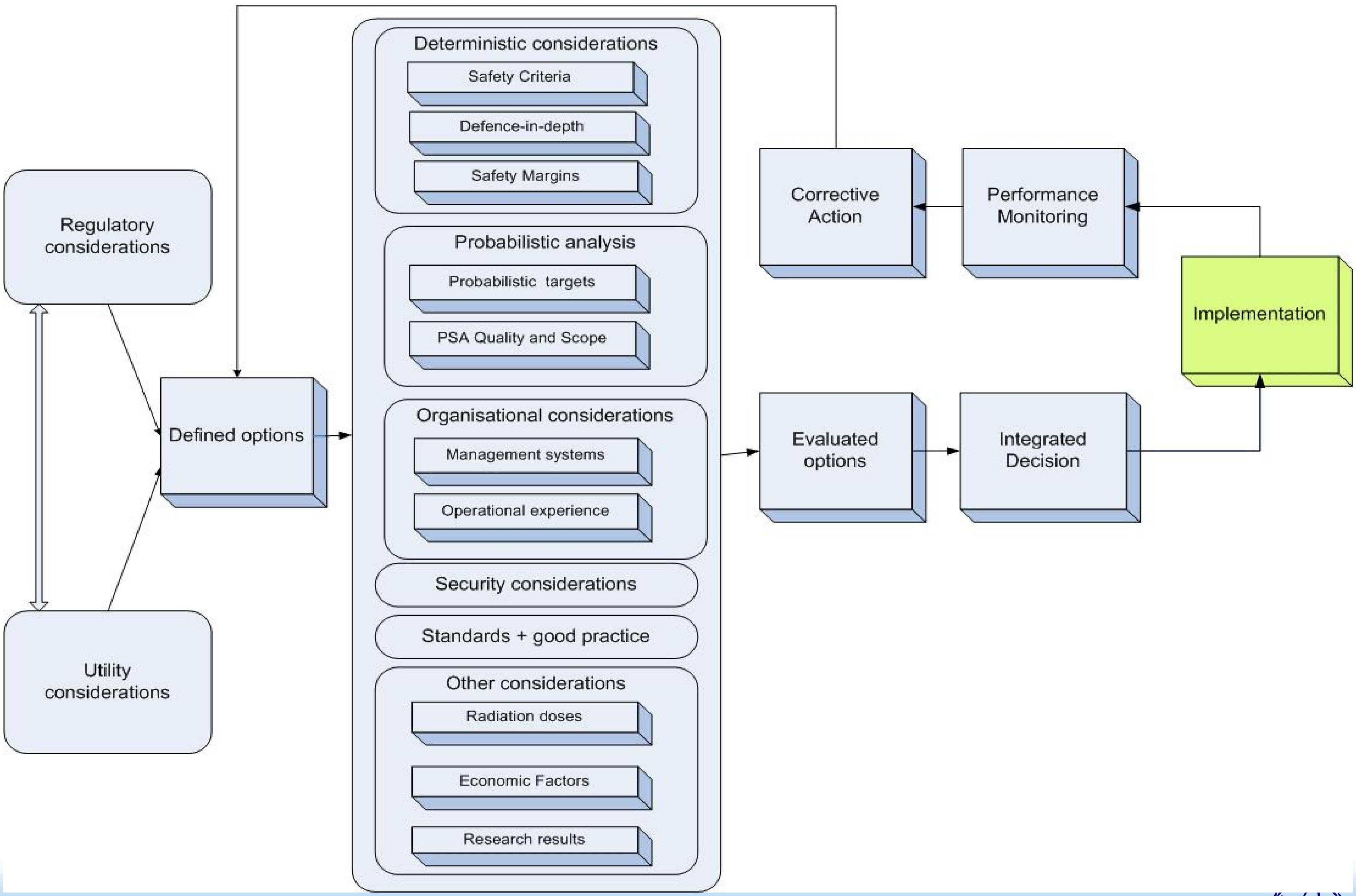
- **First CS has held in October 2009**

- **OBJECTIVE: IRIDM provides principles and suggests approaches to integrate the results of deterministic and probabilistic safety analyses as well as other important aspects to make sound, optimum, and safe decisions**

- **HIGHLIGHTS:**
  - **Principles on IRIDM**
  - **An overview of the complementary blend of deterministic and probabilistic approaches**
  - **Organizational interrelationships in IRIDM and overview of the necessary support for IRIDM**
  - **Deterministic and risk aspects of IRIDM**
  - **How to use deterministic and risk aspects to arrive at sound decisions**
  - **Documentation and presentation of IRIDM results**

- **Follows main principles listed in Draft INSAG-24 “A FRAMEWORK FOR INTEGRATED RISK-INFORMED DECISION MAKING PROCESS”**

IRIDM FRAMEWORK (Draft INSAG-24)



DIVISION OF NUCLEAR INSTALLATION SAFETY

SAFETY SERVICES

DSRS	**Design Safety Review Service**
SSRS	**Seismic Safety Review Service**
FSRS	**Fire Safety Review Service**
AMAT	**Ageing Management Advisory Team**
SWSRS	**Software Safety Review Service**

These Services, initiated in 1989, provide advice on selected engineering safety aspects of nuclear power plants in siting, design, construction and operation.

INSARR

Integrated Safety Assessment of Research Reactors

INSARR missions are an IAEA safety service offered to assist Member States in ensuring and enhancing the operational safety of research reactors.

IRS

Incident Reporting System

The IRS is a global network for the collection, analysis and dissemination of information on safety relevant events that have occured at NPPs.

IRSRR

Incident Reporting System for Research Reactors

The IRSRR is a system designed to collect, analyse and disseminate information on unusual events that have occured at research reactors.

OSART

Operational Safety Review Team

The purpose of the OSART programme, established in 1982, is to assist Member States in enhancing the operational safety of nuclear power plants by promoting performance based assessment processes and providing recommendations and assistance derived from these assessments.

PROSPER

Peer Review of Operational Safety Performance Experience

An IAEA operational safety service (derived from the former ASSET service) to peer review self-assessments by NPPs of their operational safety performance and its trends based on operating experience.

SCEP

Safety Culture Enhancement Programme

A service intended to support senior utility managers in enhancing the management of safety and safety culture. It provides training to increase the understanding of safety culture issues, to perform a self-assessment and to develop improvement initiatives.

IRRT

International Regulatory Review Team

Launched in 1989, the IRRT programme provides advice and assistance to Member States to strengthen and enhance the effectiveness of their nuclear safety regulatory body.

IPSART

International Probabilistic Safety Assessment Review Team

IPERS (now called IPSART) was established in 1988 to make international expertise available for reviewing probabilistic safety assessments (PSAs).

RAMP

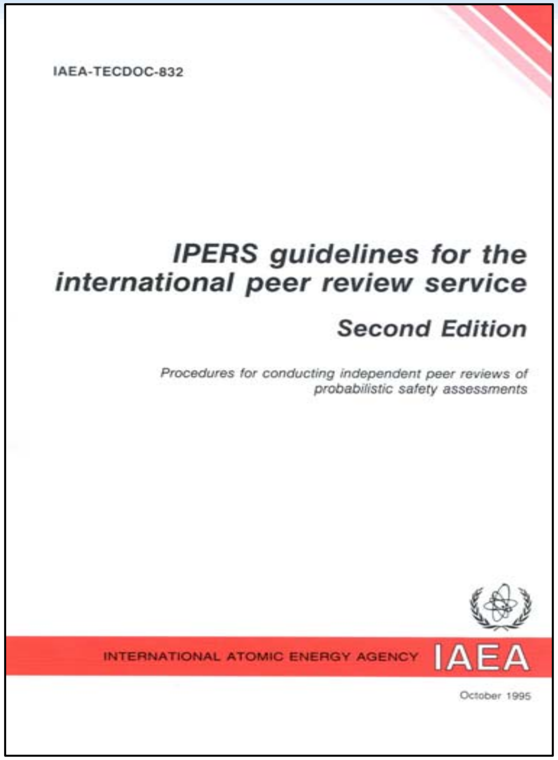
Review of Accident Management Programmes

An IAEA service to assist Member States in the preparation, development and implementation of accident management programmes for NPPs.

INES

International Nuclear Event Scale Information Service

INES is a scale aimed at putting into perspective incidents and accidents in NPPs and other nuclear installations by explaining in simple terms their significance and relative importance to the public.


IAEA-TECDOC-832

IPERS guidelines for the international peer review service

Second Edition

Procedures for conducting independent peer reviews of probabilistic safety assessments

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

October 1995

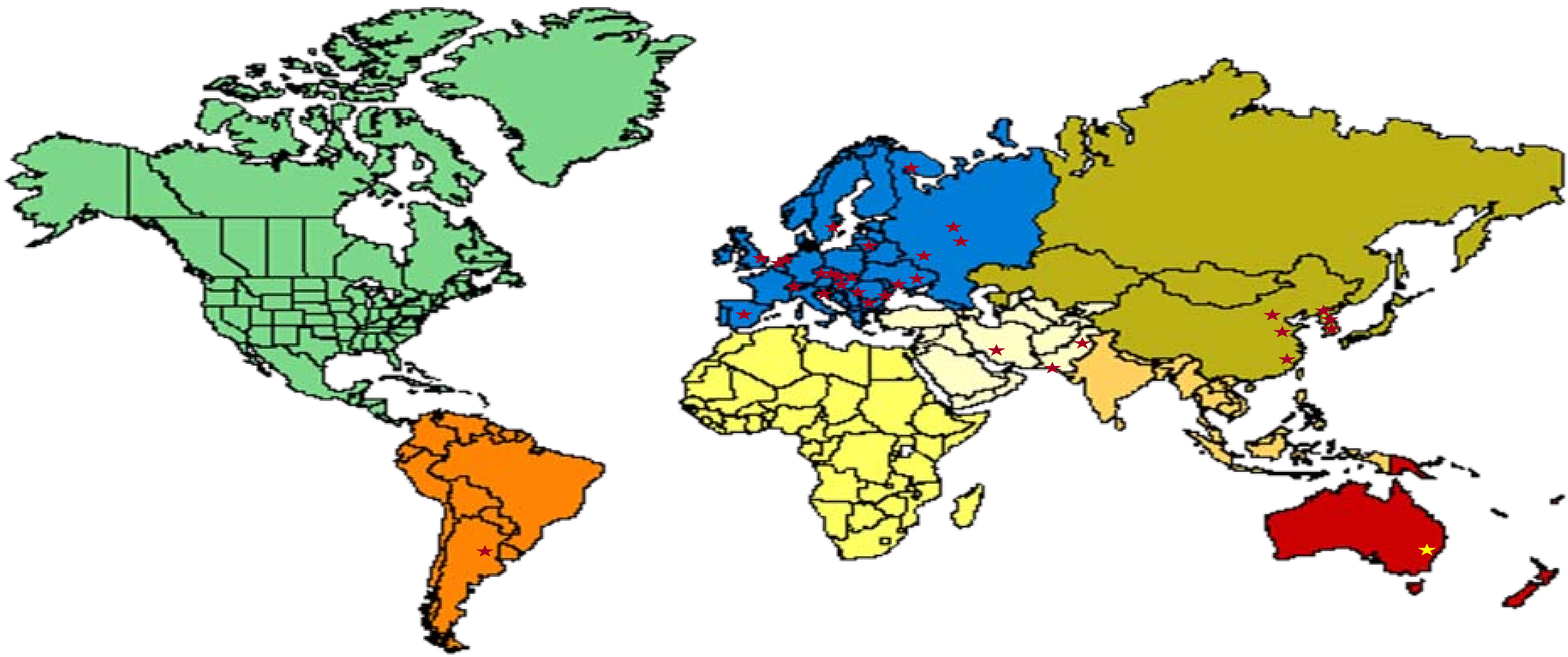
IPSART SERVICE

I nternational
P robabilistic
S afety
A ssessment
R eview
T eam

➔ ***Established in 1988***

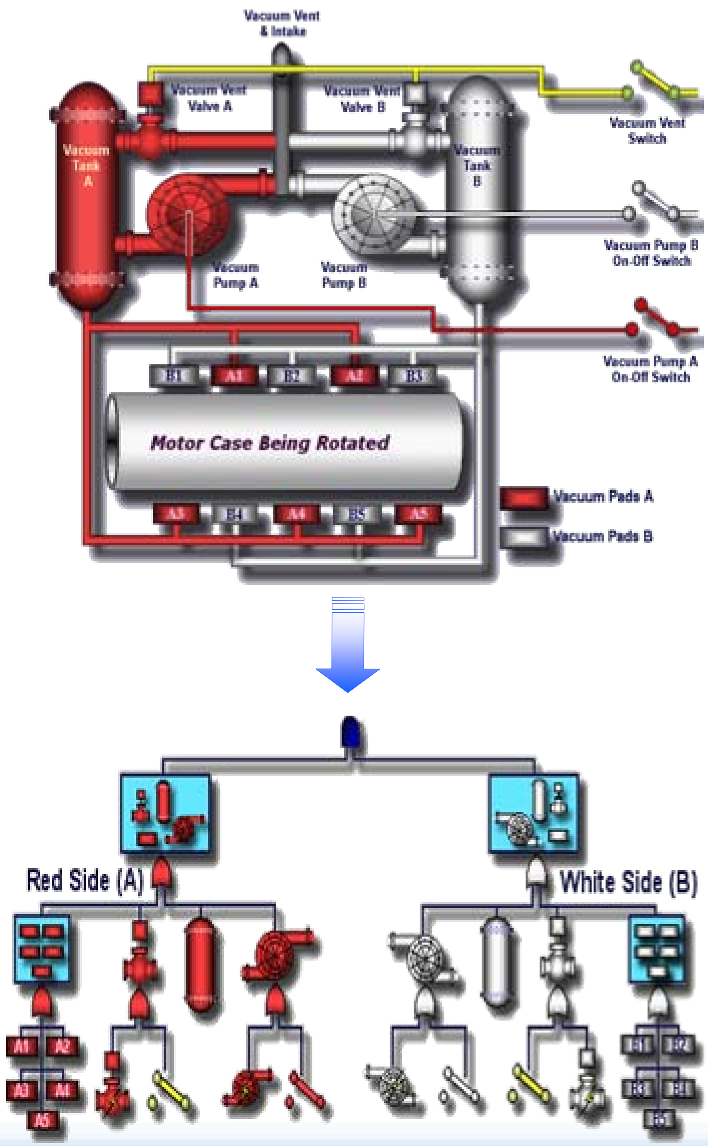
➔ ***Conducted in accordance with dedicated Guidelines***

SUMMARY OF IPSART MISSIONS CONDUCTED



- **More than 50 IPSART mission have been conducted all around the world**
- **In average, 3-5 IPSART missions are conducted every year**
- **Installations Reviewed**
  - Mostly NPPs
  - Research reactors
  - Open to other types

REVIEW APPROACH



IPSART

Surface check of methodological aspects, completeness, consistency, coherence, etc.	Detailed spot checks

- PSA objectives, purpose, scope, project plan, work and team organization
- PSA models, e.g. accident sequences and system analysis
- Database, methodology, parameters, human reliability analysis
- PSA documentation, information, results, applications, LPSA aspects

QA, internal review

IPSART MISSION REPORT

- Describes the review performed, the review findings, the technical aspects of the PSA study, strengths, and limitations
- Provides suggestions and recommendations for improvement of the PSA quality and its sound use for enhancing plant safety and risk management applications

BENEFITS

- IPSART service helps to achieve high quality of PSA and therefore assists in further enhancing the nuclear safety
  - PSA results are used widely in various risk-informed decisions by plants and regulatory authorities
- IPSART service proliferates advanced methodology and knowledge in nuclear safety assessment

RAMP SERVICE (1/2)

R e v i e w of
A c c i d e n t
M a n a g a m e n t
P r o g r a m m e

- At particular plant, on request by Member State
- Review by team of usually 4 experts, plus IAEA lead
- Duration usually one week
  - Study of documents
  - Interviews with plant staff, regulator
- Location:
  - on-site, preparation before: off-site
- At end: discussion plus detailed report with assessment and recommendations

RAMP SERVICE (2/2)

- **IAEA has prepared 'User Manual' for RAMP service: "GUIDELINES FOR THE REVIEW OFACCIDENT MANAGEMENT PROGRAMMES"**
- **Contains detailed questionnaire, with 90 questions:**
  - **Topics of questions**
    - ✓ 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
- **Separate parts for analysis and AM guidelines**

RELEAVENT ACTIVITIES

RAMP Services

- Krsko NPP, Slovenia, 2001
- Chashma NPP, Pakistan, 2005 (pre-RAMP)
- Ignalina NPP, Lithuania, 2007
- Cernavoda NPP, 2007 (Pre-RAMP)
  - Pre-Review of Accident Management Programme
- KANUPP , Karachi, Pakistan, 10.2008 (pre-RAMP)
  - Introduction of severe accident analysis and AMP for Pressurized Heavy Water Reactors

Expert Missions (2007 - 2009)

- Expert mission on severe accident analysis and accident management programme, Beijing, China 07.2007
  - Review the severe accident analysis for Chinese PWRs and develop plan for severe accident management programme (AMP)
- Expert mission to review severe accident analysis and to assist in developing severe accident management strategy, Beijing, China 07, 2008
  - Review typical severe accident analysis for Chinese PWRs
- KANUPP, Karachi, Pakistan (held in Vienna, 09.2009)
  - Review SAMG documents prepared by KANUPP

Workshops and Technical Meetings

- Regional workshop on severe accidents analysis and accident management for NPPs, Kiev, Ukraine, 06.2007
  - Sharing views and exchanging experiences on the severe accident analysis and accident management in participating countries
- TM on severe accident, accident management and PSA application of PHWRs (jointly with AECL/CNSC) Canada, 2008

CONCLUDING REMARKS

- **The NEW IAEA's Safety Standards publications will provide a common platform for performance and application of SAMPs, safety assessment, PSA, and IRIDM**
- **SG on SAMP contains extensive guidance for the setting up of an AM Program, with focus on severe accidents**
  - **E.g., type of guidance, responsibilities, setting priorities, use of I&C, dealing with incomplete information and possible negative consequences of actions**
  - **useful for new AM programs and for review of existing AM programs**
- **SGs on PSA will promote a consistent development, application, and review of PSA studies, as well as the use of PSA results and insights in the IRIDM process**
- **RAMP Service**
  - **High quality review of AM Program at individual NPPs**
    - ✓ benefit from 'fresh look', and in-depth discussions during about one week of mission
- **IPSART Service**
  - **Helps to achieve high quality of PSA**
    - ✓ proliferates advanced methodology and knowledge in nuclear safety assessment

- **All IAEA publications available at:**

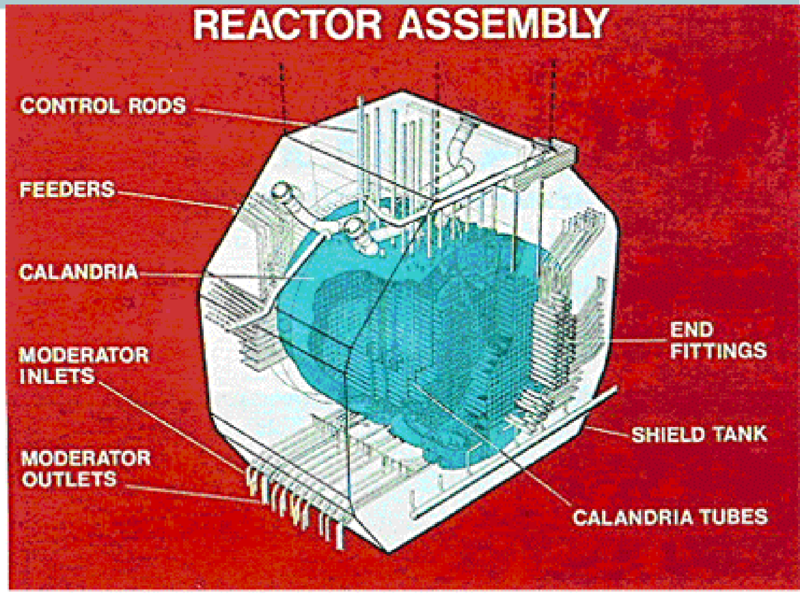
http://www-pub.iaea.org/MTCD/publications/publications.asp

Technical Challenges in Applying SAMG Methodology to Operating CANDU Plants

Keith Dinnie - AMEC NSS - Toronto, Canada

OECD/NEA Workshop on Implementation of Severe Accident Management Measures

(ISAMM-2009)



Contents

Key features of CANDU reactor design

Entry Conditions

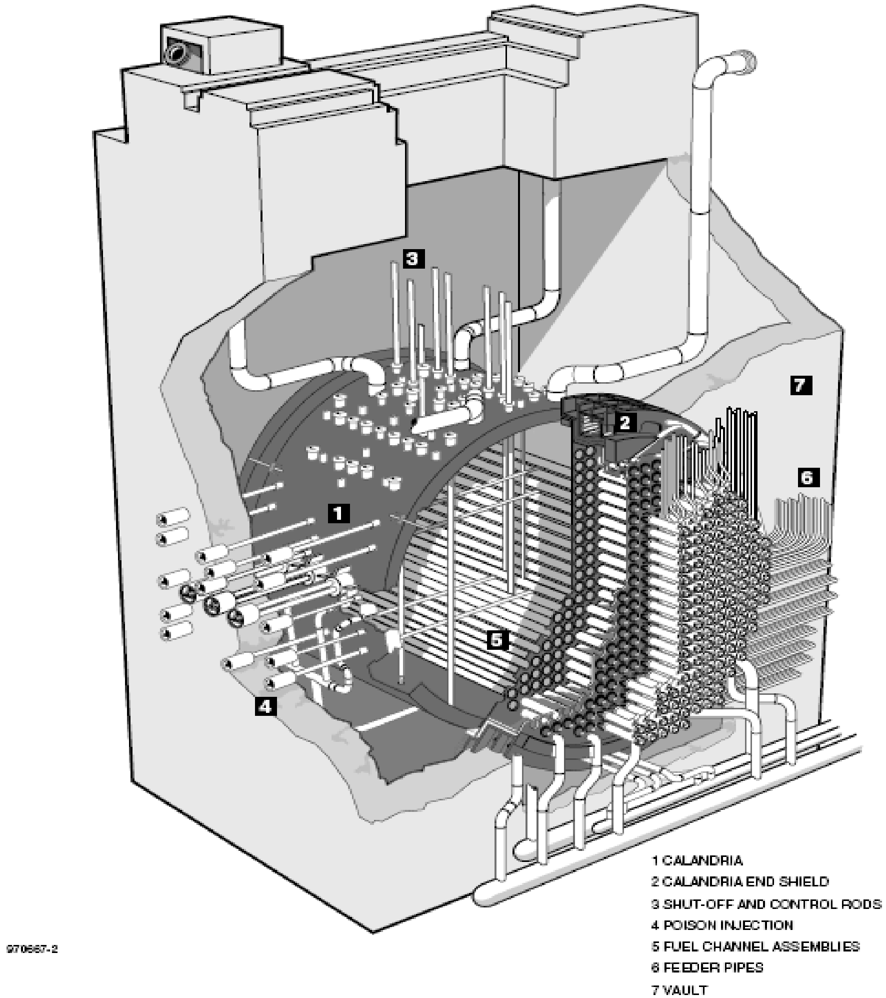
SAG Prioritization

Mitigating challenges to containment

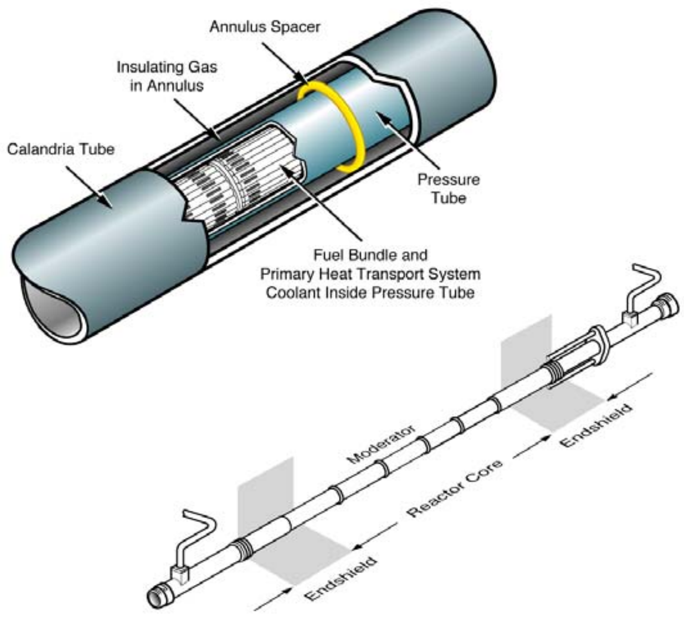
Role in PSA

Core Components

- 380 – 480 horizontal channels
- Calandria tank containing heavy water
- Shield tank containing light water

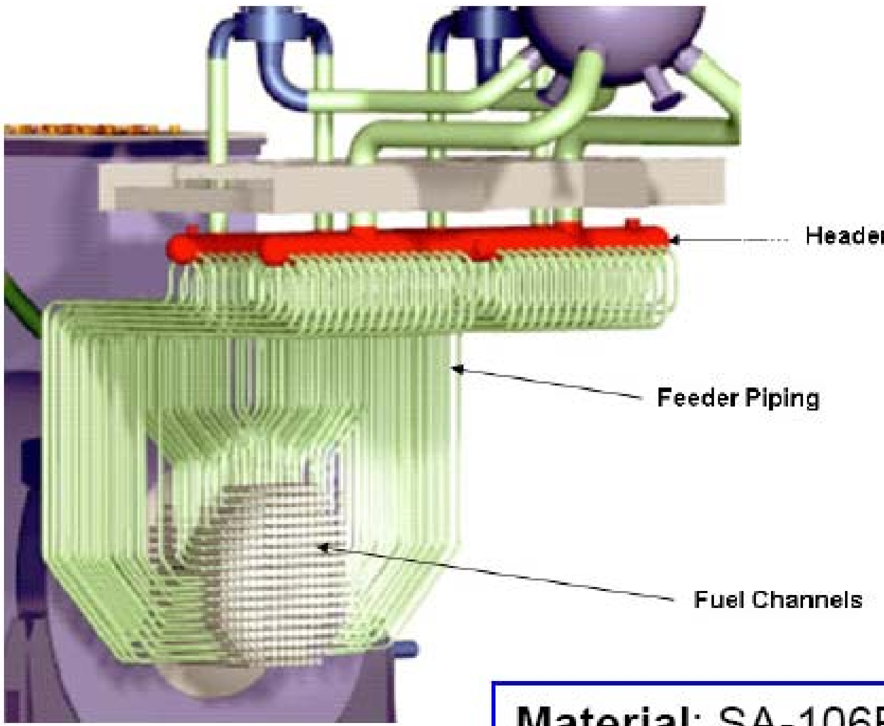


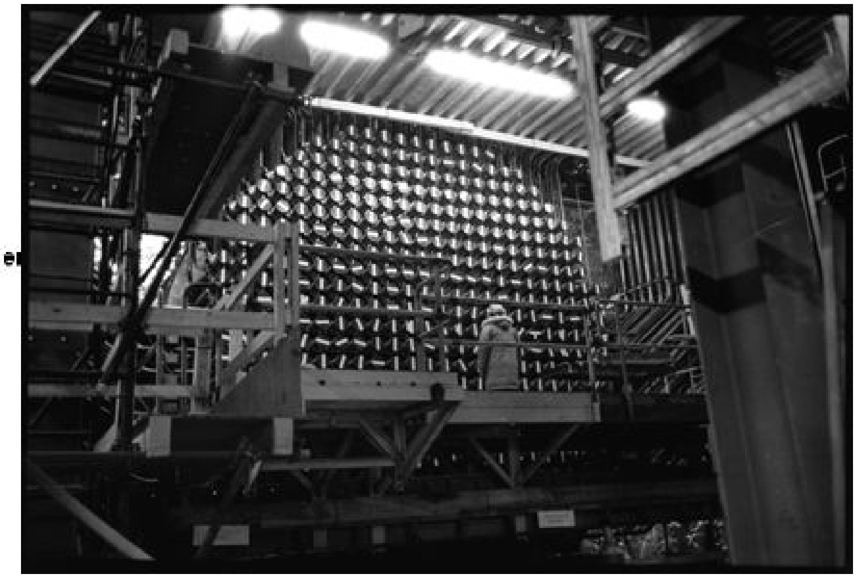
Fuel Channels



- Fuel in cylindrical bundles
- Pressure tube (PT) is the pressure boundary
- Gas filled annulus between inner PT and outer calandria tube (CT)
- Calandria tube surrounded by heavy water moderator

Heat Transport System Piping





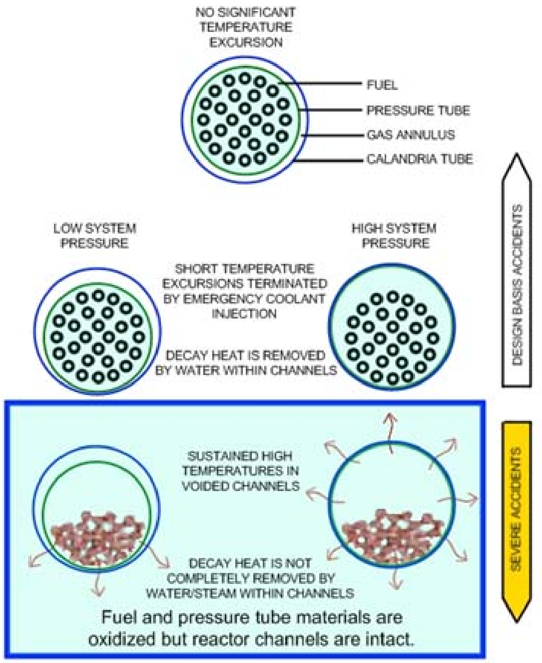
Material: SA-106B carbon steel
Pipe Size: NPS 1.5 (DN 40) through NPS 3.5 (DN 90) **Length**: 20 feet (6.1 m) through 60 feet (18.3 m)
Operating Pressure: nominally 9 to 11 MPa
Operating Temperature: nominally 250 to 310°C

Entry Conditions

- Need for CANDU-specific entry conditions:
  - No direct measurement of core temperature available
  - Wider range of accident end-states involving fuel damage for CANDU plants, including DBAs
  - Fuel damage alone is not an indication of imminent transition to a severe accident

- Entry conditions must distinguish onset of severe accident conditions from those accidents that can be effectively managed by EOPs

Figure 3: CANDU Fuel Channel Under Degraded Cooling Conditions

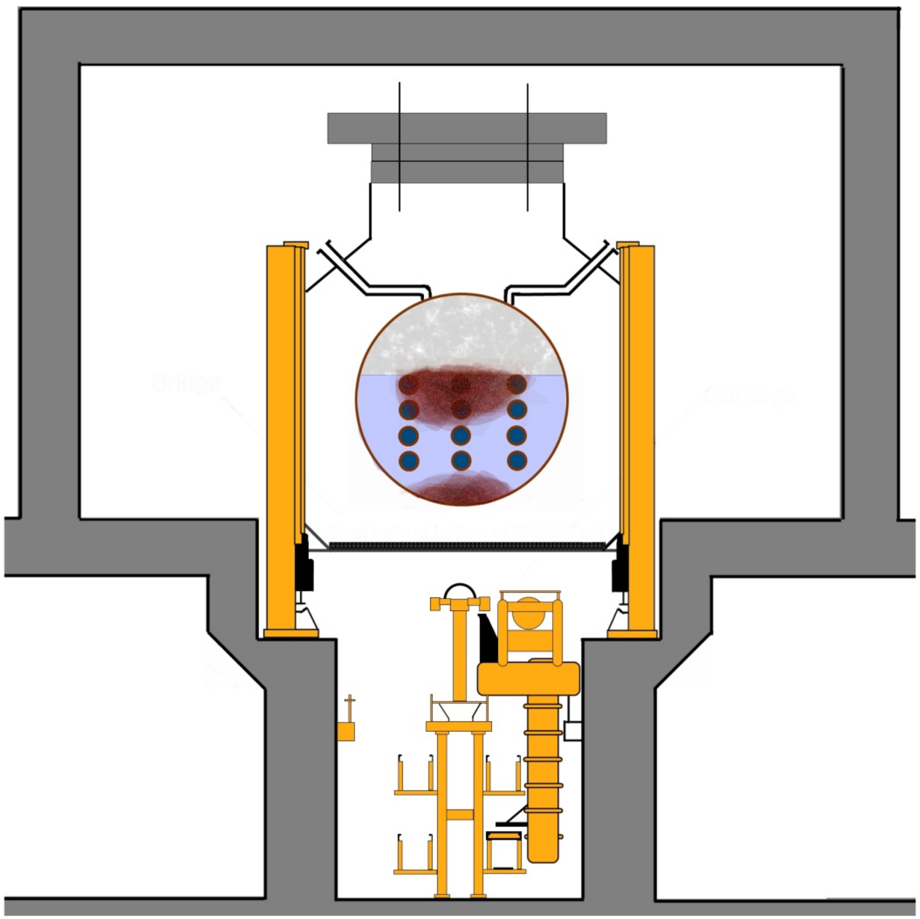


Entry Conditions

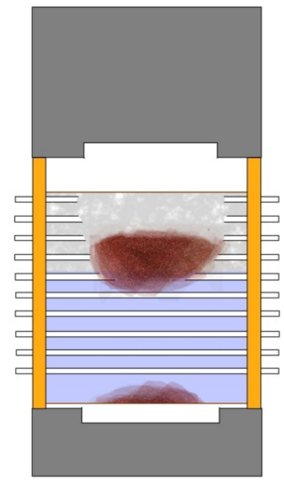
Table 1: SAMG Entry Conditions

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 ***or*** 3. Major release of fission products from the fuel	Moderator level below top of highest channels Plant radiation levels > setpoints	Moderator level instrumentation Ex-containment gamma measurements

Onset of Severe Accident



Calandria Face View



Calandria Side View

Entry Setpoint

- Design basis events ~ <1% FP release to containment if plant responds as expected (% core damage determined by correlation to dose rate measurement outside containment)

- ~ >10% FP release is clearly in severe accident range

- Setpoint = measured dose rate corresponding to calculation at specified locations assuming 3% FP release to containment

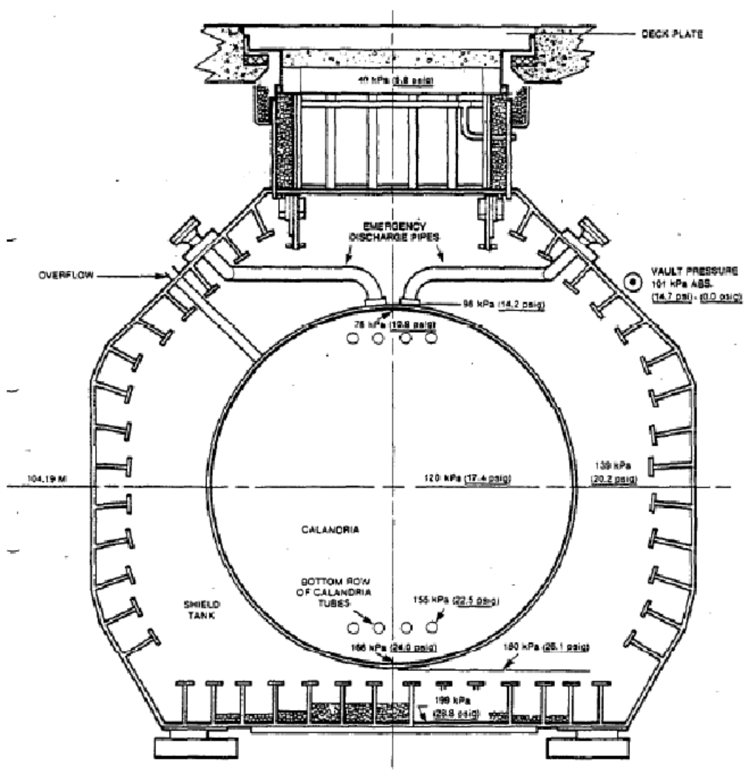
Prioritization of Barriers to Severe Accident Progression

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.

In its *initial* form, the order of the first three SAGs was reversed (i.e., 3-2-1)

Figure 4: Barriers to Accident Progression



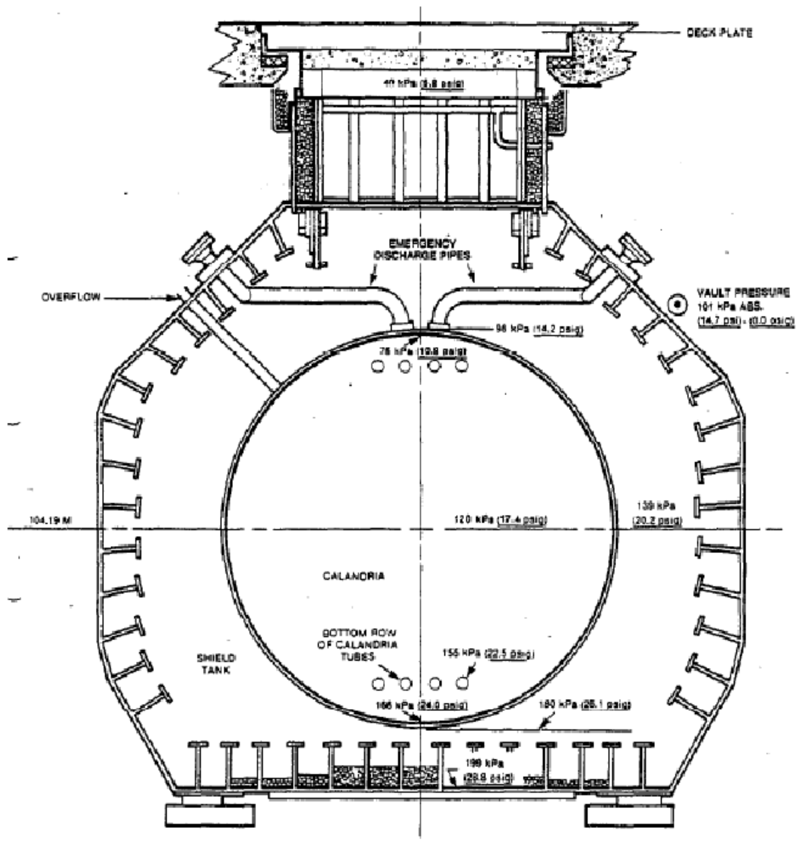
Considerations

•"Reverse" order gives priority to protecting the intact barriers by external vessel cooling

•Current order establishes priority to internal vessel cooling and for recovery actions

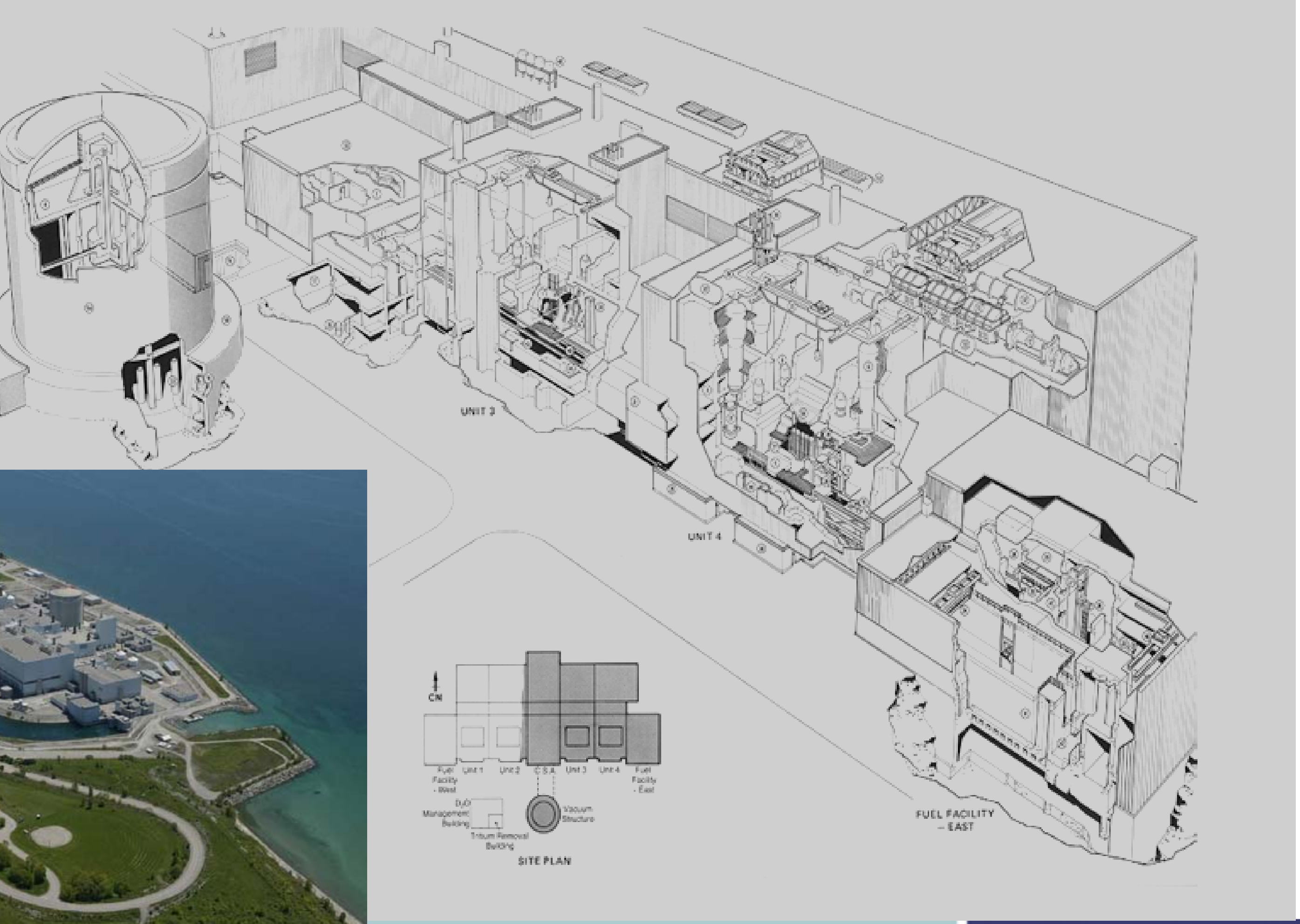
- Actions in SAG1 likely already attempted in EOPs
- Water added to HTS will find its way to the intact barrier
- Recovery of ECC in recirculation mode always a high priority

Figure 4: Barriers to Accident Progression



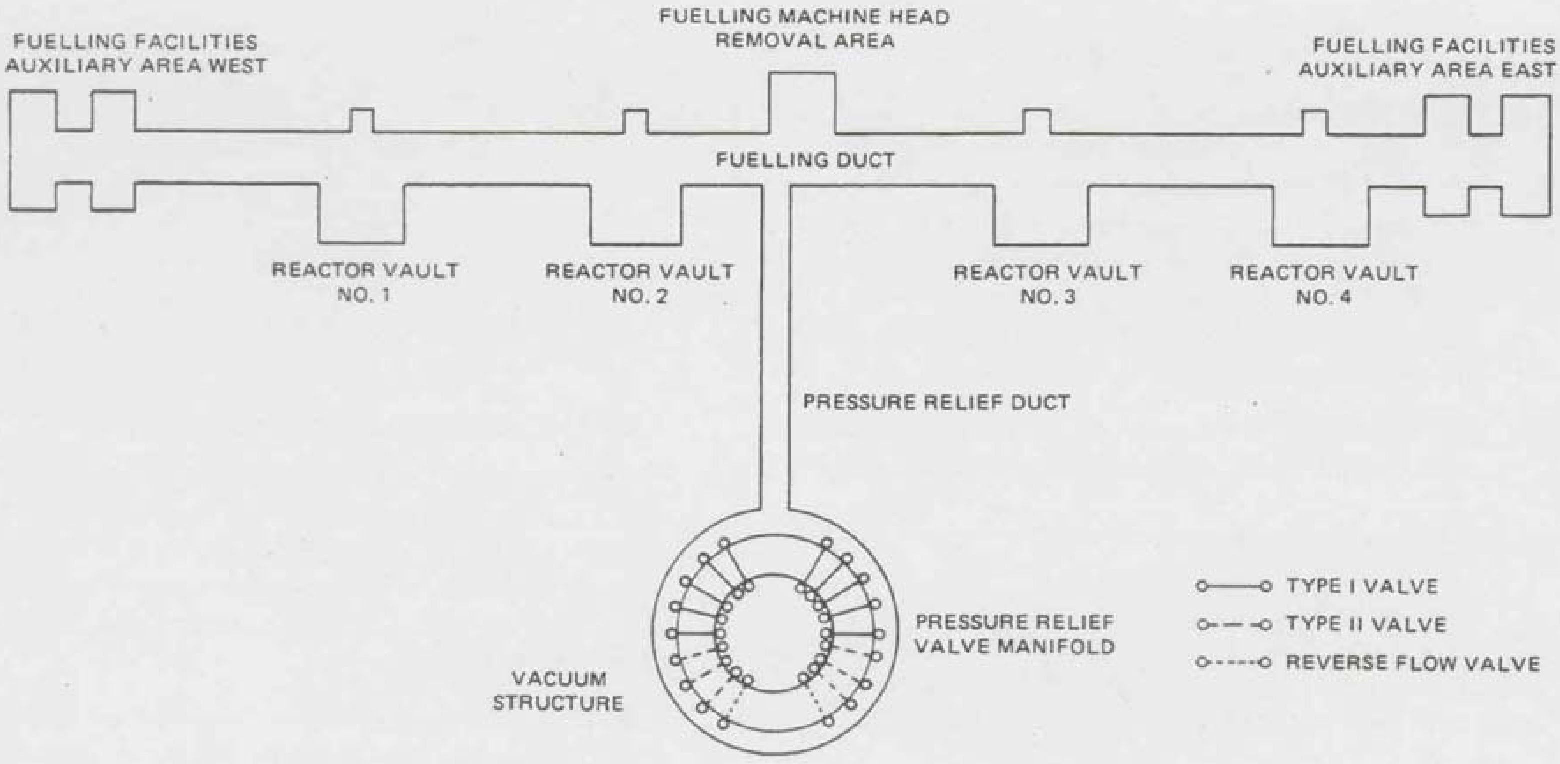
Diagnosing and Mitigating Challenges to Containment

Figure 5: Darlington Multi Unit Containment



Containment Schematic

Figure 6: Outline of the Multi-Unit Containment Envelope



Example - Basis for Hydrogen Diagnosis

- Hydrogen source term related to rate of accident progression;
- Correlation between hydrogen production and degree of fission product release to containment;
- Expectation that hydrogen will be mixed by pressure differentialst and that there will be mass transfer between the accident unit and other reactor buildings (can be estimated by comparing relative radiation measurements at similar locations outside each reactor building);
- Tracking of mass transfer to the VB (analogous to “venting” from the containment to the VB) for which pressure and temperature measurements are available;
- Assumption that igniters will maintain local hydrogen concentrations close to the flammability limit.
- Assessment of steam concentration to determine flammability

Role of Human Actions in CANDU PSA

- Level 1 PSA actions supported by EOPs
  - Alternative sources of cooling water to HTS
  - Moderator make-up
- Limited role seen for innovative Level-2 actions supported by SAMG
  - Incompatibility between PSA requirements for operator actions and SAMG decision making process
- Benefit anticipated at Entry and Exit from SAM Guidelines

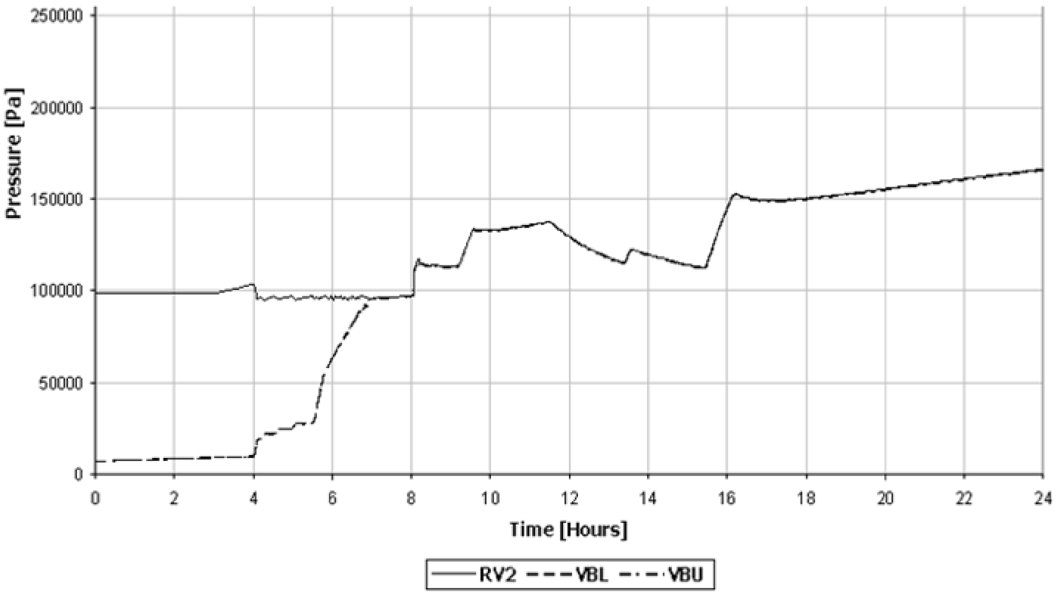
Closing Potential Early Release Pathways - Example

•In EOPs, low pressure filtered air venting is used to maintain containment subatmospheric after design basis accidents

•In SA, where containment pressure may be above atmospheric, this pathway may be at risk due to the pressure transients that accompany accident progression

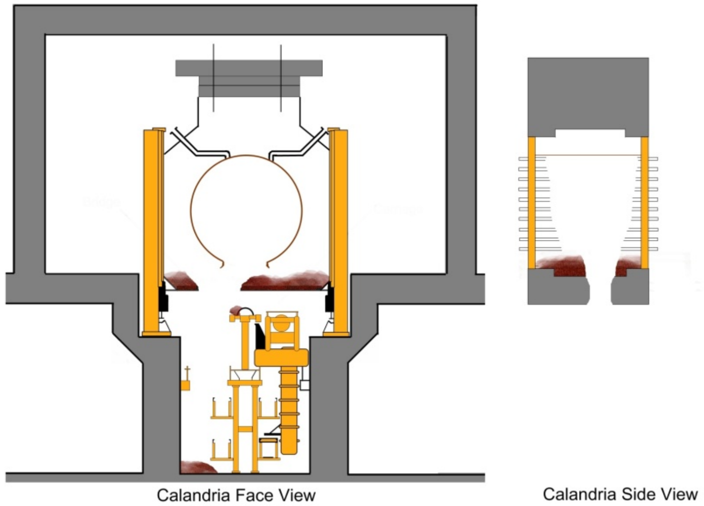
•Can be addressed in SACRG-1

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



Long-Term Stable State

- Failure to mitigate progression can result in fuel attacking the basemat
- Accumulation of water in the fuelling duct during accident progression but insufficient to cool debris
- SAMG aims to flood the duct to cover debris before exiting (only one strategy)
- Reduces likelihood of containment failure due to MCCI



Summary

Unique plant design features represent a challenge to structure of CANDU SAMG

More experience with drills will help to validate current approach or identify need for changes

Importance of SAMG to PSA is expected to be to reduce the likelihood of potential early failure pathways and reduce the impacts of MCCI by ensuring long term cooling

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

P. Scheib, M. Schneider, M. Krauß

Federal Office for Radiation Protection

Oct. 26, 2009, ISAMM2009

Böttstein

Outline

— History of AM in Germany

— PSR and PSA

— Implementation of AM-measures

— Level 2 PSA results on efficiency of SAM

— Conclusion and Outlook

History of AM in Germany

— **Initiated by Risk Study Phase B (1981-1989)***
- **Existing safety margins can be used to prevent core damage or mitigate the consequences**
- **Primary and secondary bleed and feed as preventive action**
  - **Reduction of CDF by factor of ~8**
- **Primary bleed as mitigative action**
  - **Reduction of high pressure core melt by a factor of ~7**
- **Filtered containment venting (FCV)**
- **Limitation of hydrogen-content inside the containment**

} not quantified

— **Consulting mandate to the RSK and resulting recommondations (1986-88)**
- **Safety review of all German NPPs**
- **Accident Management**
- **Recommendations for PSR**

* Deutsche Risikostudie Kernkraftwerke Phase B, ISBN 3-88585-809-6, BMFT 1990

PSR and PSA

Guide
Fundamentals on Periodic Safety Review for NPPs

Guide
Safety State Analysis

Compilation of protective goal-oriented requirements of the entire actual nuclear technical rules and standards

Guide
Probabilistic Safety Analysis

Methods for PSA of NPPs
(Methoden zur PSA für KKW)

Data for PSA of NPPs
(Daten zur Quantifizierung von Ereignisablaufdiagrammen und Fehlerbäumen)

Guide
Security Analysis

Changes in PSA requirements 1997 - 2005

— **Calculation of core damage states, taking into account preventive accident management measures**
  - **Evaluation of efficiency of preventive AM-measures**

— **Extension of the event spectrum to external hazards**

— **Extension of Level 1 PSA to LPSD**

— **Performance of Level 2 PSA for full power operation**
  - **Evaluation of efficiency of mitigative AM-measures**

Conduct of Level 2 PSA in Germany*

— **Required as part of PSR since 2005 for full power operation**

— **2-step approach (based on existing Level 1 PSA) or integrated approach possible**

— **Evaluation of effectiveness of SAM-measures**

— **Hints how to present results in order to support emergency management (but not required)**

* According to: Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke: Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, *BfS-SCHR-37/05*, 2005

Implementation of AM-measures in PWRs

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

Implementation of AM-measures in BWRs

Measure	KKB	KKI 1	KKP 1	KKK	KRB B	KRB C
Emergency management manual	●	●	●	●	●	●
Independent injection system	●	●	●	●	□	□
Additional injection and refilling of the RPV	●	●	●	●	●	●
Diverse pressure limitation for the RPV	●	●	●	●	●	●
Assured containment isolation	●	●	●	●	√	√
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

Implementation of AM-measures

- Some remarks:
  - Bleed and Feed as preventive and mitigative action for PWRs were based on PSA (Risk Study B), but not on plant specific PSA
  - Selection of SAM-measures not based on Level 2 PSA
  - Ongoing discussion about usefulness of PAR
    - Statement by RSK includes insights gained from the Level 2 PSA for the reference plant used to determine the design of PARs
- Module 7 of “Safety Criteria for NPPs”
  - Planning of AM based on “representative event sequences”
    - List of events + events taken from PSA results

Examples for PSA results on SAM efficiency

—Konvoi

— PWR 1300 MWe

— Entered commercial operation 1988 - 1989

— Study by GRS on behalf of BMU/BfS

— 2-step approach based on existing Level 1 PSA

— PARs and FCV examined for 3 selected accident scenarios

—SWR 69

— BWR 900 MWe (reference plant)

— Entered commercial operation 1976 - 1983

— Study by GRS on behalf of BMU/BfS

— 2-step approach based on existing Level 1 PSA

—GKN 1

— 3-loop PWR 840 MWe

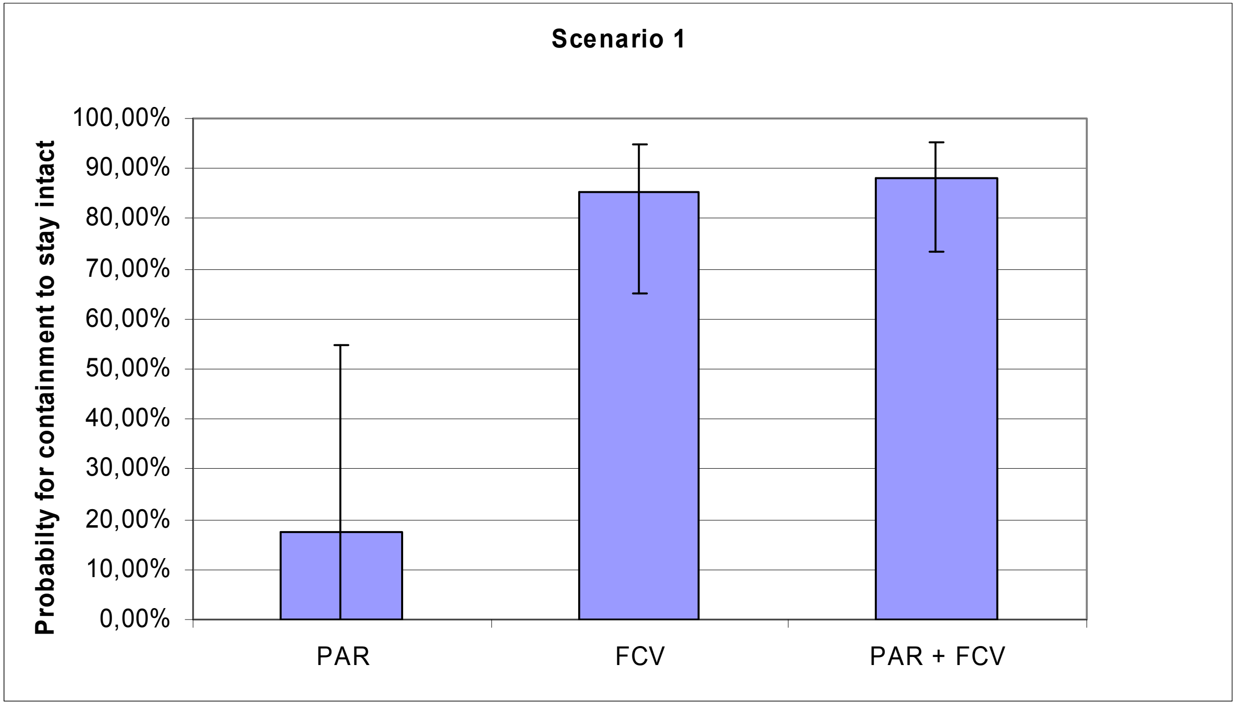
— Commercial operation since 1976

— Level 2 PSA as part of PSR (currently in review)

— 2-step approach based on existing Level 1 PSA

PSA results on SAM efficiency: Konvoi*

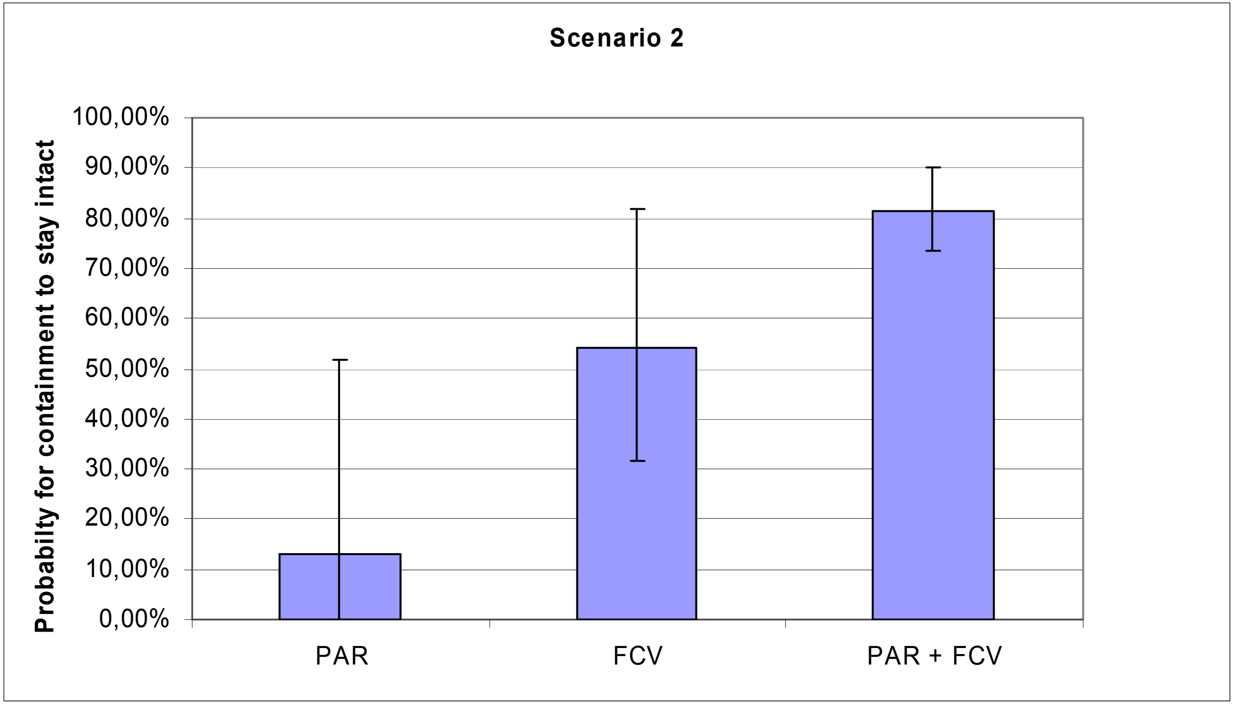
Scenario 1: total loss of steam generator feed with primary bleed



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

PSA results on SAM efficiency: Konvoi*

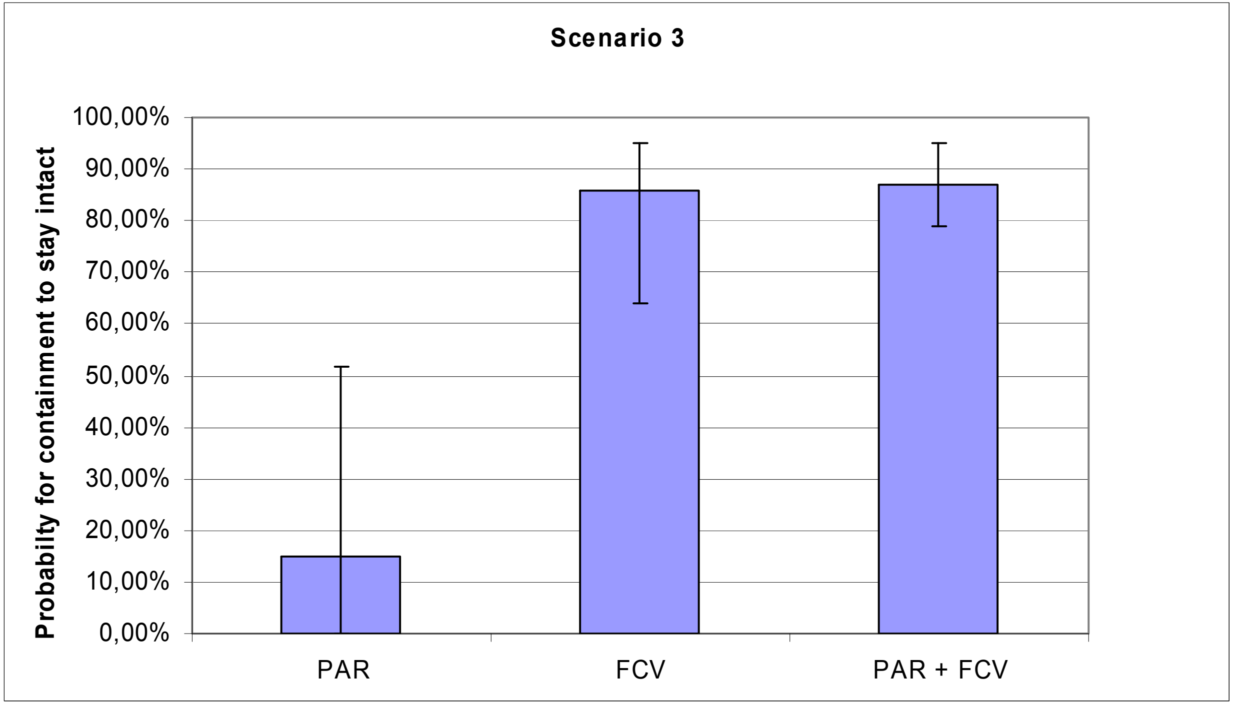
Scenario 2: break of pressurizer connection pipe



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

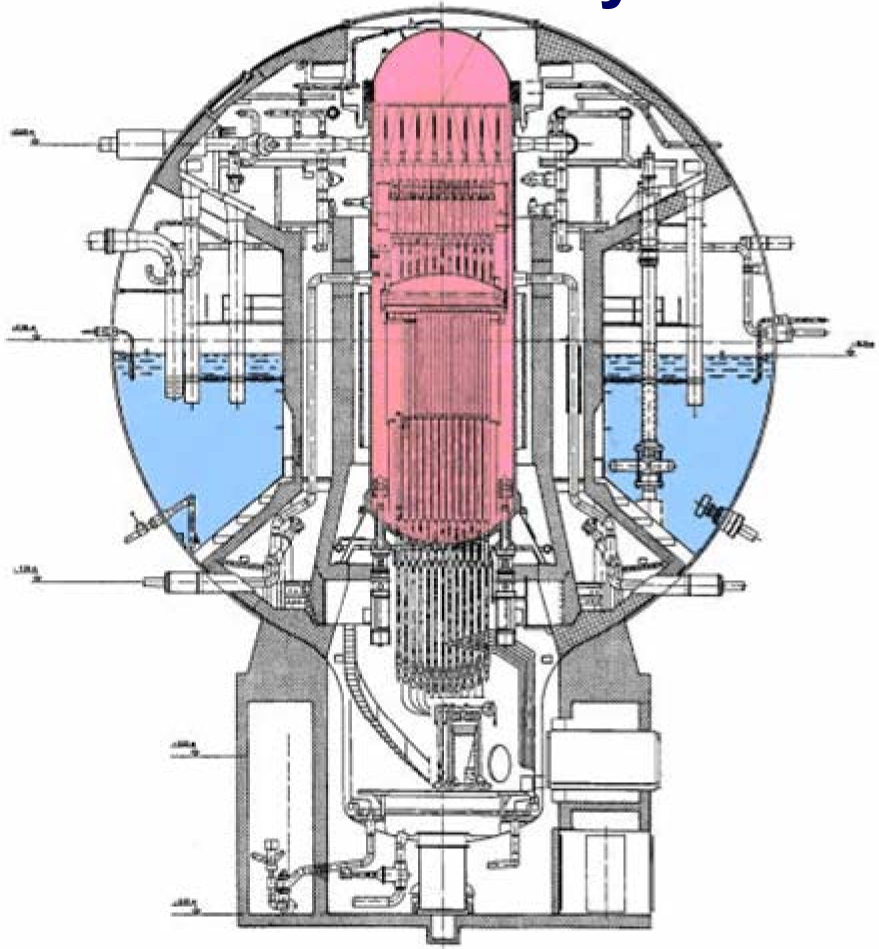
PSA results on SAM efficiency: Konvoi*

Scenario 3: small leak in the hot leg and loss of SG heat removal



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

PSA results on SAM efficiency: SWR 69*



* 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

PSA results on SAM efficiency: SWR 69*
Typical accident progression:

— **Core damage typically at low pressure (f>97%)**

— **Low probability (<2%) to retain partly molten core inside RPV**

- **Only in case of high pressure core melt**

— **Containment failure shortly after RPV failure**

- **melt-through of steel shell in control rod driving room (CRDR)**

— **Containment failure at elevated pressure, but below initiating pressure for FCV. Possibility of H_2-combustion outside containment.**

- **damage to adjacent buildings → new release paths**

— **High probability of large early release in case of core damage**

* 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

PSA results on SAM efficiency: SWR 69* Discussion on SAM-measures:

— **Flooding of CRDR to keep RPV intact:**
  - **Steam prevents water from reaching crucial parts of RPV → probably leads to large area failure of RPV**

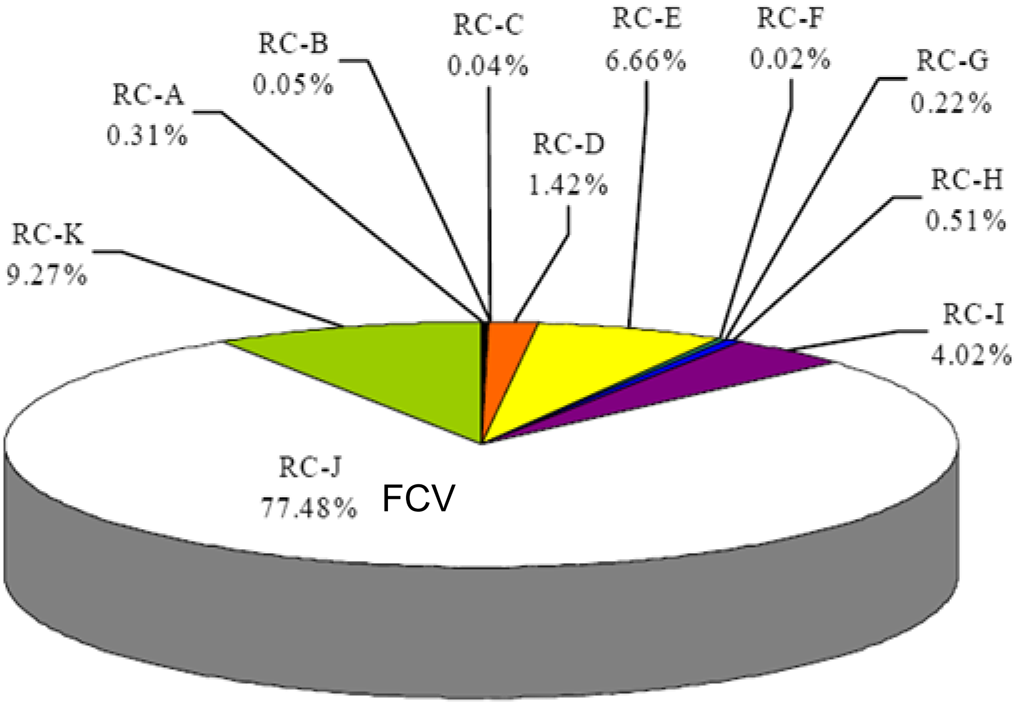
— **FCV in most cases not initiated before containment failure**
  - **Initiate more early in order to reduce pressure inside the containment and release H_2 to reduce damage to adjacent buildings**

— **Integrity of CRDR:**
  - **Modifications of the CRDR ensure fragmentation of core material**
  - **Cooling from outside**

* 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

PSA results on SAM efficiency: GKN 1*

Relative proportions of the total PDS frequency

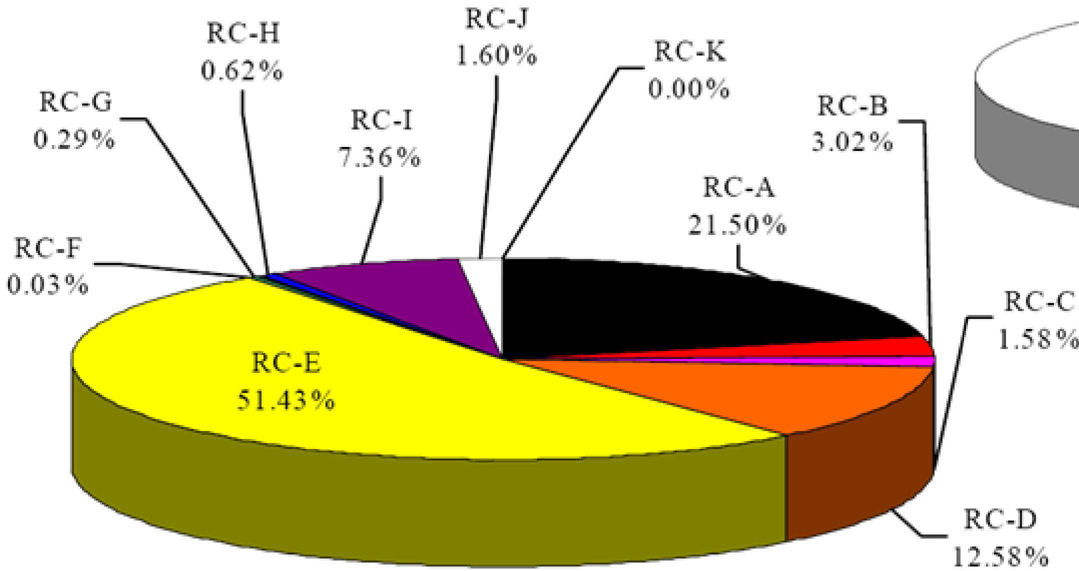


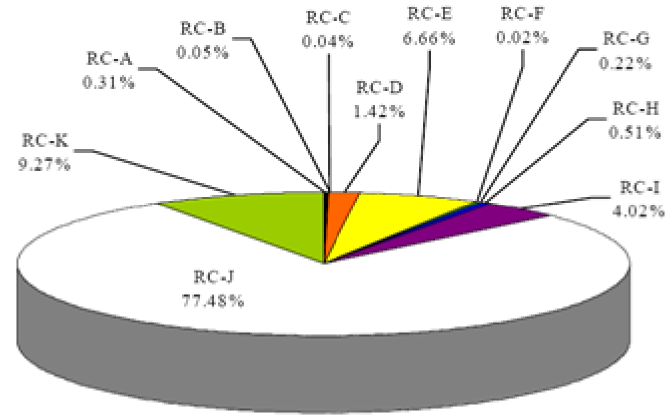
Release category	Containment (C) failure mode
RC-A	LOCA outside C
RC-B	Uncovered SGTR
RC-C	Early C rupture
RC-D	C isolation failure
RC-E	Covered SGTR
RC-F	Sump line failure
RC-G	Late C rupture
RC-H	Basemat melt-through
RC-I	Unfiltered C venting
RC-J	FCV
RC-K	No failure

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

PSA results on SAM efficiency: GKN 1*

Relative contribution of RC to the total potential released activity





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

PSA results on SAM efficiency: GKN 1*

Release category	Containment failure mode	Relative proportions of the total PDS frequency	Relative contribution of RC to the total release (excluding noble gases)
RC-A	LOCA outside containment	0.31%	21.50%
RC-B	Uncovered SGTR	0.05%	3.02%
RC-D	Containment isolation failure	1.42%	12.58%
RC-E	Covered SGTR	6.66%	51.43%
RC-I	Unfiltered containment venting	4.02%	7.36%
Sum		**12.46%**	**95.89%**
RC-J	Filtered containment venting	**77.48%**	**1.60%**

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

PSA results on SAM efficiency: GKN 1*

— **Dominant failure mode for the containment: gross failure under dynamic or static overpressure**

— **FCV effective method to avoid containment failure**

— **More detailed studies about effectiveness of AM-measures have been done by the operator as part of the sensitivity studies, but are not published**

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

Conclusion and Outlook

— **Various AM-measures implemented during the last 20 years**

— **Analysis of AM in PSA Level 1+2 required since 2005**

— **Importance of PSA regarding (S)AM increasing**
  - **e.g. Safety Criteria + safety-orientedness of PARs**

— **Feedback from development and review of Level 2 PSAs performed as part of PSR is becoming available and will be fed into PSA guidelines**
  - **Working group Level 2 PSA of the FAK starts this November**

Thank you for your attention

Patric Scheib

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Circumstances and Present Situation of Accident Management Implementation in Japan

OECD/NEA Workshop on Implementation of Severe Accident Management Measures (ISAMM-2009)
Böttstein, Switzerland

October 26 - 28, 2009

Haruo Fujimoto, Keisuke Kondo, Tomomichi Ito, Yusuke Kasagawa, Osamu Kawabata, Masao Ogino and Masahiro Yamashita

Japan Nuclear Energy Safety Organization (JNES)

Contents

1. Background and history

Date	Major events for AM
May, 1992	The Nuclear Safety Commission (NSC) of Japan issued a decision statement “Accident Management as a Measure against Severe Accidents at Power Generating LWRs,” which strongly recommended the regulatory body and utilities to introduce AM measures.
July, 1992	MITI encouraged utilities to establish AM implementation plans, using benefit of insights obtained from PSA.
March, 1994	The utilities submitted AM implementation plans to MITI. MITI reviewed utilities plans.
October, 1994	MITI made a report entitled “AM for Light Water NPPs,” in which MITI recommended utilities to undertake AM implementation plans toward 2000 and to prepare operating procedures and administrative framework.

1. Background and history (cont'd)

Date	Major events for AM
February, 2002	The utilities completed implementation of AM and reported to NISA (new regulatory body founded in January, 2001.) The effectiveness of AM for representative plants were evaluated by NUPEC (former of JNES.)
	NISA recognized that it was also important to evaluate effectiveness of AM measures for NPPs other than representative plants. And NISA requested utilities to perform evaluation of every NPPs.
March, 2004	The utilities performed evaluation of effectiveness of AM measures for every NPPs and submitted report entitled "PSA evaluation Report following AM Implementation." NISA reviewed this report with the help of JNES.
Up to now	Besides fifty-two operating NPPs, AM have been studied and implemented to four newly constructed NPPs.

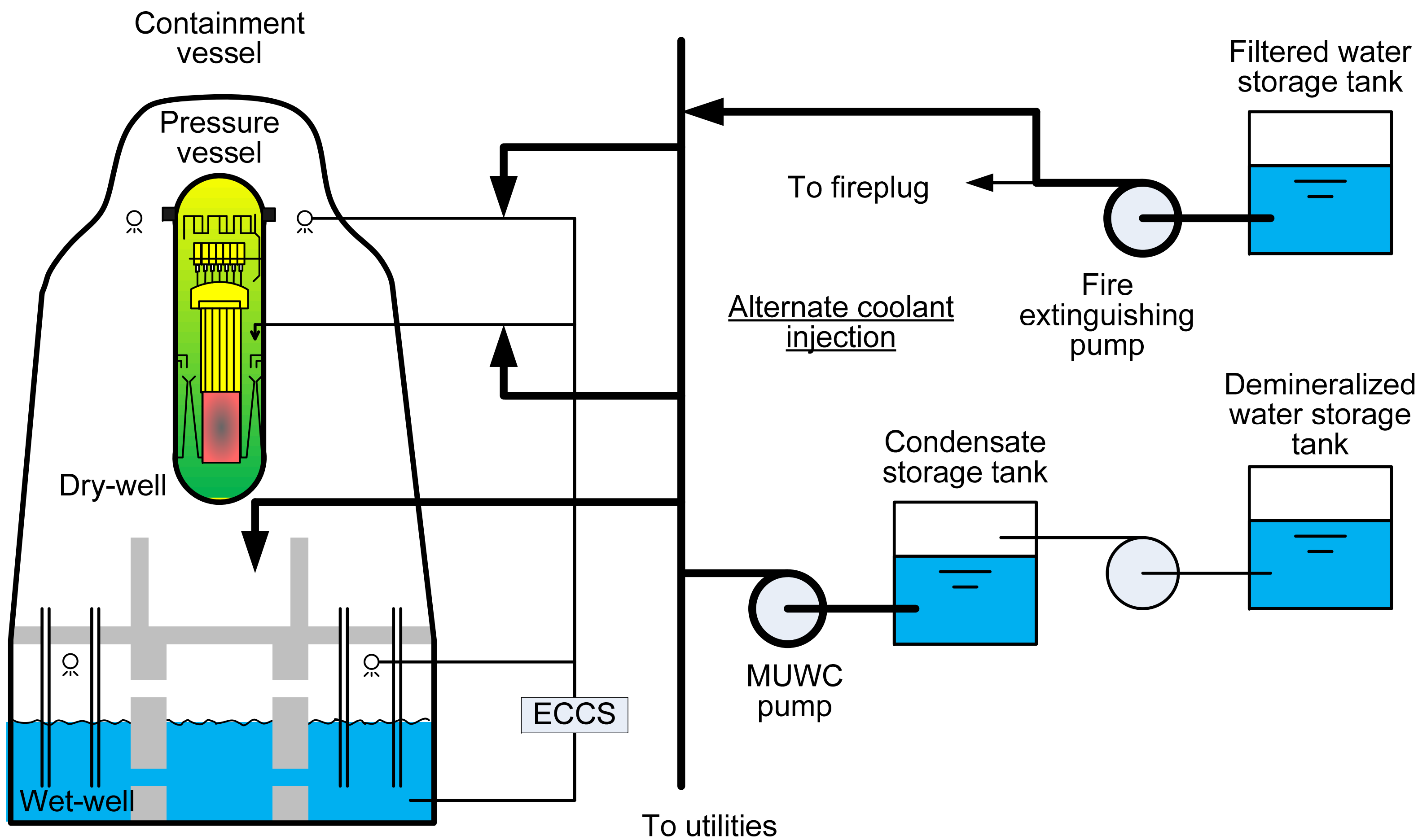
2. Accident management measures implemented to the operating NPPs

AM measures for BWR

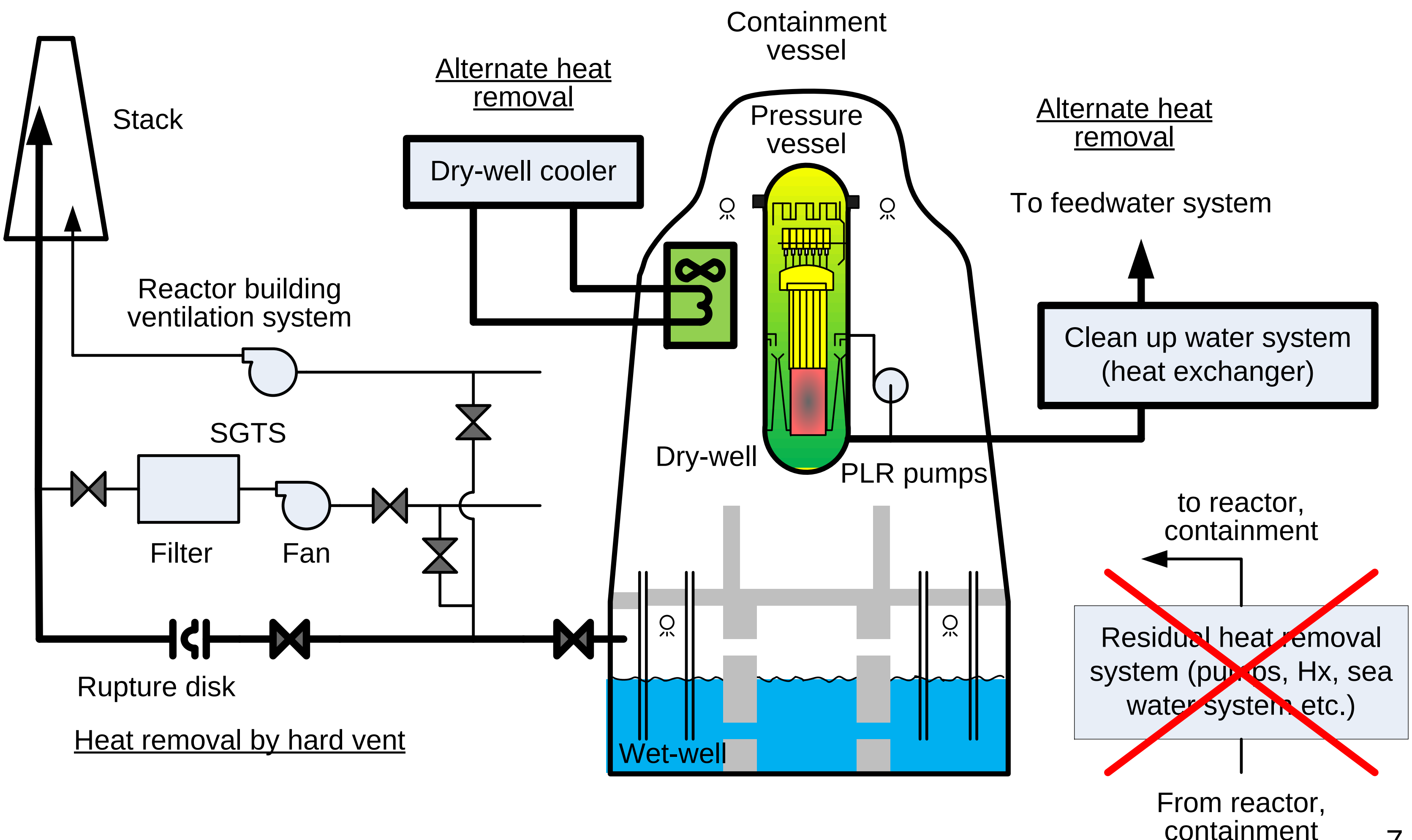
<table>
<tr><th>Safety function</th><th>Purpose</th><th>Prevention of core damage</th><th>Mitigation of core damage</th></tr>
<tr><td>Reactor shutdown</td><td>Alternate reactivity control</td><td>● ARI (except ABWR)
● RPT (except ABWR)</td><td>-</td></tr>
<tr><td rowspan="2">Coolant injection to RPV and CV</td><td>Reactor depressurization</td><td>● ADS actuation by L-1 (except BWR2, 3 and ABWR)</td><td>-</td></tr>
<tr><td>Alternate coolant injection</td><td colspan="2">● MUWC
● Fire extinguishing system or filtrate water system</td></tr>
<tr><td rowspan="3">Heat removal from CV</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
● Electric power supply from HPCS-DG (Single-unit site)
● Installation of dedicated EDG</td></tr>
<tr><td>Recovery of EDG</td><td colspan="2">● Recovery of EDG</td></tr>
</table>

ARI: Alternate rod insertion, RPT: Recirculation pump trip, ADS: Automatic Depressurization System,
MUWC: Makeup water system condensated, CUW: Reactor water cleanup,
RHR: Residual heat removal, HPCS: High pressure core spray

AM measures for alternate coolant injection (BWR)



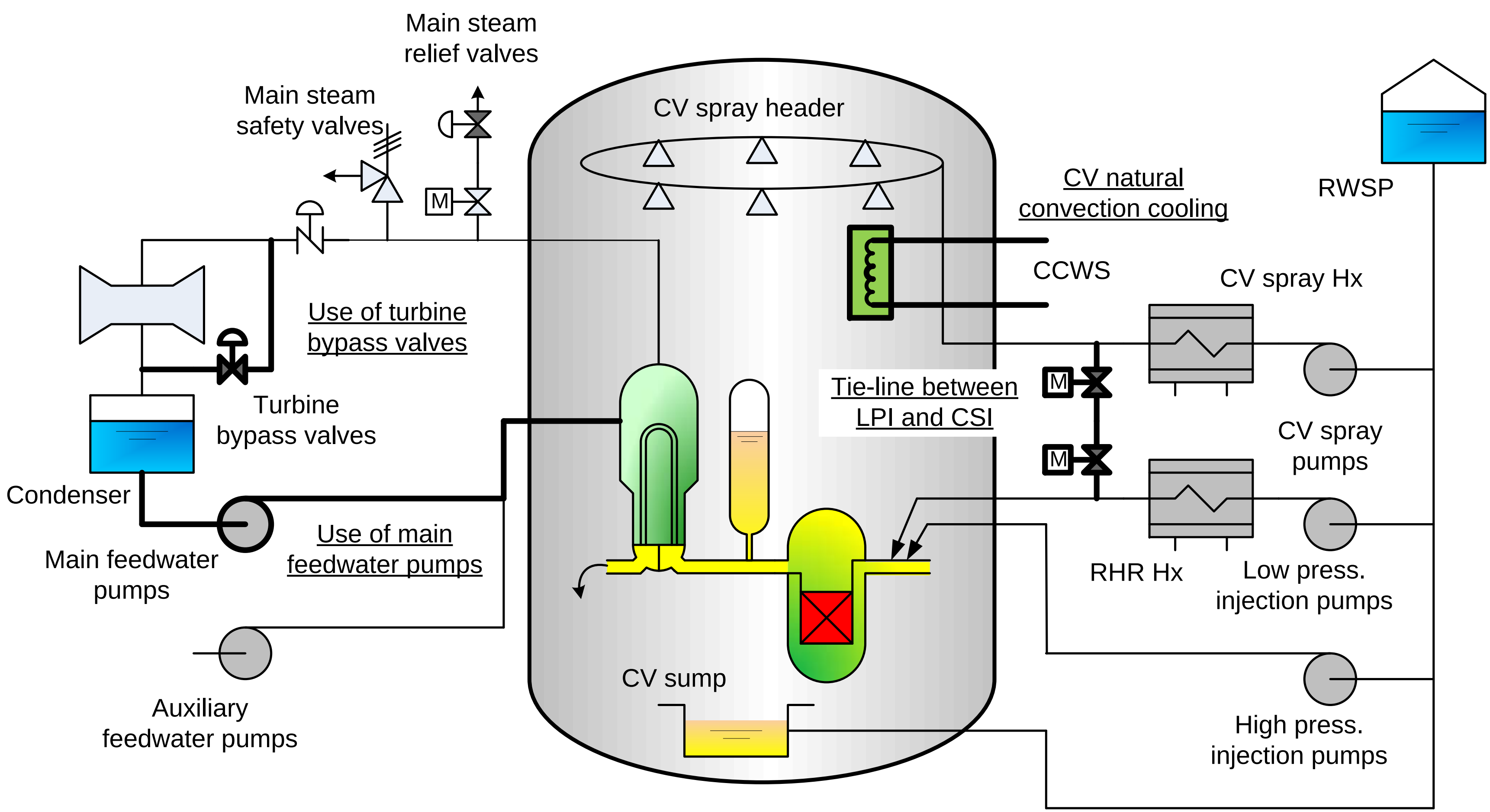
AM measures for CV heat removal (BWR)



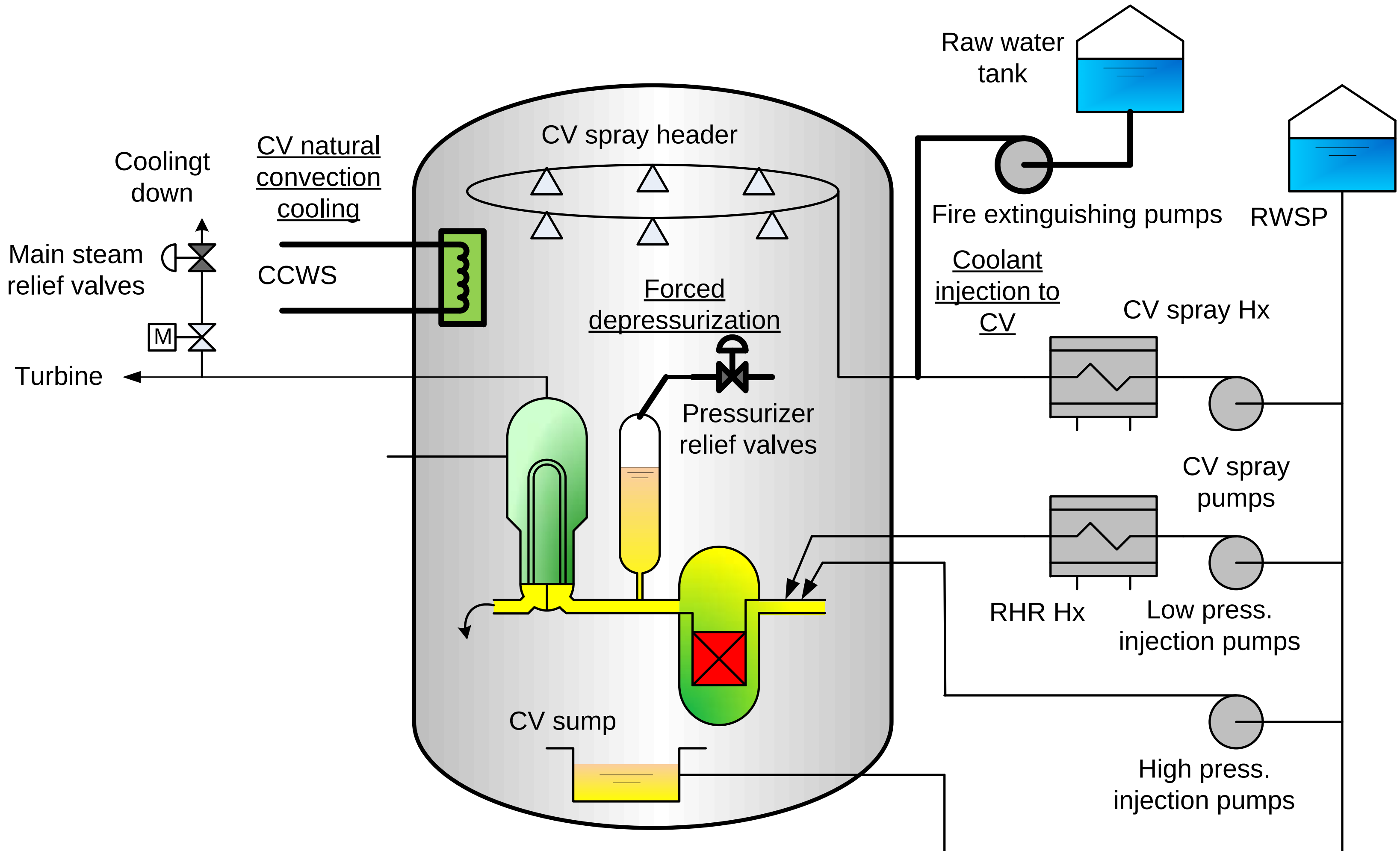
AM measures for PWR

<table>
<tr><th>Safety function</th><th>Purpose</th><th>Prevention of core damage</th><th>Mitigation of core damage</th></tr>
<tr><td>Reactor shutdown</td><td>Reactor shutdown</td><td>● Use of main feedwater pumps (ATWS)</td><td>-</td></tr>
<tr><td rowspan="3">Core cooling</td><td>ECCS injection</td><td>● LPI with turbine bypass valves</td><td>-</td></tr>
<tr><td>ECCS recirculation</td><td>● Alternative recirculation
➢ Tie-line between LPI and CSI
➢ Alternate recirculation pump
➢ Recirculation sump isolation valve bypass line</td><td>-</td></tr>
<tr><td>Isolation of coolant leakage</td><td>● Cooldown and recirculation</td><td>-</td></tr>
<tr><td>Confinement of radioactive materials</td><td>Heat removal from CV</td><td>● Natural convection heat removal
➢ Use of non-safety CV heat removal system
➢ Outside CV spray</td><td>● Natural convection heat removal
● Coolant injection to CV
● Forced depressurization of primary system
● Hydrogen igniter (Ice condenser CV plant)</td></tr>
<tr><td rowspan="2">Supporting function</td><td rowspan="2">Supporting function</td><td>● Alternate component cooling
➢ Air conditioning system
➢ BOP CCWS
➢ CV cooling system
➢ Fire extinguishing system</td><td>-</td></tr>
<tr><td>● Electric power supply from the adjacent unit</td><td>-</td></tr>
</table>

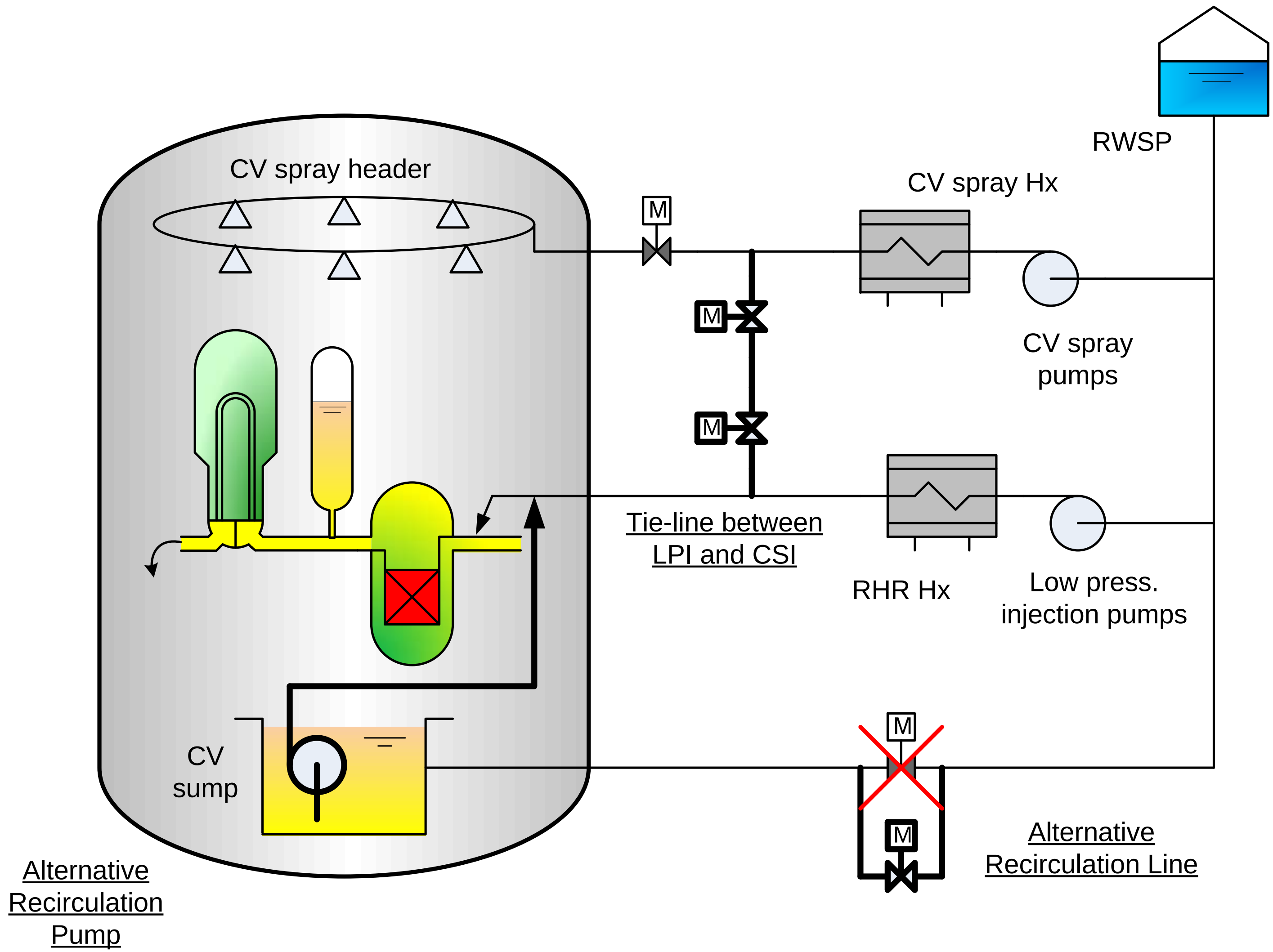
AM measures to prevent core damage (PWR)



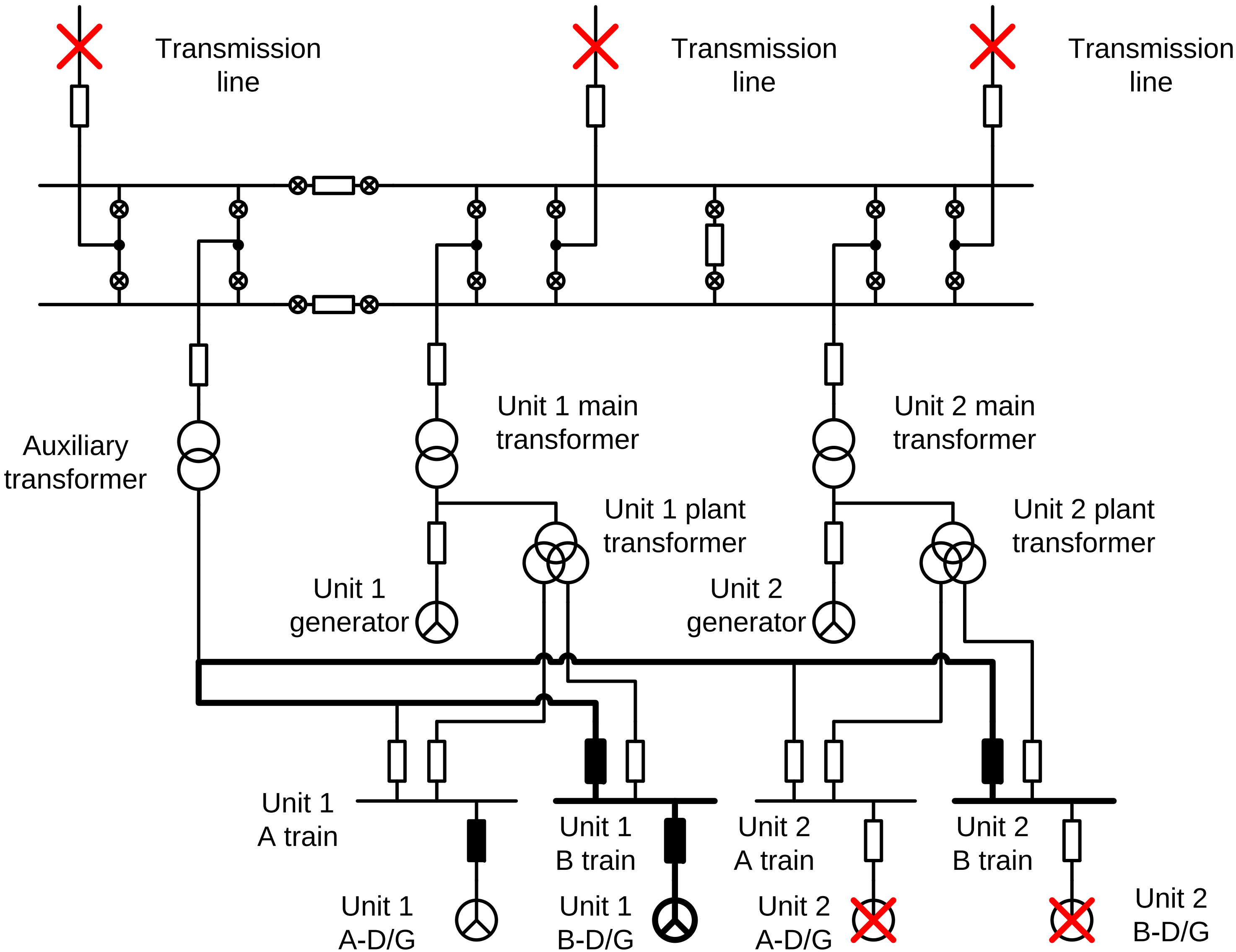
AM measures to prevent containment failure (PWR)



Comparison of alternatives for ECCS recirculation



Power supply from the adjacent unit



Reactor types and safety systems (BWR)

		type A	type B	type C	type D
Reactor		BWR2, 3	BWR4	BWR5	ABWR
Containment vessel		MARK-I	MARK-I	Mod. MARK-I, MARK-II, Mod. MARK-II	RCCV
Reactor scram		CRDHS SLCS	CRDHS SLCS	CRDHS SLCS	CRDHS SLCS, ARI FMCRD
ECCS	High pressure	HPCI IC(2)	HPCI RCIC	HPCS RCIC	HPCF(2) RCIC
	Low pressure	CS(2)	CS(2) LPCI(2)	LPCS LPCI(3)	LPFL(3)
Containment heat removal		SHC(2) CCS(2)	RHR(2)	RHR(2)	RHR(3)

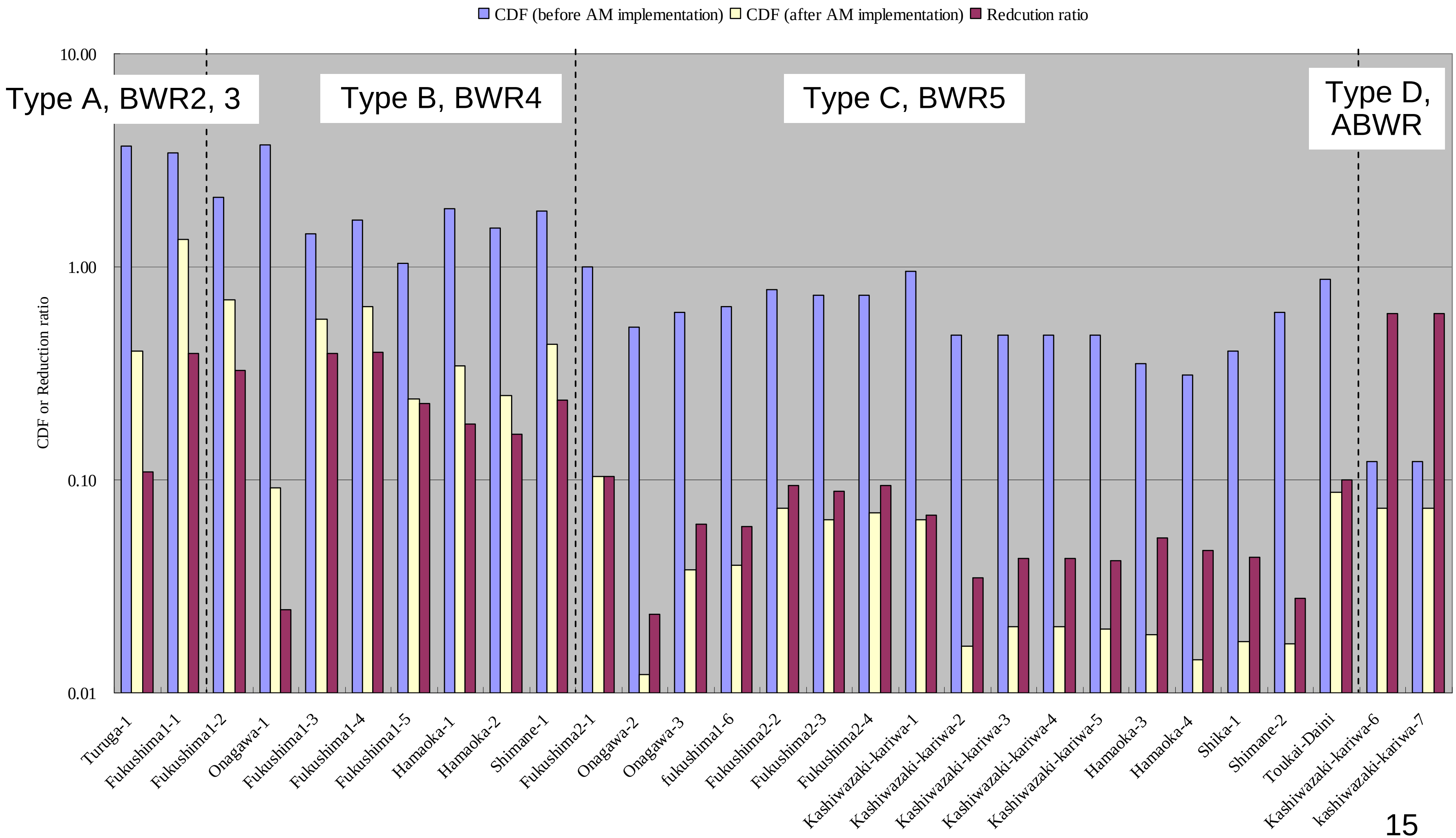
ARI: Alternate rod insertion, CCS: Containment cooling system, CRDHS: Control rod drive hydraulic system, CS: Core spray, FMCRD: Fine motion control rod drive, HPCF: High pressure core flooder, HPCI: High pressure coolant injection, HPCS: High pressure core spray, IC: Isolation condenser, LPCI: Low pressure coolant Injection, LPFL: Low pressure flooder, LPCS: Low pressure core spray, RCCV: Reinforced concrete CV, RCIC: Reactor core isolation cooling, RHR: Residual heat removal, SLCS: Standby liquid control system, SHC: Shutdown cooling

Reactor types and safety systems (PWR)

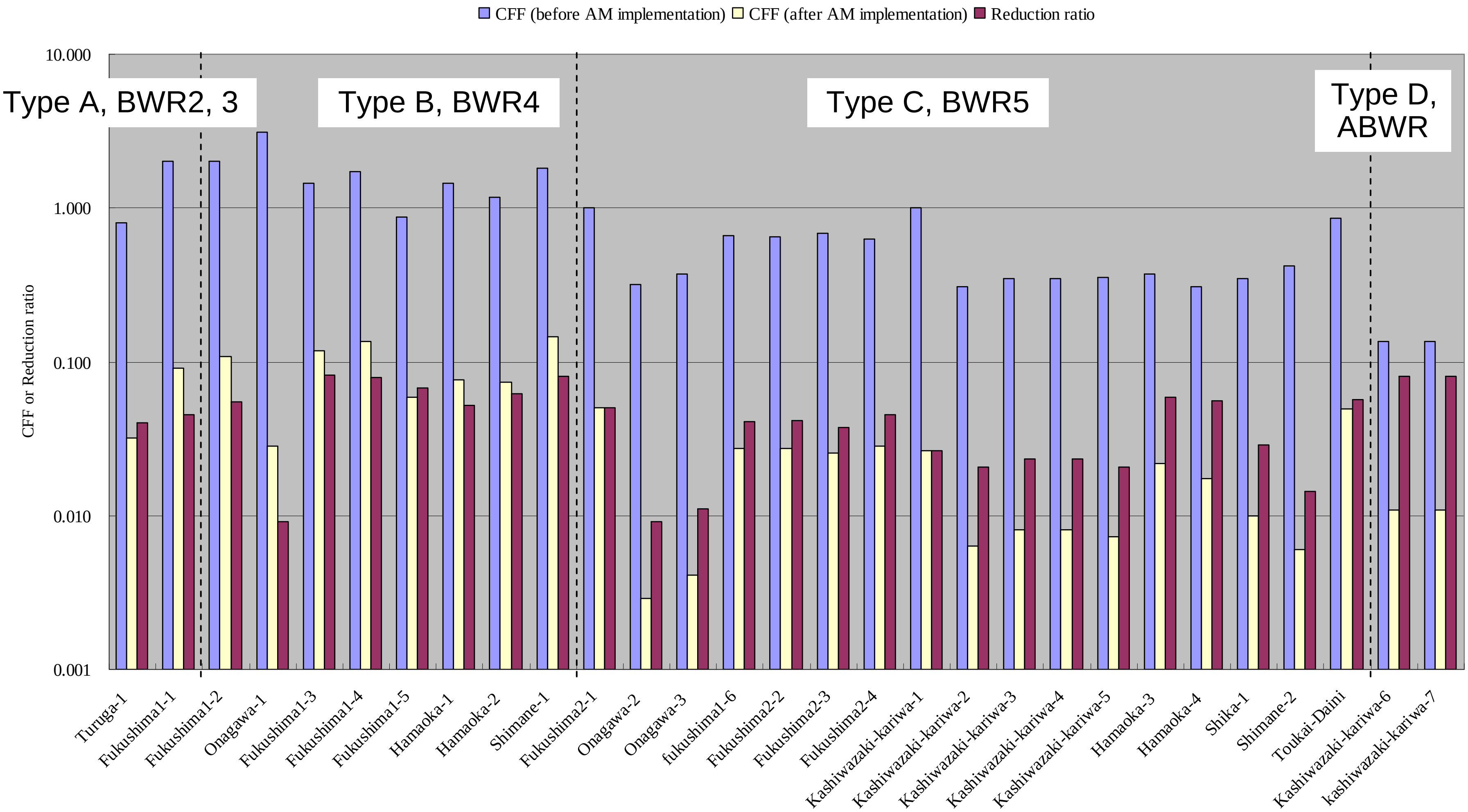
Safety systems		type A	type B	type C	type D
Reactor Type		Two-loop	Three-loop	Four-loop with ice condenser	Four-loop
ECCS	HPI	HPI(2), Boosted by LPI during recirculation	CHSI(3), Boosted by LPI during recirculation	CHSI(3), HPI(2), Boosted by LPI during recirculation	HPI(2)
	LPI	2	2	2	2
	Acc.	2	3	4	4
Aux. feedwater		M/D (2) T/D (1)	M/D (2) T/D (1)	M/D (2) T/D (2)	M/D (2) T/D (1)
CV spray		2	2	2 RHR spray(2)	2

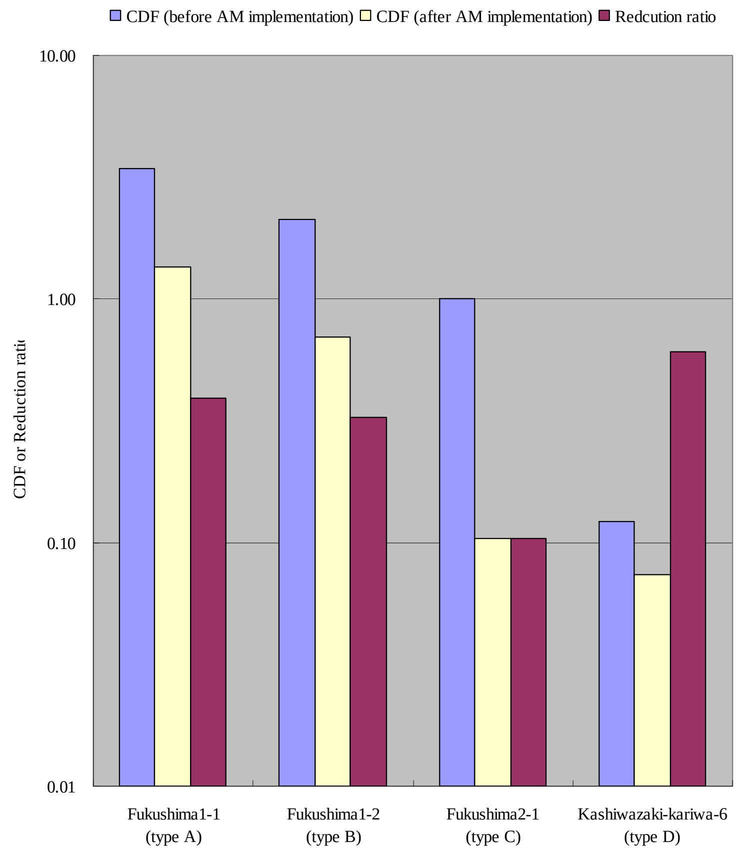
HPI: High pressure injection, LPI: Low pressure injection,
CHSI: Charging safety injection, M/D: Motor-driven, T/D: Turbine-driven,
RHR: Residual heat removal

CDF results before and after AM implementation (BWR)

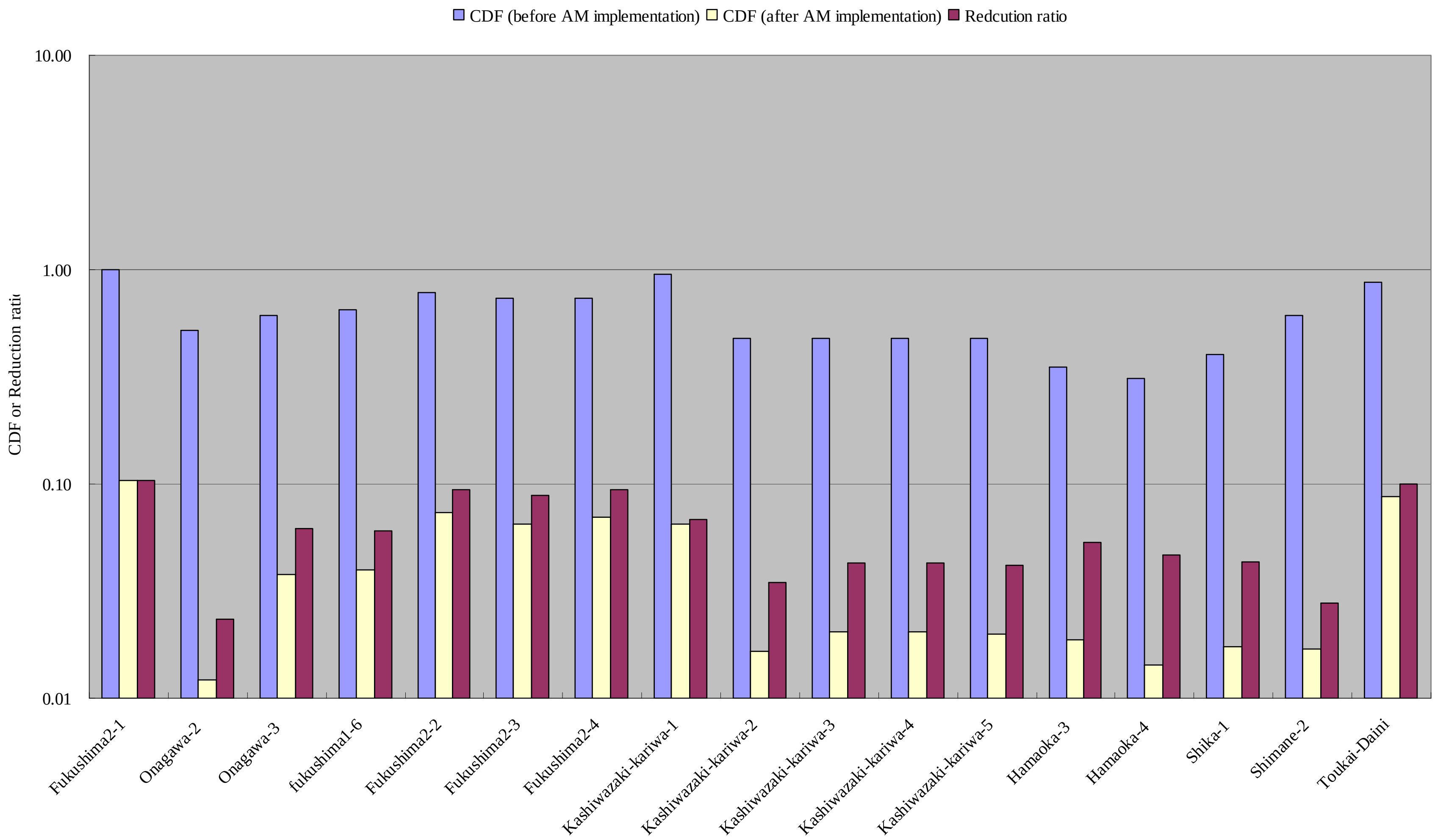


CFF results before and after AM implementation (BWR)



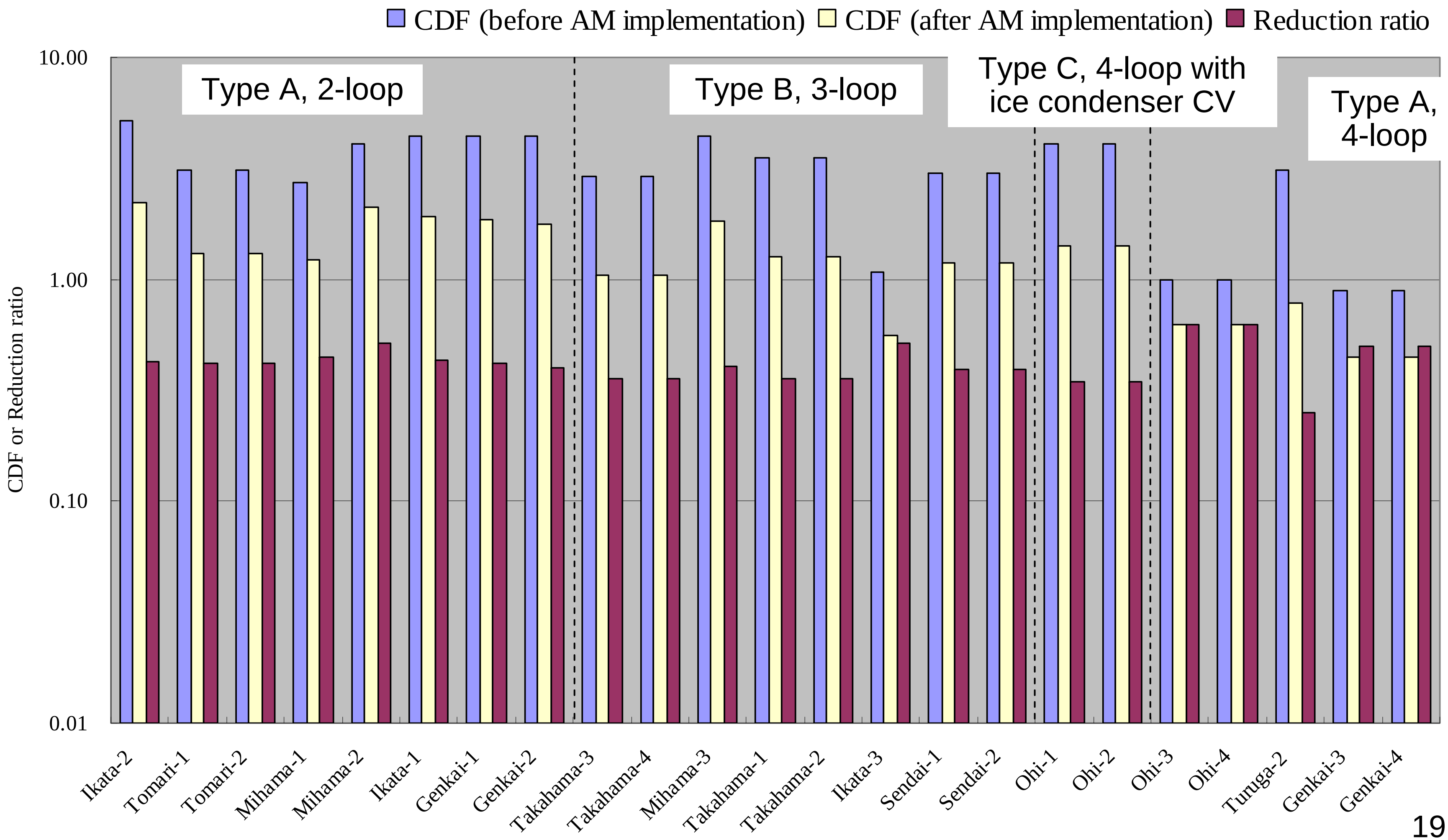


- CDFs of type D plants before AM implementation are small comparing to type A, type B, and type C plants, while the reduction ratios by AM are large, i.e. AM effect is small.
- ARI and RPT are installed, and highly redundant systems are used for the coolant injection and residual heat removal functions in type D plants, which make CDFs before AM implementation much smaller than the other.
- Additional reactor shutdown, coolant injection, and residual heat removal function are considered not needed as AM measures.

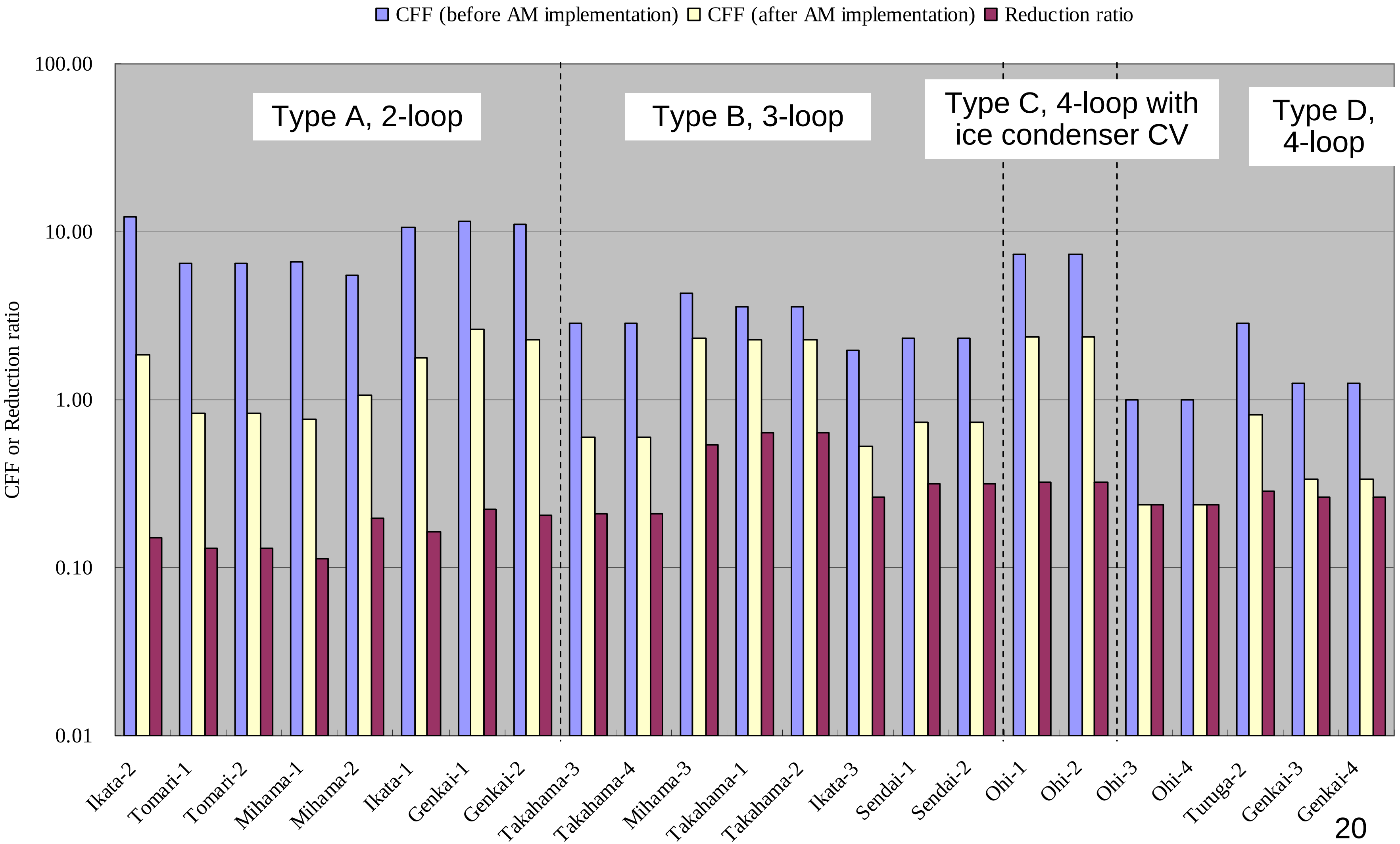


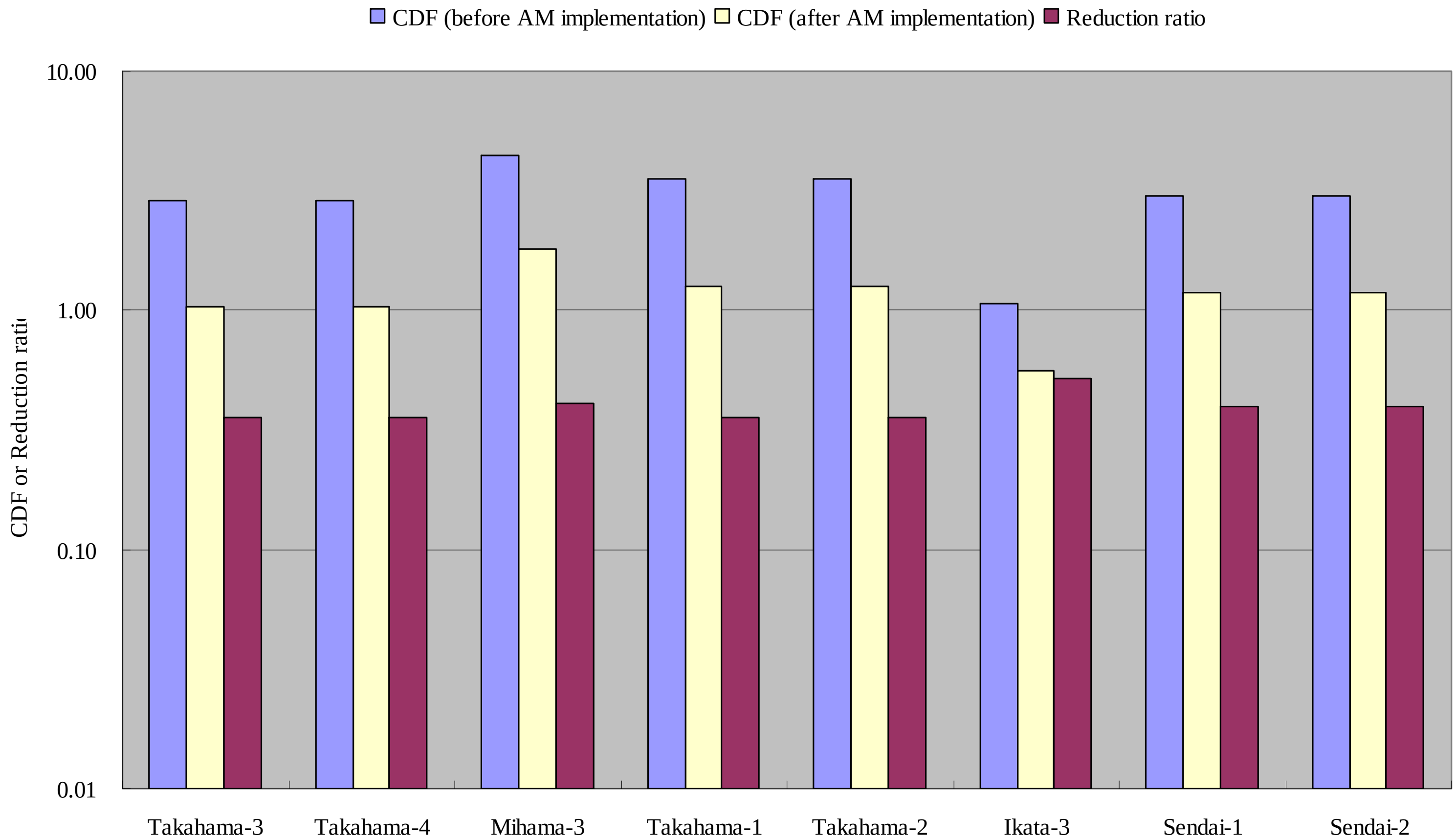
- Some variations of CDFs and CFFs can be found in the same plant type. There are some small differences in the design and operation of plants and AM measures adopted.
  Example: CDF variation due to the design and operation of CCWS in type C plants.

CDF results before and after AM implementation (PWR)



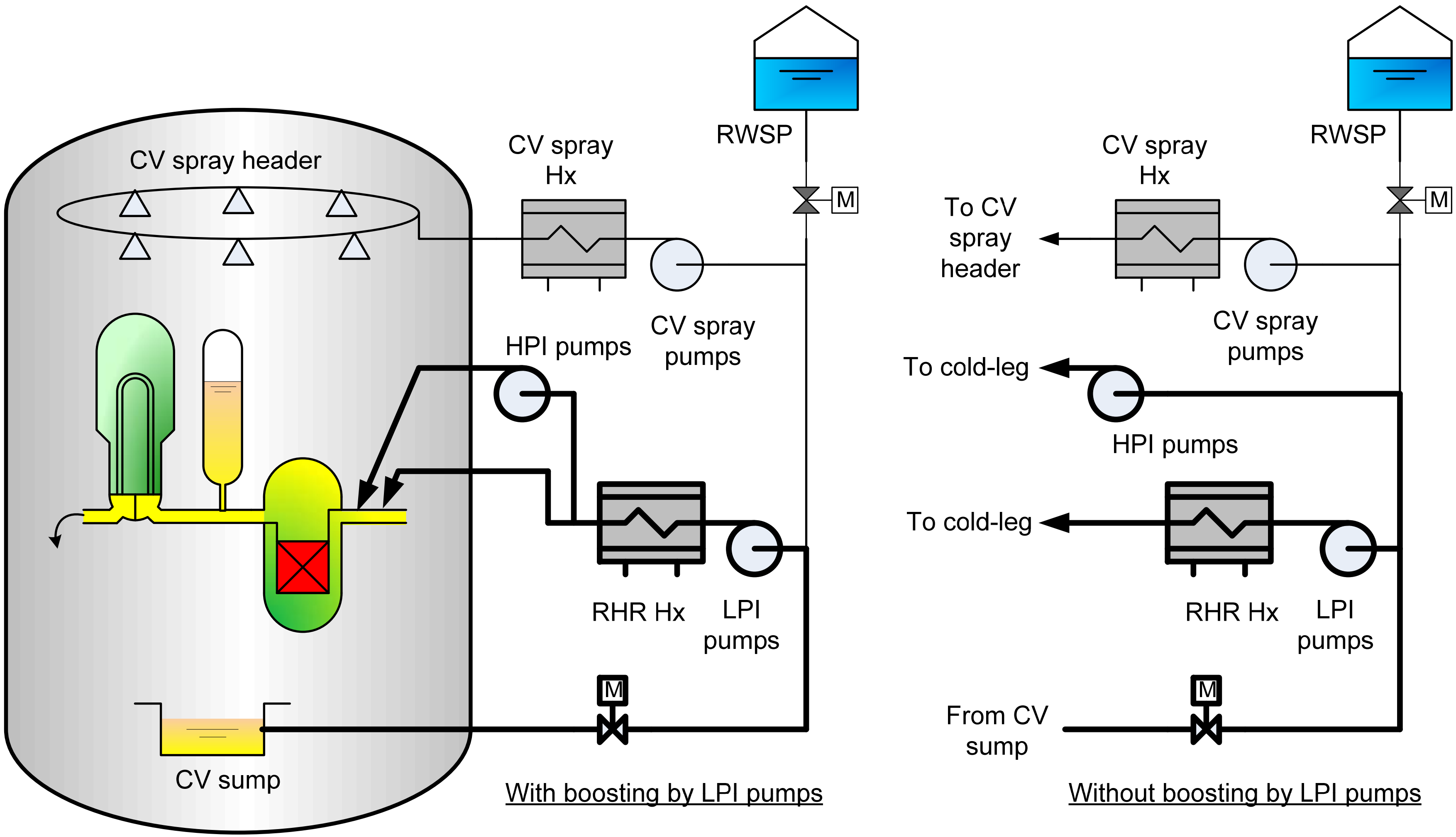
CFF results before and after AM implementation (PWR)

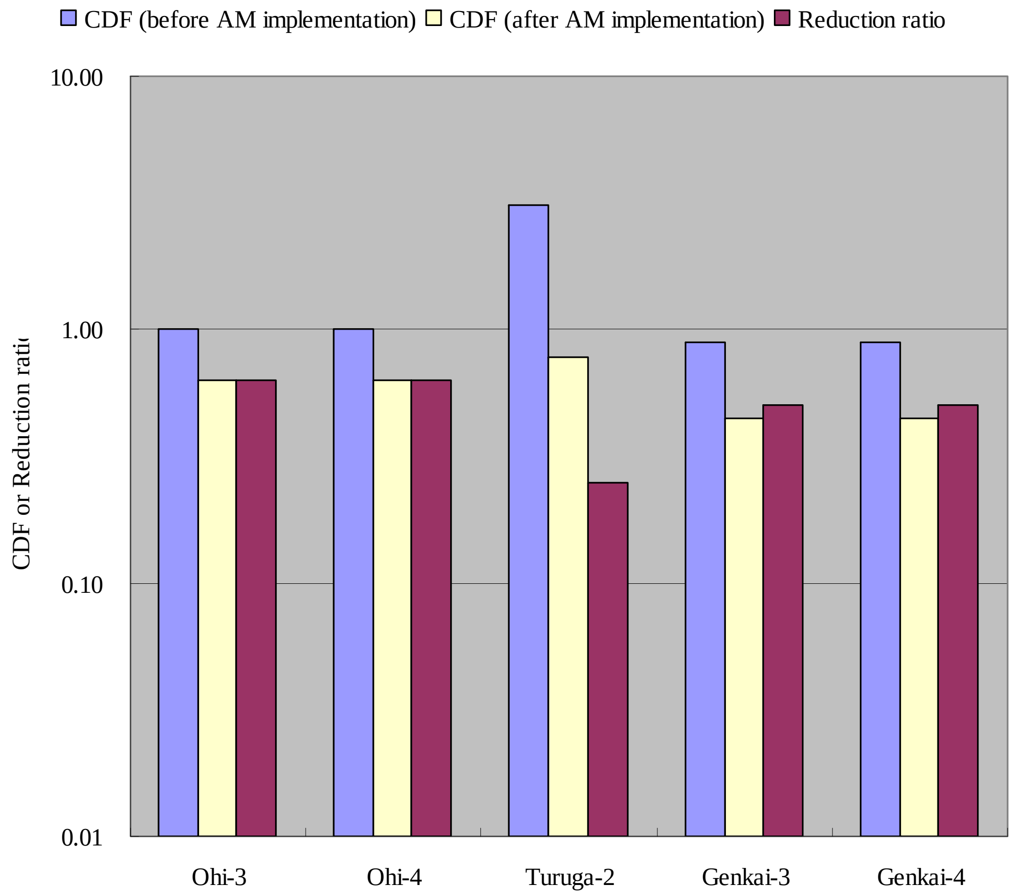




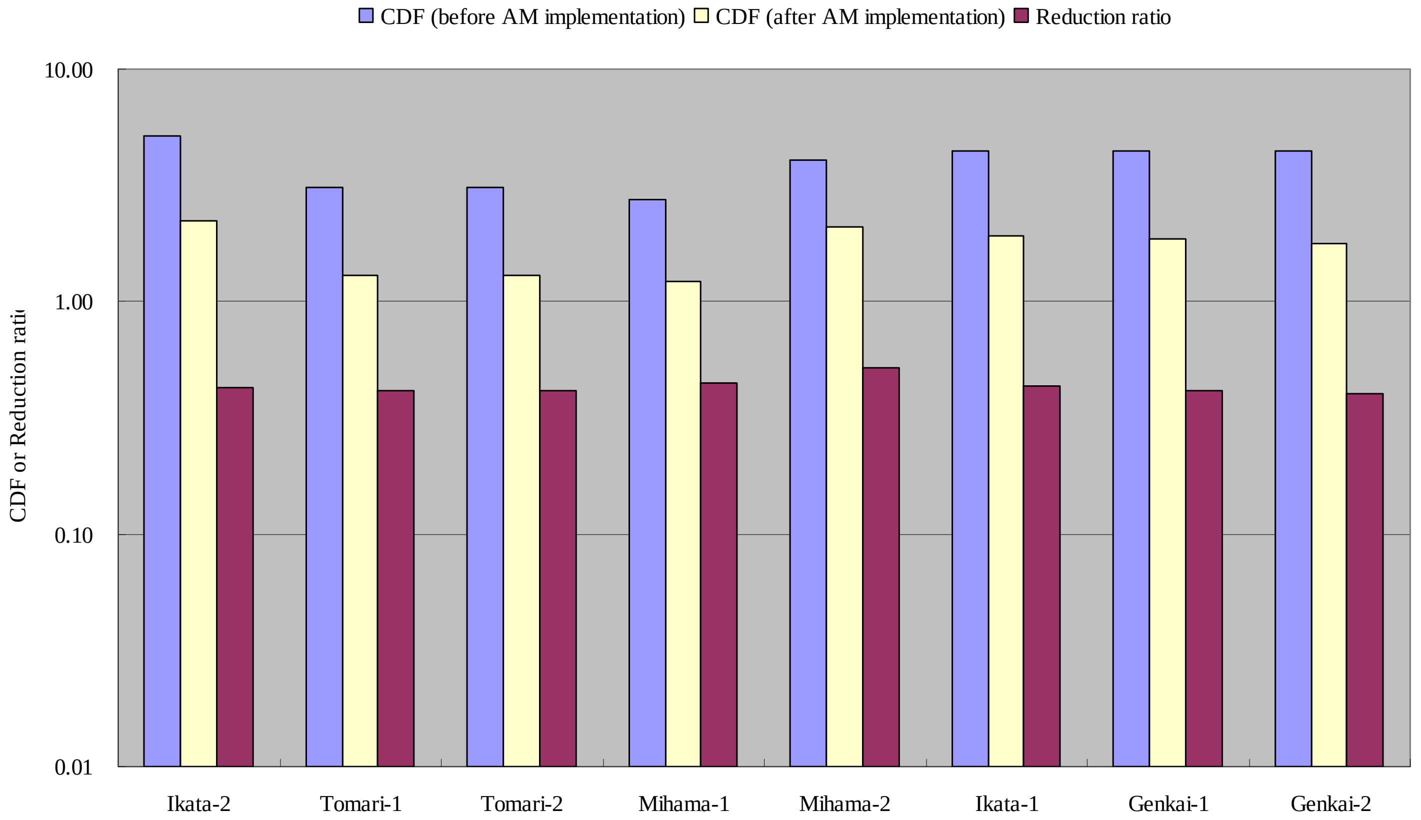
- CDF of Ikata-3 in type B group is much smaller than CDFs of other NPPs in the same group.
- In Ikata-3, the high pressure injection (HPI) pumps do not require boosting by the low pressure injection (LPI) pumps during ECCS recirculation mode while the other NPPs in the same group require boosting by 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.

HPI Pump Boosting by LPI Pump (PWR)





- Turuga-2 is the only one plant which needs boosting by LPI pump to HPI pump in type D group, which makes CDF of Turuga-2 before AM implementation greater than the other.
- In contrast, two cross-ties between LPI and CSI are used for Turuga-2, comparing one cross-tie between LPI and CSI for the others, makes small reduction ratio of Turuga-2, i.e. large AM effect.



- Another example can be find 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 for other NPPs of type A group. This design difference makes CDFs of Tomari-1 and 2 smaller than CDFs of the other plants in type A group.

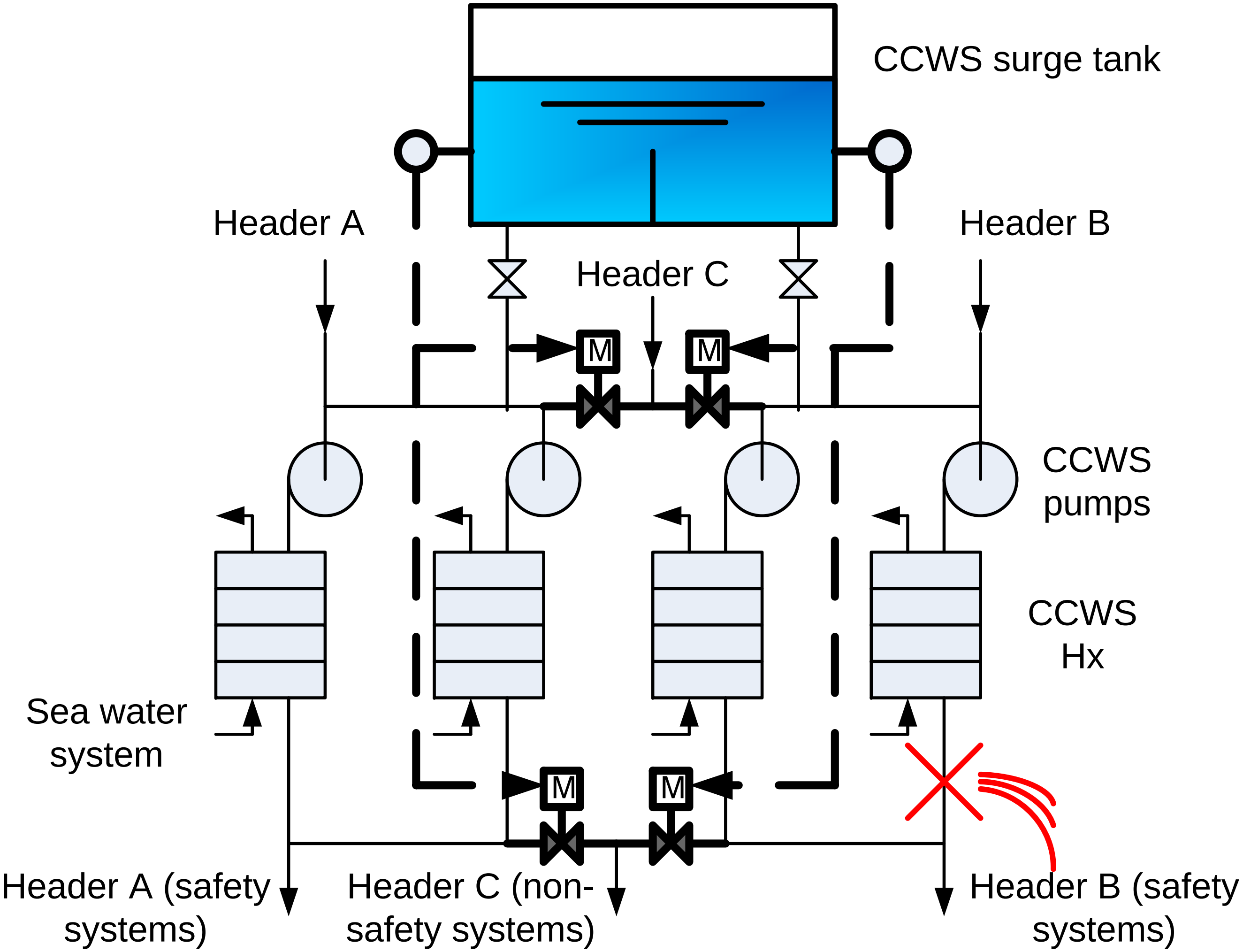
3. Accident management measures implemented to the recent NPPs

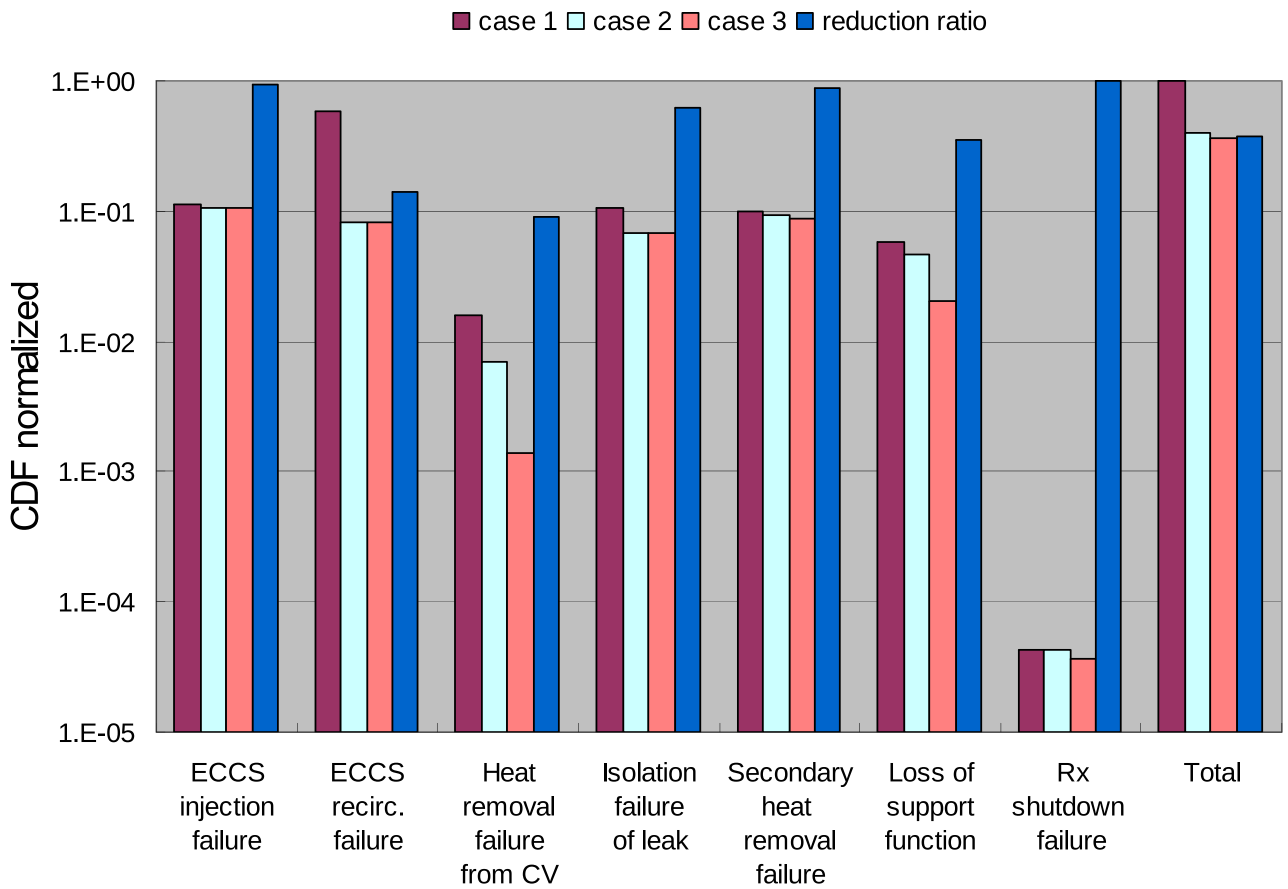
- For the newly constructed NPPs which begin commercial operation in 2002 or later, it is recommended by the NSC to establish an AM implementation plan before the first fuel loading to the core and submit the plan to the regulatory body for review.
- According to this process, AM measures for Higashidori-1, Hamaoka-5, Shika-2, and Tomari-3 have been investigated and reported to NISA until now. The results were reviewed by NISA with technical support of JNES and reported to the NSC.
- Among them, 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.
- Similar AM measures to the operating plants are used for Tomari-3, but some of them, i.e. train separation of CCWS actuated by a low CCW surge tank level against loss of CCWS function, and redundant intake lines from CV recirculation sump are incorporated as a part of basic design of the plant.

AM case studied (Tomari-3)

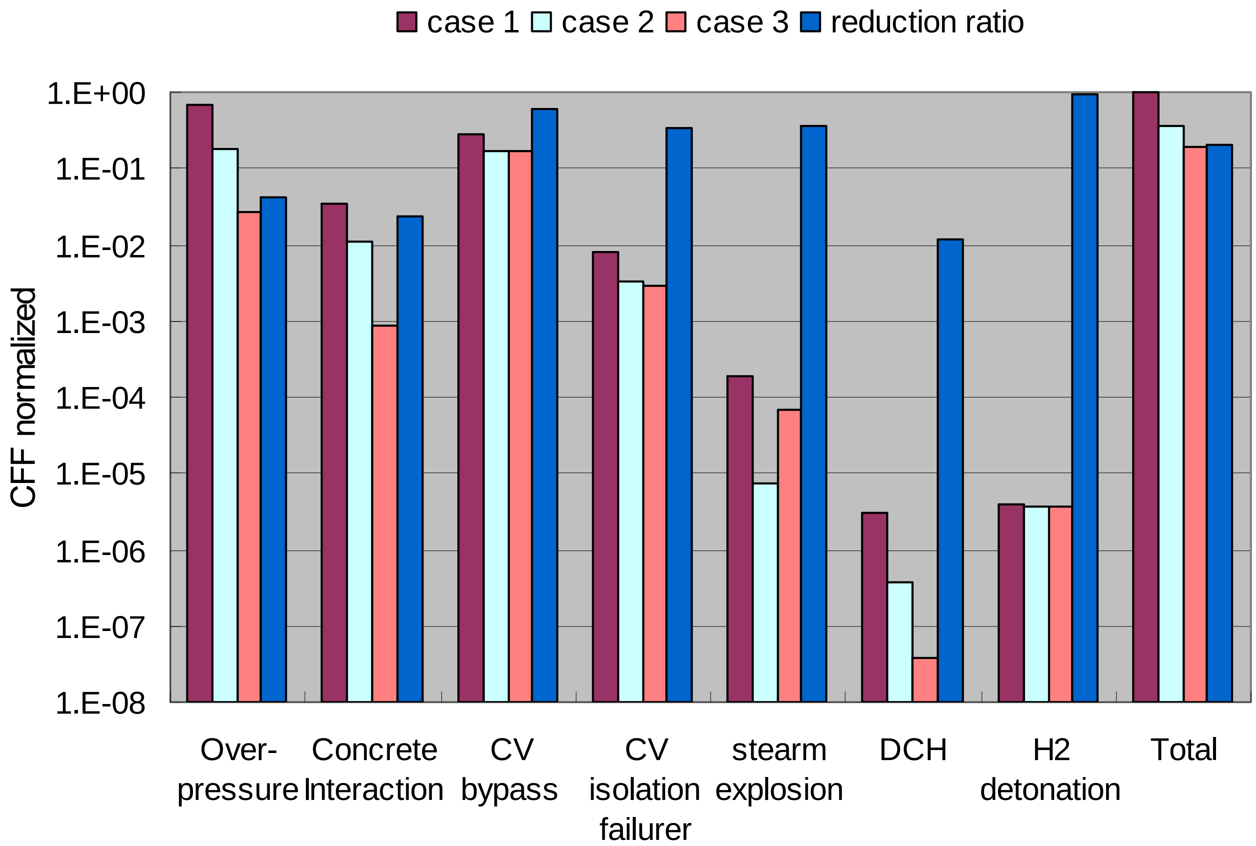
Case		Conditions for analysis
1	Base case	Basic design (without automatic CCWS train separation, alternative recirculation)
2	Basic design with AMs by operation manuals	Basic design (with automatic CCWS train separation, alternative recirculation) AMs by operation manuals (no hardware modifications) ・ Use of turbine bypass system ・ Cooldown and recirculation ・ Forced RCS depressurization
3	All AMs implemented	with AM measures ・ Natural convection cooling in CV ・ Coolant injection to CV ・ Electric power supply from adjacent unit

Component cooling water system automatic isolation





- CDFs are normalized by the total CDF of case 1.
- Overall reduction ratio of CDF, i.e. case 3 vs. case 1, is 0.37, whereas ratio of case 1 vs. case 2 is 0.41. Most of these reduction is accomplished by the adoption of alternative recirculation and automatic CCWS train separation.
- Failure of ECCS recirculation, failure of heat removal from CV, and loss of support function are reduced by the installation of alternative recirculation, natural convection heat removal, and automatic CCWS train separation.



- CFFs are normalized by the total CFF of case 1.
- Overall reduction ratio of CFF, i.e. case 3 vs. case 1, is 0.20, whereas ratio of case 1 vs. case 2 is 0.37. The latter is almost equal to the reduction ratio of CDF.
- Overpressure, Concrete interaction, and DCH are reduced by installation of natural convection heat removal, coolant injection to CV, and forced depressurization.

Basic requirements for AM

AM implementation plan is reviewed from the following points;

- Basic requirements to develop AM measures
  - Organization to execute AM measures
    Organization, Roles of related divisions, Person in charge
  - Development of infrastructure
    Preparation of facilities and equipments used by technical support center, Availability of instrumentations
  - Establishment of knowledge base
    AM manuals for operators and technical support center, Understanding of plant condition, Decision to execute AM measures
  - Communication with the outside of the plant
  - Education and training of the staffs
- Effectiveness of AM measures evaluated by PSA
- Impact to the original safety functions
  No interfering with the intended original safety functions by implementing AM measures

Related future issues to AM

- Reconsideration of the treatment of AM in the nuclear safety regulatory framework
- Efficient way of AM development
- Improvement of quality of PSA used for evaluation of the effectiveness of AM measures
- Characteristics of PSA used for AM development
- Consideration of external events
- Public communication on AM measures

4. Conclusions

- 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 execution framework and the preparation of the relevant AM procedures had been completed by 2002. The effectiveness of AM measures was evaluated by utilities and results of these evaluations were reported to the regulatory body. The effectiveness of AM measures was conformed through the reviews on these reports by the regulatory body.

4. Conclusions (cont'd)

- It was recommended to establish AM measures and to complete installation of AM measures by the first fuel loading to the core for the newly constructed NPPs. Up to now, AM plans for four newly constructed plants were studied and reviewed in this process. In some cases, AM measures were 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 measures for external events, and others.

IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

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

Reactor Safety Division

OECD/NEA Workshop on Implementation of Severe Accident Management Measures– Oct 2009 – Switzerland

1 – Introduction

- **Since the 1990's, Severe Accident Management Guidelines have been developed in France by EDF to help the PWR plant operators and emergency teams in limiting the consequences of any postulated severe accident.**

- **Severe accident knowledge, codes, PSA, methods, are still making progress ...**

- **The presentation provides some views on the current situation for the French PWR in operation**

Content

1. **French PWRs in operation**
2. **Existing severe accident measures on operated PWRs**
3. **A new tool for the safety regulation: the severe accident safety standard**
4. **Severe accident risk quantification and reduction – Present and future activities for the PWR severe accident management**
5. **Towards some higher requirements in relation with plant life extension?**

1 – French PWRs in operation

Table 1 – Some features of the French PWRs in operation

	900 MWe PWR	**1300 MWe PWR**	**1450 MWe PWR**
Started	1977 - 1984	1984 - 1992	1996-1998
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

1 – French PWRs in operation

PARS

Design P : 5 bar

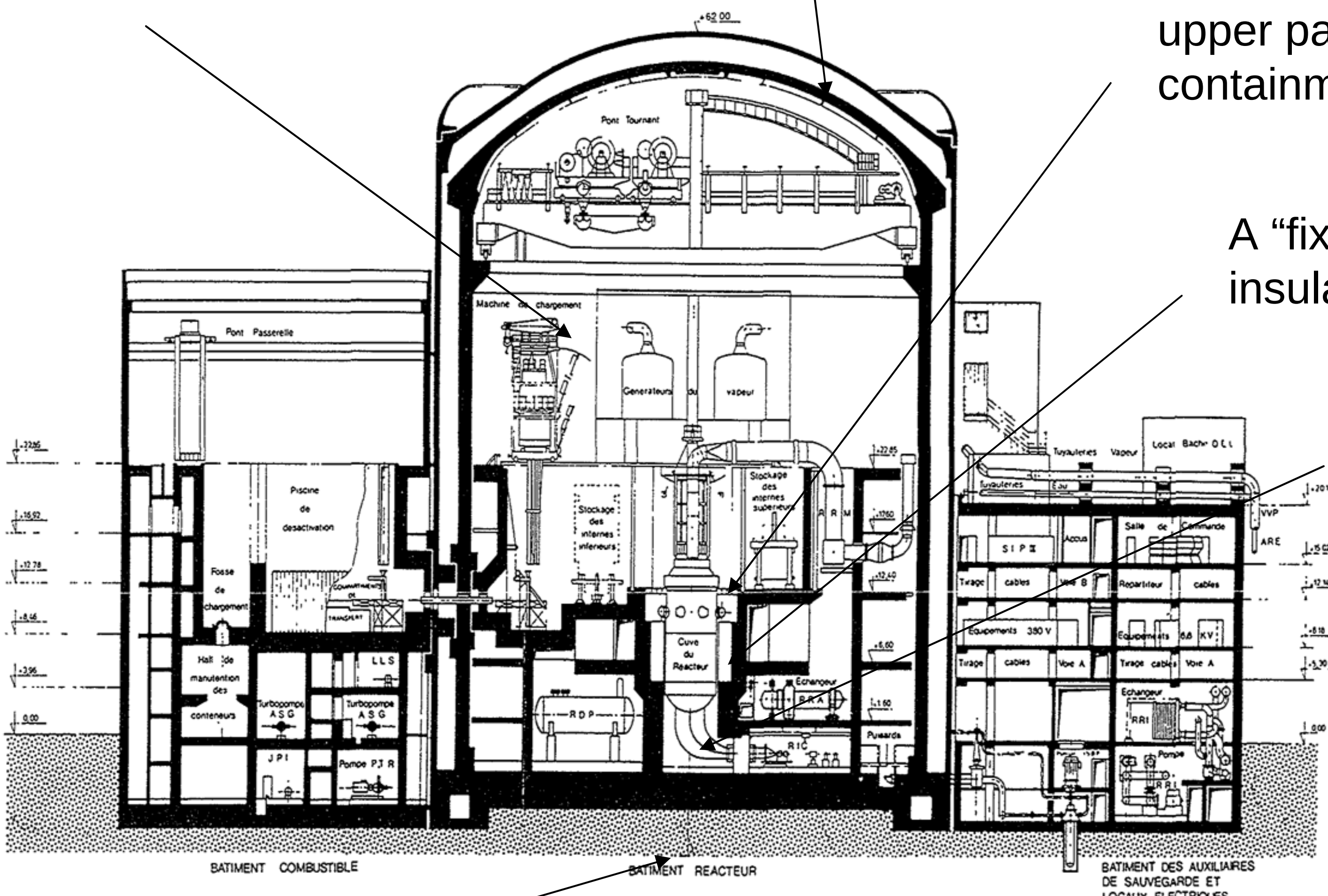
Connection between reactor cavity and upper part of containment

A “fix” vessel insulator

No draining of water in the cavity

Diversity of water injection possibilities

No Core catcher …



2 - Existing severe accident measures on operated PWRs

Part 1 -Some examples of key systems

Severe accident management guidelines

2.1 - Key systems : Reactor Containment building

- **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).**

- **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 analyses 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.**

2.1 - Key systems : Reactor Containment building

- **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 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).**

2.1 - Key systems : Pressurizer safety valves

- **Like all PWRs, 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.**

2.1 - Key systems : Containment venting system

- **A containment venting system has been installed on all French PWRs 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.**

2.1 - Key systems : 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).**

2.1 - Key systems : 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).**

2.1 - Key systems : 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 thermocouple 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.**

2.1 - Key systems : 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.**

2.1 - Key systems : Containment 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 (including case of SA)**
- **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).**

2.1 - Key systems : 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.**

2 - Existing severe accident measures on operated PWRs

Part 2 -Severe accident management guidelines

2.2 – Severe accident management 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).**

- **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.**

2.2 – Severe accident management guidelines

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;

2.2 – Severe accident management guidelines

- **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.**

3 - A new tool for the safety regulation: the severe accident safety standard

3 - A new tool for the safety regulation: the severe accident safety standard

BACKGROUND

- **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**

In that context 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.

3 - A new tool for the safety regulation: the severe accident safety standard

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.

3 - A new tool for the safety regulation: the severe accident safety standard

- **The last review by IRSN and positions of the "French Permanent Group" has conducted the Safety Authority to ask for some complements 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.**

4 - Severe accident risk quantification and reduction – Present and future activities at IRSN

4 - Severe accident risk quantification and reduction – Present and future activities at IRSN

- **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.**

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

- **The frequency of the heterogeneous dilution sequences remain relatively high, considering the potential associated impact of such accident …**

- **The calculated frequency of the loss-of-containment-integrity sequence after a steam explosion in the reactor pit appears relatively high. Additional studies regarding induced loads and containment strength under this type of loading seem to be necessary**

- **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;**

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

- **Certain sequences will be re-examined in detail :**
  - situations leading to high vessel pressure and 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;
  - situations 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);

- **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 reminds 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 closure system).**

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

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.

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.

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

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?

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

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 reflooding is now in progress in ICARE-CATHARE then ASTEC V2 codes, supported by the experimental PEARL programme;
- 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 (interest from 1300 MWe PWR L2 PSA, because the reactor cavity is connected to a corridor increasing significantly the corium spreading area). Some modelling efforts have been planned 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.

4.3 The source term assessment

In France, the emergency preparedness (distances for counter-measures applications) was consistent with 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.

In progress

- The integration of the results of the ISTP programme in the basic assumptions for the source term calculation (either in ASTEC code or in the very fast-running release code of L2 PSA) (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 accident measures may be examined to limit as far as possible the amplitude of the release.

5 - Towards some higher requirements in relation with plant life extension?

5 - 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.

5 - Towards some higher requirements in relation with plant life extension?

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 coexist 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.

Mitigation of the consequence of a severe accident is considered as a key issue.

For example, in the framework of plant life extension, the current difficulties on topics like water injection will have to be solved.

5 - Towards some higher requirements in relation with plant life extension?

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.

6 Conclusions

6 Conclusions

- **Progresses have been achieved with practical implementations of severe accident measures.**
- **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 measures allowing a better control of accident consequences.**

Thank you for your attention !

Session 2

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Protecting People and the Environment

Perspectives on Severe Accident Mitigation Alternatives for US Plant License Renewal

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US Nuclear Regulatory Commission
(presenter)*

OECD Severe Accident Management Workshop
Bottstein, Switzerland
October 26-28, 2009

Presentation Outline

- Historical context and regulatory basis
- Definition and scope
- Major steps in a SAMA evaluation
- Current status of SAMA reviews
- Insights from SAMA evaluations
- Potentially cost-beneficial SAMAs
- Conclusions / information availability

Quick terminology note

- SAMDA = severe accident mitigation design alternative
- SAMA = severe accident mitigation alternative

- Only the application is different, the process/scope is the same

Historical context and regulatory basis

- 1980 severe accident interim policy statement
  - Identify additional cases where additional features would prevent/mitigate severe accident consequences
- 1985 severe accident policy statement
  - No present basis for generic rulemaking or other regulatory changes due to severe accident risk
  - Nevertheless, perform analysis to discover instances of vulnerability to core melt or unusually poor containment performance

Historical context and regulatory basis (2)

- 1989 court decision
  - SAMDA required for plant operation
- NRC gained SAMA experience through:
  - SAMDA evaluations for Limerick, Comanche Peak and Watts Bar
  - Containment performance improvement program
  - Individual plant examinations (IPEs) and Individual plant examinations: external events (IPEEEs)
  - Implementation of severe accident management programs (US industry initiative)

Definition and scope

- SAMA = A feature or action that would prevent or mitigate the consequences of a severe accident
- Includes:
  - Hardware modifications, procedure changes, and training program improvements
  - Prevention and mitigation
  - Both internal and external events

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Major steps in a SAMA evaluation

1. Leading contributors to risk
   - Use plant-specific risk study or equivalent
   - External events considered to the extent practicable
2. Identify candidate SAMAs
   - Identify SAMAs, including low-cost ways of achieving functional objective
   - Use of PRA importance measures to identify important basic events
   - Utilize relevant past SAMA evaluations

Major steps in a SAMA evaluation (2)

3. Risk reduction / implementation cost estimates
   – Calculate maximum attainable benefit (MAB)
   – Perform benefit assessment and cost assessment
   – Screen out SAMAs that can't be cost-beneficial
   – Assess effects of uncertainties
4. Potentially cost-beneficial SAMAs
   – Estimate net value of SAMA (averted costs – cost of enhancement)
   – NUREG/BR-0058 and NUREG/BR-0184

Averted Cost Values

For completely eliminating internal events

Cost Factor	Significance	NUREG/ BR-0184 Section	Related Parameters	Average (and Ranges) of MAB from Submittals for All Approved License Renewals
APE	Offsite exposure	5.7.1	Δperson-Sv (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) Long term occupational dose (200 person-Sv)	\$17K (\$1K – \$130K)
ACC	Onsite economic	5.7.6.1	Onsite cleanup and decontamination cost (\$$1.1 \cdot 10^9$ single event, present worth)	\$870K (\$37K – \$6,300K)
ARPC	Onsite economic	5.7.6.2	Plant power level	
Total				\$1,700K (\$110K - \$8,700K)

Major steps in a SAMA evaluation (3)

5. More detailed analysis for remaining SAMAs
   - More realistic evaluation of benefits
   - More detailed implementation cost development
   - Nuclear Energy Institute document NEI-05-01, Revision A
     - endorsed by NRC Interim Staff Guidance LR-ISG-2006-03

Current status of SAMA reviews

- Completed SAMDA evaluations for 3 sites during initial licensing in 1989-1995
- Completed SAMDA evaluations for multiple advanced light-water reactors
- Completed SAMA evaluations for > 50 units for license renewal, including:
  - All BWR containment/NSSS types in US, except Mark-III / General Electric Type 6
  - All PWR containment/NSSS types in US

Insights from SAMA evaluations

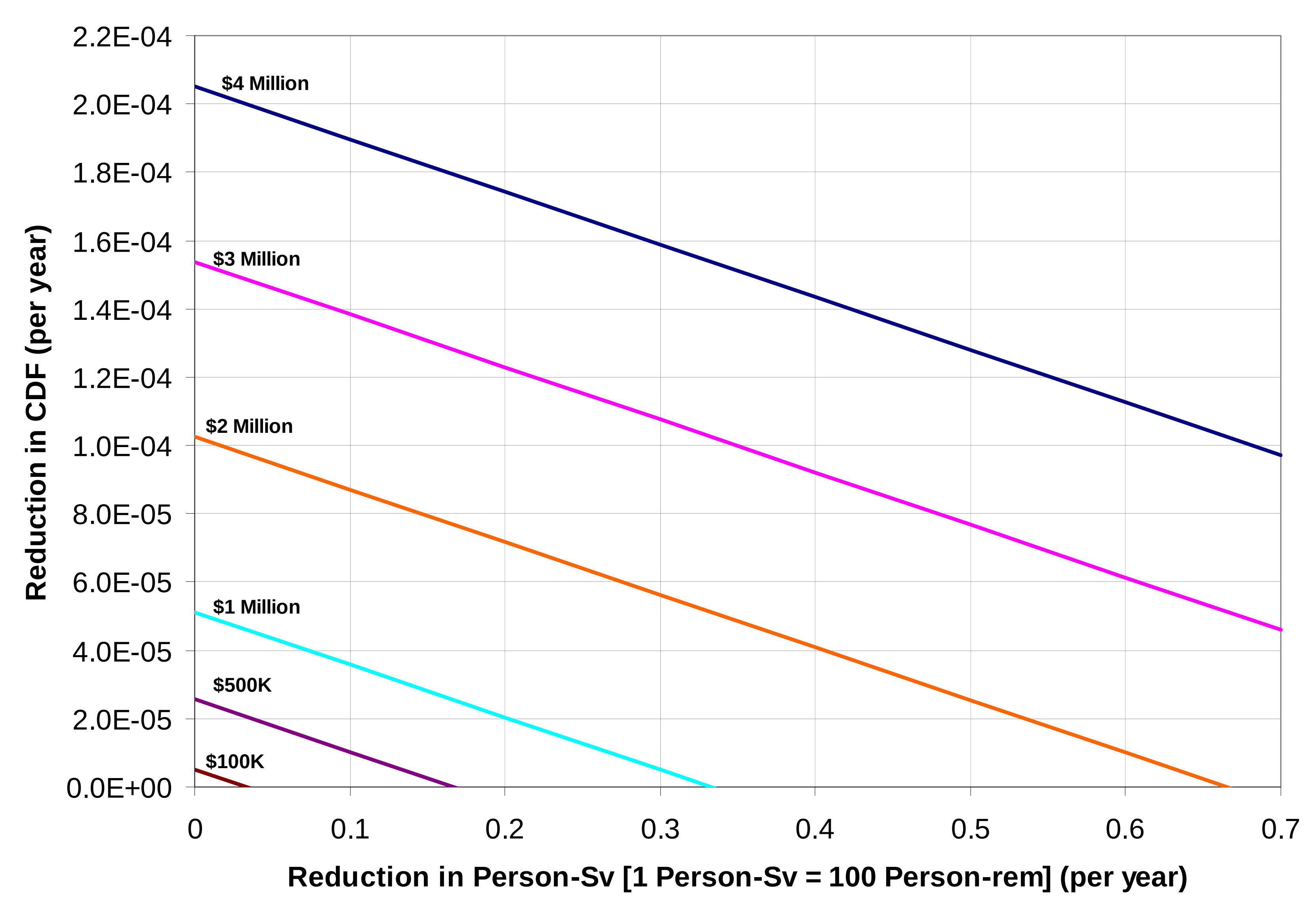
- Considerations:
  - CDFs from operating plants are relatively low
  - Past programs have addressed known weaknesses
  - SAMAs typically only act on one contributor, while risk is generally driven by multiple contributors
  - Implementation costs are high for design retrofits
  - Residual risk for advanced reactors is very low
- Therefore
  - It is difficult to identify additional changes that substantially reduce risk and are cost-beneficial
  - Cost-beneficial changes usually limited to procedural changes and limited hardware changes
  - Averted onsite costs are important – promote preventative SAMAs

Risk Reduction Values

For completely eliminating internal events

	Average	Ranges
CDF (/yr)	4.0×10^{-5}	$1.9 \times 10^{-6} - 3.3 \times 10^{-4}$
Population Dose (person-Sv/year)	0.15	0.006 – 0.69
$/event	$2.8 billion	$49 million – $12 billion
$/person-Sv	$220,000	$69,000 - $670,000
Total MAB	$1.7 million	$110,000 – $8.7 million

Typical Cost Benefit Threshold
3% Discount, 20 year term



Potentially cost-beneficial SAMAs

- Types of cost-beneficial SAMAs:
  - SAMAs related to SBO or loss of power sequences
  - SAMAs related to internal floods, fire, seismic and other external events
  - SAMAs related to protection systems
  - SAMAs related to support systems
  - SAMAs related to procedures and training

Potentially cost-beneficial SAMAs (2)

- Specific examples:
  - Procure an additional portable 480V AC station DG for backup to EDGs
  - For internal floods, install watertight doors/wall around vulnerable equipment
  - Provide an alternate/additional compressor that can be aligned to the instrument air system
  - Use firewater systems as backup for containment spray
- An extensive list of examples is provided in the associated paper

Conclusions / information availability

- PRA has been used to identify numerous cost-beneficial improvements
- PRA importance measures play a key role in this process
- Typically low cost improvements (e.g., procedure modification) are found to be more cost-beneficial

- Information related to all aspects of license renewal, including licensee submittals and Environmental Impact Statements (which include SAMA analysis) is available at:

  http://www.nrc.gov/reactors/operating/licensing/renewal.html

Acronyms

- AC = Alternating current
- ACC = Averted cleanup and decontamination costs
- AOC = Averted offsite property damage costs
- AOE = Averted occupational exposure costs
- APE = Averted public exposure costs
- ARPC = Averted replacement power costs
- BWR = Boiling water reactor
- CDF = Core damage frequency
- COE = Cost of enhancement
- DG = Diesel generator
- EDG = Emergency diesel generator
- IPE = Individual plant examinations
- IPEEE = Individual plant examinations: external events
- LR-ISG = License renewal interim staff guidance
- MAB = Maximum attainable benefit
- NEI = Nuclear Energy Institute
- NSSS = Nuclear steam supply system
- PRA = Probabilistic risk assessment
- PWR = Pressurized water reactor
- SAMA = Severe accident mitigation alternative
- SAMDA = Severe accident mitigation design alternative
- SBO = Station blackout
- Sv = Sievert
- US NRC = US Nuclear Regulatory Commission

Effect of SAMG on the Level 2 PSA of KSNP

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Contents

- Severe Accident Policy in Korea
- Status of PSA and SAMG
- Outline of SAMG
- Effect of SAMG on Level 2 PSA

Severe Accident Policy in Korea

- MOST announced in August, 2001
- Main Points
  - Set up Safety Goal
  - Implement PSA for all operating plants
  - Confirm plant capabilities to cope with severe accident
  - Establish Severe Accident Management Program

KHNP Implementation Plan

- **PSA** : Completed in 2007 for all operating plant
  - Plants in operation on Sep. 2001 : Level 2 PSA
  - Plants in construction : Level 2 PSA including Shutdown PSA
    - YGN 5&6, UCN &,6, SKR 1&2, SWS 1&2
  - Advanced Plant(Shinkori 3,4) : Level 3 PSA
- **RIMS** : Risk Monitoring System
  - Completed in 2007 for all operating plant
- **Severe Accident Management Program**
  - Completed in 2007 for PWRs
  - Will complete in 2009 for PHWRs

Status of PSA and RIMS

Back ground	89	90	91	92	93	94	95	96	97	98	99	00	01	02	03	04	05	06	07
	Post-TMI actions		Requirements for CP/OL									severe accident policy implementation							
YEAR	**89**	**90**	**91**	**92**	**93**	**94**	**95**	**96**	**97**	**98**	**99**	**00**	**01**	**02**	**03**	**04**	**05**	**06**	**07**
K # 1												11 L2-N 11					6 L2-U1/RM 5		
K # 2														1 L2-N 12				1 L2-N /RM 12	
K # 3,4	9 L1-N 8											9 L2-U1/ RM 6							
Y # 1,2	9 L1-N 8													9 L2-U1 12				7 L2-U1/ RM 12	
Y # 3,4			4 L2-N 2											1 L2-U1/RM 5					
Y # 5,6						9 L2 SD-N 12										7 L2-U1/ RM 12			
U #1,2																1 L2-N/RM 12			
U #3.,4			8 L2 -N 10											1 L2-U1/RM 12					
U #5,6										7 L2 SD -N 6							1 L2-U1/ RM 6		
W #1													10 L2-N 12					10 L2-UI/RM 12	
W #2,3,4						7 L2 SD -N 6											1 L2-U1/RM 12		

* Legend : L1(Level 1 PSA), L2(Level 2 PSA), SD(Shutdown/Lower Power PSA), N(NEW), U(Update), RM(Risk Monitoring)

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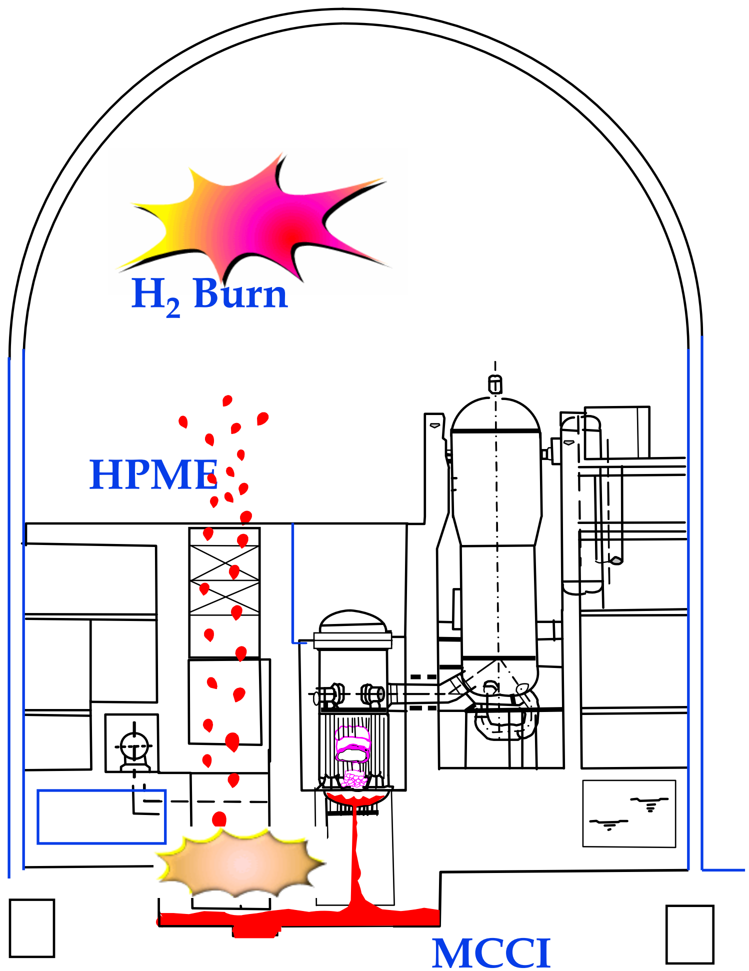
Severe Accident Management Program

	Period	Years									
		’00	’01	’02	’03	’04	’05	’06	’07	’08	’09
YGN 5,6	’99.12~’01.06										
YGN 3,4 UCN3,4,5,6	’02.04~’02.12										
Kori 1	’02.09~’03.12										
Kori 2,3,4 YGN 1,2	’03.01~’04.12										
UCN 1,2	’05.06~’07.05										
WS 1,2,3,4	‘08.01 ~ ‘09.12										
Training Program	’01.07~			C E		W H				U	PHWR
Implementatio	’02.04~										

Severe Accident Mitigation Features-OPR1000



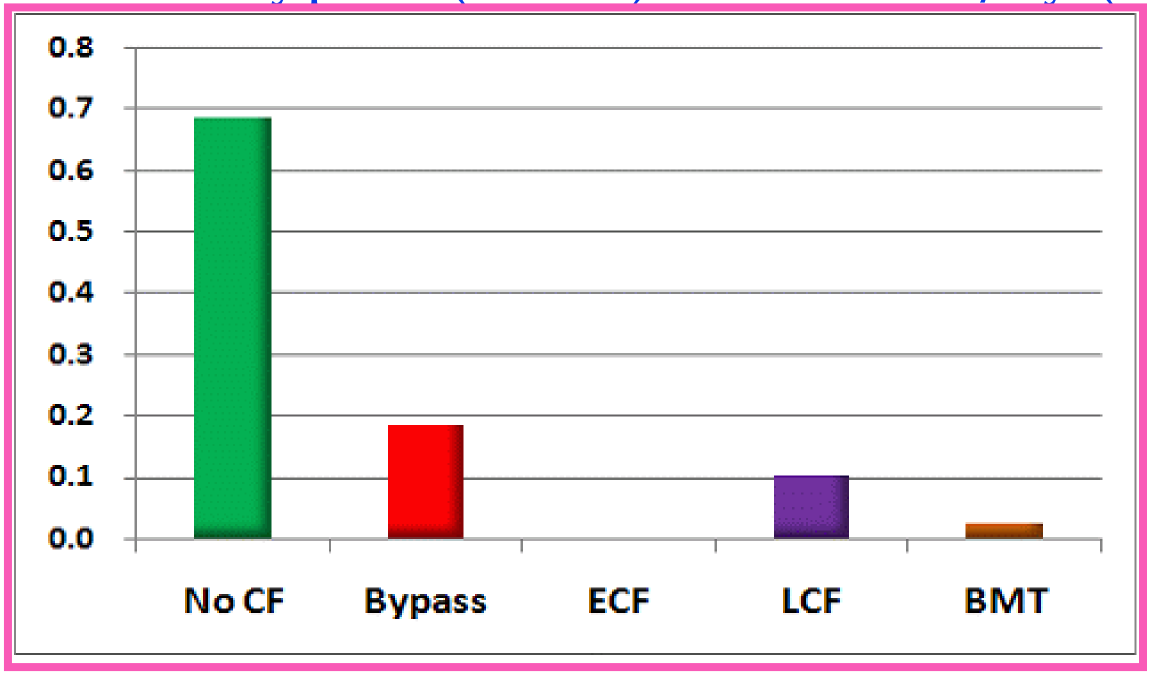
- Safety depressurization system
  - prevent DCH & TI-SGTR
- Hydrogen igniter
- Large cavity floor area
  - no dedicated cavity flooding system
- Long term containment cooling
  - spray
  - fan cooler
  - no alternating containment cooling equipment

Severe Accident Management Guidance of UCN 3&4

- Developed Based on WOG SAMG
- Guidelines
  - SACRG, Severe Accident Control Room Guideline
  - DFC, TSC Diagnostic Flow Chart
  - SAGs, Severe Accident Guidelines
    - SAG-01, Inject into the Steam Generators
    - SAG-02, Depressurize the RCS
    - SAG-03, Inject into the RCS
    - SAG-04, Inject into the Containment
    - SAG-05, Reduce Fission Product Release
    - SAG-06, Control Containment Condition
    - SAG-07, Reduce Containment Hydrogen
  - SAEGs, Severe Accident Exit Guidelines
    - SAEG-1, TSC Long Term Monitoring
    - SAEG-2, SAMG Termination

Result of UCN 3,4 PSA (2004)

- Core Damage Frequency : 5.30 x 10^{-6}/ry
- Containment Failure Frequency : 1.66 x 10^{-6}/ry
- Containment Bypass (SGTR) : 7.99 x 10^{-7}/ry (15% of CDF)



Reevaluation of UCN 3,4 PSA

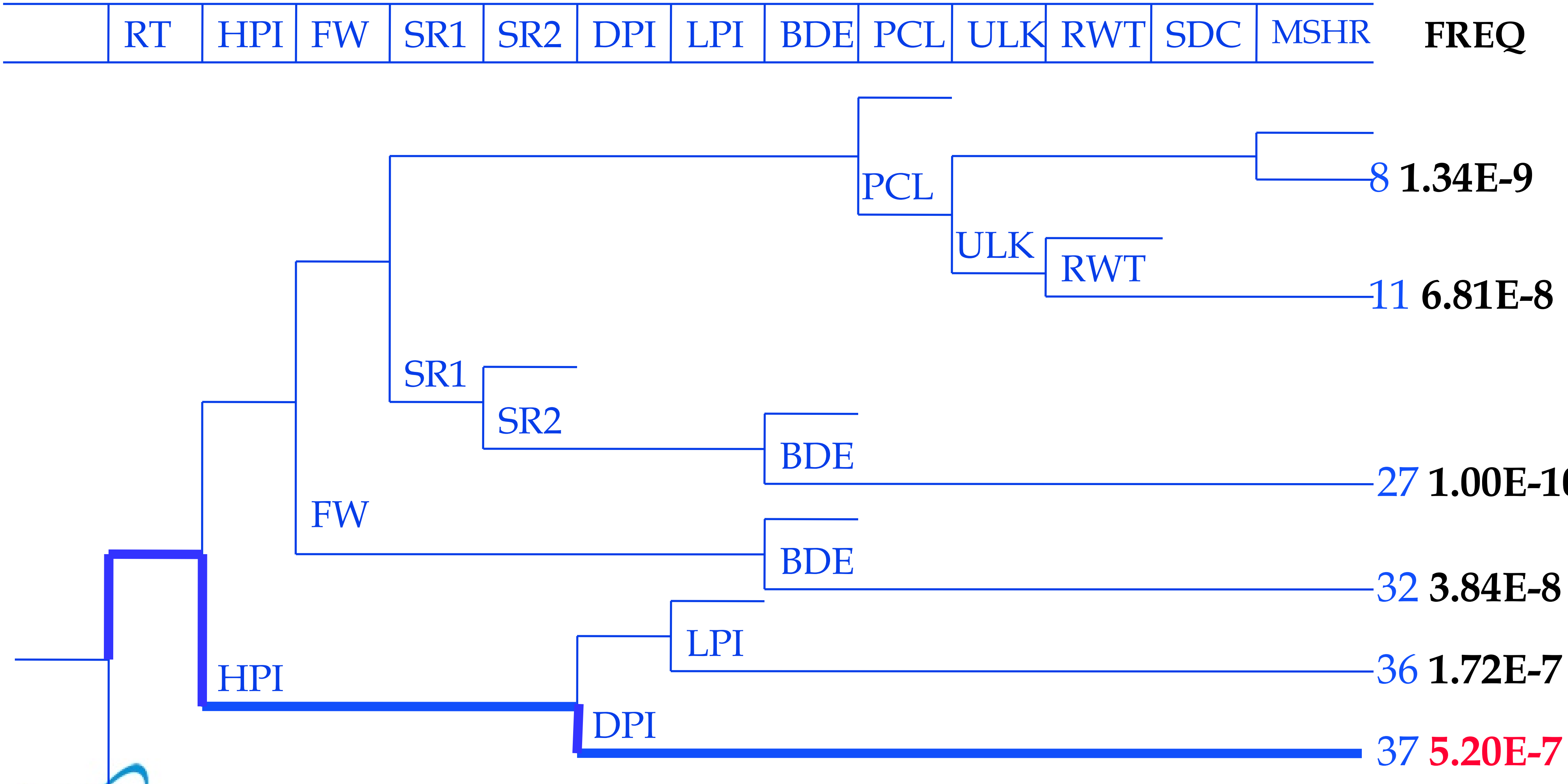
Reevaluation of bypass frequency

- Reevaluate sequence 37 of SGTR
  - Propose revision of Emergency Operation Procedure
  - Evaluate HEP base on proposed EOP

Reevaluation of late containment failure frequency

- Consider SAMG
  - Recovery of containment spray system
  - Reactor building fan cooler

SGTR Event Tree



Frequency of SGTR-37

Evaluation of operator available time

- Computer code : MARS
- RCS cooldown rate :100 ^{O}F/hr
- operator available time : about 40 minute

Evaluation of HEP

- Assumption :
  - RCS depressurization procedure is described clearly in EOP when HPI fails
  - Operator is trained for this procedure
  - Operator available time : 30 minute
- HEP : 0.0256 (cf.0.59)

Frequency of SGTR-37 : 2.734 x 10^{-8}/ry (cf. 5.20E-7)

Result of HEP Analysis

	Before EOP revision			
Available Time	30 min.	40 min.	50 min.	60 min.
MMI	Medium	Medium	Medium	Medium
Procedure	Low	Low	Low	Low
Training	Low	Low	Low	Low
HEP	5.90E-1	3.05E-1	2.39E-1	6.80E-2

	After EOP revision			
Available Time	30 min.	40 min.	50 min.	60 min.
MMI	Medium	Medium	Medium	Medium
Procedure	Medium	Medium	Medium	Medium
Training	Medium	Medium	Medium	Medium
HEP	2.56E-2	1.42E-2	1.16E-2	5.44E-3
Ratio	0.043	0.046	0.048	0.08

Reevaluation of Late Containment Failure Frequency

Recovery of Spray System

- SAG-06 "Control Containment Condition"
  - Step 1 identifies the availability of containment spray system
  - If spray system is unavailable, identify the reasons why containment spray system are not available and restore a containment spray system
- Spray pump takes the longest time to restore among components in spray system
- 47 hours are required to disassemble and assemble spray pump
- Late containment failure frequency is evaluated 72 hours after accident initiation
- Assign 0.9 for probability of spray system restoration

Use of Fan Cooler

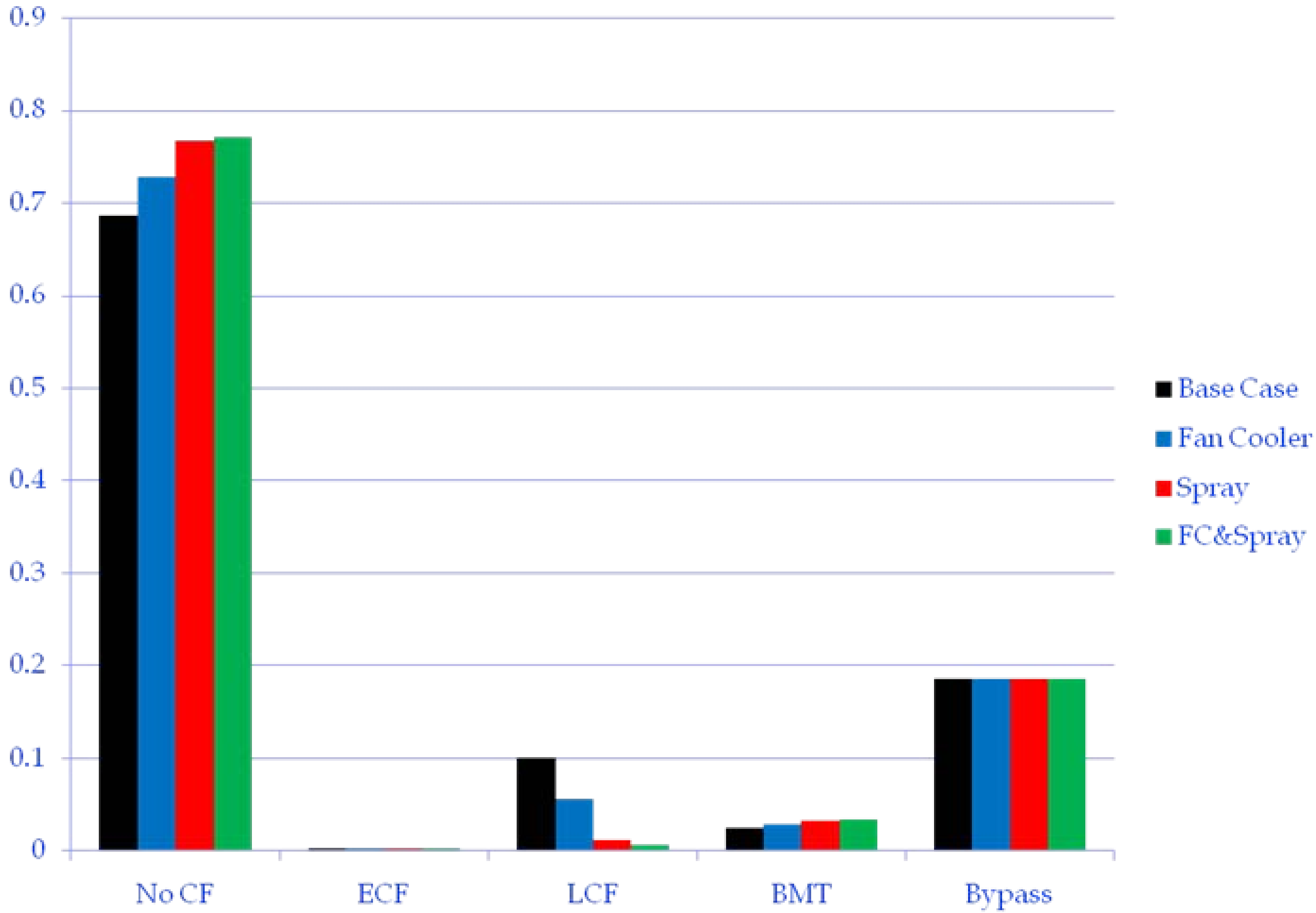
- Fan cooler is non-safety grade
- SAG-06 allows the use of non-safety grade equiment
- Failure rate of fan cooler under normal operating condition : ~ 10^{-3}/ry
- Increase of failure rate of fan cooler under severe accident condition is expected
- Use of fan cooler just after failure of sprat prevents high pressure and high temperature
- Assign 0.5 as failure rate of fan cooler under severe accident condition

Late Containment Failure Frequency

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 Meltthrough	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**

* Fraction of the total frequency

Change of Containment Failure Frequency



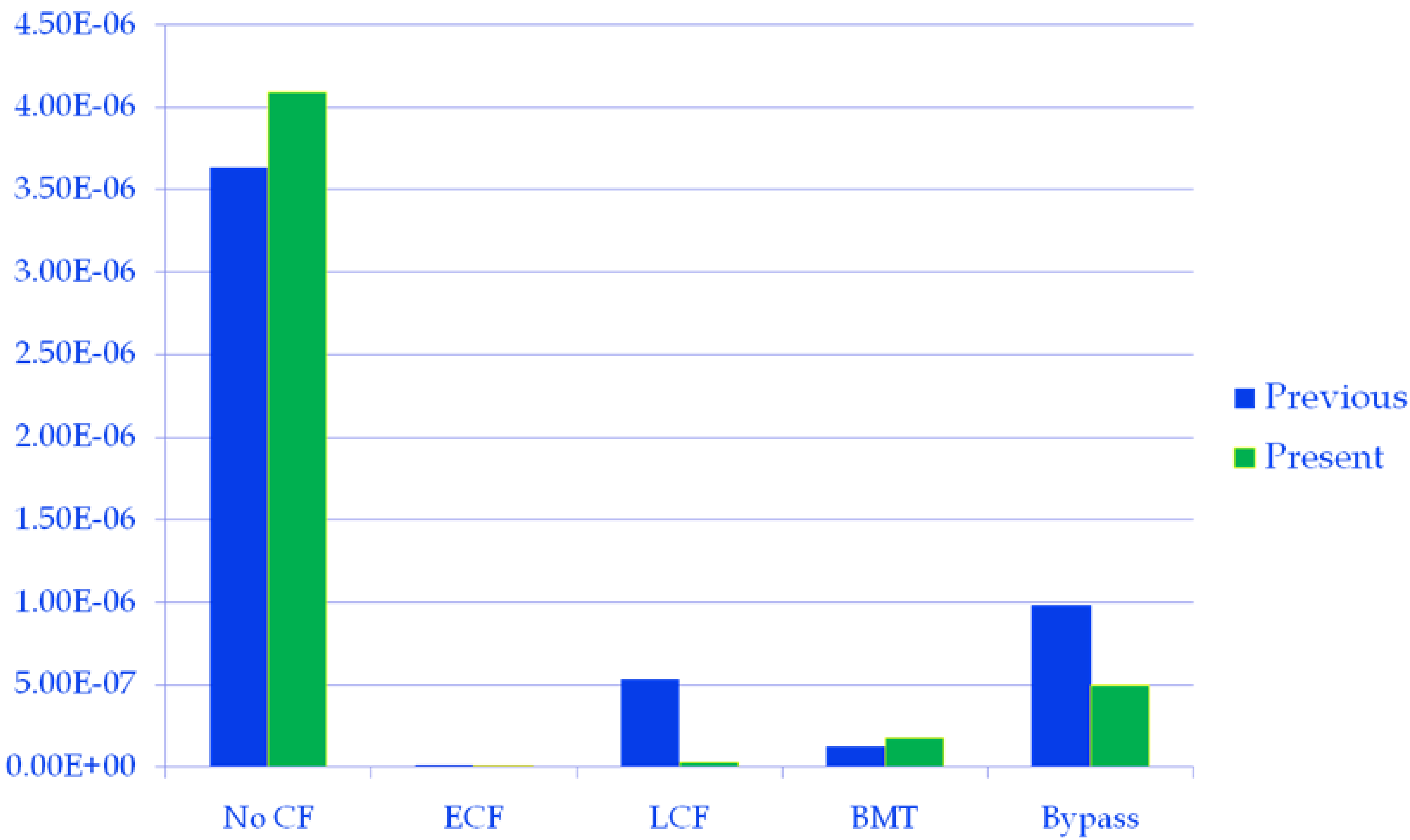
한국원자력연구원
KAERI
Korea Atomic Energy Research Institute

Result of Level 2 PSA

- Bypass Frequency
  - Previous : 0.186
  - Present : 0.093

- Late Containment Failure Frequency
  - Previous : 0.101
  - Present : 0.006
- Basemat Meltthrough Frequency
  - Previous : 0.024
  - Present : 0.034

Result of Level 2 PSA



Summary & Conclusion

❑ EOP Revision (proposed)

- ❖ Assumed that RCS depressurization procedure is described clearly in EOP when HPI fails
- ❖ Frequency of SGTR-37 sequence reduced
- ❖ Frequency of Bypass reduced
- ❖ Revision of EOP is important in the reduction of containment bypass frequency

❑ SAMG

- ❖ Restoration of spray system and use of fan cooler considered
- ❖ Frequency of late CF reduced very much
- ❖ Frequency of basemat meltthrough increased slightly
- ❖ SAMG is very effective on the prevention of late containment failure

Insights from a full-scope Level 1/Level 2 all operational modes PRA with respect to the efficacy of Severe Accident Management actions.

Klügel, J.-U., NPP Gösgen
Rao, S.B., Mikschl, T., Wakefield, D., ABSG Consulting Inc.
Torri, A., Pokorny, V. , RMA

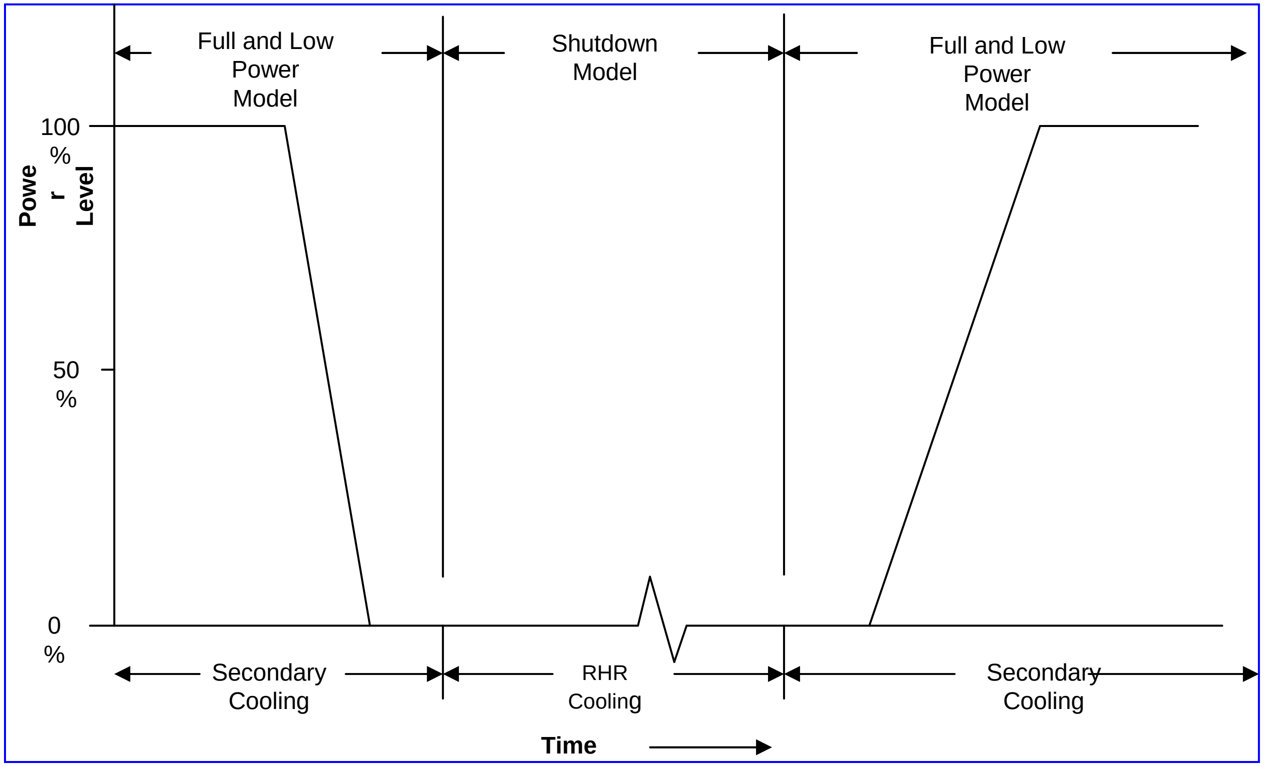
Overview

- Introduction
- Scope and Structure of the Goesgen PSA model
- Main results of the Goesgen PSA (full power, shutdown)
- Insights gained from the analysis of the results
  - Source term analysis for shutdown states
  - Importance of post accident SAM actions and of physical phenomena
- Summary and conclusions

Introduction

- NPP Goesgen is a 3-loop PWR (KWU-design) with
  P_e =1002 MW
- Commercial operation since 1979
- Integrated Emergency management since 2005 (AM +SAM integrated)
- First PSA (level 1/level 2- external events, shutdown) completed in 1994
- Periodic updates of seismic PSA – 1994, 2001/2003 and 2005/2006 (full power) as well as for shutdown conditions (2002, 2005);
- International Peer Review (2004/2005)
- Complete PSA upgrade 2008 as a part of the PSR

Scope of Goesgen PSA



Scope of Goesgen PSA

- All modes, all events integrated level1/level2 PSA study
  - 156 initiating events for power operation (including low power nn RHR-modes)
  - 173 initiating events for shutdown
- Three different outage modes
  - A repair, RCS closed
  - B repair, RCS open
  - C refueling outage
- Internal events subdivided in
  - LOCAs
  - Transients
  - SGTRs (including multiple ruptures and multiple leaks)
  - ATWS (failure of rod insert = failure of scram) for all transients, small LOCAs and SGTRs)

Scope of Goesgen PSA

- Internal hazards (explicit model)
  - Internal floods
  - Fires (more than 300 fire scenarios)
- External hazards (explicit model)
  - Airplane crash (several different classes of impactors)
  - Earthquakes (41 initiating events)
  - External floods
  - Loss of service water intakes

Scope of Goesgen PSA

- Implicit models (via „shutdown scenarios“ or manual scrams)
  - Wind and Tornado (contributions below 1E-10/a screened out)
  - Forest fire
  - Hail
  - Extreme snow loads
  - Climate change
  - Transportation and industry accidents
  - Turbine missiles (below screening threshold)

Structure of the Goesgen model

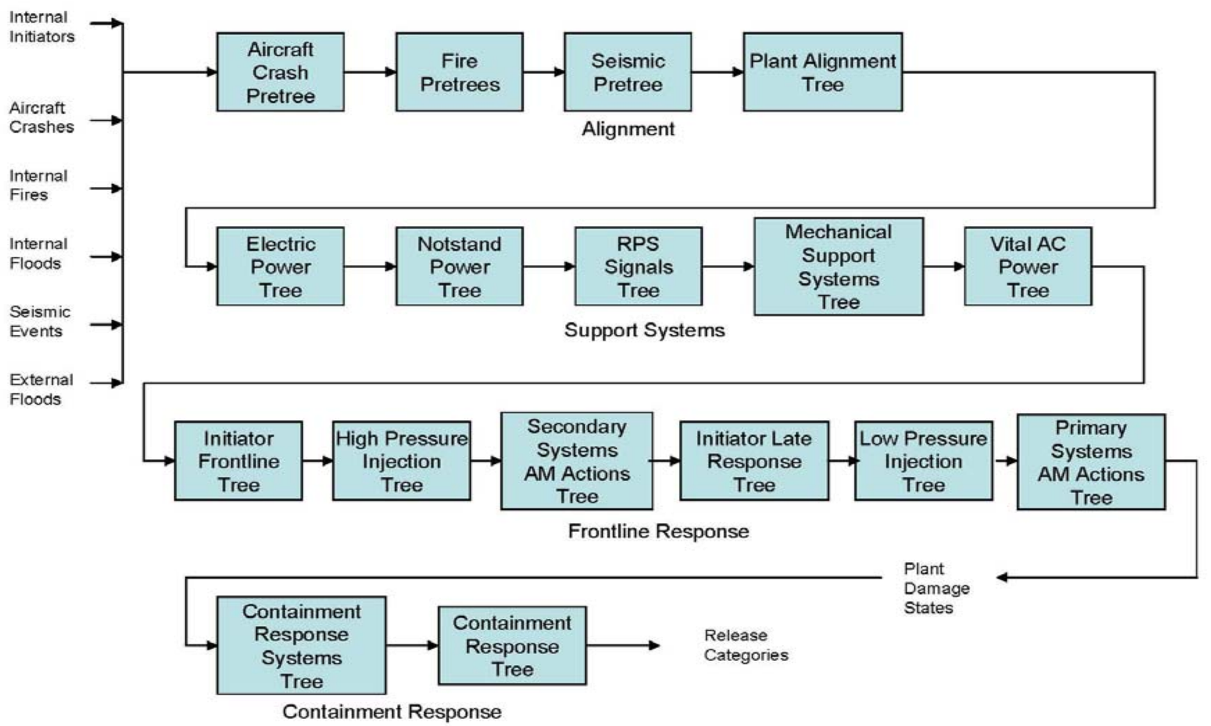


Figure 2-1. Overview of Modularized Event Tree Structure for the Full Power Model Level 1

Main results

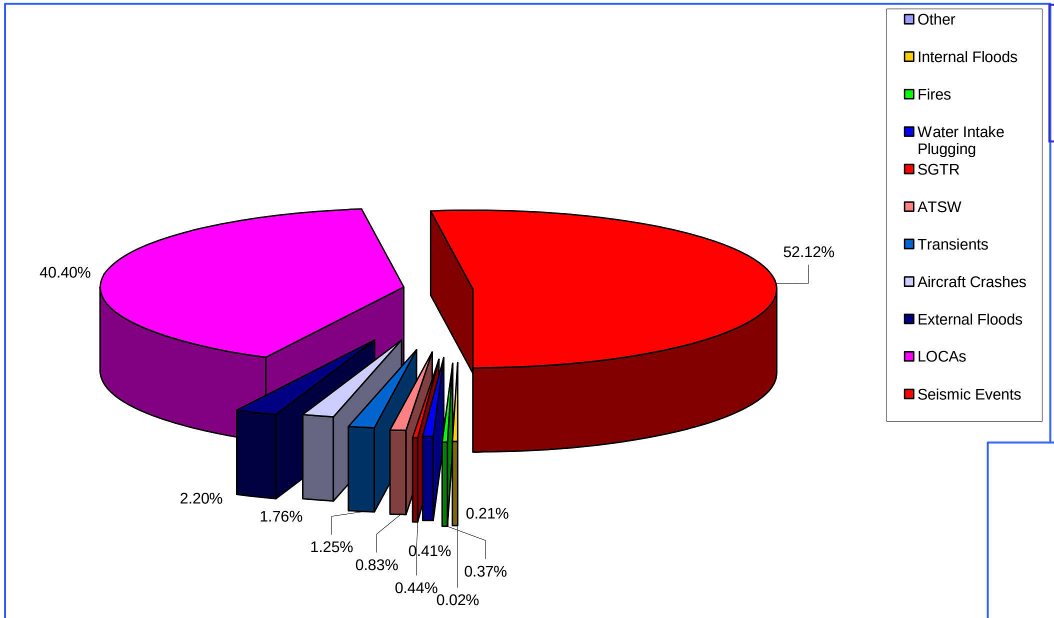
Initiating Event Group (Number of Initiators)	CDF Contribution [1 /a]	CDF Contribution, [%]
LOCAs (10)	2.61E-7	40.4
Transients (40)	8.07E-9	1.2
SGTRs (6)	2.87E-9	0.4
Internal Events (Total)	2.77E-7	42.9
Aircraft crashes (7)	1.13E-8	1.8
External Floods (1)	1.42E-8	1.8
Fires (23, more than 300 scenarios)	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%**

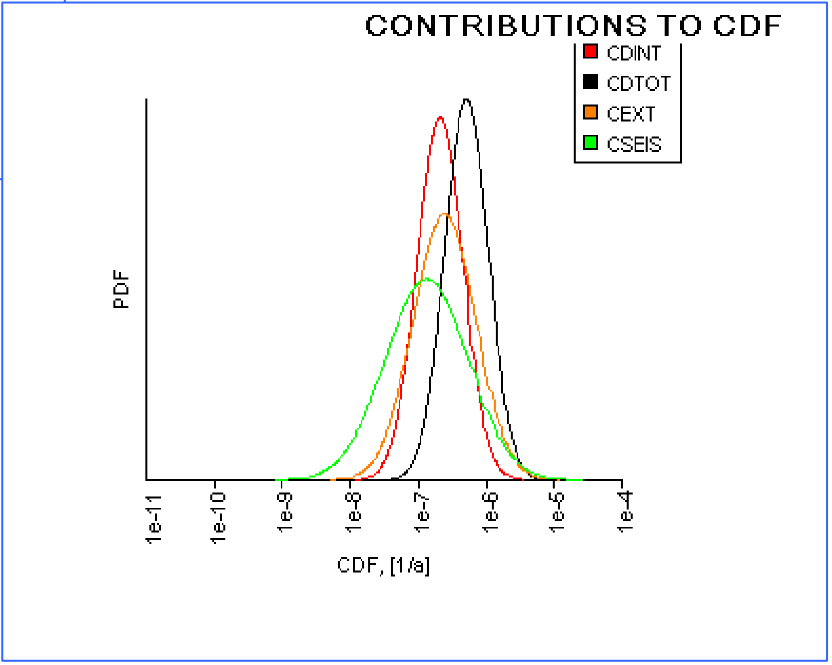
Are the results too optimistic??

Comparison with other studies, CDF

KKG (GPSA2009)	KKG (GPSA2003)	KKG (GPSA1994)	Convoy/ Preconvoy (GRS studies)	KKB (2009), Modernized Westinghouse plant
6.5E-7/a Plant State 2008	1.4E-6/a Model of 2009 Plant State 2003;	2.3E-6/a Model of 2009 Plant State 1994	1.7E-6/a (without seismic) (preconvoy– 4.6E-6/a with seismic)	1.7E-5/a (preliminary)
Results (simulating the design of other plants) using the Goesgen model (more heat sinks, more safety trains than other designs (+2 trains for most PIEs), 6x 100% for most PIEs)			4.3E-6/a	3.1E-5/a

Contribution of IE groups to CDF



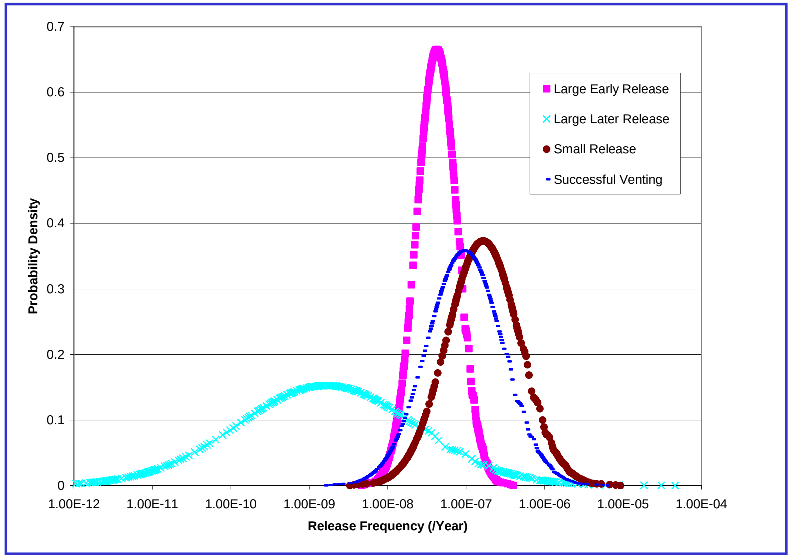


The CDF is dominated by external events,

Seismic events ca. 52%, total 56.5% of CDF

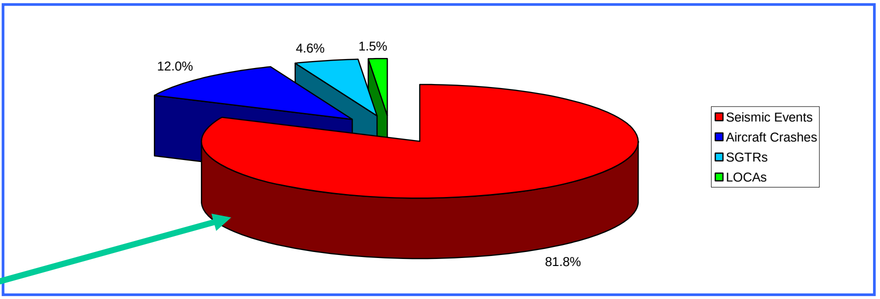
Results of Level 2 PSA, power operation

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



Containment venting is the most efficient measure to prevent large early releases

Noble gas release is not accounted for LERF

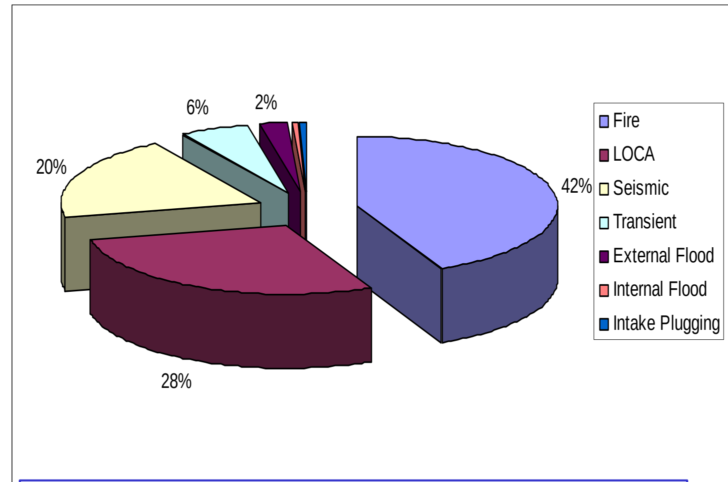


Largest contribution – Seismic ca.82%

External events – 96% of LERF

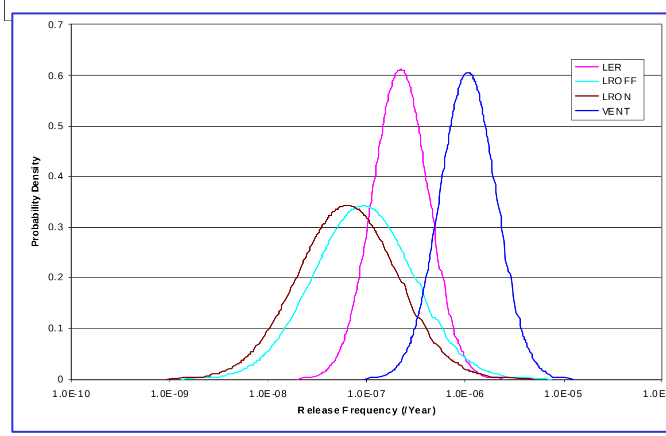
Results for Shutdown operational modes, FDF and Releases

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		



Fires largest contributor to FDF, 2.40E-6/a, FDF > CDF

Venting is also efficient for shutdown modes



Insights, Source Term Analysis

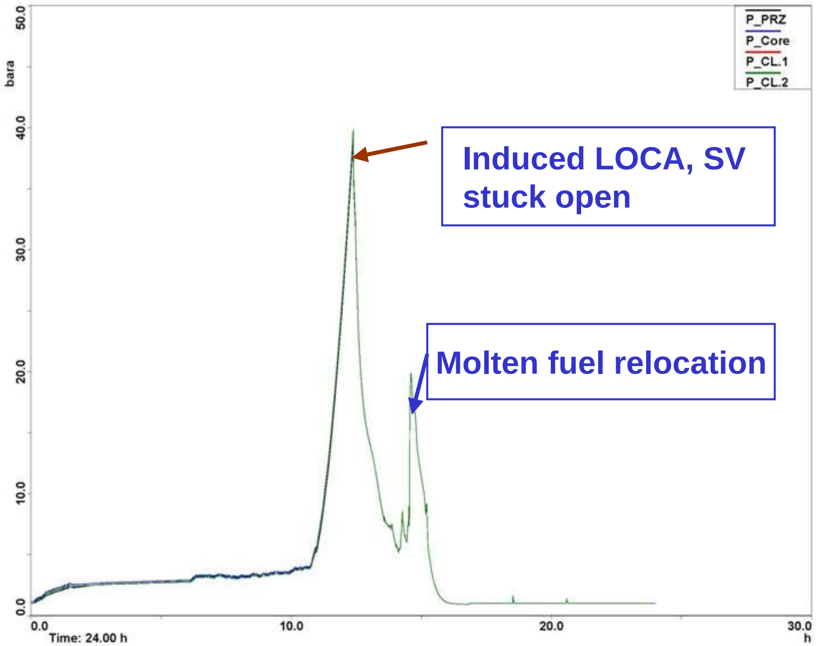
- Detailed MELSIM_KKG model (engine MELCOR 1.8.6)
- Input decks (incl. visualisation)
  - Early ¾ loop operation, vessel head in place
  - Early ¾ loop operation, vessel head removed
  - Late ¾ loop operation, vessel head removed
  - Pool configuration (fuel unloaded)
- Most critical „Post Damage Action“= Closing the Containment (material hatch);
  - difficult to close „small“ penetrations;
- Physical challenge to the containment (if closed) reduced in comparison to power conditions

Insights from accident and source term Analysis, ¾ loop, RCS closed, SBO with induced LOCA

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

Insights from accident and source term analysis, shutdown

Pressure in primary circuit, 3SGs available



- Sensitivity study confirmed the benefit of the Goesgen Operational manual (BHB) requirement, that at least two SGs shall be available before transferring the plant to reduced inventory shutdown operation modes.
- Study also confirmed the large time windows available for post-accident operator actions to prevent damage (>8 hrs)

Insights from Source term analysis, shutdown

SBO with induced LOCA, LERF source terms

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

The radioactivity releases are very similar despite the differences in accident progression

Insights from Accident and Source term analysis, shutdown, SBO, vessel head removed

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

C6 – early, C11 late configuration

Reduced time windows for post accident actions, about 3h for C6

Insights from Accident and Source term analysis, shutdown, SBO, fuel unloaded to pool

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)

Despite the large time windows the release is considered as „early“ according to Swiss guideline A05

Analysis of important operator actions (SAMG)

- Most important severe accident management measure is venting
  - Via „passive“ path
    - Isolated during normal operation,
    - „huge“ time window available to unisolate
- Most important action for shutdown is „isolating the containment“ (before core damage, normal post-accident action)
- Other actions not very beneficial from a risk perspective;
- Reason: LERF is controlled by external events failing the required hardware, preventing access to local service areas, or „shocking“ the operators

Summary and Conclusions

- Risk during shutdown is larger than during power operation (RPS not available)
- Availability of SGs during 3/4 loop operation (RCS closed) is beneficial, requirement in the operational manual (2 SGs must be available) approved
- Pre-damage post accident actions are more important for a reduction of LERF than the „direct" SAMG (mitigative actions)
- It is beneficial to remove maintenance activities from outage to on-line operation to reduce the „more risky" outage time

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

Fortum Nuclear Services Ltd.

Ms. Satu Siltanen, Dr. Harri Tuomisto, Mr. Tommi Purho

Outline of the presentation

- Introduction
- Loviisa SAM strategy as an application of Integrated ROAAM
- SAM strategy extension to shutdown states
- Fulfillment of the SAM safety functions during shutdown
  - Mitigation of hydrogen as a case example
- Level 2 PRA results
- Summary and continuation

Fortum

Introduction

- Loviisa SAM strategy

  → originally designed to cope with severe accidents starting from power operation

- Risk profile from PRA level 1* shows the importance of the shutdown states → SAM strategy extension for shutdown states going on together with shutdown extension of PRA2

- In parallel also on-going work in the area of procedures and guidelines for shutdown states

*Shutdown fire study under development, otherwise full-scope study

What makes shutdown states different (and difficult)?

- Different initial conditions (lower level of decay heat and pressures) → longer delays but core can still melt!

At the same time

- Containment might be missing
- Maintenance work, periodical testing, inspections:
  - Safety systems (prevention of core damage)
  - SAM systems (mitigation of core damage)
  - Auxiliary systems

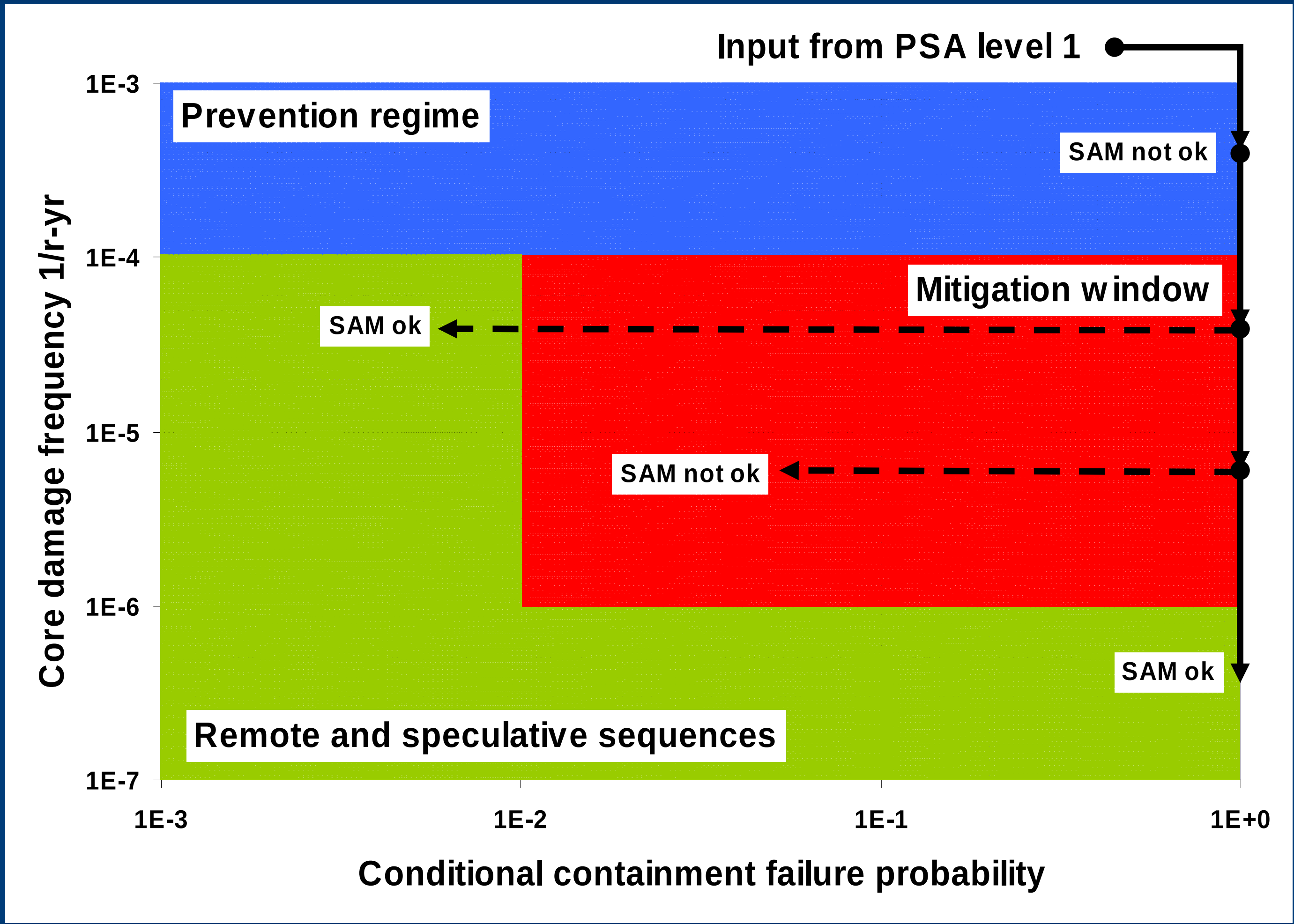
→ Main safety principles jeopardized (diversity, redundancy, separation, safety barriers..) and plant becomes more vulnerable

Fortum

Loviisa SAM strategy – idea of integrated ROAAM

For Loviisa NPP

- screening frequency 10^{-6} ^{1}/r-yr
- design target for failure of each safeguards function $< 10^{-2}$ / demand



Fortum

Loviisa SAM strategy - safety functions

- Successful containment isolation
- Primary circuit depressurization
- Mitigation of hydrogen combustion
- Reactor pressure vessel lower head coolability and melt retention
- Long-term containment cooling

→ Same safety functions have to be ensured also during shutdown states

Fortum

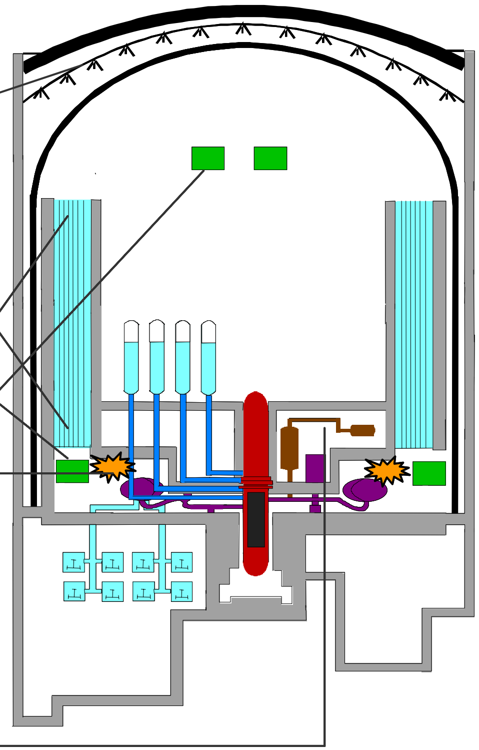
Loviisa SAM strategy - implementation

Long-term containment cooling by spraying the dome steel shell externally

Hydrogen mitigation

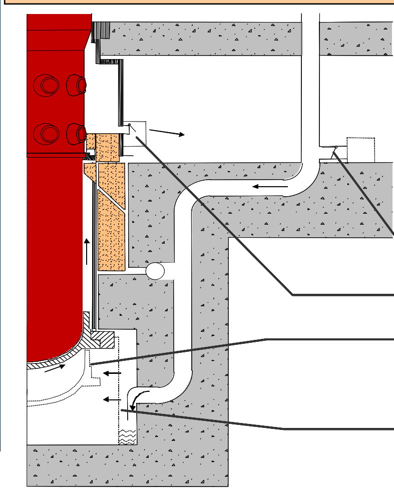
- Forcing the ice condenser doors open
- Catalytic recombiners
- Igniters (glow-plugs)

Depressurization of the RCS



Containment isolation

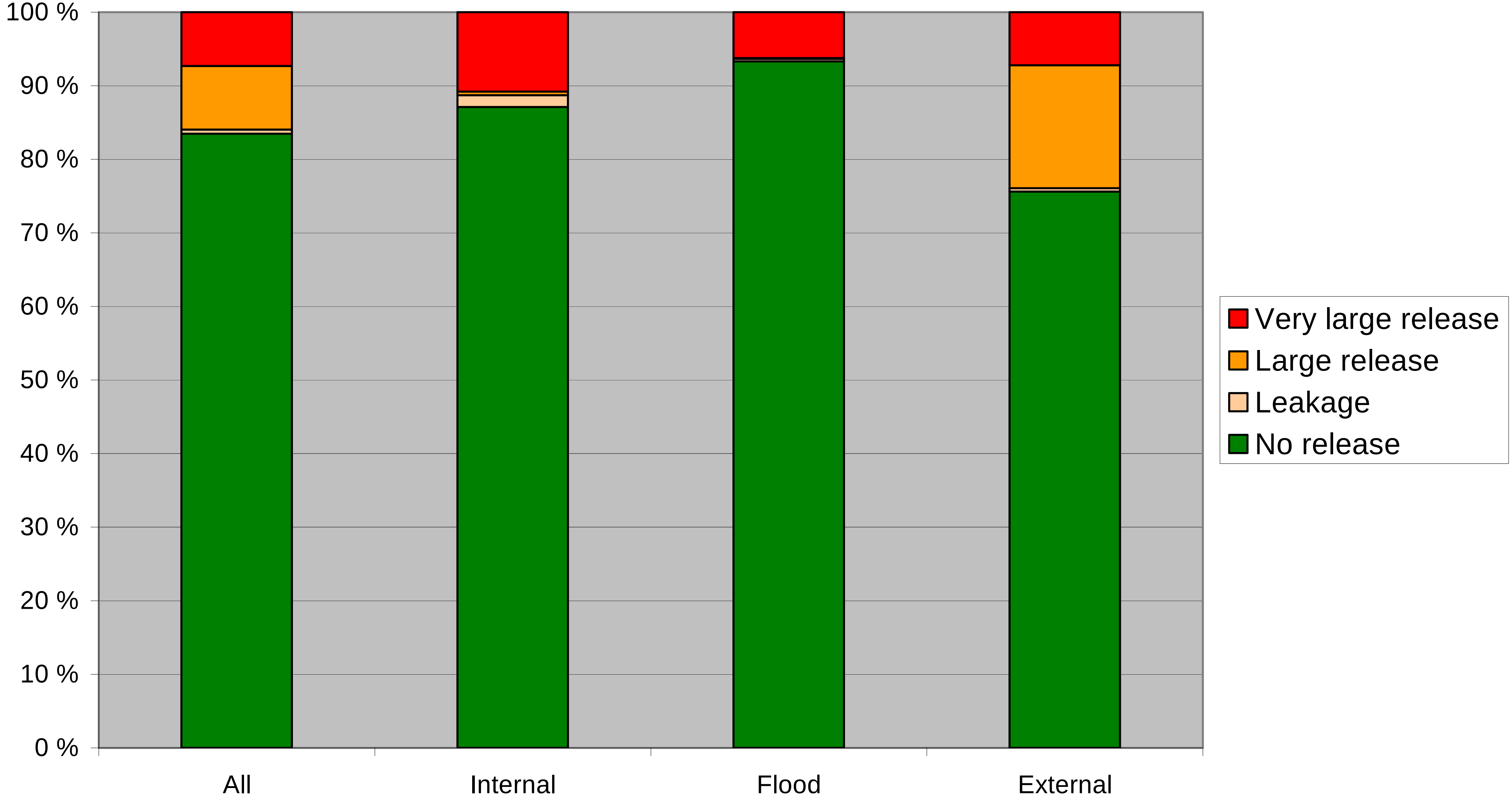
- Manual backup
- Local control centres
- Monitoring of the isolation success
- Monitoring of the leakages



In-vessel retention of corium

- Inlet valves
- Outlet valves
- Lowering of the bottom part of the thermal insulation/neutron shield
- Screening of impurities

Loviisa SAM strategy – showing the adequacy of mitigation part during power operation



SAM strategy extension to shutdown states

- Differences in containment initial conditions:
  - Containment tightness
  - Flow pattern in the containment
  - Maintenance of systems
- Accident sequences are typically slower
- Criteria for the SAM safety functions success might be different
- Situation changes depending on the stage of the outage

→ Extensive amount of work in order to analyze the state of containment and systems (especially SAM mitigation systems), and re-assessing of the success criteria of the SAM safety functions (background studies, code calculations observation during outages)

Safety functions in shutdown – mitigation of hydrogen

- Requirements are not different from power operation:
  - Forcing open the ice condenser doors → efficient mixing of containment
  - Hydrogen management with recombiners (and igniters when hydrogen production rate is high)
- Situation is different:
  - Containment flow pattern
  - Maintenance in ice condensers (filling the ice baskets)
  - Protection of recombiners against possible poisoning of the catalytic material
  - At least 1 train of igniter system is fully operable

Containment flow pattern

Ice condenser containment

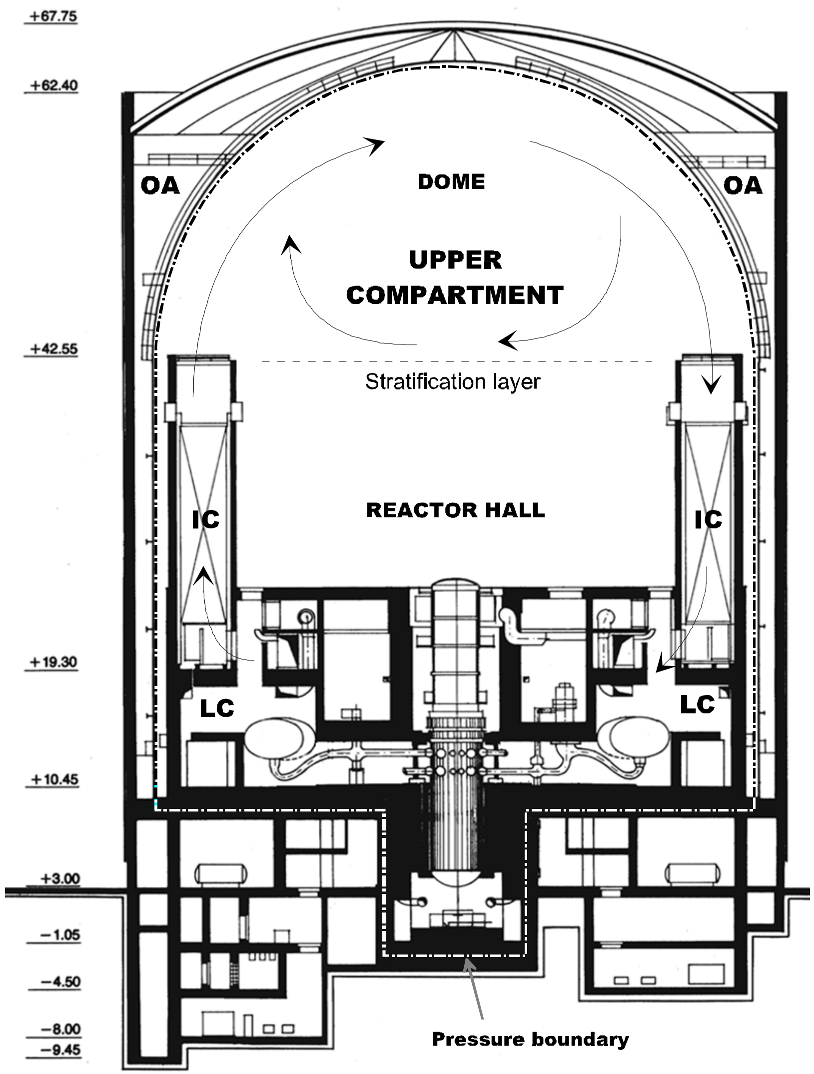
58 000 m3

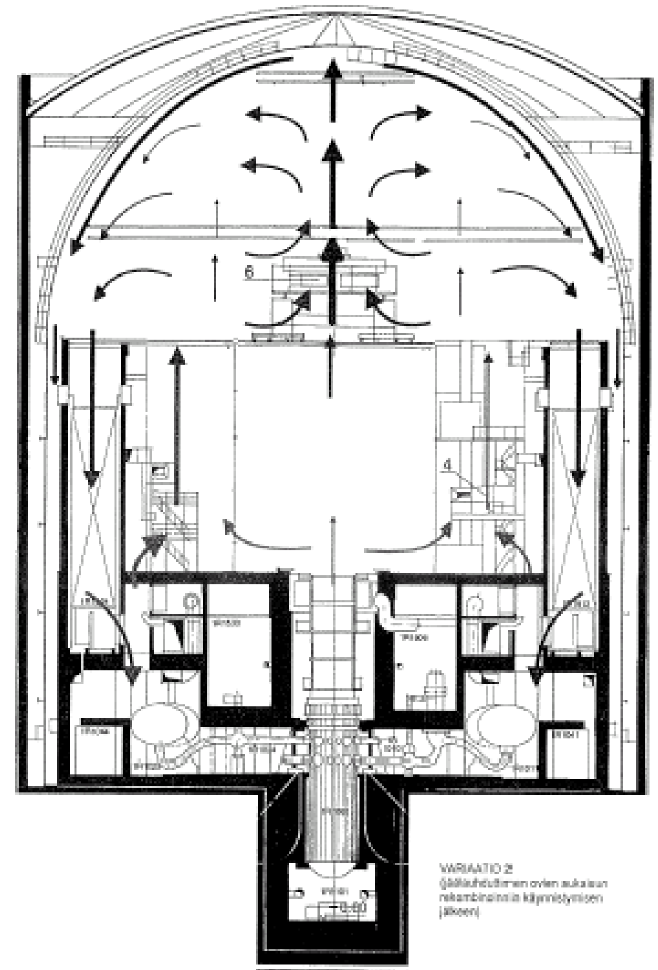
1,7 bar

Global convective loop flow

→ power (stratified)

→ → shutdown (well mixed)





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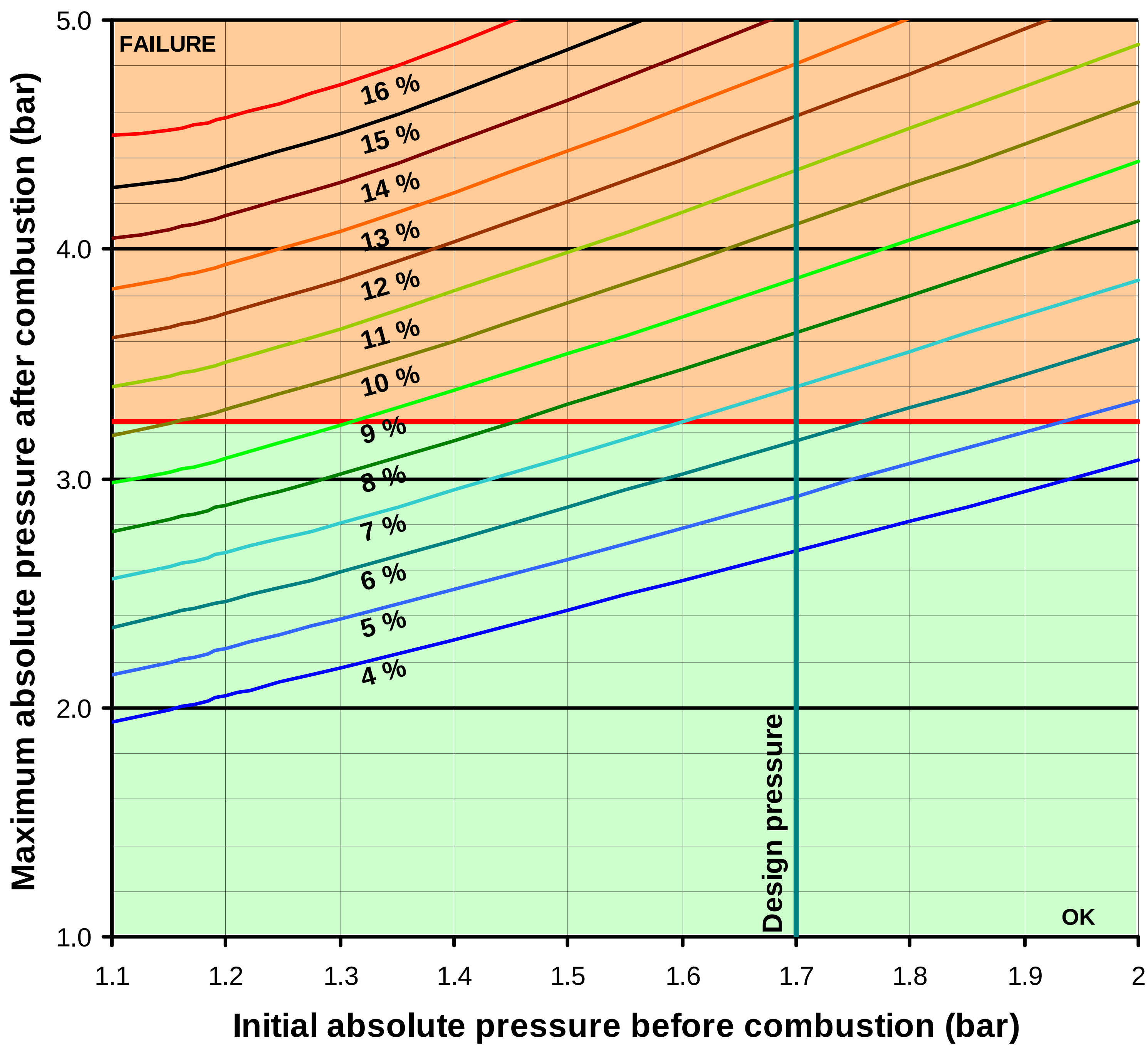
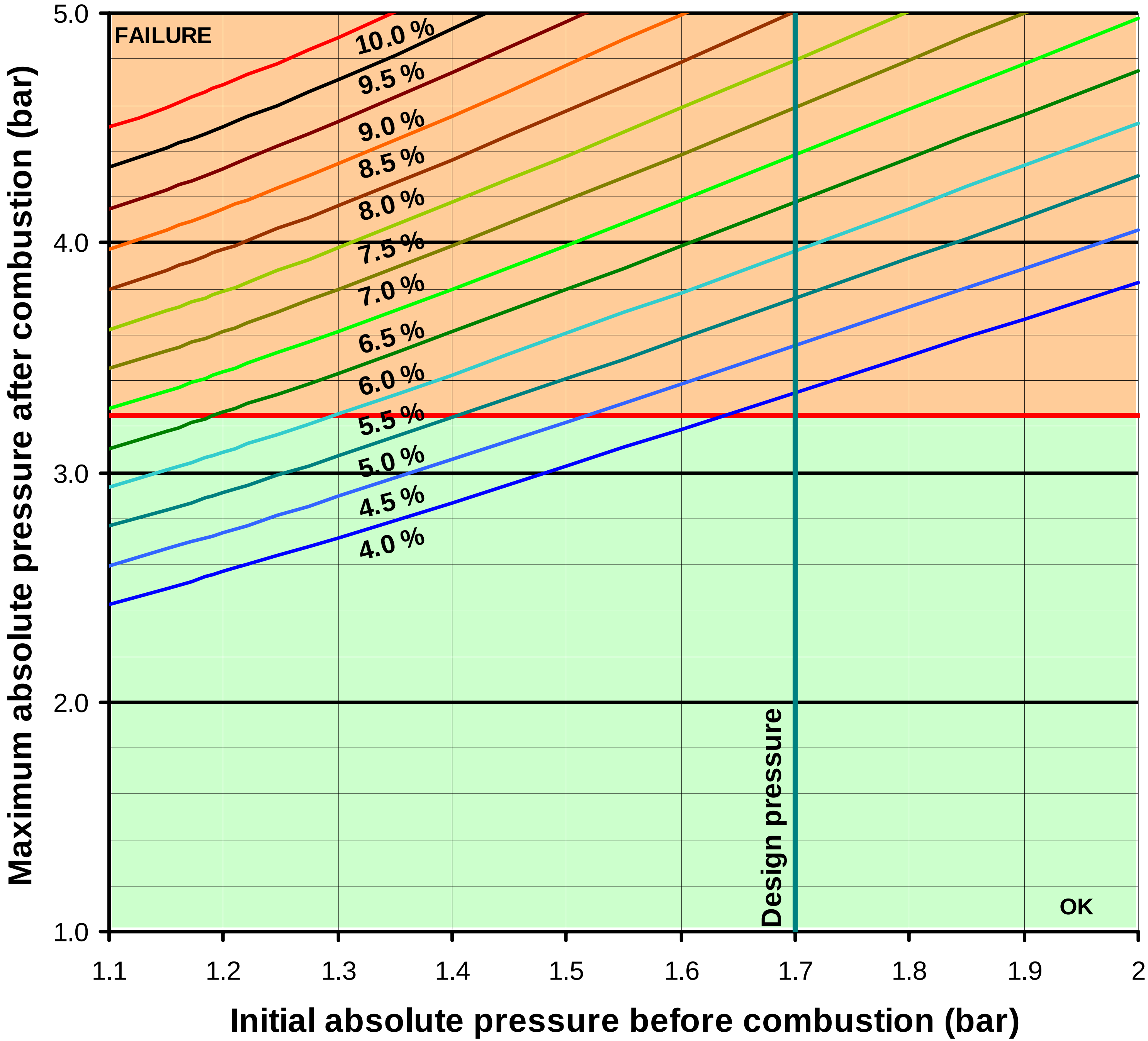
H_2 burn

- p_{AICC}
- Stratified upper comp.
- Typical for at-power states

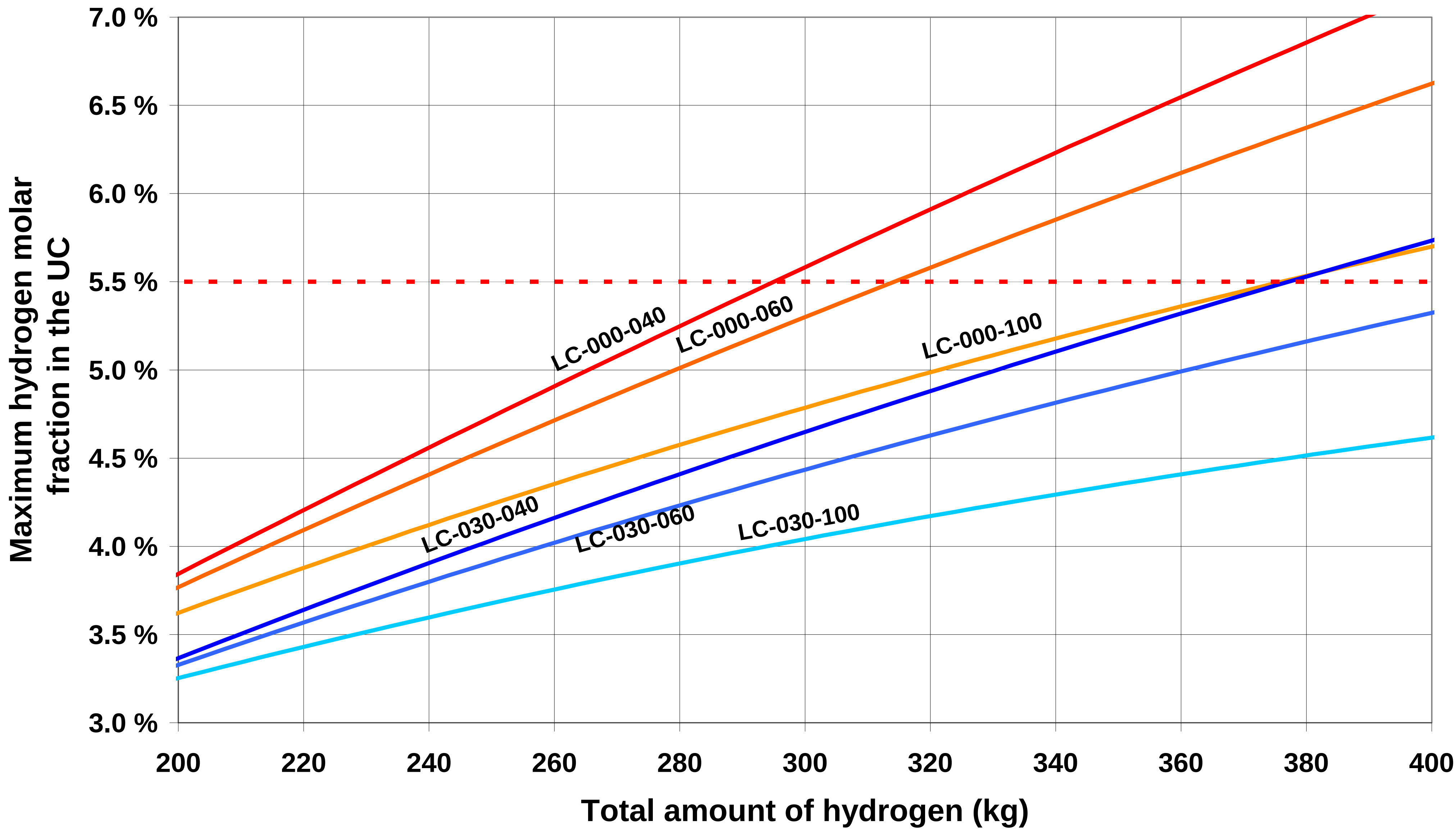


H_2 burn

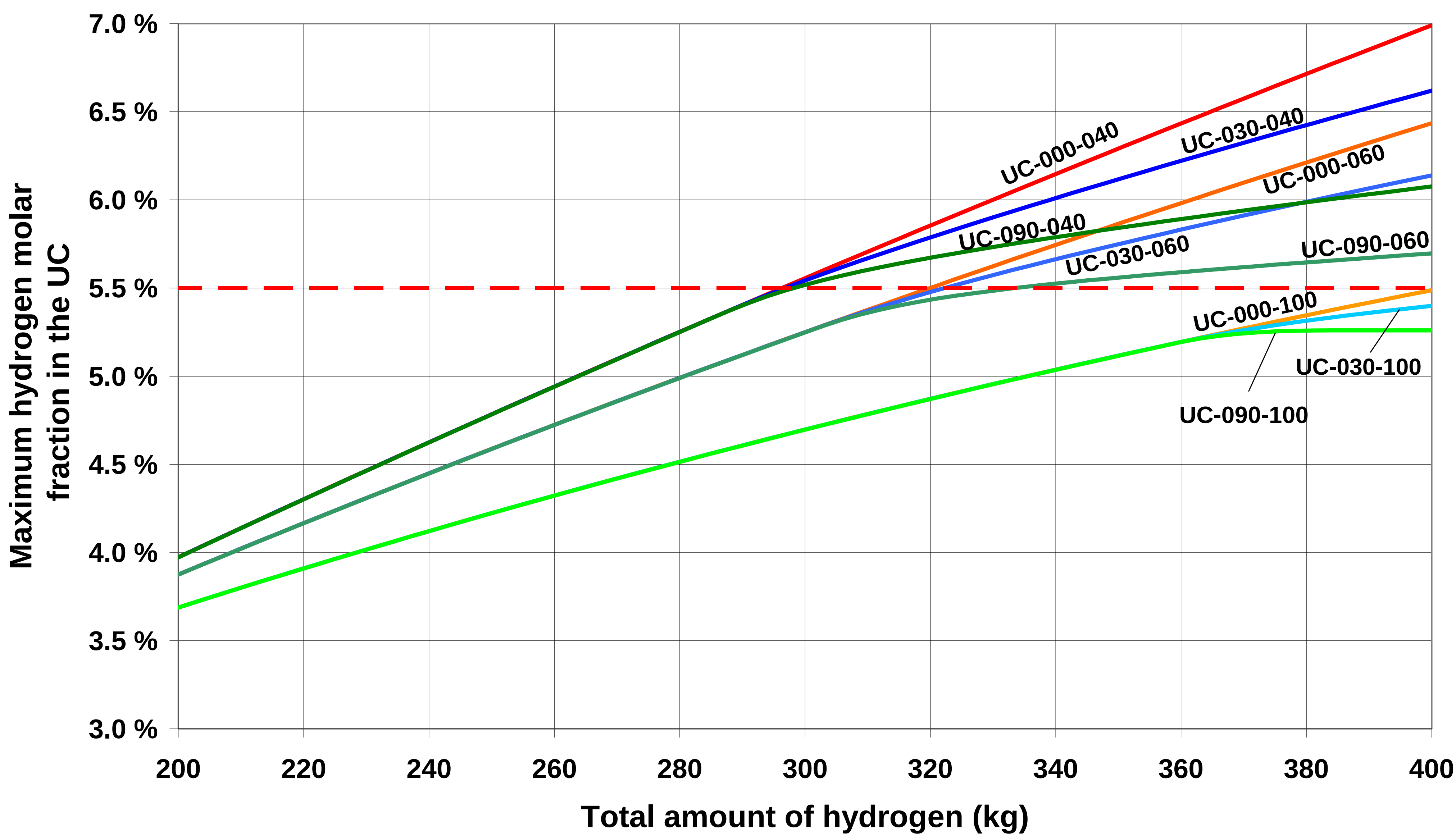
- p_{AICC}
- Well mixed upper comp.
- Typical for shutdown states



Maximum hydrogen molar fraction in the UC with source in the LC (recovery of the recombiners)



Maximum hydrogen molar fraction in the UC with source in the UC (recovery of the recombiners)



Safety functions in shutdown – mitigation of hydrogen

- Situation has been re-assessed in shutdown
  - Level 2 PRA success criteria have been defined
- Guidelines for the recovery actions needed in order to recover the operability of the ice condenser and recombiners has been made

- Recombiner protection has been found rather problematic. It has been studied whether it is possible not to protect recombiners during shutdown, some testing have been already made and work continues.

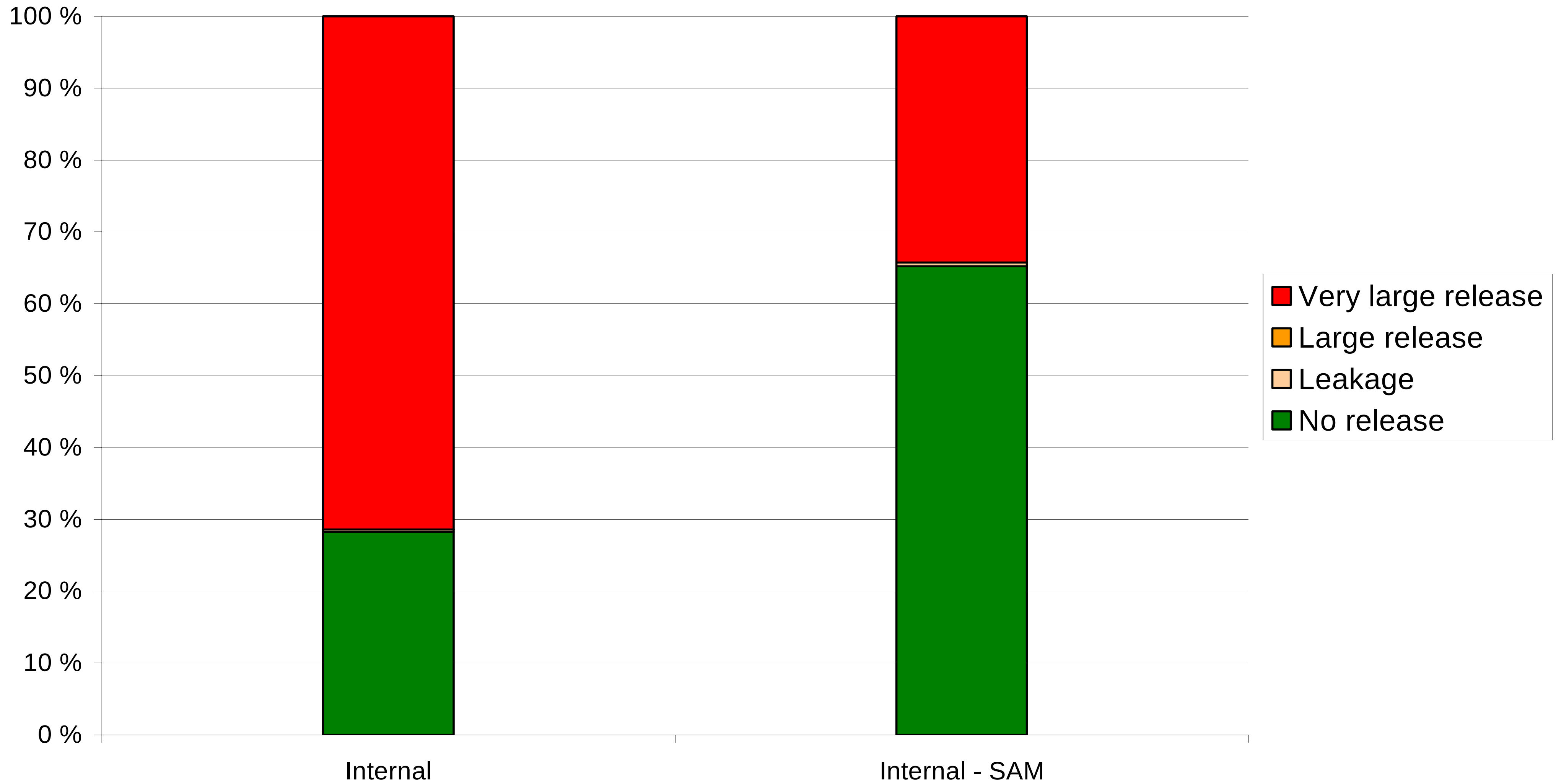
Safety functions in shutdown – other than hydrogen

- Also other safety functions have been separately studied and procedures facilitating SAM have been implemented and guidelines for SAM system recoveries have been written
- Some example of the point of interest:
  - Ensuring tightness of the lower compartment
  - Ensuring overall containment tightness (personnel hatches, penetrations which are closed and sealed during power operation)
  - Ensuring water to the cavity for in-vessel retention (lower decay power doesn’t melt the ice as effectively as during power operation)

Current situation

- Guidelines for SAM system recovery has been written (note the difference between these guides and SAMGs), validation and verification going on at the moment
- Many procedural changes have already been implemented, work with the other issues continues
- Level 2 PRA for internal initiators for refuelling outage has been done, at the moment on-going work with PRA 2 for internal flood initiators and external hazards

Loviisa SAM strategy in shutdown – going to the right direction, but work still has to be carried on



Summary and continuation

- Shutdown states have an important role in overall risk profile of the Loviisa NPP
- Loviisa SAM strategy has been originally developed for the accidents starting from power operation → extension to the shutdown has been started. Work continues

- Even though the main focus in this presentation (and in ISAMM2009 paper) has been on the extension of the mitigation part of the SAM strategy to shutdown states, the preventive part has not been forgotten. Also sequences which pose an imminent threat to the containment integrity (boron dilution, drop of heavy loads) have been studied and work goes on also around these issues.

Thank you!

Questions and comments?

mvm paks nuclear power plant

NUBIKI NUCLEAR SAFETY RESEARCH INSTITUTE

OECD/NEA workshop on
Implementation of Severe Accident Management Measures (ISAMM-2009)
26-28 October 2009, Böttstein, Switzerland

Development of the SAM strategy for Paks NPP on the basis of Level 2 PSA

Éva Tóth , József Elter

Paks NPP

Gábor Lajtha, Zsolt Téchy

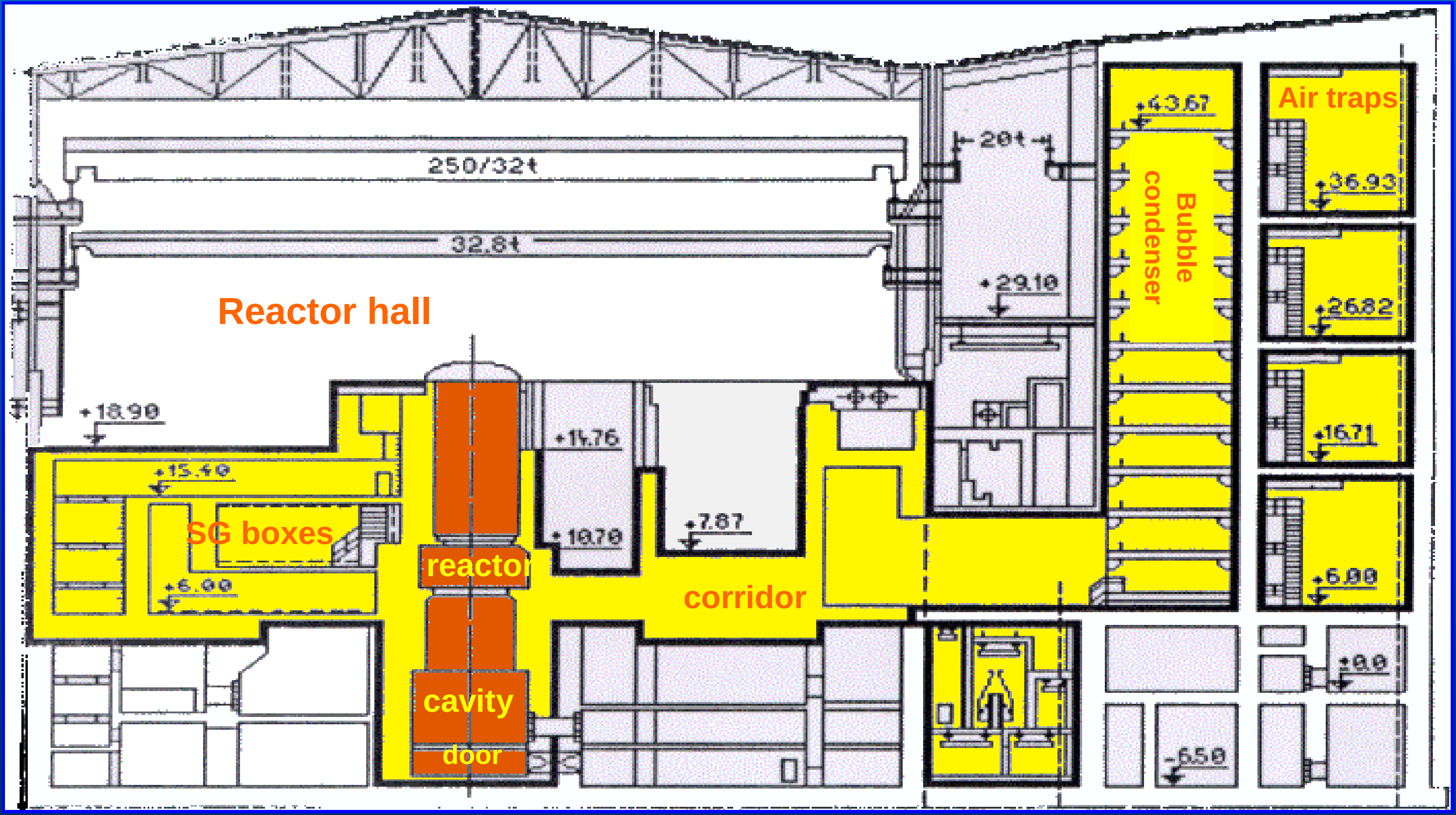
NUBIKI Budapest

HUNGARY

Table of Contents

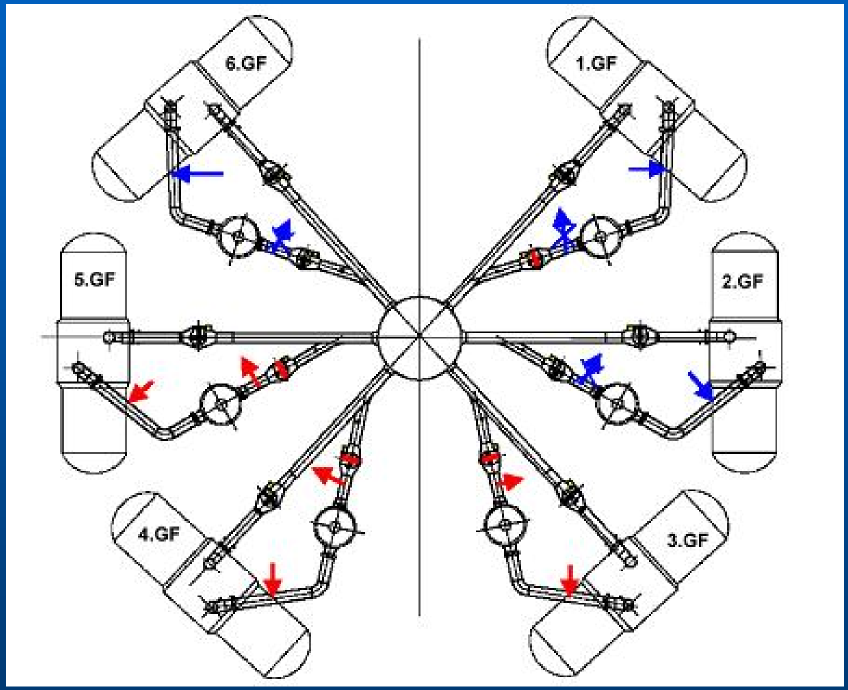
Specific design features with SAM implication
Containment structure

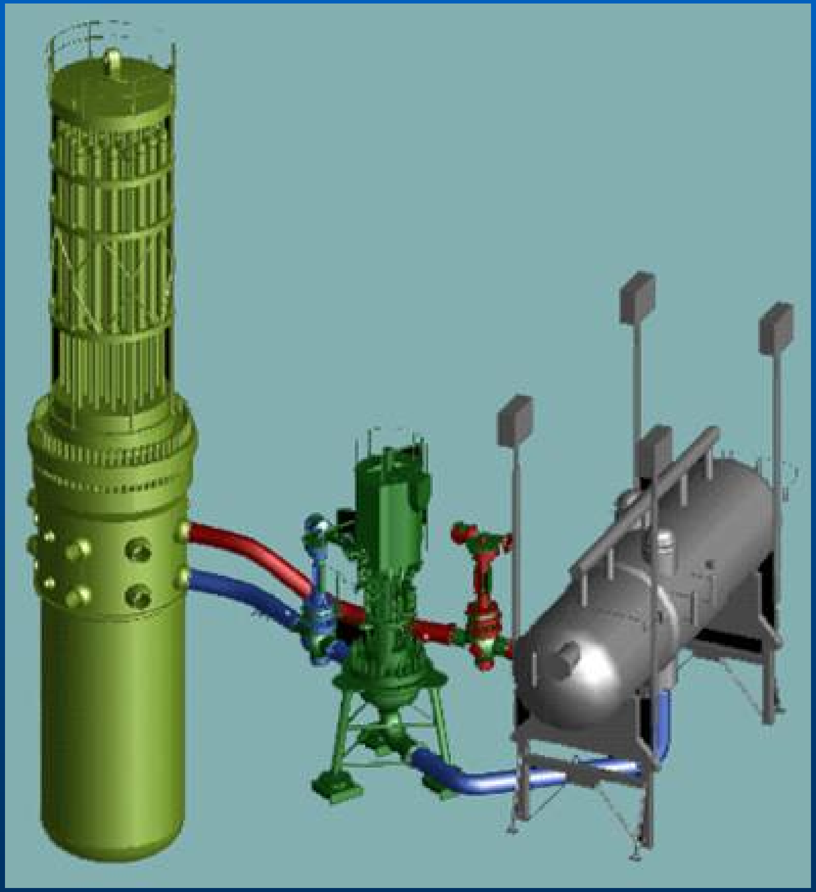
Paks NPP : 4 units VVER 440, 213 type with bubble condenser



Specific design features with SAM implication
Primary loops

6 loops with horizontal SGs, MCPs and MLIVs (loop seals)

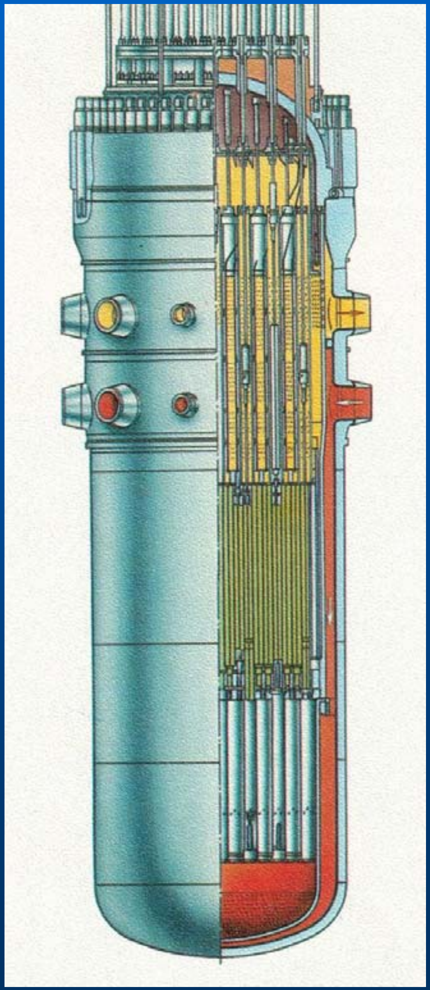




⇒ extremely large water reserves on the primary and secondary sides

Specific design features with SAM implication
Reactor pressure vessel

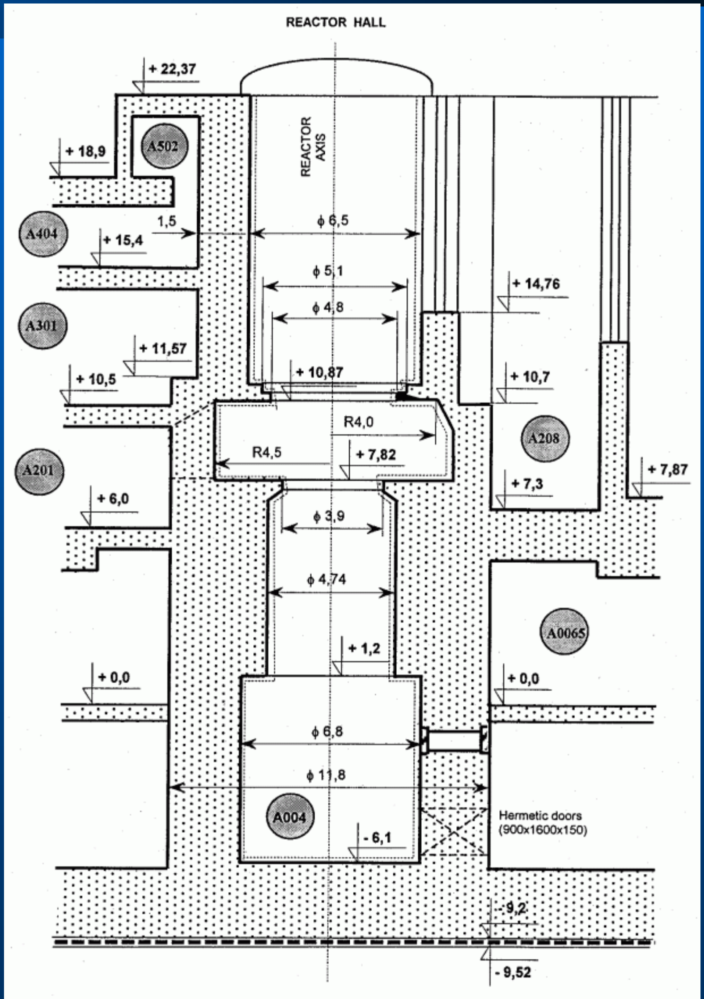
- relatively small reactor core in a long reactor vessel
- pressure vessel remains intact for a longer period even if the core remains uncooled
- RPV: relatively high surface area compared to low decay power ⇒ eventual outside cooling more effective
- nominal pressure in primary: 123 bar ⇒ primary pressure reduction! (in EOPs)



Specific design features with SAM implication

Reactor cavity and containment

- relatively narrow design with a door to non-hermetic comp.
- design pressure of cont.: 2.5 bar (ultimate cont. capability > 4 bar)
- relatively high (14.7 %vol/day) design leakage rates ⇒ now around 5-10 %vol/day
- about 1200 m^3 water reserve on the bubble condenser trays



Summary of Level 2 PSA results
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	

Summary of Level 2 PSA results
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

Summary of Level 2 PSA results
AM strategies and their components

	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)

Selection of AM procedures:

- **Release into the atmoshere: no significant differences between 2 strategies**
- **Basemat melt-through frequency:**
  - **with isolation of room A004 : 1,83·10^{-5} 1/unit/year**
  - **with cavity flooding : 1,6 · 10^{-7} 1/unit/year (in case of success)**

Summary of PSA Level 2 results

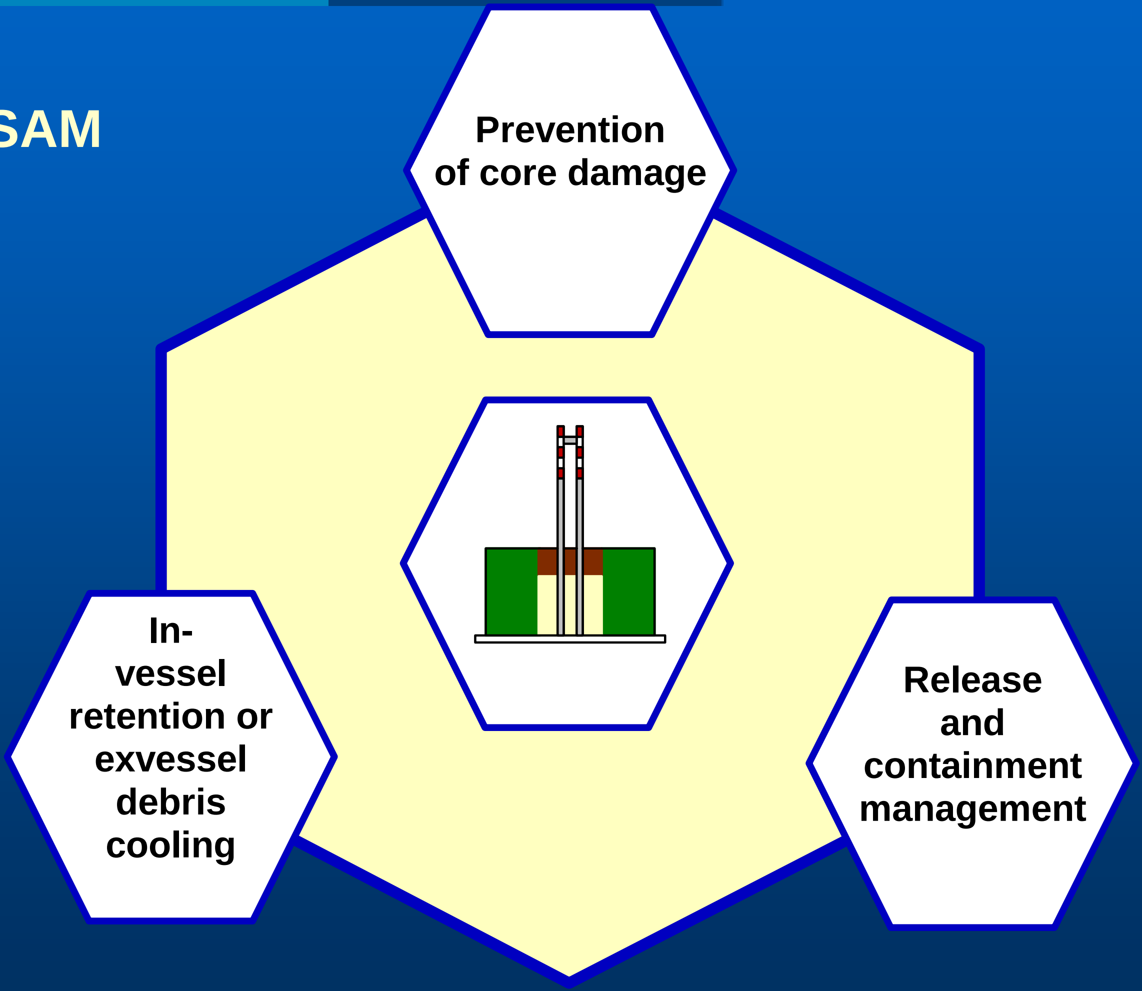
Frequency of the release categories

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	$4.13 \cdot 10^{-7}$

AM strategies and their components

Key elements of the SAM strategies

SAMG



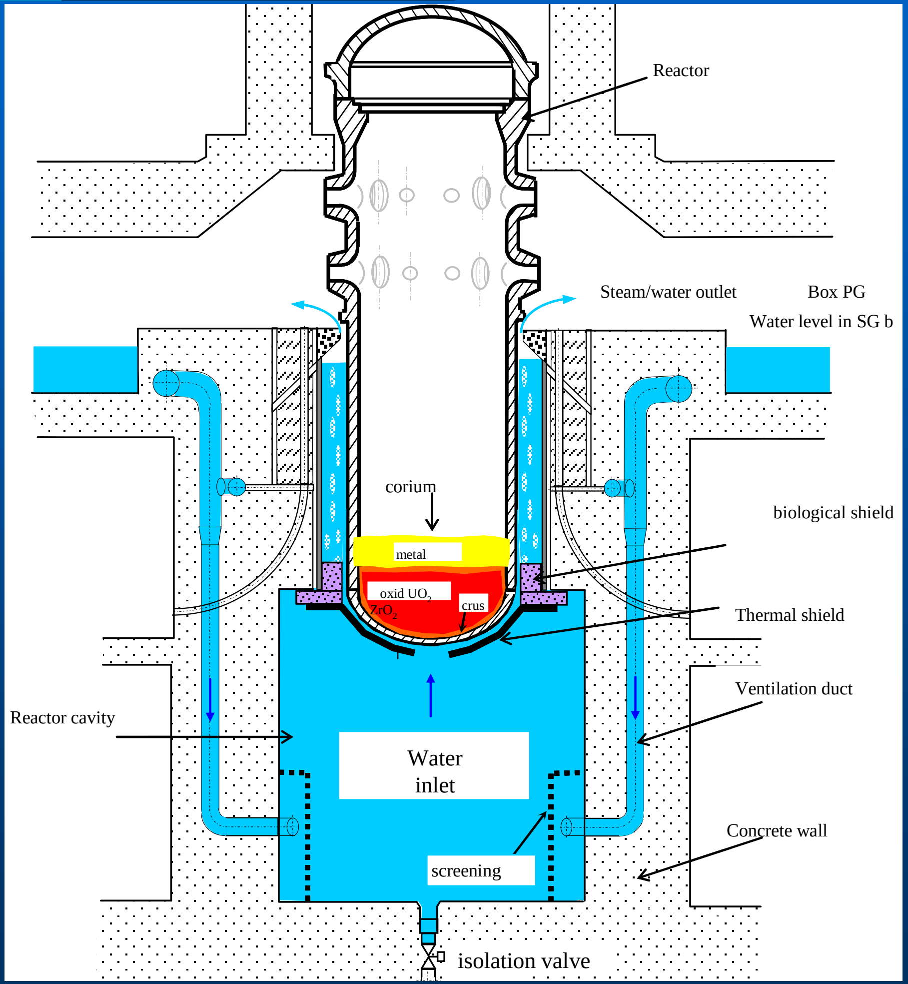
AM strategies and their components

- **Measures to prevent core damage:**
  - **Strictly perform the adequate EOPs (EOPs for shotdown state are developed)**
- **Measures to prevent RPV failure:**
  - **Primary system depressurization by opening of PRZ safety and relief valves (according to EOPs FR-C.1, ECA-0.0 and SAMG)**
  - **In-vessel corium retention by ECCS recovery and cavity flooding (in SAMG)**

AM strategies and their components
Cavity flooding

IVR concept:

- simple ERVC loop with minor modifications
- supporting analyses: proposed solution is effective in preserving RPV integrity
- engineering design: mostly passive, relatively low costs
- efficiency of the ERVC loop: will be proven experimentally by AEKI on CERES facility
- Licensing design documentation for implementation of necessary plant modifications prepared.



AM strategies and their components

- **Measures to safe cavity integrity in ex-vessel case:**
  - **Reactor cavity flooding $\Rightarrow$ not challenged**
- **Measures to safe confinement integrity:**
  - **Confinement isolation**
  - **Hydrogen treatment: application of 30 large passive recombinersre (required capacity and distribution calculated by MAAP4 and GASFLOW codes - VEIKI)**
  - **Prevention of late over-pressurization by filtered venting (modification of existing confinement vent system)**

AM strategies and their components

- **SAMG development (just finishing):**
  - **Should be based on the implemented plant modifications and measures**
  - **Should be linked with the already implemented Westinghouse type EOPs**
- **Preventive measures for open reactor and spent fuel storage pool:**
  - **Extension of EOPs for shutdown mode (just finishing)**
  - **Reinforcement of storage pool cooling system (installation of fast closing valves)**

Plant modifications
2 phase schedule

- Severe accident management measures ⇒ 2 priorities
  - 1. priority measures: will be taken anyway, independently of the life time extension of units

    (essential plant modifications, procedure development, organizational arrangements)

    scheduled: up to 2012
  - 2. priority measures: will be taken only in case of life time extension has been permitted by authority

    scheduled: after 2012

Plant modifications
1. priority measures up to 2012

Goal: to prevent core damage

- **Extend EOPs for shutdown mode and for storage pool accidents**
- **Set up PRZ valves and other SAM eq. with autonomous electrical supply**
- **Implement new PRISE strategy, plant modifications (bleed from ruptured SG to the cont. before it filled up)**
- **Reinforce storage pool cooling system**
- **Implement new strategy for ECCS tests**
- **Arrange duties for the other, non-damaged units**

Plant modifications
1. priority measures up to 2012

Goal: to prevent RPV failure and early containment failure

- Develop SAMGs (partly with provisional elements)
- Establish Technical Support Centre
- Install high capacity PARs to solve hydrogen issue
- Design and install cavity flooding flow path
- Install instruments and new, independent SAM measures
- Modify operating procedures: to ensure availability of cont.spray system and water from bubbler condenser trays for open reactor and spent fuel pool cooling

Plant modifications
2. priority measures after 2012

Goal: to prevent late containment failure

- **Increase reliability and protection of spray system from common cause failures**
- **Modify confinement vent system TN01 to use as filtered venting**
- **Finalize SAMGs on the base of hardware modifications**

Conclusions

- Unit VVER 440/213 type has specific design feauters $\Rightarrow$ plant specific SAM strategy and SAMG needed
- Selection of the possible SAM strategy based on the results of Level 2 PSA study.
- Main points of the proposed strategy:
  - hydrogen mitigation with recombiners,
  - in-vessel melt retention by flooding the cavity
  - using an existing ventilation system for filtered venting
- SAM measures for Paks NPP: 2 phase schedule
  - 1. priority measures up to 2012
  - 2. priority measures after 2012

THANK YOU FOR YOUR ATTENTION!

Development of Technical Bases for Severe Accident Management in New Reactors

ISAMM-2009 Workshop

October 26-28, 2009

Edward L. Fuller and Hossein G. Hamzehee

United States Nuclear Regulatory Commission

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Outline of Presentation

- Accident management (AM) programs for existing reactors in the United States
- The technical basis for existing AM programs
- Expanding the technical basis to address severe accident mitigation features in new reactors
- Severe accident management review for new reactors
- Insights regarding severe accident mitigation features in new reactors
- Severe accident management insights from NRC confirmatory assessments
- Conclusions

Accident management (AM) programs for existing reactors

- NRC offered an approach for implementing essential elements of a utility AM plan in SECY-89-012 and worked with the industry to develop guidelines.
- NEI 91-04 contains severe accident management (SAM) closure guidelines, describes the regulatory basis, and contains the binding implementing guidance.
- Industry technical basis for SAM stems from EPRI TR-101869, “Severe Accident Management Technical Basis Report,” which was used in developing vendor-specific guidance for use by various owners groups

The Technical Basis for Existing SAM Programs

- AM consists of those actions taken to:
  - ★ Prevent the accident from progressing to core damage;
  - ★ Terminate core damage progression once it begins;
  - ★ Maintain containment integrity as long as possible; and
  - ★ Minimize on-site and off-site releases and their effects.
- The latter three actions constitute SAM.
- The two-volume EPRI report (EPRI TR-101869), accident progression simulations using MAAP, and various computational aids, were used by the owners groups to develop severe accident management guidelines (SAMGs).

SAM review for new reactors

- The NRC's Office of New Reactors (NRO) staff expects this approach will be adopted by the applicants for new reactor licenses as well.
- The NRC staff reviews the design certification (DC) applicants' technical bases, and frameworks for procedure development and training programs, to ensure that the new features for accident prevention and mitigation are properly included.
- 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 utility's AM procedures and training programs.

Insights regarding severe accident mitigation features

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- The new reactor designs all include features that increase the capability for mitigating severe accidents. These address issues identified in SECY-90-016 and SECY-93-087 and associated staff requirements memoranda regarding:
  - ★ 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.
- Applicant evaluations of the performance of the mitigation features during severe accidents have provided a number of insights pertinent to SAM.

Insights regarding severe accident mitigation features (continued)

- Further insights result from confirmatory assessments carried out by the NRC's Office of Research (RES) for NRO.
  - ★ Severe accident scenario simulations are done using the MELCOR code, and the results are compared against the MAAP simulations.
  - ★ The insights obtained from these calculations are factored into the Safety Evaluation Reports (SER) prepared by for each design.
  - ★ Core debris coolability is a particular area of concern for all of the designs, because CCI threatens containment integrity both from overpressurization and from potential basemat melt-through.

SAM Insights for AP1000 Mitigation Features

- External reactor vessel cooling (ERVC)
  - ★ The objective of ERVC is to remove sufficient heat from the vessel exterior surface so that the thermal and structural loads on the vessel do not fail it.
  - ★ Design features include RCS depressurization, a clean lower head, reactor cavity flooding, and a RPV thermal insulation system.
  - ★ The AP1000 PRA estimates that more than 95% of core melt sequences would not lead to vessel failure.
  - ★ The NRC confirmatory assessment also concluded that the probability of vessel failure would be small, but its consequences must be taken into account from an AM perspective.
- Combustible gas control
  - ★ Monitoring of hydrogen concentration.
  - ★ Hydrogen igniters to promote burning soon after the lower flammability limit is reached.
  - ★ Decreases the probability of containment failure.

SAM Insights for AP1000 Mitigation Features (continued)

- Core debris coolability
  - ★ Design features, in case RCS depressurization and cavity flooding fail, include a large cavity floor area to promote melt spreading, a manually-actuated cavity flooding system to cover debris, and thick concrete layers to protect the containment shell and liner.
  - ★ 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.
  - ★ From <u>confirmatory assessment</u> calculations with MELCOR, the staff agreed with the applicant that the AP1000 design would provide adequate protection against early containment failure even if debris was not retained in the vessel.

SAM Insights for ESBWR Mitigation Features

- Combustible gas control
  - ★ The containment would be inerted during full-power operations.
  - ★ 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 hr.
  - ★ Combustible gas generation would need to be considered for low power and shutdown accident scenarios, because the containment may not be inerted then.
- Containment performance
  - ★ Because of the passive containment cooling system (PCCS), the three vacuum breakers between the wetwell and upper drywell are designed be essentially leak-proof.
  - ★ To prevent the possibility of containment bypass during a severe accident, each vacuum breaker is equipped with a check-type isolation valve that is normally closed.
  - ★ The vacuum breaker and the isolation valve would have to leak simultaneously for suppression pool bypass to occur.

SAM Insights for ESBWR Mitigation Features (continued)

- Core debris coolability and molten 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 water to flood the LDW when the temperature increases enough to be indicative of RPV failure and core debris in the LDW.
  - ★ The BiMAC provides a barrier to core debris attack of the LDW floor. The design features a series of side-by-side inclined pipes, forming a jacket that is passively cooled by natural circulation when subjected to thermal loading.
  - ★ Water from the GDCS pools enters the BiMAC pipes via connecting downcomers. Once the pipes fill up, the debris is also cooled from above from water that flows out of them.

SAM Insights for ESBWR Mitigation Features (continued)

- Core debris coolability and molten fuel-coolant interactions (continued)
  - ★ Flooding the LDW too soon increases the likelihood of a strong ex-vessel steam explosion that could cause structural failure of the pedestal or the BiMAC tubes.
  - ★ 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 so that water is not added too soon.
  - ★ Timely flooding of the LDW, a properly-functioning BiMAC, and a sound AM strategy, would make the issue of corium-concrete interactions inconsequential.
  - ★ MAAP 4.0.6 calculations and confirmatory assessments with MELCOR 1.8.6 show that, even if LDW flooding did not occur, containment integrity would be maintained for more than 24 hours for either limestone or basaltic concrete.

SAM Insights for U.S. EPR Mitigation Features

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- Combustible gas control
  - ★ The containment has a dedicated combustible gas control system (CGCS) with two subsystems to avoid containment failure.
    - The hydrogen reduction system consists of both large and small passive autocatalytic recombiners (PAR) installed in various parts of the containment.
    - The hydrogen mixing and distribution system ensures 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.
      - A series of mixing dampers and blowout panels would open to transform the containment into a single volume.

SAM Insights for U.S. EPR Mitigation Features (continued)

- Combustible gas control (continued)
  - ★ For either in-vessel or ex-vessel hydrogen production, both MAAP and confirmatory 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 the applicant's findings that, due to efficient recombination by PARs and by successful implementation of the hydrogen distribution system, there is little potential for formation of pockets of high hydrogen concentration inside the EPR containment and hence deflagration or detonation is unlikely.

SAM Insights for U.S. EPR Mitigation Features (continued)

- Core debris coolability and containment performance
  - ★ The Core Melt Stabilization System (CMSS) and the Severe Accident Heat Removal System (SAHRS) act 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, until a sacrificial layer of concrete is penetrated and a melt plug opens to allow molten core debris to flow to a spreading compartment.
    - 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.
    - Cooling elements form a series of parallel cooling channels through which water from the IRWST flows under the melt, along the sidewalls and onto the top of the molten core debris.
    - The melt would be cooled and stabilized.

SAM Insights for U.S. EPR Mitigation Features (continued)

- Core debris coolability and containment performance (continued)
  - ★ The SAHRS has four primary modes of operation:
    - 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.
  - ★ A properly-functioning CMSS would keep the debris cool, and prevent sustained concrete ablation in the core spreading room.
  - ★ 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 containment pressure.
  - ★ Confirmatory calculations with MELCOR found the time duration from vessel breach to reactor pit melt plug failure to be much shorter than MAAP predictions, and suggest that not all of the core debris would be in the pit yet. Subsequent delayed relocation has implications for energetic molten fuel-coolant interactions after water if water in the spreading room floods back into the pit through the connecting channel.

SAM Insights for US-APWR Mitigation Features

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- ERVC and Core Debris Coolability
  - ★ In-vessel retention of core debris by external RV cooling is considered as effective potential mechanism for severe accident mitigation. However, it is not credited because of large uncertainties.
  - ★ Flooding of the cavity would be initiated when core damage is detected, to cool molten debris after vessel failure.
  - ★ 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, retaining it and providing long-term stabilization inside the containment.
  - ★ 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.

SAM Insights for US-APWR Mitigation Features (continued)

- ERVC and Core Debris Coolability (continued)
  - ★ MAAP 4.0.6 calculations by the applicant predict that the water would quickly cool down the debris, and even if there was no water, containment integrity would be maintained for at least 24 hours.
  - ★ Calculations with MELCOR 1.8.6 <u>confirm</u> the MAAP calculations.
- High-pressure core melt ejection and containment bypass
  - ★ Severe accident-dedicated depressurization valves would be manually actuated shortly after core damage, reducing the RCS pressure to a level below that which would cause core debris to enter the upper containment atmosphere.
  - ★ In addition, the lowered RCS pressure would effectively eliminate the possibility of temperature-induced steam generator tube ruptures.

SAM Insights for ABWR Mitigation Features

- 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, where 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. ;
    - 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), to prevent catastrophic containment failure.
  - ★ MAAP calculations by the applicant and <u>confirmatory</u> MELCOR calculations by the staff indicated that the debris would be cooled using this approach, and when flooding did not occur, the time to COPS initiation would usually be more than 24 hours after accident initiation.

Conclusions

- The DC application reviews, both complete and ongoing, are confirming that the new reactors will be safer if the new severe accident mitigation systems that address the concerns expressed in SECY-90-016 and SECY-93-087 are included in the designs.
- All of the applicants claim that the new regulatory requirements emanating from SECY-90-016 and SECY-93-087 will be met by doing so.
- Both the preparations of the DC 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 bases now in place for the existing reactors.
- Using the enhanced technical bases will enable appropriate accident management procedures to 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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OECD/NEA Workshop on Implementation of Severe Accident Management Measures– Oct 2009 – Switzerland

1 – Introduction

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.

EC, NRC, OECDE, IAEA on-going activities are commented hereafter.

2. Ongoing activities within the European Framework Programmes

SARNET / ASAMPSA2

2.1 SARNET(Severe Accident Research NETwork of Excellence)

- **SARNET 1 - 2004-2008 - 51 organizations**
- **SARNET 2 - 2009-2012 - 41 organizations**
  - integration activities – ASTEC / spreading of knowledge
  - research on high priority issues
- **Activities concerning L2 PSA were performed within SARNET1 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,
  - on the L2 PSA requirements for computer codes such as ASTEC.

2.2 ASAMPSA2 (Advanced Safety Assessment Methodology : level 2 PSA)

The main characteristic of the ASAMPSA2 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.

Objective of ASAMPSA2 (From EC)

Objectives: based on research activities in previous Framework Programmes and in the Member State, to develop best practices guidelines for the performance of level 2 PSA methodologies with a view to harmonisation at EU level

Scope: best practice guidelines for

- *the performance of a level 2 PSA and the definition and clarification of the purpose, objectives and level of harmonisation for the various applications;*
- *a meaningful and practical uncertainty evaluation in a level 2 PSA.*

Expected impact: as a result of this action, the developed Level 2 PSA methodologies could be used with greater confidence in the further development of severe accident management procedures and could greatly assist in the decision making associated with plant life management.

ORGANIZATION

			END USERS GROUP		
			Establishing the needs	Verification that the guidelines fulfil the need	Definition of follow up actions proposal
TECHNICAL GROUP	Elaboration of a guideline for the limited scope methodology L2 PSA	Identification of data set needs for level 2 PSAs based on current reactor designs			
	Elaboration of a guideline for the full-scope methodology L2 PSA				
	Commentary on adaptation for Gen IV reactor				

The Partners

Participant N°	Participant organisation name	Country
1 (Coordinator)	IRSN	France
2	GRS	Germany
3	NUBIKI	Hungary
4	TRACTEBEL	Belgium
5	IBERINCO	Spain
6	UJV	Czech
7	VTT	Finland
8	ERSE	Italy
9	AREVA NP GmbH	Germany
10	AMEC NNC Limited	United-Kingdom
11	CEA	France
12	FKa	Sweden
13	Cazzoli consulting	Switzerland
14	ENEA	Italy
15	NRG	Nederland
16	VGB	Germany
17	PSI	Switzerland
18	FORTUM	Finland
19	STUK	Finland
20	AREVA NP SAS	France
21	RELCON	Sweden

Interesting stakeholders diversity

1 Safety Authority

5 TSOs

1-2 Vendor

4-6 Services/Ing. companies

3-5 Utilities

3 Research Organizations

The technical objectives (in what level of details we try to go)

The guideline should not stay at a "what a to do" step but should go deeply in a "how to do" direction.

- Example : the guidelines should present some practical approaches for the uncertainties assessment

All (?) L2 PSA issues should be covered (in fact those identified by the Partners + End-Users)

There is an interest to have an "open" process to improve the final quality of the guidelines

b) Structure of the ASAMPSA2 best-practices guidelines

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 for each issue will be built if possible.

c) Content : a guideline composed of 3 parts.

PART 1 - General

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 initiatives, 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.

c) Content: a guideline composed of 3 parts

PART 2 - Technical recommendations

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.

A very large number of issues may be examined in a L2 PSA. The treatment of each issue with enough details is another difficulty of the ASAMPSA2 project (with limited available resources) but the working plan developed and the current distribution of tasks between the Partners with the related experience shsould enable a complete coverage of all issues.

c) Content : a guideline composed of 3 parts

PART 3 – Application for GEN IV

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’ were considered as a key point :

- to establish the needs of the ‘End-Users’ for the performance of a L2 PSA,
- to assure the acceptance of the guidelines at the end of the project by a majority of the ‘End-Users’

A dedicated working group, coordinated by PSI, has been established to help in formalizing these relationships.

d) Relationship with the L2 PSA “End-Users”

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.

d) Relationship with the L2 PSA "End-Users"

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 …

d) Relationship with the L2 PSA “End-Users”

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.

d) Relationship with the L2 PSA “End-Users”

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.

e) Link with the international scientific research activities related to severe accidents

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-the-art for each topic, the guideline will be open for review by specialists involved in the SARNET Network of Excellence or NEA/CSNI members.

f) 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 hereafter are of course of high interest in relation to the ASAMPSA2 effort.

g) Schedule (to be considered as objectives …)

- **<u>Meeting 2</u> - 01st of December 2008 - (IRSN -Fontenay-aux-Roses)**
  - <u>General methods</u> (initial exchange of information – some outcomes from SARNET will have to be considered):
  - L1-L2 interface,
  - APET/CET (structure, general approach for the quantification of events, treatment of uncertainties …)
  - Release Categories (key parameters, example, screening frequency)
  - Human Risk assessment (example of actions, method for quantification)
  - Definition of representative TH sequences for each PDS
  - <u>Discussion on the End-Users workshop follow-up</u>

- **<u>Meeting 3 - 1st, 2d July 2009 - (VTT - Helsinki)</u>**
  - <u>Discussion on the first draft of guideline on general methods</u>
  - <u>Initial exchange of information on the following issues (some outcomes from SARNET will have to be considered)</u>
  - Phenomena - In-vessel core degradation
  - Phenomena - Vessel Rupture Phase
  - Phenomena - Ex-Vessel Phase
  - Phenomena - Containment performance (tightness)
  - System behaviour in severe accident conditions
  - Source term assessment

- **<u>Meeting 4 - November 2009 (2 days - postponed 28-29th of January 2010)</u>**
  - Discussion on the first draft of guideline on subjects discussed at meeting 3.
  - Identification of chapters to be improved.

- **<u>Meeting 5 - May 2010 (2 days - date to define)</u>**
  - Review of the version 1 of the guideline. This version includes conclusions of WG4 and will be submitted to End-Users review (**workshop in October 2010**)

- **<u>Meeting 6 - December 2010</u>**
  - Examination of the conclusions of the End-Users review.
  - Identification of chapters to be improved.

3 Ongoing NRC activities of interest to the international Accident Management community

(From D. Helton - NRC)

NRC - State-of-the-Art Reactor Consequence Analyses (SOARCA) project

- **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.**

NRC - Existing standards for PSA

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)[1] for internal and external hazards for light-water reactors.

The US NRC's position on this standard is articulated in Regulatory Guide 1.200[2].

[1] 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.

[2] 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.

NRC - New PSA standard in development

There are three additional light-water reactor standards (under development) that are of interest to the Accident Management community:

- low power shutdown PRA, Level 2 PRA, and Level 3 PRA
- applicable for existing and advanced light-water reactors

The L2 PSA 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 target date for a draft of the new Level 2 standard is late 2009.

NRC - New PSA standard in development

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."

Others NRC activities

Review for advanced light water reactor

- Deterministic severe accident analysis, probabilistic severe accident mitigation design alternative (SAMDA) analysis, and Level 2 PRA development

Development of the necessary guidance for operational oversight of the new reactors, including the risk metrics (in relation with the international community, e.g. MDEP)

For operating reactors : accident management issues, severe accident mitigation alternatives (SAMA), analyses for license renewal, and development of advanced Level 2/3 PRA methods.

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 has provided an opportunity to relay some of the recent references that may be of key importance for the development of L2 PSAs.

- See : Table 1. OECD references on severe accidents, severe accident management and Level 2 PSA

2 papers recent on PSA2

- 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.

extracts from the TOP

“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.”

“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

(From A. Lyubarskiy, IAEA,)

Safety standards

- **See in particular the safety guides :**

  - Severe Accident Management Programmes for Nuclear Power Plants” (NS-G-2.15)

  - Development and Application of Level-1 Probabilistic Safety Assessment for Nuclear Power Plants

  - Development and Application of Level-2 Probabilistic Safety Assessment for Nuclear Power Plants (SG 393)

Review of Accident Management Program (RAMP)

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 Accident Management Programme.

International **P**robabilistic **S**afety **A**ssessment **R**eview **T**eam (IPSART)

Service established in 1988 and conducted following IAEA TECDOC 832. Review of PSAs is performed on request by the Member State.

From one to two weeks with 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 (description of the review, findings, technical aspects of the PSA study, strengths and limitations, recommendations for improvement of the PSA quality and its sound use for enhancing plant safety and risk management applications.

More than 50 IPSART missions have already been performed.

6 - Conclusions

This overview 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).

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.

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.

THANK YOU FOR YOUR ATTENTION

Nuclear
Energy

Accident Management and Risk Evaluation of Shutdown Modes at Beznau NPP

OECD-Workshop:
Implementation of Severe Accident Management Measures,
Böttstein, Switzerland, October 26 - 28, 2009

Martin Richner and Samuel Zimmermann
Axpo AG, NPP Beznau, Switzerland

Jon Birchley and Tim Haste
Paul Scherrer Institute, Villigen Switzerland

Nathalie Dessars
Westinghouse Electric Belgium S.A., Nivelles, Belgium

Table of Contents


Beznau NPP today
Oldest operating PWR worldwide
NOK
Plant extensively backfitted

1970 Orig. NPP

1980

RWST

1990

Bunker Systems

Cont. Filtered Vent

SGs

Condensators

HP-Turbines

2. Hydro Emerg. AC

MCR

Important Valves

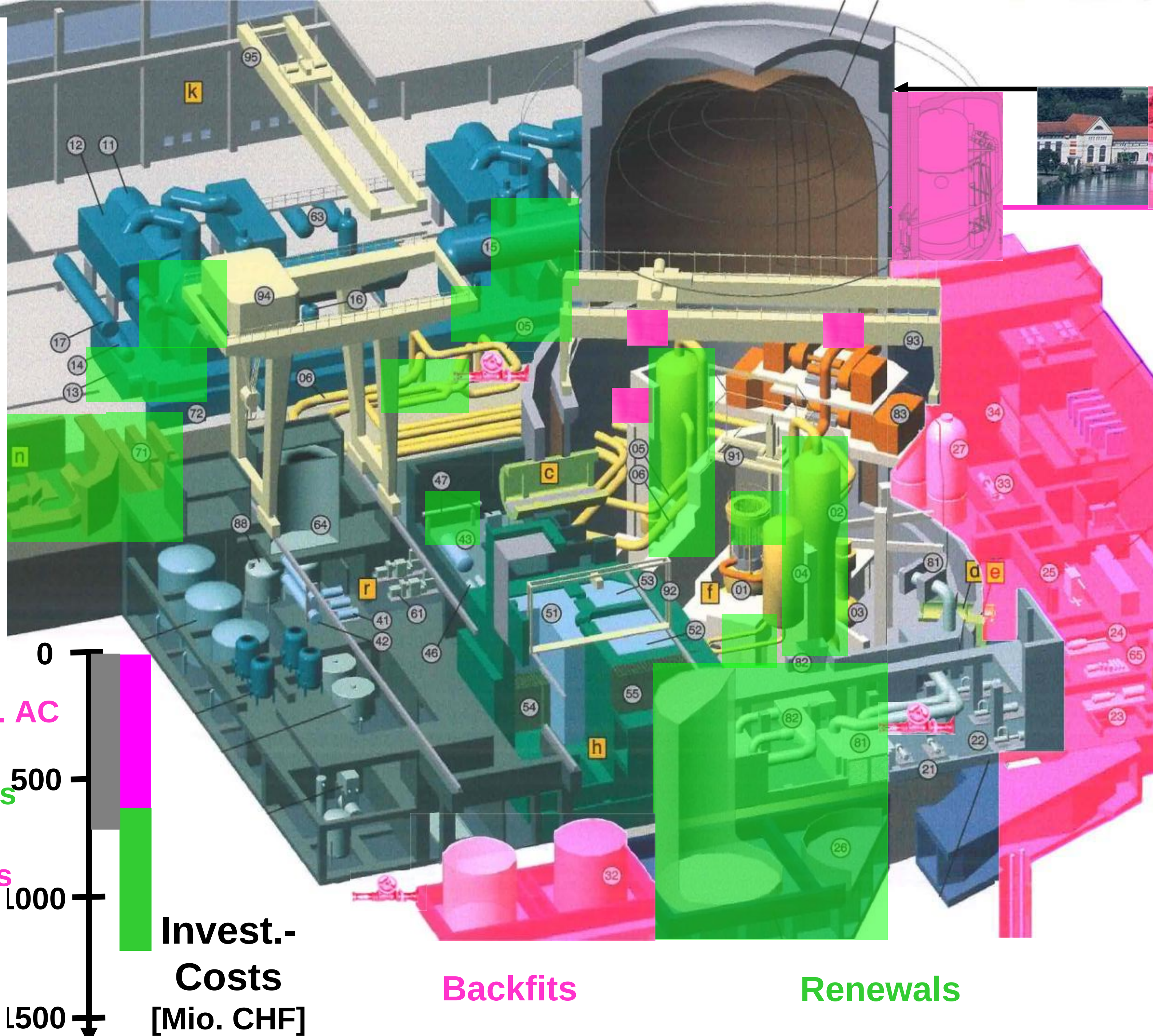
Emerg. FW

FW-Connections

RPS

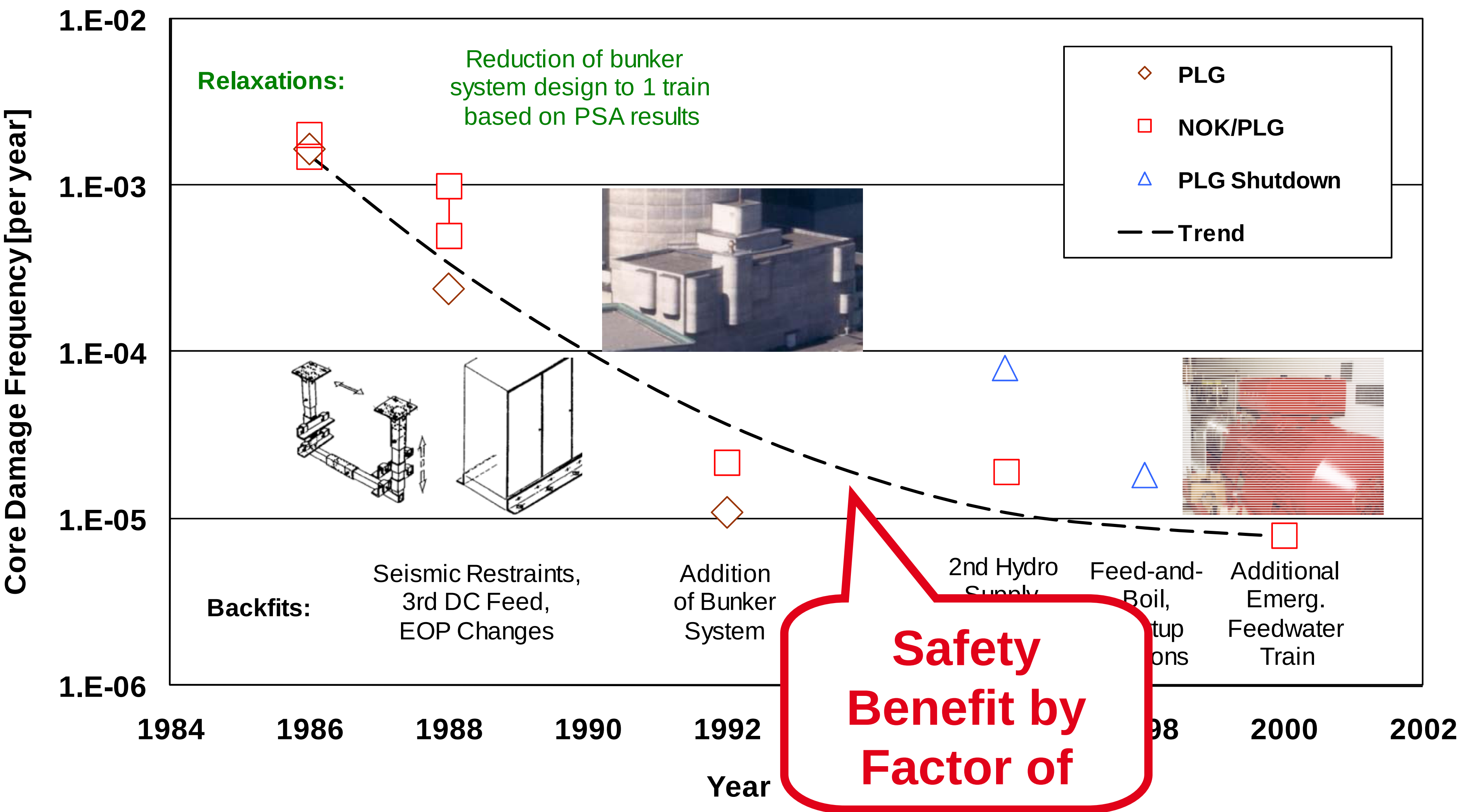
2000

H_2-Recombiners



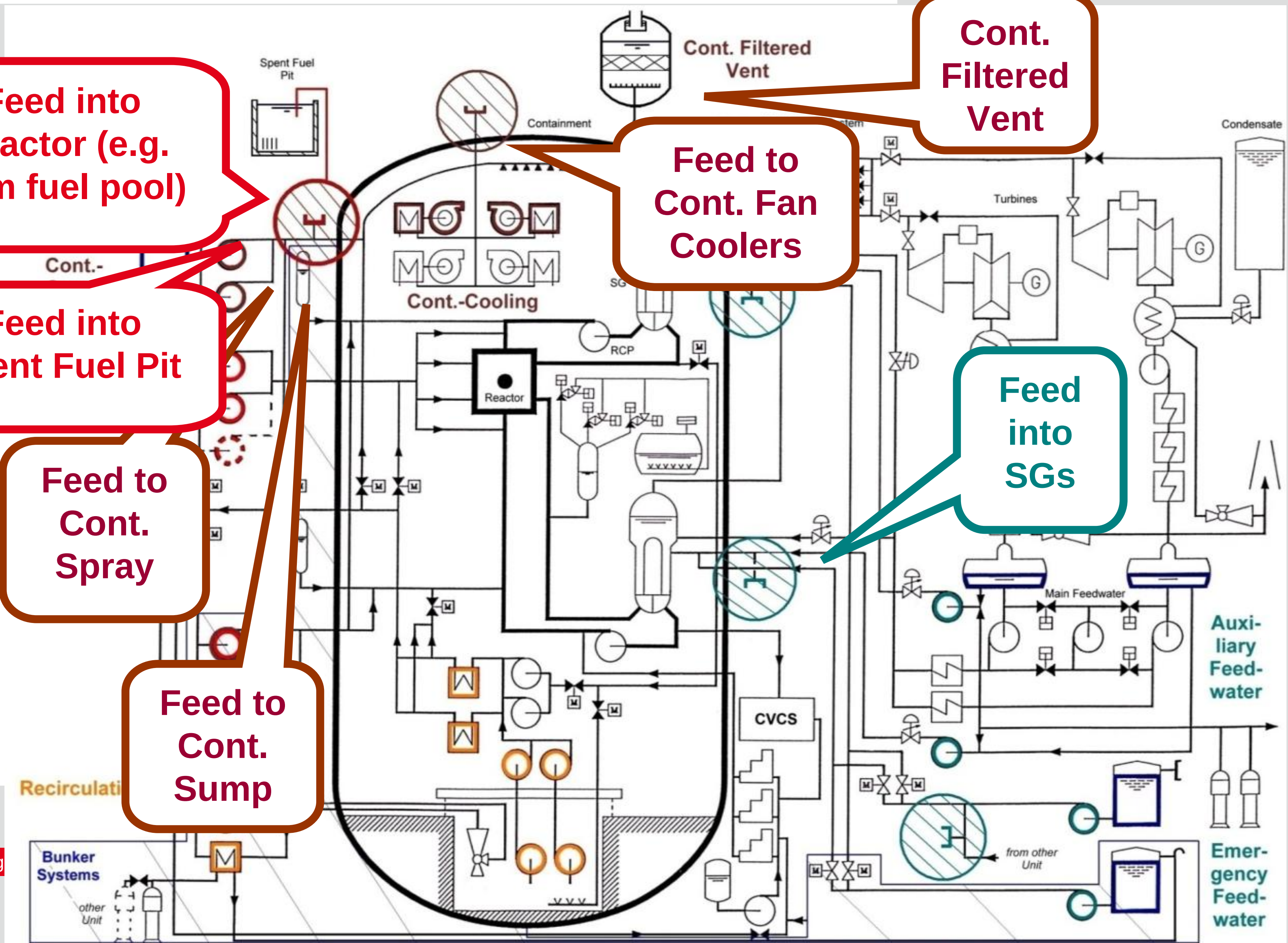
Backfits and Frequency of Core Damage
Total of Internal and External Events

aspo Nuclear Energy

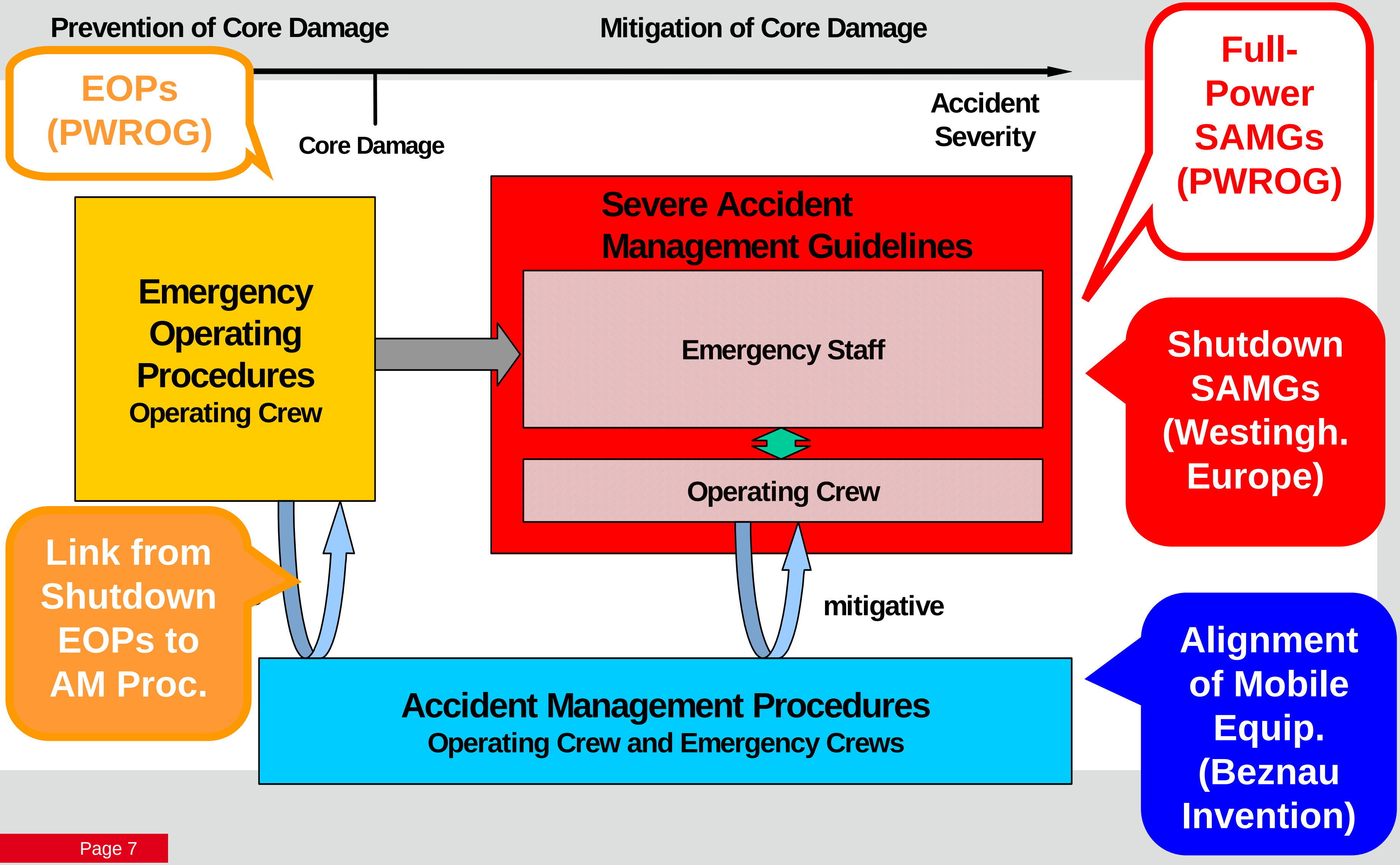


2. Beznau Accident Management Program

Hardware for AM



Beznau Accident Management Procedures



Extensions from Full-Power SAMGs to Shutdown SAMGs

1. Procedure for Spent Fuel Pool
2. Transition Evaluation Table (TET)
   - **Transition into SAMGs in configurations with the core exit thermocouples removed**
   - **Alternate parameters than core exit temperature are:**
     - **Containment radiation**
     - **Hydrogen concentration inside containment**
     - **Hot Leg and pressurizer temperatures**
     - **Reactor neutron flux**

Specific Factors of AM during Shutdown

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Long time windows during shutdown

Alternate indications from TET for core uncovery

Comprehensive shutdown AM program may significantly reduce shutdown CDF

Human errors dominate shutdown CDF

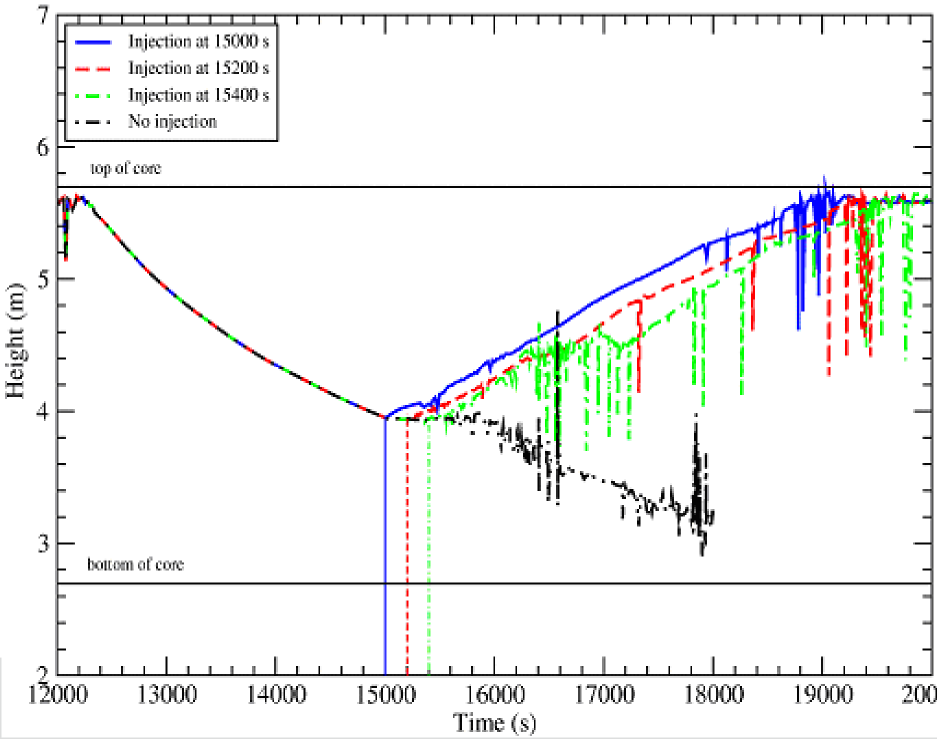
Independent responsibility of emergency staff for shutdown SAMGs

Alternate hardware such as fire water pumps

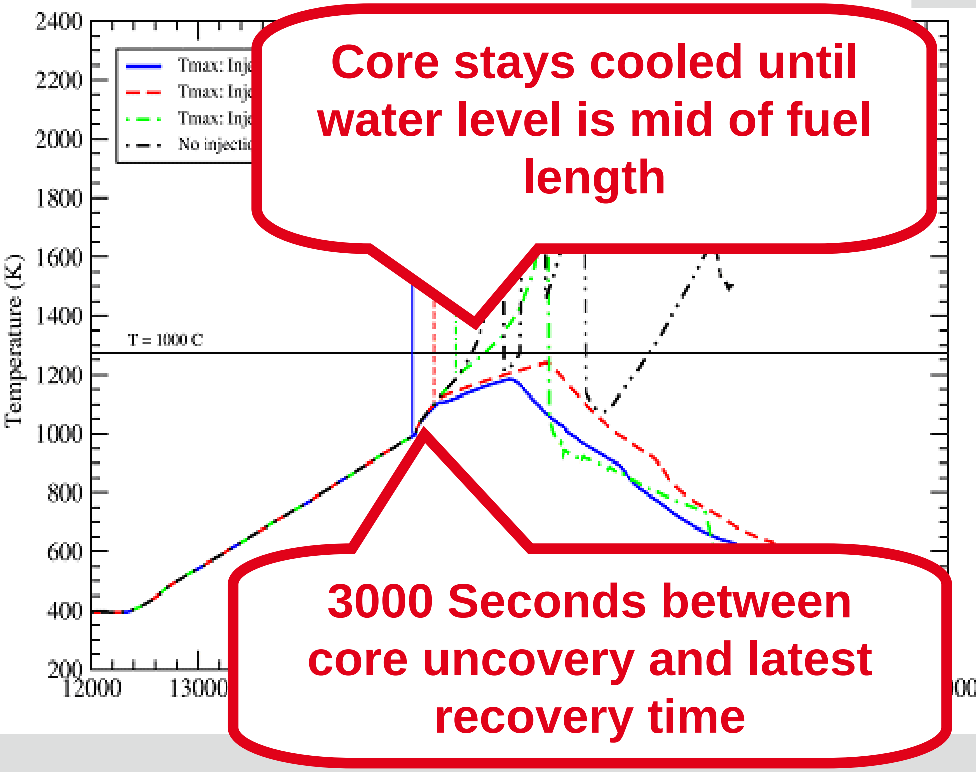
3. Realistic Evaluation of Shutdown Risk

- **MELCOR Analysis: Loss of RHR cooling at Mid-Loop 22 h after power operation**
- **Restart of charging injection when water level is mid of core length**

Core liquid level

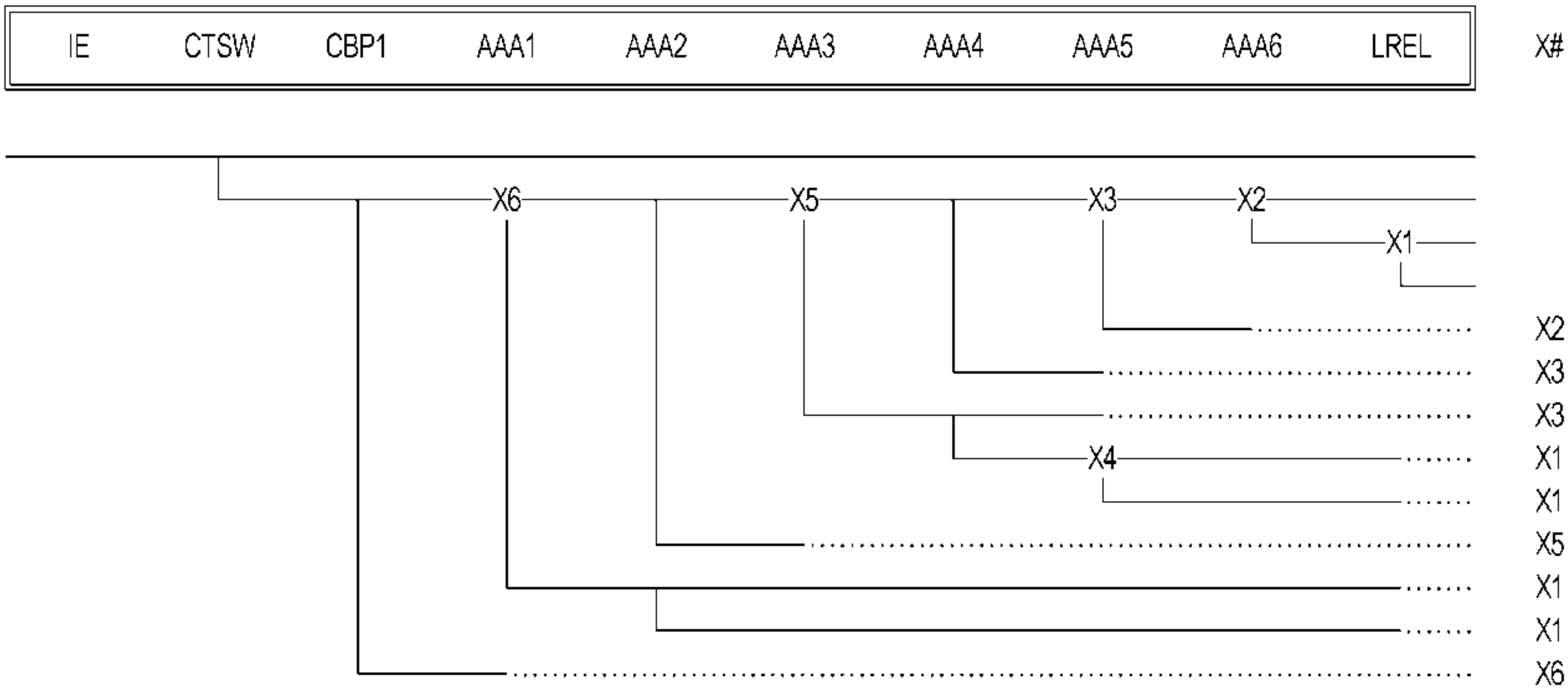


Maximum Cladding Temperature



Modeling of AM Measures in Shutdown PSA

- Consider start of charging pump after core uncovery as alternate recovery action (additional time and indications)
- Specific Accident Management and Containment Event
  - Includes AM hardware and operator actions
  - Simplified Level 2 Model
  - Fully linked with Level 1 model



Nodes in Accident Management and Containment ET

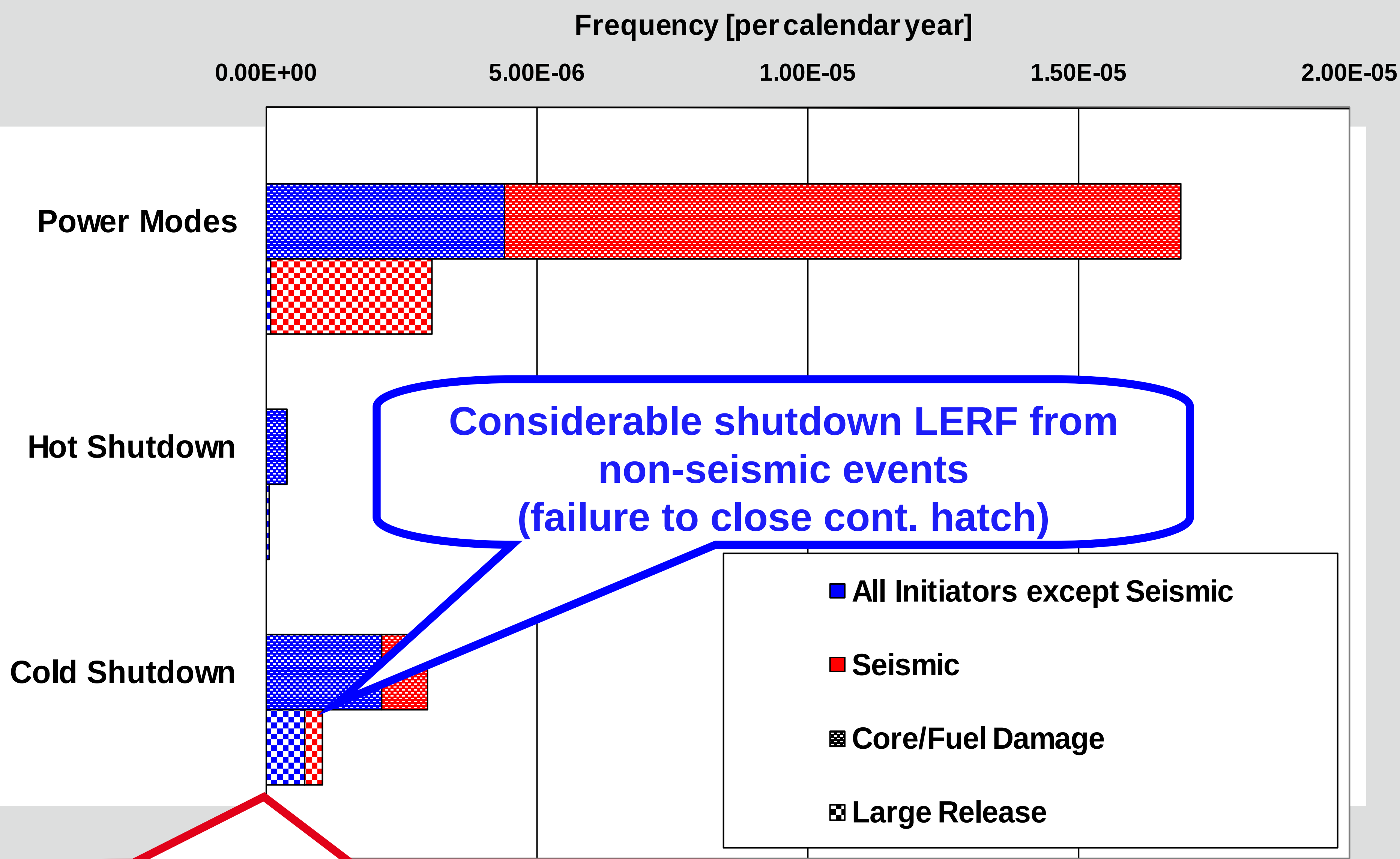
AM Part:

- Emergency staff overtakes control
- Operation of fire water pumps
- Operator actions to align mobile equipment

Level 2 Part:

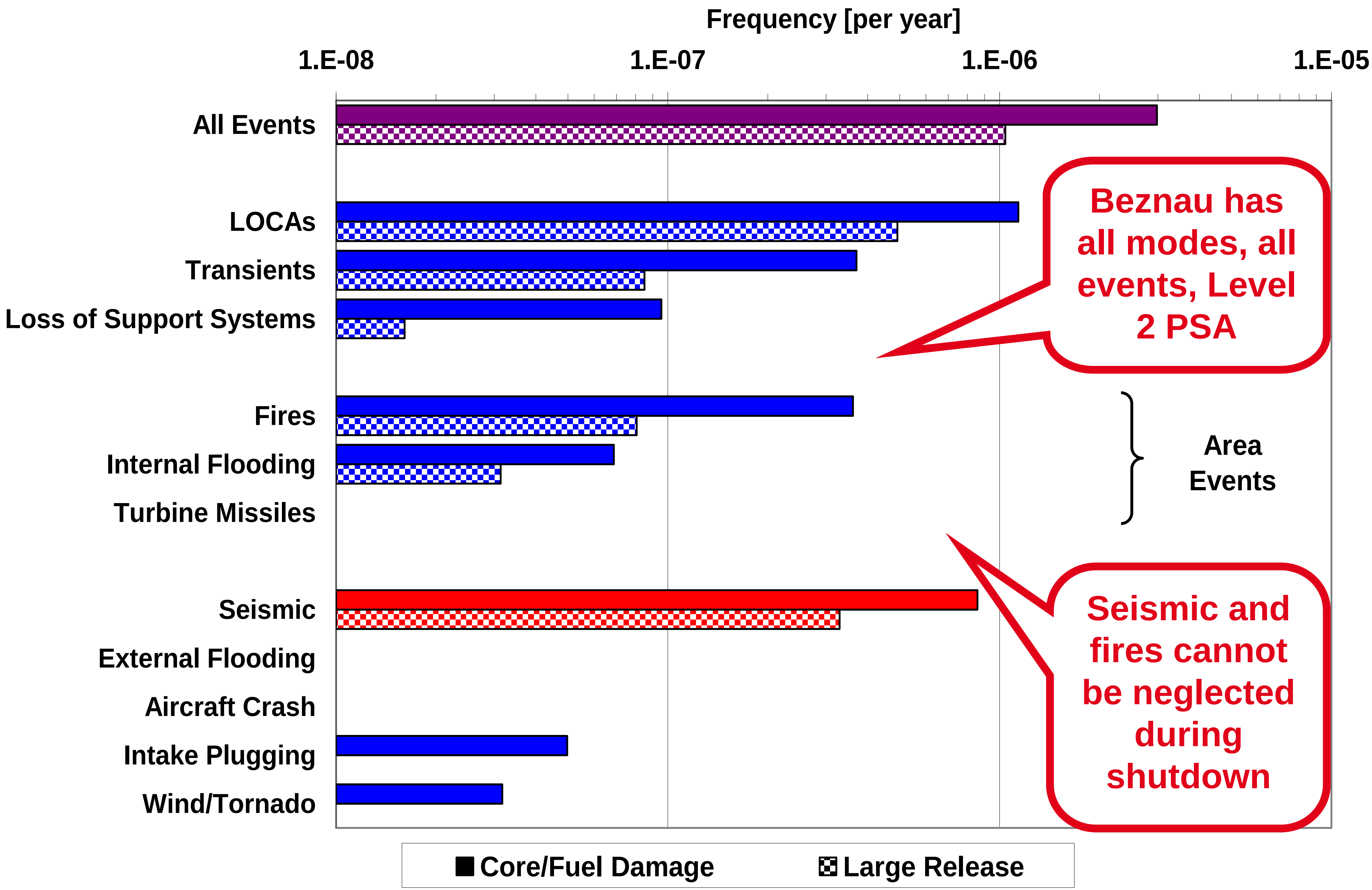
- Operator action to recover containment isolation (close hatch)
- Conditional failure rate of containment due to accident progression phenomena:
  - One single node in event tree
  - Failure rate taken from sum of failure rates of detailed full-power Level 2 PSA

4. Results of Beznau Shutdown PSA



Shutdown Risk < Full-Power Risk

Initiator Group Contributions to Shutdown CDF and LERF



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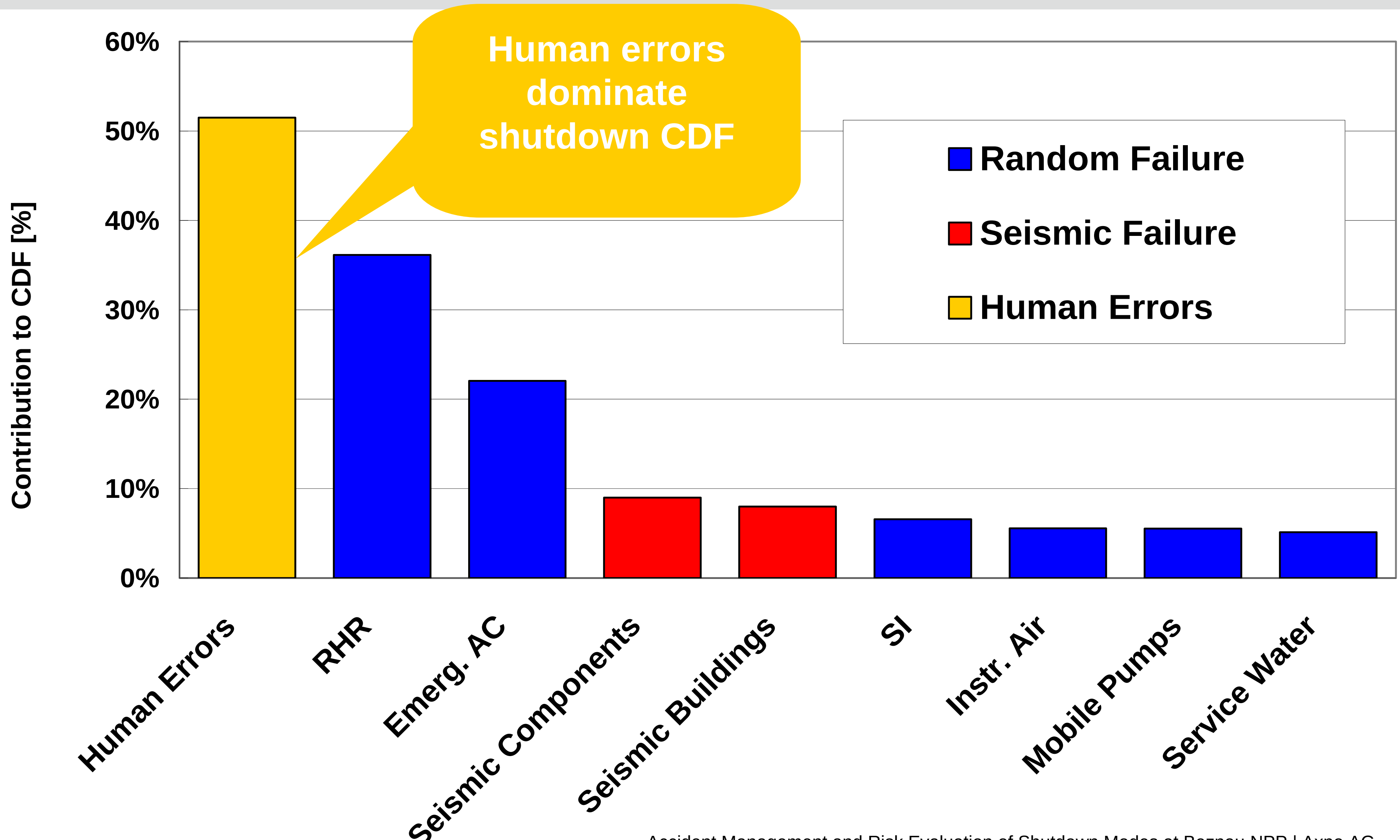
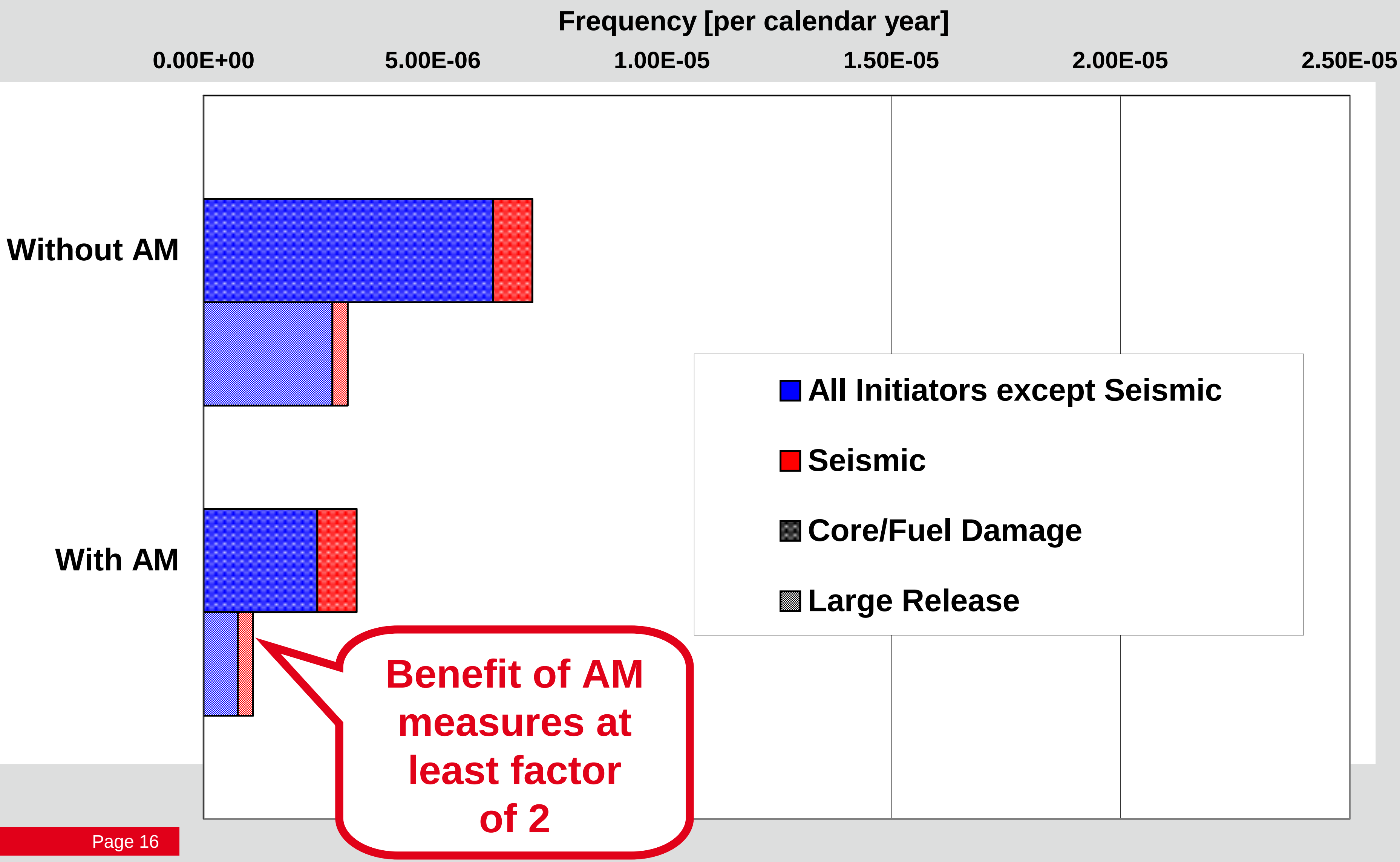
Contributions from System and Human Errors to Shutdown CDF



Cold Shutdown CDF and LERF without AM Measures



Nuclear Energy

5. Conclusions

- During shutdown modes, several conditions favor Accident Management measures to restore core cooling:
  - Long time windows
  - Core stays coolable until water level mid of core
  - Alternate indications for core uncovery
- Hardware and procedures for AM during shutdown (EOPs, SAMGs) are cost effective measures to improve plant safety
- After implementation of shutdown AM program, shutdown CDF is expected to be lower than full-power CDF
- Fires and seismic cannot be neglected during shutdown
- Simplified Level 2 PSA for shutdown modes can be performed by binning of conditional containment failure rates of full-power Level 2 PSA into one single ET node
- Shutdown LERF dominated by failure to close cont. hatch
- Shutdown LERF comparable to full-power LERF

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

Don Helton, James Chang, Nathan Siu, Kevin Coyne
US Nuclear Regulatory Commission

Mark Leonard
Dycoda, LLC

OECD Severe Accident Management Workshop
Bottstein, Switzerland
October 26-28, 2009

Presentation overview

- Current treatment of AM in PSA / PRA
- Overview of Level 2 PRA approaches
- Overview of dynamic PRA modeling
- Implementation considerations
- Potential benefits of dynamic methods for AM modeling
  - Pre-core damage benefits
  - Post-core damage benefits
  - Offsite response benefits
- Conclusions / future work

Current treatment of AM in PRA

- Historically, post core-damage operator actions are either:
  - Neglected
  - Incorporated in to subjective probability assignment
- Practical need to minimize # of sequences has outweighed desire to explicitly represent all actions
- Many applications of Level 2 PRA don’t require the degree of realism to justify more rigorous treatment

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Current treatment of AM in PRA (2)

- Studies have been conducted to assess AM effect on Level 2 PRA results
- Best practice guidance / standards encourage/require consideration of AM
  - E.g., ASME/ANS PRA standard, IAEA guidance on Level 2 PSA
- Guidance also encourages careful evaluation of viability of actions in adverse environments

Current treatment of AM in PRA (3)

- Existing approach relies on subjective mixture of deterministic analysis, experimental data and practical knowledge
- Strengths of existing approach
  - Facilitates the treatment of a large # of sequences
  - Lend well to subjective treatment
- Limitations of existing approach
  - Static event trees have difficulty with complex system/operator interactions
  - Difficulty in ensuring Level 1 / Level 2 consistency

➢ Limited-scope studies using novel methods offer an avenue for increased realism

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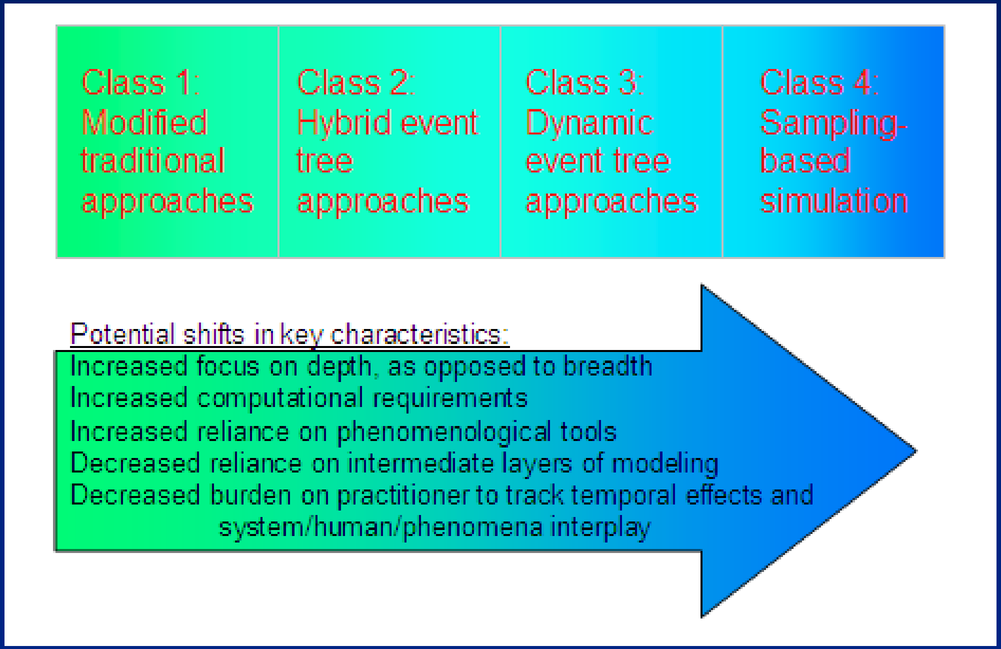
Overview of Level 2 PRA approaches

- NRC scoping study investigated potential methods
  - Traditional methods
  - Static coupling of event trees to deterministic tools
  - Dynamic event tree simulation methods
  - Sampling-based direct simulation methods
- Approach categories are broad, and implicitly include some other methods



Overview of Level 2 PRA approaches (2)

- Desirable characteristics:
  - Reduce reliance on modeling simplifications
  - Address shortcomings identified by SOARCA
  - Improve treatment of human interaction and mitigation
  - Make process / results more scrutable
  - Allow for consideration of alternative risk metrics
  - Leverage advances in computational / technology advances
  - Allow for ready characterization of uncertainty
  - Permit simplification for regulatory applications at a later time

Overview of Level 2 PRA appraoches (3)

- Approaches 3 and 4 (dynamic and sampling-based direct simulation approaches) are most promising
- Key advantages:
  - Direct use of MELCOR in event tree construction
  - Use of dynamic event trees that are not constrained to pre-determined top events
  - Direct coupling of MELCOR to the operator response model
- Approach 3 selected for further development at Sandia National Labs

Overview of dynamic PRA modeling

- Generally, dynamic methods have sought to:
  - Permit the representation of sequence evolution in a more time-based manner
  - Capture the dynamic nature of accident evolution by direct modeling of accident scenario development
    - All relevant phenomena
    - Operator decision making and actions
    - Physical accident progression
  - Use above to provide necessary context for rigorous treatment of operator decision making

Overview of dynamic PRA modeling (2)

- Numerous past efforts dating back to 1980s (discussed in paper)
  - CES
  - DYLAM
  - DETAM
- Several ongoing efforts to implement evolved approaches:
  - ADAPT/MELCOR (The Ohio State University)
  - ADS-IDAC (University of Maryland)
  - MCDET (GRS)

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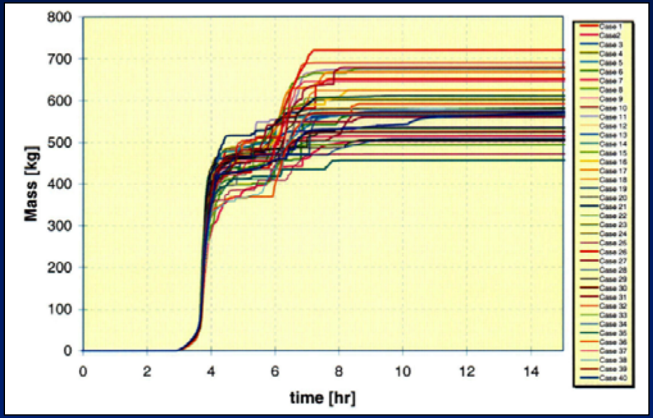
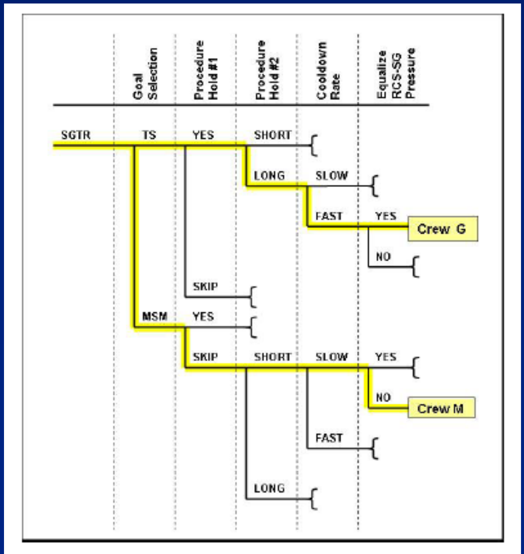
Overview of dynamic PRA modeling (3)

- Basic features of current generation:
  - Plant model (e.g., MELCOR)
    - Calculates plant response and phenomenological aspects
    - Typically include separate stochastic and deterministic sub-models
  - Crew model (e.g., IDAC)
    - Address the cognitive aspects of crew behavior
  - Simulation manager
    - Tracks branches / state transitions, calculates sequence probabilities and controls sequence development
    - Most can handle parallel processing





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Implementation considerations

- Major difficulties:
  - Exponential increase in # of unique sequences leads to:
    - Screening out “unimportant” operator actions
    - Merging sequences
    - Truncating at a prescribed frequency
  - Requires development of operator response models; very limited data for model validation
  - Application requires oversight to catch instances where the model is forced in to untested regimes
  - Strong non-linearities can magnify small errors leading to unrealistic contexts for operator actions
- Dynamic methods offer different strengths and limitations regarding uncertainty quantification

Potential benefits – pre-core damage

- Translation of beliefs regarding operator response in to computer-based routines forces re-evaluation of the basis for these beliefs
  – Relationships between behavior and the underlying reasons for the behavior must be explicit
  – Implementation in an integral simulation environment provides clearer links between actions and their proximate causes
- Core damage determination based on actual fuel response (not a pre-determined surrogate)
- Transition from EOPs to SAMGs handled on a sequence-by-sequence basis

Potential benefits – post-core damage

- Sequence-by-sequence context for SAMG decision making
- Improved consistency between Level 1 and Level 2 portions (move toward seamless Level 1/2)
- Improved resolution regarding the importance of specific operator actions on Level 3 results
- Explicit treatment of communication pathways (e.g., effects of shift changeover)

Potential benefits – offsite response

- More realistic source term owing to benefits outlined in the Level 1/2 phases
- Explicit modeling of Emergency Action Level (EAL) declarations and variability
  - Subsequent effect on timing and variability in protective actions
- Better capturing of decision making context (e.g., timing) for emergency prepraredness

Conclusions / future work

- Potential advantages exist for the use of dynamic methods in Level 2 PRA / AM
- A body of work already exists for these methods
- Additional work is needed regarding implementation of these methods
- Work by others (e.g., GRS) can be readily leveraged

Acronyms

- a.k.a. = also known as
- ADAPT = Analysis of dynamic accident progression trees
- ADS = Accident dynamics simulator
- AM = Accident management
- ANS = American Nuclear Society
- ASME = American Society of Mechanical Engineers
- CES = Cognitive environment simulation
- DETAM = Dynamic event tree analysis method
- DYLAM = DYnamic logical analytical methodology
- EAL = Emergency action level
- EOP = Emergency operating procedure
- GRS = Gesellschaft für Anlagen und Reaktorsicherheit
- IAEA = International Atomic Energy Agency
- IDAC = Information, decision, and actions in a crew context
- MCDET = Monte Carlo dynamic event tree
- MELCOR = not an acronym
- NRC = US Nuclear Regulatory Commission
- PRA = Probabilistic risk assessment
- PSA = Probabilistic safety assessment
- SAMG = Severe accident management guideline
- SOARCA = State-of-the-art reactor consequence analysis project

Overview of the Modelling of Severe Accident Management in the Swiss PSAs

Vinh N. Dang
PSI, Villigen, Switzerland

Gerhard M. Schoen, Bernhard Reer
ENSI, Villigen, Switzerland

OECD/NEA Workshop on „Implementation of SAM Measures“, October 26-28, 2009
Schloss Böttstein, Switzerland

Outline: Part 1

Regulatory Basis

Status of Implementation

- **SAMG**
- **PSA**

SAM Actions in the PSA

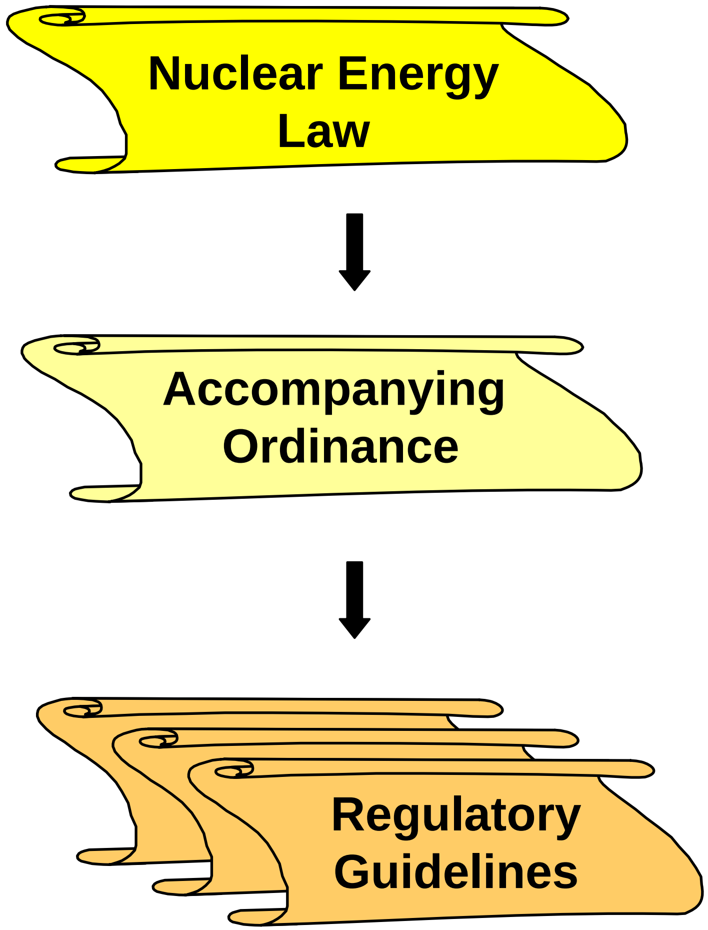
Part 2

Overview of modelling approaches and results

Performance context for SAM actions

Summary and outlook

Regulatory Basis



Regulatory Basis

Accompanying Ordinance

SAMG

- **Decision Guidance** for severe accident management

PSA

- **PSA** for relevant operating modes

- The **risk impact** of plant modifications, findings and events is to be assessed systematically.

Regulatory Basis

Regulatory Guideline ENSI-A05

- **PSA: Quality and Scope**

Regulatory Guideline ENSI-A06

- **PSA: Applications**

Regulatory Guideline ENSI-B12

- **Emergency Preparedness for Nuclear Installations**

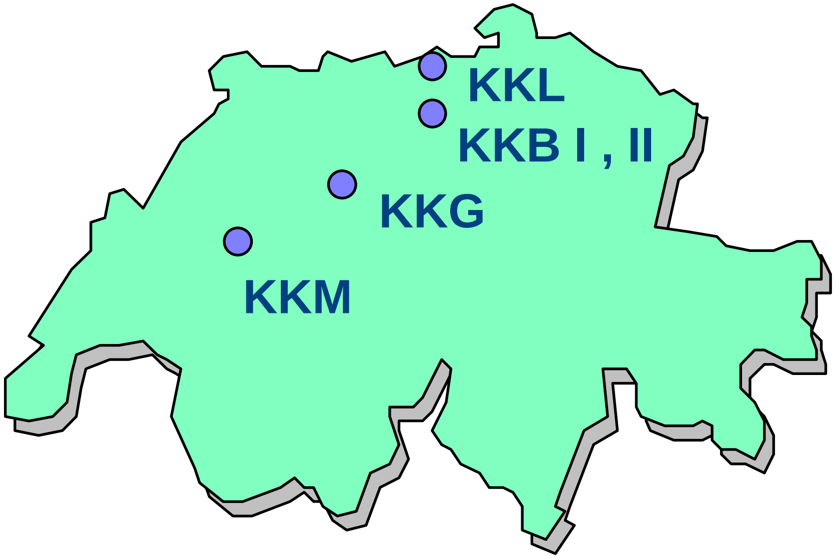
Status of Implementation: SAMG

- **1997** Start of the survey study
- **1998** General requirement to implement SAMG
- **1999** Finalization of the survey study
- **2000** Detailed specification of the requirements
- **2005** SAMG is anchored in the ordinance
- **2009** Detailed requirements are stated in regulatory guideline

Status of Implementation: SAMG

	KKB	KKG	KKL	KKM
SAMG (Full-Power)	2001	2006	2004	2004
SAMG (Shutdown)	2005	2006	(2009)	2007
Exercises	FP	FP	FP	FP

Status of Implementation: PSA



⇨ **4 plant-specific PSA Models**

Status of Implementation: PSA

Scope of the PSA Models:

Full-Power		LP & S	
L-1	L-2	L-1	L-2
External Events			
Internal Events			

SAM Actions in the PSA

SAM actions : = SAMG-guided actions to

- terminate core degradation,
- ensure containment integrity, and
- mitigate radiological releases.

Scope of the paper: SAM actions modelled in **Level 2 PSA** for **full power**

➡ **SAM actions**

SAM Actions in the PSA

Examples of SAM Actions

- Alternative water supplies, especially alignment of firewater
- Flood for heat removal, e.g. of the reactor pressure vessel, of the drywell
- Flood or spray for radionuclide retention
- Containment venting

SAM Actions in the PSA

	KKB	KKG	KKL	KKM
Types of Actions	10	7	1	3
Cases	34	19	2	7

Some PSAs are currently being updated or reviewed.

Conclusion (Part 1)

- Introduction of SAMG and associated training have contributed to an increased reliability of SAM actions.
- To obtain a realistic estimate of risk, it is important to model SAM actions in PSA.
- The development of SAMG and Level 2 PSA can be an iterative process.
- Three sites have implemented SAMG for shutdown and two consider SAMG in the Level 2 PSAs for shutdown.

Part 1

Regulatory Basis

Status of Implementation

- SAMG
- PSA

SAM Actions in the PSA

Part 2

Overview of modeling approaches and results

Performance context for SAM actions

Summary and outlook

Credit for SAM actions – types and cases

	KKB	KKG	KKL	KKM
Types of Actions	10	7	1	3
Cases	34	19	2	7

Modelling approach

HRA-type analysis (KKB, KKG)

APET model (KKL, KKM)

Some actions related to SAMG measures are not (yet?) credited in the PSAs

- as mentioned, on-going updates

Quantification of SAM Actions

"HRA-type" analysis

Probability of failure

- "diagnosis" / decision
- Implementation

Accident Progression Event Tree (APET) questions

engineering/expert judgment process

Probability of non-occurrence

- Will the ERT decide that a given SAM measure is optimal? Mitigation strategy
- Successful manual implementation of the measure
- Availability of the hardware

Dependence

- On failure of preventive actions (L1 HFEs)
- Among mitigative actions

Probabilities assigned to SAM actions in the surveyed PSAs

	P(failure) or P(non-occurrence)				
	< 0.001	0.001 < p < 0.01	0.01 < p < 0.1	> 0.1	**Total**
KKB	0	2	20	12	**34**
KKG	0	2	14	3	**19**
KKL	0	0	0	2	**2**
KKM	0	0	4	3	**7**

Performance context of SAM actions (1)

Positive factors, supporting success

- Transition to new, mitigation-oriented objectives
- Increased expertise available to and within ERT

Differences that need to be considered

- Open (by necessity) aspects of the mitigative response plan
  - Some decisions must be made in-situation
- Increased uncertainty regarding plant state
  - Symptoms do not correspond as tightly to known states
- Need for more parties to agree (for some measures)
- Personnel radiation exposure (local actions)
- Some dependencies possible due to need to rely on CR crew for information

Performance context of SAM actions (2)
Some elements relevant for HRA modeling

- Non-prescriptive nature of the guidance
  - ➢ Judgments left to the ERT (by design)
- Strategy selection
  - – Judgment of whether the SAM measure could be effective in the given severe accident condition
  - – Considered in APET. To what extent can it be (is it) addressed in HRA-type analysis?
- Option selection for a specific SAM measure
  - – One option (when many are available) is frequently modeled
- Factors affecting potential dependence of SAM actions on previous HFEs need further study
  - – New set of decision-makers should reduce dependence
  - – Their assessment, at least initially, will not be independent

Conclusion – Modelling of SAM actions (Part 2)

Part 1 conclusions (slide 13)

Important to model the actions and measures supported by SAMG in Level 2 PSA

But there are challenges

- **Uncertainties faced by ERT** in assessing plant state and expected accident progression
- **In-situation strategy** selection (informative, non-prescriptive guidance)
- **Dependence** factors
- **Option selection**, given a SAM measure has been selected
- **Timing** of decisions **–** more parties have to agree

- **Differences in PSAs may reflect**
  - differences in SAMGs **or**
  - different analyst views on the key factors

Extended use of MERMOS to assess Human Failures Events in Level 2 PSA

H. Pesme, P. Le Bot

ISAMM workshop Schlöss Böttstein
Oct. 27, 2009

CHANGER L'ÉNERGIE ENSEMBLE

Summary

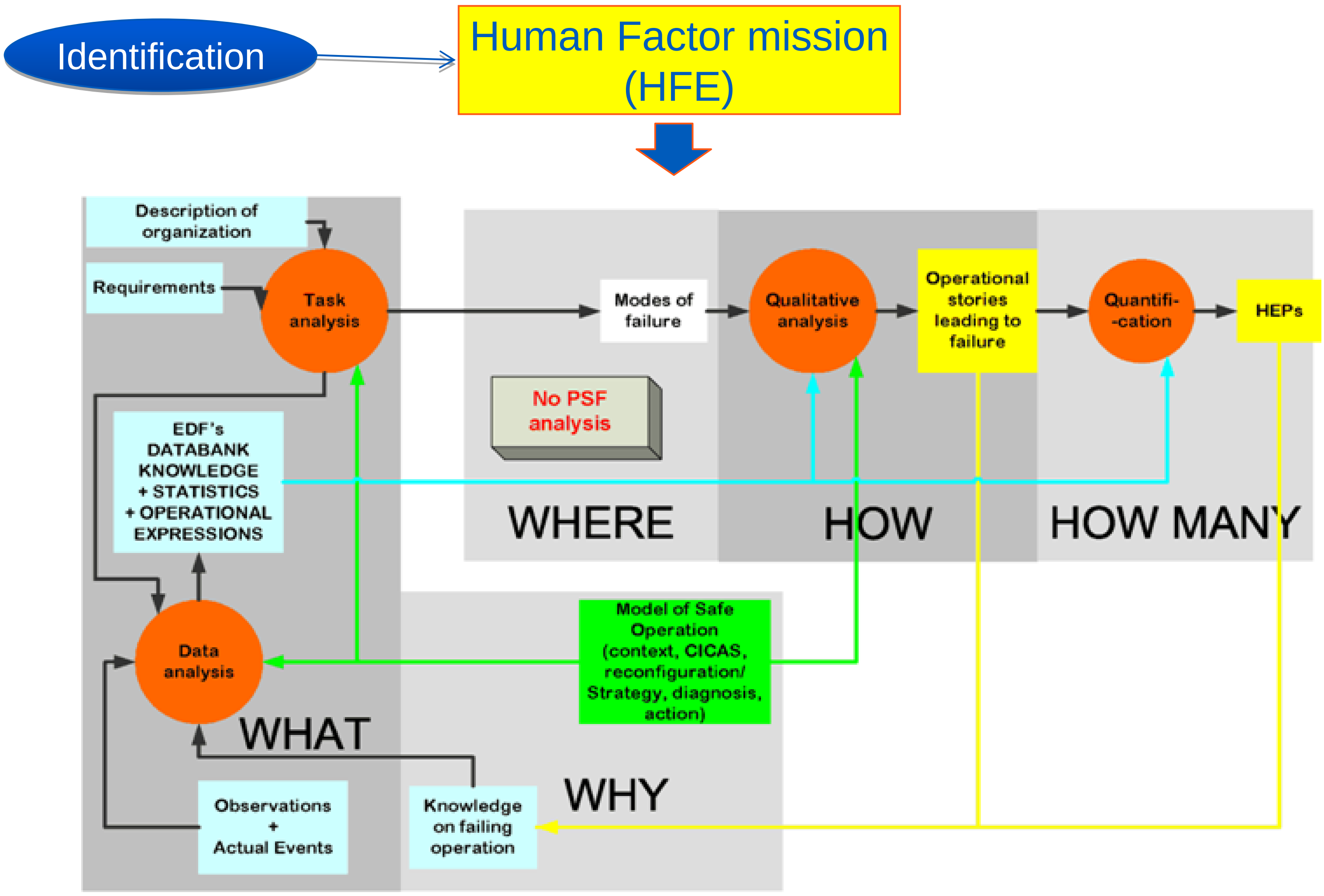
- MERMOS methodology
- LEVEL 2 specificity: the national crisis team
- Contribution of severe accident experts
- Example
- Conclusion

HRA methodology (MERMOS)

MERMOS/PSA	Simplified approach	Statistical approach	Detailed approach
Pre initiator	**MERMOS** pre initiator simplified (from FH7, under development)	**MERMOS** pre initiator statistique	**MERMOS** pre initiator détailed (under development)
Initiator		**(FH7)** Future development	
Post initiator	**MERMOS** post initiator forfaitaire	**Observation guide for MERMOS** **Time related curves**	**MERMOS post initiator detailed**
Crisis organization	**MERMOS** crisis team simplified (PSA level 1) (under development)	-	**MERMOS detailed (PSA level 2)**
Fire (under development)	**MERMOS** Fire screening	**MERMOS** Fire fighting statistical	**MERMOS** Fire fighting detailed **MERMOS** Fire operation detailed
Seism, Flood …		Future developments	

+ Application frame (choice of methods to take into account project constrains & specific objectives, HRA team organization …)

Input: HFE / Output: Quantified scenarios of failure

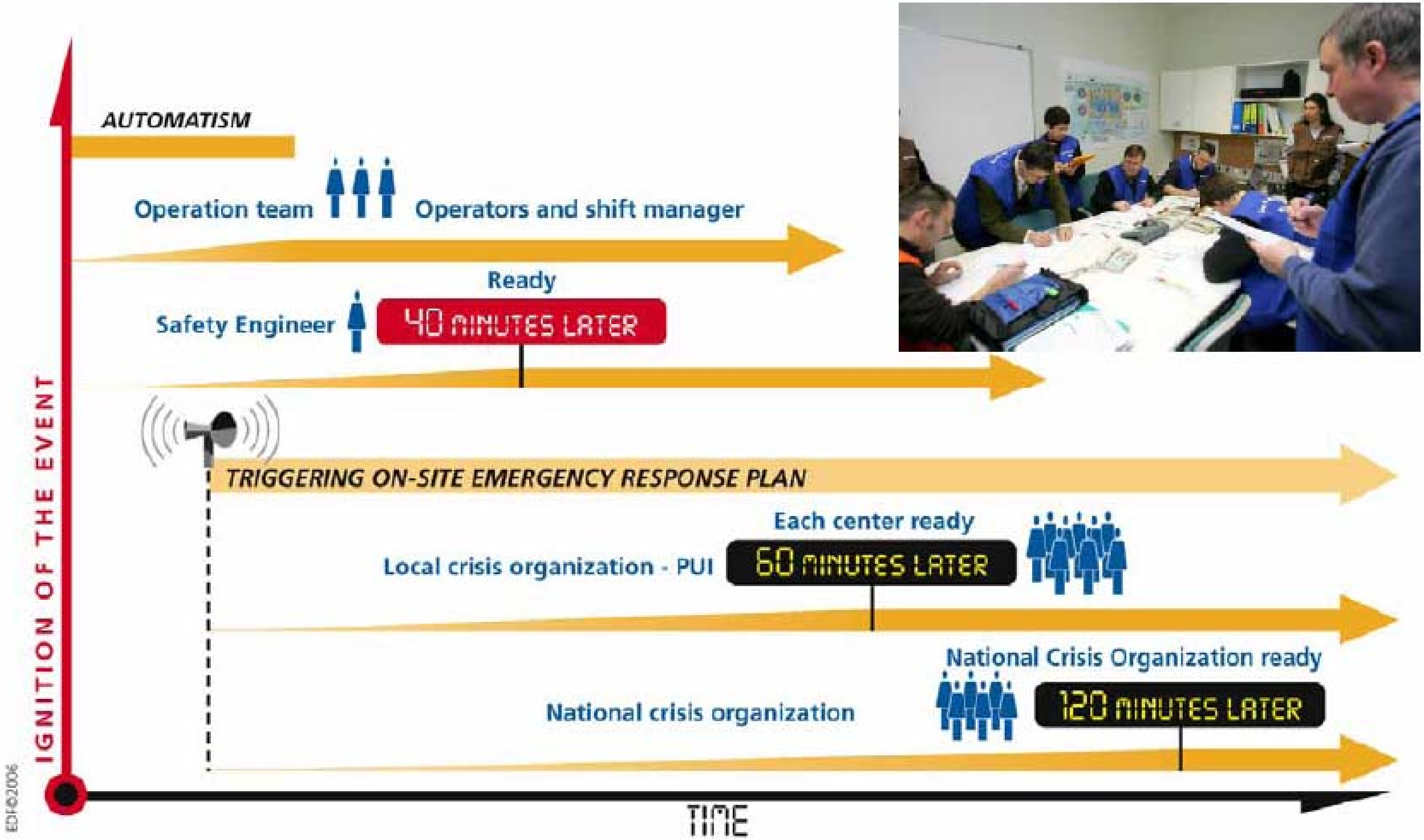


Contextual and systemic : MERMOS (post-initiator HFE : level 1 & 2, fire, precursor analysis)

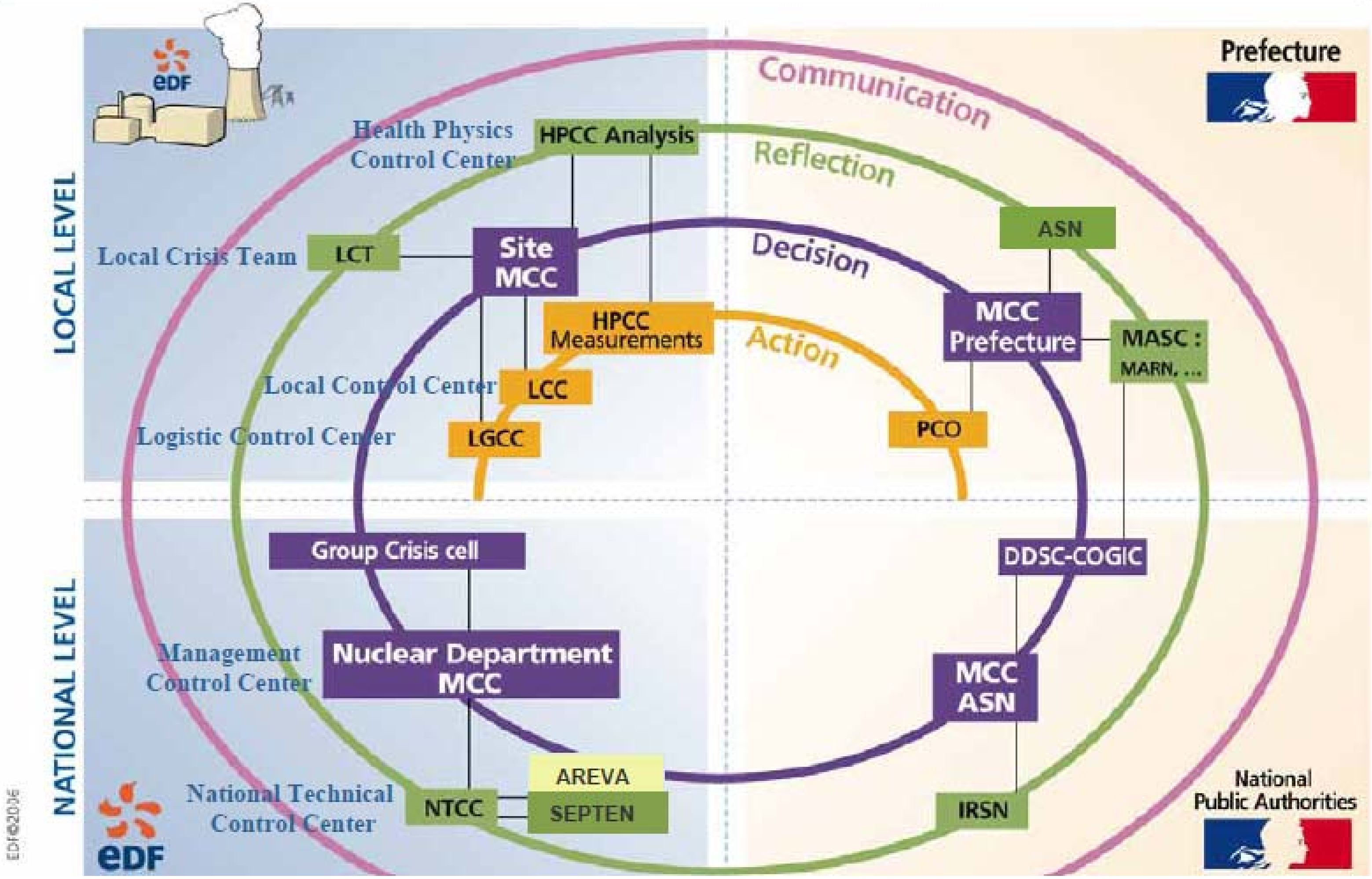
LEVEL 2 SPECIFICITY: THE NATIONAL CRISIS TEAM

(some slides from EDF presentation at the International Symposium on Seismic Safety
Feb. 27, 2008)

ACCIDENT MANAGEMENT
Building up Crisis Organization



NATIONAL CRISIS ORGANIZATION
Global Scheme



ON-SITE EMERGENCY RESPONSE PLAN TASKS

Decide for

- Staff protection
- Actions to manage

Alert, inform and communicate

- Local public Authorities
- National public Authorities
- National crisis Organization
- EDF Group
- The medias

PUI tasks

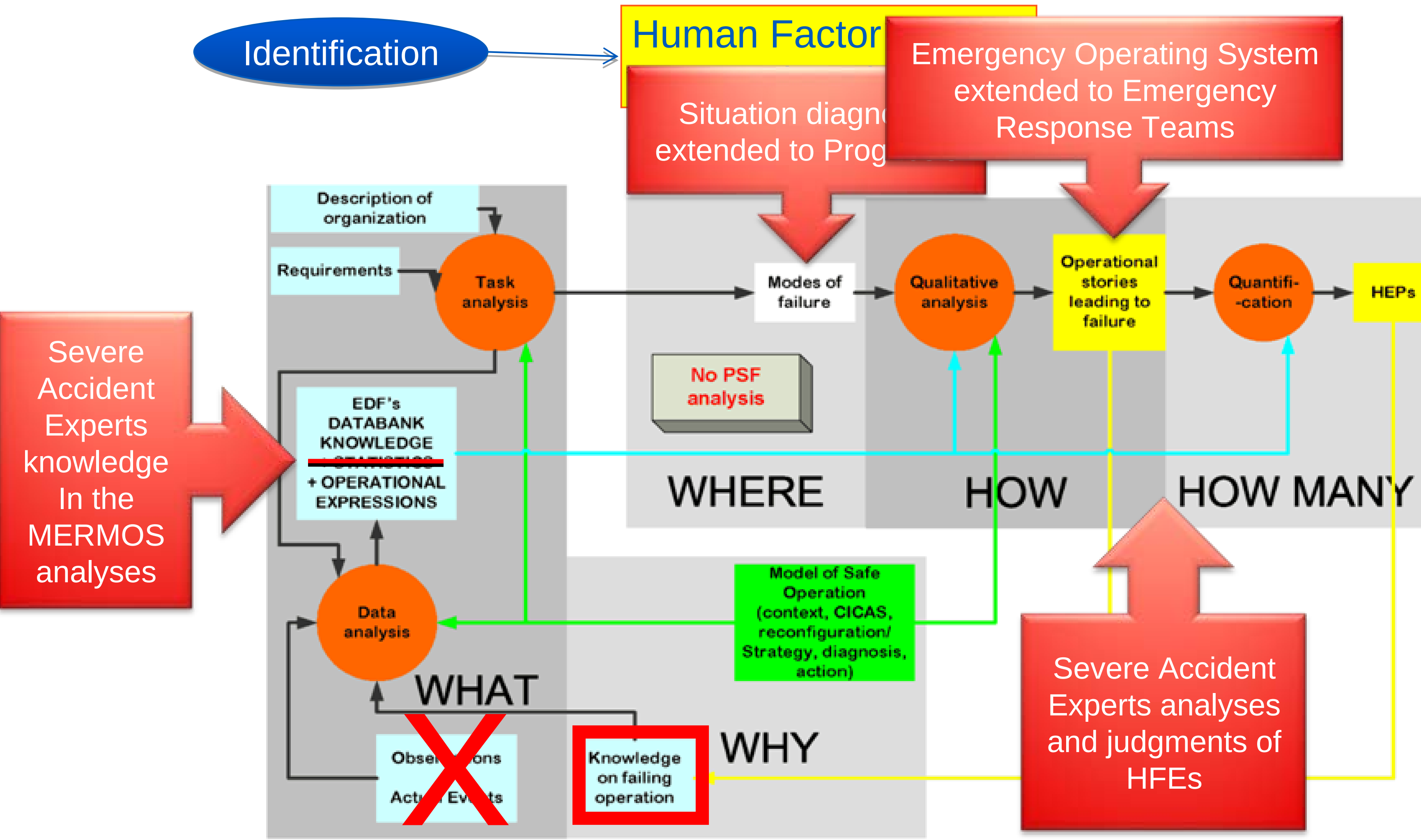
Act

- To make a diagnosis of the incident
- To operate the unit to the best safety status
- To stabilize the situation
- To limit consequences of the accident
  - on the environnement
  - on the installation

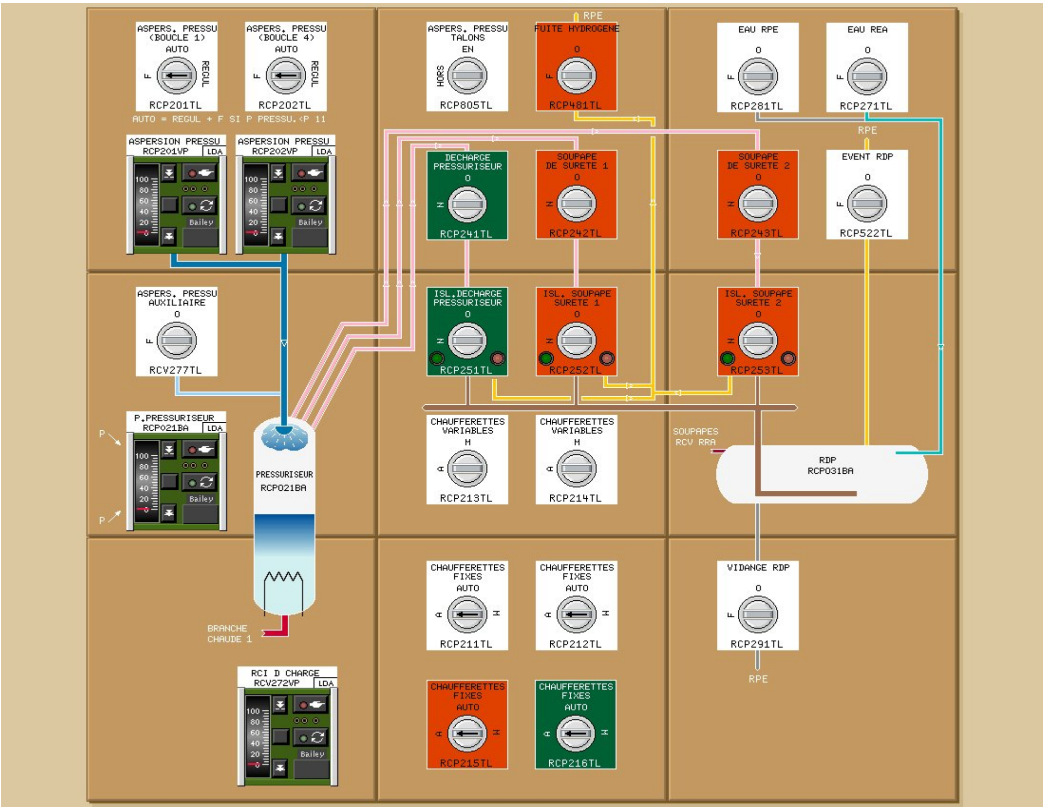
Technical reflection

- To make a prognosis of its evolution
- To assess radioactive releases
- To evaluate its impact on the population

How to take into account the Crisis organization ?



Contextual and systemic : MERMOS (post-initiator HFE : level 1 & 2, fire, precursor analysis)



MERMOS ANALYSIS PROCESS

Goal of the analyst

- To build (and upgrade) the answer to the question :
  - How could the Emergency Operation System fail ?
  - In rare situations and in a plausible way
  - By describing operational stories leading to failure (= MERMOS scenarios)

Structure of MERMOS analysis / quantification

$$P(HFE\ failure) = P_{residual} + \sum_{i=1\ to\ n} P(scenario\ i)$$

Example : HFE assessed with Severe Accidents Experts

Loss Of Feed Water + station blackout

Primary cooling system depressurisation by opening pressurizer valves in less of 15mn after Core Temperature = 1100°C (~3h from initiator)

Probability of mission failure (HEP):	**1.1 E-1**
Residual Probability	6. E-5

N°	Scenarios	Prob.
1	Suspension of actions until the diagnosis is carried out	8.1 E-4
2	Suspension of actions until their prioritization is done	2.7 E-6
5	Valve opening is not correctly confirmed	8.1 E-3
6	Power supply to the valves is not restored in time	8.1 E-2
8	Team is waiting for confirmation from the Main Control Center to depressurize (MCC is waiting for local emergency response team which is not available in time)	7.3 E-4
9	Team is waiting for confirmation from the Main Control Center to depressurize (MCC is not available in time)	2.2 E-2

No scenario identified for Wrong Strategy, No Action, Wrong Prognosis, No Prognosis

EDF

Scenario structure / quantification

$$P(scenario\ i) = P(context) \times P(operation)_{/context} \times P(non\ reconfiguration)$$

Context (or situation)

- Conjonction of situation features
- Given the initiator and aggravating events

Operation (given the context)

- Main features of the emergency operation : configuration and orientation of the EOS (coherent and justified) = CICAs

Non reconfiguration

- Wrong operation is lasting too long

Example of MERMOS scenario

SCENARIO n° 9	Probability: 2.2 E-2
Description : Team is waiting for confirmation from the Site Main Control Center to depressurize (Site MCC is not available in time)	

Situation features	
Following its procedure, the Safety Engineer needs the confirmation of Site MCC	0.9
Site MCC decision id delayed	0.3
Local Crisis Team does not decide instead of Site MCC	0.9
National Technical Center is not able to help	1

CICA (Main features of the emergency operation)	
Delegation of decision to Site MCC	0.9

No reconfiguration probability :	0.1

Conclusion

- Qualitative aspects from Severe Accident Experts participation
  - Obviously, in the example high failure probability given the time to act
  - In the two analyzed HFEs, complexity of decision circuit appears to be the weakness of the help of Crisis Organization

- With the help of MERMOS analysts,
  Severe Accident Experts produced knowledge about Level 2 HFE failures by contributing to MERMOS analyses
  - Two HFE completed as examples and references for by-delta new analyses

Session 4

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

Jason H. Schaperow, Mark T. Leonard, Charles G. Tinkler, K. C. Wagner

Presented at the OECD/NEA Workshop on Implementation of Severe Accident Management (SAM) Measures

October 26-28, 2009

Outline

- Overview of SOARCA Study
  - Background
  - Objectives
  - Approach
  - Conclusions
- Accident Progression and Source Term
- Peer Review

Background

- NRC security studies performed following 9/11 incorporated severe accident research performed over the last 2 decades
- Security studies confirmed that earlier accident consequence studies were conservative to the point that predictions were not useful for characterizing results or guiding public policy
- Earlier consequence studies used
  - Combination of conservative assumptions or boundary conditions
  - Simple bounding analysis

Objectives

- SOARCA study being performed to develop body of knowledge regarding the realistic outcomes of severe reactor accidents
- Incorporate significant plant improvements and updates not reflected in earlier assessments
  - System improvements
  - Training and emergency procedures (EOP/SAMG)
  - Offsite emergency response
  - Recent security-related enhancements (10 CFR 50.54(hh))
- Evaluate the potential benefits of mitigation improvements in preventing core damage and reducing an offsite release should one occur
- Enable NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders
  - Federal, state, and local authorities
  - Licensees
  - General public
- Update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development"

Approach

- Detailed
  - Includes operator actions beyond those critical to prevent core damage
    - reducing injection flow to preserve inventory
    - depressurizing RCS
  - Includes details of facility not included in previous studies – modeling fission product deposition in buildings adjacent to containment
  - Detailed nodalization of core and RCS
- Best-estimate
  - Represents the most likely outcome for uncertain behavior
    - Avoids biasing answer in conservative or non-conservative fashion
  - Models high-temperature failure of RCS components (BWR SRV sticking open, PWR hot leg rupture following thermally induced SGTR)
- Self-consistent
  - Integrated MELCOR analysis
  - Accounting for all relevant systems, subsystems
  - Scenario-specific EP

Approach

- Integral
  - Single code (MELCOR) provides feedback among phenomenological models and operator actions
- Current scientific knowledge and plant capabilities
  - MELCOR validation includes the latest tests such as PHEBUS and VERCORS
  - Results of ARTIST tests of fission product deposition reflected in the analysis
  - Latest security-related mitigation measures (10 CFR 50.54(hh)) credited in the analysis

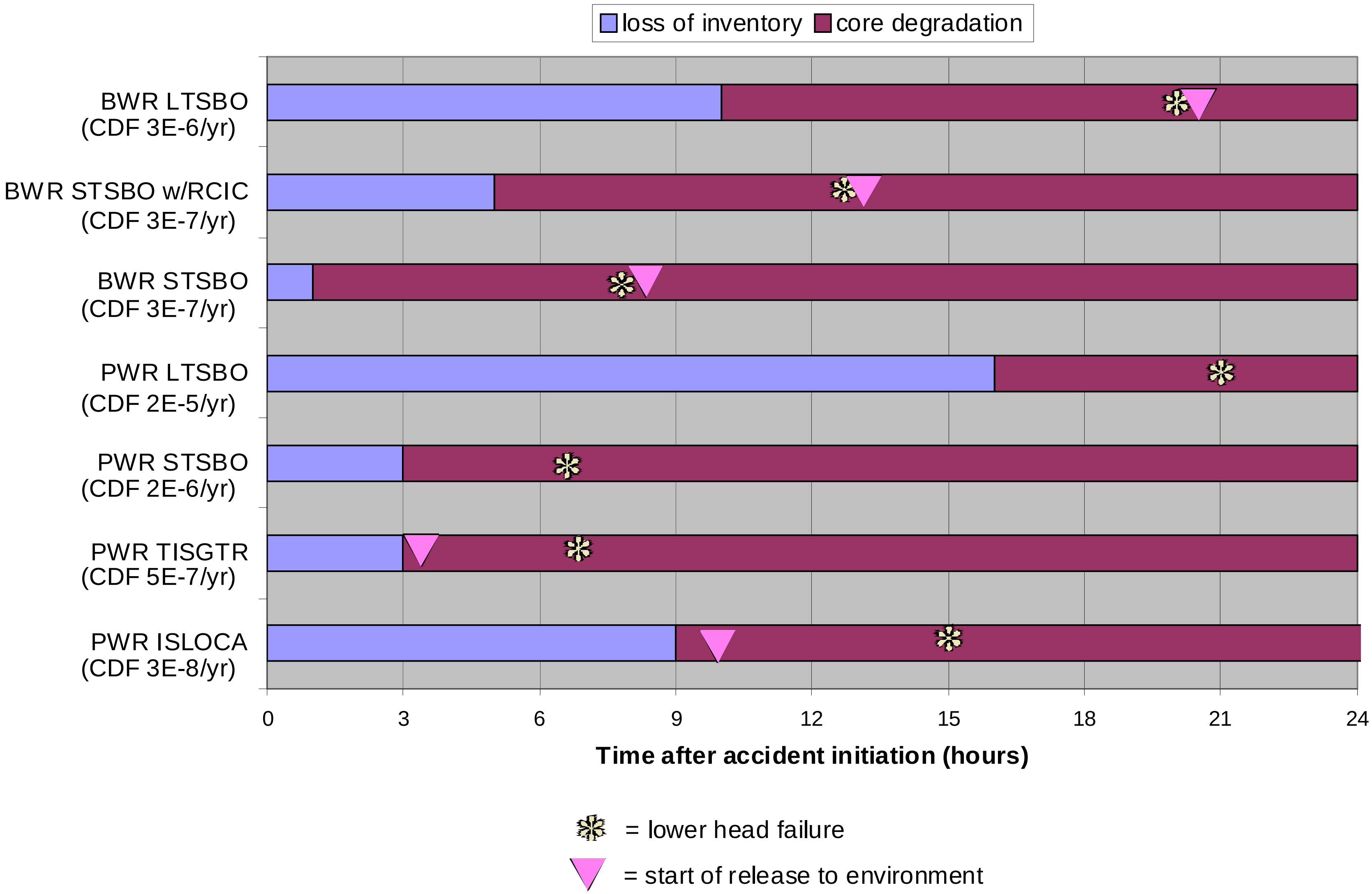
Conclusions

- SOARCA represents major change from the way people perceive severe reactor accidents and their likelihood and consequences
  - Mitigation is likely (due to time, redundancy, diversity) and, when it is implemented, effective in preventing core damage
    - Impact on existing level 1 PRA
  - Unmitigated accidents progress more slowly with smaller releases, no LERF
    - Impact on existing level 2 PRA
  - Scenarios have lower frequency and lower consequences – lower risk
  - Dominance of external events suggests need for corresponding PRA focus
    - Seismic research needed

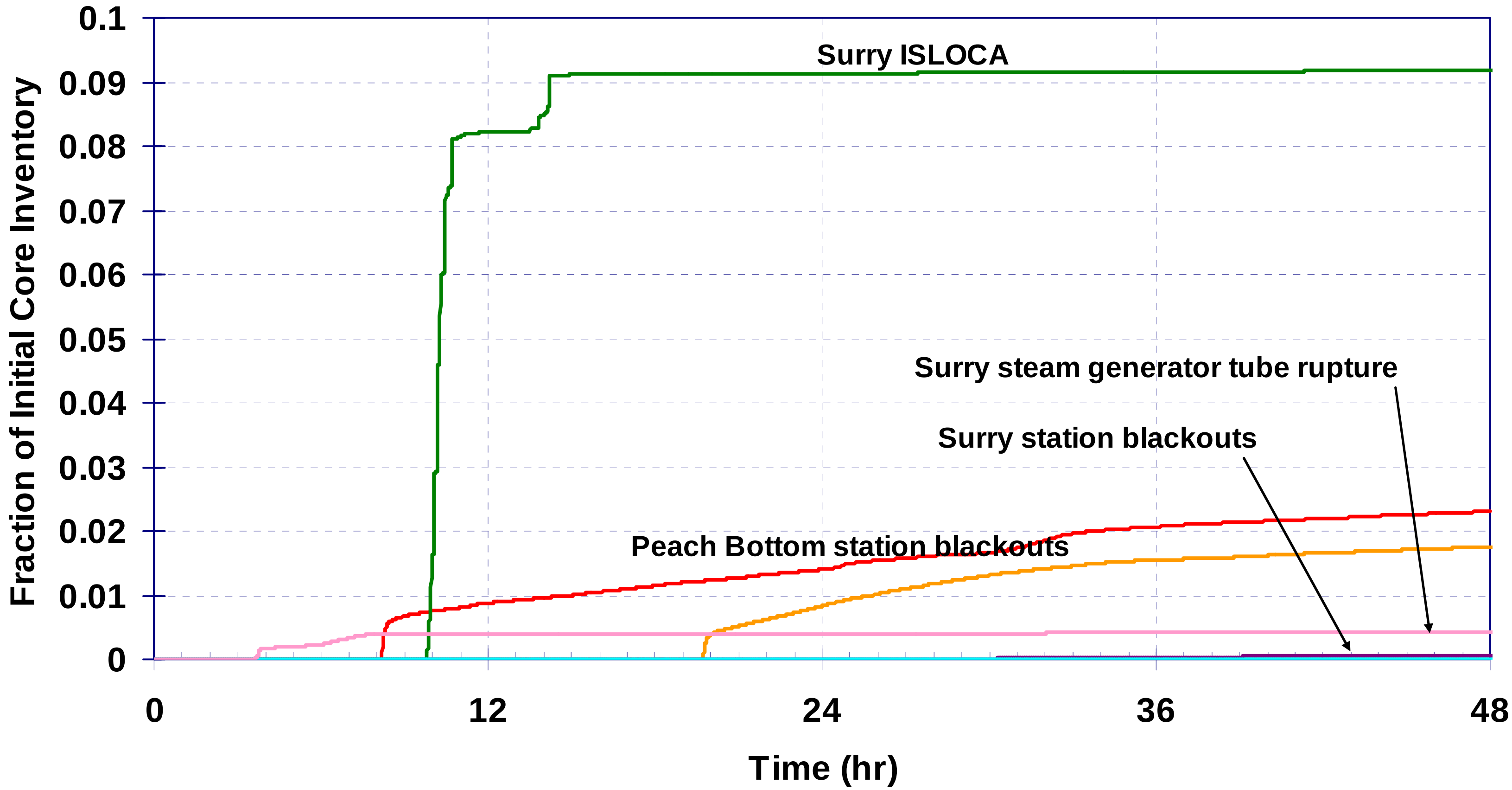
Accident Progression and Source Term

- SOARCA concluded mitigation is likely and effective in preventing core damage
- SOARCA also analyzed these same scenarios assuming they proceed unmitigated
  - To quantify benefit of mitigation measures (risk averted)
  - To provide basis for comparing to past analyses of unmitigated severe accident scenarios

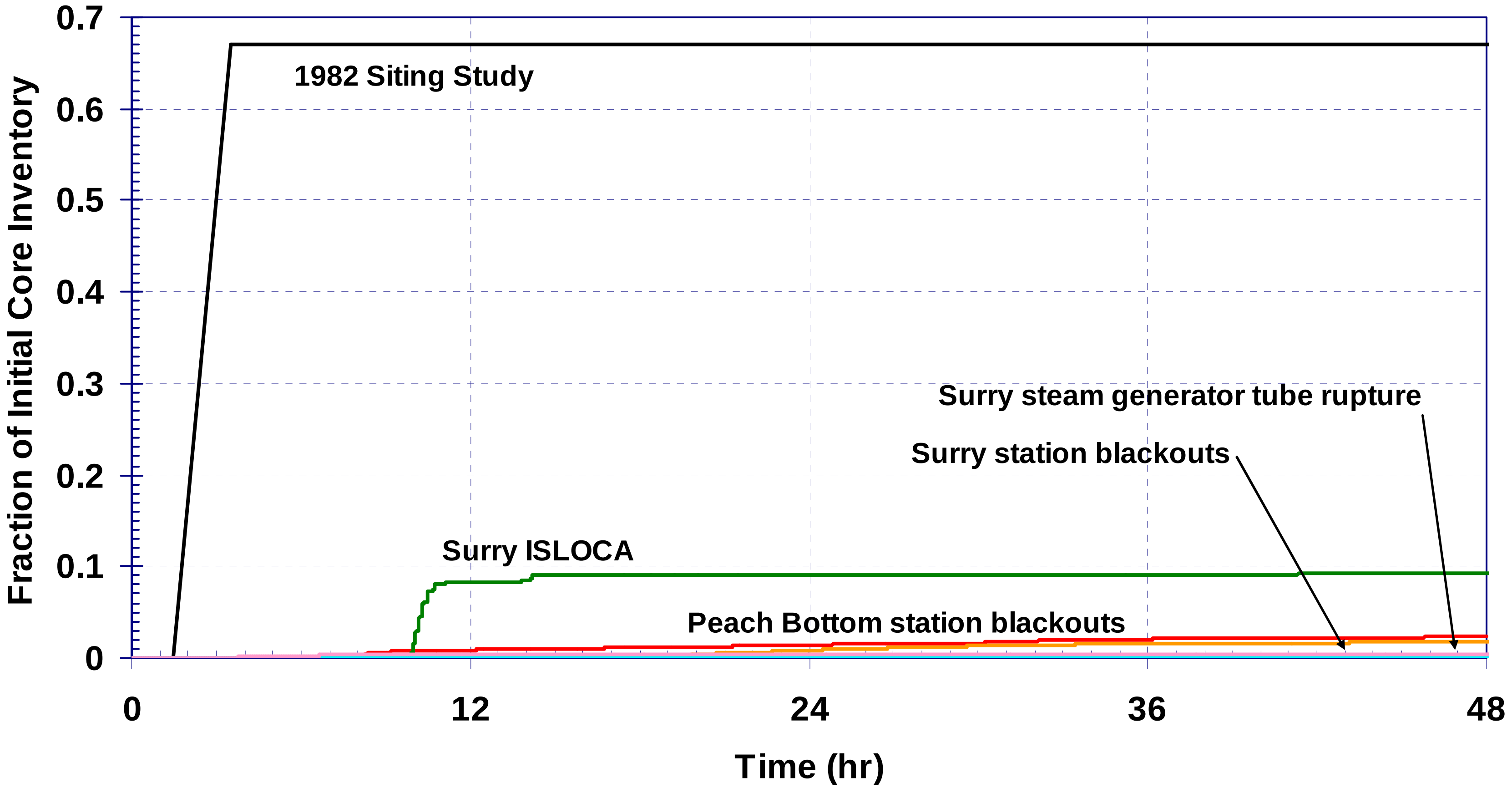
Accident Progression – Key Timing for Unmitigated Sensitivity Cases



Cesium Release for Unmitigated Sensitivity Cases



Cesium Release for Unmitigated Sensitivity Cases



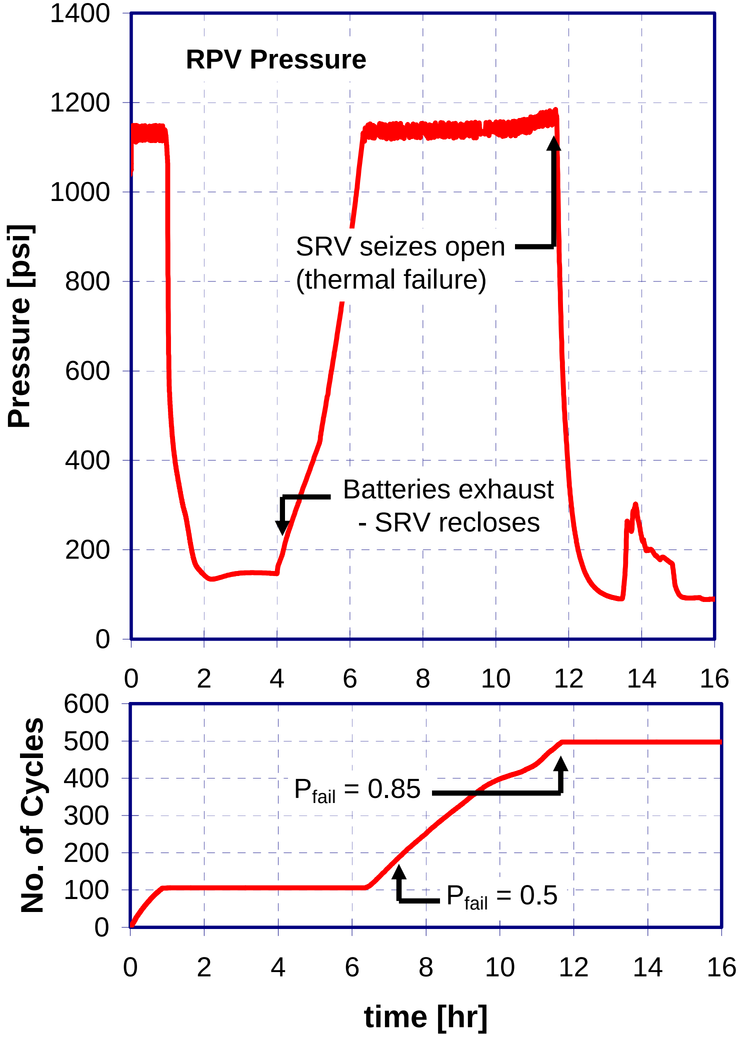
Peer Review

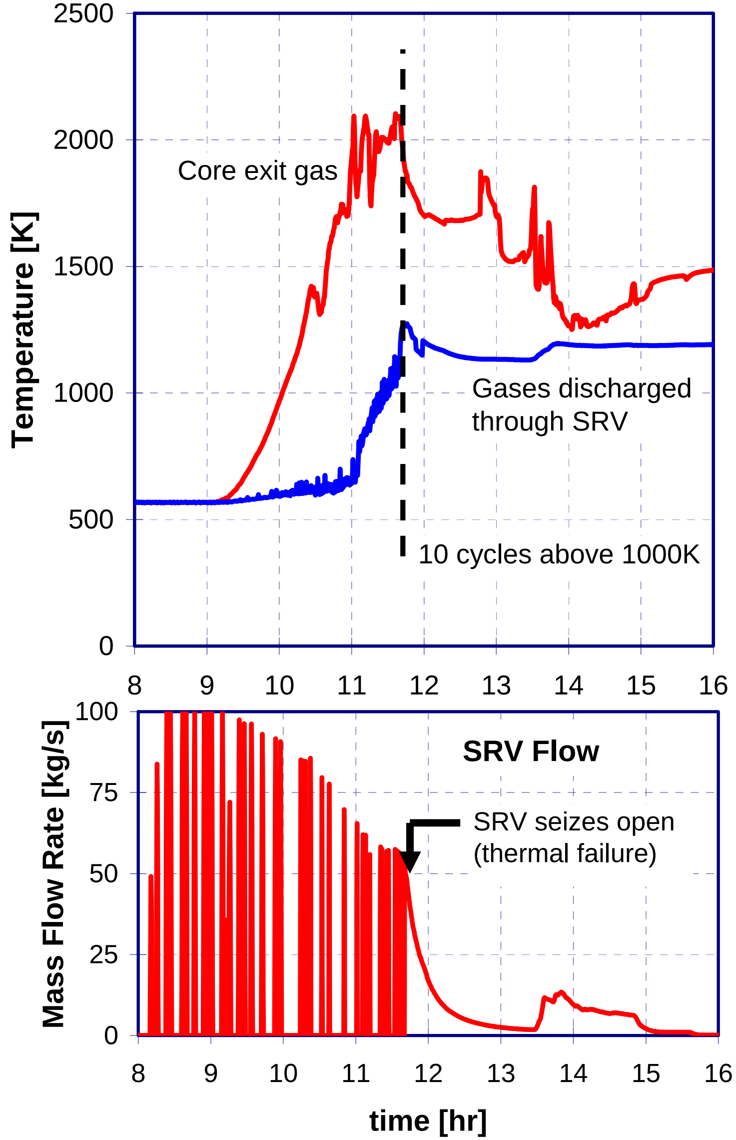
- SOARCA is being peer reviewed
- Preliminary issues raised by peer review committee
  - Safety relief valve fails open for BWR (Peach Bottom)
  - Hot leg creep rupture for PWR (Surry)
  - Alternative iodine chemical/physical forms

Safety Relief Valve Fails Open for BWR (Peach Bottom)

- Peach Bottom SBO
  - High gas temperatures during core degradation cause SRV to stick open depressurizing the RPV and transporting fission products to the suppression pool
- Preliminary peer review comment
  - Consider SRV sticking partially open or not sticking open at all
- Additional information subsequently provided to committee
  - Multiple natural mechanisms for early RPV depressurization are represented in the Peach Bottom MELCOR model
    - Stochastic failure of a cycling SRV to re-close
    - Thermal seizure of an SRV in the open position
    - Steam line or nozzle creep rupture
  - Partial open/closed positions not considered due to valve design and operation
  - Thermal seizure was the ‘lead’ or first mechanism to occur in the SOARCA calculations, but the others would follow shortly

Failure of lead SRV to Reclose -- LTSBO



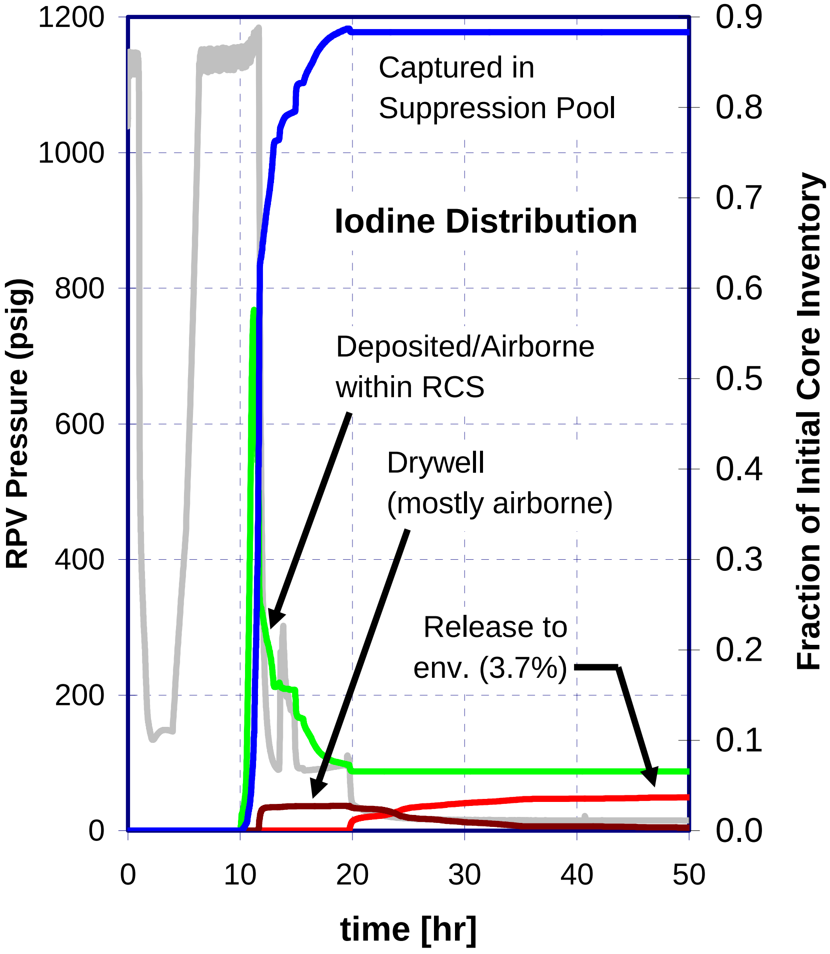


Stochastic **Thermal**

Safety Relief Valve Fails Open for BWR (Peach Bottom)

<u>Conclusions</u>

- Earlier time of depressurization possible if lower confidence level for stochastic failure is assumed
  - ‘Sweep-out’ of RPV airborne aerosols to suppression pool may be delayed until debris enters water in RPV lower head
- Later time highly unlikely due to confluence of active failure mechanisms at the time thermal seizure occurs in best estimate model (12 hrs in the LTSBO)
  - Several hour delay would be necessary to preclude depressurization prior to VB

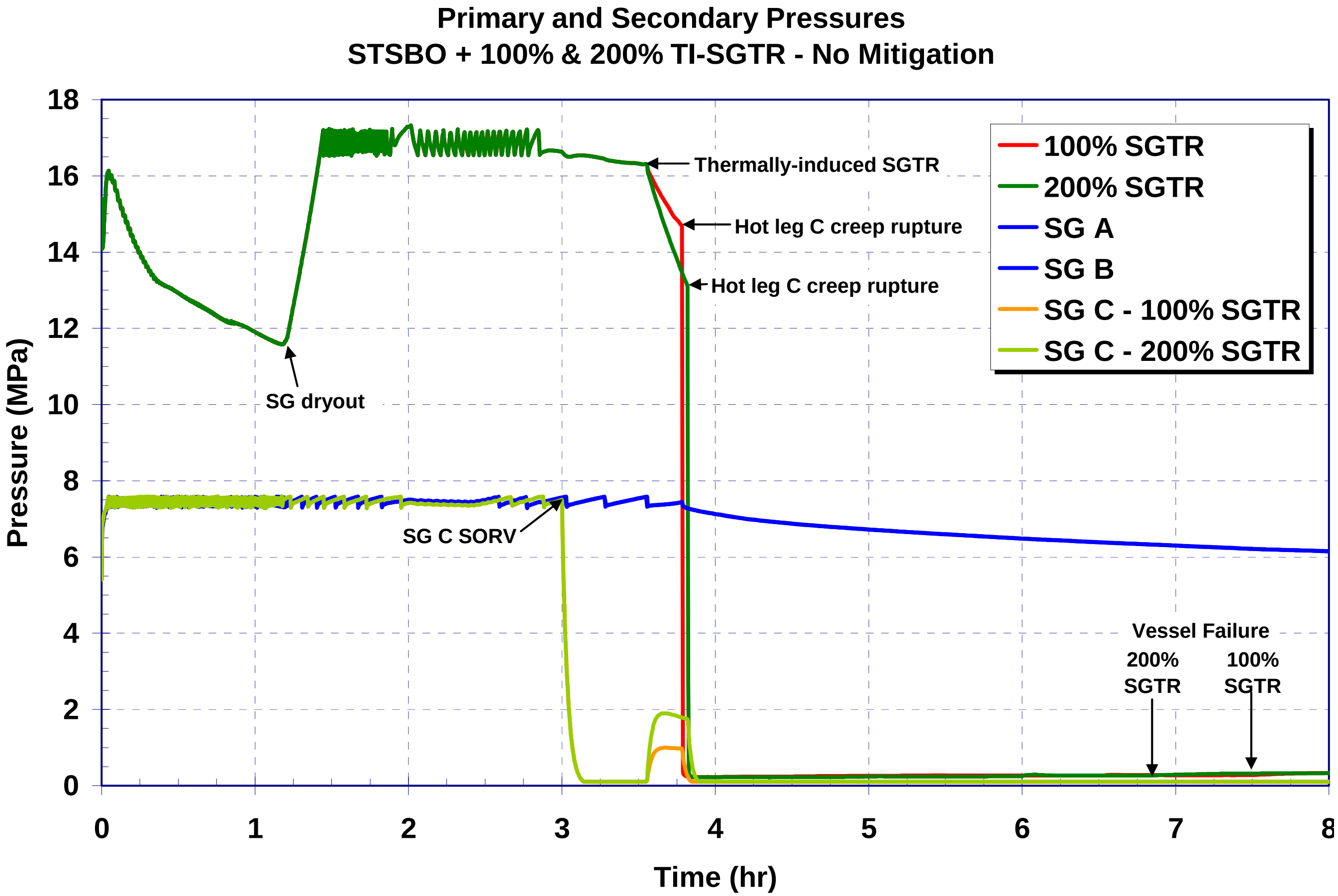


Hot Leg Creep Rupture for PWR (Surry)

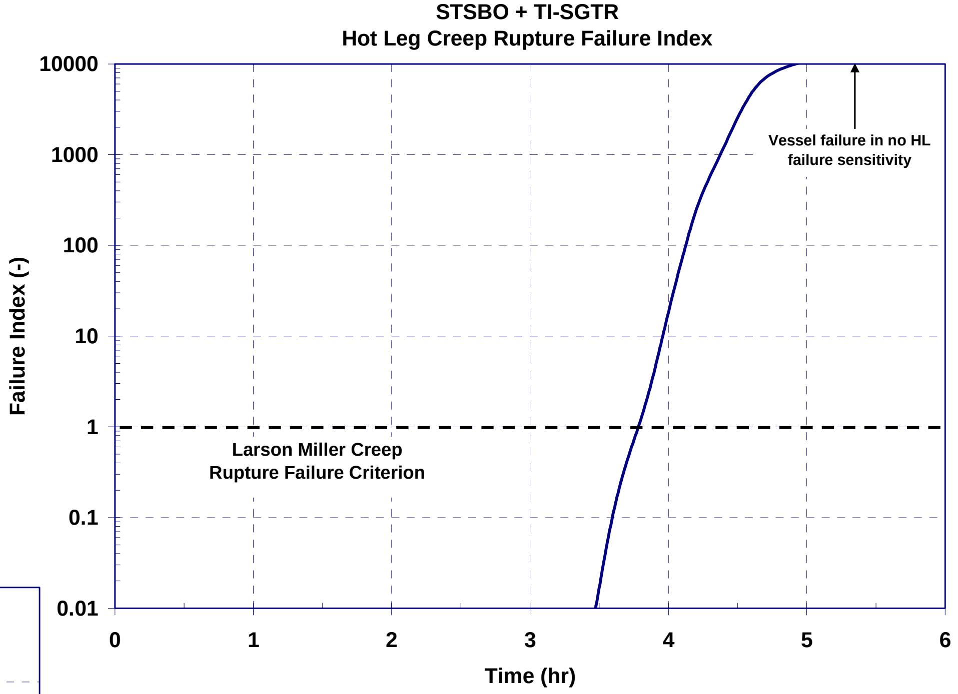
- Surry SBO with TISGTR
  - High gas temperatures during core degradation cause hot leg creep rupture depressurizing the RCS and transporting fission products to the containment
- Preliminary peer review comment
  - Consider uncertainty in the time of the hot leg creep rupture

Uncertainties in RCS Failures Unmitigated STSBO w/TI-STGR

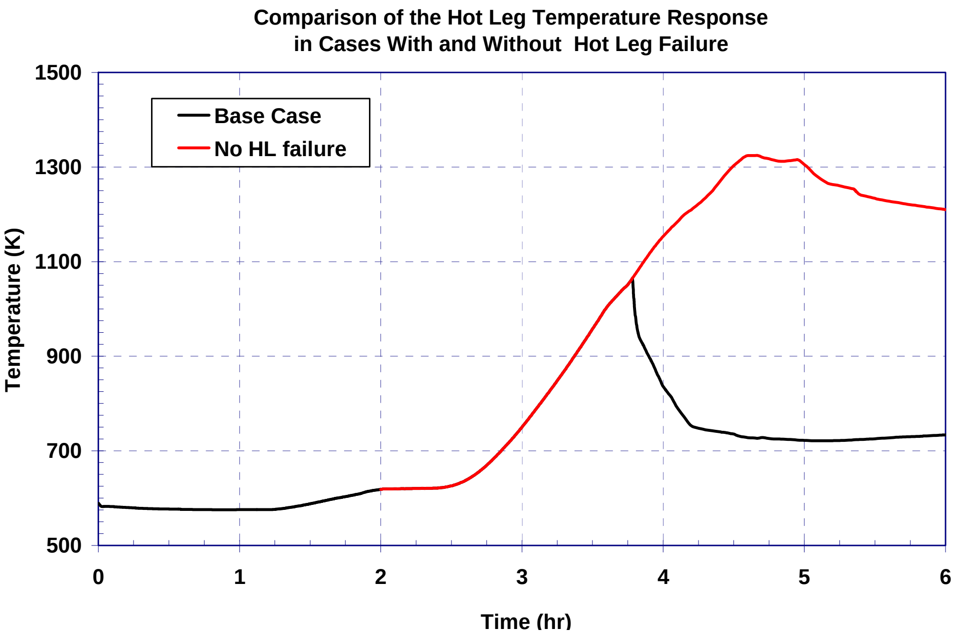
- **TI-SGTR did not preclude creep rupture of the hot leg**



Uncertainties in RCS Failures Unmitigated STSBO w/TI-STGR



- **Hot leg nozzle continues to heat following SG tube failure**
  - **~250 K hotter than base case**



- **Ignoring hot leg creep rupture is not credible**
  - **Larson Miller index 4-orders of magnitude above failure criterion**
  - **High sensitivity to thermal stress at >1000 K**

Uncertainties in RCS Failures Counterpart SCDAP/RELAP5 Analyses

- SCDAP/RELAP5 analyses performed using latest FLUENT modeling and modeling for hottest tube, NTR (normalized temperature ratio) = 0.5
- 2 cases modeled a single DE tube rupture
  - Tube rupture predicted for tube with assumed stress multiplier of 2.0 on the hottest tube (occurs at 03:46)
    - Hot leg failed 1.2 min later
  - Tube rupture predicted for tube with assumed stress multiplier of 3.0 on the hottest tube (occurs at 03:39)
    - Hot leg failed 8.8 min later
- Additional extreme case modeled as multiple tube rupture (with stress multiplier of 2.0)
  - HL failed 1.3 min later
- Counterpart SR5 hottest tube calculations confirm hot leg fails shortly after tube rupture for assumed seriously flawed tube (just above tube sheet)
  - MELCOR prediction is slightly conservative

Uncertainties in Iodine Chemical/Physical Form

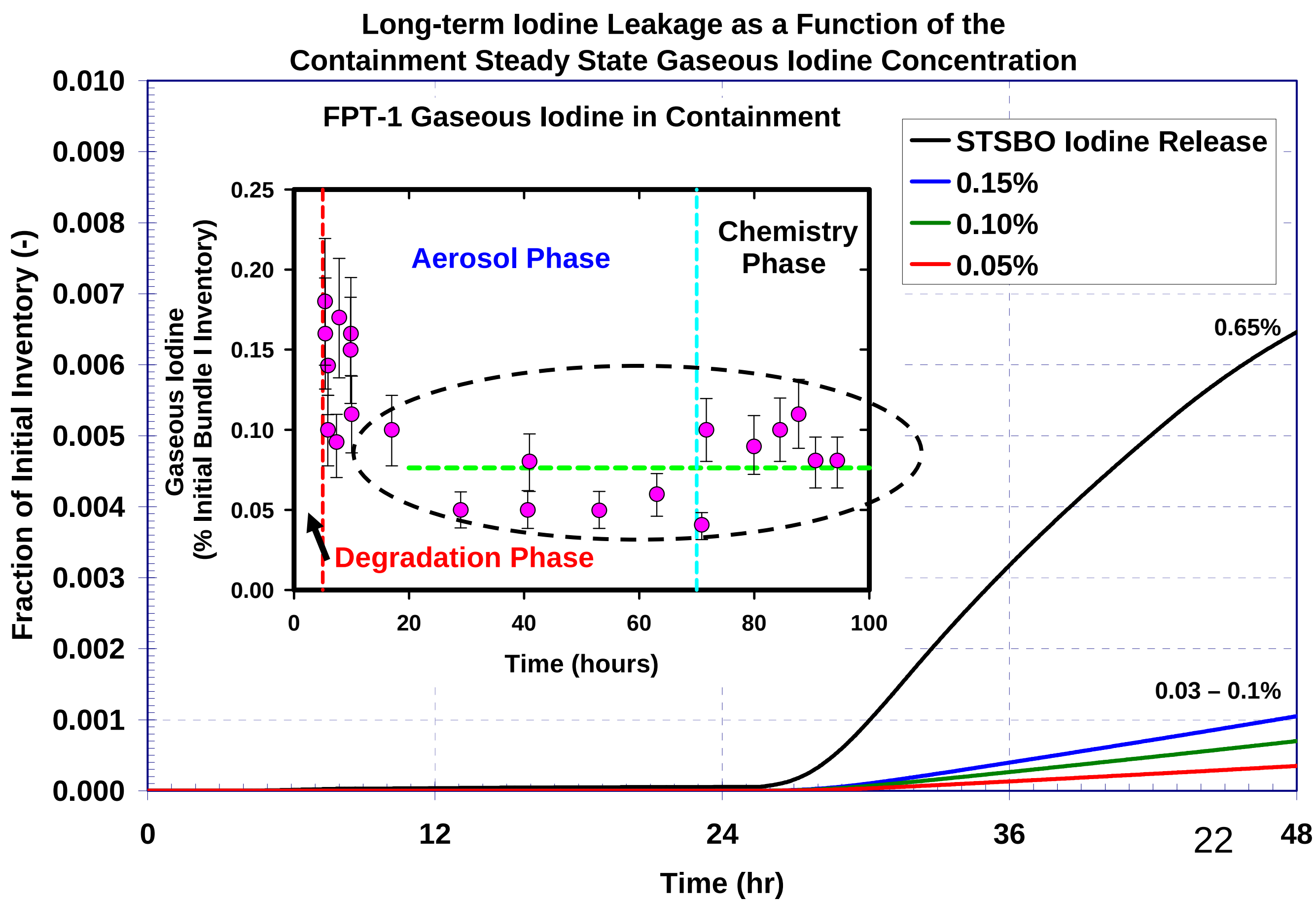
- **MELCOR calculations performed for SOARCA modeled iodine as cesium iodide and neglected iodine vapor**
- **Preliminary peer review comment: Iodine vapor was observed in the PHEBUS tests and should be considered**

Uncertainties in Iodine Chemical/Physical Form

- **Details of iodine release and subsequent behavior are complex**
- **Detailed data from Phebus is further informing our understanding of radionuclide iodine behavior**
  - **Tests show Cs being transported with Mo and I**
  - **In-vessel deposition and surface chemistry effect revaporization of iodine**
  - **Ex-vessel pools, sprays, and paint may capture gaseous iodine but mechanisms exist for re-release**
  - **Sump-wall-atmosphere exchange showing small long-term airborne concentrations**

Uncertainties in Iodine Chemical/Physical Form

- **Use Phebus FPT1 data to estimate additional STSBO source term**
- **Gaseous iodine seeks a low steady-state concentration that is largely independent of many parameters (pool pH, condensing, evaporating, etc.)**
- **Potential for a persistent low-level, long-term release**



Conclusions

- With 10 CFR 50.54(hh), mitigation is likely
- Without 10 CFR 50.54(hh), detailed more realistic modeling (MELCOR) shows more time to core damage and smaller releases
  - Treatment of complete operator response, including actions that may delay, but not prevent, core damage
  - Improved phenomenological treatment
    - Incorporated results of research programs showing that early containment failure modes of alpha mode failure and direct containment heating were physically not feasible or of extremely low probability
    - Incorporated test results from international test programs (PHEBUS, VERCORS, ARTIST)

Presented at OECD/NEA Workshop ISAMM-2009
Schloss Böttstein, Switzerland

Deterministic Evaluation of Quantitative Health Objective and Target of Severe Accident Management

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CONTENTS

I. Introduction

Introduction

❑ Background

- Policy statement on severe accident of NPPs was issued on August, 2001
- Quantitative Health Objective (QHO) as 0.1% additional risk to the sum of other base risk was proposed
  - prompt fatality
  - latent cancer fatality
- Policy Statement asks
  - utility to perform PSA
  - utility to develop Severe Accident Management Program
  - to develop performance goal of NPP to satisfy the QHO
- Utility performed PSA and developed SAMP for operating plants
- KINS reviewed the developed SAMP and evaluated the QHO for operating plants

Introduction

❑ Difficulties in Reviewing SAMP

- Main difficulties in reviewing the efficiency/feasibility of SAMP
  - uncertainties in severe accident phenomena
  - lack of success criteria for accident management activities
- Basic philosophy of the current SAMG is to do one's best with equipments available at the time of accident
  - normally installation of new hardware equipment is not required
- Regulatory review needs criterion
  - couldn't say SAM is O.K because the operators would do their best with what they have
  - wish to have a quantitative target/criterion
- QHO was thought as one possible target of SAM
  - different concepts of risk and conceptual difficulties exist in comparing risk by severe accident and other risk

Introduction

Introduction

- Different Concepts of Risk
  - Two different definitions of risk
    - Risk = Frequency × Consequence
    - Risk = Hazard + Outrage
  - For public living near the NPP at the time of accident, frequency has no meaning
- Thus, we wished to
  - ☞ evaluate whether the current NPP satisfies the QHO with the public concept of risk
  - ☞ search for a possible target of accident management activities under the current QHO

2. QHO of Different Countries

QHO of Different Countries

- Korean Quantitative Health Objective
  - Similar to US QHO

    - risk of prompt fatalities that might result from accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from other accident

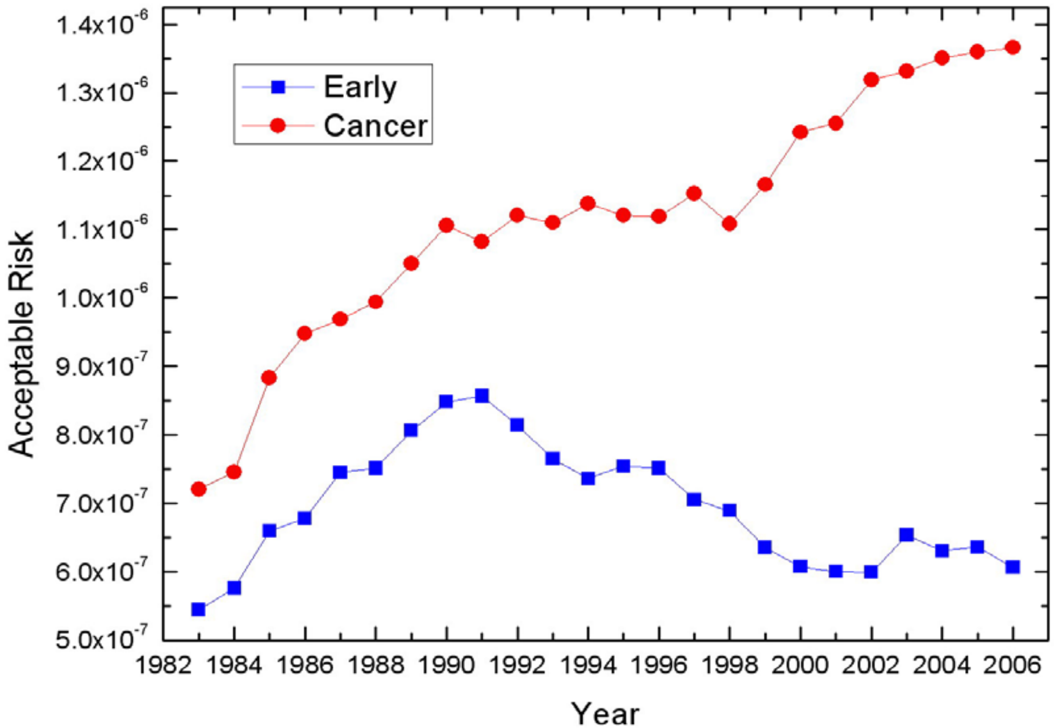
    - risk of cancer fatalities that might result from NPP operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes

QHO of Different Countries

- Data from Korean statistical information service (KOSIS) provide the concrete value. Averaged over 24 yrs.

  - 0.1% of the sum of prompt fatality from other accidents gives 6.9E-7

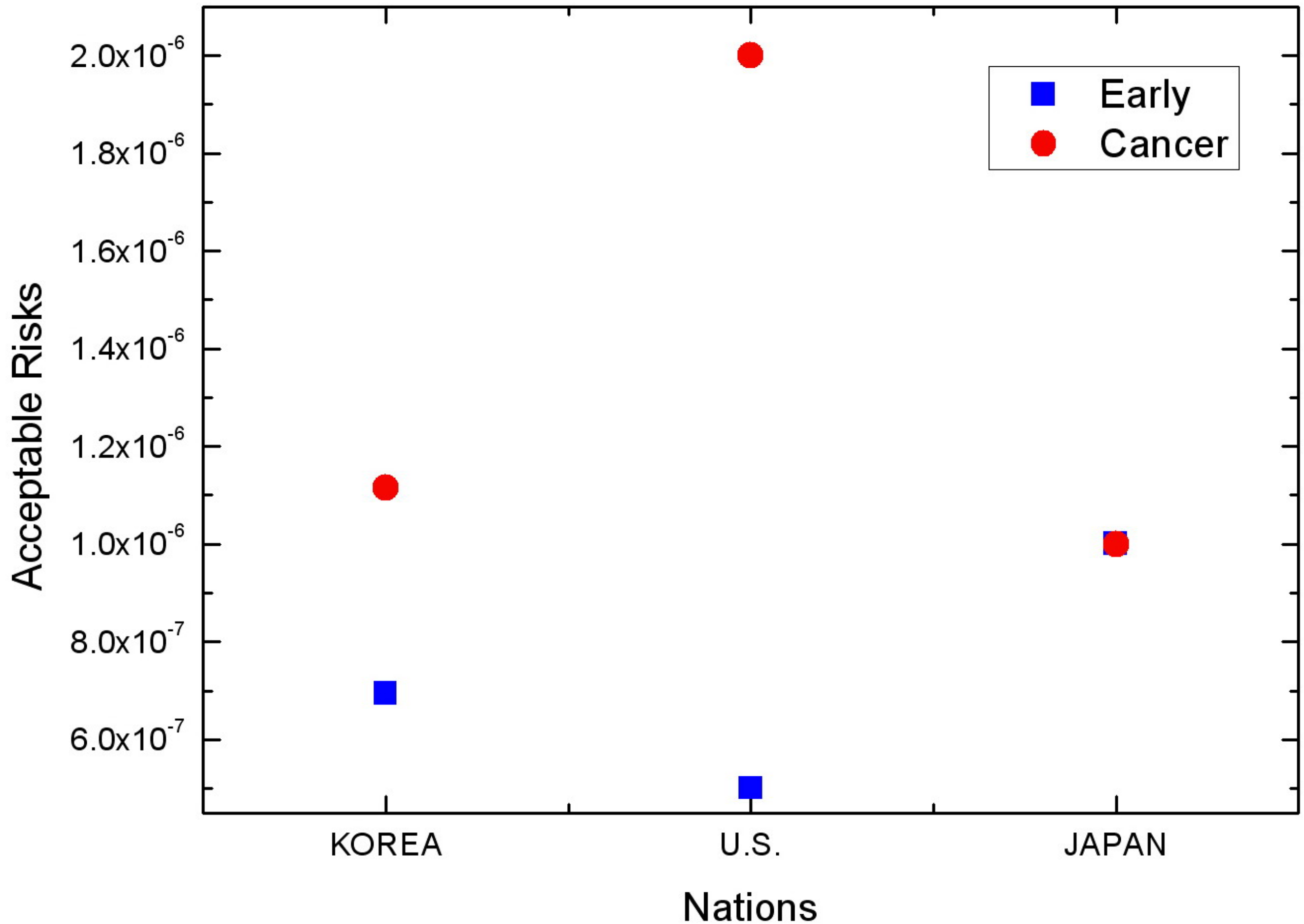
  - 0.1% of the sum of cancer fatality from other causes gives 1.1E-6



QHO of Different Countries

❑ Comparison of QHO

- Korea, U.S. and Japanese Health Objectives



3. Deterministic Evaluation of QHO using MELCOR-MACCS2

Deterministic Evaluation of QHO for Ulchin 3&4

- Deterministic evaluation means we will follow the accident as it progresses and source term release is modeled to occur when the containment pressure reaches a leak pressure

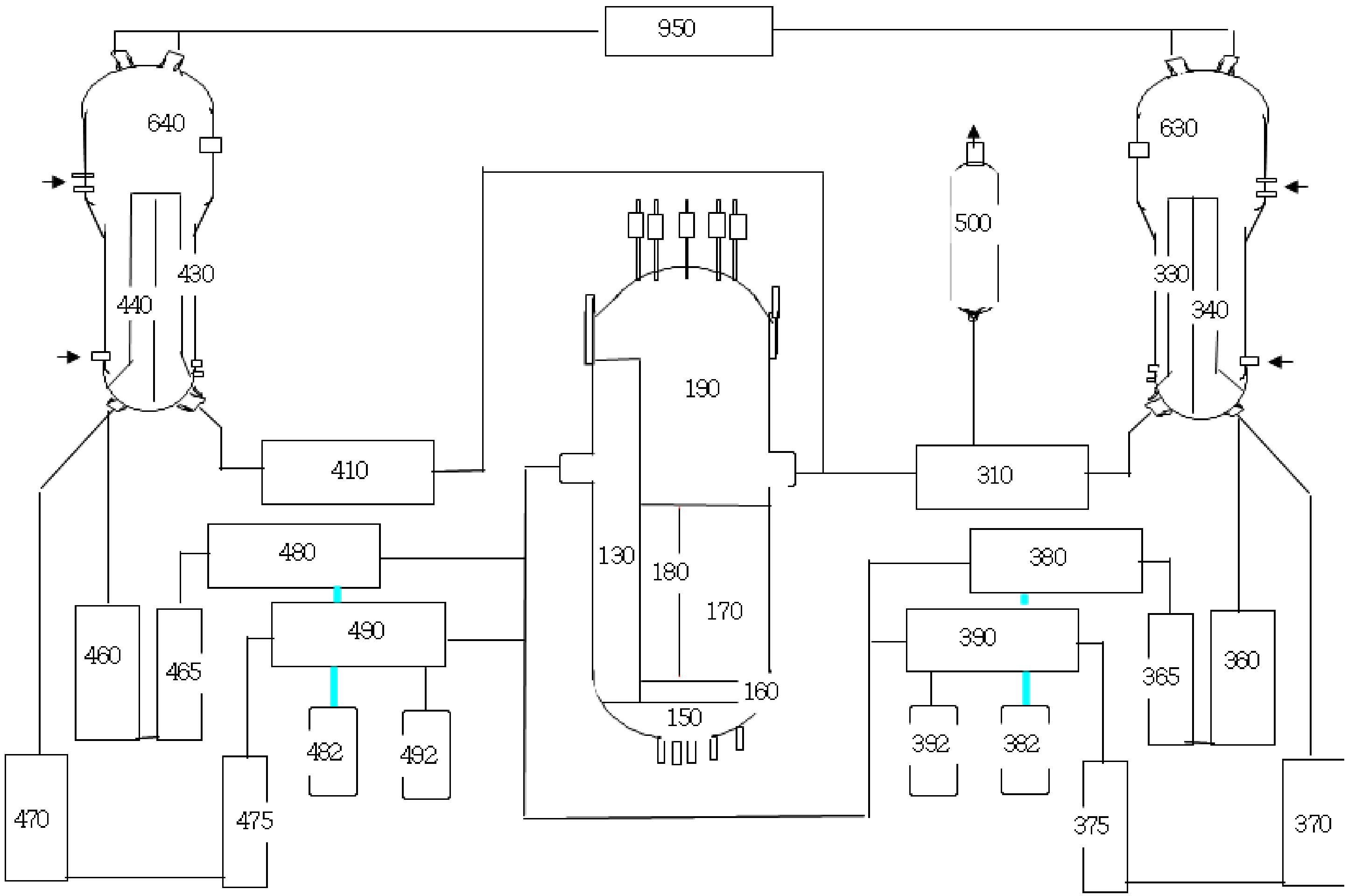
  - frequency multiplication can be removed

- Ulchin 3&4 NPP

  - 2826 MWt with 2 SG, 1 PZR and 4 RCPs
  - LBLOCA, SBLOCA, SBO scenarios were chosen for first assessment.
  - neither ESF nor operator actions are assumed for simplicity
  - purpose is to get a rough value on the magnitude of fatalities

- MELCOR 1.8.5 and MACCS2 Codes are used

- MELCOR 1.8.5 modeling is a typical one



Modeling of Containment Leak

- Leak model from structure analysis of containment
  - structure analysis using ABACUS code
  - 6.0 in^2 leak occurs near equipment hatch at median pressure of 169 psig
  - lower limit of pressure with 5% probability is 132 psig from $P_m \exp(-1.65\beta_u)$
- Leak rate at ILRT pressure
  - assumed 0.1 vol%/day leak occurs at P_{DBA}
  - ILRT is performed at this P_{DBA} = 57 psig
- Thus, source terms are modeled to be released either through 6.0 in^2 area when the pressure reaches 169 psig, or at 0.1 vol%/day when the pressure reaches P_{DBA}

MELCOR-MACCS2

- Flow Path Area to Model 0.1 vol%/day

  - assuming dry, steady condition the rate of volume change is equal to rate of mass change.

  - 0.1 vol%/day is 0.0011 kg/sec for containment volume of 7.76E4 m^3 and air density of 1.2 kg/m^3

  - MELCOR flow path area corresponding to this leak rate at P_{DBA} is calculated to be 1.0E-5 m^2

  - leak rate calculated by MELCOR is 0.002 kg/sec.

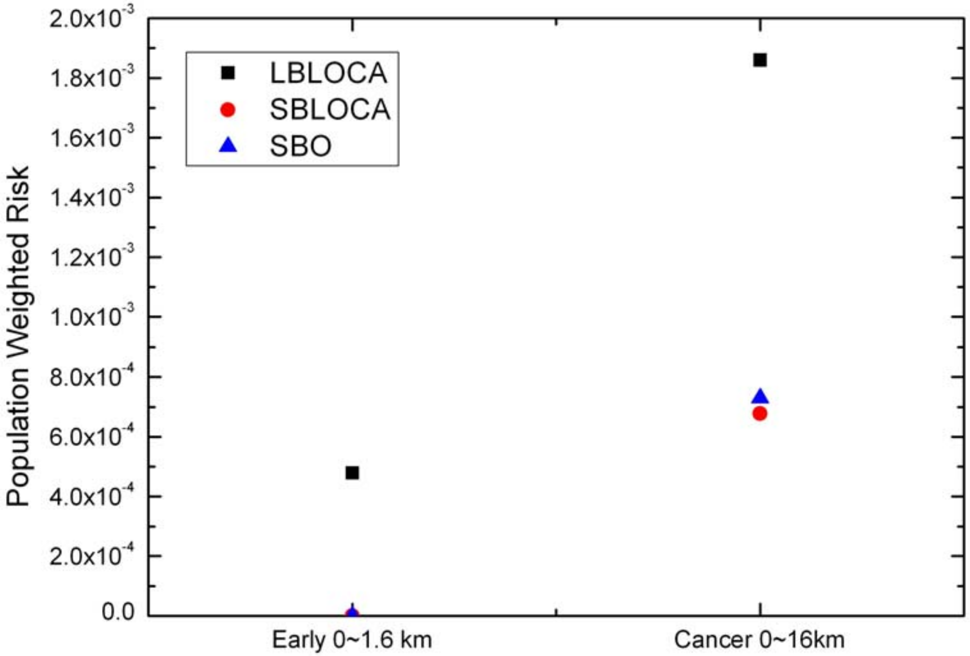
  - density in annual compartment is 2 times higher than that of dry air, thus MELCOR model of flow path at P_{DBA} is reasonable

- ORIGEN-S and MACCS2 codes are used for consequence analysis

4. Evaluation Results of QHO and Target of SAM

Evaluation Results and Target of SAM

- Initial Assessment of Fatalities
  - Assessment for accident scenario of LBLOCA, SBLOCA, SBO
    - leak modeled to occur through 6.0 in^2 (3.9E-3 m^2) area when the pressure reaches 132 psig
    - release data are used as inputs for MACCS2 code and fatalities are calculated
  - results show that neither early nor cancer fatalities satisfy the QHO



Evaluation Results and Target of SAM

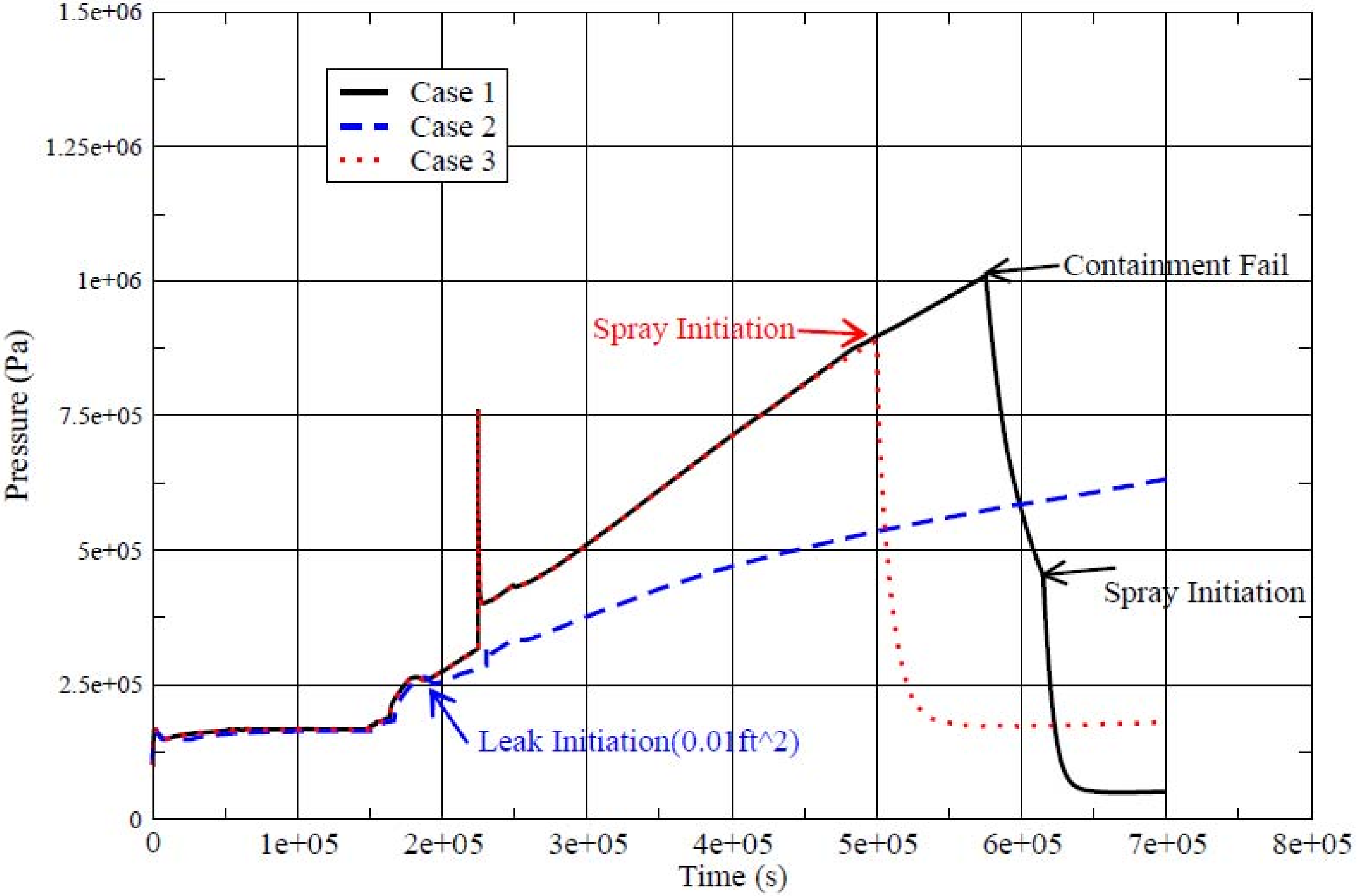
❑ Sensitivity Evaluation for SBLOCA Accident

- More realistic SBLOCA accident scenario
  - SBLOCA * Rx Trip *HPSI Injection *AFW *MS ADV /HPSI Recirculation *RCS Depressurization using Aux. Feed / LPSI Recirculation
  - * means success action, / means failed action
  - leak area was changed to see the effect
- Insights
  - QHO is not satisfied for 3 cases
  - case 2 simulates venting strategy with 0.01 ft^2 area.

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.	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.	Can. Fat. / 0 - 1.6 km 2.64E-3 Can. Fat. / 64 - 80 km 2.53E-7

Evaluation Results and Target of SAM

- Pressure behaviour for 3 cases



Evaluation Results and Target of SAM

❑ Sensitivity Evaluation for Case-3

Cases	Average Individual Risk
3-1 leak at the rate of 0.1 vol%/day at P_{DBA} (2.3E5 sec) and sprayed at 5E5 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%/dat at P_{DBA} and sprayed at 2.7E5 sec. (12 hrs 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 2.4E5 sec. (3hrs after leak begins)	Can. Fat. / 0-1.6 km 6.63E-6 Can. Fat. / 64-80 km 9.27E-10

☞ Insights from the sensitivity evaluation

- QHO could be satisfied if spray is activated within 3 hrs after reaching P_{DBA}
- saying other way, pressure should be maintained below P_{DBA}
- this could be target of SAM, viewed from the current QHO

5. Summary

SUMMARY

❑ Preliminary Insights

- Current QHO has a conceptual difficulty in applying
- Uncertainty of MACCS2 code is high

  - order is easily changed depending on inputs and how we model the source term release, plume position and energy
- The only way to satisfy the QHO is to maintain the containment pressure below P_{DBA}

  - venting strategy is not effective from QHO viewpoint, if not a filtered venting
- Target of SAM should be to maintain the containment pressure below P_{DBA} under the current QHO

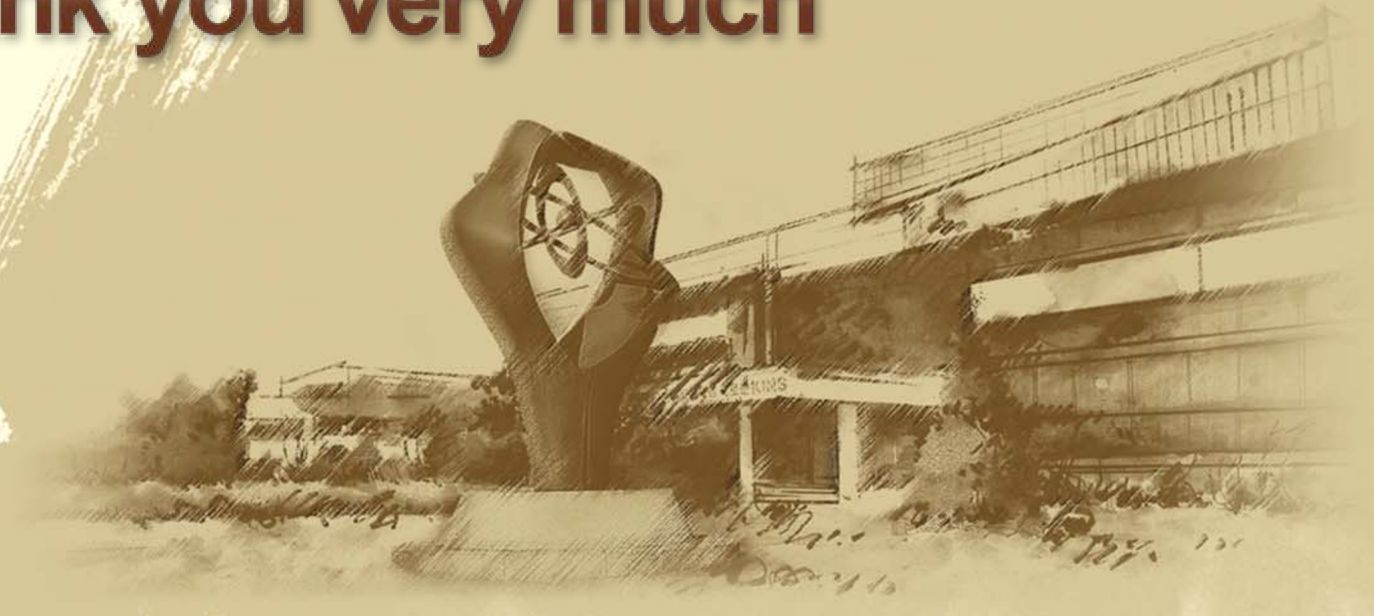
SUMMARY

- Suggestion for Further Study
  - Uncertainties in consequence analyses should be reduced
  - Quantitative target of SAM activities is possible

    \- having a quantitative target of AM satisfying the QHO could provide more logical framework for developing the AM strategies and also to convince public on NPP safety
  - QHO needs to be assessed again seriously

KINS

Thank you very much



Verification of the SAMG for Paks NPP with MAAP code calculations

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OECD/NEA Workshop on Implementation of Severe Accident Management Measures
PSI, Villigen, Switzerland, October 26-28, 2009

Contents

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Introduction

- Paks NPP implemented a severe accident management program for the VVER-440/213 units. The program includes plant modifications and development of procedures.
- A project on the development of Severe Accident Management Guidelines (SAMG) was launched with the lead of Westinghouse Electric Belgium Co.
- As a complementary effort, a domestic project on the verification of the guidelines was initiated to check and support the development of the SAMG.
- MAAP4/VVER code calculations were performed with assumptions of SAMG actions within the project.

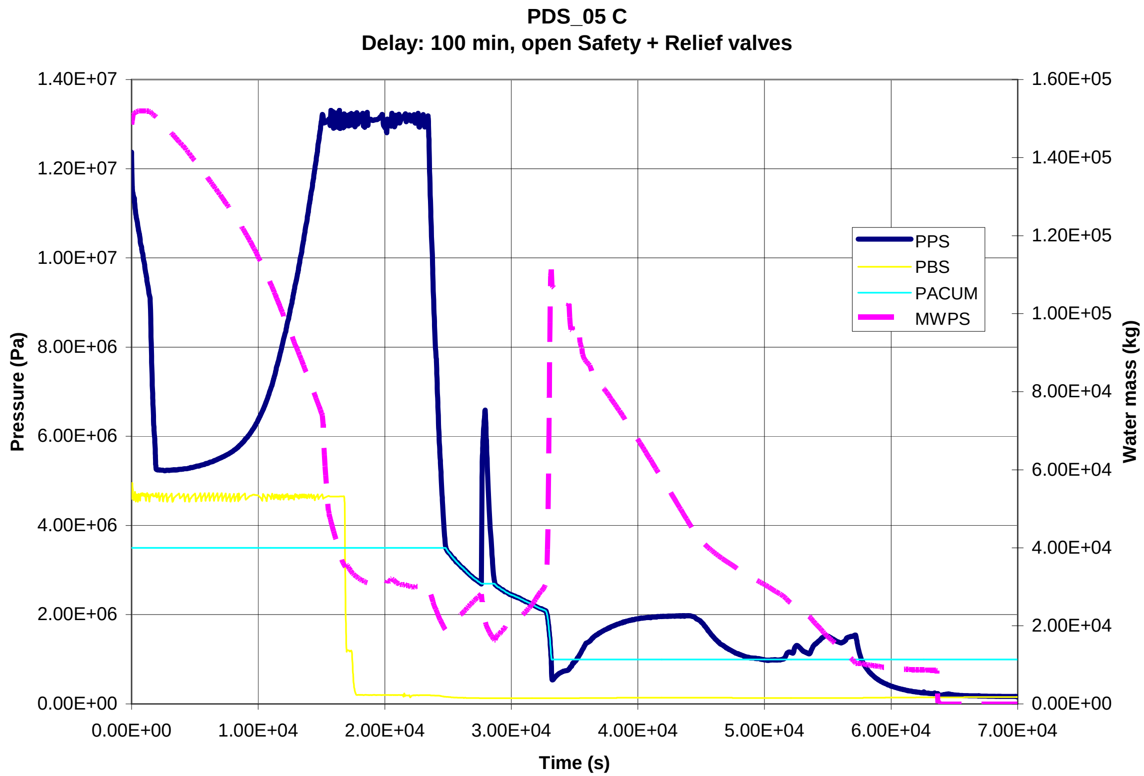
Depressurisation/1

- **Purpose of the SAG-1 guideline (Depressurize the RCS):**
  - 1. Decrease the potential of a high pressure melt ejection (HPME) event and creep rupture of SG tube
  - 2. Making available injection sources into the primary system at lower pressure
- **Initial LOCA or SGTR events with an equivalent break size larger than 40 mm do not lead to HPME**
- **The dominant sequence according to Level 2 PSA was selected for the calculations:**
  - PDS_05C sequence: 11 mm LOCA with loss of ECC and secondary heat removal

Depressurization /2

- **Base case calculation w/o primary pressure reduction for PDS_05C**
  - Vessel failure on high system pressure (103 bar), HPME, catastrophic consequences
- **PDS_05C, primary pressure reduction with the pressuriser valves:**
  - Number of valves (2 safety valves and 1 reduction valve (PORV) are available), and time delay of the intervention from the TEXIT = 550 C signal were varied.
  - Core melt can be prevented, if all (3) valves are opened within maximum 20 minutes during the Emergency Operating Procedures (EOP). In this case the primary pressure is reduced to the level of p=7.5 bar, a condition for starting LPIS.
  - Vessel failure can be prevented, if at least 2 valves are opened within 100 min after the TEXIT=550 C signal. In this case the primary system pressure is reduced to p=10 bar.

Depressurization /3



- **Primary system pressure (PPS) and water mass (MWPS) in the vessel in case of primary system depressurization at 100 min after the signal TEXIT = 550 C**
- **Core melt starts at 20000 s, depressurisation at 23300 s**

Depressurization /4

NUBIKI

60 minutes delay from severe accident signal							
			Primary system pressure				
	Letdown cross section	Equvalent 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

- Depressurization via different letdown valves in case of the PDS_05C sequence
- At least 20 mm vent size is necessary to avoid HPME
- At least 40 mm vent size is needed for the actuation of LPIS

Green: effective depressurization, red: failure to reduce pressure

Injection into primary system/1

- **According to the SAG 3 guideline (Inject into RCS), water injection from alternative sources is suggested after the depressurization of the primary system**
- **The effectiveness of the alternative water injection options were studied in different phases of the severe accident sequence:**
  - after core heat up, but before melt down,
  - after core melting, but before the lower support plate failure,
  - when the core debris was relocated into the bottom of the vessel.
- **Dominant sequences of the Level 2 PSA and LBLOCA sequences were selected for the verification study**

Injection into primary system/2

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

- **Effectiveness of LPIS injection for the PDS_05C sequence depending on the restoration time**
- **Vessel failure can be prevented, if LPIS recovered within 10 hours**

Injection into primary system/3

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

- **Influence of the water injection rate on the progression of a LBLOCA sequence**
- **Water injection starts after core melt**
- **At least 18 t/h is necessary to prevent vessel failure**
- **This rate is more than the amount necessary for decay heat removal (12 t/h)**

Injection into primary system/4

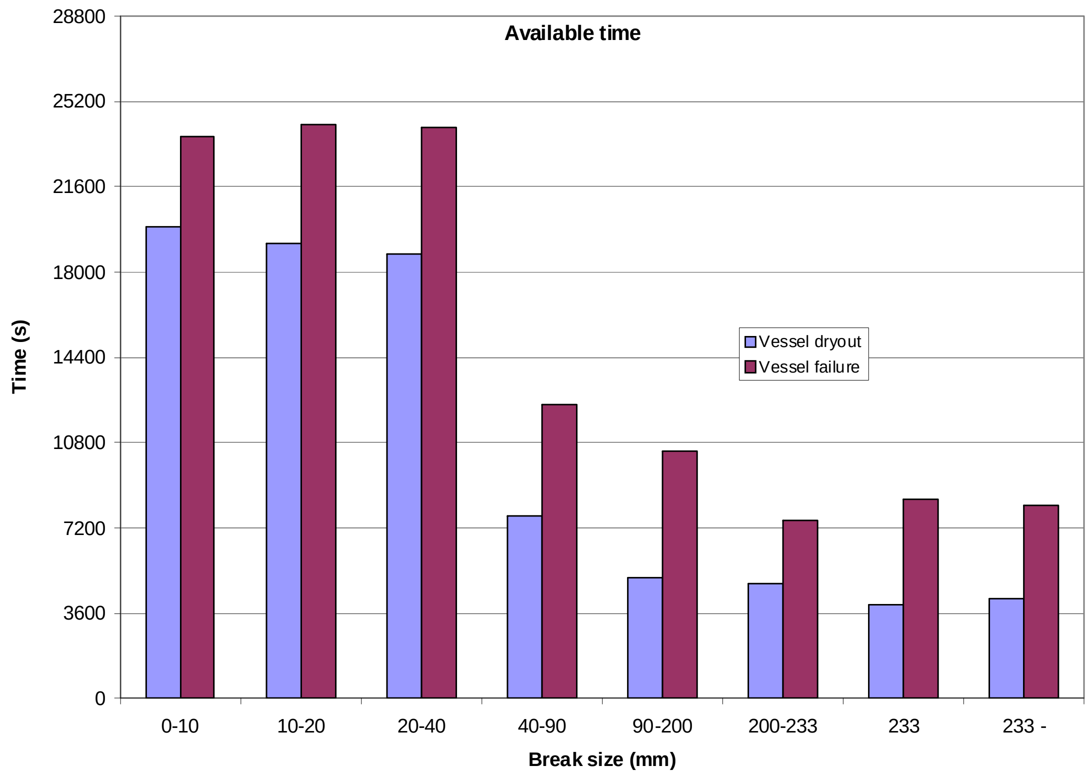
Conclusions:

- **Water injection should be initiated as soon as possible in case of availability of any water resources**
- **The flow rates needed to arrest the severe accident sequence progression are usually higher than the amount necessary for decay heat removal**
- **The negative impacts of water injection into the primary system (e.g. hydrogen production) was over-predicted in the SAG 3 guideline**

In-vessel melt retention/1

- **By design the VVER-440/213 reactor cavity is dry. For external cooling of the vessel two actions should be performed: (1) drain the water from the localisation system to the containment sump and (2) flood the reactor cavity from the containment sump. The procedure is described in the SAG 2 guideline (Inject into containment and cavity flooding).**
- **The goal of the study was to determine the time window available for the intervention**
- **The following sequences were analyzed in the study:**
  - Dominant sequences of the Level 2 PSA: PDS_05, PDS_02
  - Other, deterministic sequences: LBLOCA 500 (200 %), LBLOCA 233, LBLOCA 200, medium and small breaks in the range of 20 - 100 mm

In-vessel melt retention/2



- Available time for flooding the cavity from the signal T_{EXIT}=1100 C
- Flooding should be started until vessel dry-out at latest
- There is ample time to perform the action for the most probable (Level 2 PSA) sequences
- Less time is available for the larger LOCA sequences

- The limiting sequence is the LOCA 200 mm. This sequence is quite fast, but the water inventory of the localisation tower does not flow back automatically to the sump, so the water should be drained manually.

In-vessel melt retention/5

- For the limiting LOCA 200 mm case the available time until the vessel dry-out is 65 min from the T=1100 C signal and 80 min from the T= 550 C signal.
- The time between the vessel dry-out and vessel failure is the safety margin of the intervention
- Manual draining of the water from the localisation system also takes a substantial time of around 80 min. Therefore the procedure should be started upon reaching the T=550 C signal, as a last step of the EOP.
- For the most probable sequences predicted by the Level 2 PSA, the available time to perform the procedure is more than 5 h.

Preventing excessive vacuum/1

- **Excessive depression may be established in the containment - a VVER-440/213 specific feature. (Limiting depression value: 200 mbar)**
- **Depression may be established due to the following physical phenomena:**
  - Relocation of air from the main building into the air-traps and concurrent steam condensation
  - Release of non-condensing gases through the containment leakage
  - Decreasing the fraction of hydrogen and oxygen as a result of the operation of catalytic recombiners or hydrogen burn

Preventing excessive vacuum/2

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- **Containment depression may be intensified by the operation of the containment spray system**
- **Excessive vacuum can develop in case of 3 operating trains of the spray system**
- **The remedy is quite simple: stopping one or two trains of the spray system when containment pressure reaches the atmospheric level**

Preventing overpressure/1

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- According to the SCG 2 guideline, a filtered venting procedure will be started to prevent containment overpressure
- Filtered venting is planned to be implemented via the upgrade of the existing TN 01 ventilation system with appropriate motor operated valves, a rupture disc and severe accident filter. Presently the system is in the design phase.
- The filtered venting procedure is designed to start as soon as containment pressure reaches 3.3 bar abs. (HCLPF value). This pressure level is expected to occur no sooner than 24 h after the start of a severe accident.

Preventing overpressure/2

- Calculations have been performed for the PDS_05C sequence and a LBLOCA sequence with and without IVR.
- According to the calculations the filtered venting is capable to decrease containment pressure.
- A dedicated filter with filtering efficiency of 99.9 % would limit the Cs and I release to 0.01 %.
- Spurious opening of the line would not lead to excessive Cs and I releases, although noble gas release will increase in this case.
- Steam condensed in the venting line and the filter will be cooled and redirected to the containment.

Decreasing fission product release/1

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- The SAG 4 (Decreasing the release of fission product release) and SAG 6 (Control of containment parameters) guidelines suggest the use of the ventilation systems in the frame of the SAMG.
- There are 6 recirculation ventilation systems available in the plant with a variety of flow rates and design - some are equipped with heat exchangers, others - with filters.
- Calculations have been performed for the PDS_05C sequence with the MAAP code to check the effectiveness of the ventilation systems for FP retention.
- One example is the TL01 Recirculation ventilation system (3 x 60000 m^3/h), designed to cool the containment. The system is equipped with heat exchangers, but no filter is available.

Decreasing fission product release /2

- Aerosols will be deposited mainly in the heat exchangers of the TL01 ventilation system. The maximum retention of the system is around 30 % related to the amount entering the inlet.
- Other ventilation systems have similar features, but their capacities are lower.
- Another benefit of using the ventilation systems is the moderation of the pressure gradient in the containment, thus time can be gained until the start of filtered venting procedure
- Conclusion: Ventilation systems can be used to decrease the FP release by a few percents. For comparison, the containment spray system can reduce the release by an order magnitude.

Summary/1

- **MAAP code 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 the accident management actions.**

- **Highlights of the conclusions of the study:**
  - Depressurization of the primary system: this is a very important intervention, which can be performed with different valves
  - Water injection into the primary system: water injection should be performed with any water source available
  - In-vessel retention: draining the water from the localisation system should be started as early as possible to flood the reactor cavity and to provide effective cooling to the vessel

Summary/2

  - Preventing excessive vacuum: arresting the operation of the spray system for some time removes the problem
  - Preventing containment overpressure: if the spray system is not available, then a filtered venting can be effectively used as a remedy
  - Decreasing fission product release: the primary option is the use of the containment spray system. If sprays are not available, then ventilation systems can be used to moderate the release.
- **The verification study led to the conclusion that fine-tuning and some modification of the existing guidelines are needed to meet the specific challenges represented by severe accidents at Paks NPP**

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Treatment of Accident Mitigation Measures in State-of-the-Art Reactor Consequence Analyses

Jason H. Schaperow, Mark T. Leonard, Charles G. Tinkler, K. C. Wagner

Presented at the OECD/NEA Workshop on Implementation of Severe Accident Management (SAM) Measures
October 26-28, 2009

Outline

- Overview of SOARCA Study
  - Background
  - Objectives
  - Approach
  - Conclusions
- Scenario Selection
- Accident Mitigation
  - Approach
  - Results
  - Conclusions

Background

- NRC security studies performed following 9/11 incorporated severe accident research performed over the last 2 decades
- Security studies confirmed that earlier accident consequence studies were conservative to the point that predictions were not useful for characterizing results or guiding public policy
- Earlier consequence studies used
  - Combination of conservative assumptions or boundary conditions
  - Simple bounding analysis

Objectives

- SOARCA study being performed to develop body of knowledge regarding the realistic outcomes of severe reactor accidents
- Incorporate significant plant improvements and updates not reflected in earlier assessments
  - System improvements
  - Training and emergency procedures (EOP/SAMG)
  - Offsite emergency response
  - Recent security-related enhancements (10 CFR 50.54(hh))
- Evaluate the potential benefits of mitigation improvements in preventing core damage and reducing an offsite release should one occur
- Enable NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders
  - Federal, state, and local authorities
  - Licensees
  - General public
- Update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, “Technical Guidance for Siting Criteria Development”

Approach

- Detailed
  - Includes operator actions beyond those critical to prevent core damage
    - reducing injection flow to preserve inventory
    - depressurizing RCS
  - Includes details of facility not included in previous studies – modeling fission product deposition in buildings adjacent to containment
  - Detailed nodalization of core and RCS
- Best-estimate
  - Represents the most likely outcome for uncertain behavior
    - Avoids biasing answer in conservative or non-conservative fashion
  - Models high-temperature failure of RCS components (BWR SRV sticking open, PWR hot leg rupture following thermally induced SGTR)
- Self-consistent
  - Integrated MELCOR analysis
  - Accounting for all relevant systems, subsystems
  - Scenario-specific EP

Approach

- Integral
  - Single code (MELCOR) provides feedback among phenomenological models and operator actions
- Current scientific knowledge and plant capabilities
  - MELCOR validation includes the latest tests such as PHEBUS and VERCORS
  - Results of ARTIST tests of fission product deposition reflected in the analysis
  - Latest security-related mitigation measures (10 CFR 50.54(hh)) credited in the analysis

Conclusions

- SOARCA represents major change from the way people perceive severe reactor accidents and their likelihood and consequences
  – Mitigation is likely (due to time, redundancy, diversity) and, when it is implemented, effective in preventing core damage
    - Impact on existing level 1 PRA
  – Unmitigated accidents progress more slowly with smaller releases, no LERF
    - Impact on existing level 2 PRA
  – Scenarios have lower frequency and lower consequences – lower risk
  – Dominance of external events suggests need for corresponding PRA focus
    - Seismic research needed

Scenario Selection

- Select scenarios that are important to risk for the purpose of performing state-of-the-art accident progression, source term, and consequence analyses
  - Central focus of SOARCA is to introduce use of a detailed, best-estimate, self-consistent quantification of sequences based on current scientific and plant capabilities
- Plant-specific for Peach Bottom and Surry based on latest PRA information available
- Internal and external events included

Scenario Selection

- Group sequences according to similar equipment availabilities and analyze the more probable and important severe accident sequence groups
  - Enhanced realism in the analysis requires specifying initial and boundary conditions for clearly defined scenarios
  - Screen in sequence groups that PRAs have shown are important contributors to risk (e.g., station blackout)
  - Screen out sequence groups that PRAs have shown are small contributors to risk (e.g., internally initiated event of large break LOCA with sustained loss of injection)

Scenario Selection

- Peach Bottom scenarios analyzed in SOARCA
  - Long-term SBO (seismic initiator) loss of AC – $1x10^{-6}$ to $5x10^{-6}$ /year
  - Short-term SBO (seismic initiator) loss of AC, loss of DC – $1x10^{-7}$ to $5x10^{-7}$ /year
  - Loss of Vital AC Bus E12 – ~$5x10^{-7}$/year

Scenario Selection

- Surry scenarios analyzed in SOARCA
  - Long-term SBO (seismic initiator) loss of AC – $1x10^{-5}$ to $2x10^{-5}$/year
  - Short-term SBO (seismic initiator) loss of AC, loss of DC, gross rupture of ECST – $1x10^{-6}$ to $2x10^{-6}$/year
  - Short-term SBO (seismic initiator) with thermally induced SGTR – $2.5x10^{-7}$ to $5x10^{-7}$/year
  - ISLOCA – $7x10^{-7}$/year (licensee PRA), $3x10^{-8}$/year (SPAR)
  - Spontaneous SGTR – $5x10^{-7}$/year

Accident Mitigation – Approach

- Plant-specific and scenario-specific for Peach Bottom and Surry
- Extensive cooperation from licensees
- Table-top exercises with SRO's, PRA analysts and other licensee staff
  - Based on recent MELCOR analysis of unmitigated event to establish RCS conditions, timing
  - Walk through of scenario timeline and operator actions based on EOPs, SAMGs and other mitigation, considering also activation of TSC and EOF
    - Assessment of adequacy of available time for operator action
    - Considered aggravation by seismic condition

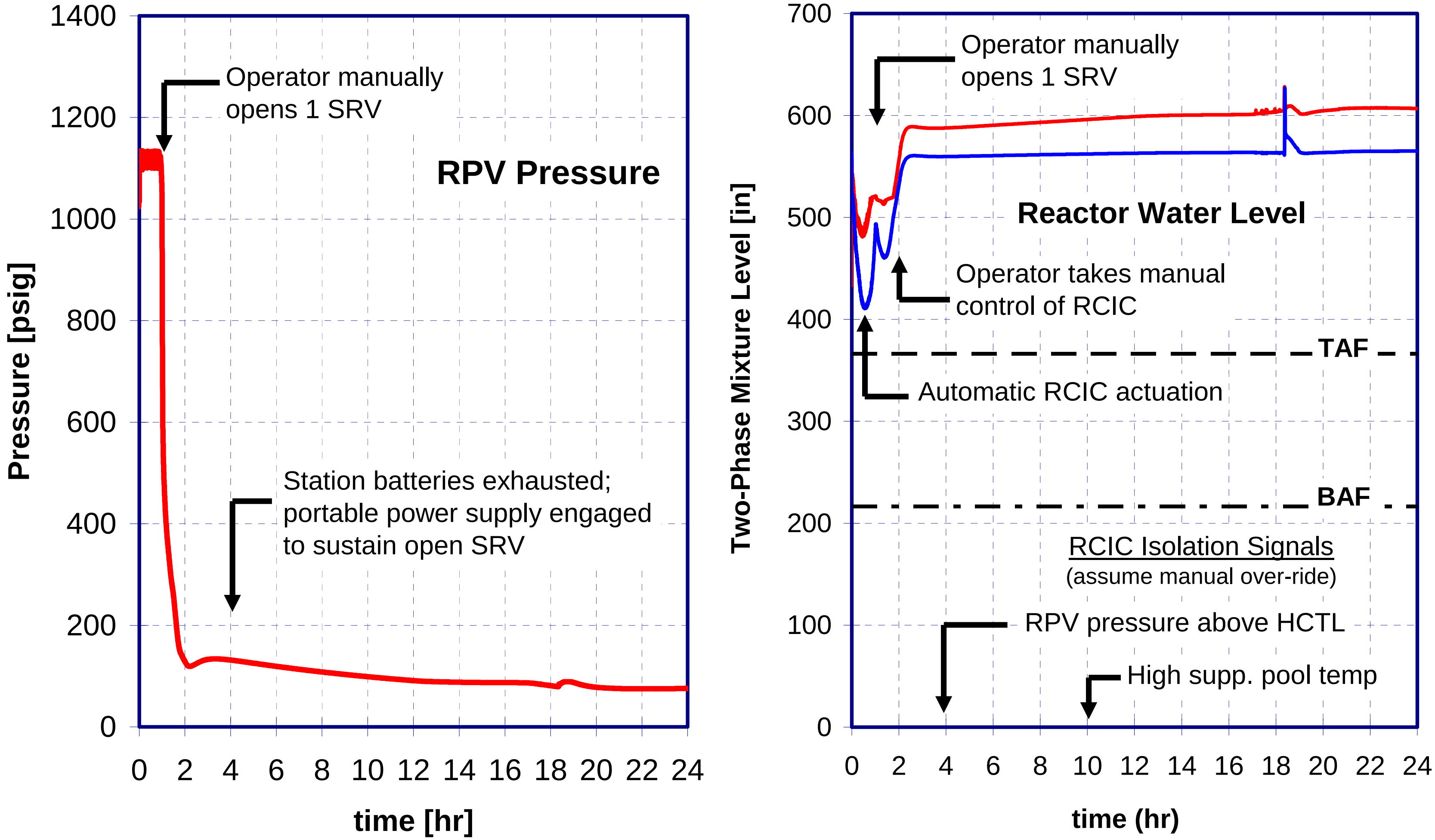
Accident Mitigation – Approach

- Table-top exercises were used to develop detailed timeline of operator actions for each scenario
- Timelines included multiple possible actions to mitigate
  - Peach Bottom LTSBO
    - Manual operation of RCIC without electric power
    - Depressurization of RPV together with portable diesel driven 10 CFR 50.54(hh) pump
- Performed MELCOR calculations with mitigation times from the table-tops
  - Confirmed mitigation times sufficient to prevent core damage
  - Confirmed mitigation measures had sufficient capacity (pressure, gpm)

Peach Bottom Long-Term SBO

- First 4 hrs (duration of dc power):
  - Reactor coolant makeup via operation of RCIC
    - Automatic actuation
    - Manual flow control to stabilize level within target range
  - RPV pressure control
    - Open SRV reduces pressure to approx. 150 psi (above low-pressure isolation setpoint for RCIC)
- Within 4 hrs:
  - Portable power supply is positioned, connected to emergency bus (dc) and operating
  - Portable air and power supply positioned and connected to isolation valves for torus hard-pipe vent
- Long-term response:
  - RCIC operation maintains RPV water level
  - Containment pressure controlled by periodic opening of hard pipe vent

Peach Bottom Long-Term SBO



Accident Mitigation – Results

- Peach Bottom
  - LTSBO (seismic initiator) loss of AC
    - Mitigated by either manual operation of RCIC or 10 CFR 50.54(hh) equipment
    - No core damage
  - STSBO (seismic initiator) loss of AC, loss of DC
    - Mitigated by either manual operation of RCIC or 10 CFR 50.54(hh) equipment
    - No core damage
  - Loss of Vital AC Bus E12
    - Mitigated by CRDHS (demonstrated by MELCOR calculation alone) – no need for 10 CFR 50.54(hh) measures
    - No core damage

Accident Mitigation – Results

- Surry
  - LTSBO (seismic initiator) loss of AC
    - Mitigated by either manual operation of TDAFW or 10 CFR 50.54(hh) equipment
    - No core damage
  - STSBO (seismic initiator) loss of AC, loss of DC, instantaneous gross rupture of ECST
    - Mitigated by using 10 CFR 50.54(hh) pump to supply containment spray
    - Small release of volatile fission products to environment
  - STSBO with TISGTR (seismic initiator) loss of AC, loss of DC, gross rupture of ECST
    - Same as above

Accident Mitigation – Results

- Surry
  - ISLOCA
    - PRA indicated core damage due to operator failure to refill RWST or switch to unaffected unit's RWST
    - Mitigated by normal equipment and ample time (8 hours to core damage) – no need for 10 CFR 50.54(hh) measures
    - No core damage
  - Spontaneous SGTR
    - PRA indicated core damage due to operator failure to depressurize RCS, isolate the faulted SG, and refill RWST
    - Mitigated by normal equipment and ample time (24-48 hours to core damage) – no need for 10 CFR 50.54(hh) measures
    - No core damage

Accident Mitigation –Conclusions

- All events can reasonably be mitigated
- 10 CFR 50.54(hh) mitigation and more realistic treatment of other mitigation together with detailed realistic modeling (MELCOR) has significant benefits
  - Scenarios that current PRAs say result in core damage were shown to not be core damage scenarios (or even lower frequency)
    - Peach Bottom long-term SBO, short-term SBO, loss of vital ac bus E12
    - Surry long-term SBO, ISLOCA, spontaneous SGTR
  - Surry short-term SBO resulted in core damage, because we assumed seismic event was severe enough to result in ECST rupture and preclude operator action for several hours

Accident Mitigation –Conclusions

- Dominant contributors to CDF and their consequences are better characterized with integrated, best-estimate accident mitigation modeling (e.g., the SOARCA approach)
  - Internal events scenarios – core damage reasonably prevented by normal equipment and ample time
    - no need for 10 CFR 50.54(hh) measures
  - External events – core damage reasonably prevented by redundant and diverse security-related measures
    - manual operation of turbine-driven pump (RCIC, TD-AFW), portable diesel-driven pump

BEST PRACTICES APPLIED TO DETERMINISTIC SEVERE ACCIDENT AND SOURCE TERM ANALYSES FOR PSA LEVEL 2 FOR GERMAN NPP´S

Dr. M. Sonnenkalb, Dr. N. Reinke, Dr. H. Nowack
GRS Cologne

OECD/NEA Workshop
Implementation of Severe Accident Management (SAM) Measures
26. – 28.10.2009
Villigen, Switzerland

Status of German SAM Program and PSA Level 2

SAM implementation - Legal Requirements and Status

- In past no formal requirements on SAM.
- Utilities offered in 1986 voluntarily to realize recommendations of German RSK on SAM.
- Implementation of SAM since 1986 mainly with significant hardware modifications: bleed&feed, PARs/inertisation, filtered cntm. venting, secured cntm. isolation, additional power supply, cntm. sampling system.
- Implementation of SAM measures is almost completed.
- Severe Accident Management Guidelines to be developed/implemented in future.
- Review of legal requirements was done at GRS on behalf of BMU between 2006 - 08.
- New German regulations are in test phase.

Status of PSA Level 2

- PSA Level 2 for three main German NPP types have been performed by the GRS within R&D projects (1998 – 2006), exploring PSA Level 2 methodology.
  -> **MELCOR** was mainly used for severe accident and source term analyses.
- PSA Level 2 recently has become part of the periodic safety review:
  -> **German PSA Guidance** document was updated and published in 2005.
  -> Integral codes like **MELCOR, ASTEC** are recommended to be used.
  -> German utilities started to perform PSA Level 2 studies (since 2006).
- PSA Level 3 is still not required in Germany.

Status of Source Term Analysis and Prognosis

Source Term Analysis

- Deterministic analyses to calculate the release of fission products from different locations in an NPP into the environment
- Different release locations (core, cavity, spent fuel pool) and release paths (water path, filtered venting system, containment leaks, air ventilation ducts) are possible
- Usually performed within a PSA Level 2; basis for a PSA Level 3
- Deterministic integral codes used: **MELCOR, ASTEC**

Source Term Prognosis

- Prognosis of source term to initiate external AM measures (e.g. sheltering)
- Important and difficult topic to be performed early in a (severe) accident by NPP crisis team
- Often very simple methods applied to estimate releases from NPP
- New systems for source term prognosis under development at GRS:
  - **STERPS**: probabilistic tool based on a „bayesian belief network“
  - **ASTRID**: deterministic fast running severe accident tool
- Existing deterministic system **RODOS** is used to calculate the radiation exposure / off-site consequences outside the NPP; used for off-site emergency planning

Status of German PSA Level 2 Guidance

Objectives

- To support a systematically development of PSA studies and the assessment of branching probabilities for severe accident progression event tree (APET) analysis.
- To reduce the potential of controversial expert views in the frame of the Periodic Safety Review Process on complex and not well known severe accident phenomena.

There are two volumes, representing the status of knowledge; *published in 10/2005*:

- The volume on ***"Methods for PSA"*** deals with:
  - Level1/2 interface (core damage state properties)
  - **Quality requirements for integral deterministic accident and source term analysis (MELCOR, ASTEC)**
  - Accident progression event tree (APET), issues to be considered
  - **Definition of release categories - source term**
  - Handling of uncertainties
- The volume on ***"Data for PSA"*** gives advice:
  - Quantification of branching probabilities in the APET for complicated issues
  - Specification when to use of generic, or plant-type specific or plant specific branching probability values

Some recommendations are explained in next slides

Best Practice - Severe Accident and Source Term Analyses

Develop adequate input for used codes – MELCOR & ASTEC used

- Requires high knowledge of code user on severe accident phenomena
- Need for adequate and sufficient information on plant specifics and design
- **Use real plant data without conservative assumptions** as for DBA analyses
- Need for **appropriate modelling of relevant plant specifics** and **all probable fission product release paths into the environment**
- Need for sufficient detail of nodalisation schemes for all components and buildings to allow a realistic simulation of NPP behaviour under severe accident conditions

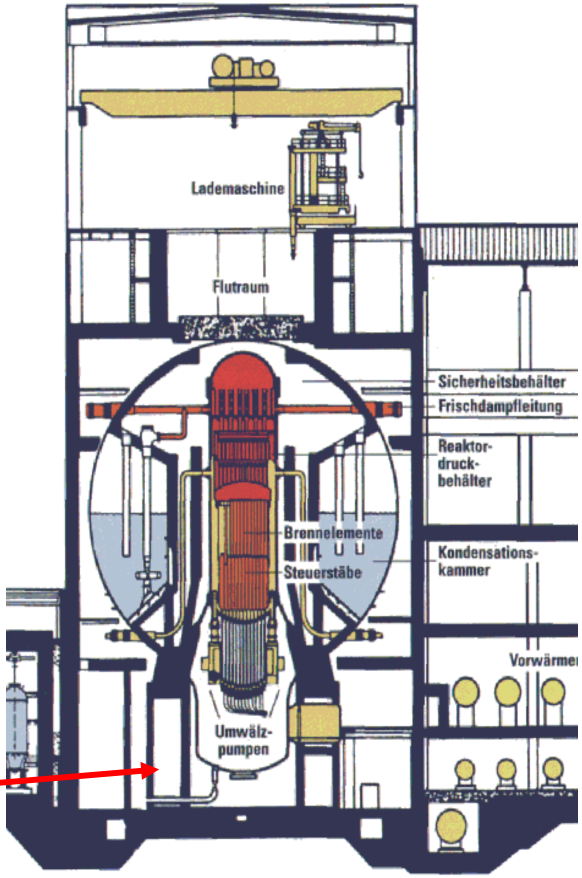
Validate developed input deck – MELCOR & ASTEC used

- Against real plant data for normal plant operating conditions
- By code to code comparisons with detailed codes (ATHLET-CD, COCOSYS, …)
- Main integral code results for different accident phases and timing of sequence should be in good agreement to detailed codes

Best Practice - Severe Accident and Source Term Analyses

Take into account all severe accident phenomena and source term aspects.

- **BWR-69 plant features**
  - Steel containment with:
    ~5500 m³ free volume
    ~2500 m³ water in wetwell
  - Internal air circulation system
  - Containment N2-inerted
  - Filtered containment venting connected to wetwell
  - RPV not coolable by flooding from outside spray system in drywell
  - RPV penetration failure expected
  - Containment head sealing made from organic material – low failure temperature
  - **Shortly after melt release from RPV**
    -> Failure of containment in lower position expected
    -> Releases through adjacent buildings

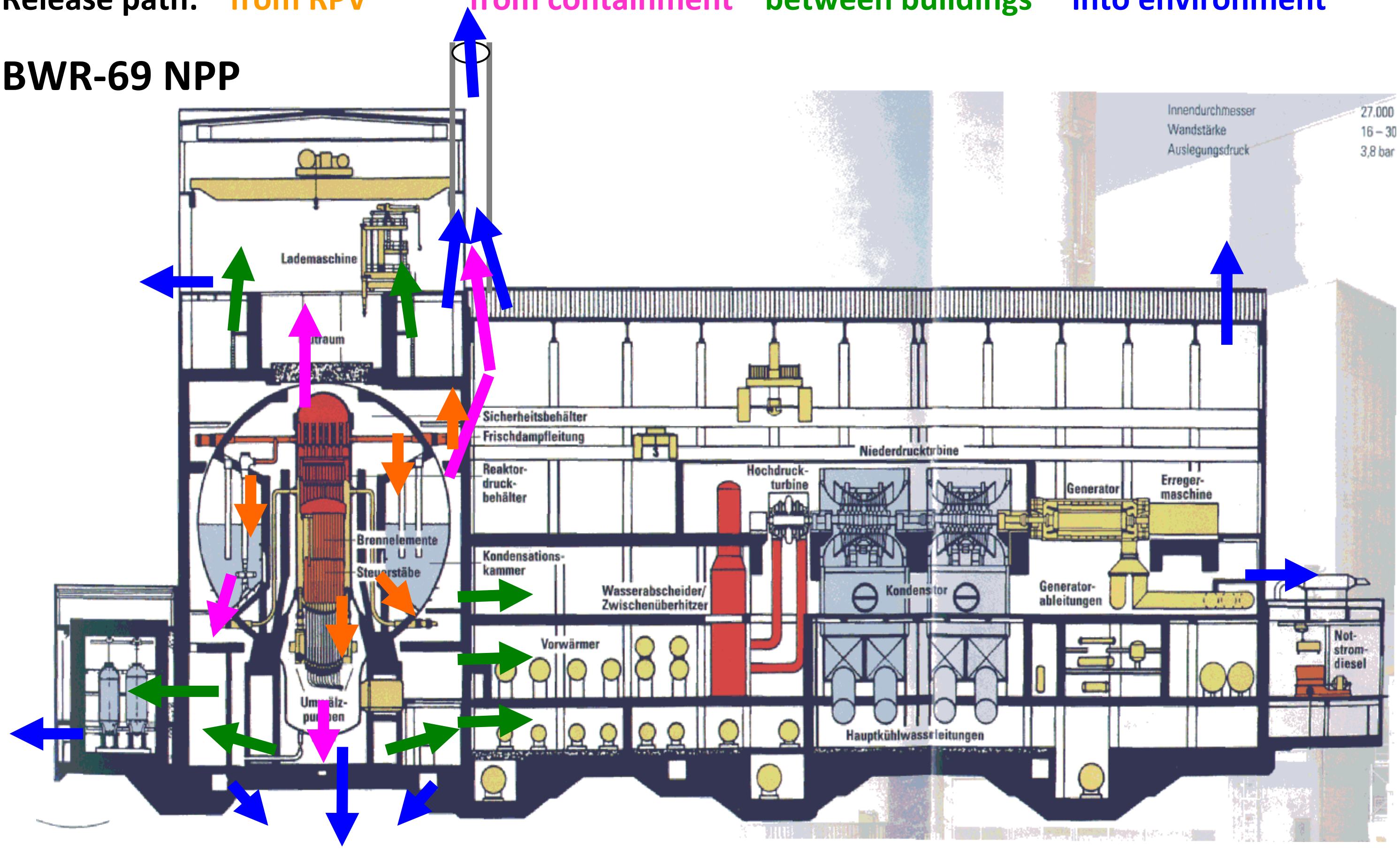


Best Practice - Severe Accident and Source Term Analyses

Determine all probable release paths for radio nuclides into the environment.

Release path: from RPV   from containment   between buildings   into environment

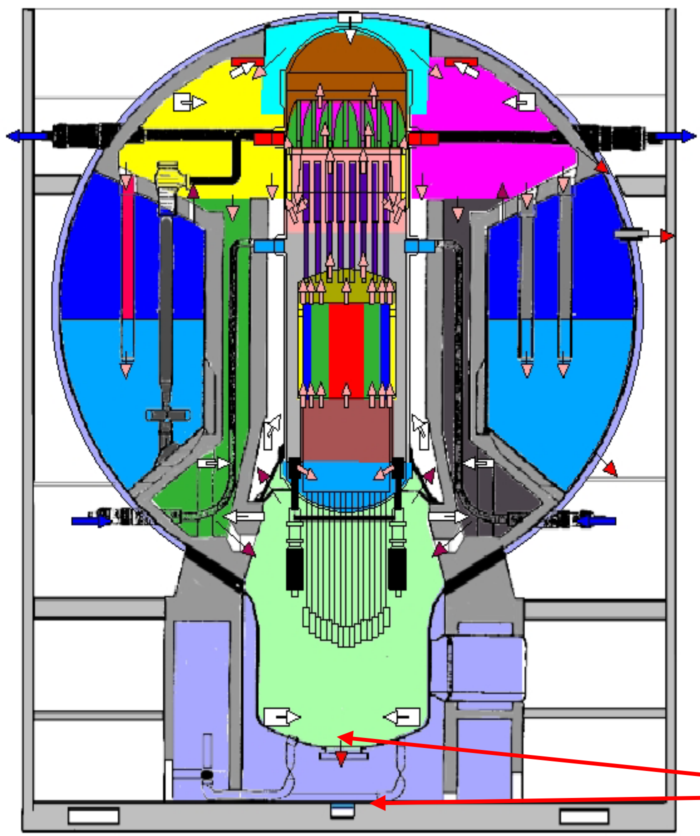
BWR-69 NPP



There exists more release paths than expected; relevance is not clear before study is made.

Best Practice - Severe Accident and Source Term Analyses

Develop adequate plant nodalisation schemes (MELCOR example) …



each coloured cell = one CV node of input deck

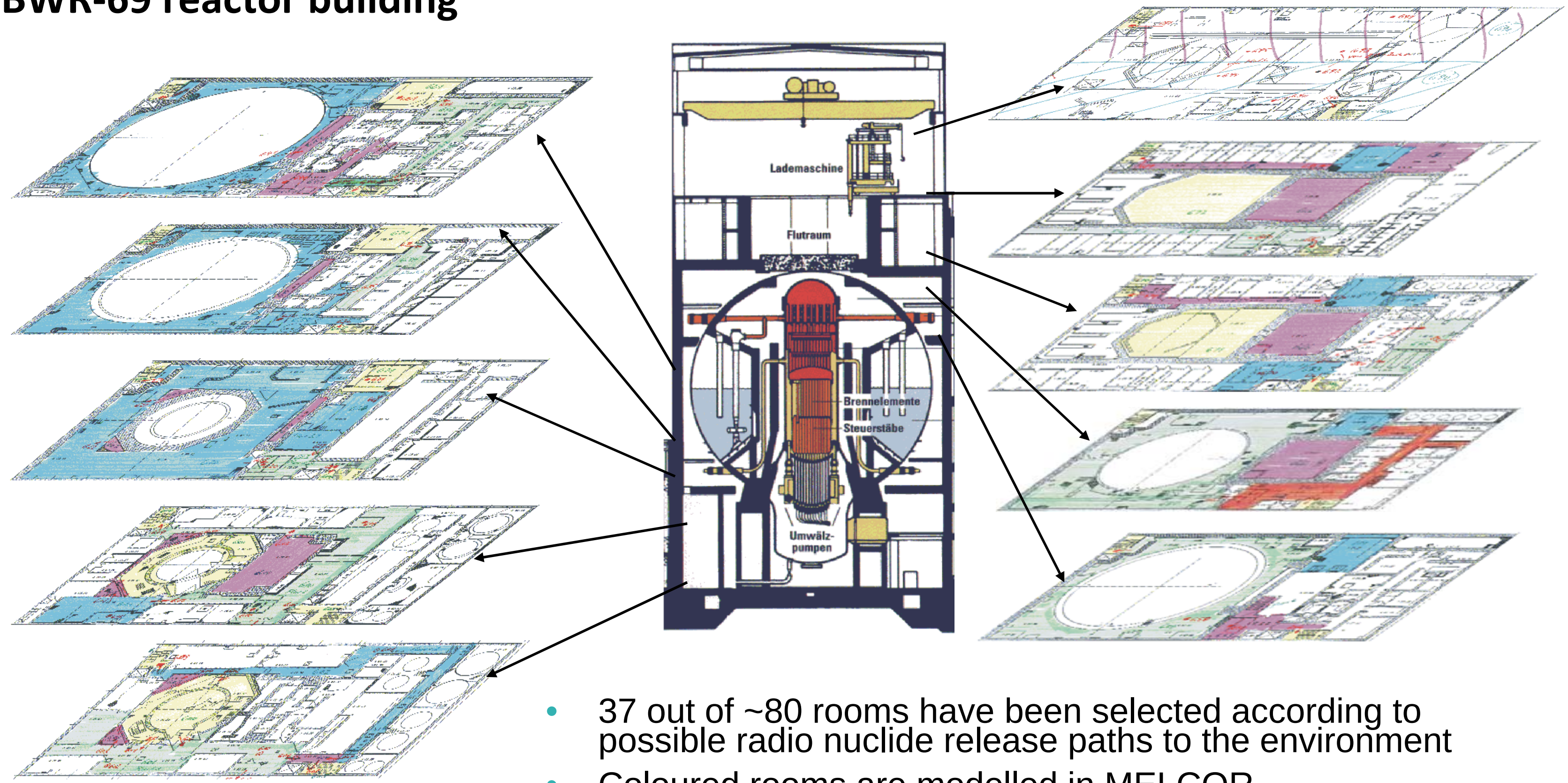
BWR-69 - RPV and containment

- Detailed RPV model to calculate void fraction in core, steam separation, RPV water level (15 volumes, 25 junctions, 85 structures)
- Detailed core model with 6 non-uniform radial rings and 15 levels + 6 levels in lower plenum **-> lessons learned from experiments**
- Definition of plant specific radio nuclide inventory and decay power
- Detailed containment model to consider plant specifics
(12 volumes, 33 junctions, 70 structures)
- Air ventilation systems in containment considered **-> contributes to gas mixing**
- Inertisation, filtered venting system, wetwell cooling systems considered as well
- 3 cavities, 2 of them outside containment in reactor building

Best Practice - Severe Accident and Source Term Analyses

Develop adequate plant nodalisation schemes (MELCOR example) …

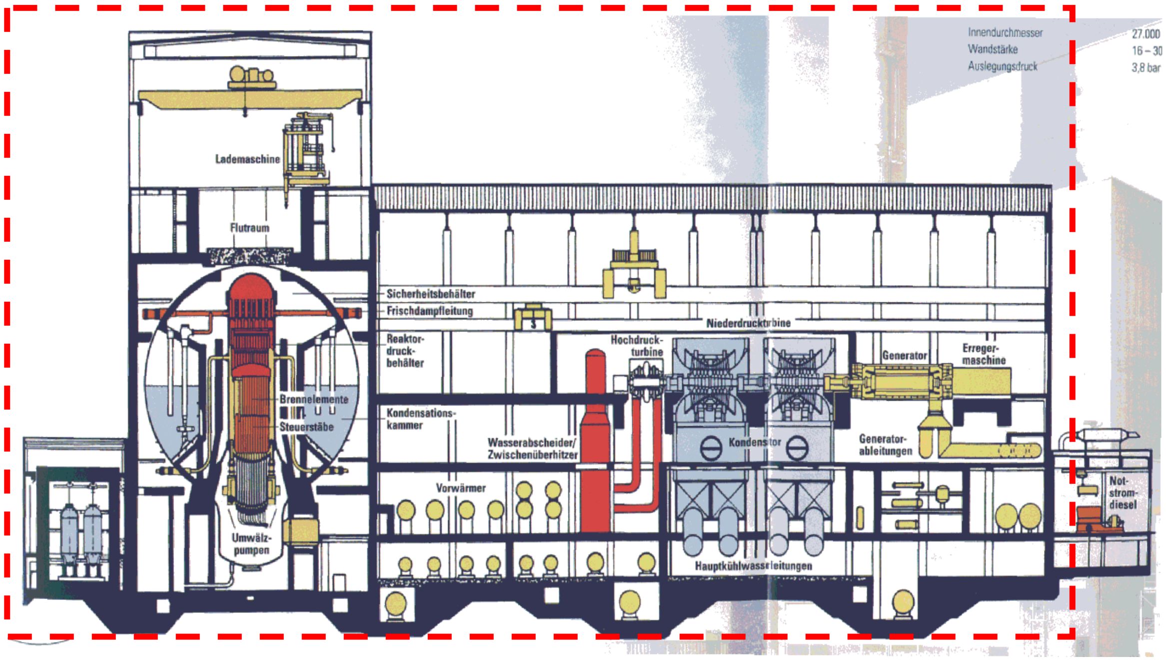
BWR-69 reactor building



- 37 out of ~80 rooms have been selected according to possible radio nuclide release paths to the environment
- Coloured rooms are modelled in MELCOR

Best Practice - Severe Accident and Source Term Analyses

Develop adequate plant nodalisation schemes (MELCOR example) …



BWR-69

Reactor Building:
- 37 volumes in 10 levels,
- 85 flow path (many doors, burst discs, etc.),
- **2 release path**
-160 heat structures

Turbine Hall:
- 15 volumes in 5 levels,
- 30 flow p., **2 release path**
- 65 heat structures

BWS Building:
- 1 volume, **1 release path**

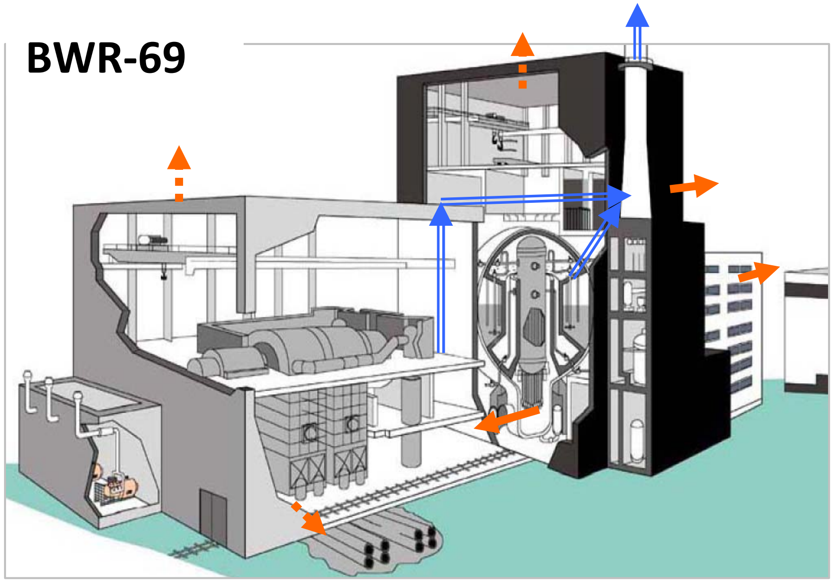
Off-gas System + Stack: 1 volume, **1 release path**

Environment: 4 volumes dependent on possible radio nuclide release paths

Best Practice - Severe Accident and Source Term Analyses

Make an appropriate model of relevant plant specific details …

- **Simulation of Doors and Burst Membranes between Rooms**
  - Many doors and burst membranes exist inside reactor and turbine building and between them
  - Failure of many doors, burst membranes, etc. due to containment failure at elevated pressure and H_2 combustions



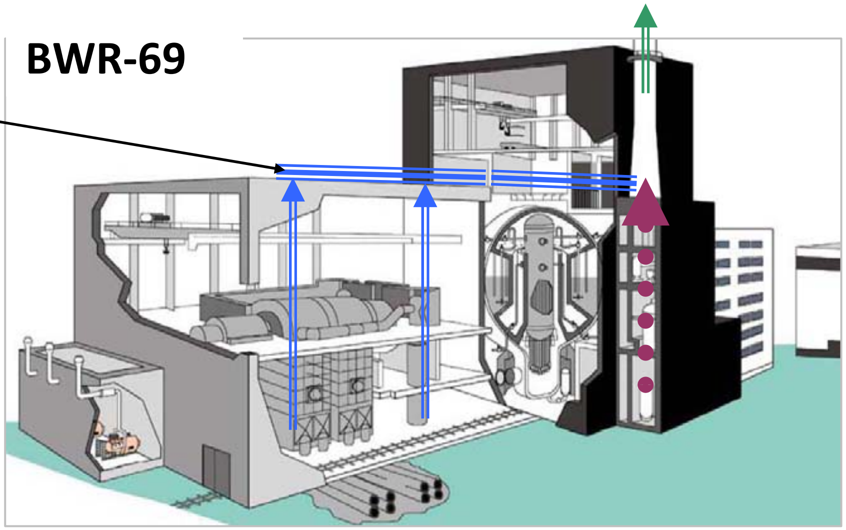
- **Modelling approach:**
  - Doors are not leak tight - small gaps simulated - simplifies pressure balance inside building
  - Failure of doors dependent on Δp according to door opening direction and design
  - No failure of doors in case of high water level on floor (doors not leak tight)
  - Re-closure of doors in case of stronger reverse flow modelled (10 % remain open)

  -> Influence on source term was analysed by sensitivity study

Best Practice - Severe Accident and Source Term Analyses

Make an appropriate model of relevant plant specific details …

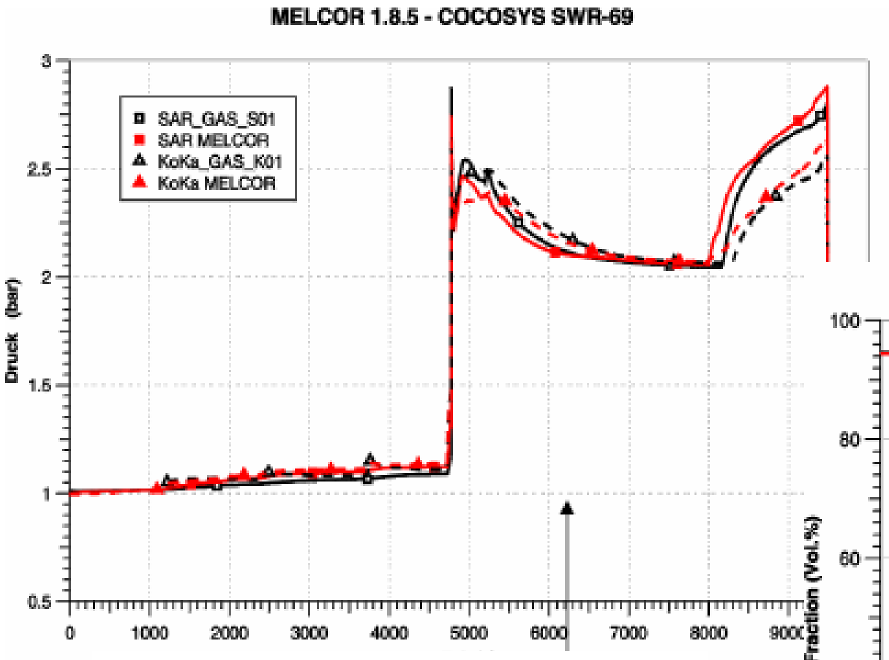
- **Air Ventilation Systems (example: Turbine Building)**
  - Sub-pressure in building during normal operation – systems switched off latest after containment failure
  - Off-gas line stays open
  - Enhanced mass flow from turbine building through stack into environment at containment failure



  - Buoyancy force driven mass flow through stack during long term
  - Sub-pressure build up in turbine and reactor building
    -> Reverse mass flow direction into buildings though leaks, open doors, etc.
  - **Details are important for source term calculation**
    -> Off-gas system and stack modelled separately

Best Practice - Severe Accident and Source Term Analyses

Validate developed input deck versus detailed code results …

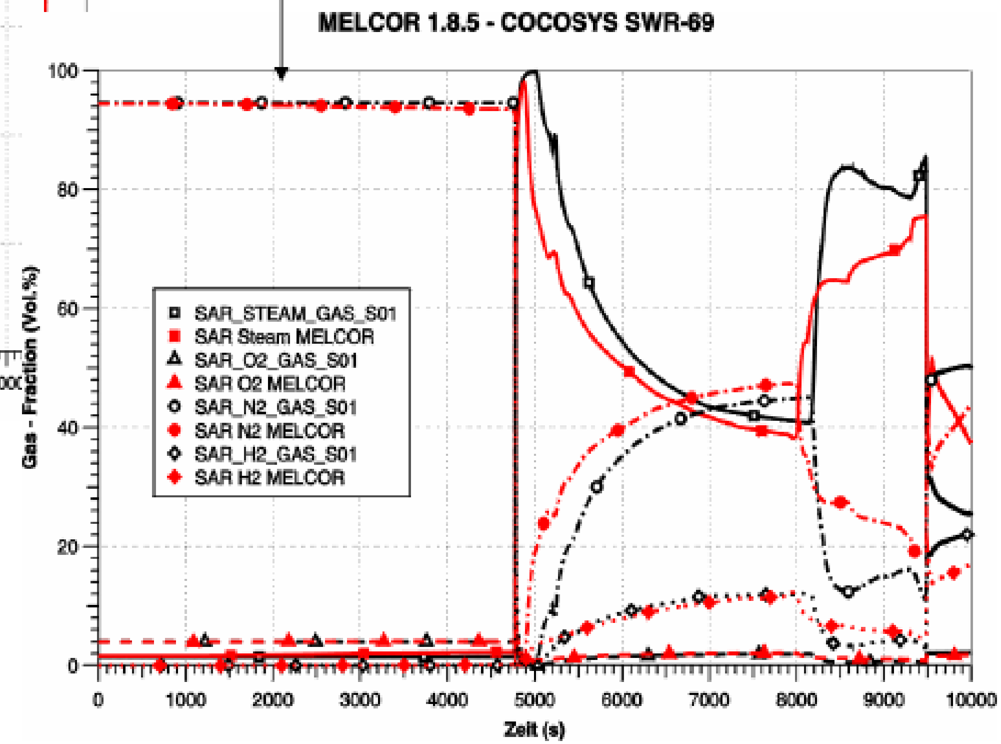


Drywell & wetwell pressure

MELCOR - COCOSYS

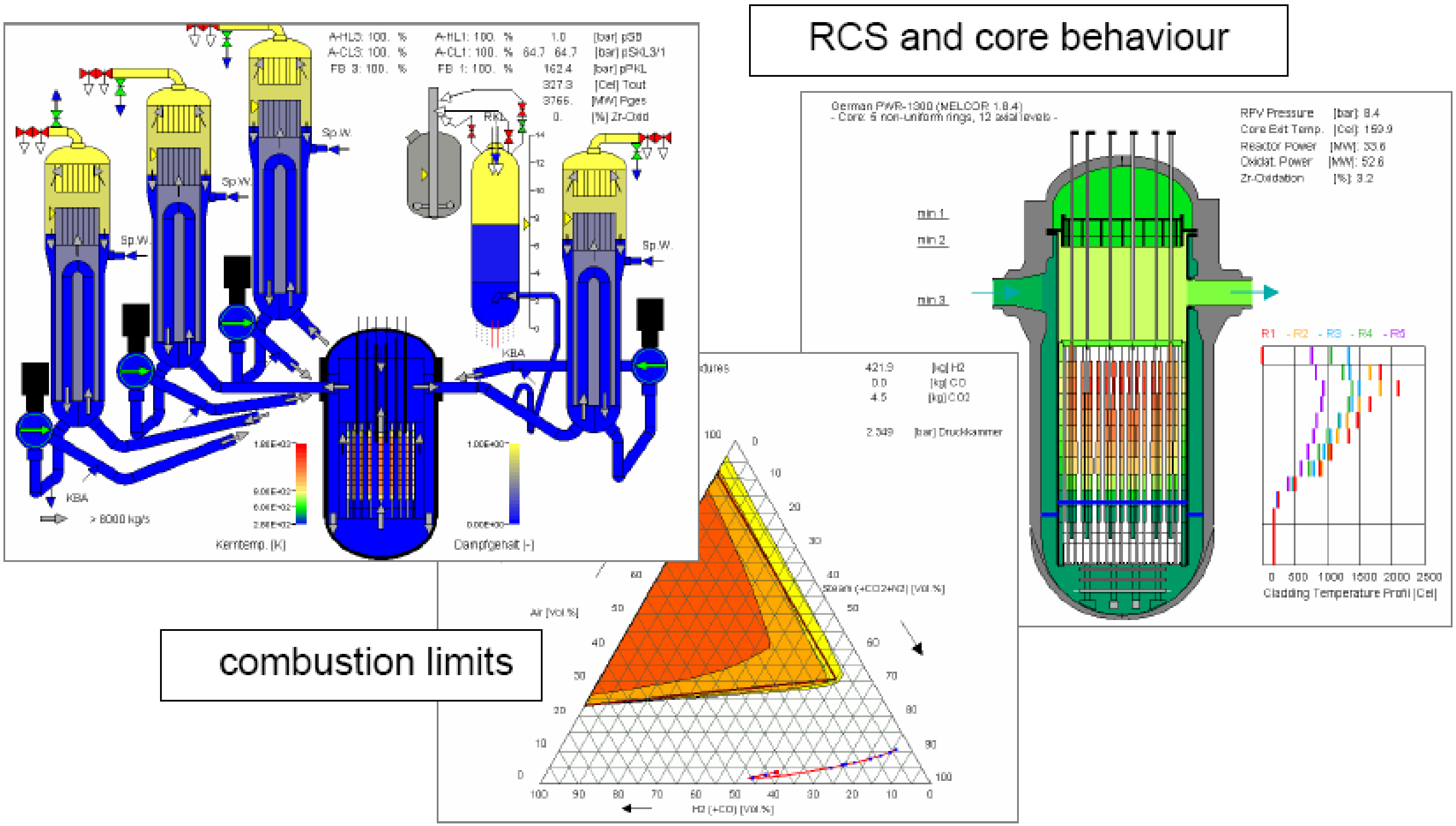
Lower drywell gas composition

MELCOR - COCOSYS



Best Practice - Severe Accident and Source Term Analyses

Use visualization tools to check appropriate modelling of relevant plant specifics …



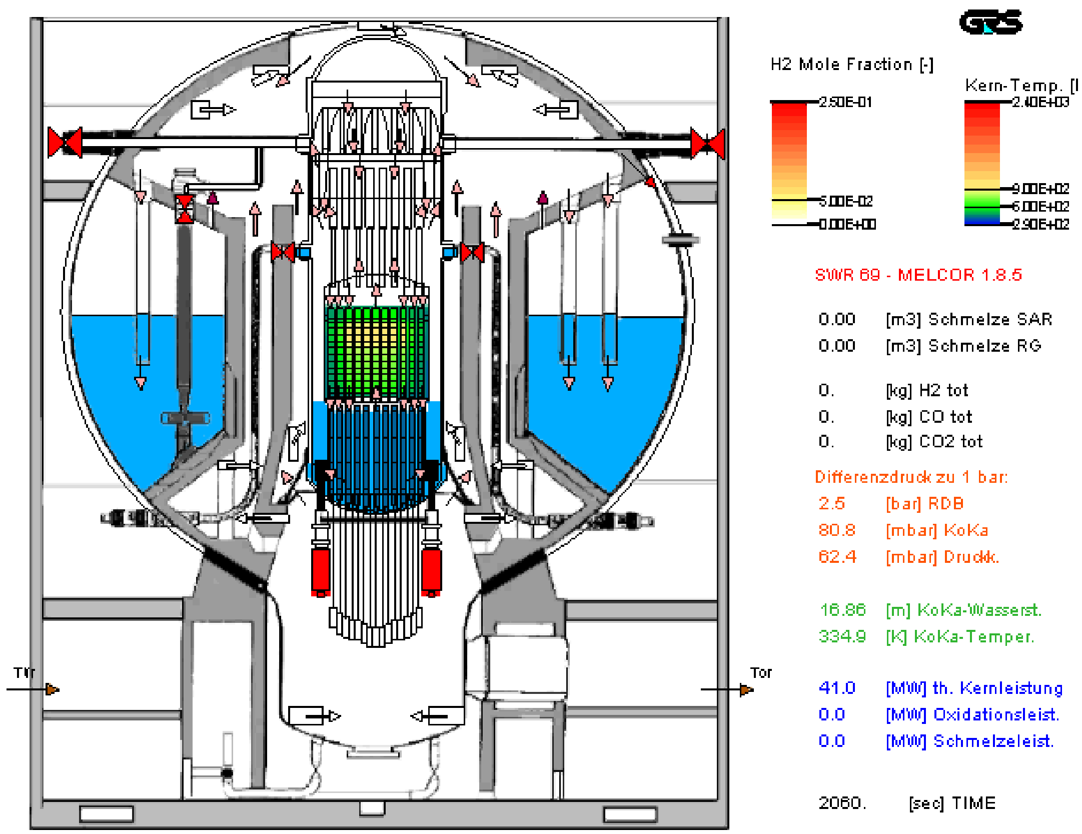
Best Practice - Severe Accident and Source Term Analyses

Assess the results carefully and determine relevant phenomena ...

BWR-69:

H_2 release through RPV-SV into wetwell before RPV failure

- Low H_2 generation in this case due to steam starvation
- Water siphon in lower plenum and low water temperature due to injection of two service systems for CR and MCP
- Extended H_2 generation after melt relocation into lower plenum water pool -> quenching and evaporation
- Early local RPV failure
- Very high H_2 generation in HP cases and other LP cases with over-feeding the RPV before core melting
- Containment inertised -> strong combustions in buildings after containment failure possible



ATLAS Simulator of GRS

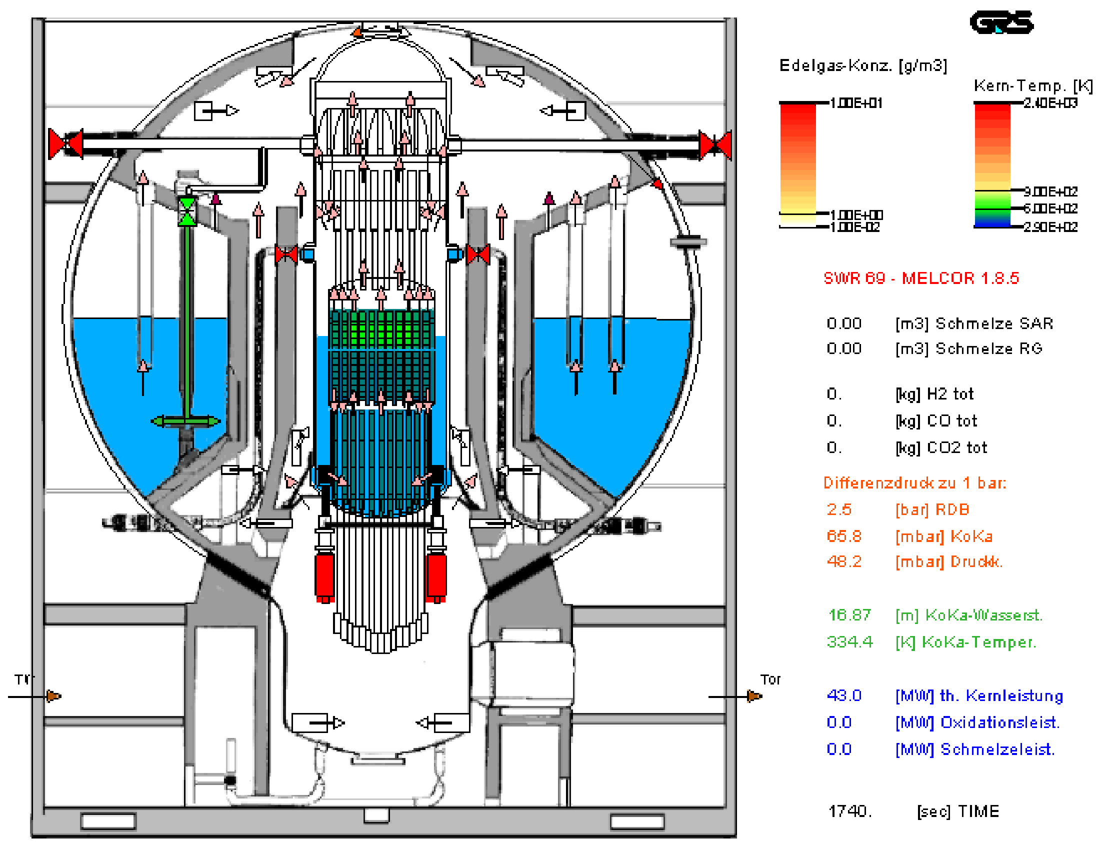
Best Practice - Severe Accident and Source Term Analyses

Assess the results carefully and determine source term relevant phenomena.

BWR 69:

Noble gas (NG) release from RPV before / after containment failure

- NG and aerosol accumulation in wetwell in early phase
- Aerosol retention in wetwell
- NG transfer from wetwell to drywell through small pressure equalisation pipes
- High NG concentration in whole containment after RPV failure
- High peak release of NG and aerosols into buildings at containment failure
- Limited release after containment failure -> still high NG and aerosol content in containment in long term phase

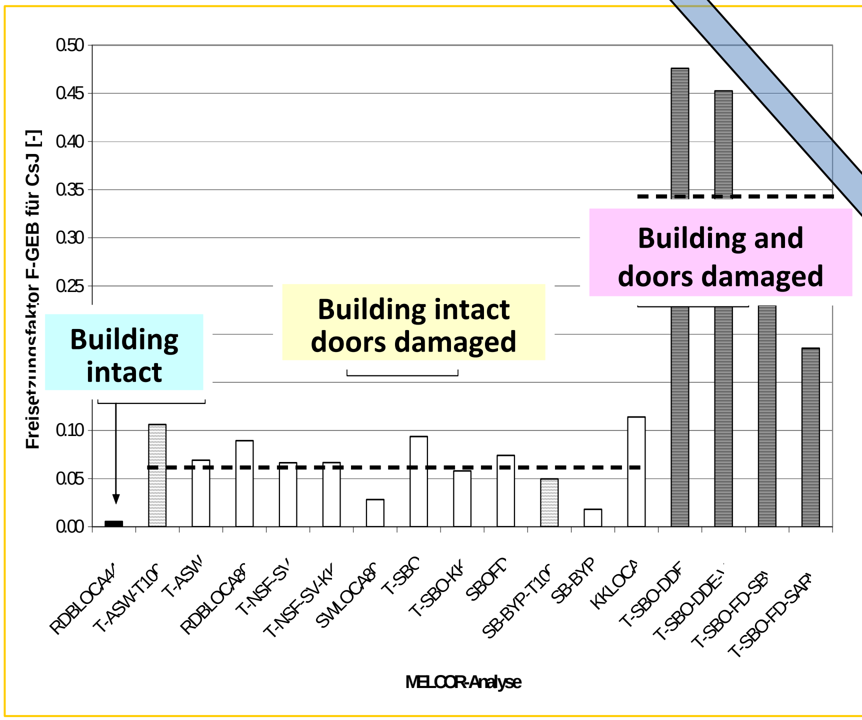


ATLAS Simulator of GRS

Best Practice - Severe Accident and Source Term Analyses

Determine source term data for PSA L2 dependent on plant status/behaviour.

Deterministic analysis results:
Release fractions of CsI from buildings to environment for various simulations



	CsI	CsOH	Te	Kr
Building intact	**0.005**	**0.01**	**0.005**	**0.24**
Door damaged	**0.07** (0.02 - 0.11)	**0.07** (0.02 - 0.13)	**0.10** (0.02 - 0.21)	**0.88** (0.69 - 0.99)
Door and building damaged	**0.34** (0.19 - 0.48)	**0.33** (0.20 - 0.46)	**0.37** (0.19 - 0.53)	**1.0**

Event tree input:
Release fractions of CsI, CsOH, Te and Kr from buildings to environment

Summary – Best Practice

- MELCOR was the main tool used at GRS within PSA level 2 studies and to support the development of SAM measures in the past.
- Knowledge is transferred to ASTEC applications.
- Detailed MELCOR nodalisation schemes have been used always to simulate plant specific details and relevant radio nuclide release paths.
- Extensive validation of MELCOR input deck performed by code to code comparisons with detailed codes.
- “Best estimate” data/results have been used/gained by analyses.
- Recommendations given in German PSA Guidance document are applicable and very helpful.
- Long(er) CPU time needed for MELCOR input was accepted to get higher quality of results (factor of 5 – 10 of process time).
- Visualisation of analyses results with ATLAS was very helpful to understand NPP behaviour under severe accidents.
- Results are ready for use for SAMG development and training.

Session 5

Severe Core Damage Accident Analysis for a CANDU Plant

P.Mani Mathew, S.M. Petoukhov, M.J. Brown and B. Awadh

Severe Accidents Section

Reactor Safety Division

AECL Chalk River, Canada

ISAMM09

26-28 October 2009

Paul Scherrer Institute, Switzerland

AECL EACL

UNRESTRICTED / ILLIMITÉ

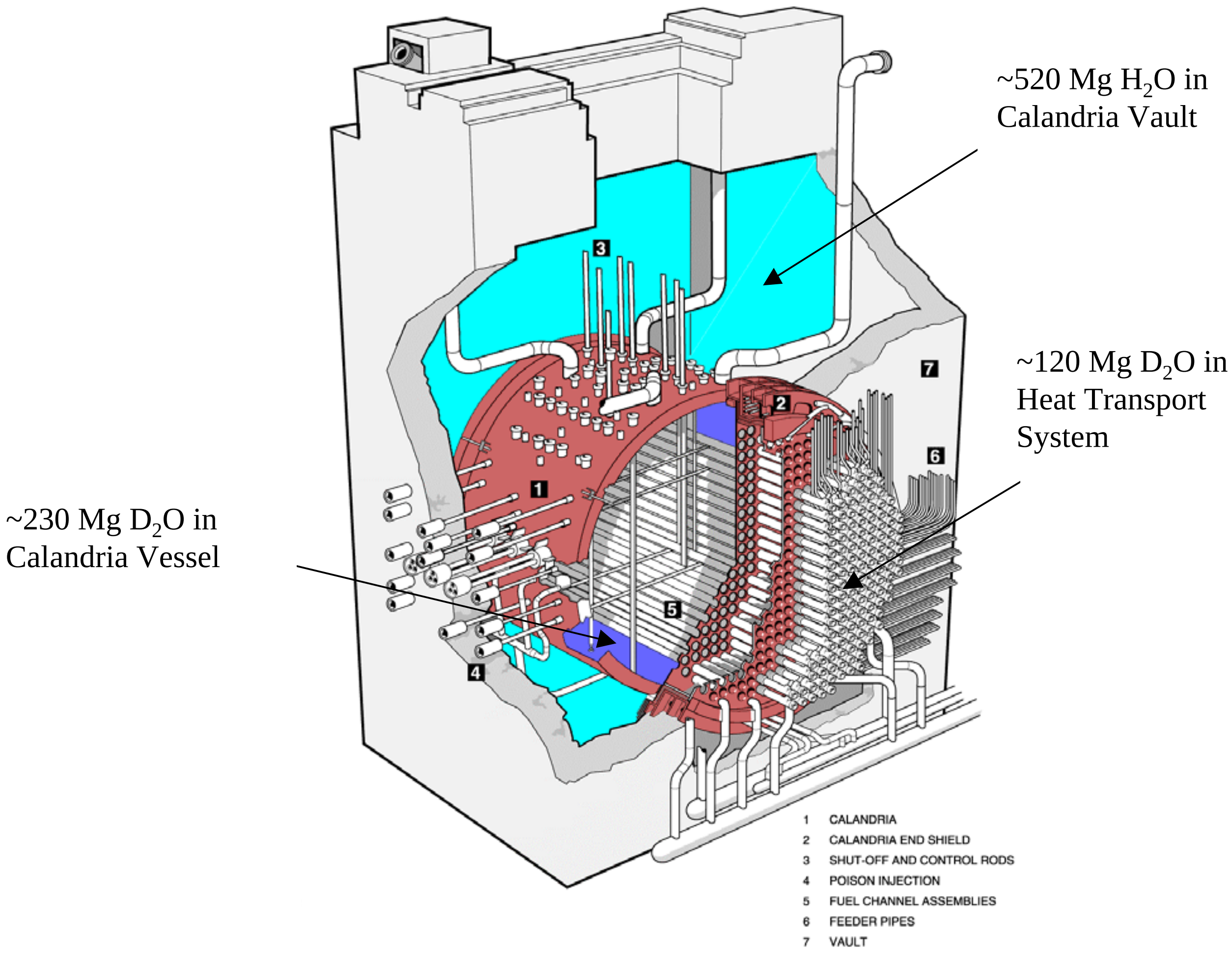
Outline

- **Introduction**
  - Definitions, conditions for core damage accident
- **Accident progression phenomenology**
  - CANDU-specific phenomena
- **MAAP4-CANDU Code**
  - General description and models
- **Application of MAAP4-CANDU code for Point Lepreau station refurbishment project**
  - Accident sequences analyzed
  - SAM measures based on analysis
- **Summary**

Introduction

- **Severe Core Damage Accident (SCD)**
  - **Accident in which substantial damage is done to the reactor core structure whether or not there are serious off-site consequences**
    - **SCD when Reactor Cooling System and Moderator back-up heat sinks are unavailable in a CANDU.**

CANDU 6 Reactor Core



Severe Core Damage Progression

- **Slow progression of Severe Core Damage in CANDU-6**
  - **Significant quantity of water surrounds the core**

- **Moderator Plays an Important role as a Heat Sink in LOCA/LOECC (Design Basis Accident)**

Design Basis Accident: LOCA/Loss of ECC but Moderator Heat Sink Available

- Primary system depressurizes, cooling to fuel reduced
- Fuel heats up, deforms and transfers heat to pressure tubes
- Pressure tubes heat up and sag into contact with calandria tubes
- Heat from fuel is removed by moderator circulation system
- Core geometry is maintained, but fuel can be severely damaged

Moderator Plays an Important role as a Heat Sink

Severe Core Damage Accidents: In-Vessel Core Damage

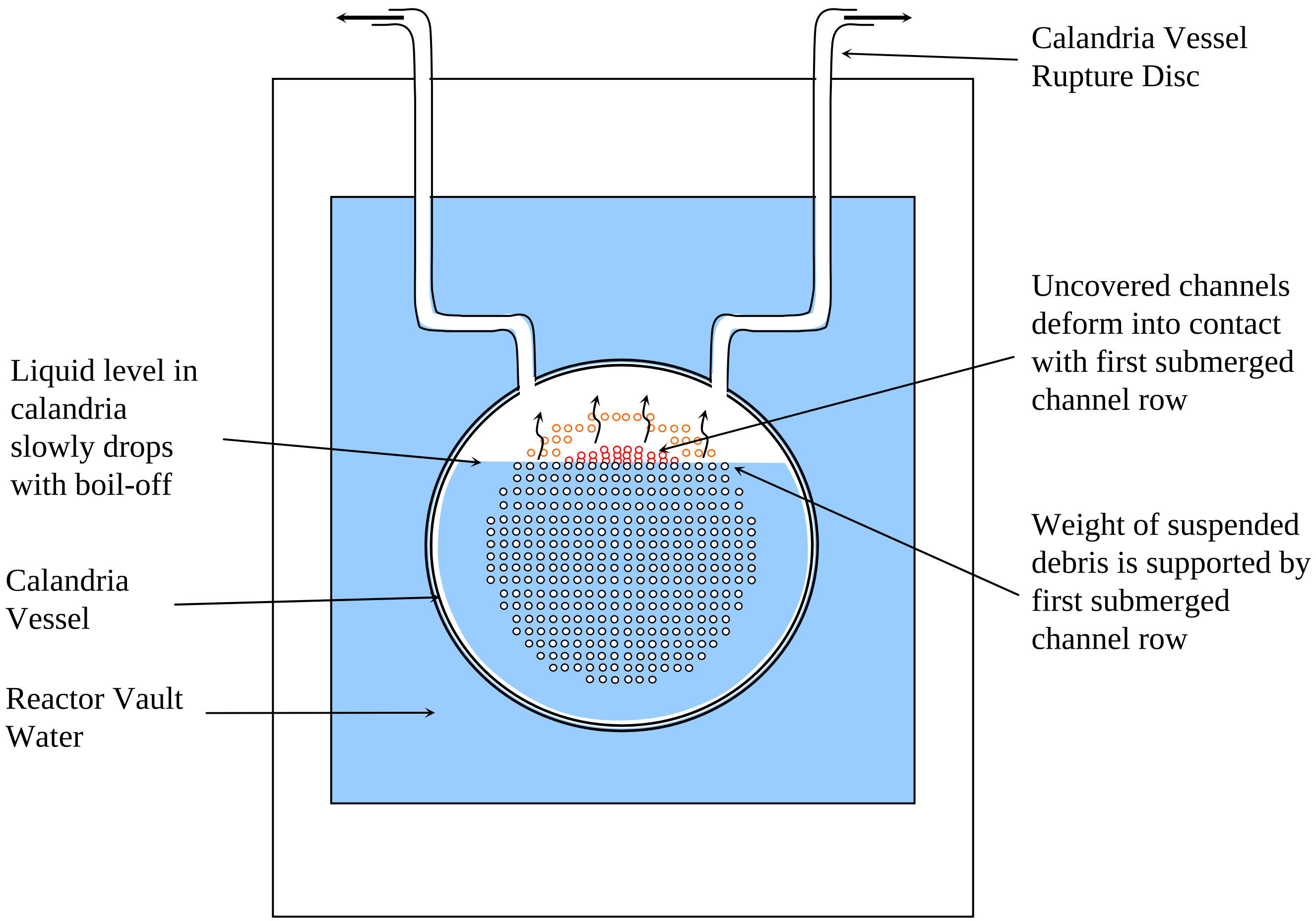
- **Loss of Coolant Events with ECC impairments and Loss of Moderator Heat Sink**
  - **Fuel Channels Heat Up**
  - **Moderator Boils Off**
  - **Core Disassembly Occurs**
  - **Debris relocate to water-cooled Calandria Vessel Bottom**
- **Reactor Vault (Calandria Vault) Cooling and Make-up Water systems Play an Important role as a Heat Sink**

Severe Core Damage Accidents: LOCA-LOECC, loss of Moderator heat sink

- **Typical sequence of events:**
  - **Primary system depressurizes, cooling to fuel reduced**
  - **Fuel heats up, deforms and transfers heat to the pressure tube**
  - **Pressure tubes heat up and sag into contact with calandria tubes**
  - **Heat load from fuel channels slowly boils off the moderator**
  - **Uncovered fuel channels gradually collapse, break up and are quenched in remaining moderator**
  - **After all moderator is expelled, debris bed heats up**
  - **Reactor vault water inventory keeps calandria vessel intact**

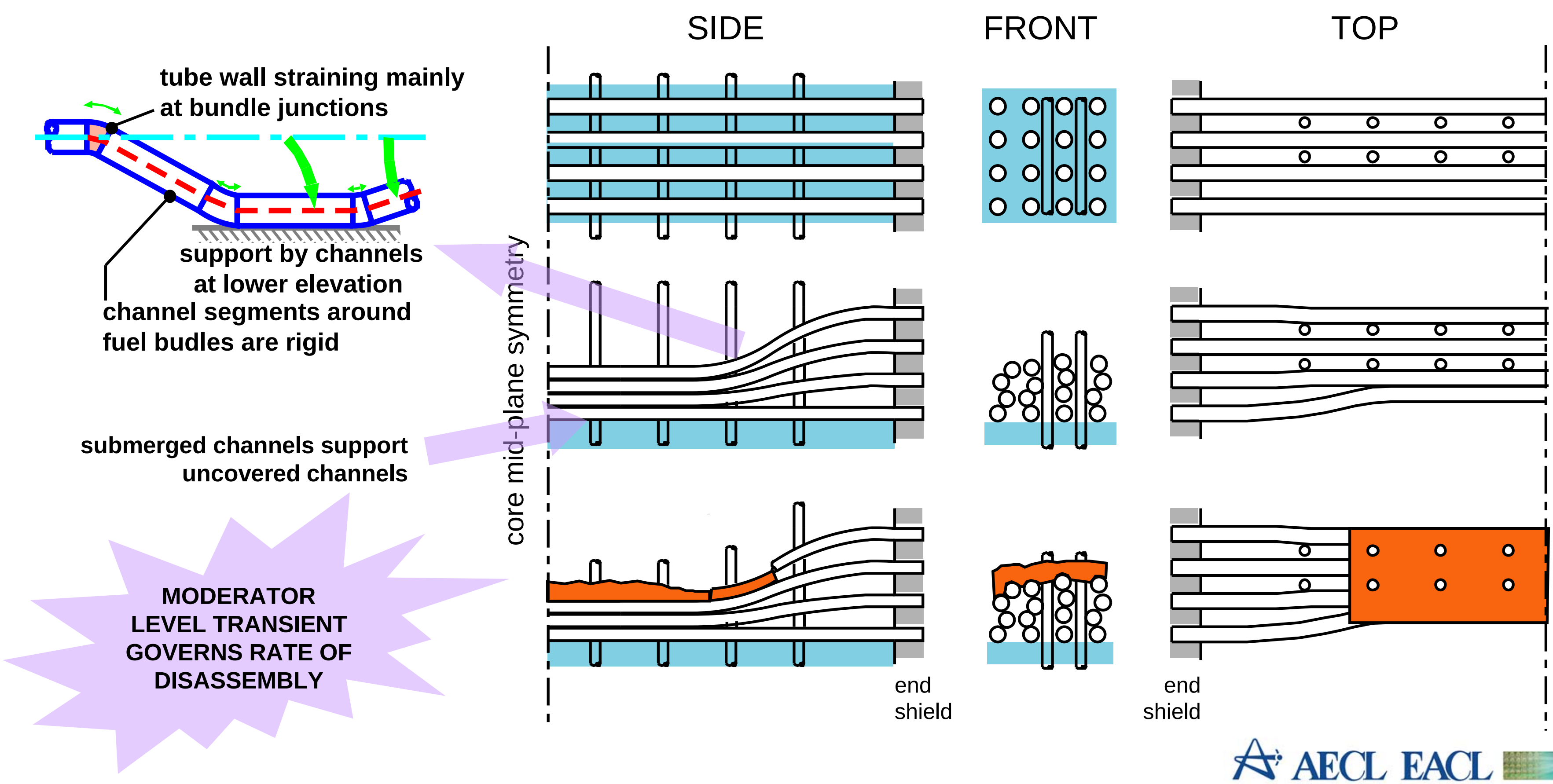
    - **RCS inherently depressurized before Core Disassembly**

A schematic showing the uncovery of top fuel channels following moderator expulsion for CANDU 6



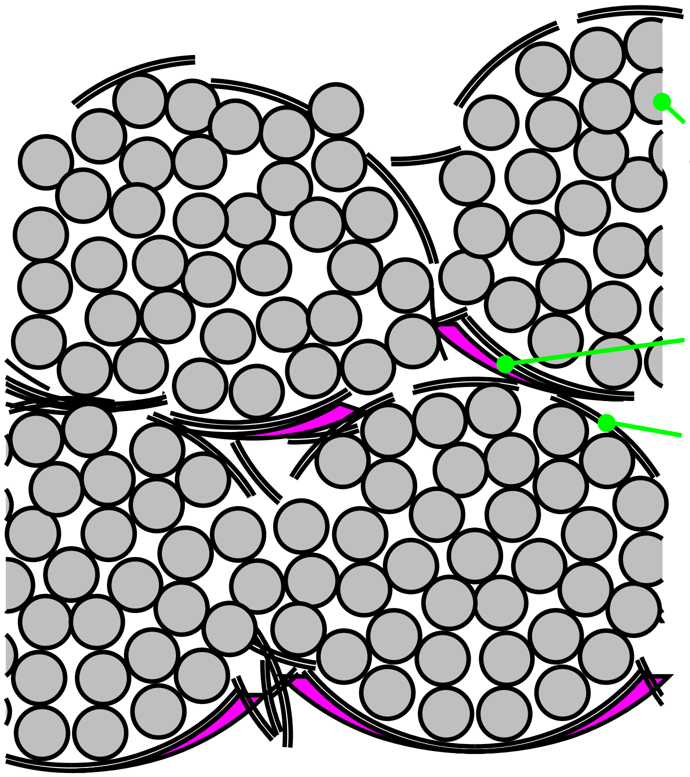
CHANNELS BREAK UP BY SAGGING

Analyses & Small Scale Tests

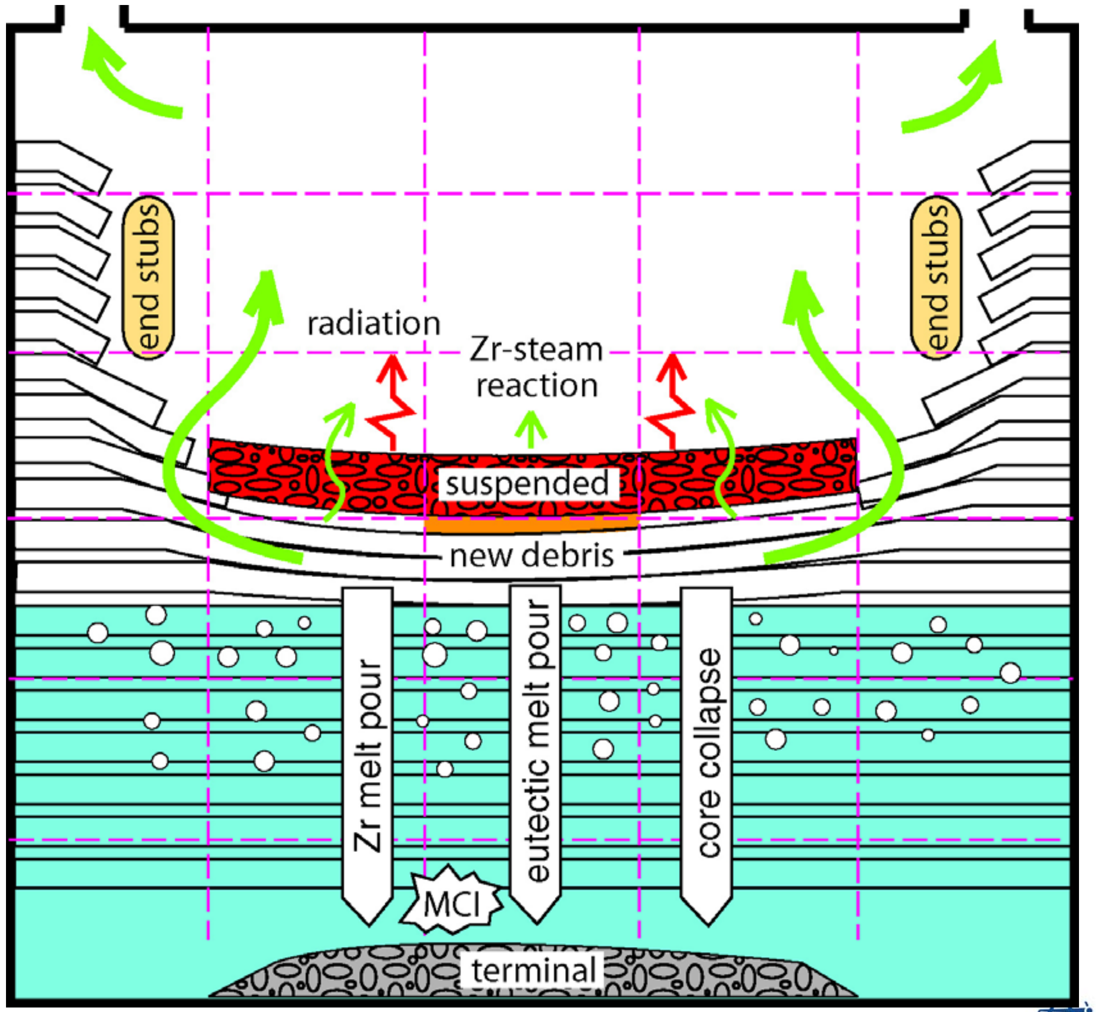


SUSPENDED DEBRIS

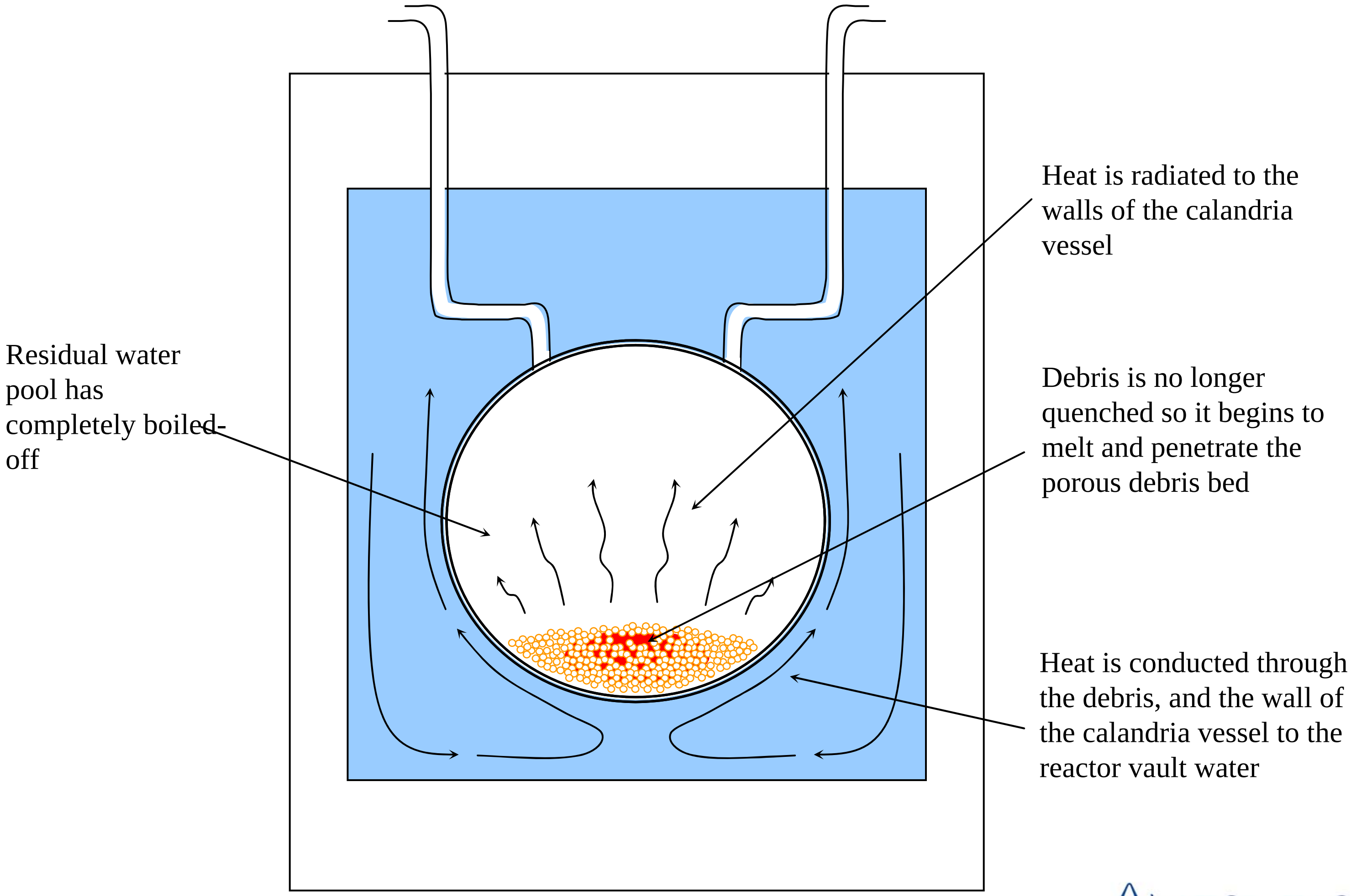
- **suspended debris mass builds up with time**
- **steam access into debris interior more difficult with time**
- **debris weight supported by first submerged row of calandria tubes**
- **load-bearing capacity of CT is not unlimited**



A schematic showing the various phenomena inside the calandria vessel during the transient

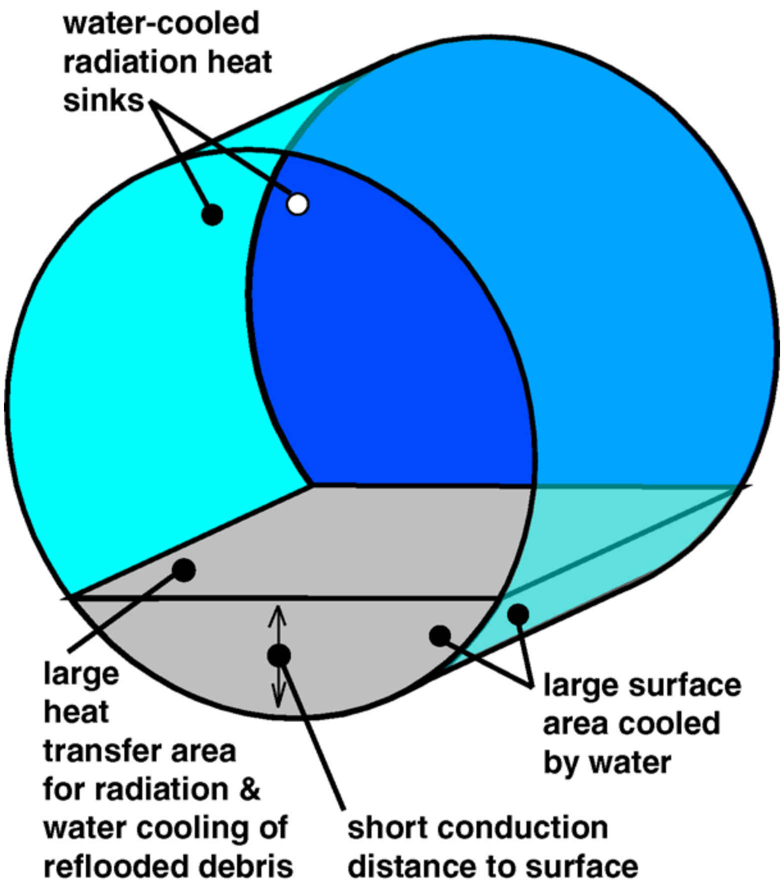


Consolidated terminal debris bed; beginning of molten corium formation near the top surface and the evolution of natural circulation in the reactor vault water



DEBRIS COOLABILITY

- **Cylindrical geometry well-suited for external cooling & flooding**
  - **large surface-to-volume ratio**
  - **surrounded by water jacket**
  - **Make-up to reactor Vault**



MAP4-CANDU: Background

- MAAP (Modular Accident Analysis Program) is an integrated code designed for Severe Accident Consequence Analysis in nuclear plants
- MAAP is owned by EPRI
- MAAP developed by Fauske & Associates Inc. (FAI), used by 50+ international PWR/BWR utilities

- MAAP-CANDU, based on MAAP-PWR / BWR, developed by FAI/OPG/AECL
- Ontario Power Generation (OPG) is the code licensee (code holder)
- AECL holds a sub-license from OPG

MAAP4-CANDU: Background (cont'd)

- MAAP-CANDU is the primary tool for for assessing severe core damage accident progression and severe accident management in CANDU plants
- The main distinguishing features of MAAP-CANDU are models of the horizontal CANDU-type fuel channels and CANDU-specific systems such as:
  - Calandria vessel
  - HTS
  - Containment Systems: dousing spray, local air coolers, etc.
- MAAP-CANDU contains CANDU core module developed by Ontario Power Generation (OPG)
- Lumped parameter code
- MAAP-CANDU is a Canadian Industry Standard Toolset (IST) code

MAAP4-CANDU Code

- **MAAP4 CANDU can assess the influence of Severe Accident Management (SAM) strategies to mitigate and recover from an accident state**

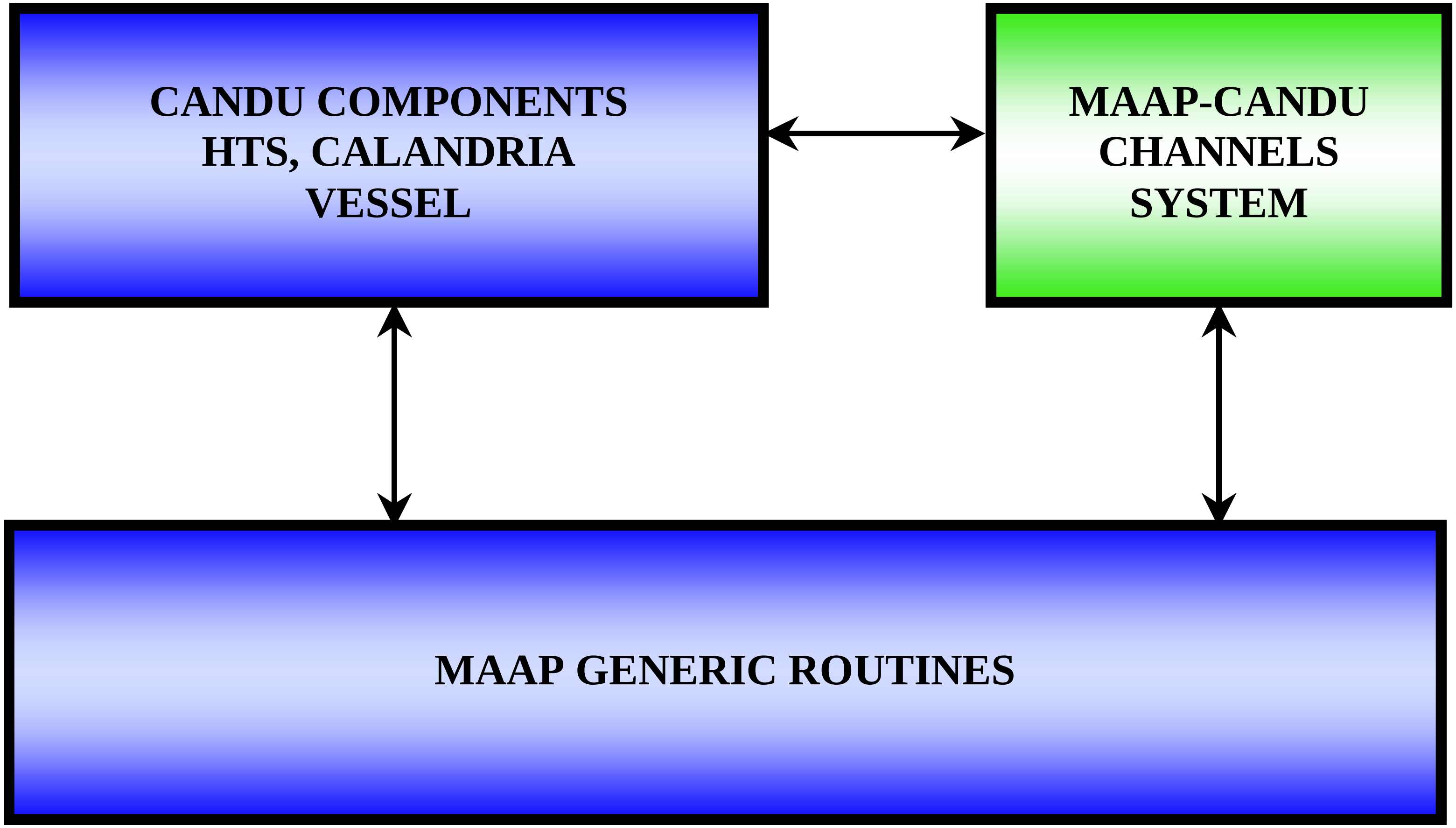
Sequences, resulting in severe core damage, that can be simulated by MAAP4-CANDU:

- **Station Blackout sequence**
- **Large LOCA**
- **Small LOCA**
- **Steam Generator Tube Rupture**
- **Feeder Stagnation Break**
- **Main Steam Line Break**

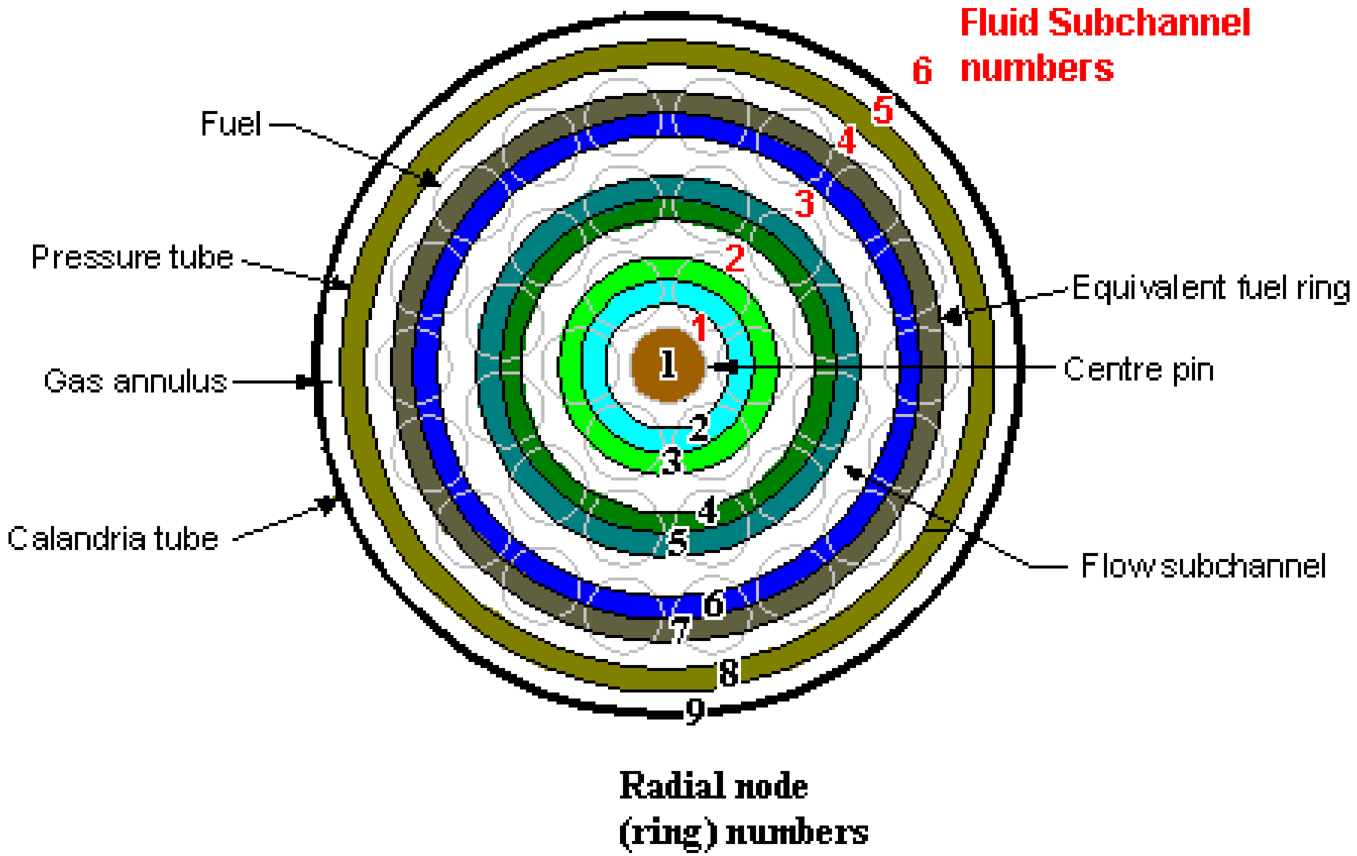
Key Generic Phenomena Specific to CANDU

- **Fuel Channel behavior**
- **Core disassembly**
- **Calandria vessel behavior**
- **Reactor Vault behavior**

MAAP-CANDU Basic Architecture

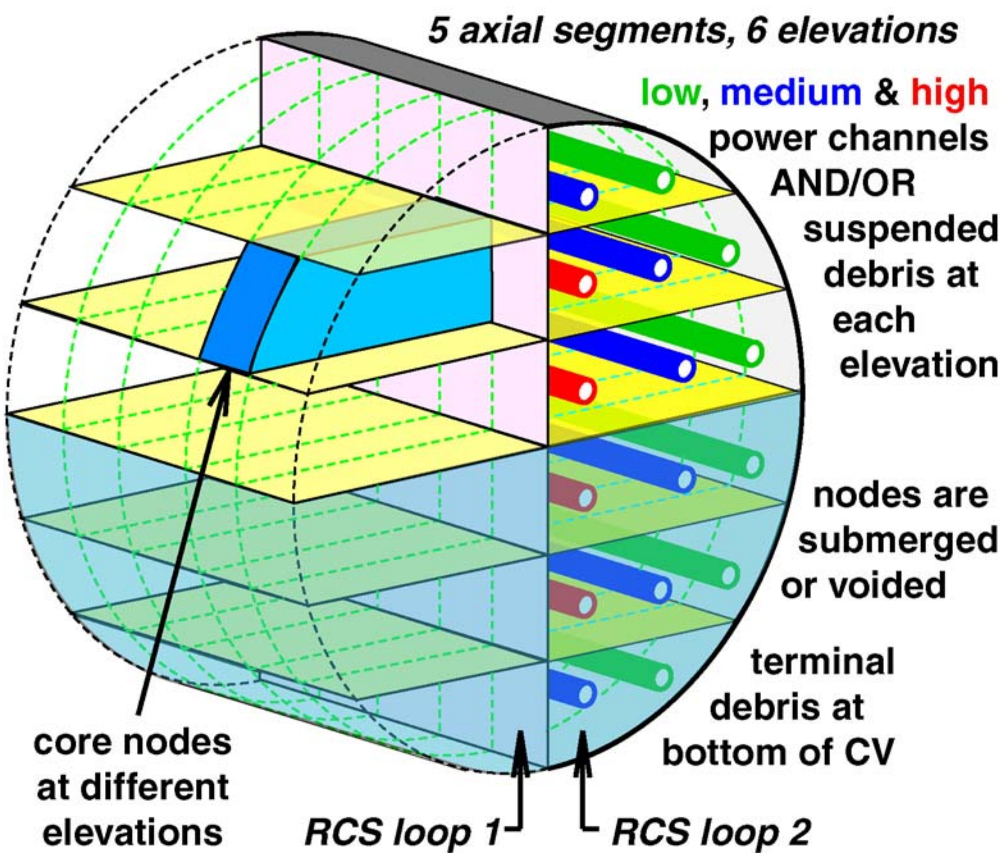


Nodalization of Fuel Channel



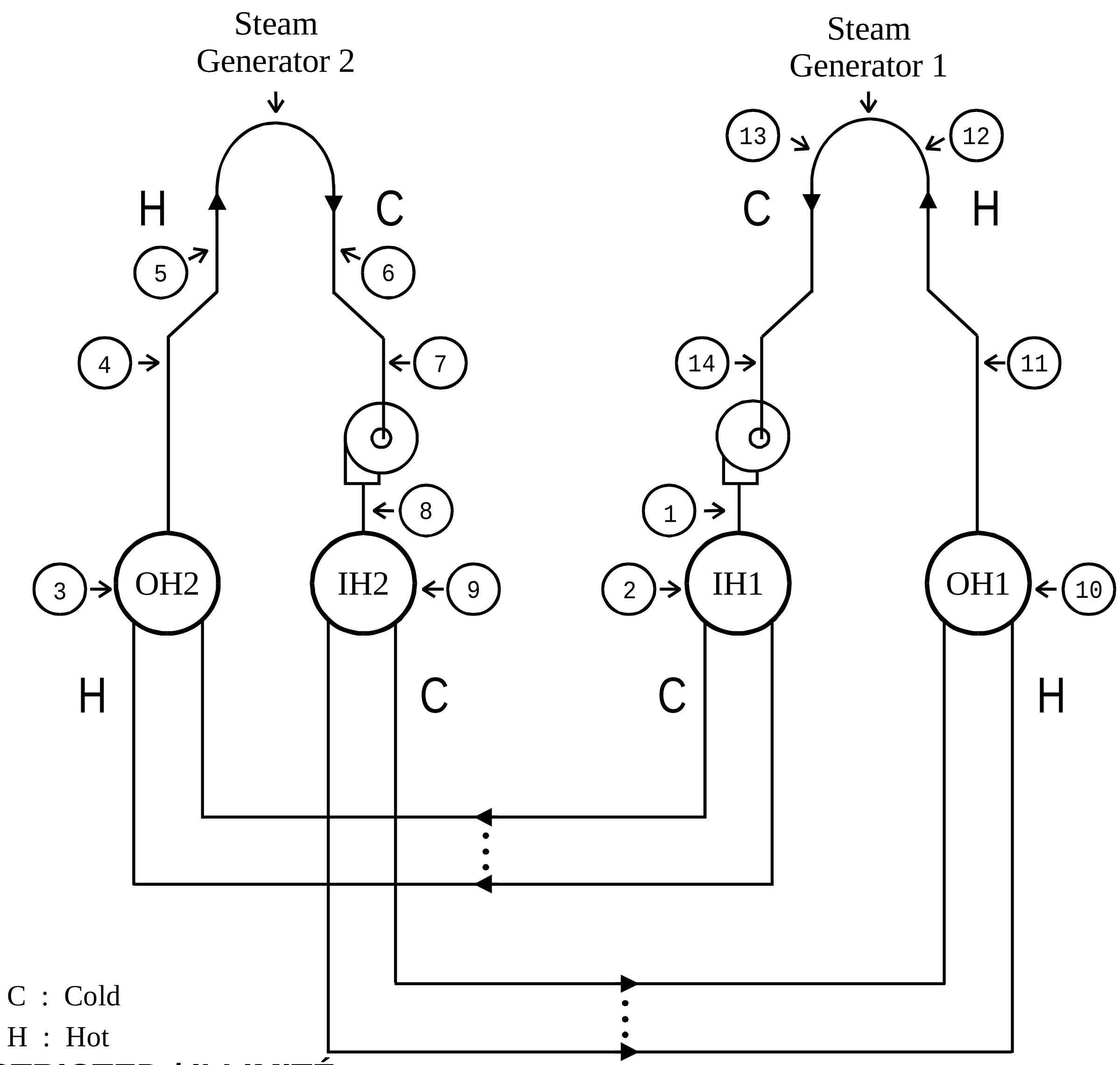
CANDU 6:COMPLEX NODALIZATION FOR CORE DISASSEMBLY

- Channels heat up & break up at different rates (380 channels represented by 36 characteristic channels)
- Intact channels & debris coexist
- Same CV water level in all axial nodes
- Suspended debris mass differs in axial nodes



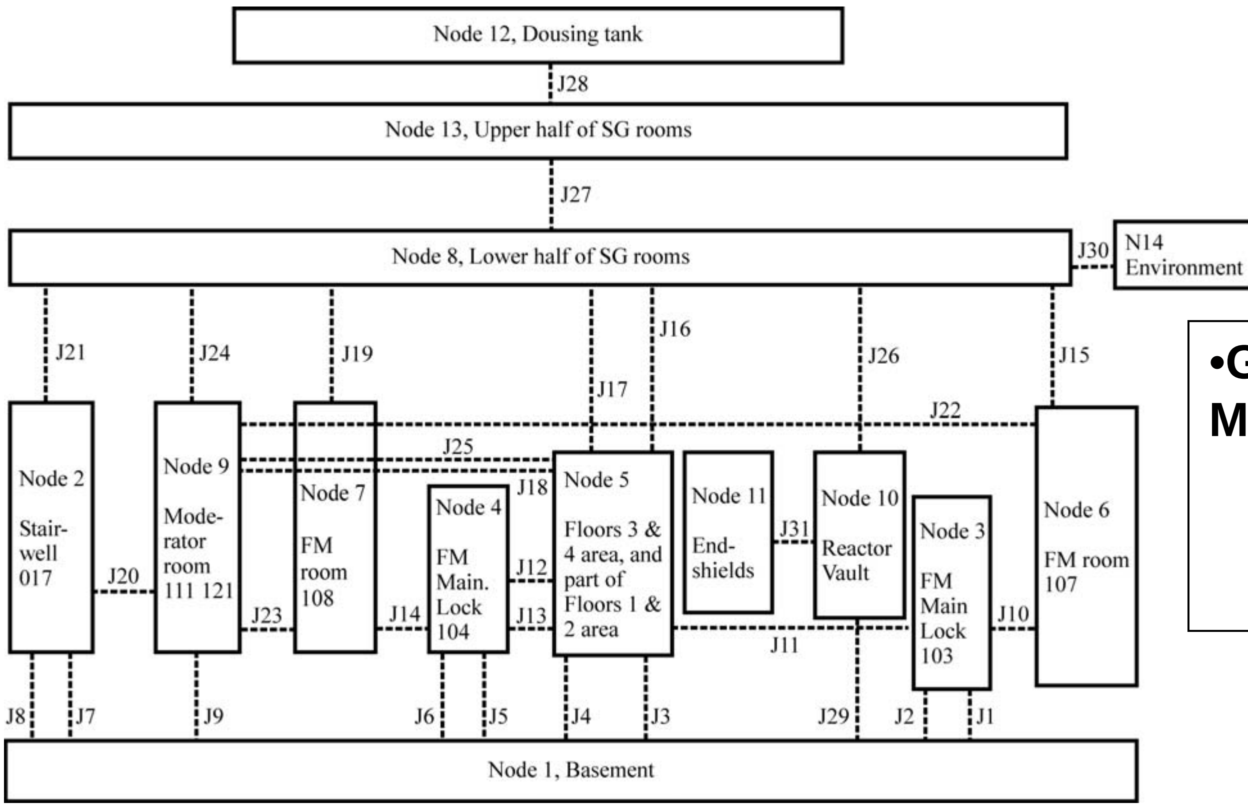
AECL EACL

Nodalization of CANDU 6 PHTS



- -14 Nodes/Loop
- -SG primary side 2 nodes (hot & Cold), secondary side 1 node

Nodalization of CANDU 6 Containment

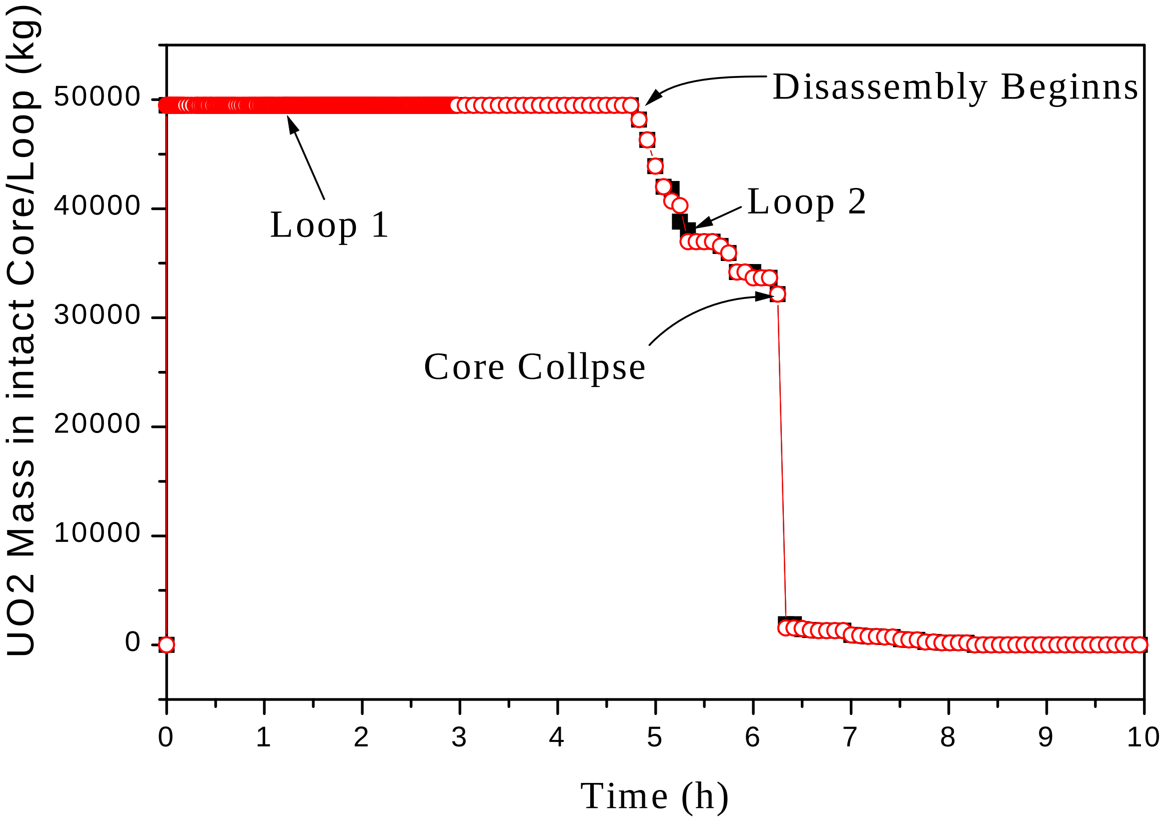


- **Generalized containment Model**
  - **13 Nodes**
  - **31 Flow Junctions**
  - **90 Heat sinks**

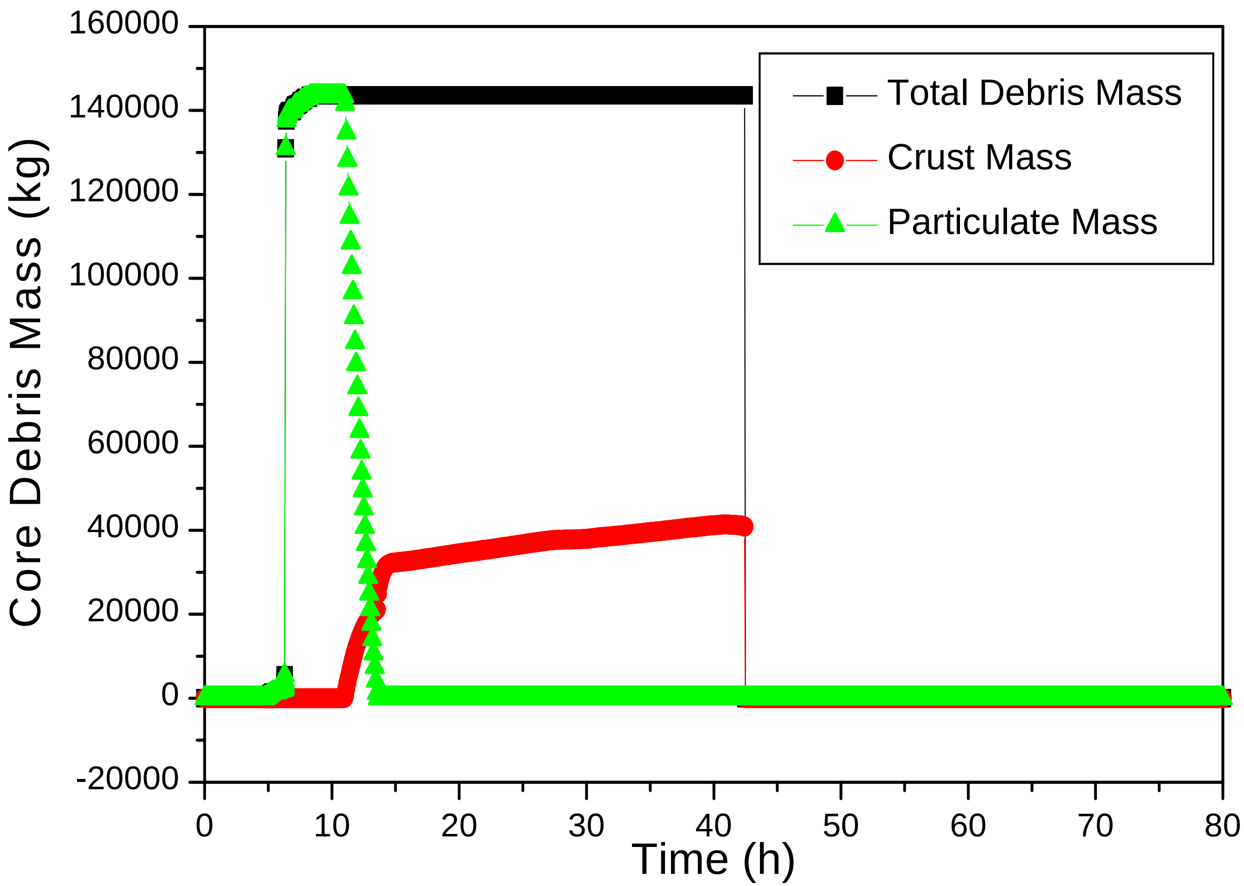
Generic CANDU 6 SBO Analysis Assumptions

- AC power and all onsite standby/emergency power unavailable
- Reactor shutdown after accident initiation
- Moderator-, Shield-, Shutdown cooling unavailable
- Main and Auxiliary Feed water unavailable
- ECCS (high, medium and low pressure) unavailable
- Dousing and Crash cool-down not credited
- LACS not available
- No Operator Interventions are credited
- Failure criteria used to fail certain components/systems
- **No make-up to the Reactor Vault**

UO2 Mass/Loop (Generic CANDU 6 SBO)



CANDU 6 Calandria Vessel Behavior



MAAP4-CANDU Analysis: Point Lepreau Plant (CANDU 6) Refurbishment

- **AECL performed consequence analysis using MAAP4-CANDU to support refurbishment activities of Point Lepreau Station**
- **The following five SCD accident scenarios were selected:**
  - **SBO with loss of all cooling systems; in some cases moderator drain through failed channel bellows**
  - **SLOCA with LOECC, Loss of moderator cooling and loss of other safety-related systems**
  - **SFB (Stagnation Feeder Break) LOCA with LOECC, loss of moderator cooling and moderator drain**
  - **SGTR with LOECC and loss of moderator cooling**
  - **SSA (Shutdown State Accident): IE: Leak from bearing seal of shutdown cooling system pumps and simultaneous loss of shutdown cooling system. PHTS drains to reactor header level, combined with LOECC, loss of moderator cooling, shield cooling and other safety-related systems**

MAAP4-CANDU Analysis: Point Lepreau Plant (CANDU 6) Refurbishment

- **Reference case assumed no operator interventions and credited only limited number of safety-related systems**
- **Sensitivity cases were performed assuming availability of certain systems to assess their effects on accident progression**
- **More than 50 cases were analyzed; timing of major events and fission product releases to the environment were obtained**
- **Results of representative sequences with highest frequencies shown here**

MAAP4-CANDU Analysis Results: PL Refurbishment Representative Sequences

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	not applicable
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%

MAAP4-CANDU Analysis Results: Point Lepreau Refurbishment

- Most efficient system to delay core disassembly: Low Pressure ECC, Steam generator auxiliary feed water
- Most efficient system to delay containment failure: LACs and Low Pressure ECC
- Most efficient system to prevent calandria vessel failure: shield cooling

MAAP4-CANDU Analysis: Point Lepreau Refurbishment SAM Measures

- **Two new systems being installed for SAM**
  - **Systems to add water with a flow rate of ~ 3kg/s from an external source to reactor vault 24 h after accident initiation when no other systems to prevent containment failure available. SAM measure initiated by operator on low water level in reactor vault.**
  - **Use a filtered containment venting system based on high containment pressure set points when no other systems to prevent containment failure was available, provided water make up was added to the reactor vault after 24 h from accident initiation to prevent calandria vessel failure.**

Summary

- The CANDU core damage progression is slow
- MAAP4-CANDU has the necessary models for severe core damage accident analysis for a CANDU plant
- MAAP4-CANDU assisted in level 2 PSA studies and in developing SAM measures

Availability of various systems credited

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

Time Window for Steam Generator Secondary Side Reflooding to Mitigate Large Early Release Following SBO-Induced SGTR Accidents

Y. Liao, S. Guentay

Paul Scherrer Institute, Villigen PSI, Switzerland

Outline

- Background
  - Introduction to SBO induced SGTR accidents
  - Aerosol fission product retention on SG secondary side
- Thermal-hydraulic response
  - SBO transient prior to induced tube failure
  - SGTR transient post tube failure
- Fission product release mitigation
- Summary

Background

- Characteristics of SGTR severe accidents
  - Consisting of spontaneous and temperature or pressure induced SGTR
  - Containment bypass for fission product release
  - SG secondary side is the last barrier to environmental release
- SBO induced SGTR typically poses greater threat than spontaneous SGTR
  - Unavailability of power; engineered systems disabled
  - Faster accident progression
- Objectives for analysis of temperature induced SGTR transient
  - To characterize thermal-hydraulic response and consequent aerosol behavior
  - To estimate the time window for carrying out accident management to avoid large release

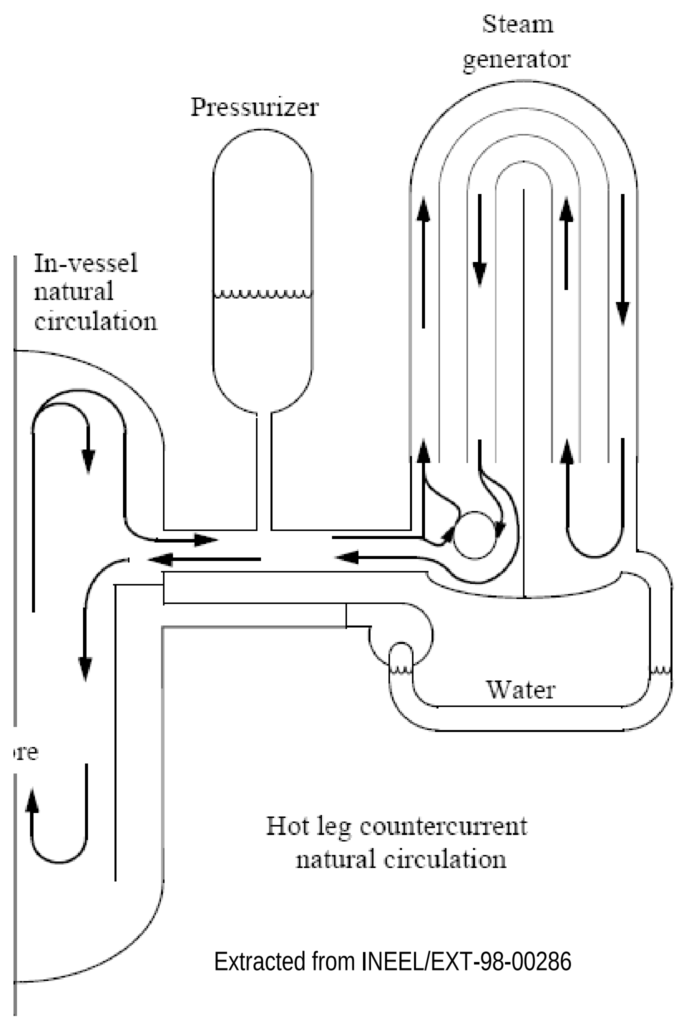
Introduction to SBO induced SGTR accidents

- TMLB' station blackout sequence
  - High pressure primary side, dry secondary side
  - Cold leg loop seal plugged with water

- Assuming SG steam relief valves fail to reclose in TMLB'

- Hot leg counter-current natural circulation (HLNC)
  - Transfer heat to hot leg, surge line and SG tubes
  - SG inlet plenum mixing causes tube temperature to be lower than that of other components

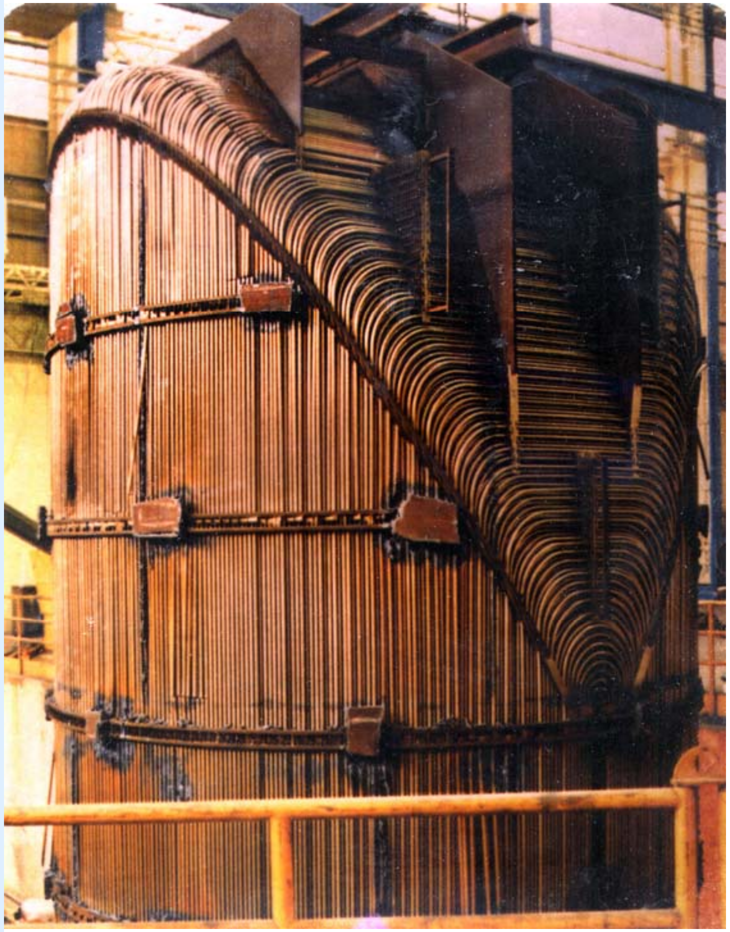
- Probability of temperature induced SGTR
  - Unlikely for intact tube due to inlet plenum mixing
  - Probably for severely degraded tube



Extracted from INEEL/EXT-98-00286

Fission product retention on SG secondary side

- SG secondary side characteristics
  - Thousands of tubes contribute to substantial deposition surfaces
  - A large space for rupture flow to expand and decelerate
  - Numerous internal structures to divert and recirculate flow
  - At temperature lower than that of primary side
- Aerosol retention mechanisms
  - Inertial impaction with cross flow
  - Turbulence deposition
  - Thermal diffusion
- Condensation of FP vapor on structure surface
- Pool scrubbing enhanced by internal structures



SG tube bundle, extracted from Wikipedia

Aerosol fission product retention on SG secondary side

- PSI ARTIST-I experimental program
  - A tube bundle facility with separator and dryer
  - A 7-phase test program, with each phase dealing with a different SG component for aerosol (or droplet) deposition, except for the last phase acting as an integral test
  - Aerosol size was identified as one of the key parameters governing deposition
  - Retention is significant when Dae>1um, and increases sharply with larger Dae
- PSI ARTIST-II experimental program
  - To generate data for conditions not studied but with importance recognized in ARTIST-I, such as: flooded bundle or separator, low water submergence and other aerosol particles, etc.

D_{ae}=1.4um

D_{ae}=3.7um

Aerosol fission product retention on SG secondary side

- Best-estimate aerosol size (AMMD) suggested by Phébus FP experiments (Kissane, 2008, Nuclear Engineering and Design)
  - At 973K (hot leg), close to 2um
  - At 423K (cold leg), around 3um
  - Subject to some uncertainty

- Knowledge of thermal-hydraulic response during the transient is a prerequisite for
  - Partition of fission product between vapor and aerosol phases
  - Characterization of aerosol size and consequent deposition efficiency

Thermal-hydraulic response: prior to induced tube failure

- TMLB' sequence was analyzed using MELCOR for a 4-loop W NPP
  - Initiated by station black-out
  - SG degraded tube induced to leak or rupture by high temperature
  - Fission product release to environment because of SG tube failure
  - Retention of fission product on SG secondary side
  - Calculation terminates when lower head temp. >1273K

- Probability of induced tube failure depends mainly on
  - Characterization of SG tube degradation
  - SG tube temperature level relative to those of surge line and hot leg

Characterization of SG tube degradation

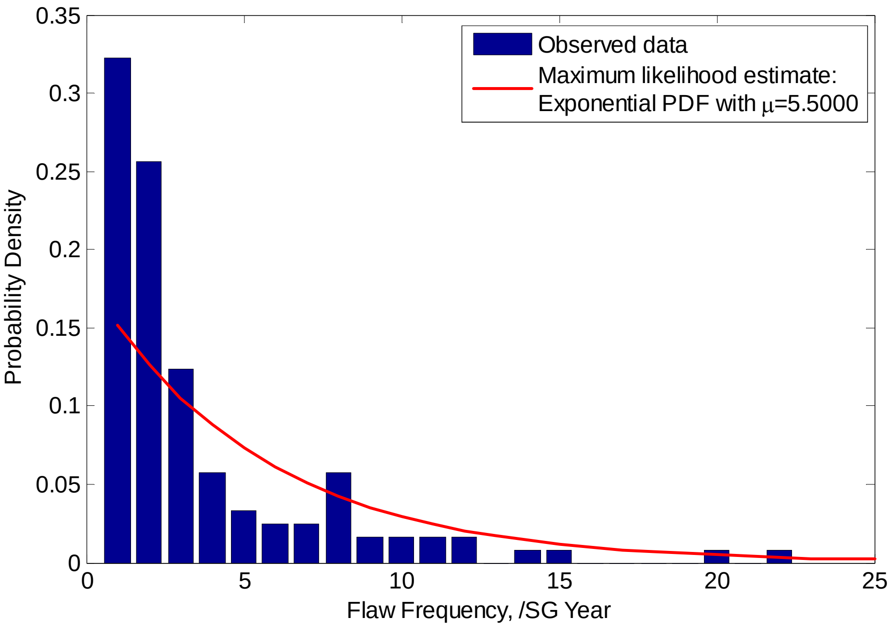
- Mill Annealed Alloy 600 old generation SGs
  - Stress corrosion cracking (SCC) was a major degradation mechanism (NUREG/CR-6365)
- Thermal Treated Alloy 600 and 690 new generation SGs
  - SCC becomes less important due to improved tubing materials, as well as better design and operating practice
  - Foreign object damage and support structure fretting emerge as the major inservice degradation mechanisms (NUREG-1771, -1841)
- Foreign object damage may be a concern of induced tube failure for new generation SGs
  - Foreign object damage mostly occurred at the top of tubesheet, where temperature is highest in the tube bundle during HLNC
  - Foreign object may cause severe tube degradation in an unpredictable way (has caused 5 out of a total of 6 tube leakage events in new generation SGs, NUREG-1771, -1841)

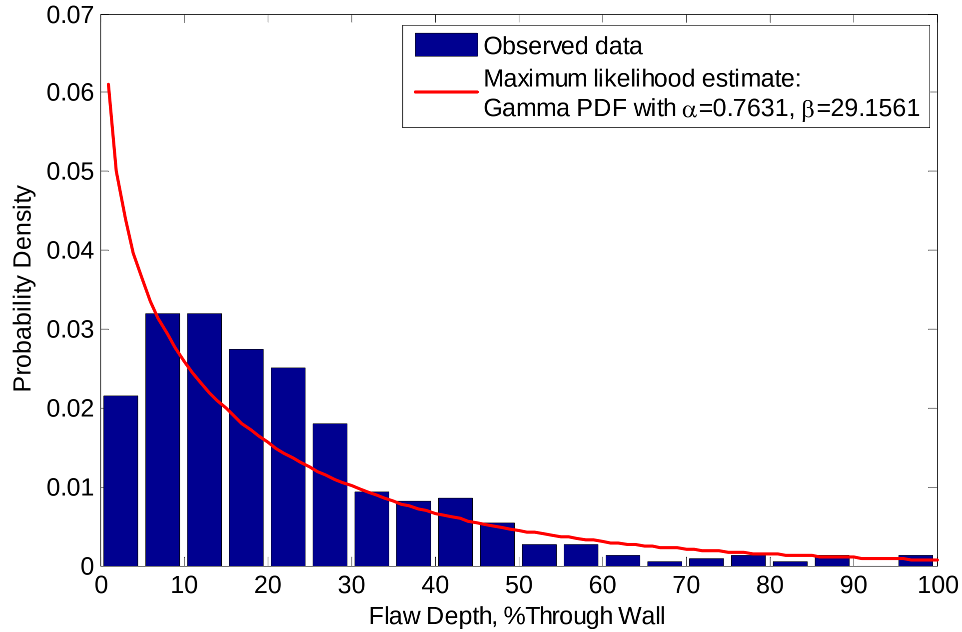
SG tube degradation caused by foreign object damage

- Probability density function (PDF)
  - Derived from SG inservice inspection reports (Liao and Guentay, 2009, NED)

Flaw frequency

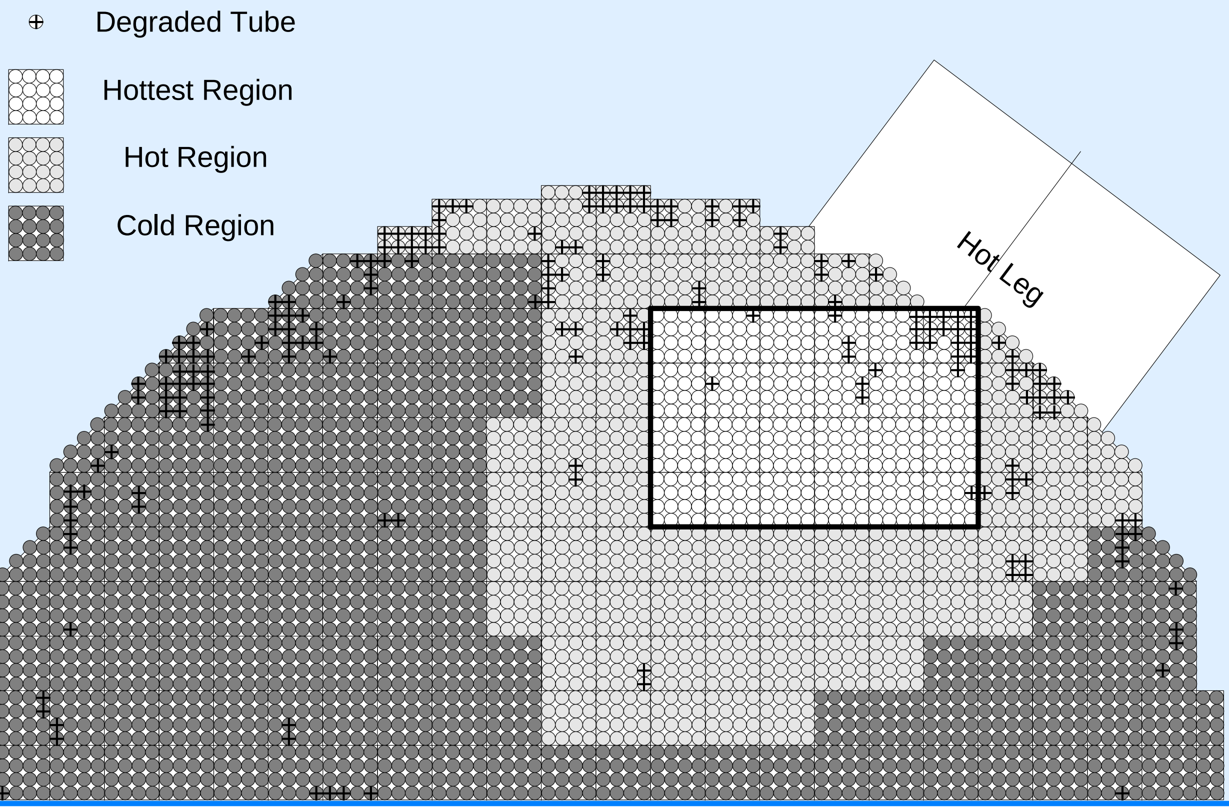
Flaw depth





Degraded tube and temperature distribution at top of tubesheet

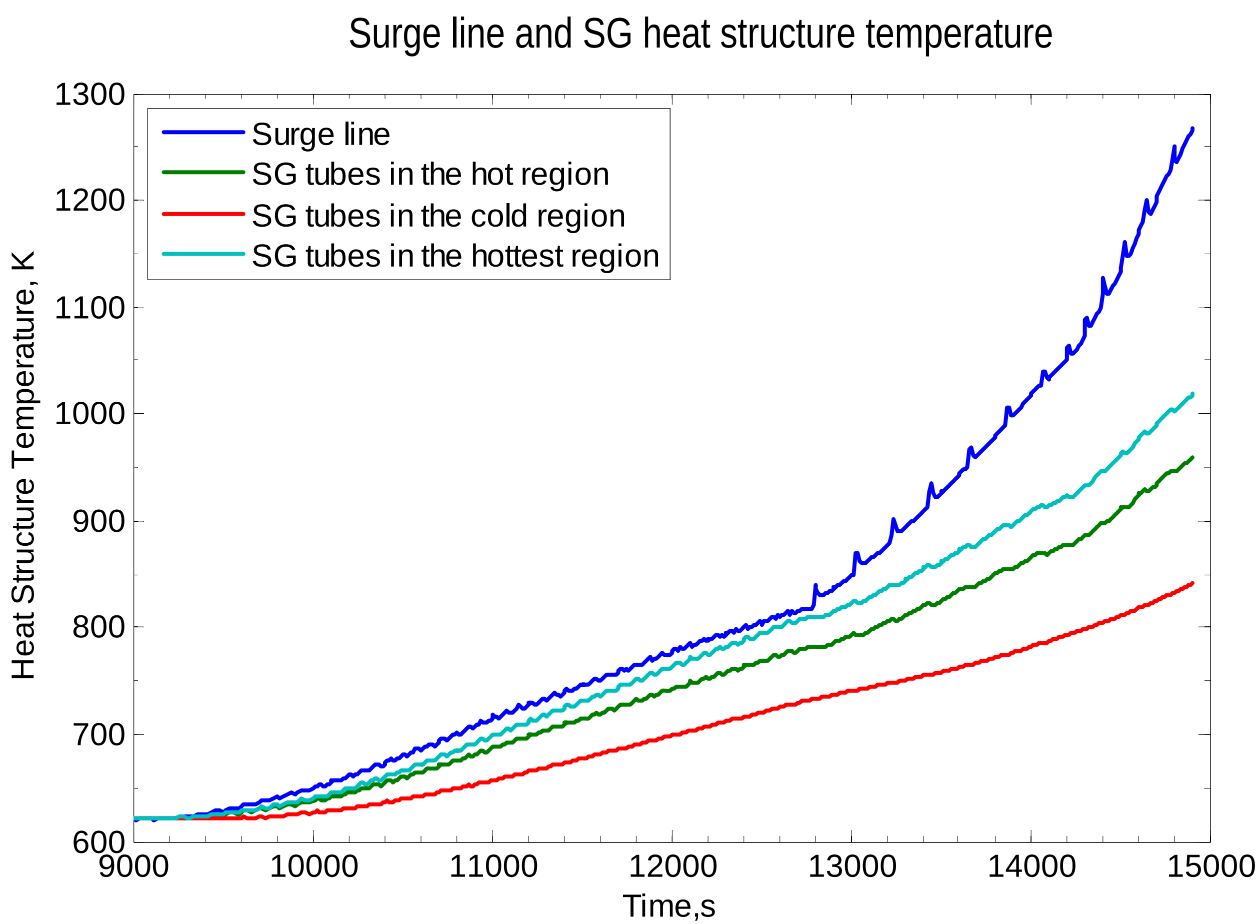
- Degraded tube distribution obtained from SG inservice inspection reports
- Temperature distribution taken from CFD analysis (NUREG-1788)



Thermal response history prior to induced tube failure

- Average temperature predicted by MELCOR

- Hottest region temperature predicted using a peaking factor derived from CFD analysis (NUREG-1788)



Assessment of probability of induced tube failure

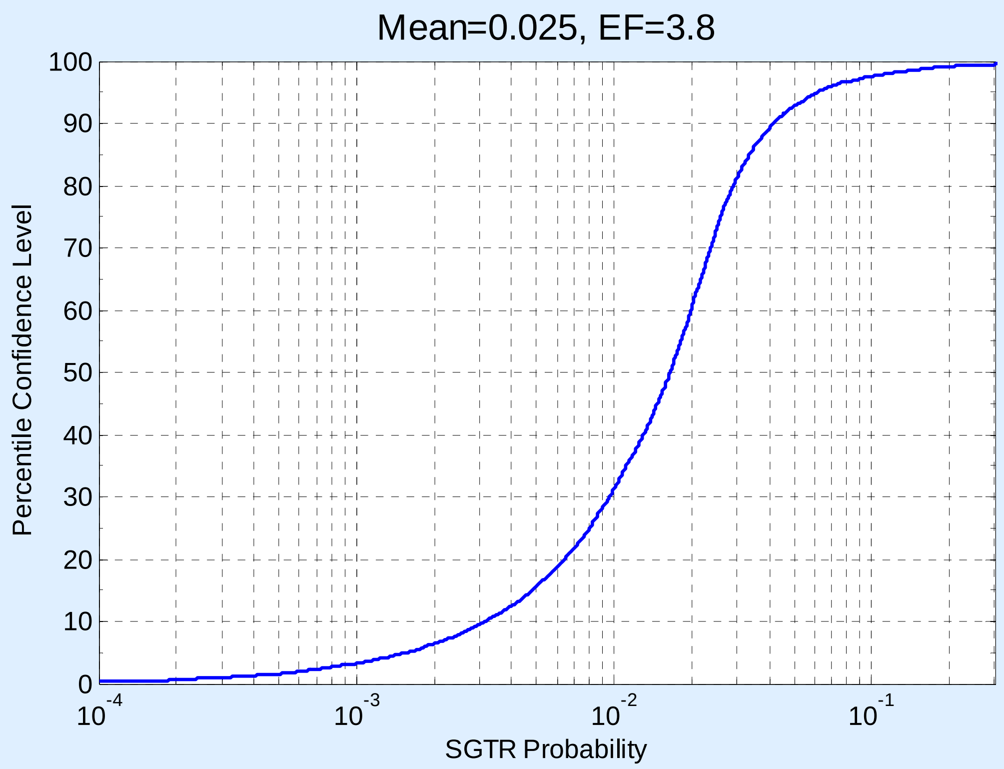
- SG tube may or may not fail before surge line/hot leg
  - Depending on the severity of tube degradation and the relative magnitude of thermal challenge
- Creep rupture model was used to evaluate the failure order among SG tube, surge line and hot leg

$$\int_0^{t_f} \frac{dt}{t_R(T, M_p \sigma)} = 1$$

  - Component failure governed by life fraction rule
  - Employing the respective component's thermal-hydraulic history (T, σ)
  - Depending on SG tube flaw location as well as degradation severity (M_p)
- A Monte Carlo probabilistic approach was adopted to deal with variations in flaw characterization and thermal-hydraulic response uncertainties

Failure probability (tube flaw due to foreign object damage)

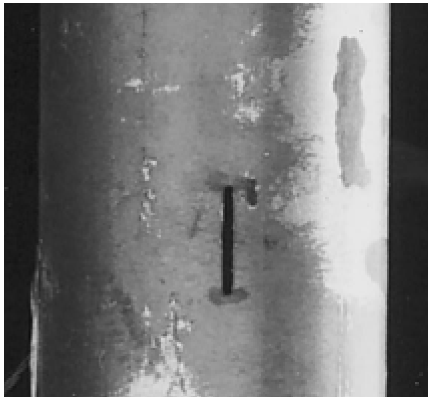
- Cumulative PDF of induced tube failure
  - Derived from 10,000 Monte Carlo simulations (Liao and Guentay, 2009, NED)



Thermal-hydraulic response: post induced tube failure

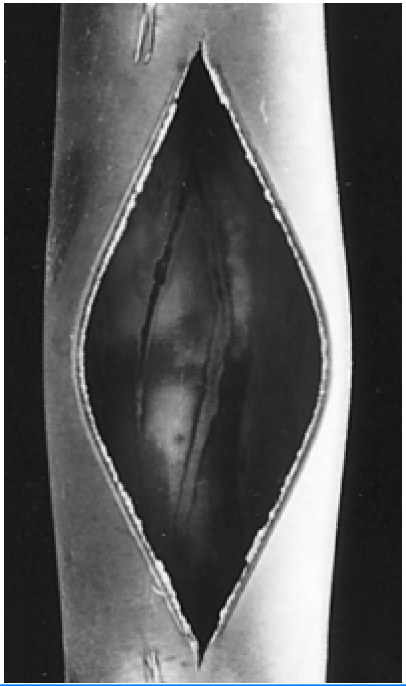
- Leak versus rupture at tube failure
  - Tube with short crack induced to leak
  - Tube with long crack induced to rupture

0.25in



2in

Majumdar, 1999, NED



- Cascading tube failure due to leakage jet impingement initiated from a short crack
  - NUREG-1570 preliminary scoping analysis
    - Cascading failure might occur even for crack length as short as .25in
  - NUREG/CR-6756 reassessment
    - From a .25in long crack, jet impingement damage is insignificant due to small crack opening
    - Cascading failure would be avoided by subsequent depressurization through surge line/hot leg

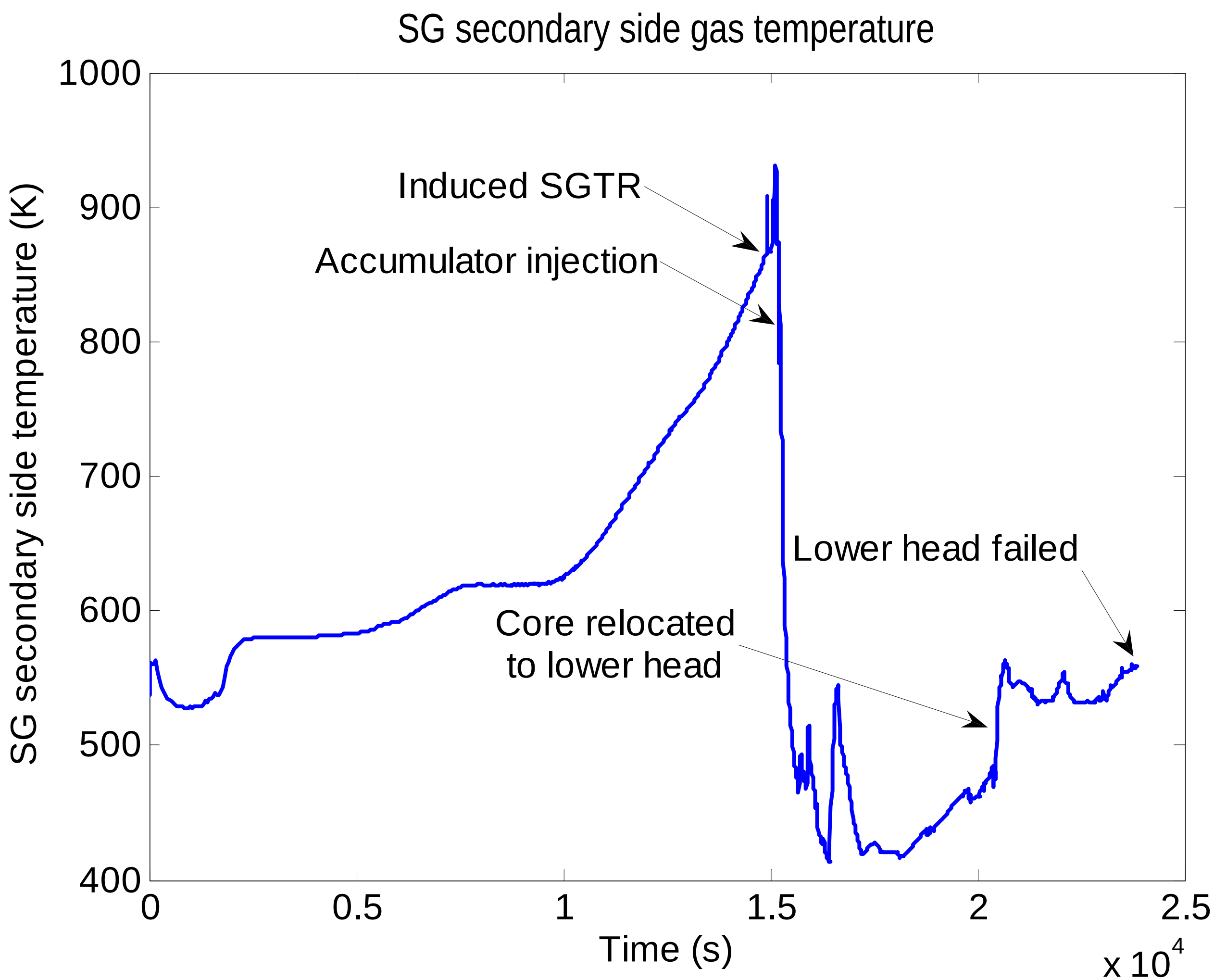
Thermal-hydraulic response: post induced tube failure

- Cascading tube failure due to rupture initiated from a long crack
  - Crack opening rate is higher for longer crack (NUREG/CR-6756),resulting in larger jet impingement velocity
  - Enlarging rupture flow might compromise HLNC and inlet plenum mixing, resulting in higher tube temperature
  - Crack opening rate is higher with elevated temperature (NUREG/CR-6756)
  - Bending and whipping driven by rupture flow momentum may cause tube-to-tube contact and damage (Mihama SGTR event, NUREG/CR-6365)
- Cascading tube failure is considered probable, especially for rupture initiated from a very long crack
- A scoping MELCOR analysis was done assuming cascading tube failure
  - Two rings of tubes surrounding the ruptured tube (25 tubes) damaged in cascading failure (about 6in break)
  - 6in break is sufficient to depressurize the system in a few minutes, making a lot more tubes to be damaged less likely

Thermal-hydraulic response: post induced tube failure

Secondary side gas

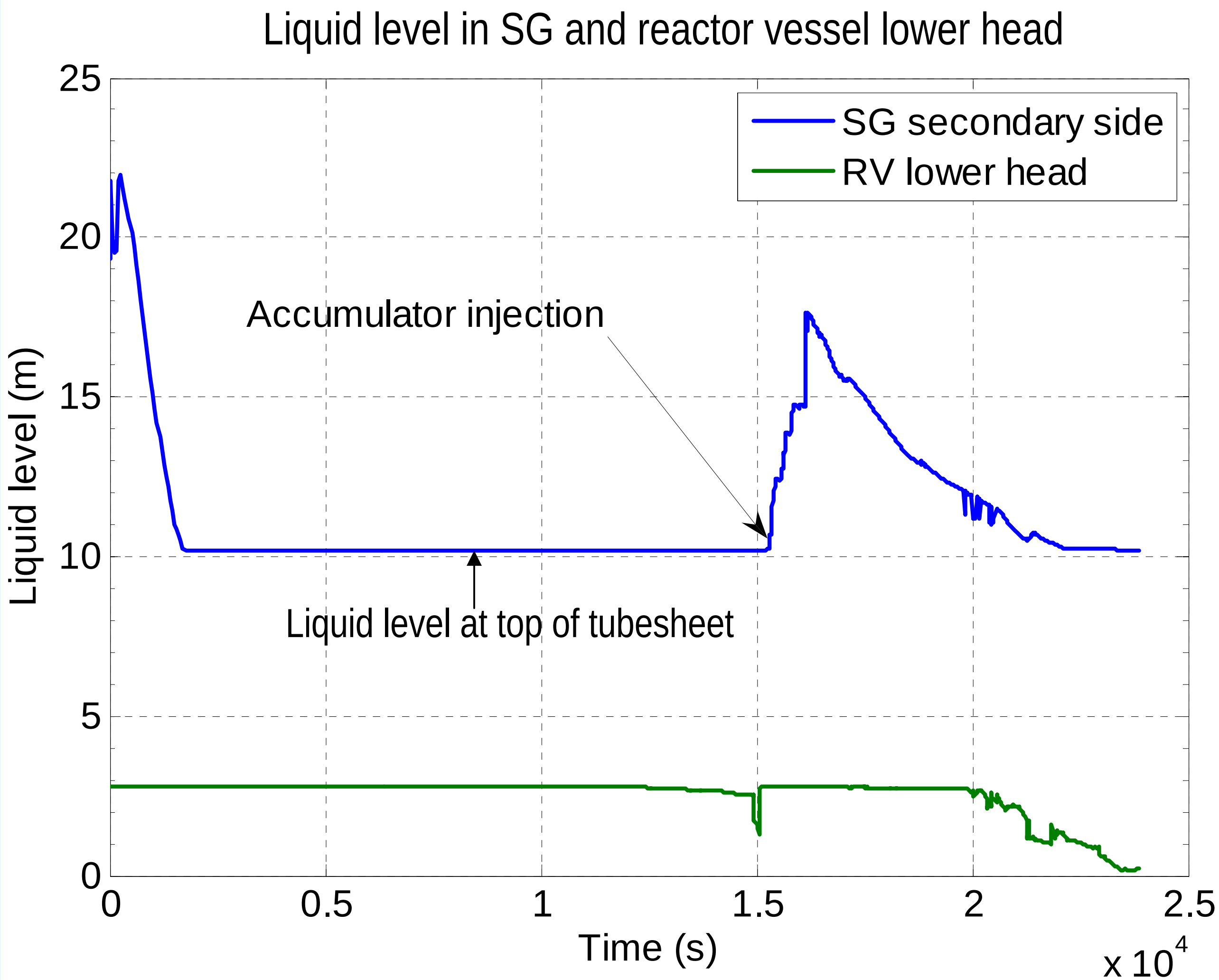
- Cooled down due to injection of subcooled water
- Then heated up again after water above core evaporated



Thermal-hydraulic response: post induced tube failure

Secondary side pool

- No loss of injected water out of system (unlike LOCA)
- Void generated above core as boiling restarts
- Counter-current flow limitation
- Similar to a pool established in pressurizer when PORV opened



Fission product release mitigation

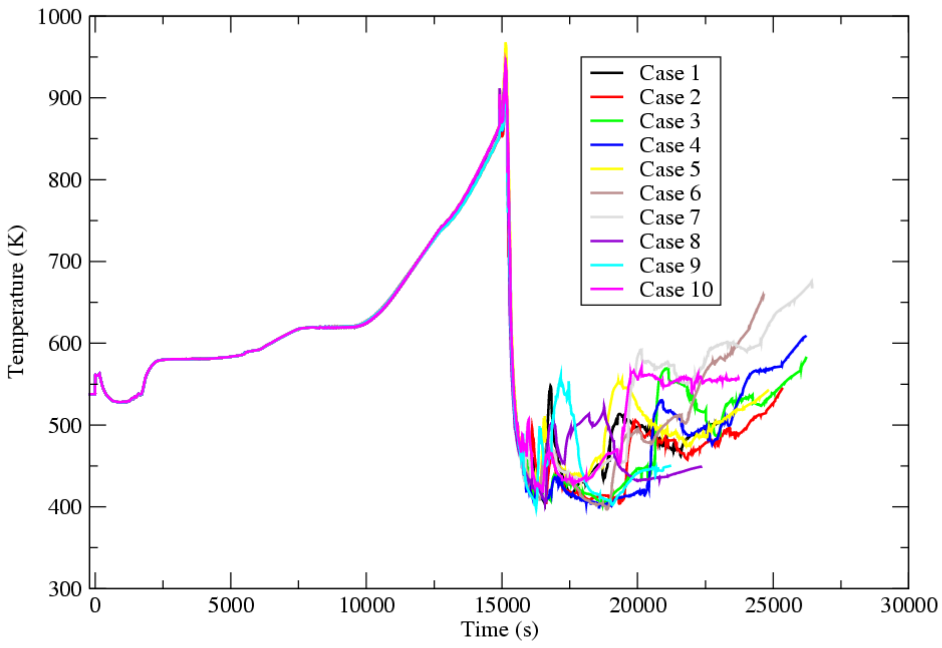
- Mitigation by inherent safety features
  - SG water pool established by accumulator injection
    - Liquid level quickly rises to 7m above TTS, then drops to 1m within 2hr,
    - Most aerosol is removed by pool scrubbing and retention onto tube and structure surfaces of contaminated droplets rising above the pool
  - After SG secondary side dries again
    - Temperature ranges from 500 to 600K
    - Aerosol size about 2.5um
    - Significant aerosol retention by inertial impaction and turbulence deposition is expected (ARTIST experimental evidence)
- MELCOR sensitivity study was done to locate uncertainty of parameters affecting mitigation by inherent safety features

Sensitivity study

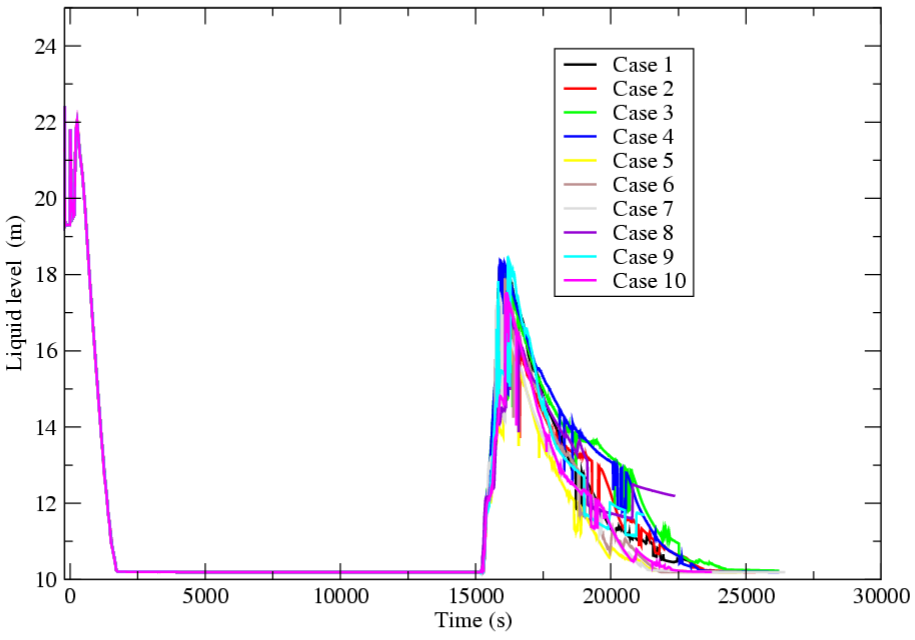
- Sensitivity to core release model
  - CORSOR-Booth model for low or high burn-up fuel
- Sensitivity to bubble rise model with or without RN aerosol scrubbing
- Sensitivity to parameters in core degradation models (SNL LHS technique)
  - Molten Zr breakout temperature
  - Fuel collapse temperature
  - UO2 fraction dissolved in molten Zr
  - Candling melt freezing heat transfer coefficient
  - Core and lower plenum debris diameter
  - Debris porosity
  - Falling debris quench heat transfer coefficient
  - Core thermal radiation view factors

Sensitivity study

SG secondary side gas temperature



SG secondary side pool liquid level

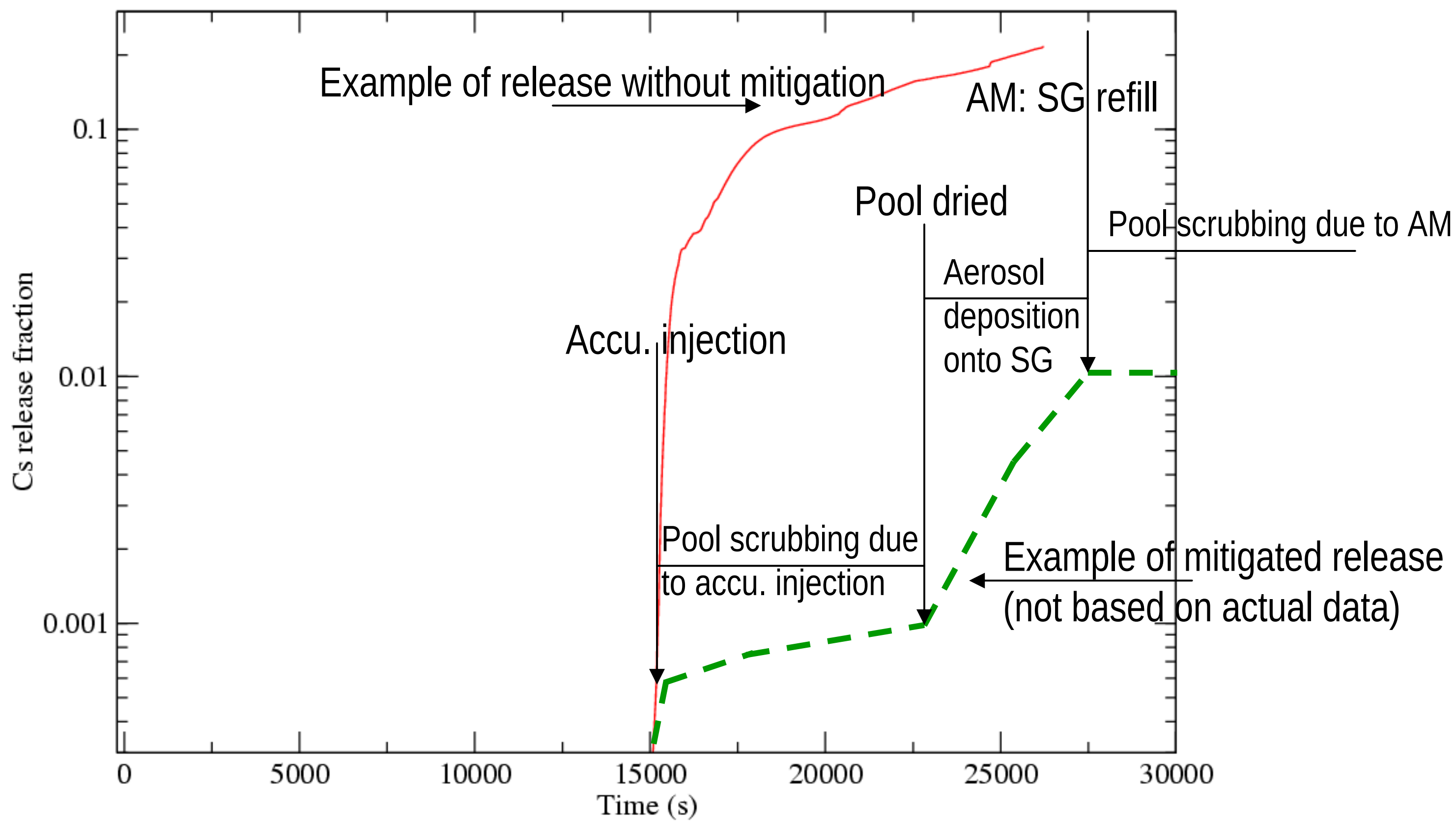


Fission product release mitigation

- Results from MELCOR sensitivity study on parameters governing FP retention on SG secondary side (most probable value)
  - Pool peak liquid level established by accumulator injection: about 7m
  - Duration of pool above top of tubesheet: about 2hr
  - Secondary side gas temperature after SG dries again: about 500K
- Large FP release (1% core inventory) would be postponed by aerosol scrubbing in the pool, with the delay time dependent on pool liquid level and lifetime
- Large release would be further postponed by aerosol deposition on SG tube and structure surfaces, with the delay time dependent on fluid temperature, aerosol size and others
- Delay of large release makes time for accident management execution
  - Injection into SG has the highest priority for accident mitigation (Westinghouse SAMG)
- ARTIST project will generate data for retention in dry and flooded conditions to assess the probability to avoid large early release

An example showing effect of fission product release mitigation

Cs cumulative release fraction to environment



Summary

- A scoping MELCOR analysis was carried out for thermally induced SGTR
- Inherent safety features would postpone large early release by a number of hours, making more time for accident management to take effect to more probably avoid large release
- Future work may integrate MELCOR analysis and ARTIST experimental data for a more detailed assessment of
  - Probability of accident management to avoid large early release
  - Fission product release fraction for various induced SGTR scenarios

On the Effectiveness of CRGT Cooling as a Severe Accident Management Measure for BWRs

Weimin Ma, Chi-Thanh Tran

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Stockholm, Sweden

Outline

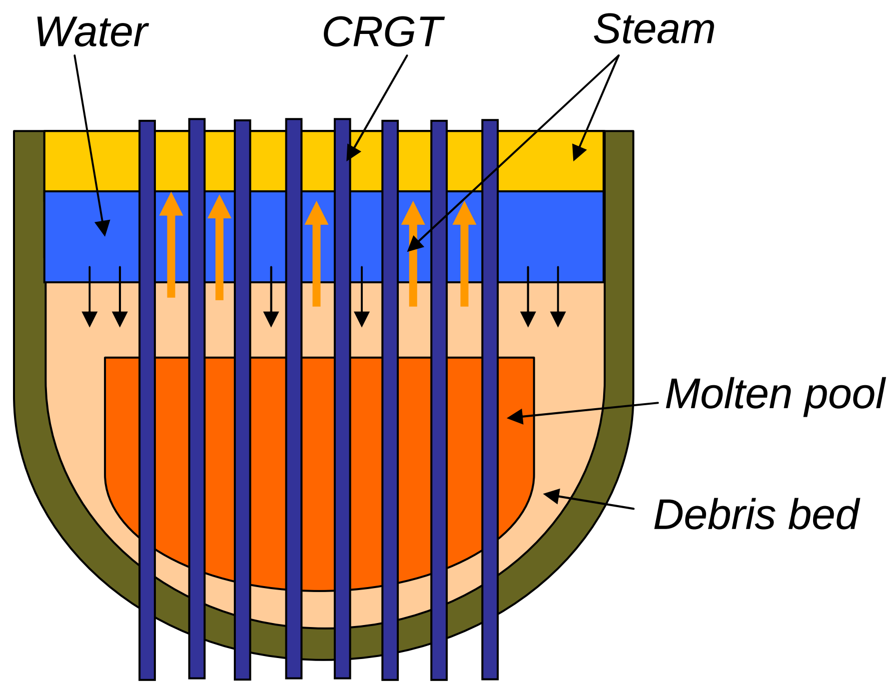
- Motivation
- Objectives
- MELCOR modeling of CRGT cooling
- Preliminary results
- Implication to reactor safety
- Outlook

Motivation (1)

- The in-vessel retention (IVR) has been implemented in a few PWRs, but not applied to any BWR so far.
- 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:
  - √ The modification will be minimal by capitalizing on the existing cooling system;
  - √ The forest of CRGTs provides large area for heat transfer from corium to coolant;
  - √ The flowrate of the CRGT cooling (~10kg/s) is small so that it can be ensured by introducing a battery-driven pump.

Motivation (2)

- CRGT cooling as a SAM measure is under investigation at KTH.
- The challenges are:
  - √ Very high Rayleigh number (10^{15}-10^{17})
  - √ Long transient of severe accident progression
  - √ Complex flows
  - √ Complex 3D geometry
- We need a tool which is
  - √ Sufficiently accurate (i.e. preserving the key physics)
  - √ Computationally affordable (effective) for melt pool heat transfer simulation
  - √ Capable of long time transient simulation, and
  - √ Simulation of 3D complex geometry of BWR lower plenum
  - √ Applicable for thermal fluid-structure interaction problems



Homogeneous melt pool formation in a BWR

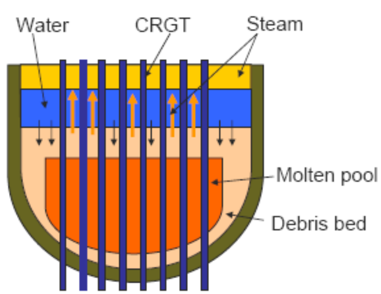
Motivation (3)

- The efficacy of CRGT cooling has been addressed by ECM/PECM approach, in term of the melt pool behavior in the lower head, with assumptions of melt conditions.

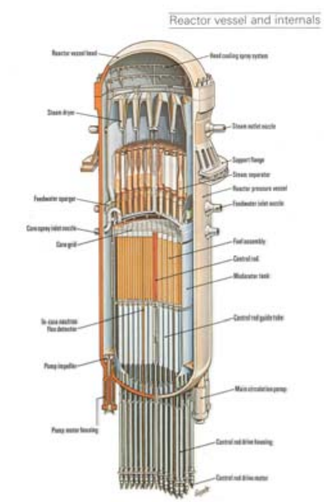
  – C.T. Tran, The Effective Convectivity Model for simulation and analysis of melt pool heat transfer in a light water reactor pressure vessel lower head, Doctoral Thesis, Royal Institute of Technology, Stockholm, Sweden 2009.

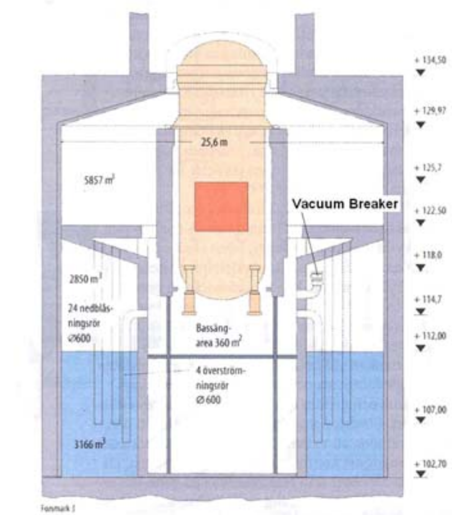
- To provide the realistic melt conditions and account for the influence of CRGT cooling on progress of whole SA scenario, especially of core degradation, it is necessary to perform analysis at system level.

For a 2500MWth BWR
Melt pool<0.7, coolable.



ECM/PECM



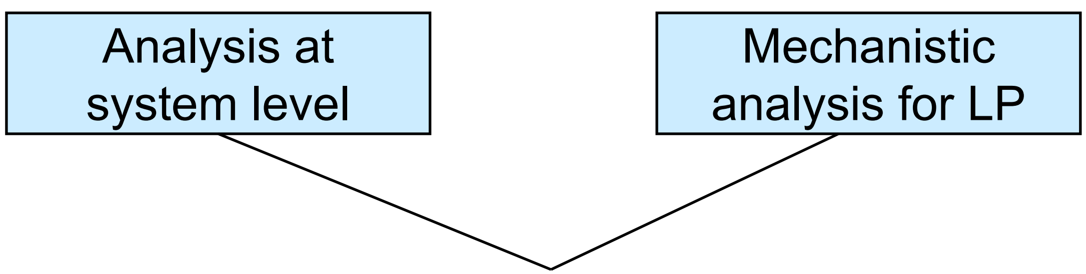


Objectives

- To examine the capabilities of MELCOR code for simulation of CRGT cooling.

- To investigate the efficacy of CRGT cooing, with modeling of whole reactor system and calculation of an entire SA scenario.

- To develop a methodology to perform integral safety analysis by using system code (MELCOR) and CFD code (PECM).

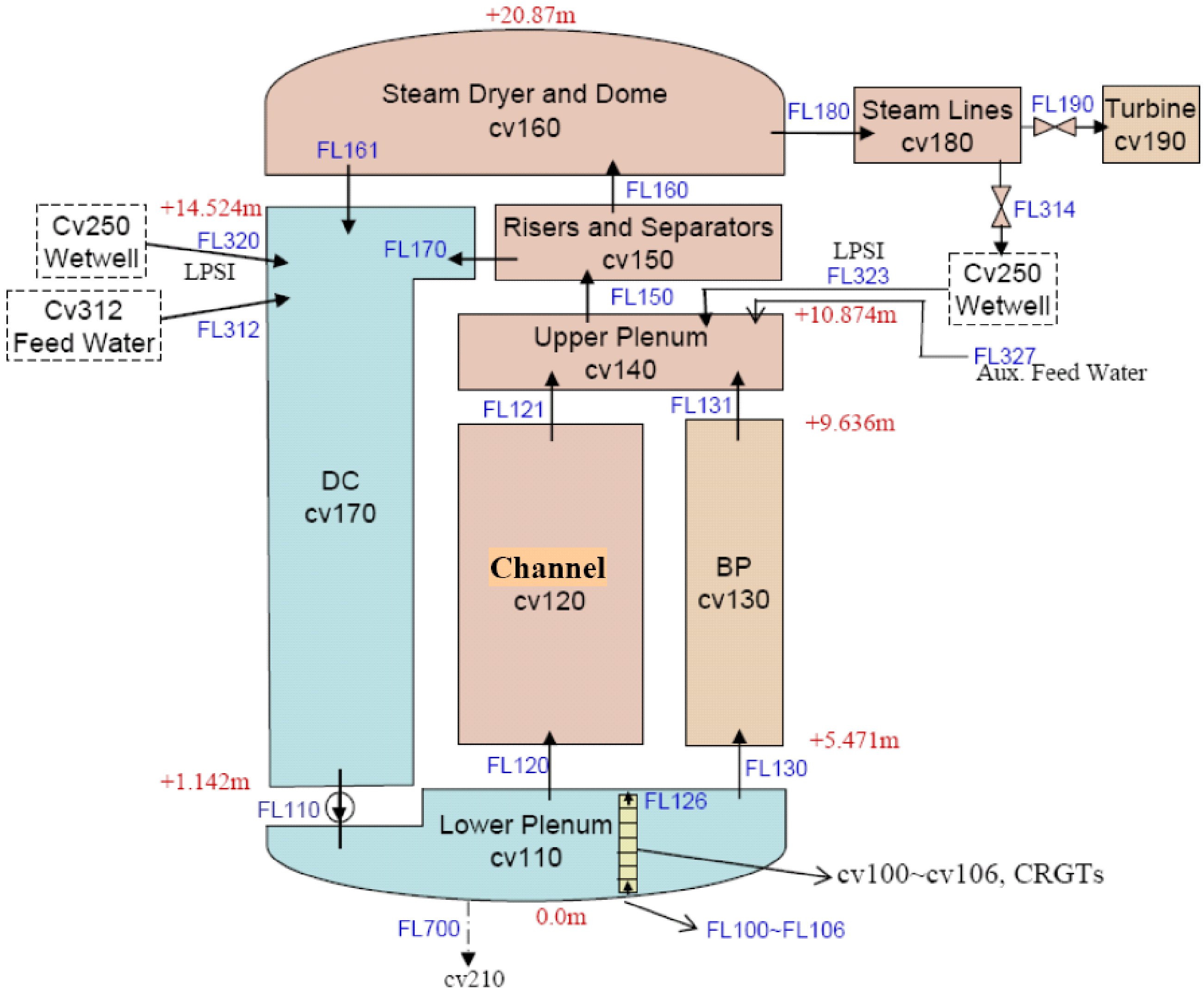


Increase the reliability of the assessment

Reference reactor

No.	Parameter	Value
1	Thermal power, MWt	3900
2	Operating pressure in vessel, bar	70
3	Reactor vessel outside height, m	21
4	Internal vessel diameter, m	6.4
5	Vessel wall thickness, m	0.198
6	Effective core height, m	3.68
7	Number of CRGTs	169
8	Nominal flow rate per CRGT, g/s	62.5
9	Nominal flow rate in entire CRGT, kg/s	10.5
10	Initial UO_2 mass, kg	146000
11	Initial *Zr* mass, kg	52680
12	Initial steel mass, kg	100400

Nodalization



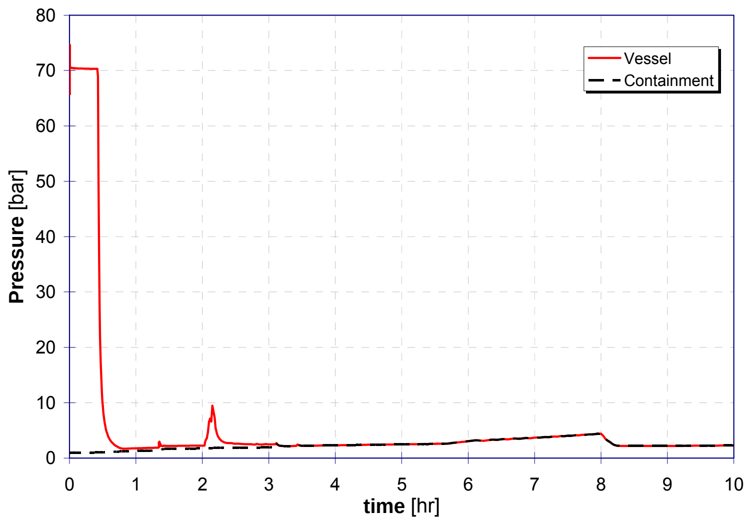
Calculation matrix

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***

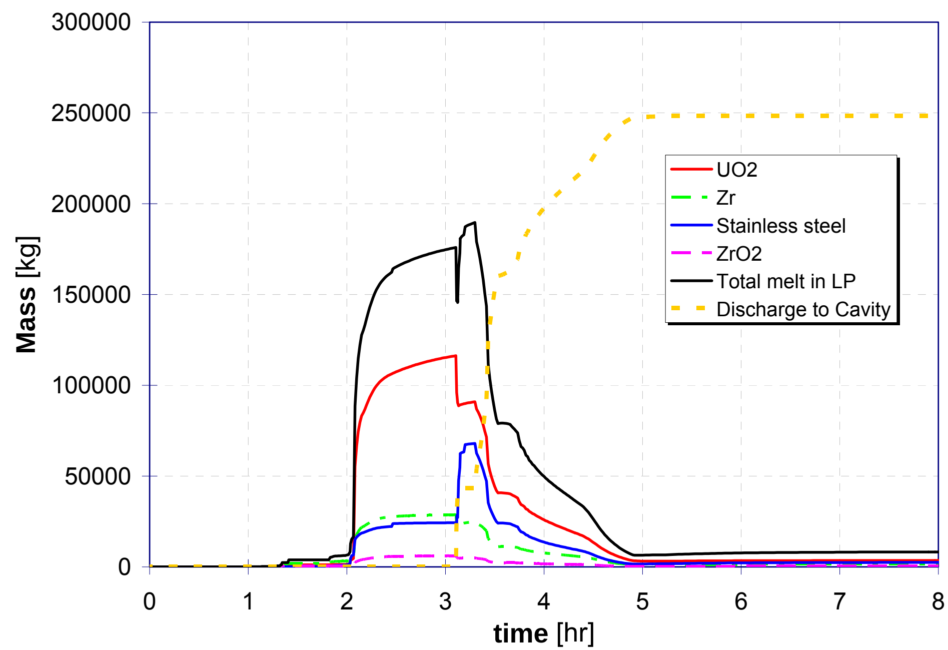
Results (1)

- **Scenario-1**: *SBO without CRGT cooling*

  - √ Station blackout.
  - √ No activation of other emergency injections.
  - √ Activation of ADS during the entire SA sequence.
  - √ No flow in CRGTs.



Pressure evolution

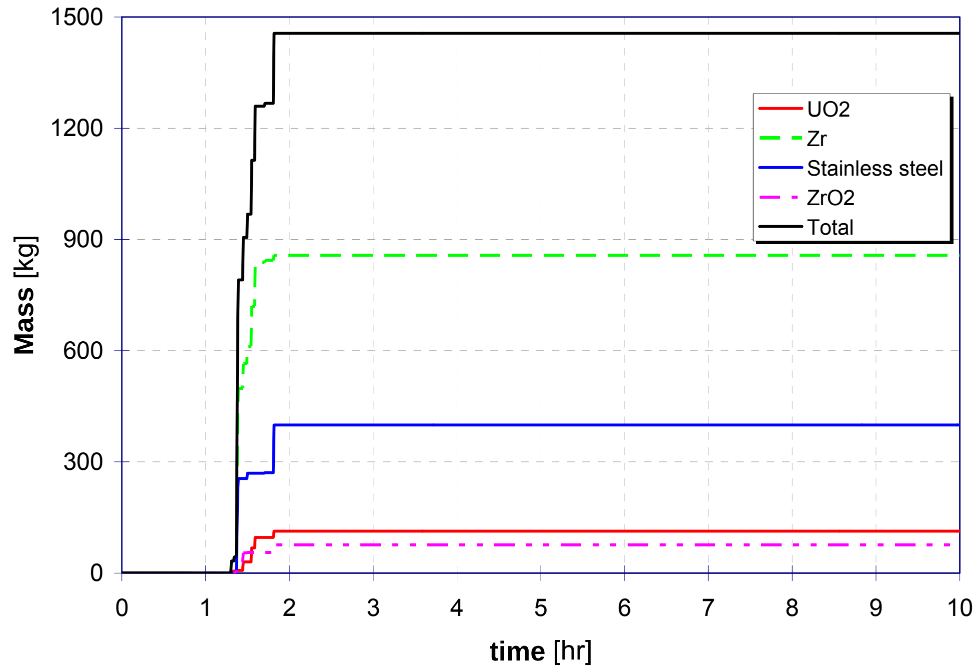


Melt mass in the LP and Melt ejection

Results (2)

- ***Scenario-2**: SBO + CRGT cooling at **10.5kg/s from time 0***

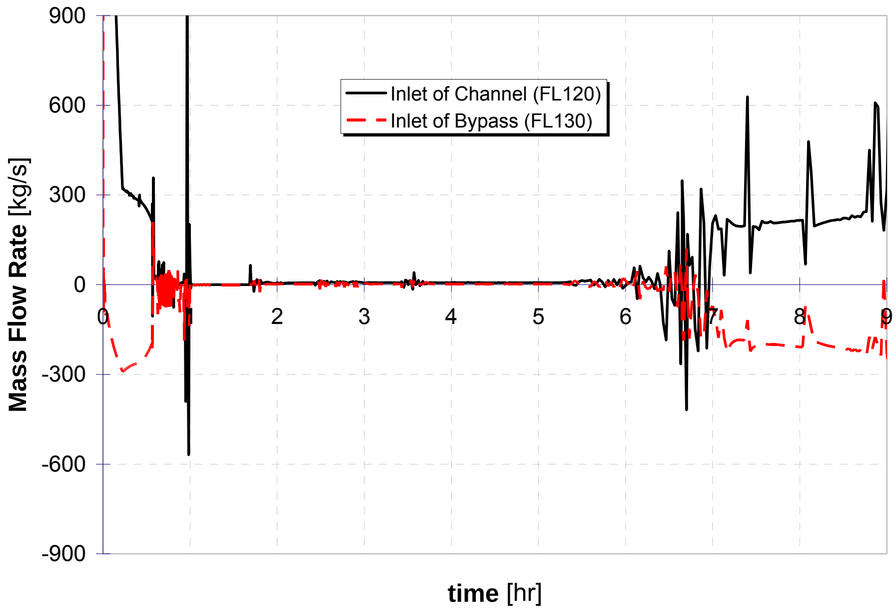
  - √ Water injection through CRGT cooling system.
  - √ Flow rate in CRGTs at the nominal rate 10.5 kg/s.
  - √ Starting time of CRGT cooling: t = 0 h.
  - √ Water temperature = 20ºC.

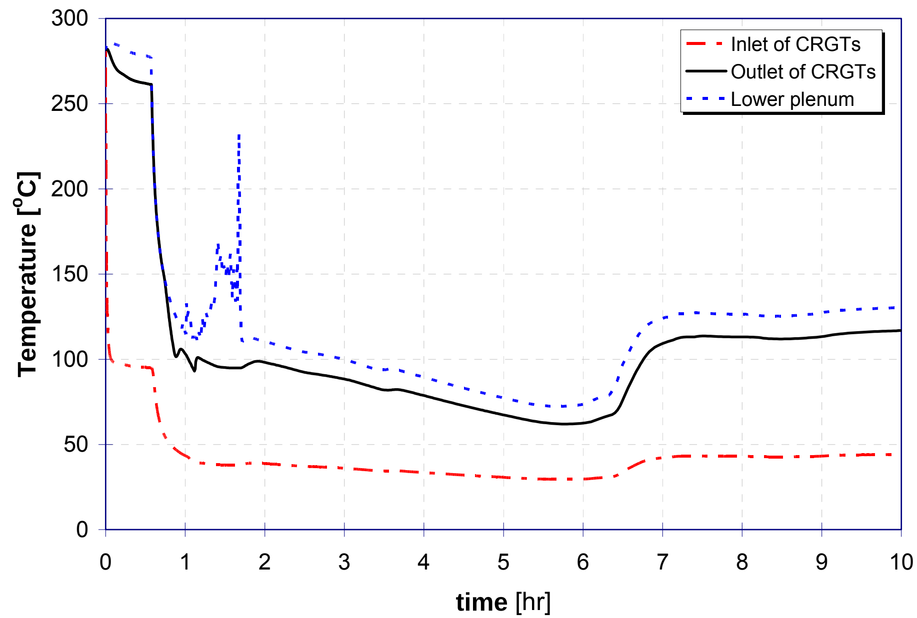


Melt mass in the LP

Results (3)

- ***Scenario-2**: SBO + CRGT cooling at **10.5kg/s from time 0*** (contd.)



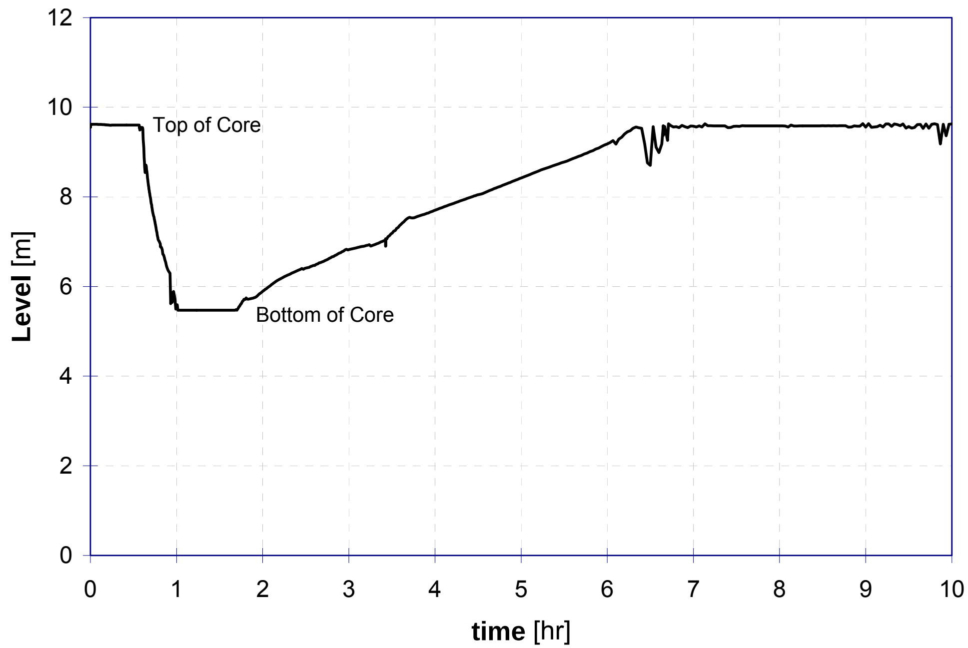


Flows between the core and LP

Temperature in CRGTs and LP

Results (4)

- ***Scenario-2**: SBO + CRGT cooling at **10.5kg/s from time 0*** (contd.)

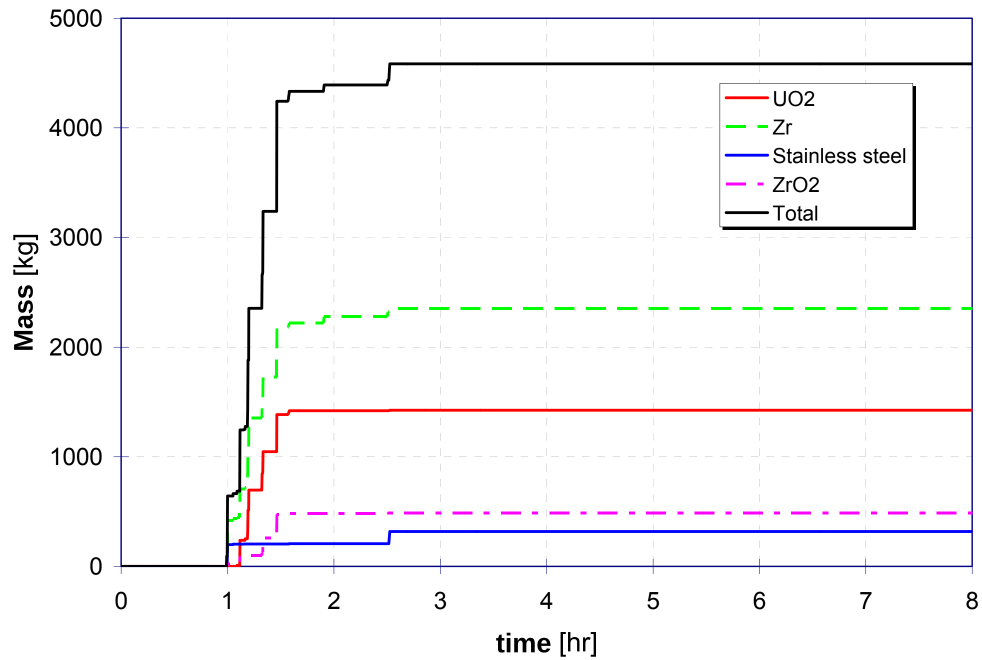


Water level in the core

Results (5)

➢ ***Scenario-3**: SBO + CRGT cooling at **10.5kg/s from time 1 hr***

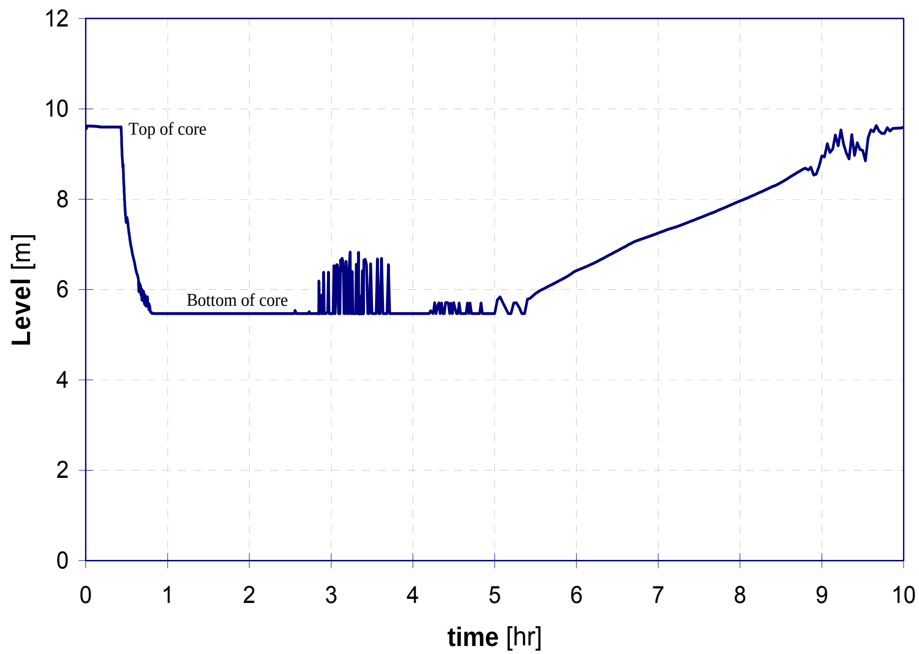
- √ The starting time of CRGT cooling: t = 1 h.
- √ Flow rate in CRGTs at the nominal rate is 10.5 kg/s.



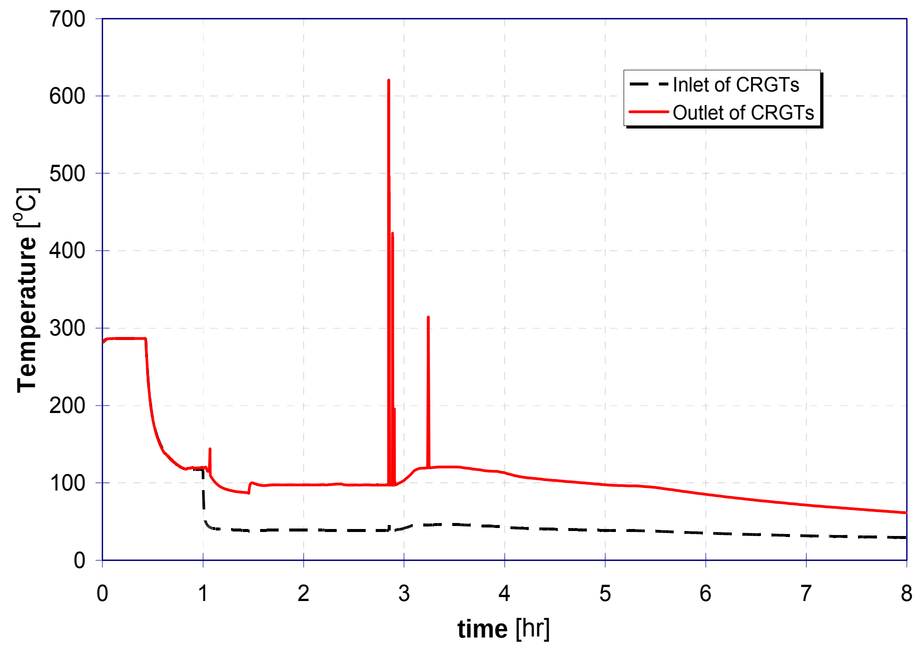
Melt in the LP

Results (6)

- ***Scenario-3**: SBO + CRGT cooling at **10.5kg/s from time 1 hr*** (contd.)



Water level in the core



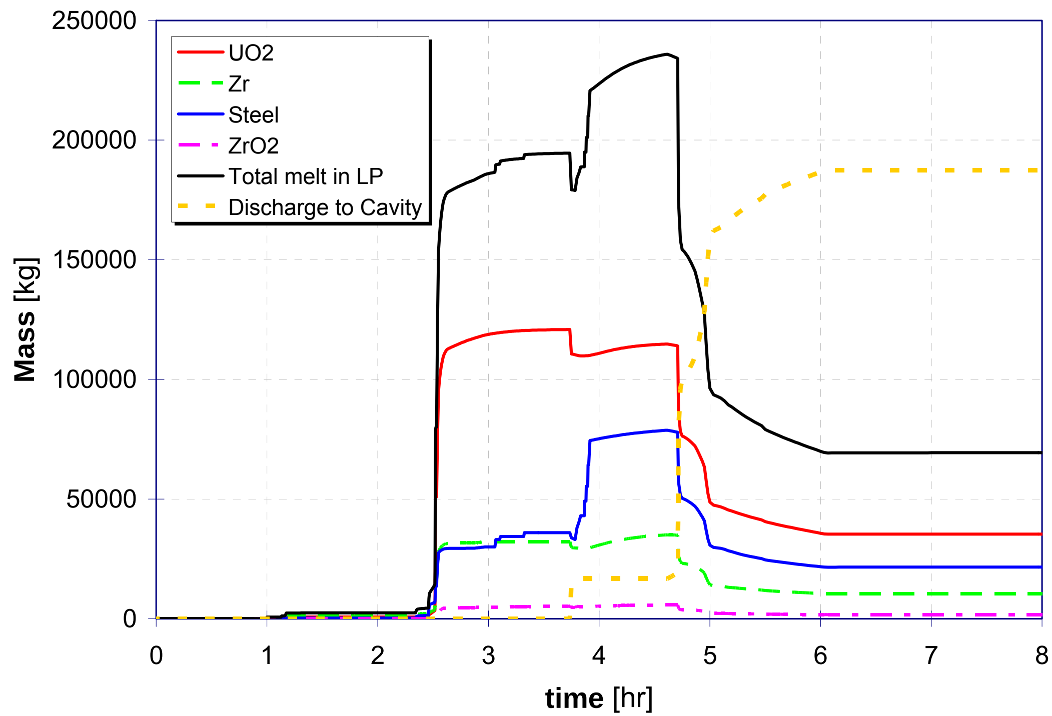
Temperature in CRGTs

Results (7)

- ***Scenario-4**: SBO + CRGT cooling at **10.5kg/s from time 2 hrs***

  √ The starting time of CRGT cooling: $t = 2$ h.

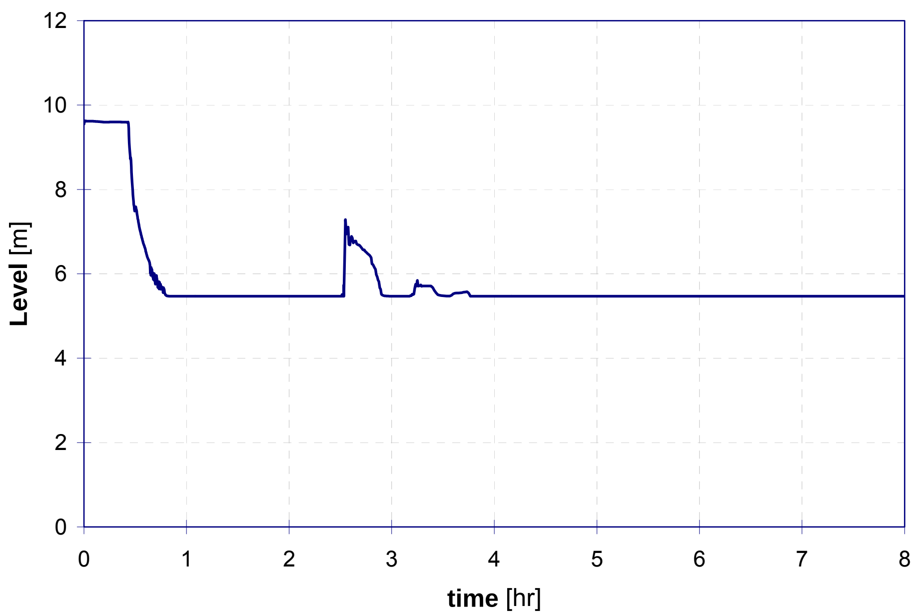
  √ Flow rate in CRGTs at the nominal rate is 10.5 kg/s.



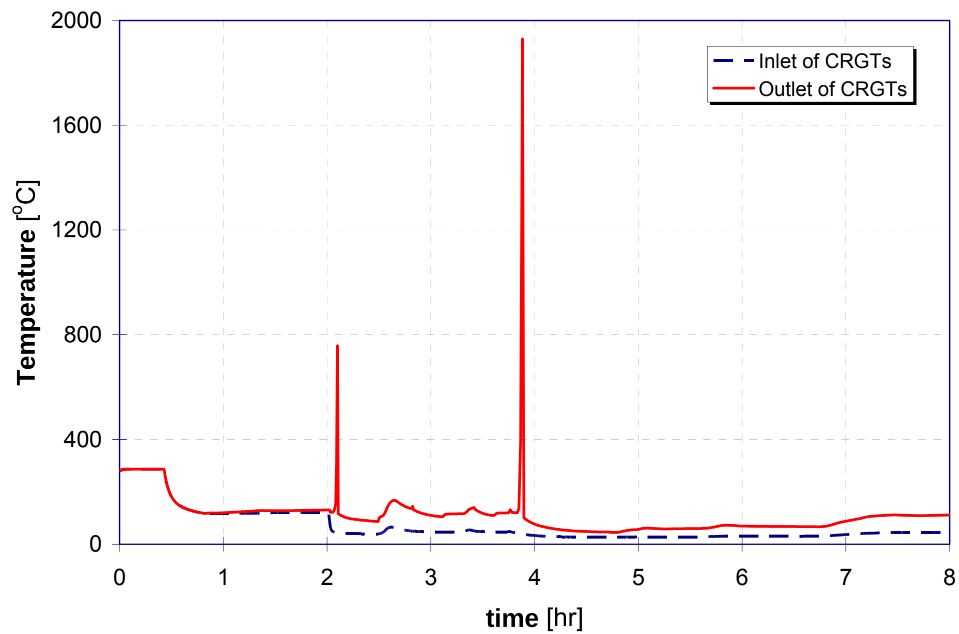
Melt mass in the LP and Melt ejection

Results (8)

- ***Scenario-4**: SBO + CRGT cooling at **10.5kg/s from time 2 hrs** *(contd.)



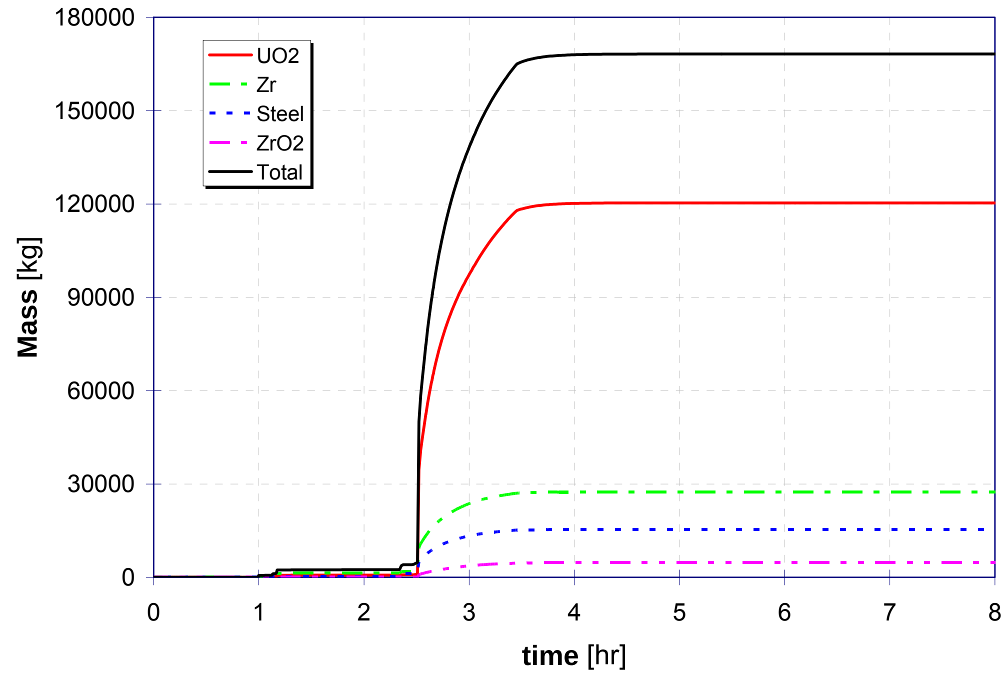
Water level in the core



Temperature in CRGTs

Results (9)

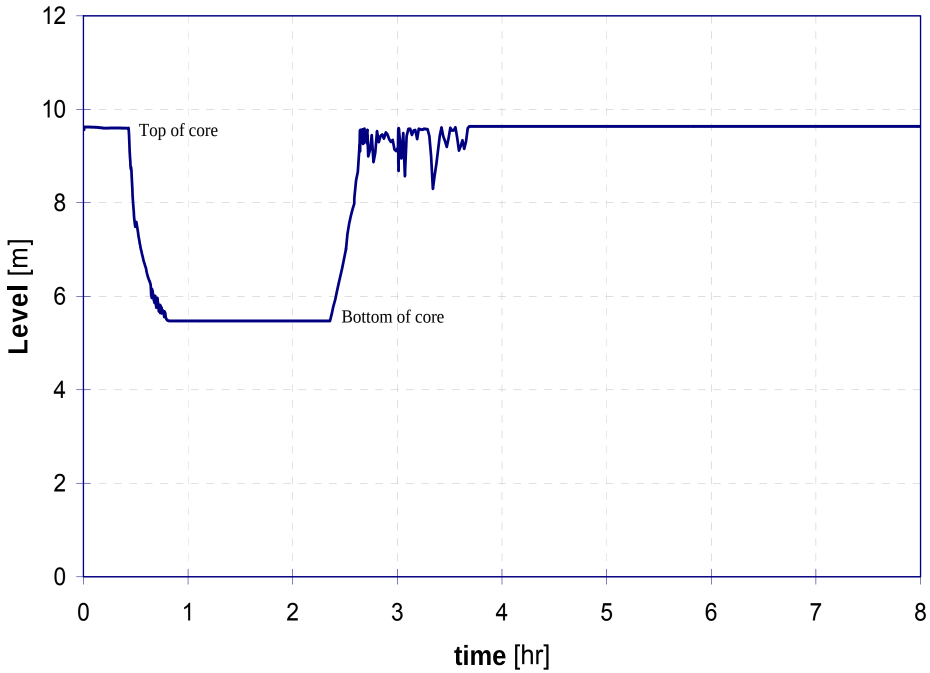
- ***Scenario-5**: SBO + CRGT cooling at **42kg/s from time 2 hrs***
  - √ The starting time of CRGT cooling: t = 2h.
  - √ Flow rate in CRGTs is 4×10.5 =42 kg/s.



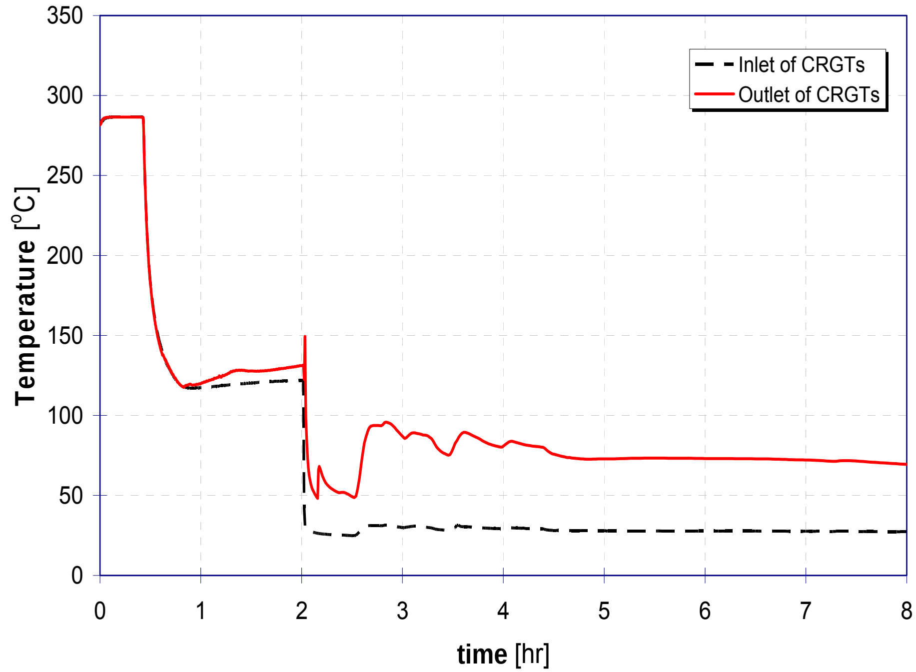
Melt mass in the LP

Results (10)

➢ ***Scenario-5**: SBO + CRGT cooling at **42kg/s from time 2 hrs*** (contd.)



Water level in the core



Temperature in CRGTs

Implication to reactor safety

- The nominal flowrate (~10kg/s) of CRGT cooling is sufficient for in-vessel coolability, if the water injection is activated no later than 1 hours after scram.
- If water injection through CRGTs is activated after 2 hours, much higher flowrate (~40kg/s) is needed to contain the melt in the vessel.
- Melt discharge can be reduced substantially by CRGT cooling even if the water injection is activated at flowrate less than 40kg/s after 2 hours.

Outlook

- Sensitivity analysis with the selections of modeling parameters and timing.
- Methodology development to assess the effectiveness of CRGT cooling as a SAM measure, by lumped-parameter analysis (MELCOR) at system level and mechanistic analysis (ECM/PECM) at detailed level.
- The dual approach leverages on the strength of the two methods (MELCOR and /PECM), and therefore increases the reliability of the assessment.

Ex-Vessel Corium Management for the VVER-1000 Reactor

Bohumír Kujal

Nuclear Research Institute Rez plc, Czech Republic

OECD/NEA Workshop on

Implementation of Severe Accident Management Measures

Bottstein,Switzerland, Oct. 26-28,2009

Contents

- **Background**
- **Ex-vessel corium management strategies**
- **Assessment of the strategies**
- **Melting through of containment basement**
- **Summary**
- **Ultimate corium management strategy**

Background

<u>VVER-1000 containment</u>: modern PWR design

- design pressure: 0.49 Mpa
- free volume: 66000 m3
- leak rate: 0.1 % /24 hours

<u>Inconvenient design features</u>:

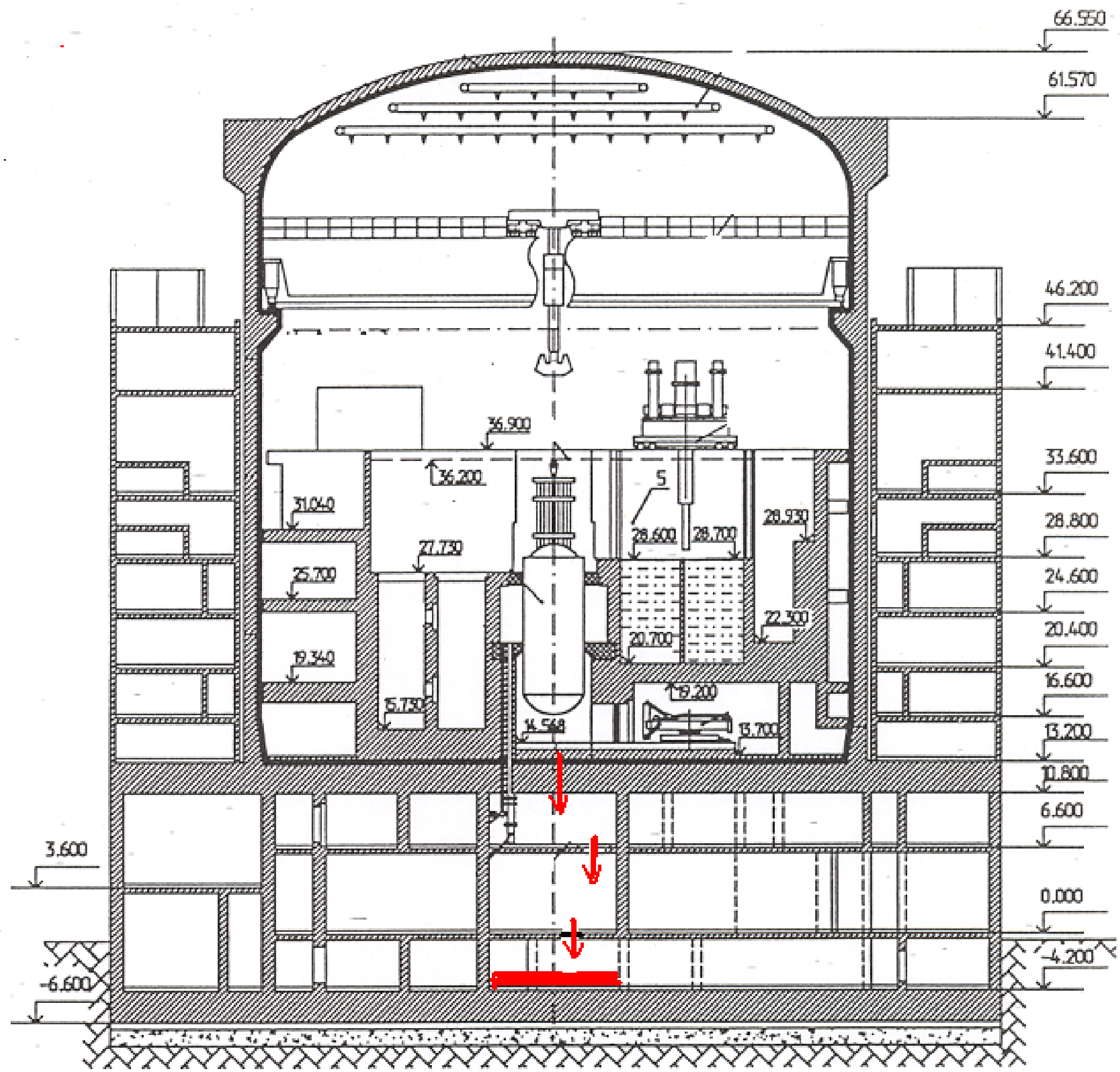
- built on non-hermetic lower part of reactor building
- thickness of containment basement slab: 2.4 m

<u>Real threat:</u>

- melting through of containment basemat slab in a few days
- massive release of FP into non-hermetic rooms and finally into environment

<u>Advantageous design feature</u>:

free room on containment floor for corium spreading out of the reactor cavity (over 100 m2)


66.550
61.570
46.200
41.400
36.900
36.200
33.600
31.040
5
28.930
28.600
28.700
28.800
27.730
25.700
24.600
22.300
20.700
20.400
19.340
19.200
16.600
15.730
14.568
13.700
13.200
10.800
6.600
3.600
0.000
-4.200
-6.600

Strategies

<u>Two strategies proposed</u>:

- **Corium spreading out of the cavity (total corium flooded area was about 100 m2)**
- **Water cooling of melt pool (water is poured from above)**

<u>Corium management: 4 scenarios (strategies) studied:</u>

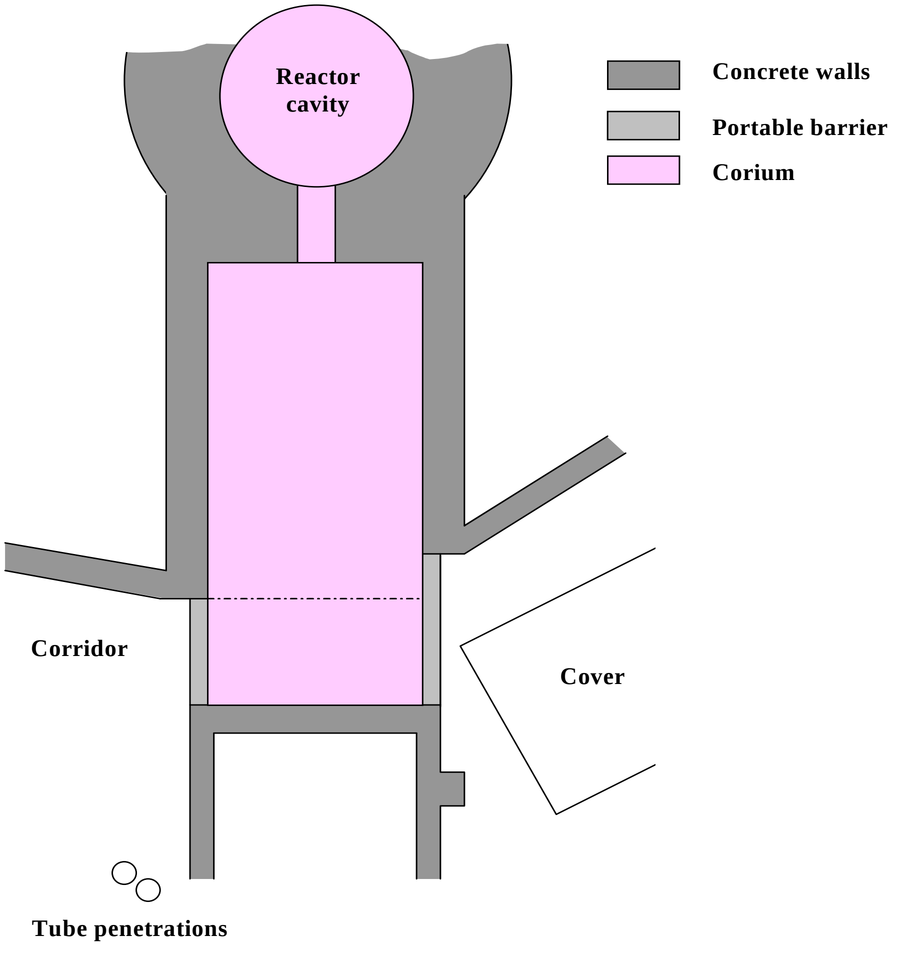
- TB2:reference (basic): no remedial measures

- TB3: corium spreading out of the cavity

- TB4: corium cooling with water

- TB5: corium spreading out of the cavity and water cooling: combination of both of the strategies

Flooded area shape



Assessment of strategies

<u>SA scenario</u>: blackout

<u>Codes used</u>:- CORCON 3.01h (part of MELCOR 1.8.5)

- MEDICIS 1.3.2 (part of ASTEC code)

<u>Common assumptions:</u> - homogeneous melt pool

- siliceous concrete in the cavity

<u>Initial and boundary conditions</u> for MEDICIS and CORCON codes: provided from integral MELCOR 1.8.5 calculation:

- **initial melt pool mass and composition**
- **Initial melt temperature**
- **decay heat in melt pool**
- **thermodynamic state of volume over melt pool**

Assessment of strategies

Calculation: - default input values for CORCON
- recommended input values and models for MEDICIS
- no adjustments of MEDICIS input parameters to match CORCON results
- 24 hours development of scenarios was analyzed

Criteria for strategy assessment:

Minimization of:
- vertical corium penetration depth
- horizontal corium penetration depth
- total mass of ablated concrete

Assessment of strategies

Comparison of the MEDICIS and the CORCON vertical penetration depths (m)

Scenario	CORCON	MEDICIS
TB2: no corium spreading no water injection	1.802	1.803
TB3: corium spreading no water injection	0.780	0.940
TB4: no corium spreading water injection	1.797	1.792
TB5: corium spreading water injection	0.805	0.823

Assessment of strategies

Comparison of the MEDICIS and the CORCON horizontal penetration depths (m)

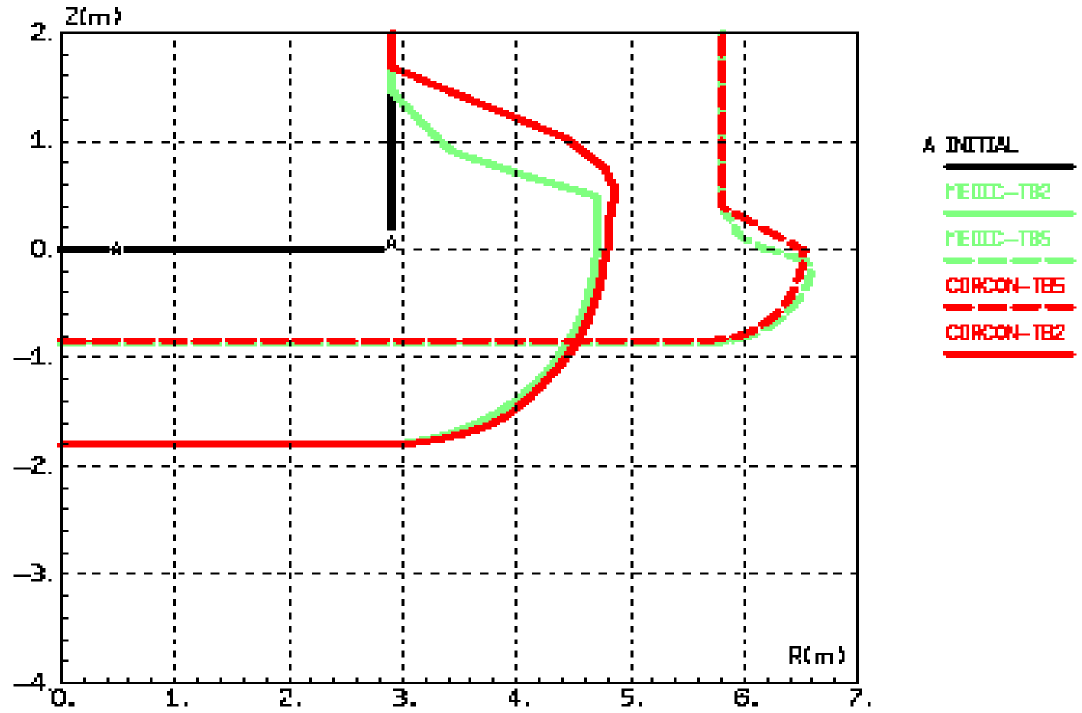
Scenario	CORCON	MEDICIS
TB2: no corium spreading no water injection	1.962	1.804
TB3: corium spreading no water injection	1.366	0.921
TB4: no corium spreading water injection	1.833	1.793
TB5: corium spreading water injection	0.729	0.804

Assessment of strategies

Comparison of MEDICIS and CORCON ablated concrete mass (ton)

Scenario	CORCON	MEDICIS
TB2: no corium spreading no water injection	471.0	334.3
TB3: corium spreading no water injection	398.1	313.3
TB4: no corium spreading water injection	439.7	329.7
TB5: corium spreading water injection	285.7	266.3

Assessment of strategies



VVER–1000

TB2 and TB5 CAVITY SHAPEs COMPARISON

Assessment of strategies

Effectiveness of corium management strategies:

Corium spreading out of the pit and water cooling is the best strategy to reduce corium penetration into concrete.

Application of this strategy resulted in

- **reduction of vertical penetration depth by factor 0.45 - 0.46,**
- **reduction of horizontal penetration depth by factor 0.37- 0.45,**
- **reduction of ablated concrete mass by factor 0.61- 0.79**

!!!!!! Nevertheles even this best strategy is not able to terminate CCI, it can only slow down corium penetration into concrete

Melting through of CNT basement

MELCOR study of corium behaviour:

<u>Reference scenario</u> without containment basement melting through
<u>Modified scenario</u> with melting through of containment basement

<u>Assumptions common for both of the scenarios</u>:
- **Severe accident scenario initiated by plant blackout**
- **Homogeneous corium pool configuration**
- **No ex-vessel corium management strategies applied**

<u>Assumptions for modified scenario:</u>
- **Extended model of VVER-1000 was used**
- **Containment basement slab broke down when residual thickness of the containment basement slab fell below 1 m**
- **Final deposition of corium is on basement slab of reactor building**

Melting through of CNT basement

Extended integral MELCOR model of VVER-1000 was prepared including models of:

- lower part of reactor building (non-hermetic)
- second melt pool cavity in the lower part of reactor building (on reactor building basement slab)

Melting through of CNT basement

Corium spreading in lower part of reactor building:

- **after melting and break down of containment basement corium penetrates into storey (level +6.6 m) under containment**
- **corium melts thin cover on the floor (that hides square opening with area of 1.9 m2)**
- **melt penetrates into storey at level 0.0 m**
- **corium melts two thin lids covering two square openings in the floor (area 2 x 1 m2)**
- **melt penetrates into storey at level –4.2 m (final destination)**
- **corium pool is formed on the reactor building basement slab**
- **corium – concrete interaction starts here**

Melting through of CNT basement

MELCOR Results:

Ref. scenario: - **RPV failure: 4 ½ hour**

- vertical corium penetration depth: 2.35 m (after 48 hours)

Mod. scenario: **- containment basement slab failure: 20 ½ hour start of corium transfer into reactor building, formation of corium pool (about 225 t) on the reactor building basement slab**

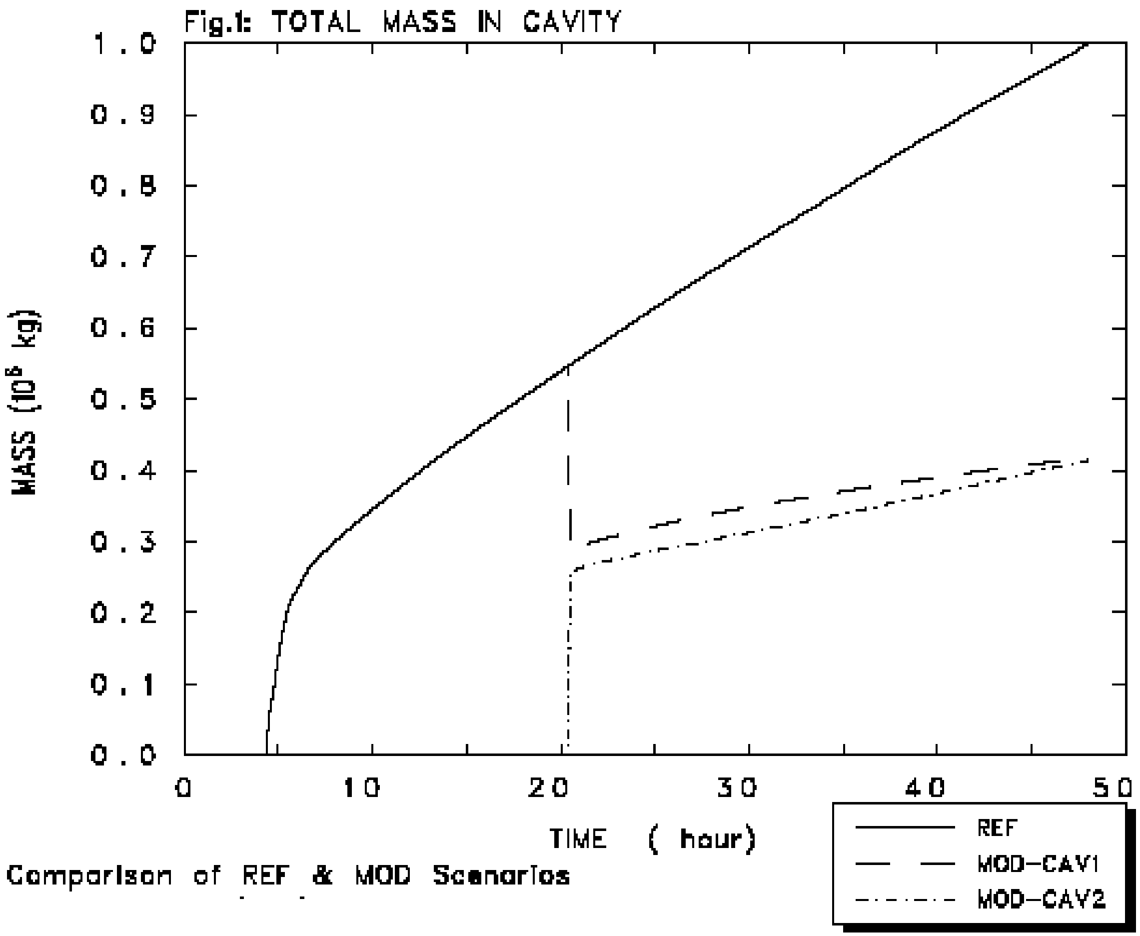
- CNT pressure at slab failure time: 200 kPa

- overpressure forced doors in the lower part of reactor building leading to environment

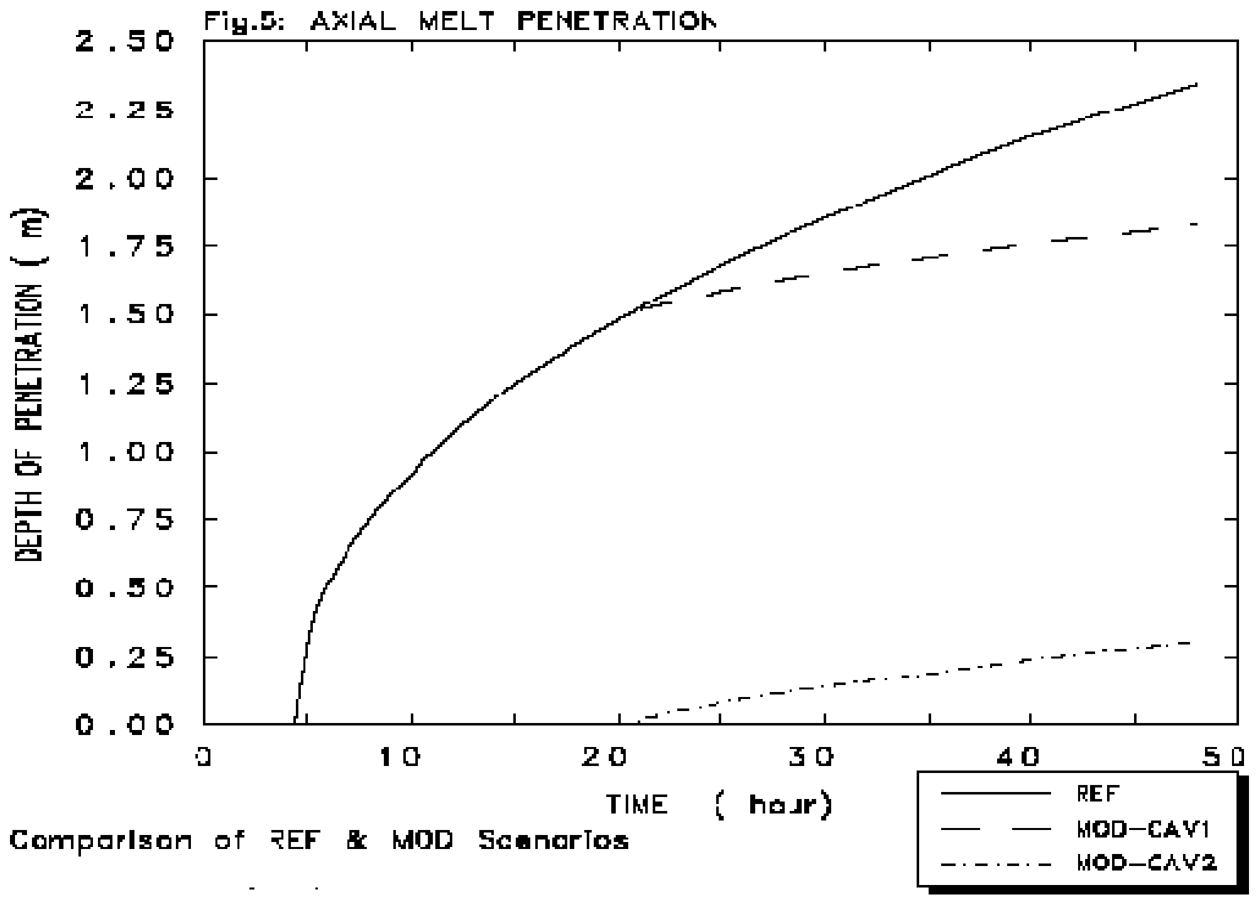
=> massive FP leak into environment

!!!!!! **increase in FP leak was in the range from 1 to 3 orders**

Melting through of CNT basement



Melting through of CNT basement



Summary

- The best strategy for ex-vessel corium management is corium spreading out of the cavity and cooling with water.
- The strategy can reduce vertical corium penetration depth by factor 0.45.
- But neither this strategy does not terminate corium-concrete interaction, it is able only to slow down concrete ablation.
- The melting through and break down of containment basement slab can be expected in a few days after the start of accident.
- Containment basement failure results in corium penetration into non- hermetic lower part of reactor building.
- This can lead to massive release of FP into environment namely in the case of overpressure in containment (opening the new leakage paths).

Ultimate corium management strategy

Objective of the ultimate strategy:

to prevent massive leakage of FP into environment

Proposal for remedial measures:

- **reinforsing and additional sealing of 7 doors leading from lower part of reactor building into environment (preventative measures),**
- **upkeeping containment leaktight during SA,**
- **removal of cover and lids on the floors of storeys +6.6 and ± 0.0 m to facilitate corium transfer to the final destination (during accident before containment basement failure),**
- **containment depressurization (before containment basement slab failure)**
- **assuring of long term heat removal from containment/reactor building**
- **prevention of H2 detonation (recombiners)**

Session 6

Jacobsen
Engineering
Ltd.

Criteria for the Transition to Severe Accident Management

OECD/NEA Workshop on Implementation of
Severe Accident Management Measures
ISAMM 2009
Schloss Boettstein, Switzerland
October 2009

R. Prior
Jacobsen Engineering Ltd

Jacobsen Engineering Ltd.

Outline

Symptoms for Transition

Comparison of Transition Temperatures

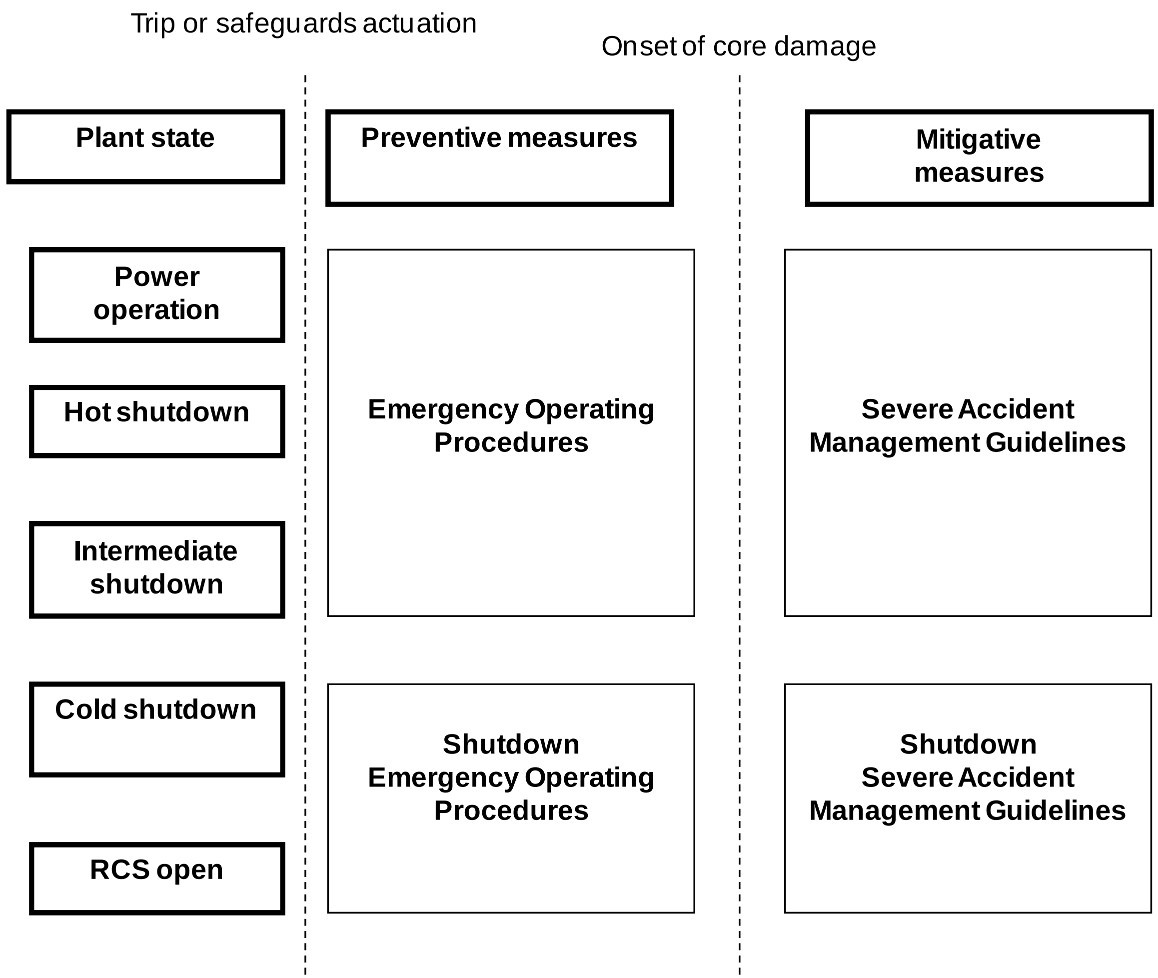
Influencing Factors

WGAMA CET Working Group

Conclusions

Jacobsen Engineering Ltd.

The Transition to Severe Accident Management



Detection of Core Damage

Concern: clad / fuel temperature

Possible measures:
-Core exit temperature
-Coolant level
-Containment hydrogen concentration
-Containment radiation

Some issues:
-Available range
-Survivability
-Interpretation
-Time response
-Suitability to cover multiple scenarios

Most (but not all) approaches use core exit temperature for transition
Some use others (either as primary indication or as backup)

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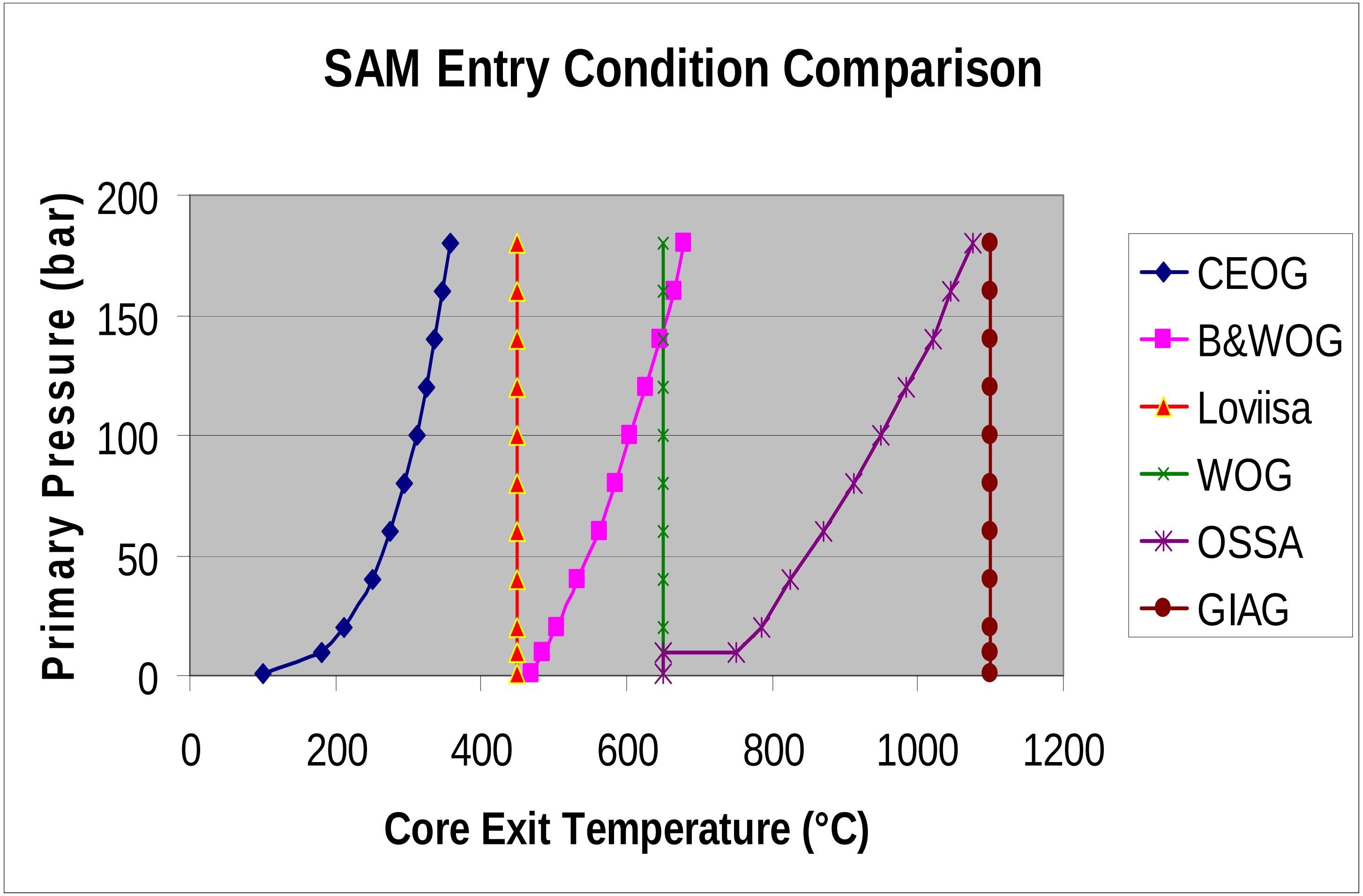
Comparison of SAMG Entry Conditions



Influencing Factors

Core Physical Condition
-Uncovered
-Deeply uncovered / significant superheat
-Fuel pellet damage / fission product release

Structure and scope of the AM:
-Simultaneous use of EOP and SAMG
-Treatment of severe accident phenomena

Strict (fixed) transition criterion

Decision making for transition

Application of margins
-Simplicity of use vs. ‘accuracy’

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The WGAMA CET Working Group

ROSA 6.1 test: apparent delayed response of CETs

GAMA CET WG set up to investigate

Survey on use of CET in AM
Review of applicable experimental evidence

Some conclusions:

All countries surveyed use CET widely in AM.
Normally margins are applied to setpoints.
Specific situations may need investigation.
Capability of analysis (and its validation) to develop CET setpoints

Working group detailed report this year.

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Conclusions

-Core exit temperature is widely used to detect the symptom for EOP to SAMG transition

-A wide range of temperature setpoints is used depending on the SAM approach

-However, this can be explained by consideration of other key characteristics of the SAM approach

-This interaction between factors must be carefully considered when developing SAMG

-There is no single best way to do this!

-But given the range of approaches, the importance of exercises and validation is emphasised.

Use of the Software Module SPRINT in the Netherlands for Prediction of the Source Term

Marcel Slootman
NRG The Netherlands

OECD/NEA Workshop on Implementation of Severe Accident Management Measures,
October 2009

SPRINT: Contents

- Main objective
- Main benefits
- Example of input screen / output screen
- Example of Belief Network
- Belief Network Development
- Overview of Borssele NPP (KCB)
- Establishment of KCB SPRINT model
- Overview of KCB ERO
- Example of using SPRINT
- Organizational aspects
- Feedback and findings during exercises
- Conclusions

Main objective for development

- To develop a flexible and adaptable system capable of generating plant specific Source Terms
  - develop a probabilistic model (a “Belief Network”) to rapidly infer the likely plant status from information on a number of key plant observables
  - a pre-calculated Source Term is assigned to each plant status

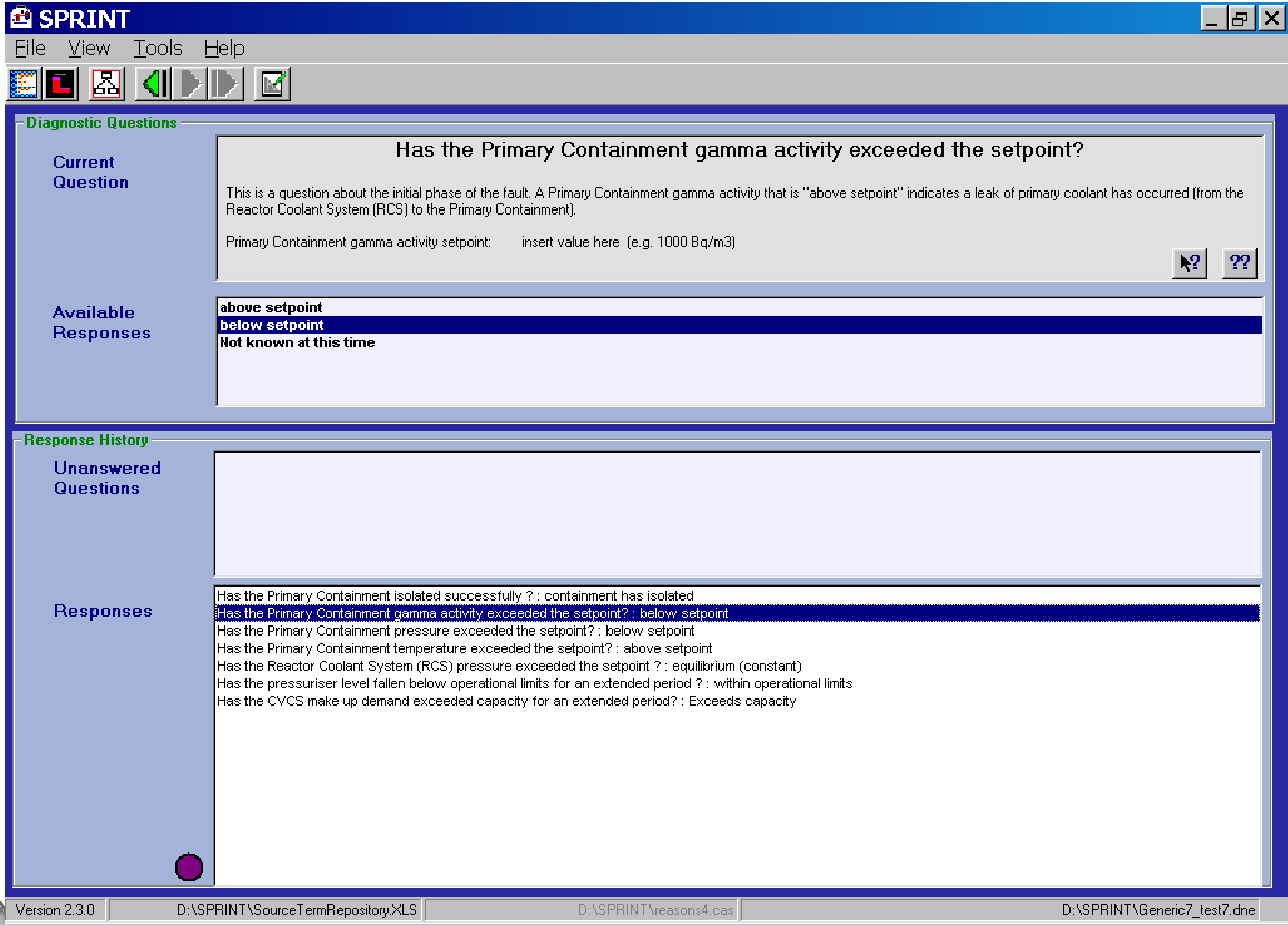
- Software module developed within the Euratom Framework Programs FP4, FP5 and FP6

Main Benefits

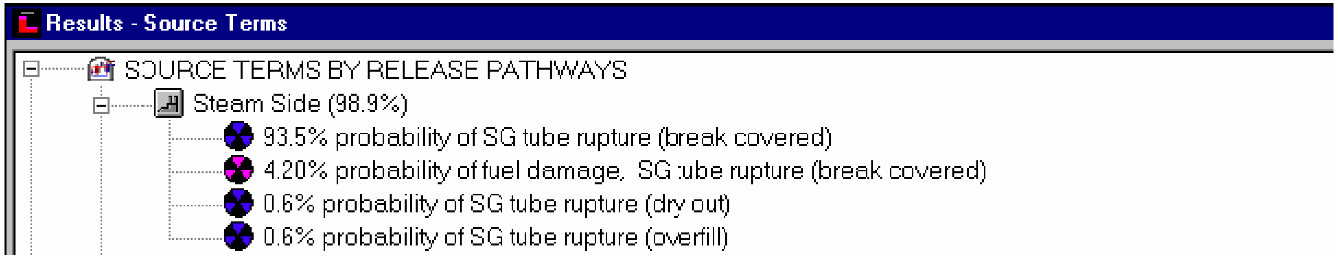
- It alerts the user to existence of other possible final plants states, based on “known” and “unknown” plant status parameters

  (in contrast to the deterministic approach, where assumptions have to be made about the “unknown” parameters)
- It functions in Beyond Design Basis conditions where the instrumentation may not be operating in its designated range e.g.
  - conflicting / unreliable reading
  - complete failure, i.e. no readings
- Rapid and early diagnosis

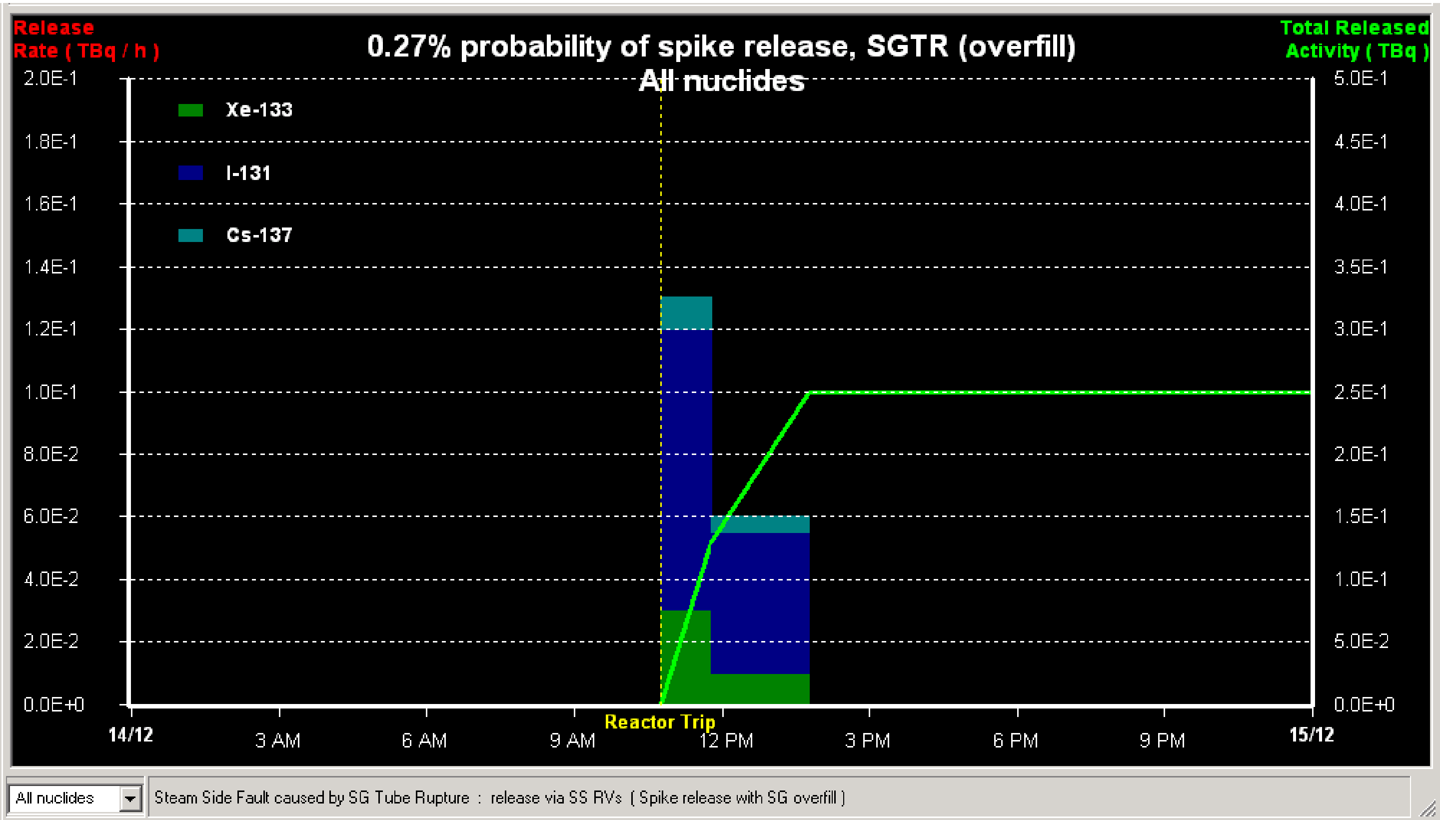
Example of SPRINT input screen



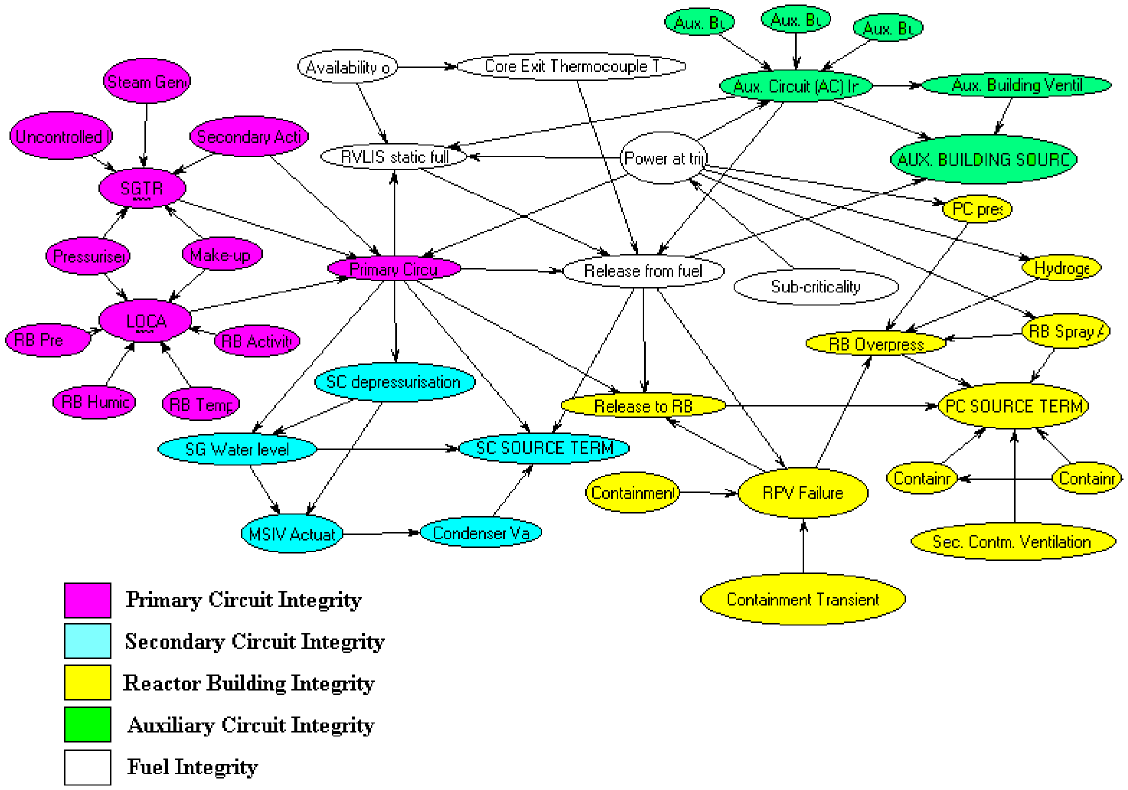
Example of Source Term Probability results



Example of one of the Source Terms



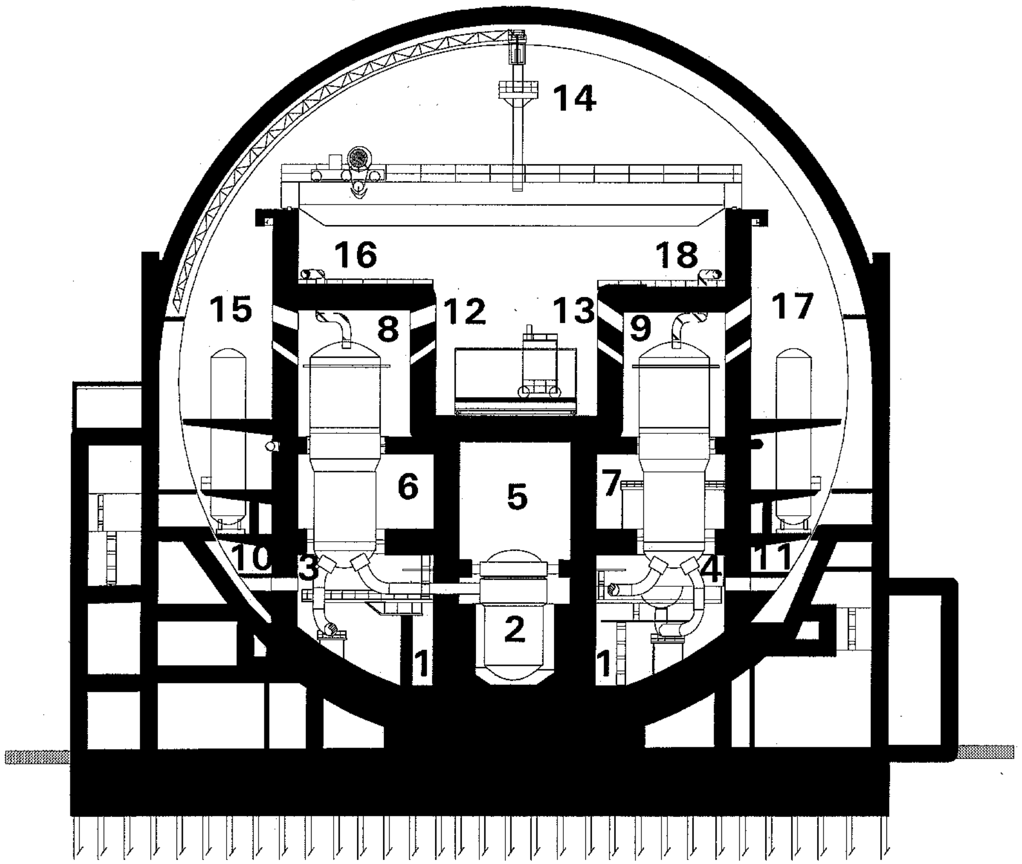
Example of a Belief Network



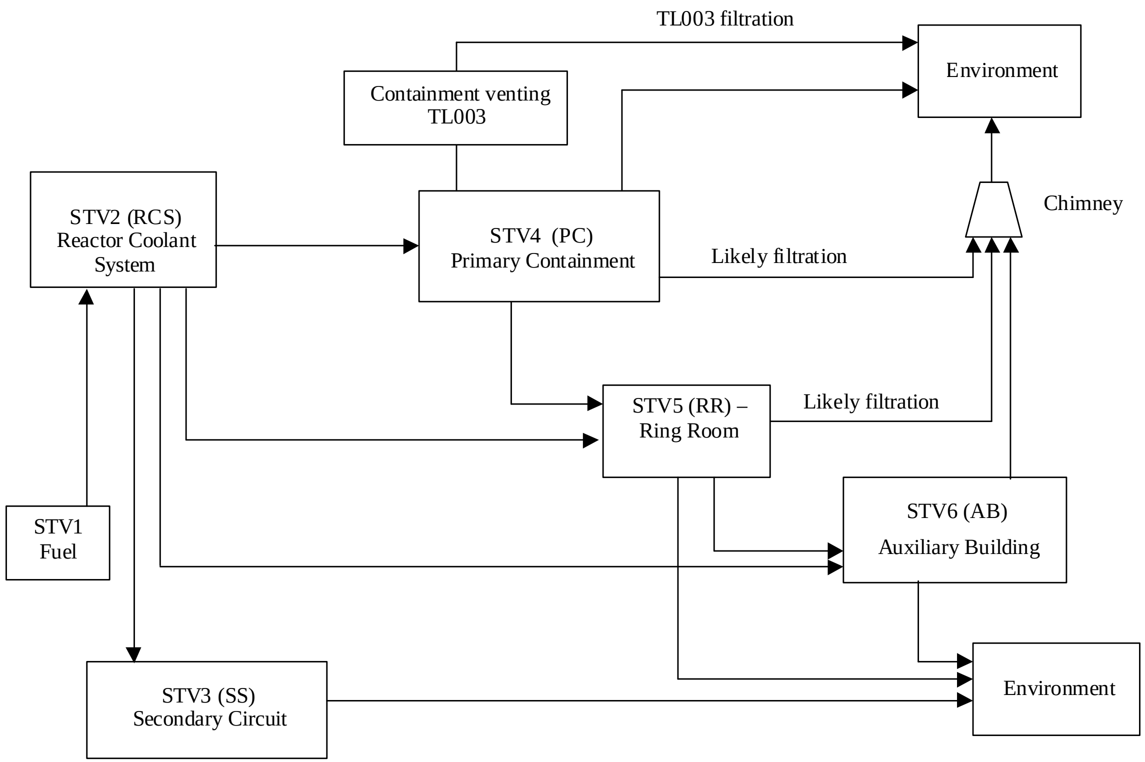
Belief Network Development

- Input and output data requirements:
  - definition of transport / release pathways
  - key plant systems
  - observable plant parameters
  - SAM measures and EOP measures
- Development of sub-networks:
  - Causal relationships between nodes
  - Input of Conditional Probabilities
  - Determination of question paths
- Testing sub-networks: does it behave as expected?

Borssele NPP (KCB): two loop PWR



Physical volumes and Fission Product Transport Routes for KCB NPP



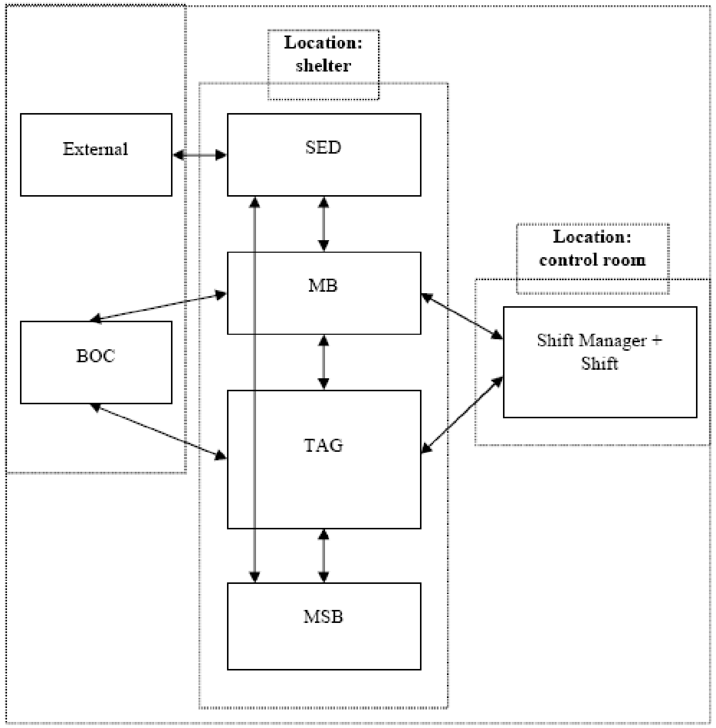
Conditional Probability Tables for KCB

- Values of Conditional Probability Tables based on:
  - PSA level 1 results
  - PSA level 2 results
  - MAAP analyses
  - Thermal hydraulic analyses
  - Specific KCB system knowledge
  - Expert judgement

Verification of the KCB SPRINT model

- Basic checks
- Comparison between the network and best-estimate event progressions (MAAP analyses)
- Comparison between the network and PSA results
- Representation of events with high consequences and low probabilities
- Important issue for verification: change of results due to the progression of an accident
- Verification showed reasonable/good results
- Documentation

Overview of ERO at KCB



BOC	Person responsible for actions in the plant
MB	Person to which the TAG reports
MSB	Source Term Manager
SED	Site Emergency Director
TAG	Technical Support Group

Use of SPRINT, organizational aspects

- Use during exercises of EOPs, SAMGs and Full Scale Emergency Exercise
- Additional TAG person recommended during SAMGs
- Location of SPRINT user at TSC: TAG room with plant process computer
- Preparation of KCB specific User Manual
- Coaching of SPRINT users
- Data sources:
  - information from computer screens
  - Information from shift/MB
- Update of SPRINT input every hour reasonable
  - At fast progression more often
- Periodical communication/discussion of the SPRINT results within the ERO

Emergency Exercise, some main steps

- Steam line break between SG2 and MSIV
- Failure of the electrical power
- After 1 min: SGTR in SG2
- After 15 min: failure of all emergency diesels
- After 9 hrs and 30 min: core heat up starts
- After 9 hrs and 50 min:

  - start of diesel EY050, so one TW pump available

  - maximum PCT about 800 ℃

Emergency Exercise, some main SPRINT results

- 1 hr (no electrical power and SGTR):
  - 22 % early release from a dry non-isolated SGTR, core melt
  - 76 % SGTR without core melt

- 9 hrs and 40 min: (core exit temp. increasing, low RPV level)
  - 85 % early release from a dry non-isolated SGTR, core melt
  - 14 % SGTR without core melt

Use of SPRINT, organizational aspects

- Use during exercises of EOPs, SAMGs and Full Scale Emergency Exercise
- Additional TAG person recommended during SAMGs
- Location of SPRINT user at TSC: TAG room with plant process computer
- Preparation of KCB specific User Manual
- Coaching of SPRINT users
- Data sources:
  - information from computer screens
  - Information from shift/MB
- Update of SPRINT input every hour reasonable
  - At fast progression more often
- Periodical communication/discussion of the SPRINT results within the ERO

SPRINT results at the plant

- ERO was alerted in the first phase to the existence of low probability / high consequences end states
- Start use of SPRINT at early stage of accident
- Prediction of initiating event satisfactorily
- Changes in Source Term predictions during accident progression were logically
- Explanation of the probabilities to the MSB, SED and authorities is important (events/ acc. progression)
- Use of existing Source Term decision tree on paper also recommended
- Some AM measures not yet included in SPRINT
- Performance is less accurate in case of a **temporary** restoration of system(s) or **temporary** AM measure(s)

Feedback from authorities

- SPRINT results are very useful for authorities
  - planning of emergency measures
  - decision making
- Timing of delivered source term information to authorities good
- Policy Team interested in SPRINT results
- Establish a strategy how to deal with probabilities
- Training of authority is important
  - for understanding SPRINT results
  - for insights in accident progression
- Communication during transfer of shift

Conclusions

- A SPRINT model for the KCB NPP has been developed and is used within the ERO and by the authorities during exercises
- ERO was alerted in the first phase to the existence of low probability / high consequences end states. For this purpose SPRINT is well suited
- Experiences during the exercises and some findings for improvement have been discussed
- SPRINT results are useful for authorities
  - planning of emergency measures
  - decision making

OECD/NEA Workshop on Implementation of Severe Accident Management Measures (ISAMM-2009)

Development, Validation and Training of Severe Accident Management Measures

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CONTENT

- Preamble
- SAMM/G Historic Perspective (Skip)
- Development of SAMM/Gs in Europe (Skip)
- Limitations of Current Approach
- Approach to a More Complete EOP/SAMG Validation
- Demonstration of ActiveChart Validation Model
- ActiveChart Applications and Insights

Preamble

Hardware vs. Operators:

Do we have our Priorities right?

Reactor Safety with SAM

Reactor Safety with Severe Accident Management

HARDWARE

ECCS: Prevent CD
SAMMs: Prevent ECF and Mitigate Release

EMERGENCY CONTROL TEAM

OPERATORS - EOPs
TSC: - SAMGs

Is the Deck stacked against Operators?

	PREVENTION: Prevent CD	MITIGATION: Prevent ECF	
			Emphasis this Workshop
HARDWARE	HPI-1, HPI-2 | DEPRESSURIZE 1 - 16: LPIS-1, LPIS-2, RHR-1, RHR-2, RHR-3, RHR-4	Piped Firewater, ESW Injection, Portable Firewater, Cavity Flooding, Containment Spray, Filtered Containment Vent, ETC. …..	
	Typical Level of Redundancy: 5 to 8 Trains	Typical Level of Redundancy: 3 to 5 Trains	
	Typical Reliability: No CCF < 10-8 With CCF < 10-5	Typical Reliability: No CCF < 10-4 With CCF < 10-3	80 to 90 %
OPERATORS	EOPs	SAMGs	

MCR: 1 Supervisor, 2 to 3 Operators
each with his own Assignment

10 to 20 %

TSC: 5 to 10 each with his own
Assignment

Off-site Support

Typical Level of Redundancy: 1

Reliability required to match Hardware: 10-8

What is the Evidence ?

	Statistical Evidence TMI/Chernobyl	Evidence from PSA	NRC
Core Damage Frequency	2x10-4 / RY	10-5 to 10-6/RY	1x10-4 to 1x1-05 / RY
Conditional Probability of ECF given CD	0.5	0.1 to 0.01	0.01 to 0.1
CAUSE			
Errors of OMMISSION	0 %	Considered	
Errors of COMMISSION	100 %	Not Considered	

CONCLUSION 1: THE PURPOSE OF THE (EOP/SAMG)s HAS TO BE TO MAKE SURE THAT ERRORS OF COMMISION CAN BE ELIMINATED FROM CONSIDERATION BY PROVIDING INSTRUCTIONS AND TRAINING TO THE OPERATORS THAT REPLACE AD HOC DECISION MAKING DURING AN ACCIDENT

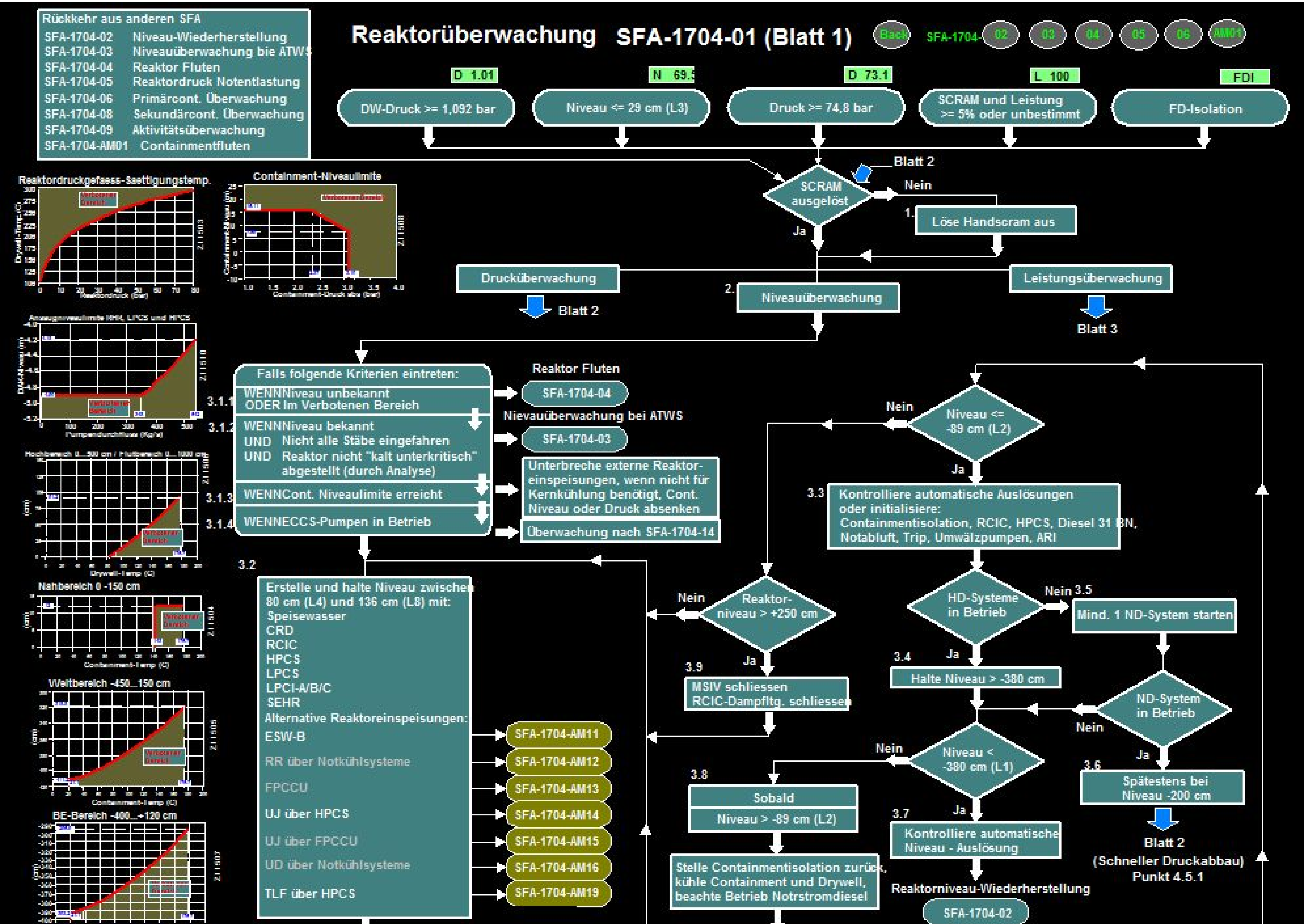
CONCLUSION 2: FOR A BALANCED SAFETY PICTURE THESE INSTRUCTIONS MUST BE NEARLY "BULLET PROOF" WITH A SINGLE REDUNDANCY HUMAN SYSTEM AGAINST A MULTI-REDUNDANCY HARDWARE SYSTEM

SAMM Historical Perspective

- SAMMs were triggered by TMI Accident
  - Beyond Design Basis conditions can develop from benign initiating events
  - Goal: Provide guidance to lead the Accident Management (AM) team to take the corrective actions necessary to:
    - Prevent core damage
    - Mitigate the consequences of core damage in the environment
  - Initially developed as SAMGs in USA by Owner's Groups, supported by manufacturers and severe accident experts
  - SAMGs are Symptom based rather than procedure driven
  - US chose "New Look" approach:
    - Accident management is taken over by Technical Support Center (TSC) when an accident proceeds beyond the design basis.
    - TSC re-evaluates the plant condition from ground up
    - SAMGs have been implemented for about 10-15 years

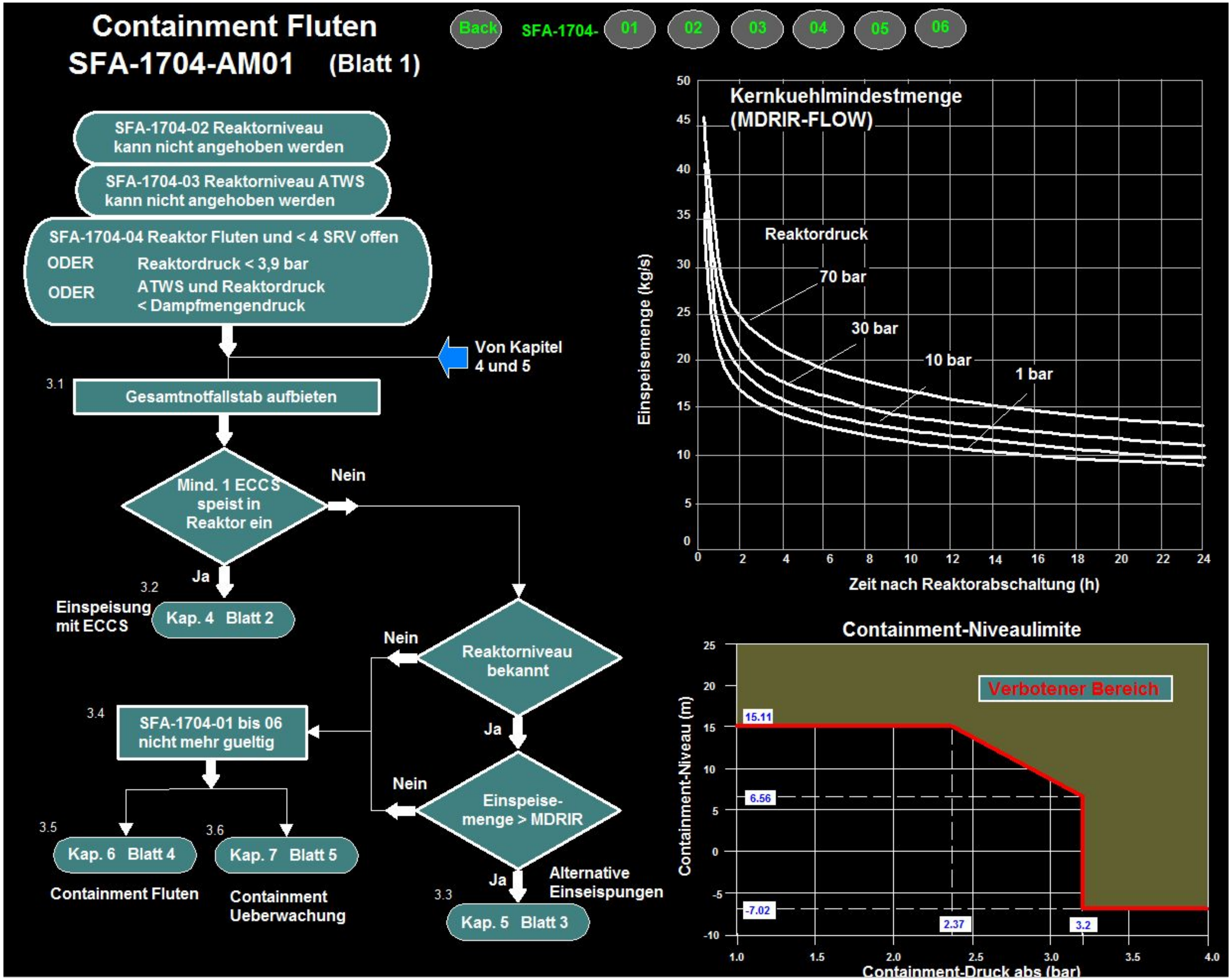
SAMM Development in Europe

- In Europe SAMMs are at various stages of being developed and implemented
  - Taking mostly the "Continuity" approach
  - Accident management is continuous, it is the senior operations person on site at the time of the incident
  - TSC is an advisor to accident manager
  - SAMMs link to and continue from the design basis emergency procedures or EOPs
  - SAMMs are developed by plant operations or by the manufacturer and with support from severe accident experts
- SAMG/Ms usually consist of decision and action flowcharts backed up by more detailed information and procedures
  - Flowcharts incorporate Symptom based decisions:
    - If pressure is > x bar do this, otherwise do that
    - What constitutes a "Yes" answer can change to a "No" answer quickly
    - Flowcharts need be "looped" until accident is under control



Reaktorüberwachung SFA-1704-01 (Blatt 1)
Back
SFA-1704-
02
03
04
05
06
AM01
Rückkehr aus anderen SFA
SFA-1704-02 Niveau-Wiederherstellung
SFA-1704-03 Niveauüberwachung bie ATWS
SFA-1704-04 Reaktor Fluten
SFA-1704-05 Reaktordruck Notentlastung
SFA-1704-06 Primärcont. Überwachung
SFA-1704-08 Sekundärcont. Überwachung
SFA-1704-09 Aktivitätsüberwachung
SFA-1704-AM01 Containmentfluten
D 1.01
DW-Druck >= 1,092 bar
N 69.5
Niveau <= 29 cm (L3)
D 73.1
Druck >= 74,8 bar
L 100
SCRAM und Leistung >= 5% oder unbestimmt
FDI
FD-Isolation
Blatt 2
SCRAM ausgelöst
Nein
1.
Löse Handscram aus
Ja
Drucküberwachung
Blatt 2
2.
Niveauüberwachung
Leistungsüberwachung
Blatt 3
Reaktordruckgefaess-Saettigungstemp.
Reaktordruck (bar)
Containment-Niveaulimite
Containment-Druck abs (bar)
Ansaugniveaulimite RHR, LPCS und HPCS
Pumpendurchfluss (Kg/s)
Hochbereich 0...500 cm / Flutbereich 0...1000 cm
Drywell-Temp (C)
Nahbereich 0 -150 cm
Containment-Temp (C)
Weitbereich -450...150 cm
Containment-Temp (C)
BE-Bereich -400...+120 cm
Falls folgende Kriterien eintreten:
3.1.1 WENN Niveau unbekannt ODER Im Verbotenen Bereich
Reaktor Fluten
SFA-1704-04
3.1.2 WENN Niveau bekannt
UND Nicht alle Stäbe eingefahren
UND Reaktor nicht "kalt unterkritisch" abgestellt (durch Analyse)
Nievauüberwachung bei ATWS
SFA-1704-03
3.1.3 WENN Cont. Niveaulimite erreicht
Unterbreche externe Reaktor-einspeisungen, wenn nicht für Kernkühlung benötigt, Cont. Niveau oder Druck absenken
3.1.4 WENN ECCS-Pumpen in Betrieb
Überwachung nach SFA-1704-14
3.2
Erstelle und halte Niveau zwischen 80 cm (L4) und 136 cm (L8) mit:
Speisewasser
CRD
RCIC
HPCS
LPCS
LPCI-A/B/C
SEHR
Alternative Reaktoreinspeisungen:
ESW-B
SFA-1704-AM11
RR über Notkühlsysteme
SFA-1704-AM12
FPCCU
SFA-1704-AM13
UJ über HPCS
SFA-1704-AM14
UJ über FPCCU
SFA-1704-AM15
UD über Notkühlsysteme
SFA-1704-AM16
TLF über HPCS
SFA-1704-AM19
Nein
Niveau <= -89 cm (L2)
Ja
3.3
Kontrolliere automatische Auslösungen oder initialisiere: Containmentisolation, RCIC, HPCS, Diesel 31 BN, Notabluft, Trip, Umwälzpumpen, ARI
Nein
Reaktor-niveau > +250 cm
Ja
3.9
MSIV schliessen
RCIC-Dampfltg. schliessen
HD-Systeme in Betrieb
Nein
3.5
Mind. 1 ND-System starten
3.4
Ja
Halte Niveau > -380 cm
ND-System in Betrieb
Nein
Ja
3.6
Spätestens bei Niveau -200 cm
Blatt 2 (Schneller Druckabbau) Punkt 4.5.1
Nein
Niveau < -380 cm (L1)
Ja
3.8
Sobald
Niveau > -89 cm (L2)
Stelle Containmentisolation zurück, kühle Containment und Drywell, beachte Betrieb Notstromdiesel
3.7
Kontrolliere automatische Niveau - Auslösung
Reaktorniveau-Wiederherstellung
SFA-1704-02

SAMM/G Development in Europe 2



SAMM/G Development in Europe 3

- SAMM development is supported by limited accident progression analyses (MAAP, MELCOR, ASTEC in the future)
  – MAAP developed by US Industry in IDCOR program. Now developed, maintained and distributed by EPRI. Nearly all US plants use MAAP.
  – MELCOR developed by USNRC. Available free to countries participating in USNRC’s code development and validation program. Used by most European organizations.
  – ASTEC under development by IRSN (France) and GRS (Germany)
- SAMMs are introduced at a plant by:
  – Training the TSC staff in the systematic use of SAMMs and in severe accident phenomena.
  – Training for the operating shift to the extent necessary to manage the initial phases of severe accidents until the TSC is assembled and up to speed.
  – SAMM drills are conducted in typically 1-2 year intervals
    - Based on pre-calculated accident scenarios
    - Slide plots for relevant parameters (hide curves for times greater than current time)
    - Train communication between TSC and operating shift and outside organizations
    - A real-time SAMM Simulator tool is used by very few plants

Limitations of Current Approach

- SAMM/G implementation approach differs significantly from EOPs
  - EOPs are continuously trained in real-time training exercises on plant simulator
  - EOP validation is only limited by fidelity of plant simulator
  - Backed by analyses from first principle codes (RELAP, RETRAN, COCOSYS)
  - SAMM/Gs have more limited validation through severe accident analyses and table top exercises but without real-time training exercises
- Should SAMM/G training employ simulator-like real-time training?
- Should the Simulator be extended for severe accident phenomena?
  - TSC is not in the Main Control Room but communicates with MCR
  - Many TSCs use Safety Parameter Display Systems (SPDS) for plant status information
  - TSC training needs to be conducted in the TSC environment
  - Extended simulator would not be useful for TSC training
  - For SAMM/G training, SPDS should include severe accident phenomena

SAMM/G Validation

- EOP/SAMG flowcharts should be complete.
  - Fast developing accident can reach core damage in less than 2 hours. These sequences are both important and they place the limiting demands on the EOP/SAMGs.
  - The accident manager does not have time to consult backup information.
  - Backup information should be mostly for education and background knowledge.
- EOP/SAMG Validation is no trivial matter. Consider:
  - Path through flowcharts and possible actions are not known a priory for a given sequence
  - SAMM/Gs can not be validated in isolation. They link to EOPs and must be validated as an integral EOP/SAMG package.
  - **EOP/SAMG validation needs to demonstrate that the flowcharts guide the ERO to possible actions that prevent core damage or mitigate the consequences for accidents where this is possible, regardless of when it occurs (10 AM on a work day or 3 AM on New Years Day).**
- What does this mean in practice? Consider:
  - No actions are needed if only one train of one ECCS safety system functions as designed (for some systems this includes depressurization)
  - Operator actions to protect the core and environment are needed and are possible if:
    - All ECCS systems fail, some mechanically and others by automatic actuation failure. Most favorable case: First train of first ECCS system asked in flowcharts has automatic actuation failed. Most limiting case: Only last train of last ECCS system asked in flowcharts has automatic actuation failed and the train needs to be lined up locally.
    - All trains of all ECCS systems fail mechanically. ERO must get through charts to where RCS is depressurized and auxiliary systems (i.e. firewater) are lined up. These systems are usually considered last in the charts.
    - Even if core damage can not be prevented the ERO can still mitigate consequences by manually isolating open containment lines, flooding the containment, etc.

SAMM/G Validation 2

- **Operators are trained to systematically follow the EOP/SAMG instructions**
  - Each step in the EOP/SAMGs takes time. Manual local actions take longer than MCR actions.
  - The time required to take an action is the sum of the times required for each step in the charts leading to that action. It can range from minutes to hours.
  - Timing and an optimal success oriented structure of the charts are critical
  - Timing is different from sequence to sequence
  - Some EOP/SAMGs employ 2 – 6 parallel tracks (i. e. pressure control, level control, power control, etc) to speed up processing. This brings a risk of conflicting decisions and actions.
- EOP/SAMG validation is important because the purpose of SAMGs is to lead the ERO to prevent core damage or mitigate the consequences if needed & possible.
- EOP/SAMG validation by the plant simulator is not practical because:
  - Validation by real-time simulation takes too long
  - The simulator does not model severe accident phenomena.
- EOP/SAMM validation by MAAP/MELCOR analyses is a cumbersome and inherently iterative process
  - Run an accident sequence without any corrective action
  - Trace thru charts to determine first possible action and time when point in chart is reached
  - Re-run sequence with action modeled at indicated time. Repeat for each action.
  - Not practical for a thorough validation, and for long running sequences (MELCOR)
- Is the current generation of SAMGs adequately validated, i. e. do they lead ERO successfully to corrective actions in severe accidents where such actions are both necessary and possible?
  - At this time we do not have this evidence in hand

Approach to More Complete EOP/SAMG Validation

- Define Validation Matrix
  - List the functional sequences that require manual action
  - Consider sequence variations (MCR vs. remote actions, Failure at t=0 vs. delayed or staggered failures
  - Operations tabulates the times required for each step in the Charts
  - Check against PSA for completeness (Note: PSAs don't give functional sequences)
- Analyze validation matrix sequences
  - Can involve large number of sequences. Need to automate the process.
  - Develop dynamic model of charts and link to running severe accident code.
  - Execute chart decisions and actions automatically with appropriate time delays.
  - Execute validation matrix in batch mode
  - Automatically extract necessary data with logic to determine success/failure
  - Review sequences with failure (core damaged, consequences not mitigated) for insights to optimize EOP/SAMGs.
- MELSIM/MAAPSIM ActiveChart demonstration with these features
  - Automated EOP/SAMM chart validation in automatic mode
  - Training and drills in manual mode
  - Complete EOP/SAMG training and drills environment, including communication, with Multi-Station Setup
  - Optimize EOP/SAMGs by addressing root causes of failure sequences

MAAPSIM ActiveChart Demonstration

- MAAPSIM INTERACTIVE Simulator driven by MAAP
- ActiveChart Logic Module for EOP/SAMG Charts
- Automatic Mode for Analysis Validation
- Manual Mode for Training
- Multi-Station Module for Exercises

ActiveChart Applications and Insights

- ActiveCharts has been used to mini-validate two BWR sites
- EOP/SAMG Charts performed reasonably well, but in all cases sequences with core damage were identified.
- Main causes were:
  - Extreme nature of sequences that require ERO intervention
  - Time required to execute EOP/SAMG instructions
  - Sensitivity cases with time delay multiples
  - Getting stuck in a Loop
  - Conflicting timing actions in parallel tracks
- Insights
  - No two sequences are the same
  - Indicated chart improvements are in may cases self evident (Chart completeness issue)
  - More parallel tracks than available people at night
  - EOP/SAMG optimization for execution speed is expected to yield additional success sequences

Severe Accidents Training in Spain: Experiences and Relevant Features

Severe Accidents Training in Spain: Experiences and Relevant Features

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Villigen-PSI, Switzerland, 26-28 October 2009

Presentation Contents

- ✓ Introduction
- ✓ Training Activities
- ✓ SAMG Revision
- ✓ Simulator models development
- ✓ Conclusions

Severe Accidents Training in Spain: Experiences and Relevant Features

Introduction

- ✓ SAMG official implementation in Spanish NPPs (2001-2002).
- ✓ SAMG development based on generic guidance of Owners Groups (W, GE and Siemens).
- ✓ Specific technical documentation
  - Methodology Manual
  - Verification and Validation Plan
    - Scenarios based on PSA results or specific MAAP calculations
  - **Training Modules.**

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

Introduction

- ✓ Main activities since implementation programs:
  - Retraining of plant personnel (basically plant operators and TSC members)
  - Updating and improvement of the plant specific SAMG.
- ✓ Objective of this presentation is to **summarize the experience** obtained along these years.

Training Activities

✓ Training Program in the implementation process includes the modules:

- Phenomenology and sequence of events associated to SA evolution
- Technical Basis of SAMG and Computational Aids with practical applications
- High-level actions and performance analysis of the instrumentation involved in SAM
- Exercises developed from PSA calculations introducing operator actions contemplated in SAMG.

Training Activities

- ✓ Nuclear Safety Council (CSN) requires an annual retraining program based on:
  - SAM drills development using and following the SAMG
  - Individual emergency exercises for the groups included in the Emergency Plan framework.
- ✓ Retraining global objective
  - Knowledge maintenance and upgrading related to phenomenology and management
  - Performance of the different plant groups in a severe accident situation.

Training Activities

✓ Specific retraining programs are addressed to <u>different personnel profiles</u>:

- **Technical Support Centre** members
  - Emergency Director
  - Evaluation Group
  - Radiological Control Group
- **Control Room** crew
- Operation auxiliaries (Trillo NPP)
- Instrumentation and Control Group (Trillo NPP).
- **Specific SAM Team** (Garoña NPP).

Training Activities

✓ Annual retraining considerations

- Two basic issues:
  - To remind the use rules of guidelines and procedures in the accident management
  - To evaluate the simulated scenarios.
- Complete response to an emergency scenario leading to a severe accident condition
  - Design Basis (EOP domain)
  - **Core degraded (SAMG domain)**
  - Transitions from EOP to SAMG.
- Relation existing between PSA Level 2 and strategies proposed in SAMG.

✓ TSC retraining items:

- Summary of the main physical and operational features of the scenario to be treated
- Evaluation according to Emergency Plan
  - Initial event identification
  - Emergency classification
- Accident management according to DBA conditions (questions from EOP)
- Summary of the main physical features of SA, with a particular aspect considered every year: hydrogen issues, fission products release, ….

Training Activities

✓ TSC retraining items:

- Strategies related to the degraded scenario and contemplated in the appropriate SAMGs
- Practical applications of the required diagnostic diagrams and computational aids
- Analysis of the proposed or real changes in the SAMG current version
- Evaluation and Radiological Control Groups performance
- Training drills for the emergency management, before and after severe accident threshold.

SAMG retraining planning

EOPs domain	Transition to severe accident	SAMGs domain
MSLB / SGTR	FR-C.1	SAG-5 / SCG-1
LOCA	ECA-1.3	SAG-7 / SCG-3
SBLO	ECA-0.0	SAG-3 / SAG-4
FWLB	FR-C.1	SAG-1 / SAG-2
LOCA	FR-C.1	SAG-6 / SCG-2

Training Activities

✓ Improvement plans of TSC for Spanish NPP

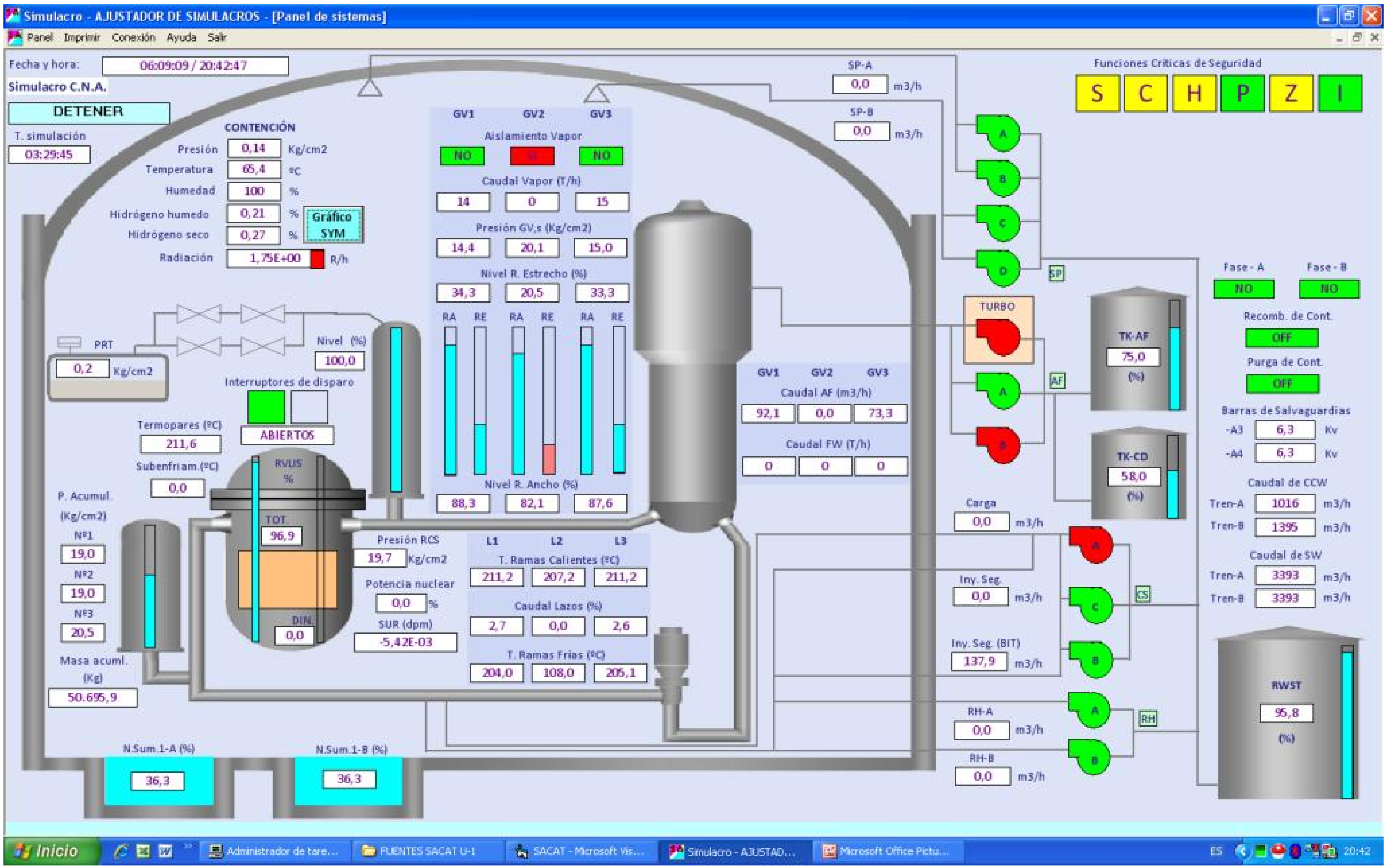
- Redesigning or updating of software/hardware tools (Almaraz NPP)
- Simple computer tools for emergency training purposes:
  - Multi-media software to physical phenomena and basic strategies
  - Hydrogen curves showing the correspondence between “dry” and “wet” measures
  - Changes in Safety Parameters Display System (SPDS)

Training Activities

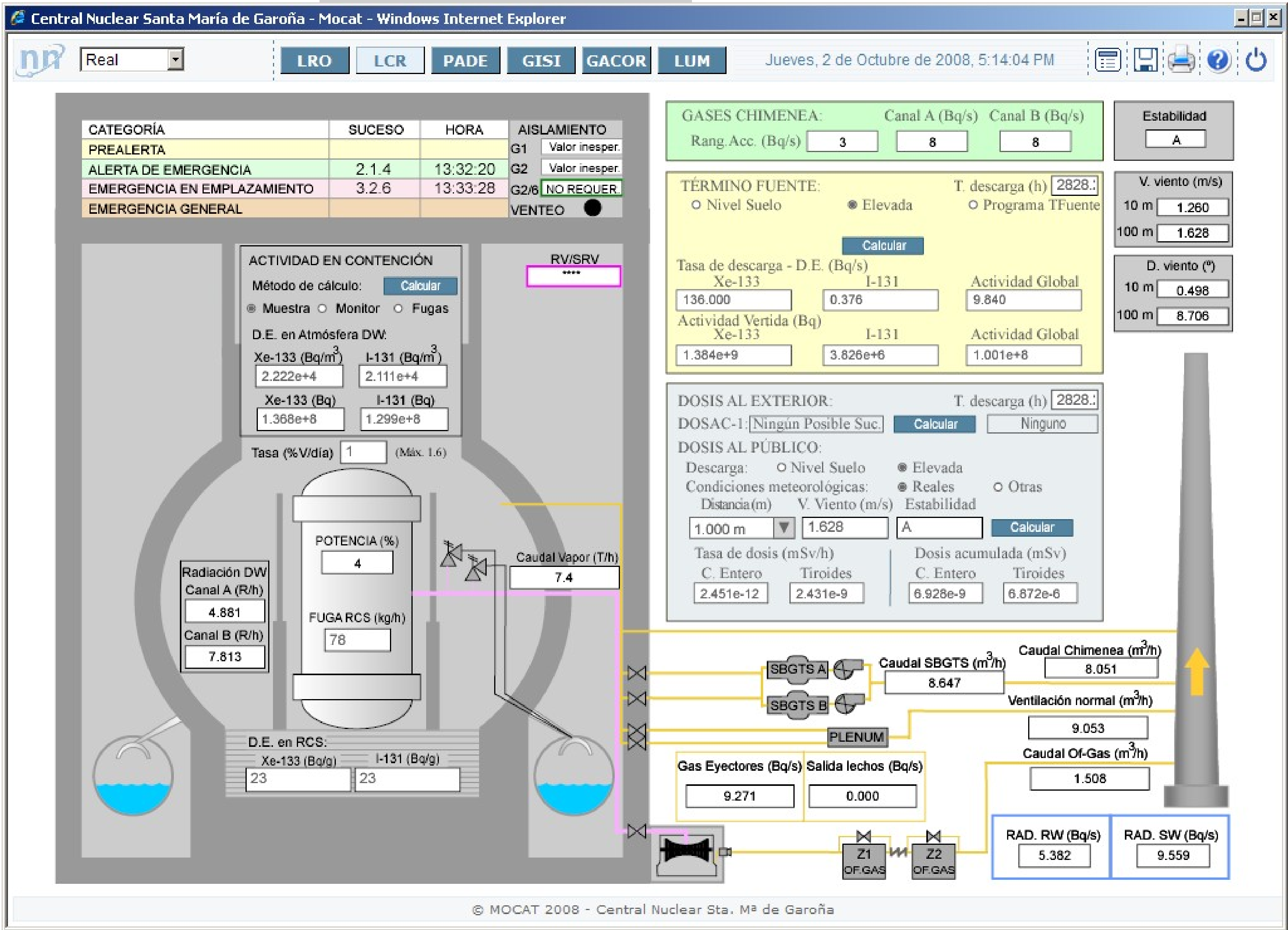
- ✓ Improvement plans of TSC for Spanish NPP
  - Computer tool developed by Tecnatom for TSC training in emergencies
    - Following and evaluating plant parameters
    - Radiological group (source term and doses estimation)
    - Tasks and responsibilities of TSC members
    - SAMG accomplishment.
    - Nowadays in
      - Almaraz NPP (SACAT Project)
      - Garoña NPP (MOCAT Project)
      - Development process in Trillo NPP.

Systems Screen (Almaraz NPP)

Severe Accidents Training in Spain: Experiences and Relevant Features



Radiological Screen (Garoña NPP)



Severe Accidents Training in Spain: Experiences and Relevant Features

SAMG Revision

✓ <u>Basic issues in these revisions</u>

- Applicable changes package included in the generic guidelines revision.
- Results and <u>experiences of training courses</u> and exercises carried out during last years (<u>6-7 years of real experience</u>).
- Methodology to make easier the future updating (Maintenance Control Sheets remarking the change cause).
- Increase of applicability and efficiency (PSA revisions, plant design modifications).

SAMG Revision

✓ Present situation in Spain

- Official implementation of SAMG Revision 1
  - Almaraz (July 2008)
  - Ascó and Vandellós (foreseen December 2009)
- Official implementation of SAG Revision 2A, based on EPG/SAG Revision 2
  - Garoña (2007)
  - Cofrentes (2008)
- Relevant instrumentation changes have not been considered necessary (ranges mainly).
- Hydrogen concentration measurement based on continuous, spatially distributed system.

SAMG Revision

- Identification of some areas to improve SAM possibilities (Ascó and Vandellós NPPs)
  - o <u>Passive Autocatalytic Recombiners (PAR), Trillo has this system.</u>
  - o Analysis to improve the filling capability of the reactor cavity ("dry cavity")
  - o Analysis to improve the fast filling capability of the Refuelling Water Storage Tank (RWST)
- Changes in Trillo NPP (<u>severe accident RSK recommendations)</u> at implementation date
  - o Control Room Air Filtering
  - o Secondary "Feed and Bleed"
  - o Emergency Power Supply
  - o Containment Hydrogen Control (PAR)

Simulator Models Development

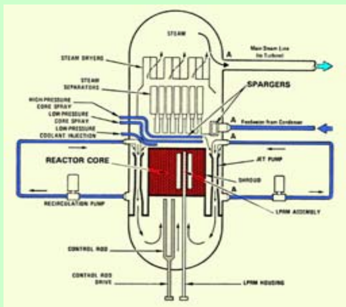
- ✓ Tecnatom has carried out the implementation of a <u>SA module for the full scope simulator</u> of Laguna Verde NPP (Mexico, 2003-2005).
  - GE design BWR/5, owned by CFE (Electricity Federal Commission).
  - MAAP-4 based <u>"Containment Advanced Model" (MAC)</u> integrated with the plant models
    - TRAC-RT thermalhydraulic code for reactor coolant and main steam system calculations
    - NEMO for core neutronics and instrumentation modelling tool.

Simulator Models Development

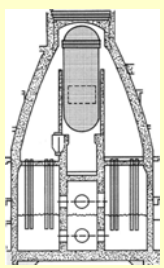
- Improvement of the modelling package
  - Scope to beyond DBA conditions
  - Training range to degraded core situations
- Simulator capabilities enhancement
  - Evaluation and validation of plant specific SAMG
  - Training sessions on SAMG for the different personnel profiles.
- Supporting tool for:
  - Definition and evaluation of SA mitigation strategies
  - Analysis of available or alternative instrumentation.

Integration with Simulator Models

Severe Accidents Training in Spain: Experiences and Relevant Features

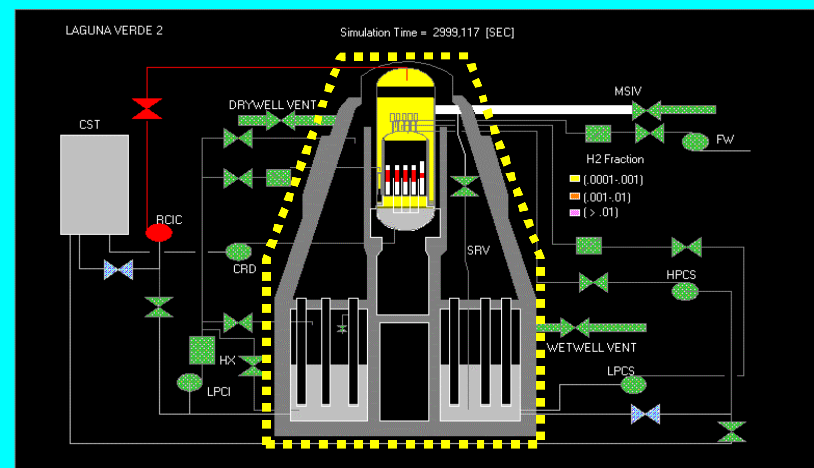


NSSS / Core Models (TRAC-RT/ NEMO)



Containment Model

Severe Accident Module (MAAP-4)



Conclusions

✓ **Main conclusions of Tecnatom experiences**

- Appreciable improvement and **familiarisation with the SAMG use.**
- Feedback related to **degraded conditions**:
  - Strategies
  - Unusual alignments
  - Working teams actuation
- Usefulness of **dynamic exercises** with an increasing **degree of participation** of the involved personnel.
- Feedback of the **PSA and SAMG**: data and experience.

Conclusions

✓ **Main conclusions of Tecnatom experiences**

- **Simple tools** supporting to TSC actuation increase the interactivity and **make more dynamic the training process.**
- **Feedback** of obtained experience and participants suggestions **to future retraining courses and guidelines improvement.**
- Increasing of the plant participation degree in the **"severe accident culture"**
  - Efficient communication between TSC and Control Room
  - **Compromising of different plant organisations**
  - Decision making process.

Conclusions

✓ **Main conclusions of Tecnatom experiences**

- **Extension of the PSA** to different groups in the plant.
- Improvement of **SPDS**.
- **Experience interchange** between similar plants.

***Session** 7*

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

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ISAMM 2009

Schloss Böttstein Switzerland

October 26-28, 2009

Outline

- Iodine issue
- Synapses of Current Status on Containment venting
- PSI Approach to address the Iodine Issue
- Requirements deemed from the new process
- Limitations of commonly used oxidant: Sodium Thiosulphate
- Basis of PSI process and results
- Implementation of PSI process in NPP containment venting filter systems
- Anticipated global safety benefits
- Conclusions

Iodine Issue

- Iodine is a fission product and each LWR core has several 10 kg. During a severe accident a large fraction will be released from the core.
- Iodine with 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 which are extremely complex and dependent on a large number of parameters:
  - Temperature and pressure
  - 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 with surfaces having different natures,
  - 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.

Iodine issue (Cont.)

- Many small and large scale separate effect tests 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, provided the complexity of the iodine behavior in various phases of the simulated severe accidents;
  - the release of iodine and other fission products and structural materials from the melting fuel bundle with AgInCd or B_4C control rod
  - 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
  - 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 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.

Iodine speciation

- 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.

Synapses of current status on iodine management

- 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, 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, 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.

Synapses of current status on iodine management (Cont.)

- 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 prevailing conditions that may occur in severe accident conditions, such as, high temperatures and radiation fields, low pH, 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.

Synapses of Current Status on Containment venting

- 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 neither secured nor systematically investigated.

PSI Approach to address the Iodine Issue

- PSI has chosen a different direction in managing the gaseous iodine from a containment equipped with containment venting filter system
  - irrespective of how it is generated and (independent of type and the origin of generated iodine species)
  - without knowing its magnitude with deemed accuracy.
- 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.

PSI Approach to address the Iodine Issue (cont.)

PSI launched a fundamental iodine chemistry project in 2002 and continued until 2008 to:

- Generate data on the basic decomposition of CH_3I for the demonstration of repeatability of literature data and extend it using in-situ β-/external γ-radiation,
- Study use of many different oxidation agents to decompose CH_3I,
- Establish a process for fast and efficient decomposition of CH_3I in aqueous solution and fixing iodide ions by utilizing a phase transfer catalyst and a reducing chemical reagent,
- Produce a large database (conducted over 1000 tests) covering a wide range of boundary conditions feasible under all possible accident scenarios.

Requirements deemed from the new process

- Implementation should secure significant reduction in the amount of released volatile organic iodides and gaseous elemental iodine into the environment.
- In addition to any technical requirements for the implementation in engineered systems, the process should also clearly require:
  - Demonstration of robustness with respect to possible large variations in parameters affecting iodine chemistry,
  - Demonstration of the guaranteed effectiveness under operational conditions of the existing Containment Filtered Venting Systems (CFVS),
  - Long term sustained effectiveness in the presence of other possible constituents in the solution of CFVS 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.

Limitations of Commonly Used Oxidant: Sodium Thiosulphate

The results of the PSI research project have confirmed the conclusion of past research on the use of alkaline thiosulphate solution, which demonstrated 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 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.

Basis of PSI Process

The PSI research demonstrated that the concurrent use of a phase transfer catalyst, specifically, Aliquat336®9 together with thiosulphate eliminates these problems.

Aliquat336 (ALI) was characterized as a versatile chemical additive, since:

- successfully already applied to nuclear technological processes, such as, spent fuel reprocessing and other metallurgical processes for metal extraction from ores.
- high stability to ionising radiation

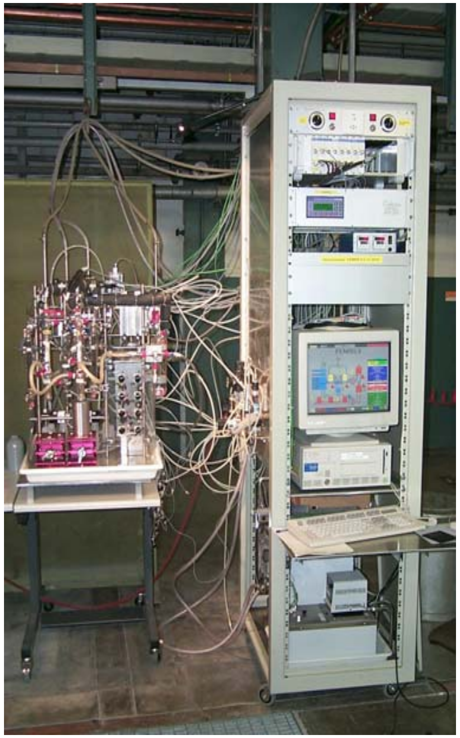
As a co-additive to alkaline thiosulphate solutions (THS),

- it increases the thermal decomposition rate of CH_3I
- and additionally binds the iodide ions formed from the decomposition process by which

  the oxidation of iodide ions in uncontrollable boundary conditions anticipated to occur in the system is suppressed effectively,

The new procedure for the retention of all volatile iodine species is already patented[1]

(1) S. Güntay and H. Bruchertseifer, European and International patent applications, 2005.



the reaction vessel, the apparatus for distillation, the sampling and activity control systems and control units are made ready for transfer to the hot

cell for in-situ β irradiations

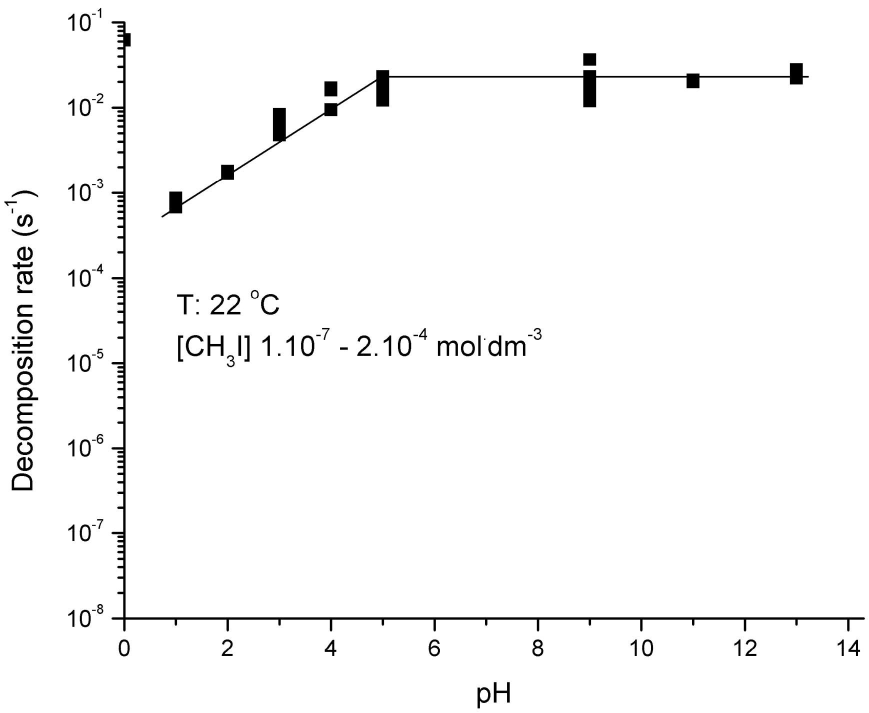
In the shielded cell of the Hot Lab… a computerized, remote-operating system has

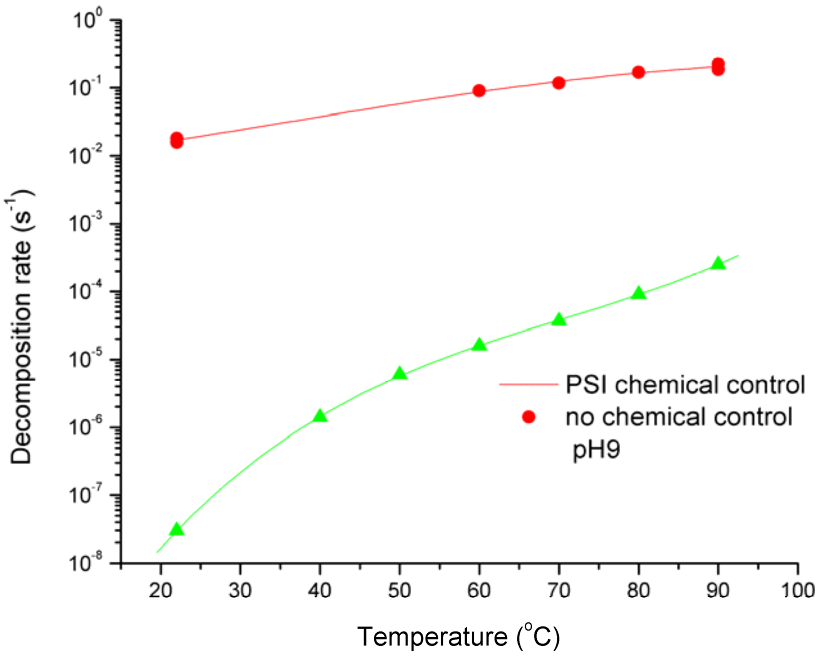
been installed

Photo of reaction vessel in γ-irradiation chamber

and gamma-cell

Effectiveness of PSI Process



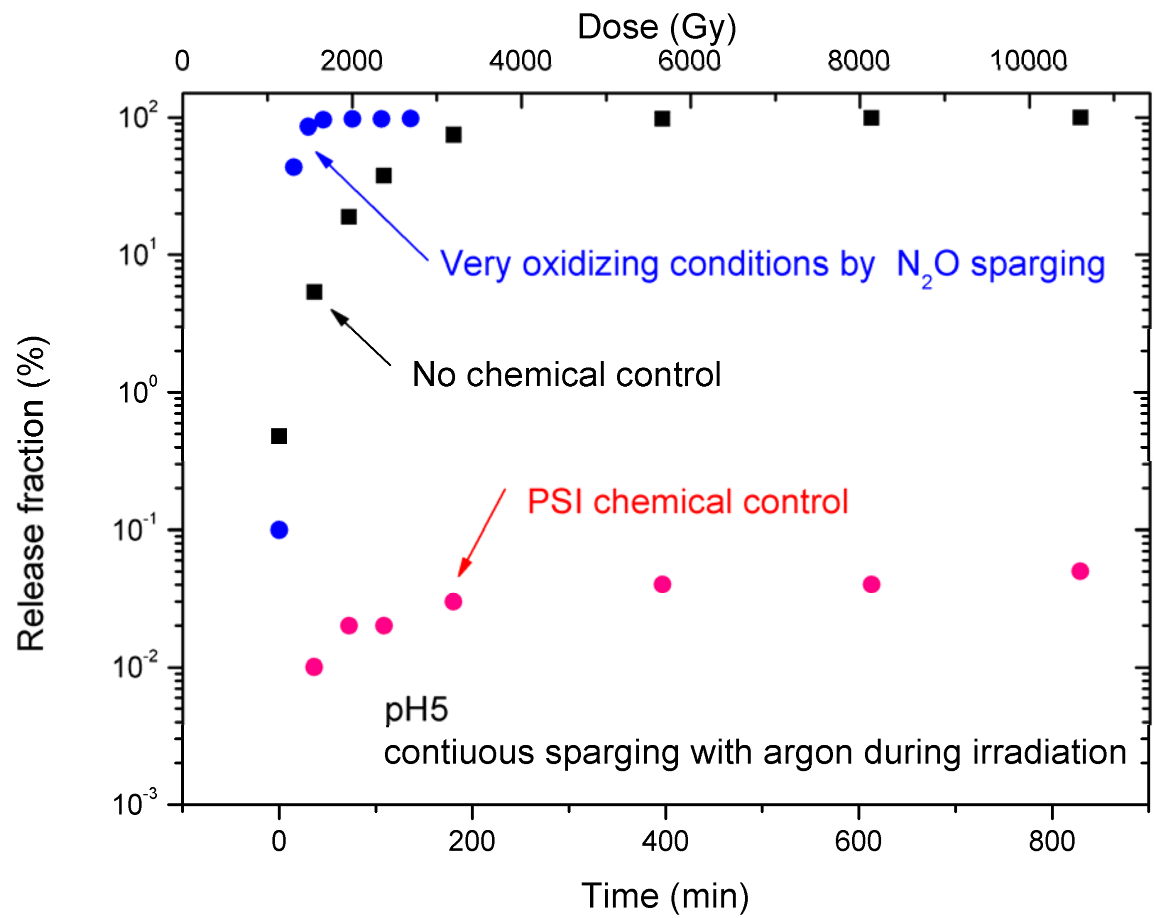


Quantification of the Enhanced Decomposition of CH_3I

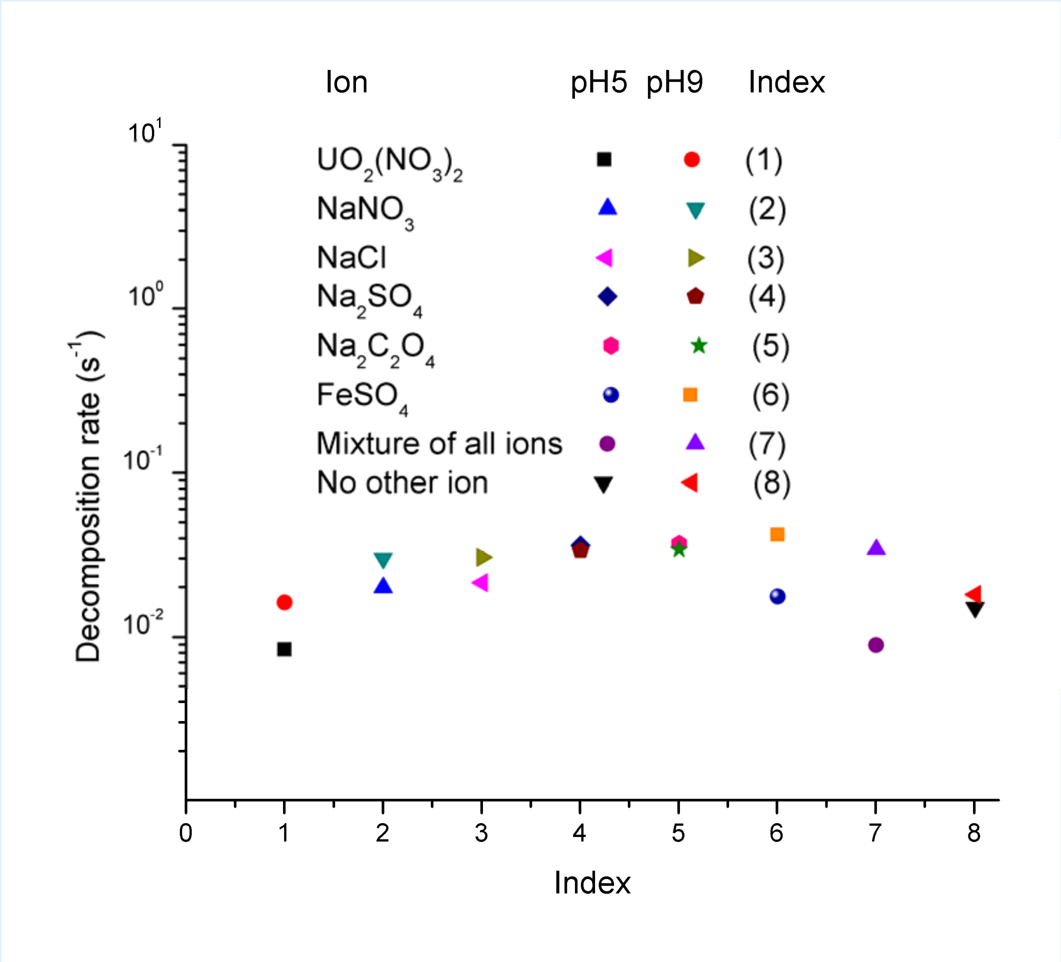
Reaction mechanism	Enhancement factor in reaction rate with respect to that at 25°C or 80°C	
	25 °C	80 °C
No additives (hydrolysis alone)	1	$1.4 \cdot 10^3$
Radiolysis + Hydrolysis	$11 \cdot 10^3$	$12 \cdot 10^3$
Hydrolysis + THS alone	$15 \cdot 10^3$	$1200 \cdot 10^3$
Hydrolysis + THS+ALI	$200 \cdot 10^3$	$>2000 \cdot 10^{3*}$
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

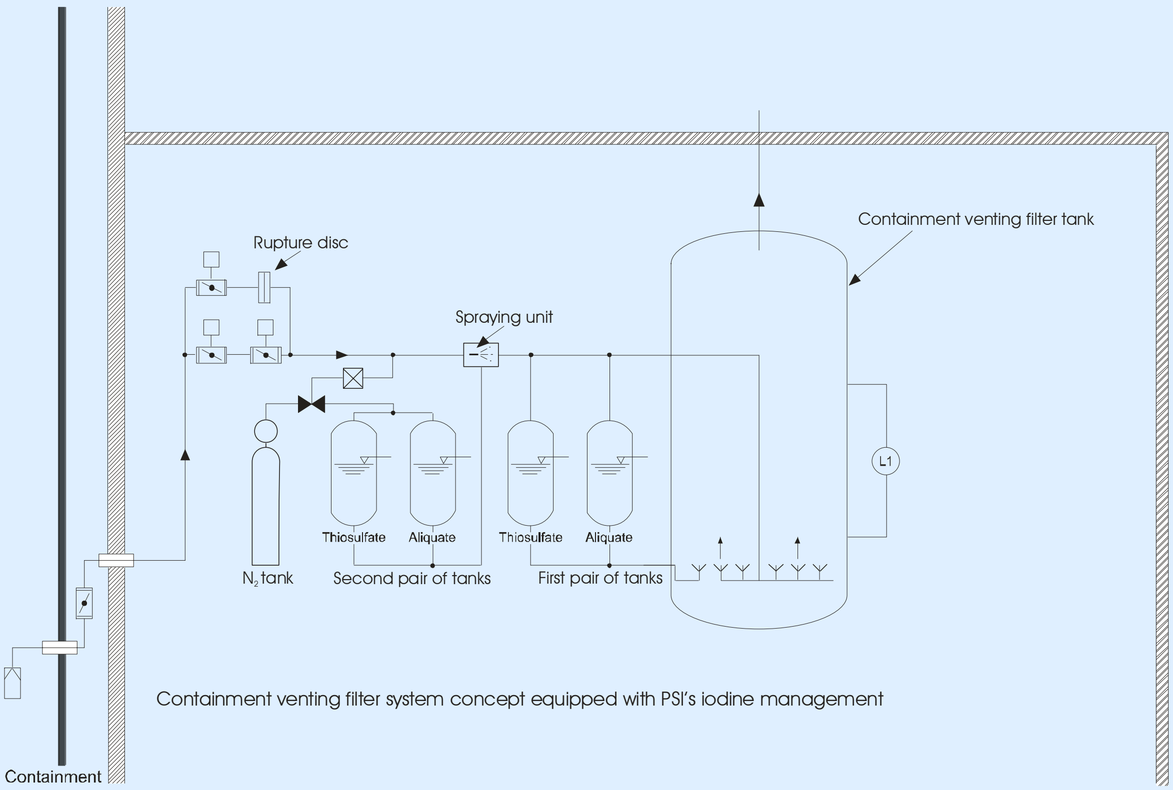
Effective suppression of radiolytic oxidation of iodide ions



No degradation due to presence of other ions



Implementation of PSI process in NPP containment venting filter systems



Containment venting filter system concept equipped with PSI's iodine management

- Independent of NPP systems
- Easy implementation of required tanks and valves
- Passive operation

Anticipated global safety benefits

The exact global safety benefit by implementing the novel system developed at PSI for the iodine management as a part of the containment venting system:

- depends 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 7.10^{-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 iodine source term to the environment will be by a factor of several 1000.

Therefore, achievement of 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.

Conclusions (1)

- 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.

Conclusions (2)

- Over 1000 tests demonstrated fulfillment of all requirements preset:
  - 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 flowing gas into the aqueous phase during a sparging application such as in a containment venting filter operation.
- Feasibility of engineering of implementation of the process for back-fitting existing wet containment venting filters or implementation in a new containment venting filter system prepared.
- Safety benefit of implementing the PSI novel system for iodine management during containment filtered venting clearly shown.

Thank you for your attention

Leibstadt Nuclear Power Plant

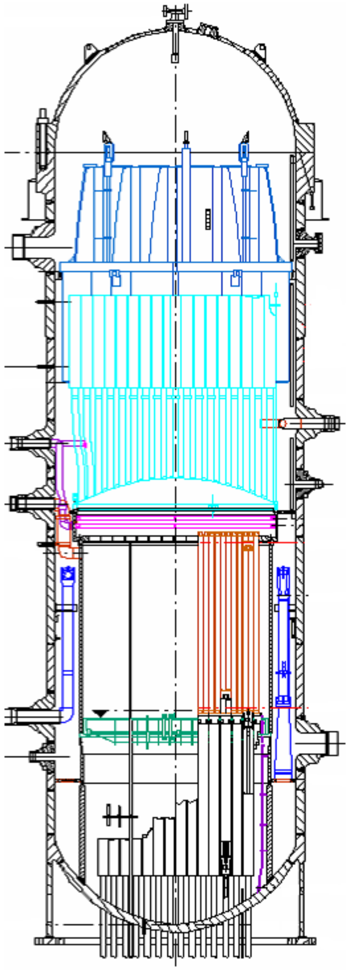
Development of Severe Accident Management Guidelines for Shutdown Conditions (SSAMG)

Wolfgang Hoesel, Peter Keller

Leibstadt Nuclear Power Plant

Leibstadt BWR-6 Technical Data

Reactor Type:	GE BWR-6
Power:	3600 MWth / 1165 MWel
Initial Startup:	1984
Recirculation:	2 external pumps 20 internal jet pumps
Total Core Flow:	11151 kg/s
Control Rods:	149
Fuel:	648 bundles, 10x10
	113.5 t (uranium)



Leibstadt BWR-6/Mark III Containment

(1) Reactor Vessel

(2) Drywell

(3) Suppression Pool

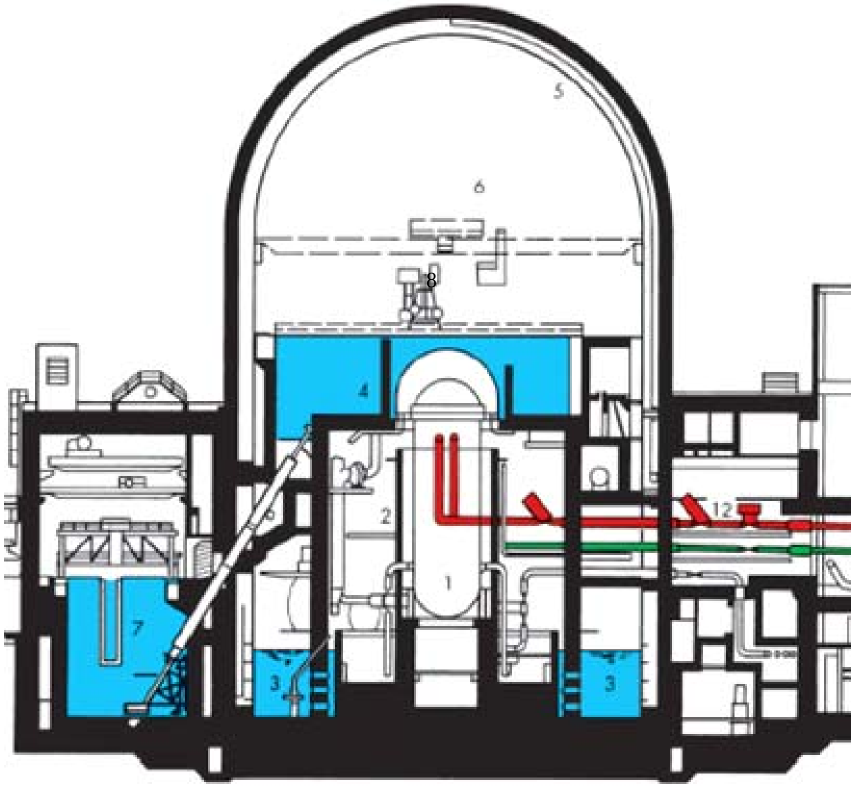
(4) Upper Containment Pool

(5) Containment

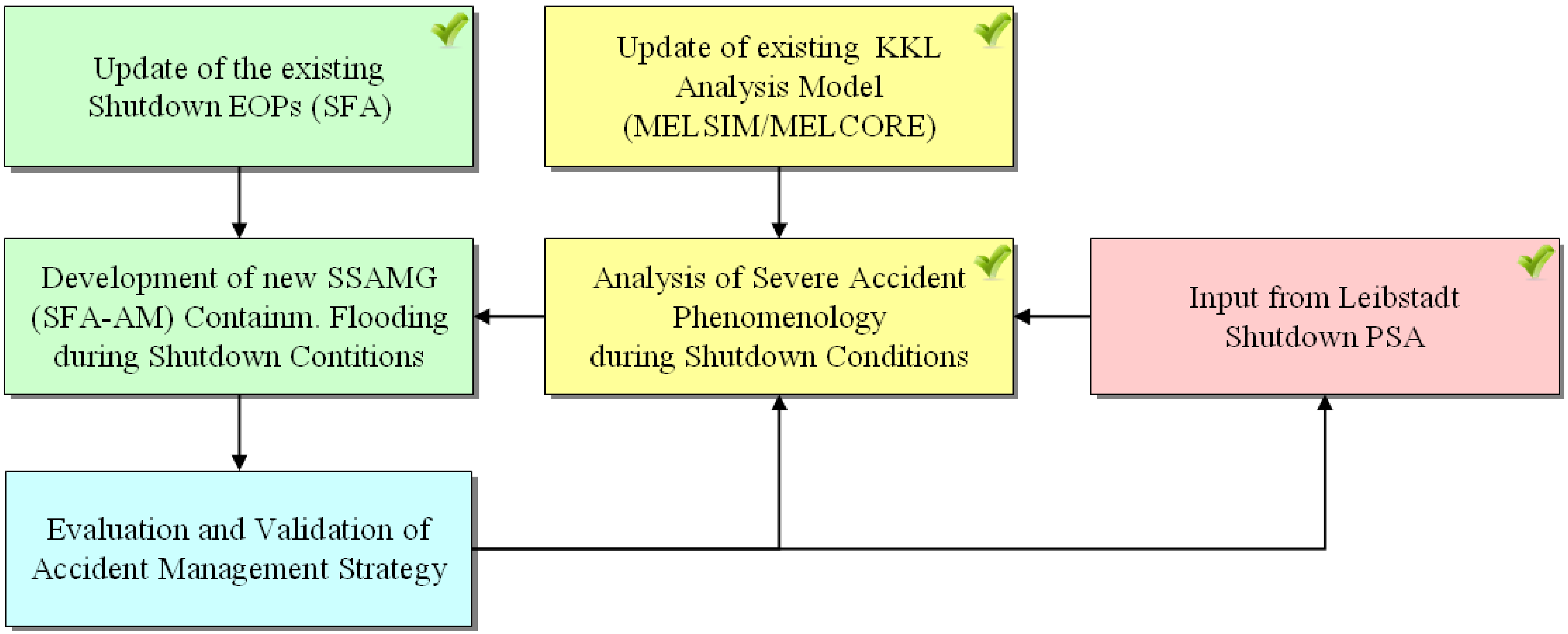
(6) Polar Crane

(7) Fuel Storage Pool

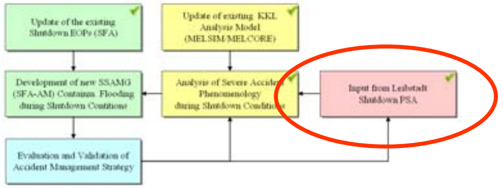
(8) Refueling Machine



SSAMG Development Strategy



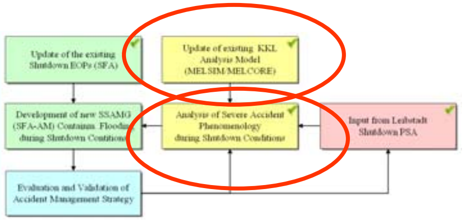
Leibstadt NPP Shutdown PSA Results



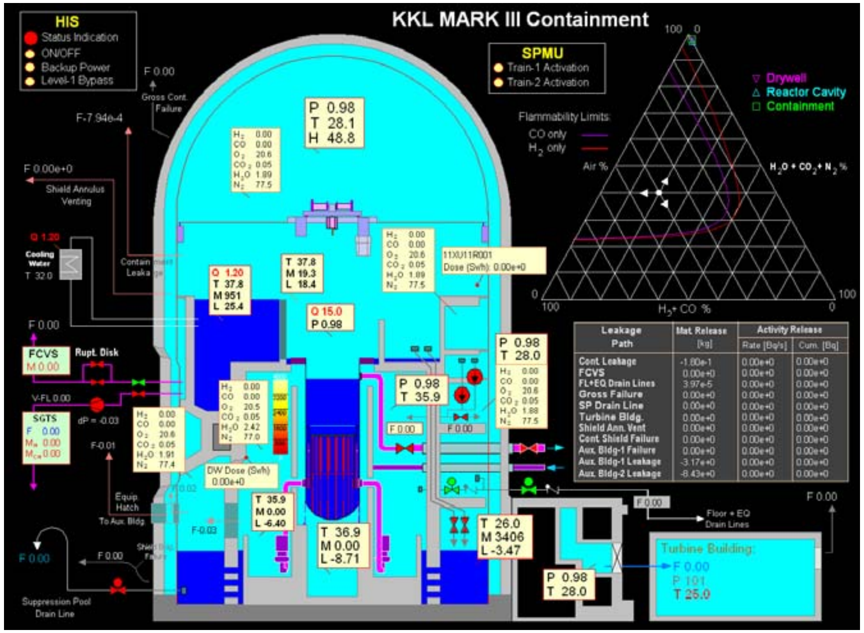
Event Type	Description	CDF At-Power	FDF Shutdown	Overall CDF
Internal Events	All LOCA Events	1.04E-07	3.34E-08	1.37E-07
	Transients and special initiators	3.26E-07	5.92E-09	3.32E-07
	Total:	**4.30E-07**	**3.94E-08**	**4.69E-07**
External Events	Earthquakes	2.14E-06	3.01E-07	2.44E-06
	High winds and tornadoes	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**

Of the overall Core Damage Frequency (CDF) of 5.24E-06 per year approximately 25% is contributed at reduced load or shutdown.

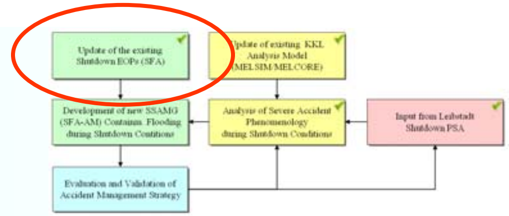
Analysis of Leibstadt NPP Shutdown Scenarios



- Update of the existing At-Power model for shutdown conditions
- The accident scenarios initiated during shutdown were analysed using the new MELCOR 1.8.6 based Leibstadt Shutdown Model
- Evaluation of the behaviour and timing of selected sequences in order to determine the time available for corrective actions



Update of the existing Shutdown EOPs (SFA)

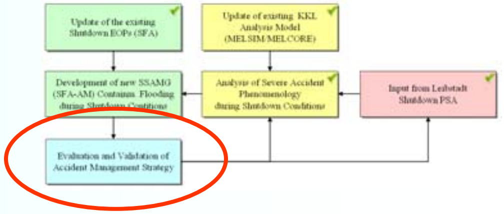


Shutdown EOPs (SFA) (prevention)
SFA-1704-43 Loss of Shutdown Cooling (RHR/SEHR) during Shutdown
SFA-1704-44 Loss of Coolant during Shutdown
SFA-1704-46 Loss of Power Supply during Shutdown

Shutdown SAMG (SFA-AM) (mitigation)
SFA-1704-AM02 Reactor Vessel and Containment Flooding during Refueling and Shutdown Conditions

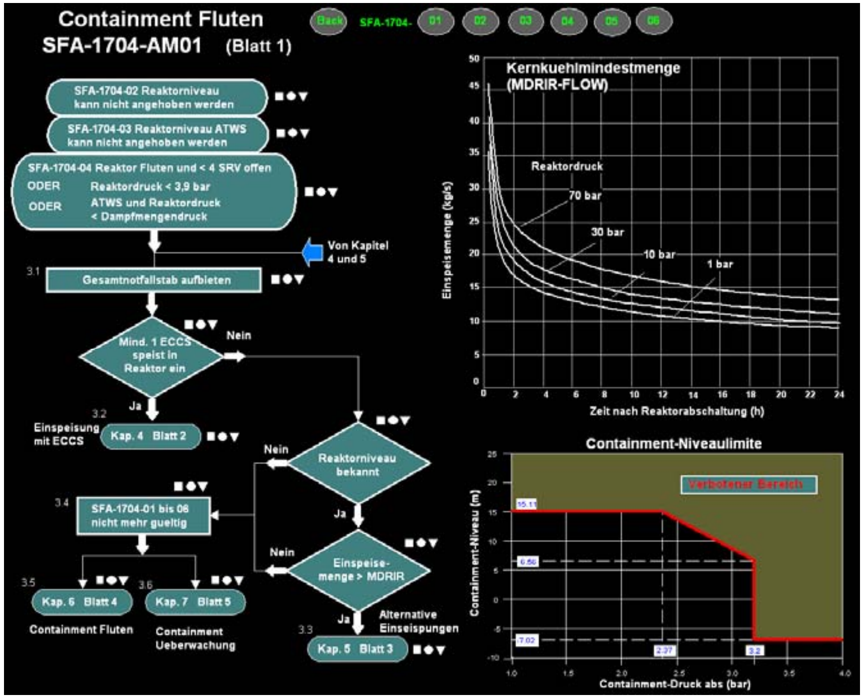
- Prior to the development of the new Shutdown SAMG the already existing Shutdown EOPs were revised and optimized
- While the objective of the EOPs is to prevent a potential severe accident condition, the objective of Shutdown SAMGs is to mitigate core melting and the effects of a vessel break through

Verification and Validation of the Leibstadt NPP Shutdown SAMG

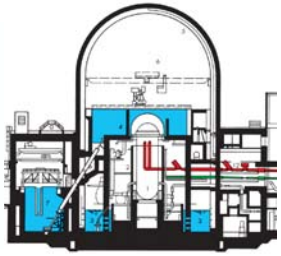


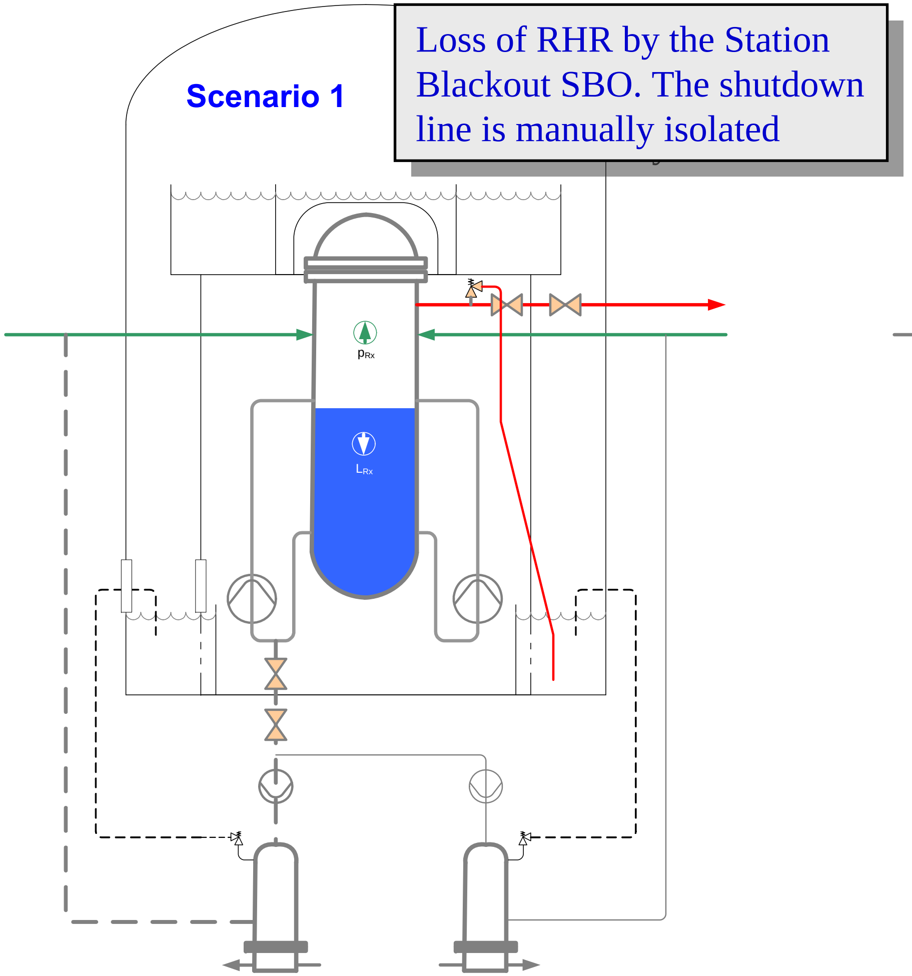
- The effectiveness of the SSAMG corrective actions will be verified using the new MELSIM shutdown model

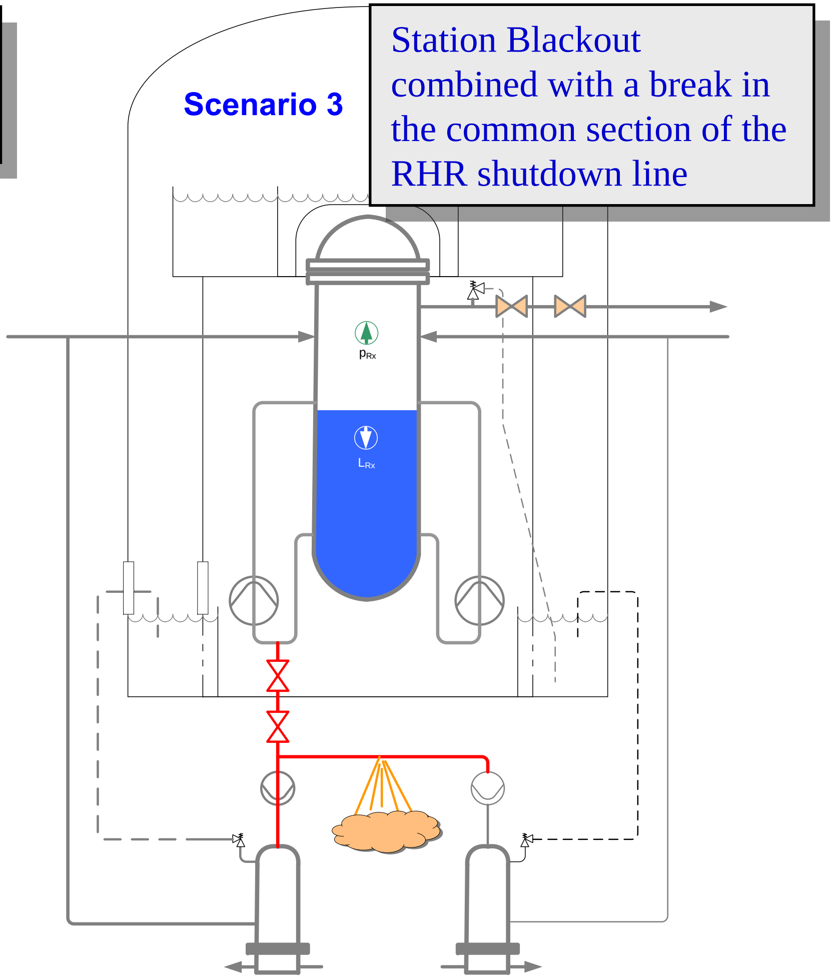
- Following a successful verification, the introduced accident mitigation measures will be validated with the Leibstadt Shutdown PSA model



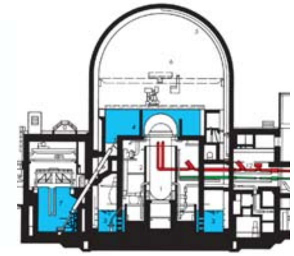
Analyzed Scenarios – Vessel Closed

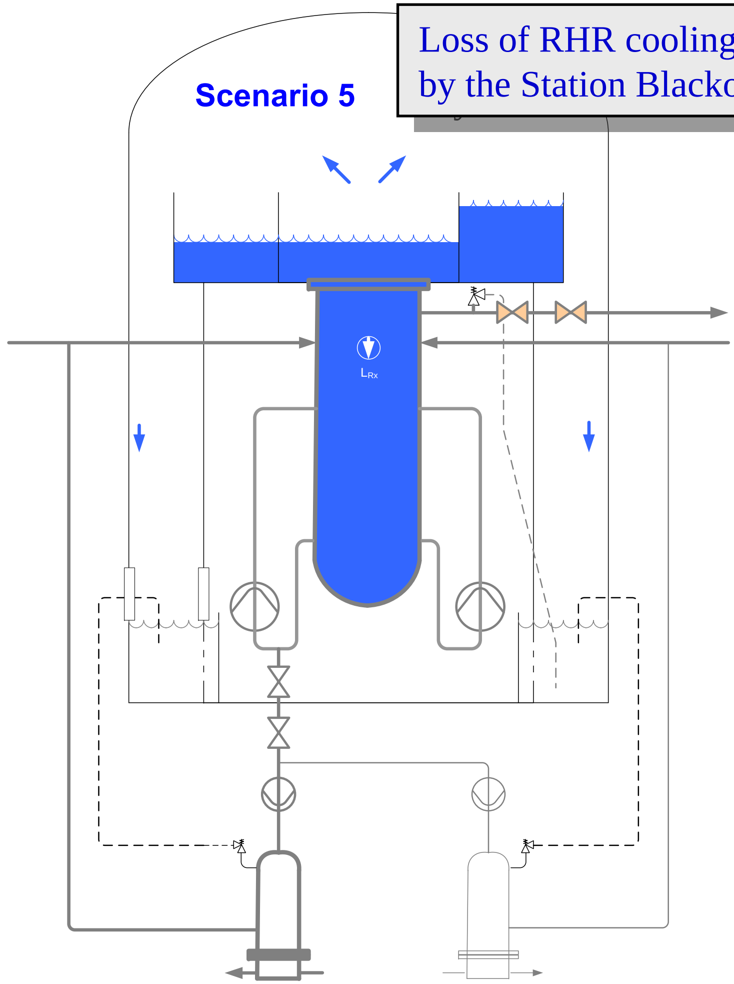




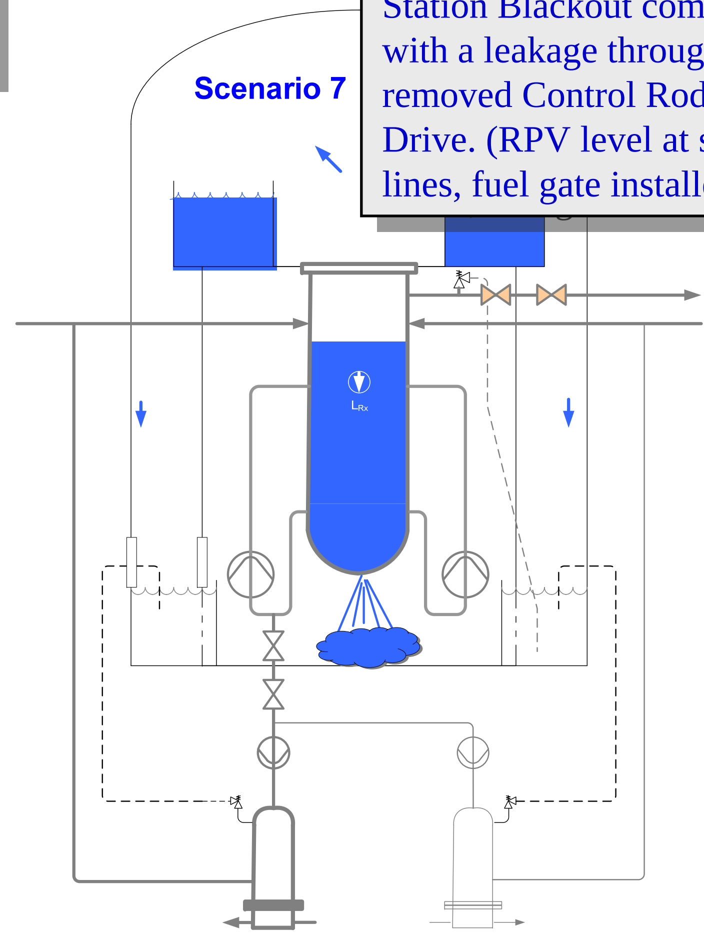


Analyzed Scenarios – Vessel Open



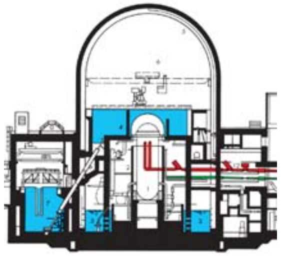


Loss of RHR cooling caused by the Station Blackout



Station Blackout combined with a leakage through a removed Control Rod Drive. (RPV level at steam lines, fuel gate installed)

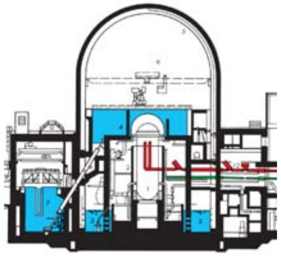
Timing of the Key Core Damage Related Events



Scenario	1	2	3	4	5	6	7	8
	SBO, RHR isolated	SBO, RHR not isolated	SBO, RHR Leak, not isolated	SBO, RHR Leak, isolated -30'	SBO	SBO, CRD Leak	SBO, Cavity drained, CRD Leak	SBO, RWCU leak
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

- The time to core uncovery 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 short core uncovery times have a significant loss of coolant inventory due to a break or leak.

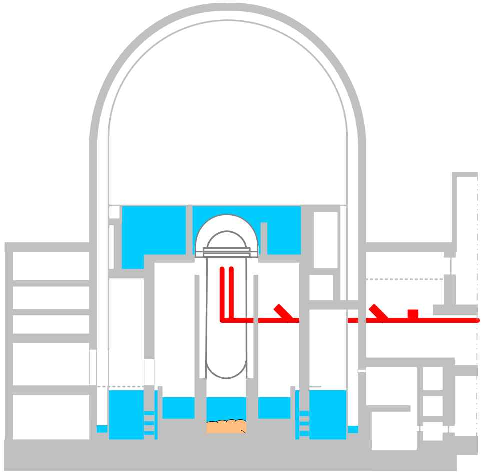
Insights gained from the Shutdown Scenario Analysis (1)

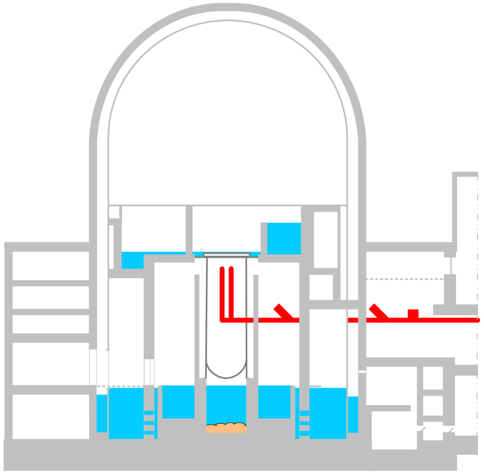


- All analyzed cases lead to severe core damage, with relocation of debris to the lower plenum, progressing to vessel breach and ejection of debris to the Reactor Cavity.

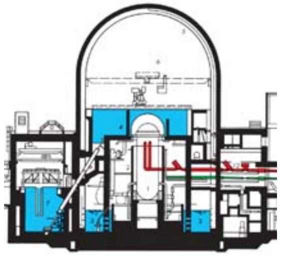
- The RPV pressure remains low in all cases with RV open to the Containment. In the other cases the RPV pressure increases up to the opening pressure of the SRVs

- The radionuclide release to the environment is large for Noble Gases and for the Aerosols as well.

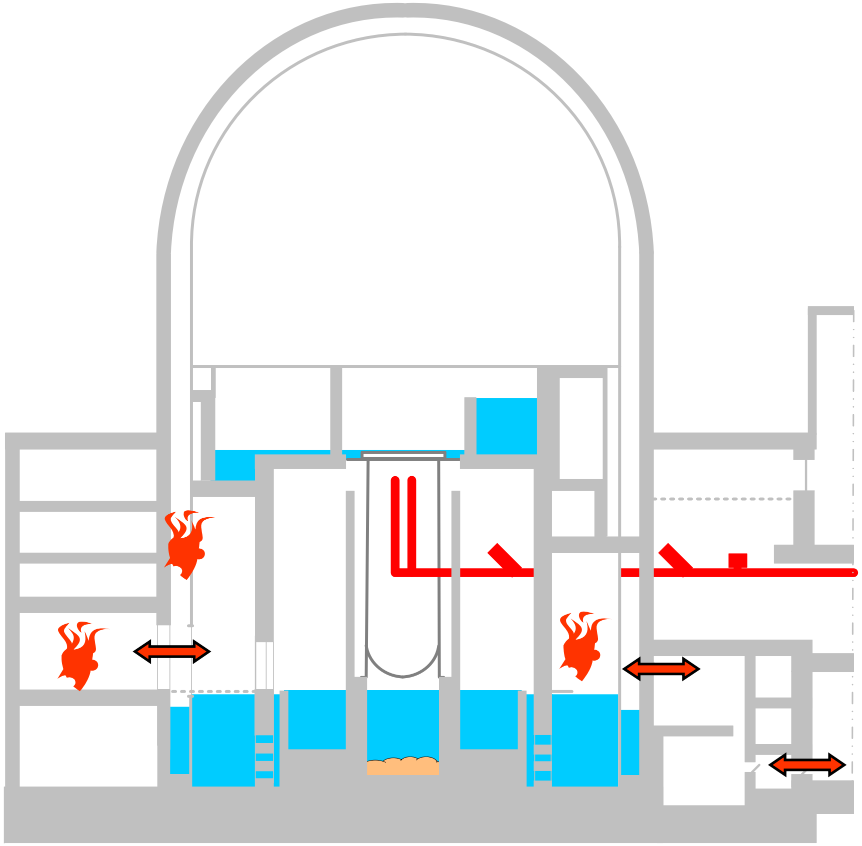




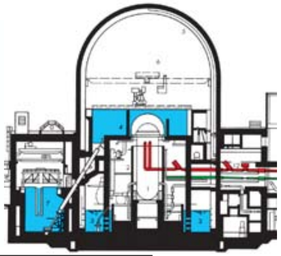
Insights gained from the Shutdown Scenario Analysis (2)



- The containment pressure remains low, because the Equipment Hatch is open to the Annulus and to the Secondary Containment
- A sizeable venting path from the Annulus to the environment opens early during the Containment pressurization
- Combustion of the hydrogen and CO occurs in the rooms of the Containment, the Annulus and the Secondary Containment



The Barrier Integrity defines the Scope of the Shutdown SAMG



Operating Modes	Reactor Temp.	Reactor Cavity flooded	RPV Integrity	Drywell Integrity	Contain-ment Integrity	Fuel above TAF	Gates Open
4	< 93 °C	F	intact	intact	intact	--	yes
4	< 93 °C	--	intact	intact	open	--	no
4	< 93 °C	--	intact	open	open	--	no
5	< 60 °C	--	open	open	open	--	no
5	< 60 °C	F	open	open	open	--	no
5	< 60 °C	F	open	open	open	yes/no	yes

- 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 apply.**

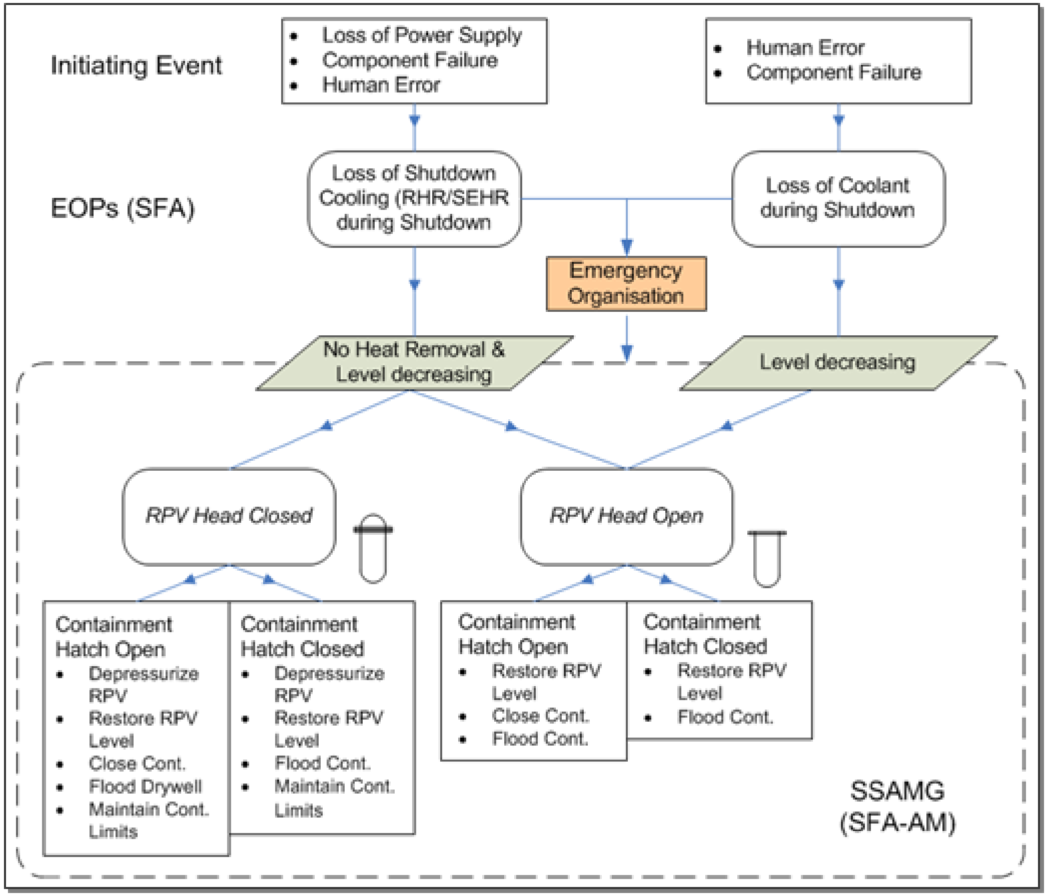
Event Handling Strategy

As during power operation, containment flooding remains the basic strategy to cope with core melting scenarios.

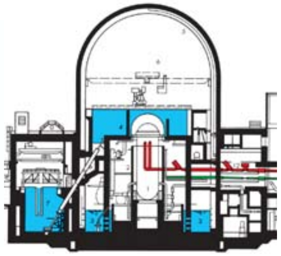
The objectives of primary containment flooding are consequently identical.

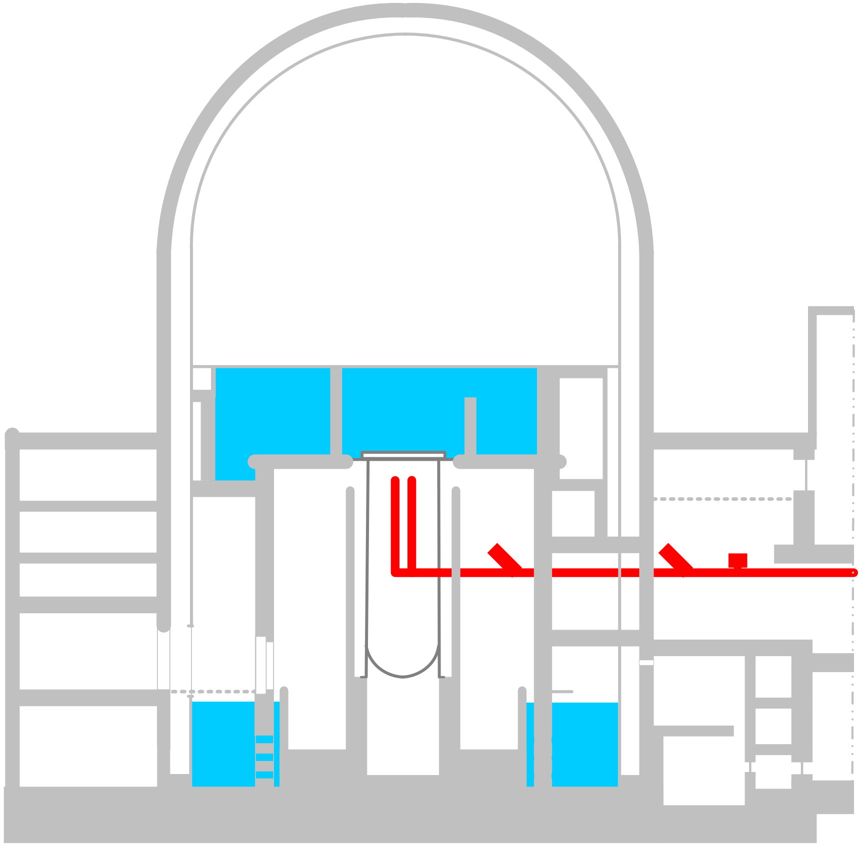
Entry conditions for Shutdown SAMG are more restrictive than during power operation.

→ **Need to re-establish containment integrity!**

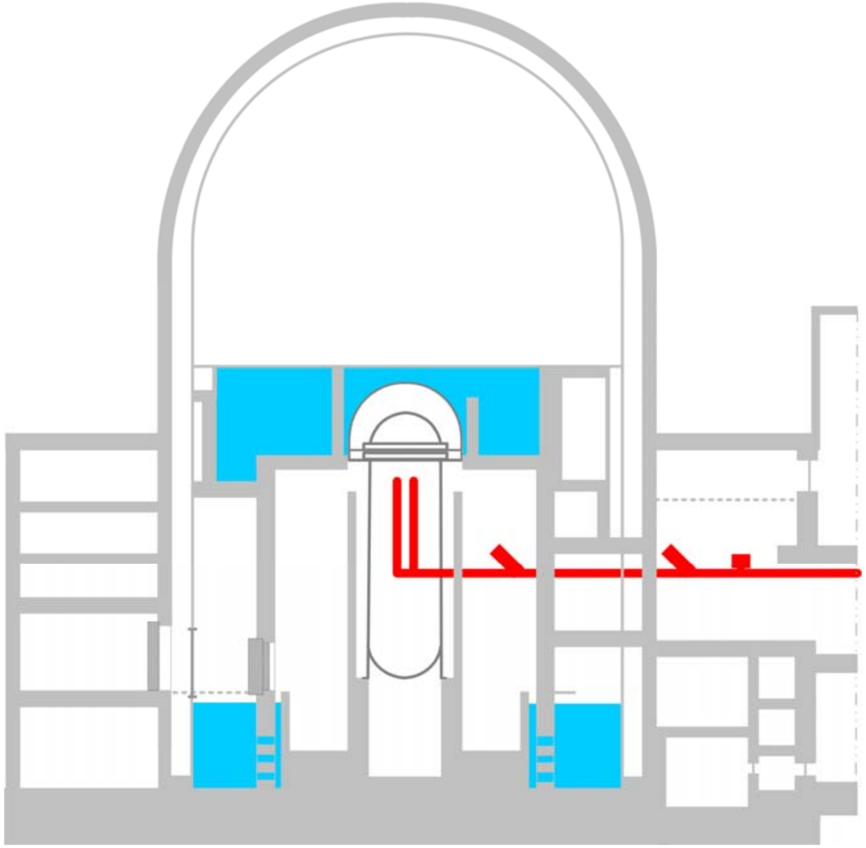


Mark III Containment Integrity Status



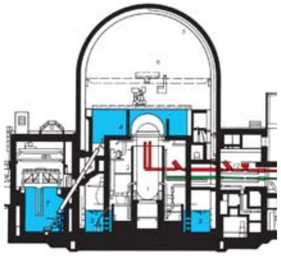


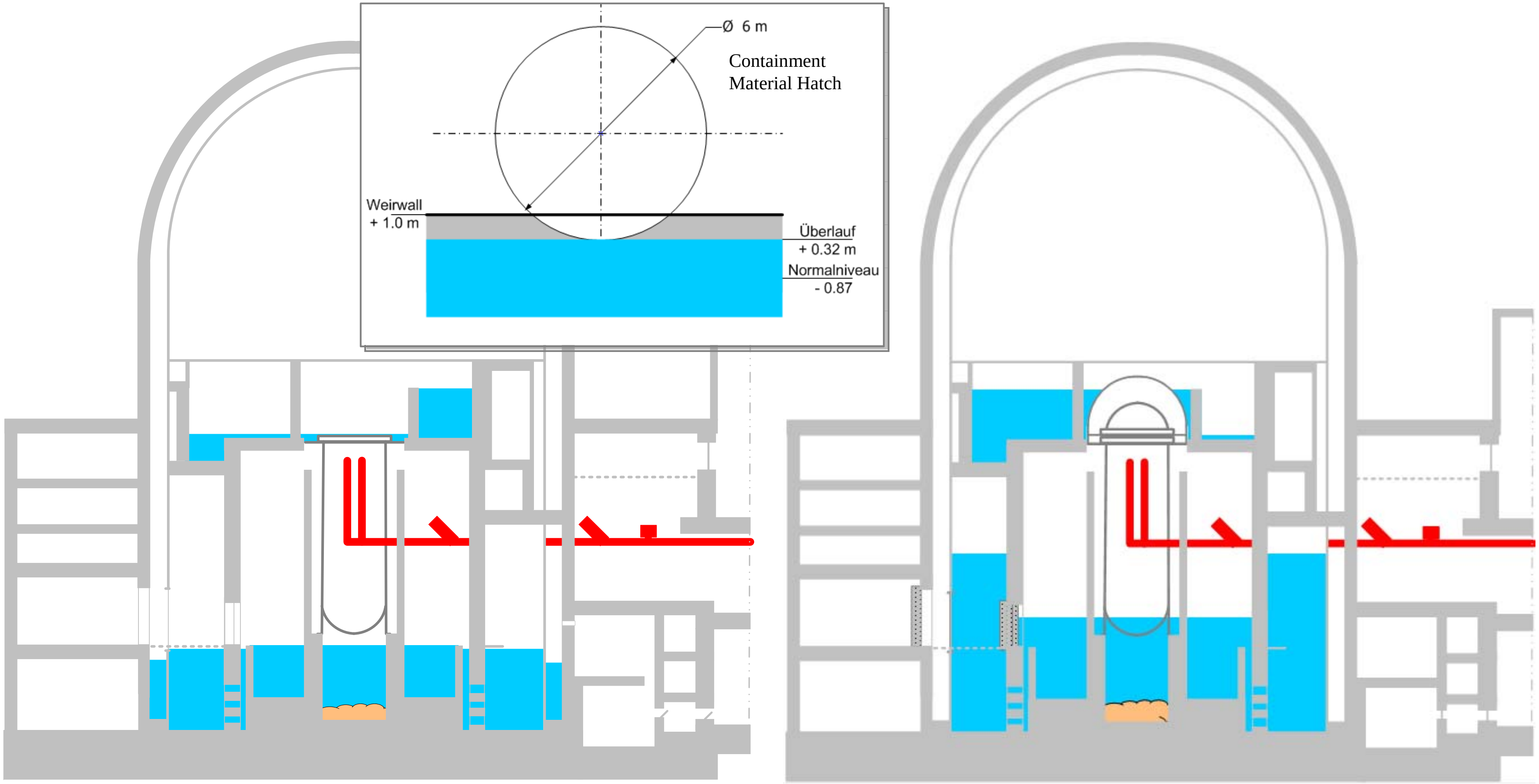
Hatch open



Hatch closed

Mark III Containment Flooding Limitations



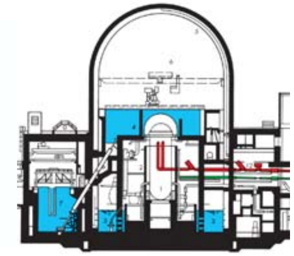


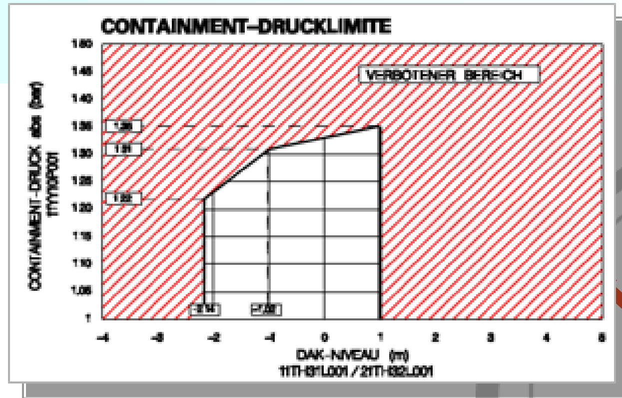
Hatch open

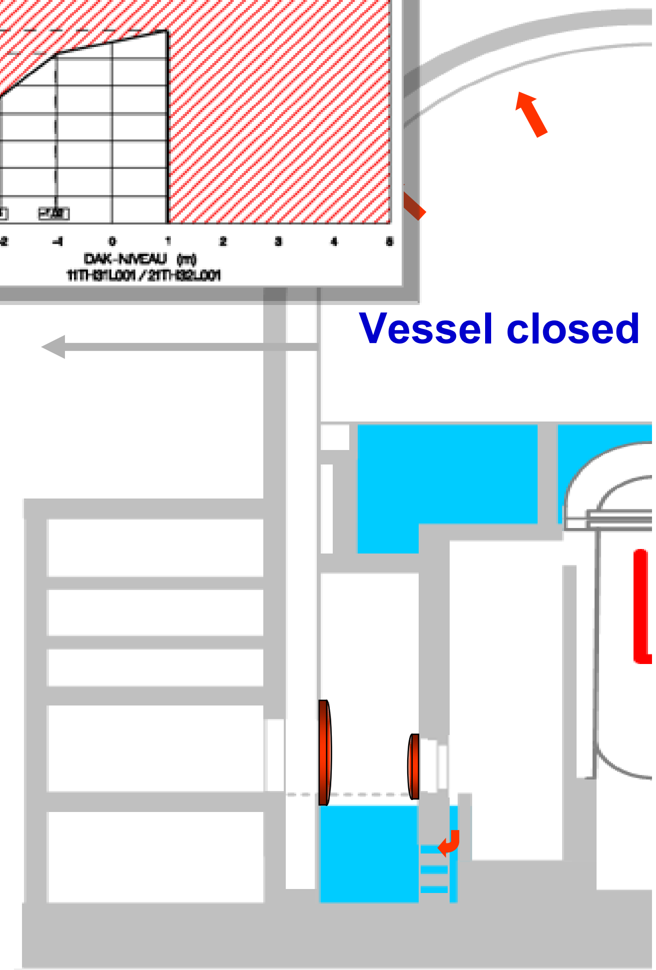
Hatch closed

Containment Recovery Strategy
Vessel Closed

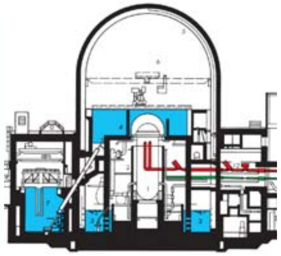
- Close Drywell Equipment Hatch (without shielding blocs)
- Close Containment Equipment Hatch
- Verify secondary containment integrity (all doors closed)
- Establish primary containment integrity to the extend possible
- Establish drywell integrity
- Stay within the Containment Pressure Limits
    - ➢ Containment Venting



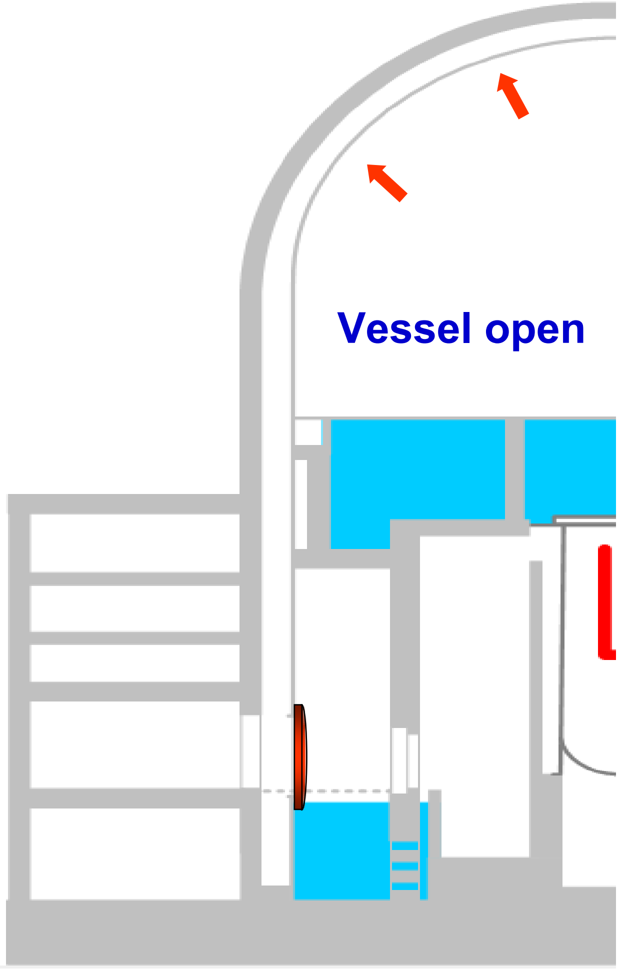




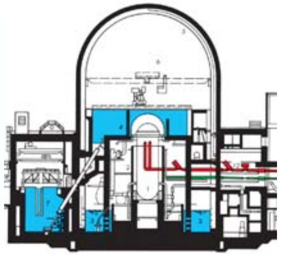
Containment Recovery Strategy Vessel Open



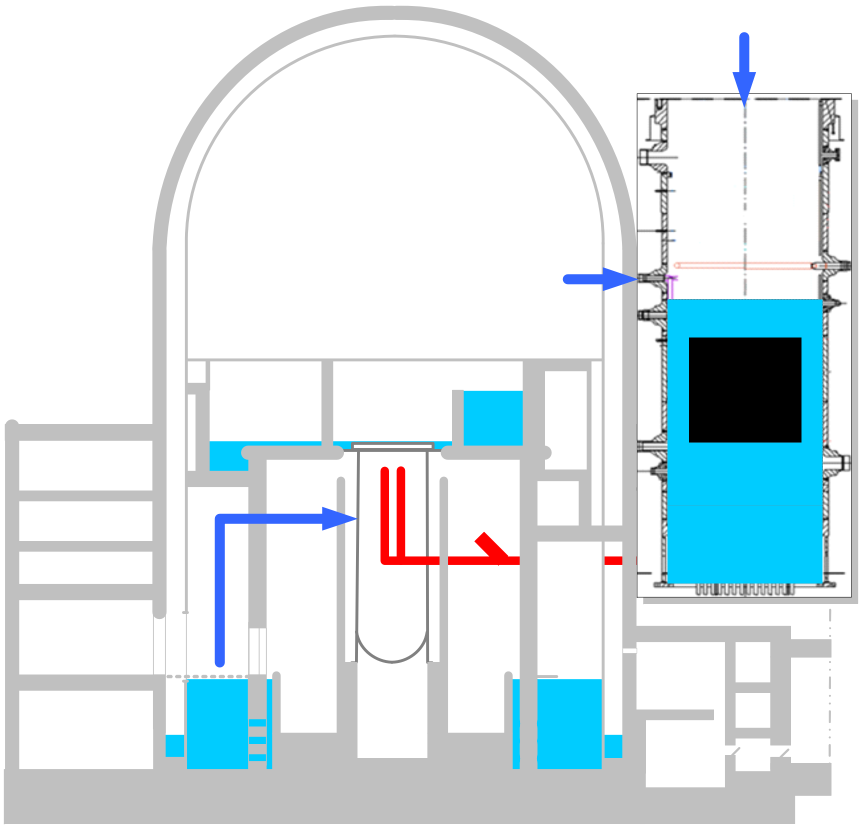
- Close Containment Equipment Hatch
- Verify secondary containment integrity
- Establish primary containment integrity to the extend possible
  - close at least one (inboard or outboard) isolation valve
- close test valves between isolation Valves



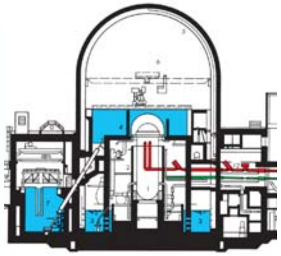
Level can be restored and maintained above Top of Active Fuel (TAF)



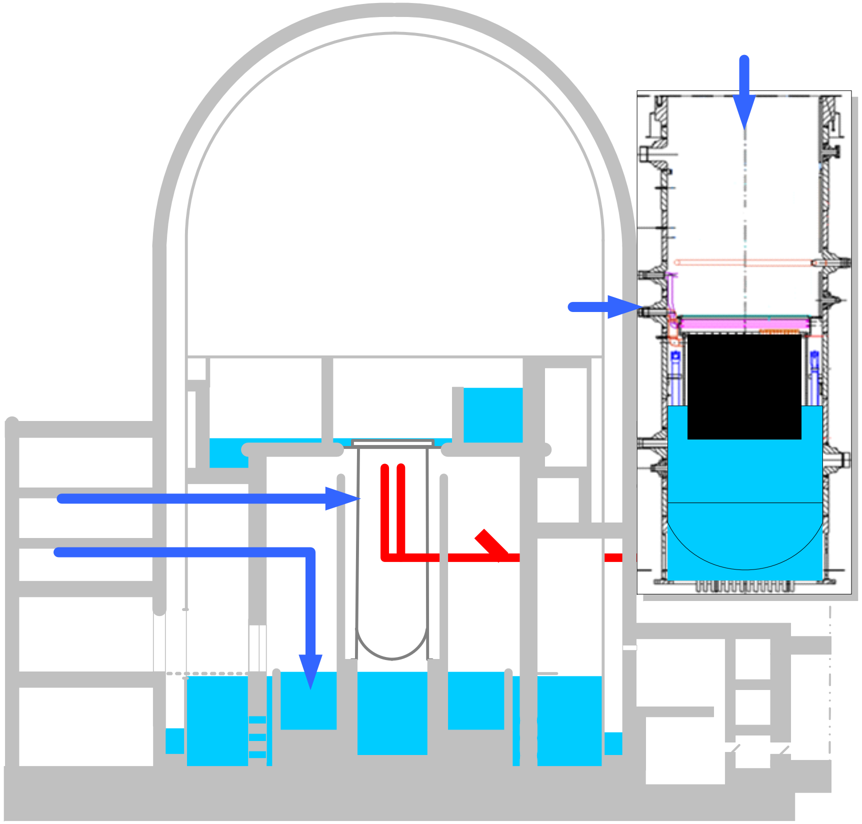
- Restore and maintain RPV water level > TAF
- Use external sources and in-shroud injection only if required
- Limit containment water level to -32 cm if Containment Hatch is not installed
- Re-establish containment integrity
- Cool Suppression Pool



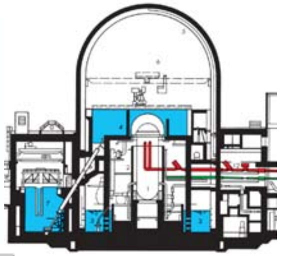
Level can be restored and maintained above bottom of active fuel (BAF)



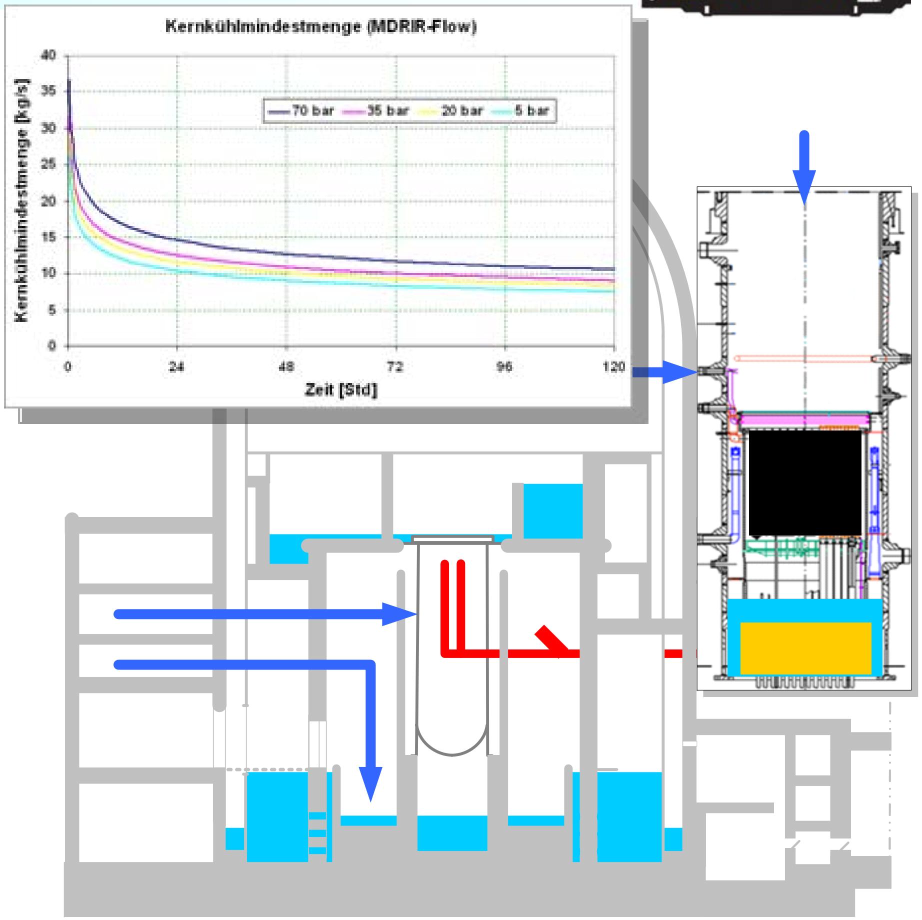
- Debris expected to remain in RPV
- Restore and maintain water level > BAF
- Priorities:
    1. Operate core spray
    2. Maximize injection of external sources
- Re-establish containment integrity
- Restore essential systems
- Flood drywell to the Minimum Debris Submergence Level (above the top of the weir wall or at least 1.5 m)
- Limit containment water level to -32 cm if containment hatch is not installed
- Cool Suppression Pool



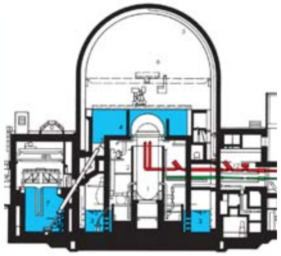
Injection can be restored and maintained above the Minimum Debris Retention Injection Rate



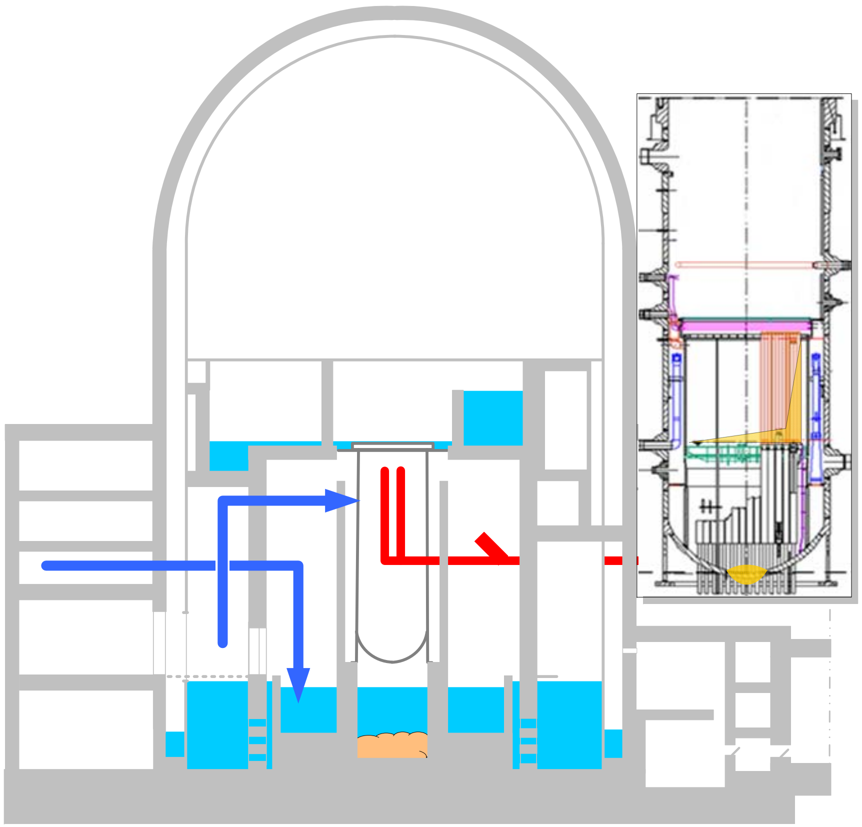
- Debris expected to remain in RPV
- Restore and maintain water injection > MRDIR
- Maximise injection of external sources to the RPV
- Re-establish containment integrity
- Restore essential systems
- Flood drywell to the Minimum Debris Submergence Level (above the top of the weir wall or at least 1.5 m)
- Cool Suppression Pool



Core debris has breached the RPV



- Pressure suppression no longer required
- Flood drywell/containment at least to the Minimum Debris Submergence Level (between 1.5 m 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 (Cont. Hatch closed)
  3. Maximize RPV injection from suppression pool (Containment Hatch open)
  4. Cool Suppression Pool

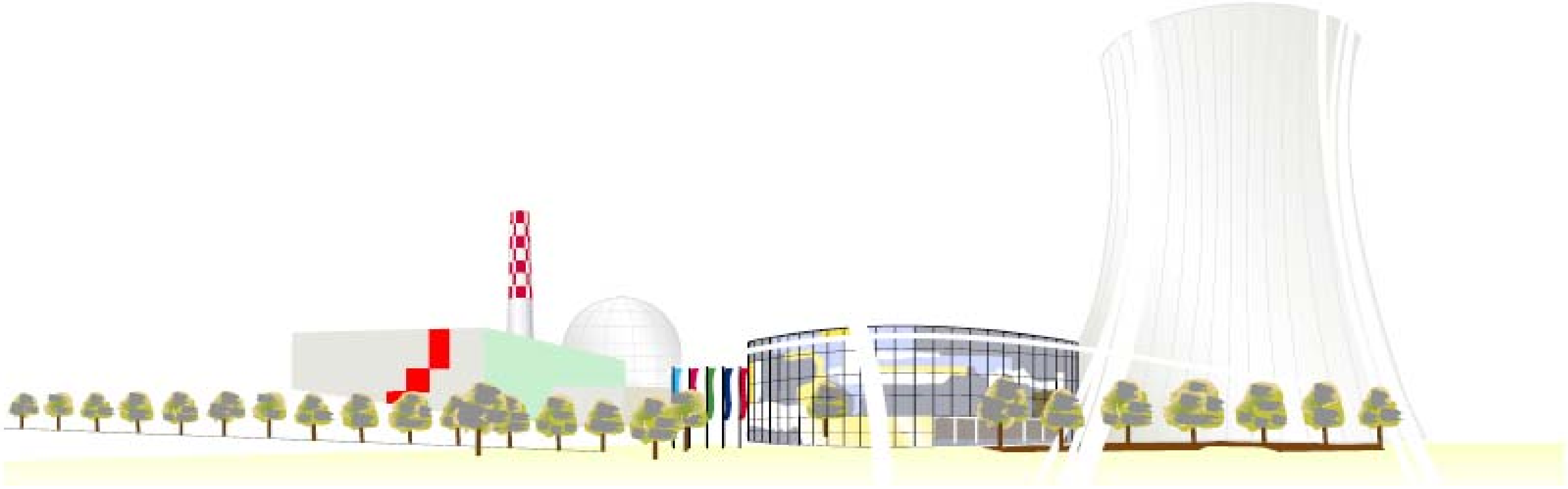


Conclusions

- The shutdown specific risks are identified
- 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
- Depending on the current maintenance schedule and operational readiness, unavailable systems need to be restored

THANK YOU FOR YOUR ATTENTION

QUESTIONS?



Design Modifications of the Mochovce Units 3 & 4 Dedicated to Mitigation of Severe Accident Consequences,

Providing Conditions for Effective SAM

OECD/NEA Workshop

Implementation of Severe Accident Management Measures (ISAMM-2009)

October 2009

Milan CVAN
VUJE, Inc., Slovakia

Dušan Šiko
SE, a.s., Slovakia

Introduction

Initiation of activities dedicated to enhancement of Slovak nuclear units regarding severe accident mitigation **is dated to around 2005**

Originally intended for units in operation, with draft SAMG already available

Since decision to continue in construction of Mochovce 3 & 4 units of VVER440/V213, these units have been the priority

The complex process started with identification of deficiencies,
through initial proposal of „ideal“ structure and extent of modifications,
seeking an optimum for all involved parties and views, **up to basic design**

Detail design is being developed, plant operation scheduled to late 2012.

Starting point

Large database of diverse severe accident scenarios

(Phare4.2.7.a, PSA level 2, SAMG development support)

Experience from development of SAMGs for units in operation

(performed by Westinghouse, with intensive contribution of Slovak specialists, including analytical support)

Both PSA 1st and 2nd level for units in operation available

No specific requirements of the Slovak Regulatory Authority

IAEA/EUR/WENRA general requirements

Limitations from already constructed buildings and structures

Initial proposal

1. **Interruption of core degradation and relocation focused to:**
    - Severe accidents by open reactor
    - Severe accidents in spent fuel pool
    - Isolation of open reactor or spent fuel pool in severe accident conditions
2. **Reliable indication of severe accident conditions and initiation of reactor cavity flooding**
3. **Preservation of reactor pressure vessel integrity by external cooling (core in-vessel retention)**
4. **Management of composition of atmosphere including controlled oxidation and burn of hydrogen inside containment**

Initial proposal

5. Filtered venting of the containment

6. Additional systems for long term heat removal from containment

7. Sufficient inventory of borated coolant for severe accident measures

8. Reliable and fast enough depressurization of primary circuit

9. Monitoring systems dedicated to severe accident control for all phases

10. Prevention of deep subpressure in containment

Management of containment atmosphere

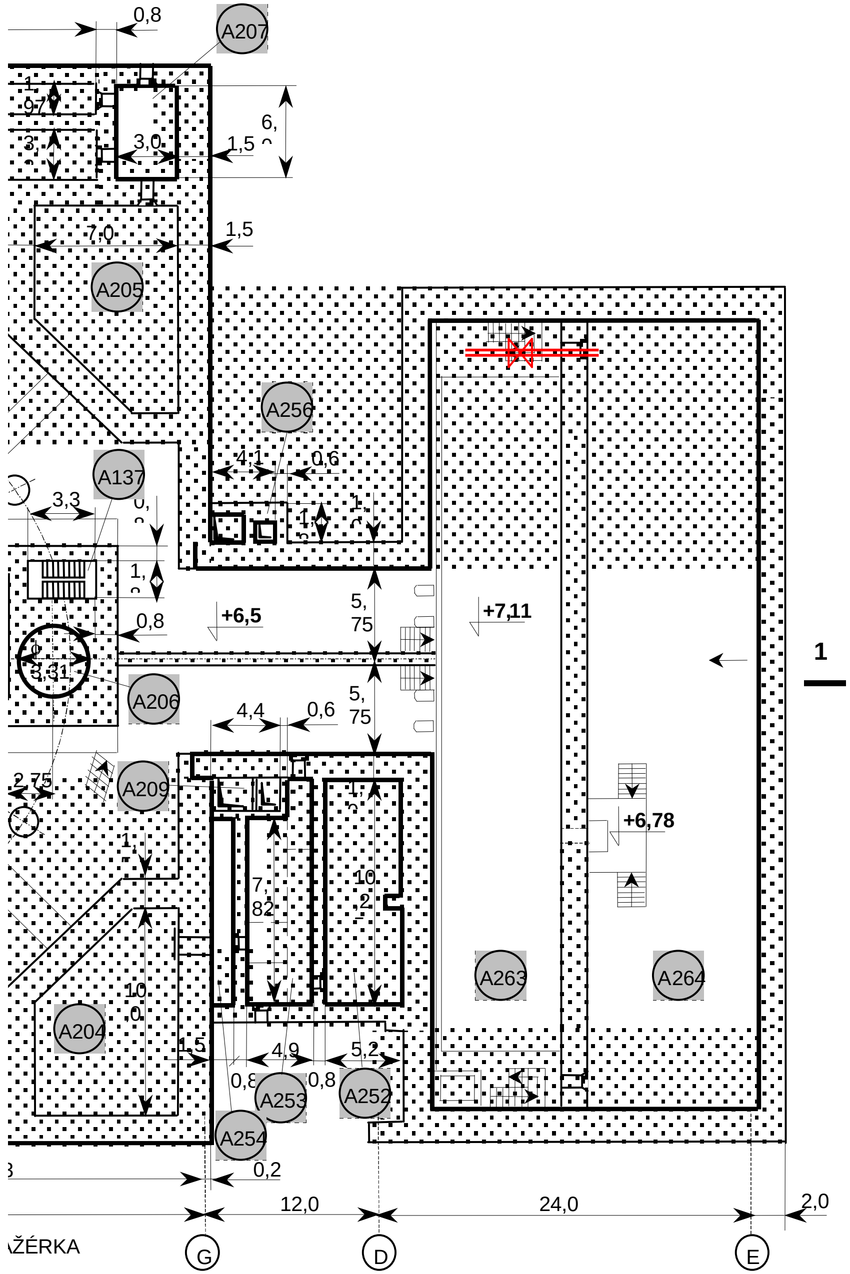
Group of measures to manage hydrogen concentration inside containment

- monitoring of the containment atmosphere composition in selected rooms
- installation of recombiners with severe accident capacity.
- installation of igniters

Vacuum breaker (addition of a system for containment deep subpressure prevention)

- modification of existing pipelines leading from the air traps
- installation of flaps, which will be included in the ESFAS structure
  (after release of locks of the flaps – only passive action of the breaker)

vúje


A207
A205
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+6,78
1
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E
12,0
24,0
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0,8
1,5
7,0
3,0
3,3
4,1
0,6
4,4
5,75
4,9
5,2
2,75
3,31
7,82
10,2
10,0

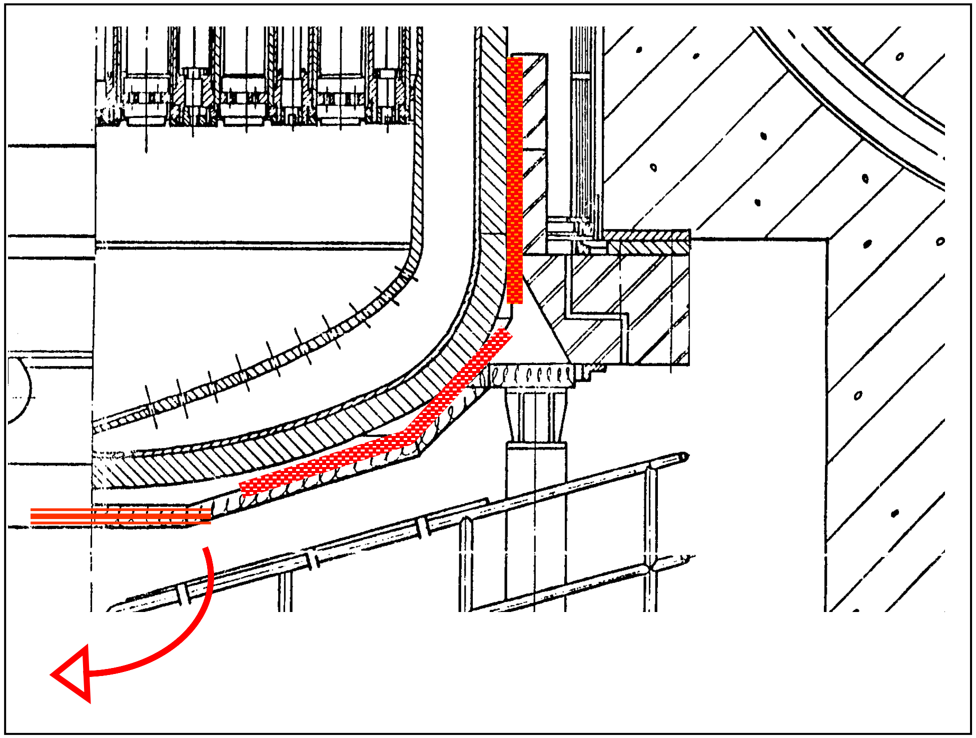
In-vessel retention of corium (1/3)

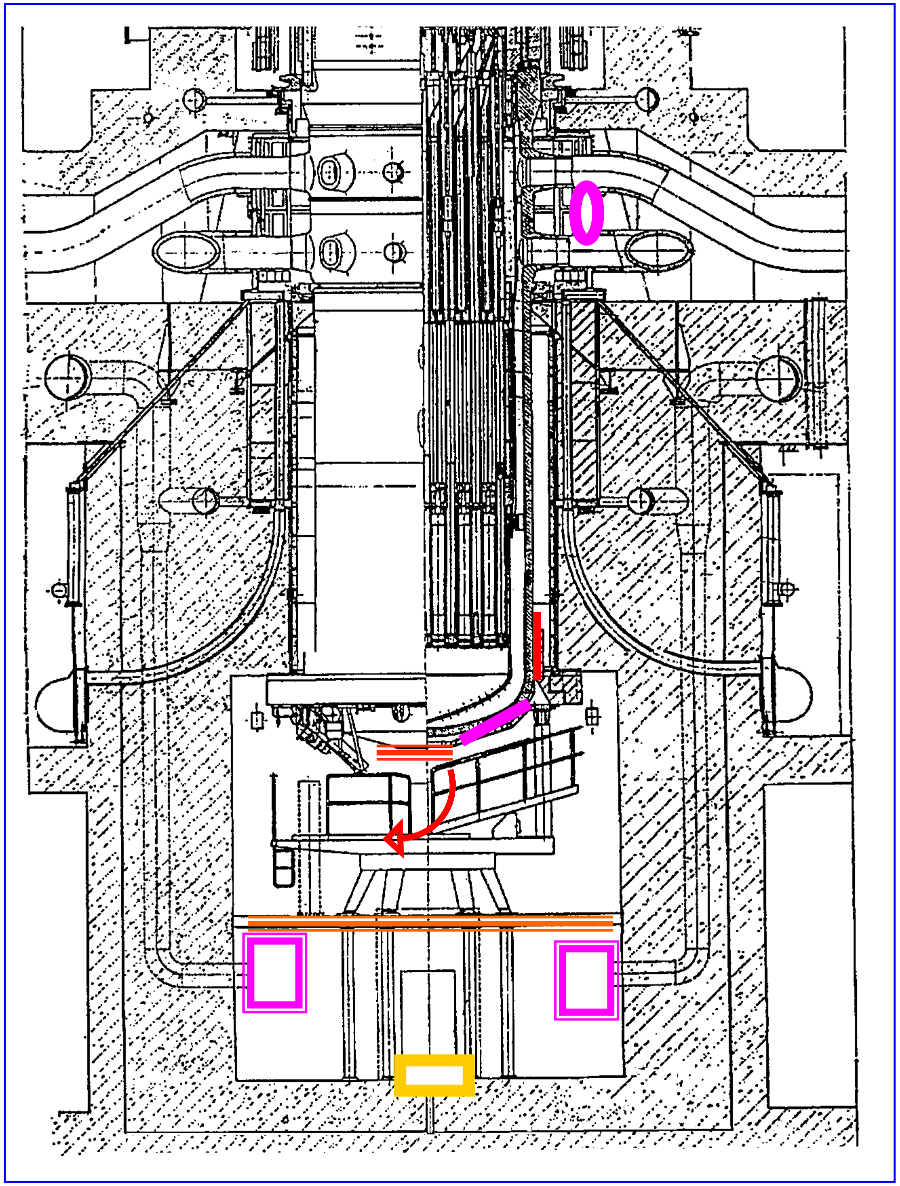
Modification of shielding at the bottom of the reactor pressure vessel

- Enlargement of the gap between RPV wall and bottom shielding structures
- Central opening in the shielding, with buoyancy driven (passive) opening system
- Reinforcement of the shielding for operation with flooded cavity and long term cooling
- Modification of the manipulation platform, to provide free access of coolant to the RPV
- Addition of filtration grid constructions at the inlet of coolant into the reactor cavity
- Modification of penetrations of the reactor cavity
- Modification of the cavity access door and sealing

Corium In-vessel Retention

New design measures





In-vessel retention of corium (2/3)

Sufficient coolant inventory and circulation in the channel along the RPV wall

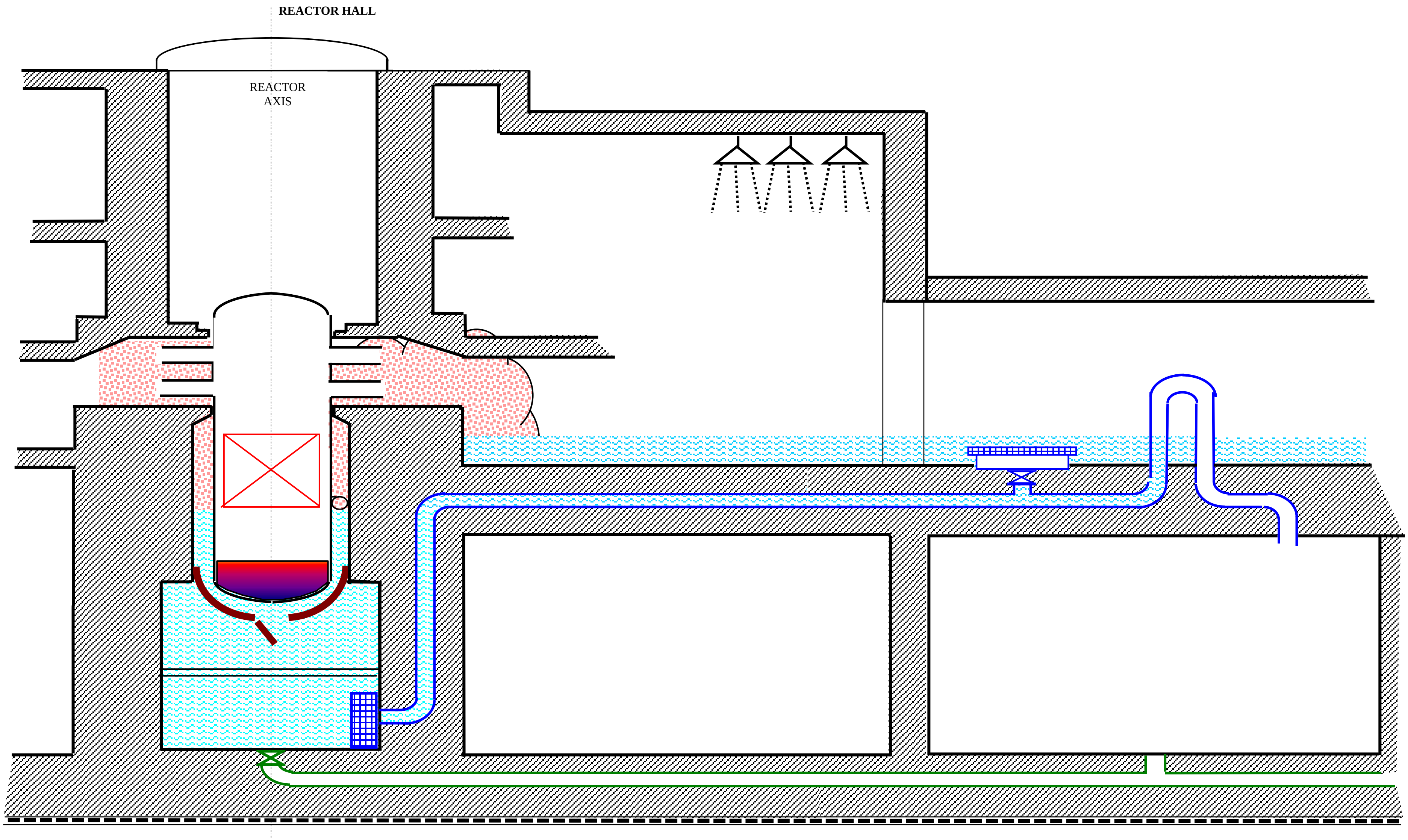
- Modifications of the drain system of the bubble tower trays for drain down capability
- Inlet opening for coolant into the existing ventilation system pipeline below the floor of the connecting corridor, with filtration of impurities
- Installation of closing valve, including control and monitoring
- Installation of U-tube (siphon) at ventilation system pipelines
- Partial reconstruction of the structures around the reactor pressure vessel nozzles

In-vessel retention of corium (3/3)

Modification of the drain line from the reactor cavity

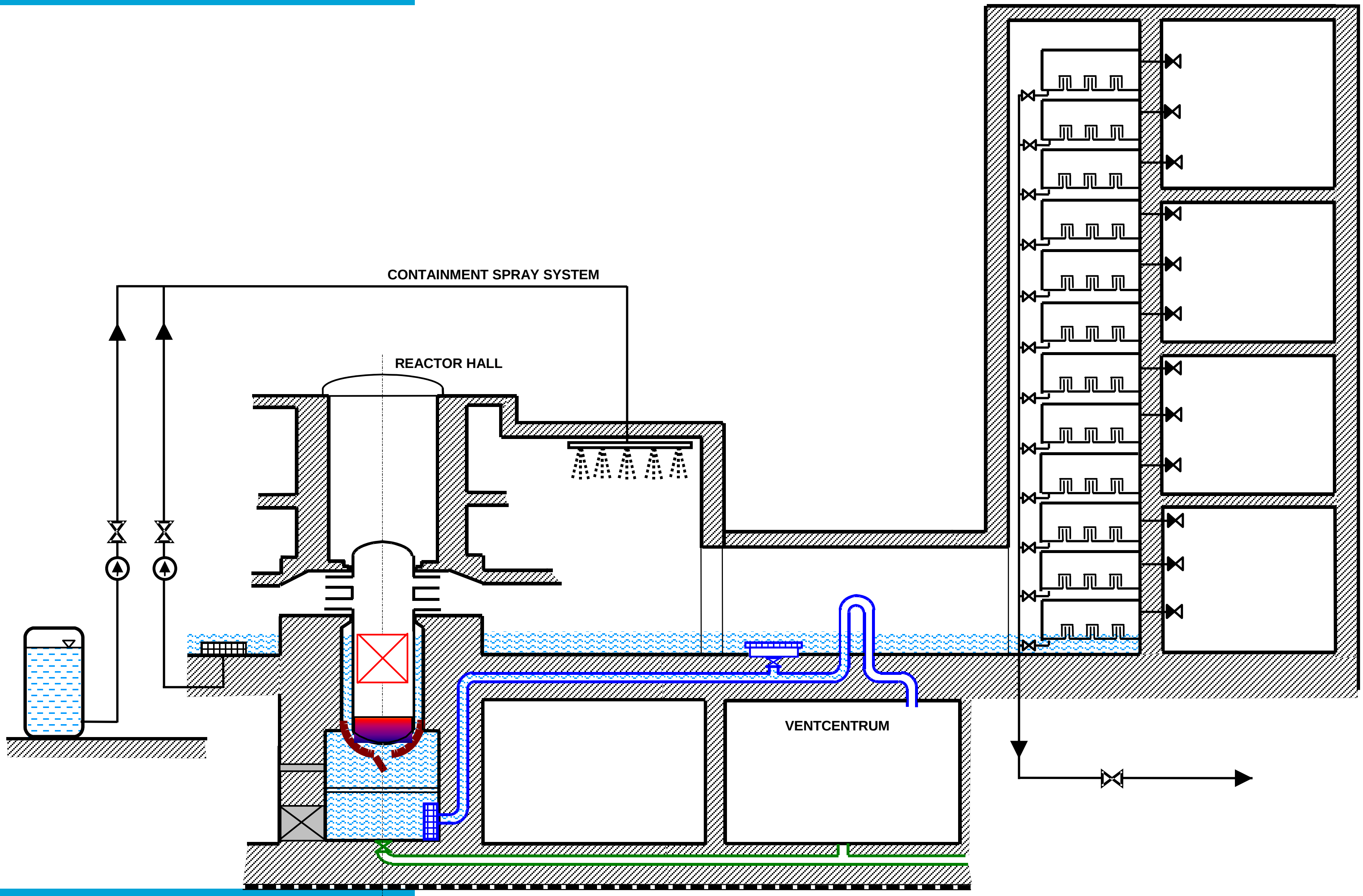
- **Addition of new closing valve inside the reactor cavity at the inlet into the drain line**
- **Installation of control of the valve (from the neighbouring room)**

General layout of the external cooling concept, based on flooding through the venting system pipeline



Corium In-vessel Retention

New design measures



Management of open reactor severe accidents

- 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

- Boric acid solution taken from new External source of coolant

- Installation of necessary pipelines, valves and control of the devices

External source of coolant

- Installation of three tanks outside the containment, common for both units, together with all necessary auxiliary systems

- Installation of appropriate pipelines from the tanks and interconnections to both low pressure ECCS system, to containment spray system and into the pipeline of the spent fuel pool cooling system

- Addition of corresponding valves and their control

Additional measures for mitigation of severe accidents

Controlled depressurization of primary circuit during severe accident

- Additional branch of existing pipeline from pressurizer into the steam generator boxes
- Installation of two closing valves with measurement of pressure between the valves, as well as a drain system

Ultimate heat sink (long term heat removal from containment)

- Modifications limited to procedures for revisions and operative maintenance of the spray system to enable permanent operation of the system

Additional measures for mitigation of severe accidents

Electricity supply for the systems for severe accidents mitigation

- **Modification of corresponding sections of non-emergency source**
- **Additional diesel generator to cover all power supply of relevant equipment**
- **Most important systems powered from backup sources (accumulators) of the DC power**
- **Pumps from the new external source of coolant connected directly to the dedicated diesel generator.**

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 RPV
- New measurement of coolant level inside reactor cavity
- New measurement of coolant level inside steam generator boxes
- Replacement of original containment pressure monitoring system
- Replacement of containment temperature sensors and measurement chains
- New measurements of hydrogen concentration at different rooms of containment

Monitoring of parameters needed for control of severe accidents

- 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

Assumed impact to SAMGs

Operator actions required

Transition from EOPs to SAMGs based on core exit temperature

Initiation of dedicated diesel operation

Control of coolant inventory for recirculation and cavity flooding

(both external source of coolant and bubble tower trays drain down)

Control of containment pressure using external source of coolant for sprays

Control of primary pressure (depressurization of primary circuit)

Initiation of cavity flooding by opening the inlet valves

Monitoring systems dedicated to severe accident control for all phases

Restoration of containment spray functions (in recirculation mode)

Perspectives

Detailed design activites ongoing, no substantial problems reported

Development of SAMGs already initiated

both for full power and shut down conditions

lead by Westinghouse Electric Belgium

Start up scheduled for first reactor to 2012/2013

both hardware, procedures and training

Session 8

Experimental Investigation of Melt Debris Agglomeration with High Melting Temperature Simulant Materials

P. Kudinov, A. Karbojian and C.-T. Tran

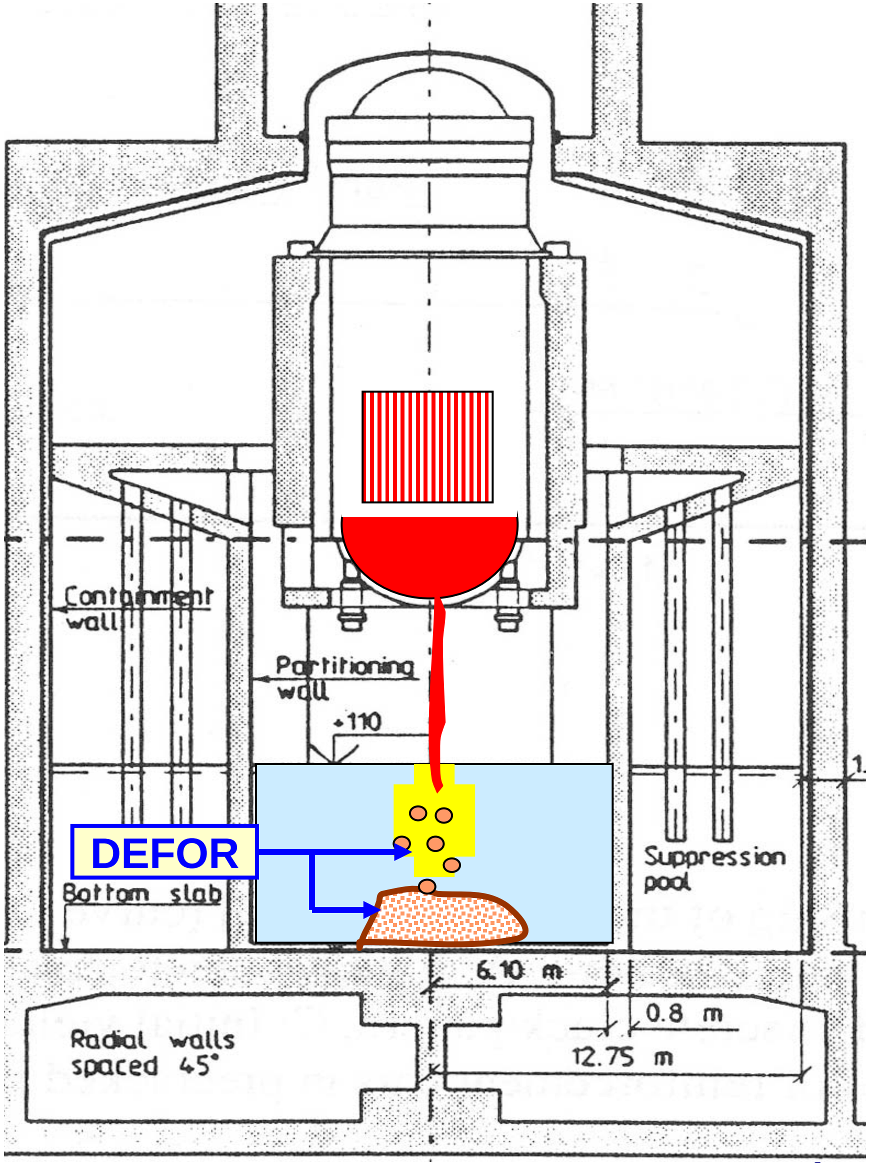
Division of Nuclear Power Safety, Royal Institute of Technology (KTH), Stockholm, Sweden

Debris Bed Formation (DEFOR)

- **Severe accident mitigation strategy in Swedish BWRs:**
  - Core melt poured in a deep (7-12m) water pool is expected to fragment quench and form a **coolable debris bed**

- **Is debris bed coolable?**
  - *Spatial configuration of the bed?*
  - *Porosity?*
  - *Particle size distribution?*
  - *Particle morphology?*
  - *Particle agglomeration?*

- **DEFOR program Goal:**
  - Establish methods to predict prototypical debris bed properties important for coolability



Problem Decomposition in Severe Accident Analysis

Fuel Coolant Interaction (FCI)

Debris Bed Coolability

How ?

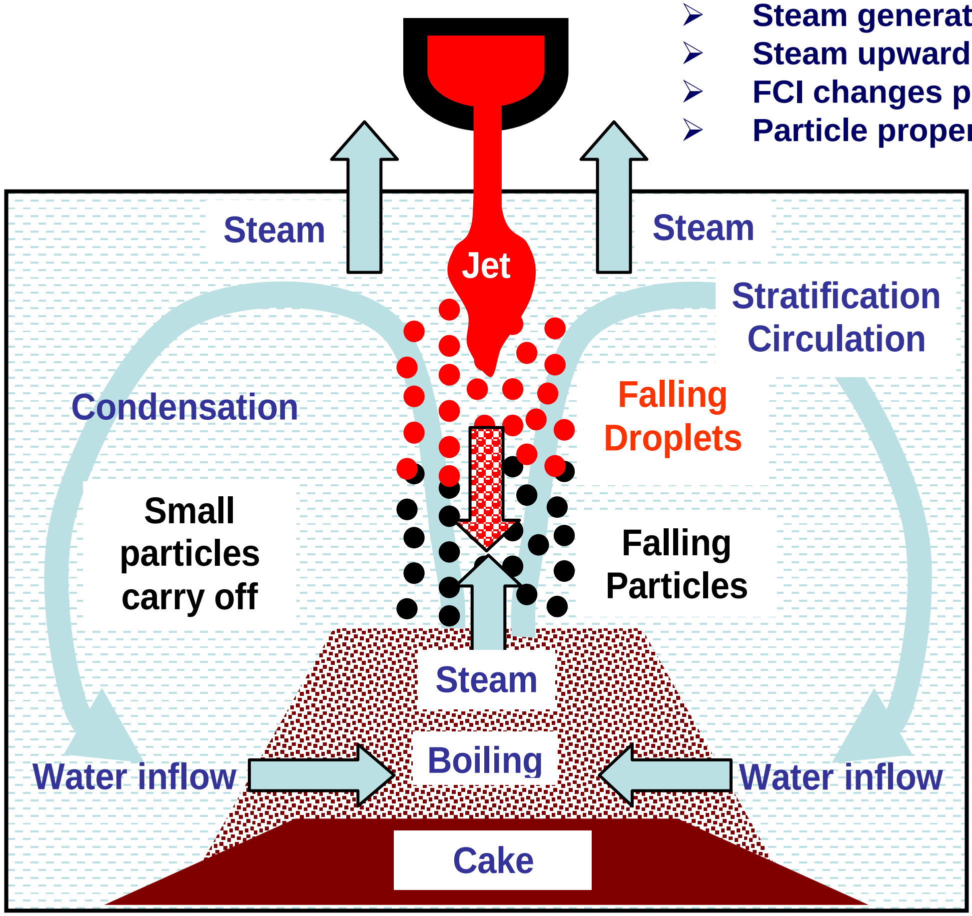
Data

Gap in knowledge

Data

Debris Bed Formation in a LWR Severe Accident

- The cooling of the debris bed is provided by heat transfer to the water that ingresses into the porous bed interior
- Steam generated inside debris bed is escaping upwards
- Steam upward flow changes conditions for FCI
- FCI changes particle properties (size distribution and morphology)
- Particle properties affect the debris bed coolability phenomena



Particle formation
- Hydrodynamic fragmentation
- Solidification and fracture
- Size distribution, morphology

Jet fragmentation
- Formation of droplets

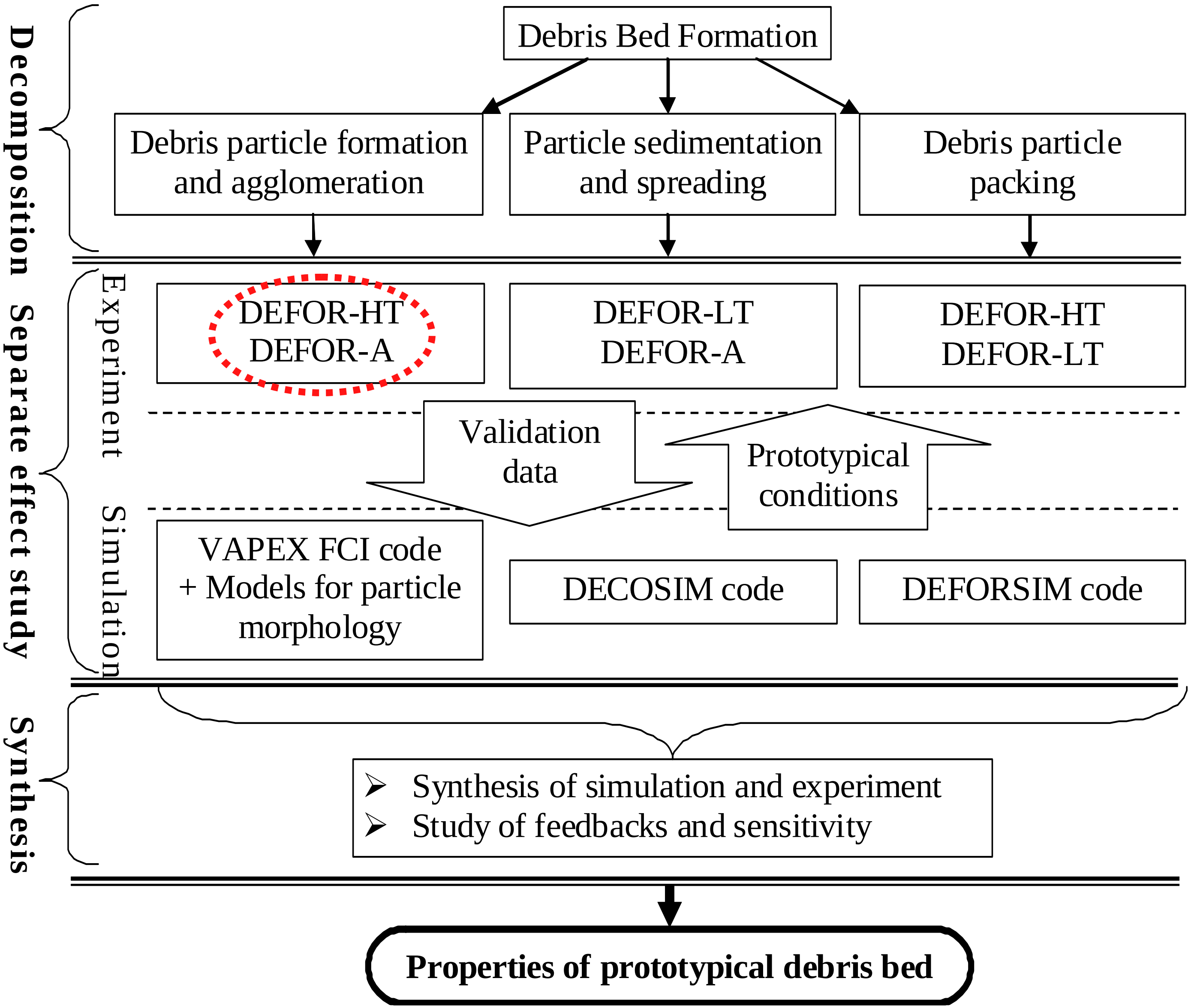
Feedback

DEFOR Phenomena
- Deposition and Packing
- Particle levitation, spreading
- Shape of debris bed
- Agglomerates and cakes
- Porous media properties
  - Porosity (void fraction)
  - Pores size distribution
  - Pore morphology
  - Non-homogeneity
  - Non isotropy

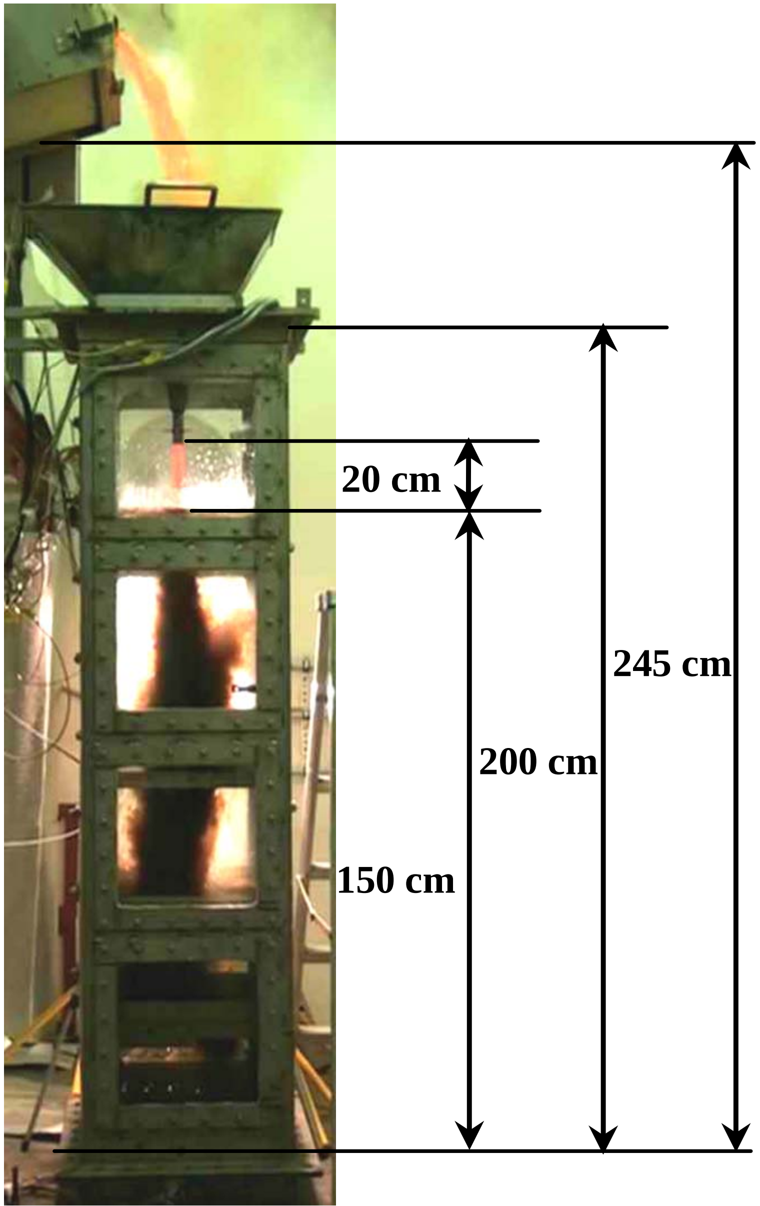
Coolability
- Steam / water flow
  - Inside the debris bed
  - In the pool
- Debris bed dryout

Strong **feedback** between FCI, debris bed formation and coolability

DEFOR Research Program
"To Fill the Gap in Knowledge"



DEFOR-HT (High Temperature) experimental program



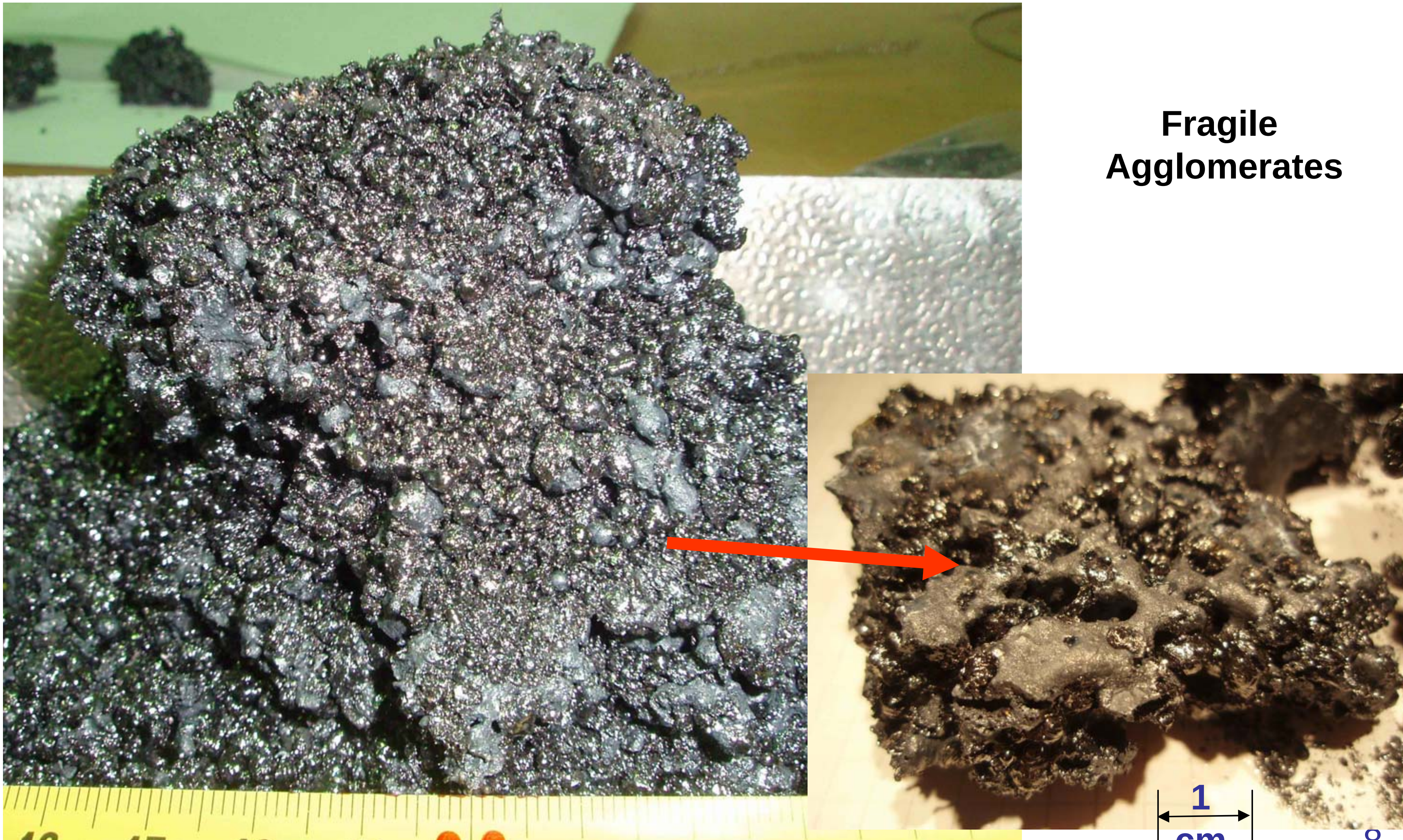
Containment: 4x4x4 m, 5 bar max pressure

DEFOR-S (Snapshot) Test Matrix

N	Experiment	Simulant	**Mixture**	Pool depth, m	Melt temp., C	Melt superheat, C	**Water temp., C**	**Porosity**	**Fraction of agglomerates, %**
1	DEFOR-S1	MnO-TiO2	**Eutectic**	0.65	1450	81	16	**71**	0
2	DEFOR-S2	MnO-TiO2	**Eutectic**	0.65	1400	31	18	**71**	0
3	DEFOR-S3	Bi2O3-WO3	**Eutectic**	0.65	950	70	21	**70**	0
4	DEFOR-S4	WO3-TiO2	**Eutectic**	0.65	1400	167	20	**69**	0
5	DEFOR-S5	Bi2O3-WO3	**Eutectic**	0.65	980	100	**75**	**59**	**20**
6	DEFOR-S6	Bi2O3-WO3	**Non-eutectic**	0.65	1060	-20	20	**68**	0
7	DEFOR-S7	Bi2O3-WO3	**Non-eutectic**	0.65	1010	16	19	**62**	0
8	DEFOR-S8	Bi2O3-WO3	**Non-eutectic**	0.65	1020	26	**75**	**46**	**90**
9	DEFOR-S9	Bi2O3-WO3	**Non-eutectic**	1.1	1070	45	11	**68**	0
10	DEFOR-S10	Bi2O3-WO3	**Eutectic**	1.0	1080	210	**73**	**62**	**8**
11	DEFOR-S11	Bi2O3-WO3	**Eutectic**	1.1	1070	200	53	**57**	0
12	DEFOR-S12	Bi2O3-WO3	**Non-eutectic**	1.1	1150	125	50	**65**	0
13	DEFOR-S13	Bi2O3-WO3	**Eutectic**	1.1	1100	230	35	**68**	0

DEFOR-S5: Water 75C,
Eutectic melt, T_{melt}= 980 C
Porosity 59.3%, Mass of agglomerates 20%

Fragile Agglomerates



DEFOR-S8: Water 75C,
Non-Eutectic melt, T_{melt}= 1020 C
Porosity 45.7%, , Mass of cake 90%

Agglomerates and Cakes

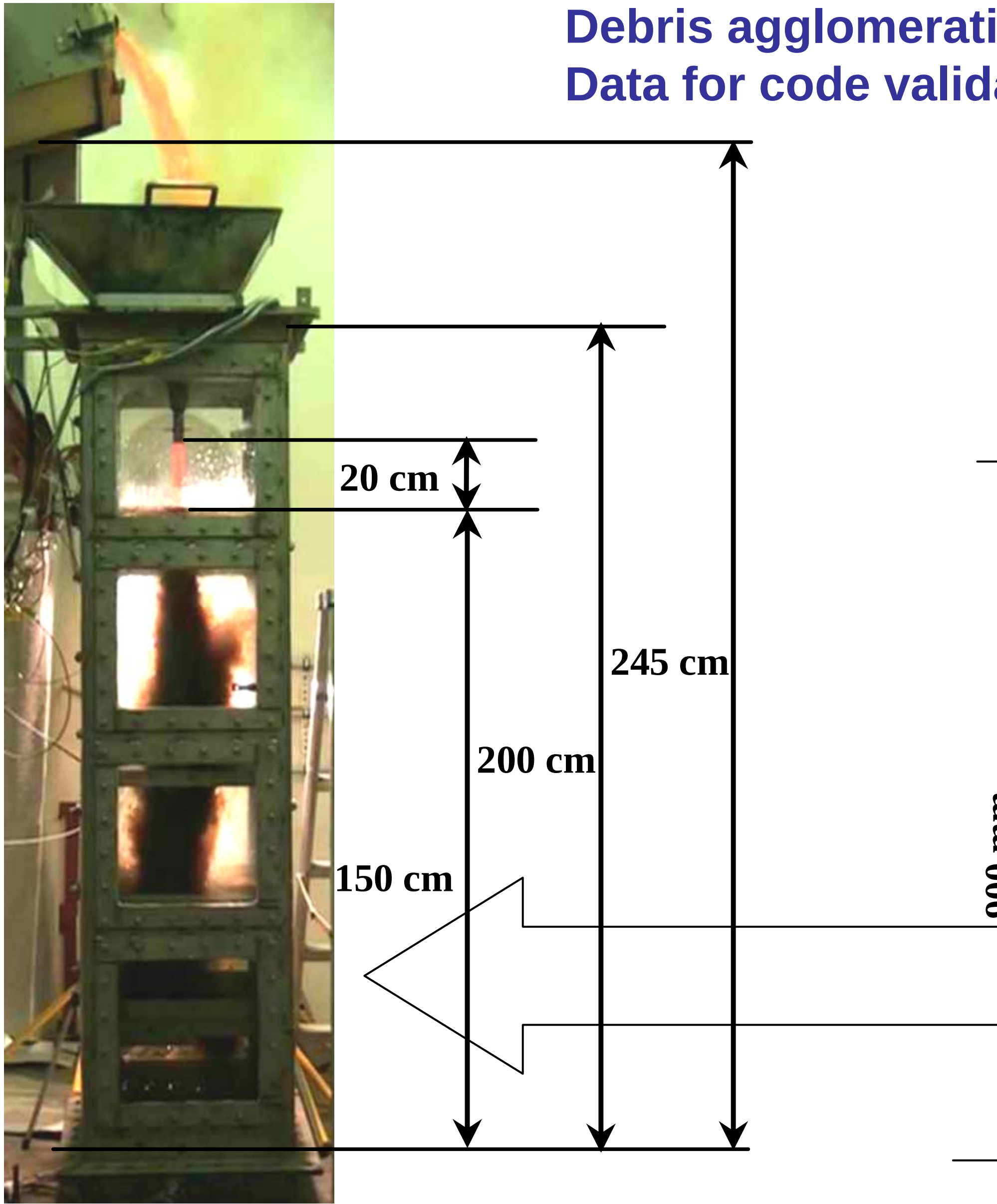
- Agglomerates - “soldered” together groups of particles

- “Cake” is formed when liquid melt fraction is bigger than fraction of solid particles

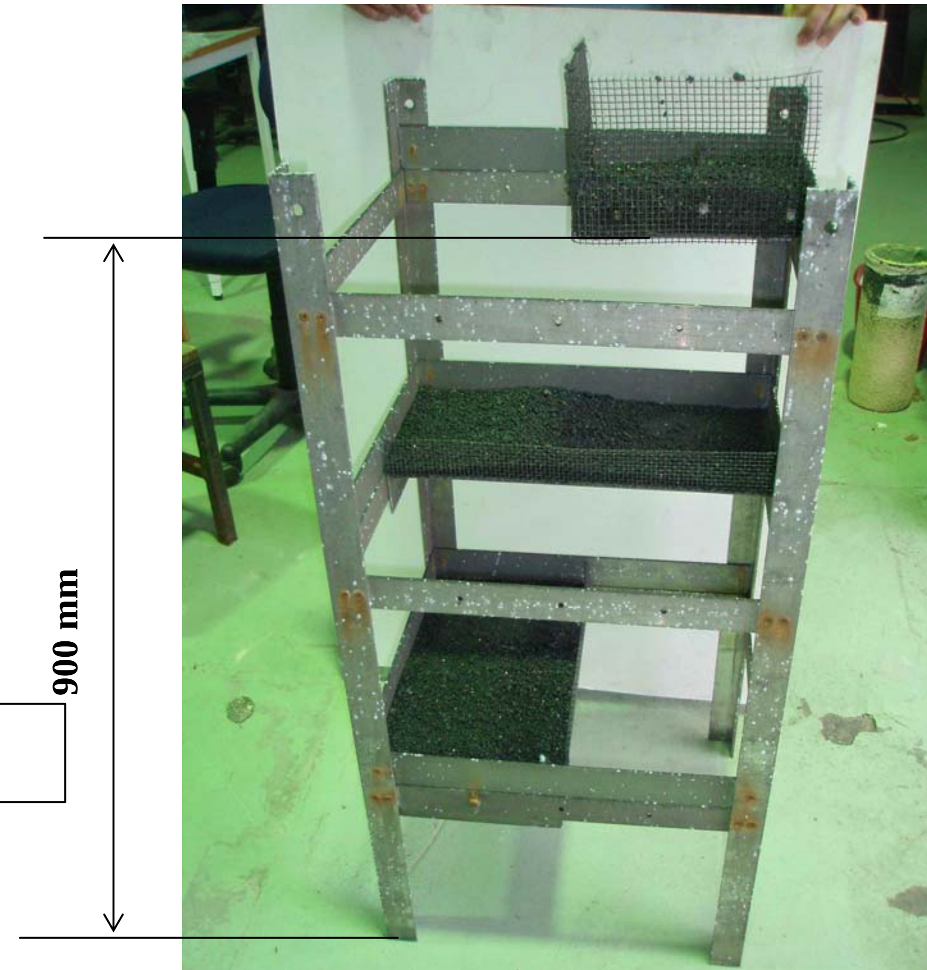
5 cm

DEFOR-A (Agglomeration) experiment

Debris agglomeration as a function of pool depth.
Data for code validation.



Debris Catchers



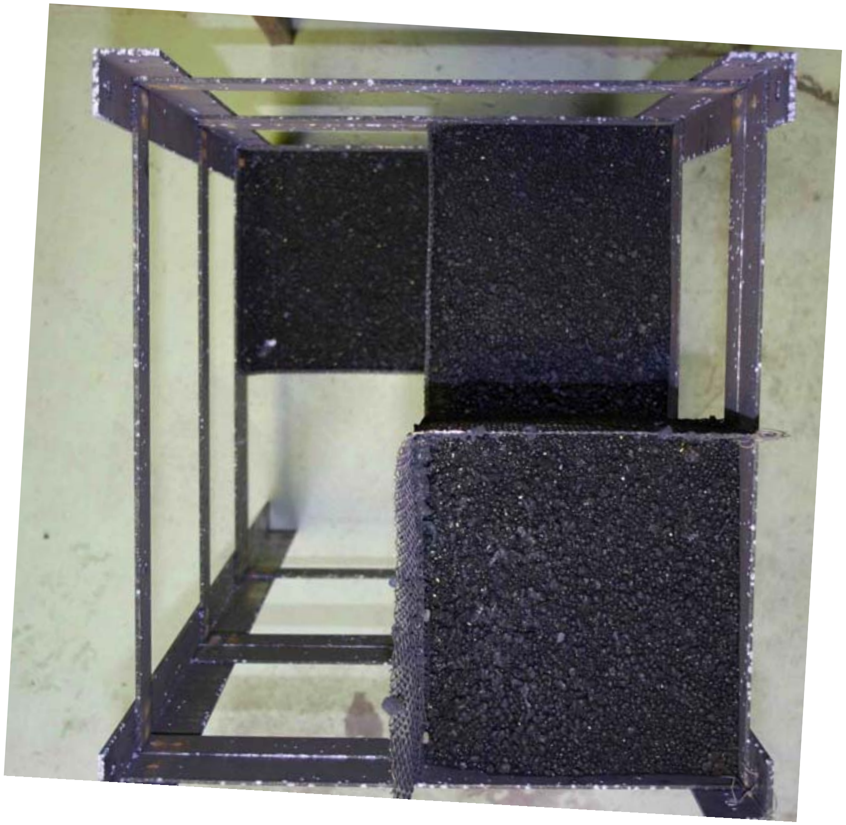
DEFOR-A Test Conditions

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/m3	7811	7811	7811
7	Melt volume, liter	3	3	3
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

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

- Influence of water subcooling and jet diameter

Bed Shape: DEFOR-A2

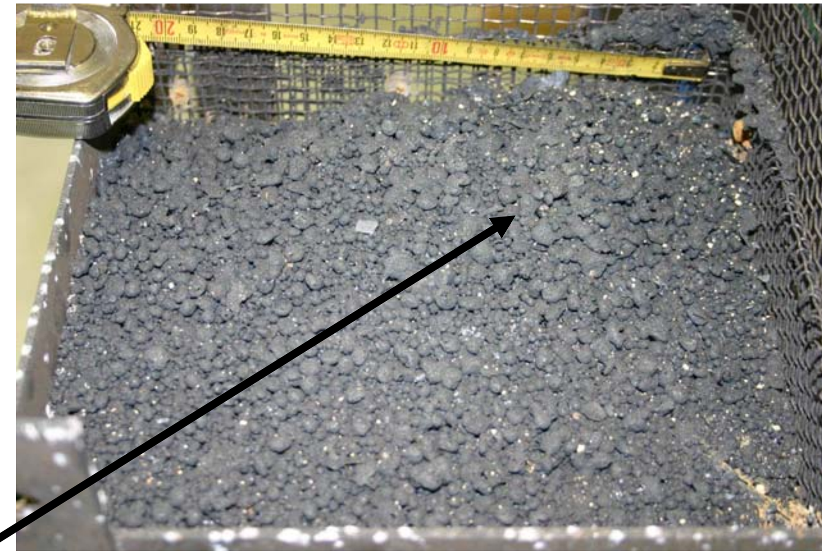


DEFOR-A2: Top view on catchers
Beds are spread uniformly

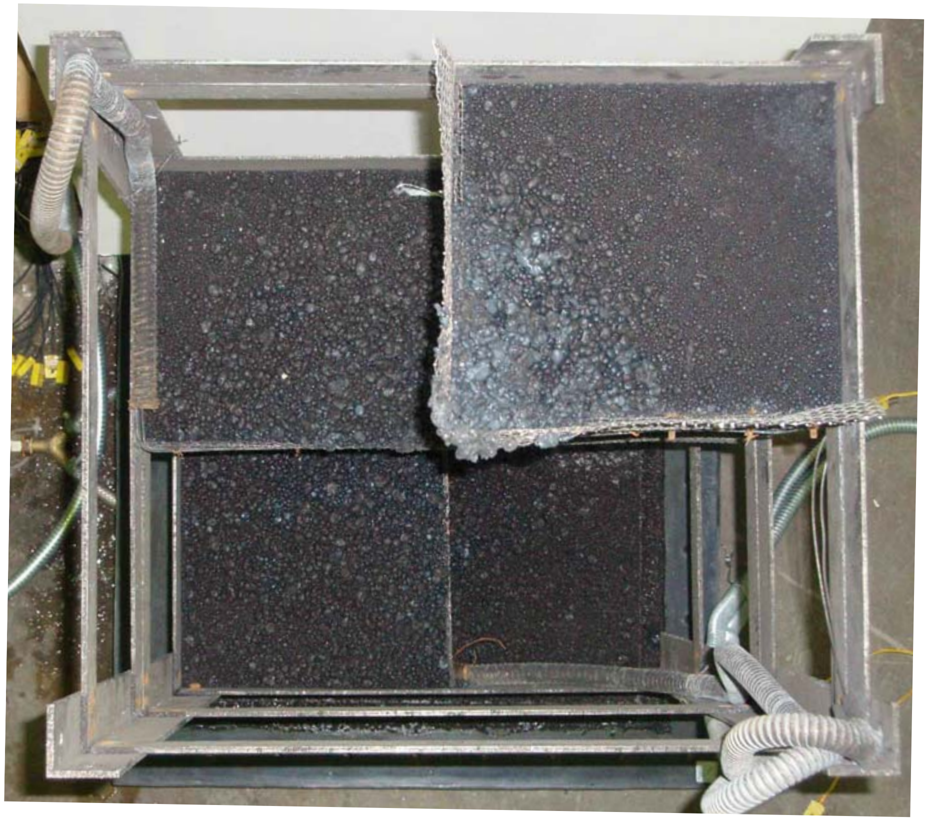
- **DEFOR-A2**
  - Melt 24 kg
  - Pool depth 1.52 m
  - Melt superheat 110K
  - Djet=22 mm
  - Water subcooling 2 K

Bed Shape: DEFOR-A2

- Debris are uniformly spread over the catchers



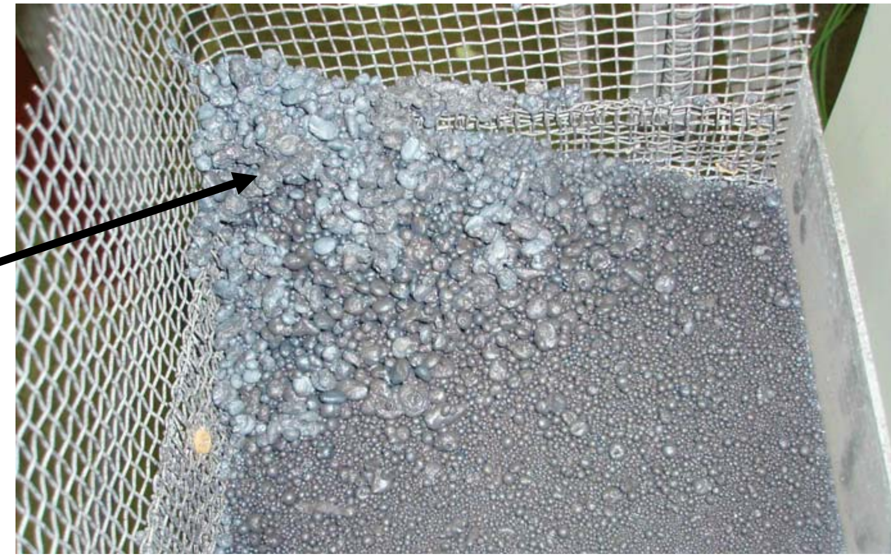
Bed Shape: DEFOR-A5



- **DEFOR-A5**
  - Melt 24 kg
  - Pool depth 1.52 m
  - Melt superheat 100K
  - Djet=10 mm
  - Water subcooling 4 K

- Debris beds are heap-like
- Small particles (<1mm) are spread over the catchers

Bed Shape: DEFOR-A5



- Debris beds are heap-like
- Small particles (<1mm) are spread over the catchers

Bed Shape: DEFOR-A6

DEFOR-A6: Top view on catchers
Beds are heap-like

- **DEFOR-A6**
  - Melt 24 kg
  - Pool depth 1.52 m
  - Melt superheat 136K
  - Djet=12 mm
  - Water subcooling 22 K

Bed Shape: DEFOR-A6

- No significant debris spreading

DEFOR-A6

- DEFOR-A6 movie

- **DEFOR-A6**
  - Melt 24 kg
  - Pool depth 1.52 m
  - Melt superheat 136K
  - Djet=12 mm
  - Water subcooling 22 K

Jet Diameter and Water Subcooling Effect on Agglomeration

DEFOR-A6,

- Djet=12 mm
- Water subcooling 22 K.

Steam **condenses** rapidly.

No violent steam production.

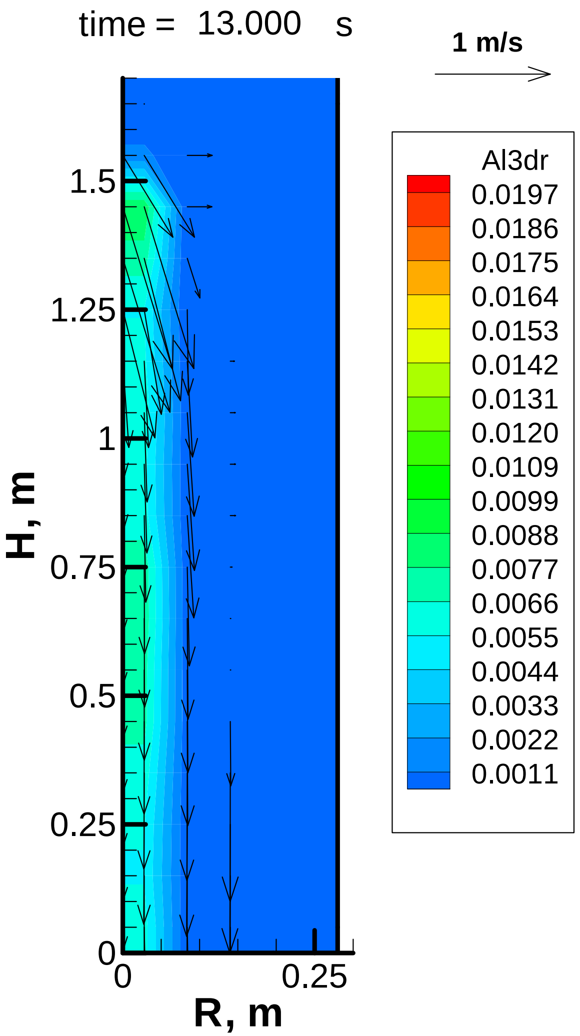
No mixing.

No debris spreading.

Debris bed is heap like.

No upward motion of debris.

Higher fraction of agglomerates



DEFOR-A2

- DEFOR-A2 movie

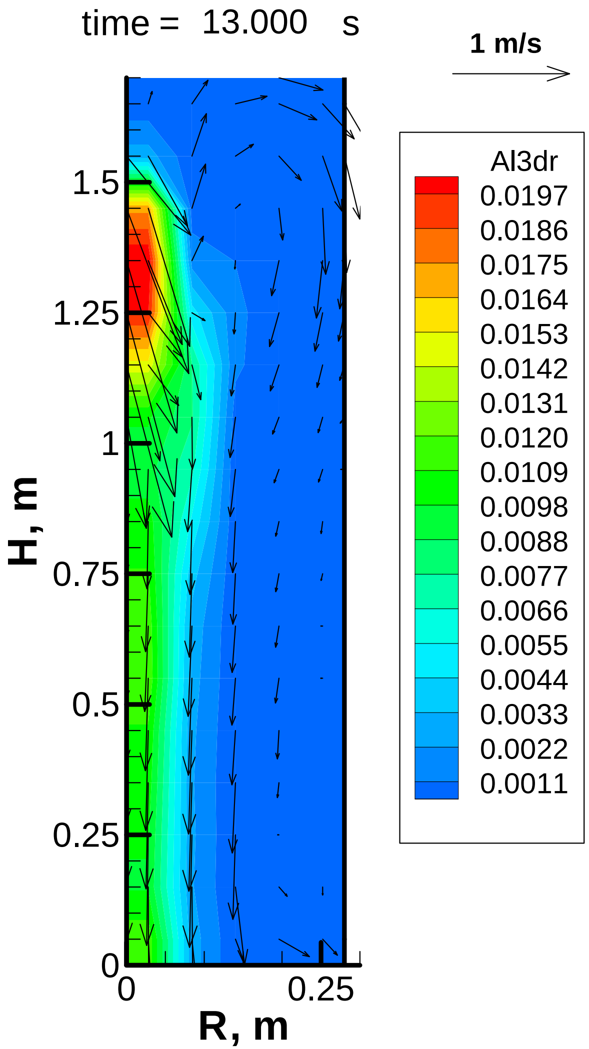
- **DEFOR-A2**
  - Melt 24 kg
  - Pool depth 1.52 m
  - Melt superheat 110K
  - Djet=22 mm
  - Water subcooling 2 K

Jet Diameter and Water Subcooling Effect on Agglomeration

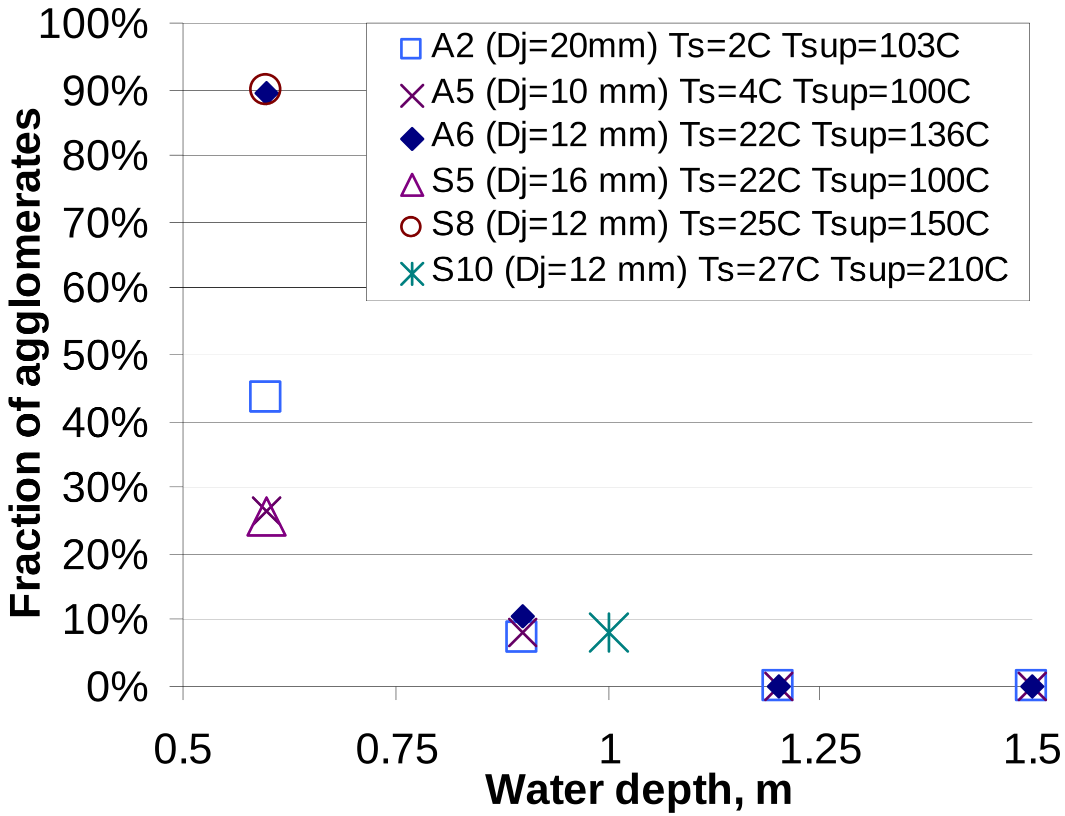
DEFOR-A2,

- Djet=22 mm
- Water subcooling 2 K.

Violent steam production.
Violent mixing.
Debris spreading.
Debris bed is flat.
Debris move upward first.
Lower fraction of agglomerates



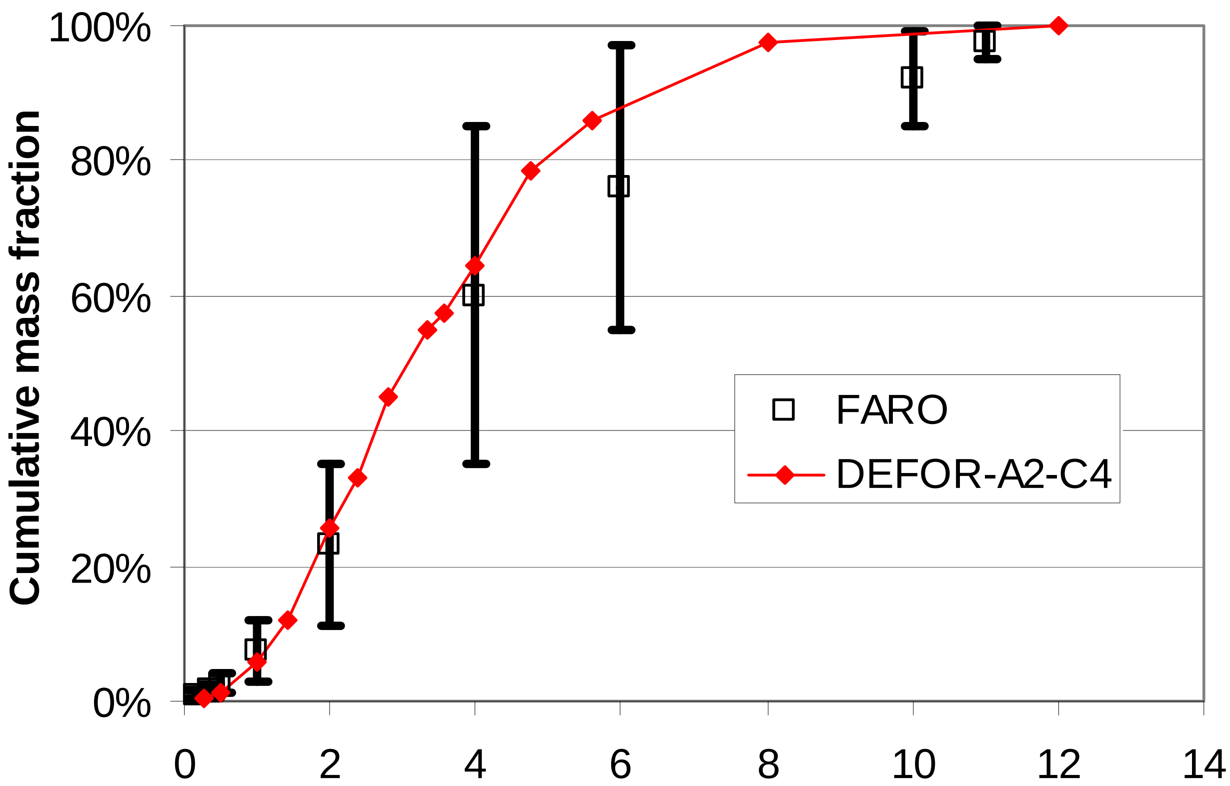
Fraction of Agglomerates versus Pool Depth



Fraction of agglomerated debris is lower in tests with low water subcooling and with bigger jet diameter

Fraction of agglomerated debris reduces rapidly with increasing of the pool depth

Are Results Prototypical?
Particle Size Distribution



- Particle size distribution in DEFOR agrees well with data obtained in the FCI experiments with real corium

Summary and Outlook

- 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 experiment 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 behavior
- Fraction of agglomerated debris decreases rapidly as the depth of the coolant is increasing. Debris collected in Catcher-4 (1.5m deep) are 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 is necessary

DEFOR Publications 2007-2009

Experimental

1. Karbojian A., Ma W.M., Kudinov P., Davydov M., Dinh N., “A scoping study of debris formation in DEFOR experimental facility”, 15th International Conference on Nuclear Engineering, Nagoya, Japan, April 22-26, 2007, Paper number ICON15-10620.
2. Kudinov P., Karbojian A., Ma W.M., and Dinh T.-N., “An experimental study on debris formation with corium stimulant materials,” Proc. ICAPP’08, Anaheim, CA USA, June 8–12, 2008, paper 8390.
3. Karbojian A., Ma W., Kudinov P., and Dinh T.-N., “A Scoping Study of Debris Bed Formation in the DEFOR Test Facility”, Nuclear Engineering and Design, 239, 2009, 1653-1659.
4. Kudinov P., Karbojian A., Tran C.-T., “Experimental Investigation of Melt Debris Agglomeration with High Melting Temperature Simulant Materials,” Proceedings of ISAMM-2009, Böttstein, Switzerland, October 26 - 28, 2009.
5. Kudinov P., Karbojian A., Ma W., and Dinh T.-N. “The DEFOR-S Experimental Study of Debris Formation with Corium Simulant Materials,” Nuclear Technology, 2009 (accepted, in press).

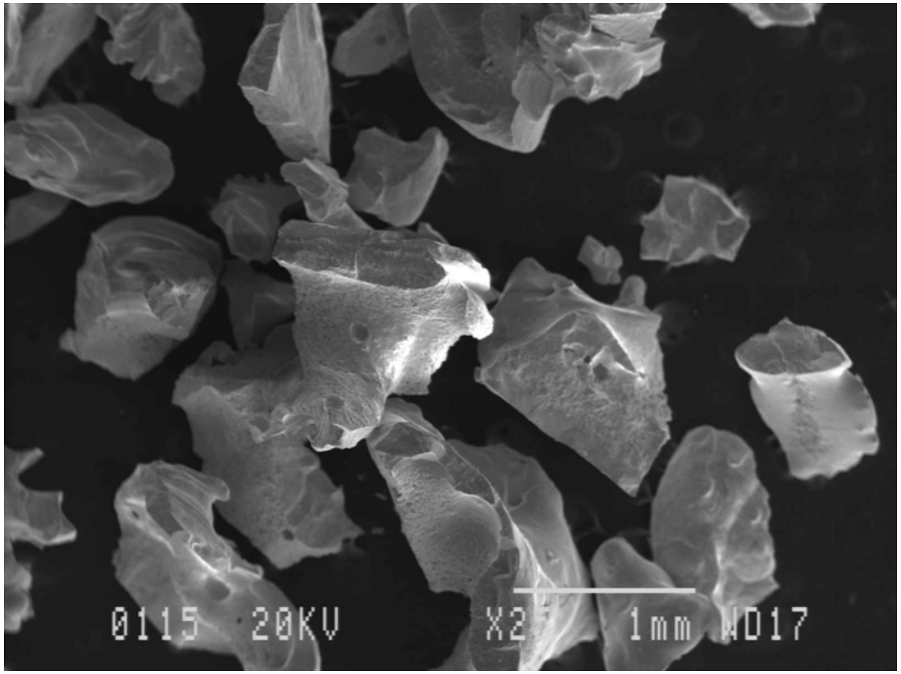
Analytical

6. Kudinov P., Karbojian A., Ma W.M., Davydov M., and Dinh T.-N., “A Study of Ex-Vessel Debris Formation in a LWR Severe Accident”, Proceedings of ICAPP 2007, Nice, France, May 13-18, 2007, Paper 7512.
7. Kudinov P. and Dinh T.-N., “An analytical study of mechanisms that govern debris packing in a LWR severe accident”, The 12th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-12), Sheraton Station Square, Pittsburgh, Pennsylvania, U.S.A. September 30-October 4, 2007. Paper 247.
8. Kudinov P. and Dinh T.-N., “A Computational Study of Debris Bed Formation,” *ANS Transactions*, 2008, paper 193463.
9. Yakush S., Kudinov P., and Dinh T.-N., “Modeling of Two-Phase Natural Convection Flows in a Water Pool with a Decay-Heated Debris Bed,” Proc. ICAPP’08, Anaheim, CA USA, June 8–12, 2008, paper 8409.
10. Dombrovsky L.A., Davydov M.V., and Kudinov P., Thermal radiation modeling in numerical simulation of melt-coolant interaction, Proc. Int. Symp. Adv. Comput. Heat Transfer (CHT-08), May 11–16, 2008, paper 155.
11. Dombrovsky L.A., Davydov M.V., Kudinov P., “Thermal radiation modeling in numerical simulation of melt-coolant interaction,” Comp. Therm. Sci. 1 (1) 2009, pp. 1-35.
12. Kudinov P., Davydov M., “Development of Ex-Vessel Debris Agglomeration Mode Map for a LWR Severe Accident Conditions,” The 17th International Conference on Nuclear Engineering, July 12-16, 2009, Brussels, Belgium, Paper ICONE17-75080.
13. Yakush S., Kudinov P., and Dinh T.-N., “Multiscale Simulations of Self-organization Phenomena in the Formation and Coolability of Corium Debris Bed,” Proc. The 13th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-13), September 27-October 2, 2009. Kanazawa City, Ishikawa Prefecture, Japan, Paper N13P1143.
14. Yakush S. and Kudinov P., “Transient Phenomena of Ex-vessel Debris Bed Formation in a LWR Severe Accident,” ANS Transactions, 2009, paper 210830.
15. Kudinov P. and Davydov M., “Approach to Prediction of Melt Debris Agglomeration Modes in a LWR Severe Accident,” Proceedings of ISAMM-2009, Böttstein, Switzerland, October 26 - 28, 2009.
16. Yakush S. and Kudinov P., “Simulation of Ex-Vessel Debris Bed Formation and Coolability in a LWR Severe Accident,” Proceedings of ISAMM-2009, Böttstein, Switzerland, October 26 - 28, 2009.
17. Kudinov P. and Kudinova V., “Influence of Water Subcooling on Fracture of Melt Debris Particle,” ANS Transactions, 2009, paper 210646.

Are Results Prototypical?
Particle morphology

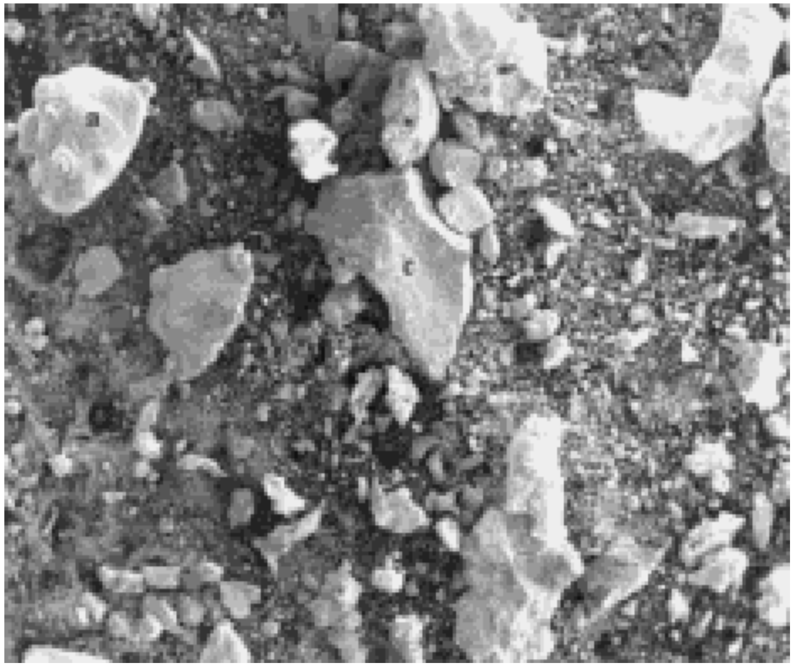
High solidification rate + solid fracture due to thermal stresses –> rock-type particles

DEFOR Simulant



DEFOR-S7 experiment
WO_3-Bi_2O_3

Corium



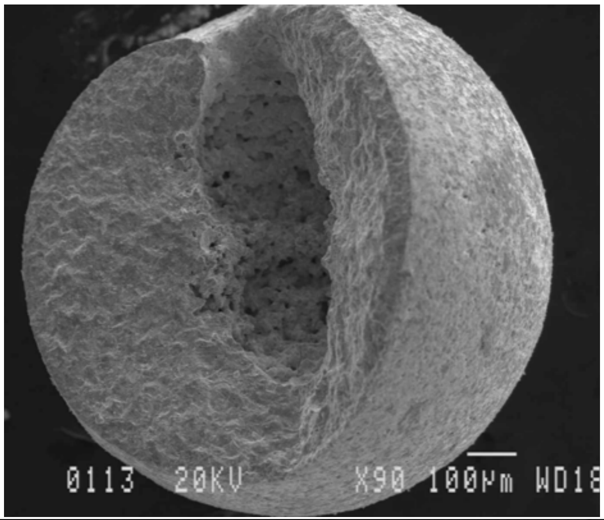
TROI 13 experiment
Corium (UO_2=70%, ZrO_2=30%)

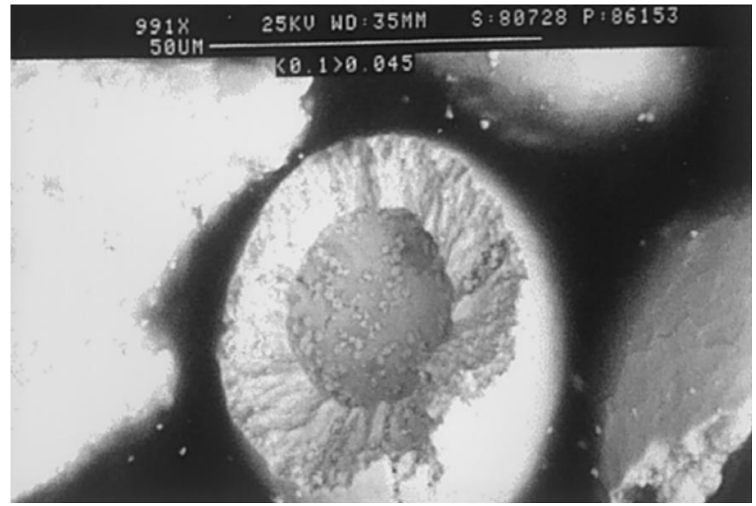
Are Results Prototypical?
Particle morphology

Slow solidification rate –> smooth surface particles + internal porosity

DEFOR Simulant | **Corium**

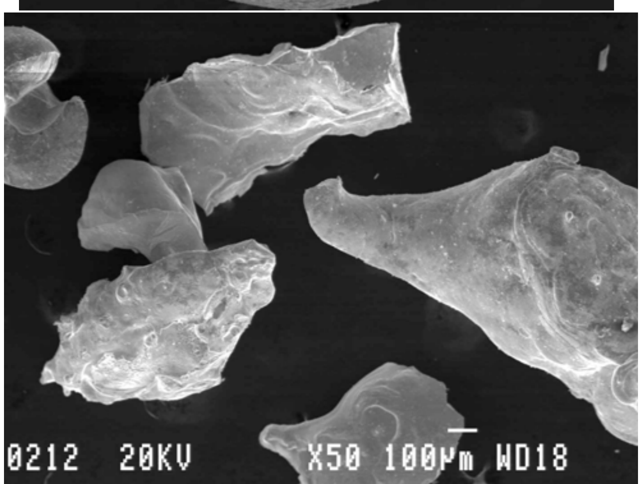
DEFOR-S8

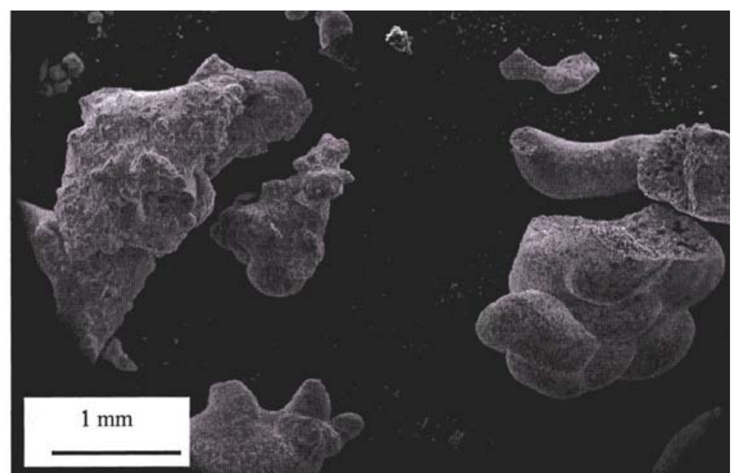




KROTOS
K-58

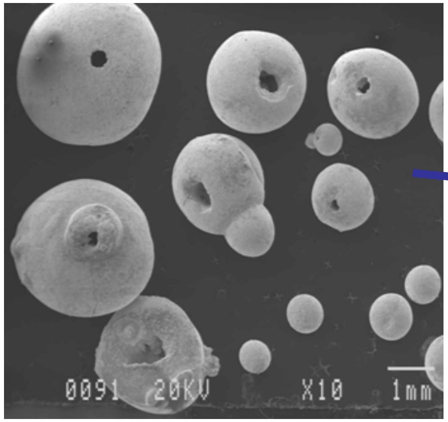
DEFOR-S9

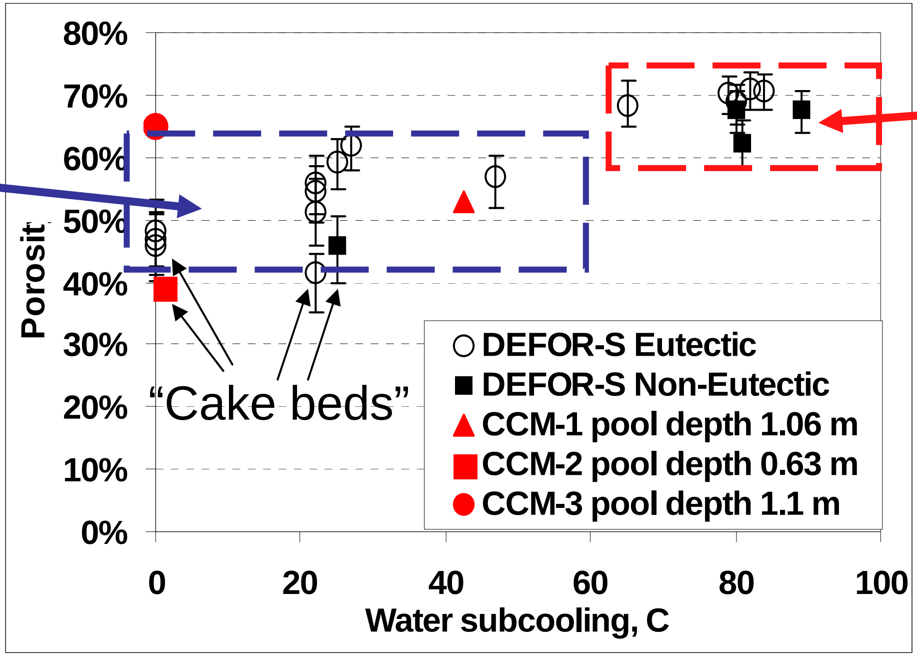


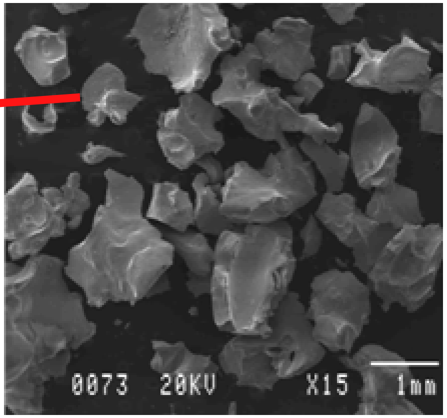


FARO
L-24

Particle Morphology Affects Bed Porosity







- Two characteristic values of porosity
  - Round-shape particles ~ 45-60%
  - Rock-like particles ~ 60-70%
- Both values are much higher than previously assumed

Approach to Prediction of Melt Debris Agglomeration Modes in a LWR Severe Accident

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Division of Nuclear Power Safety, Royal Institute of Technology (KTH), Stockholm, Sweden

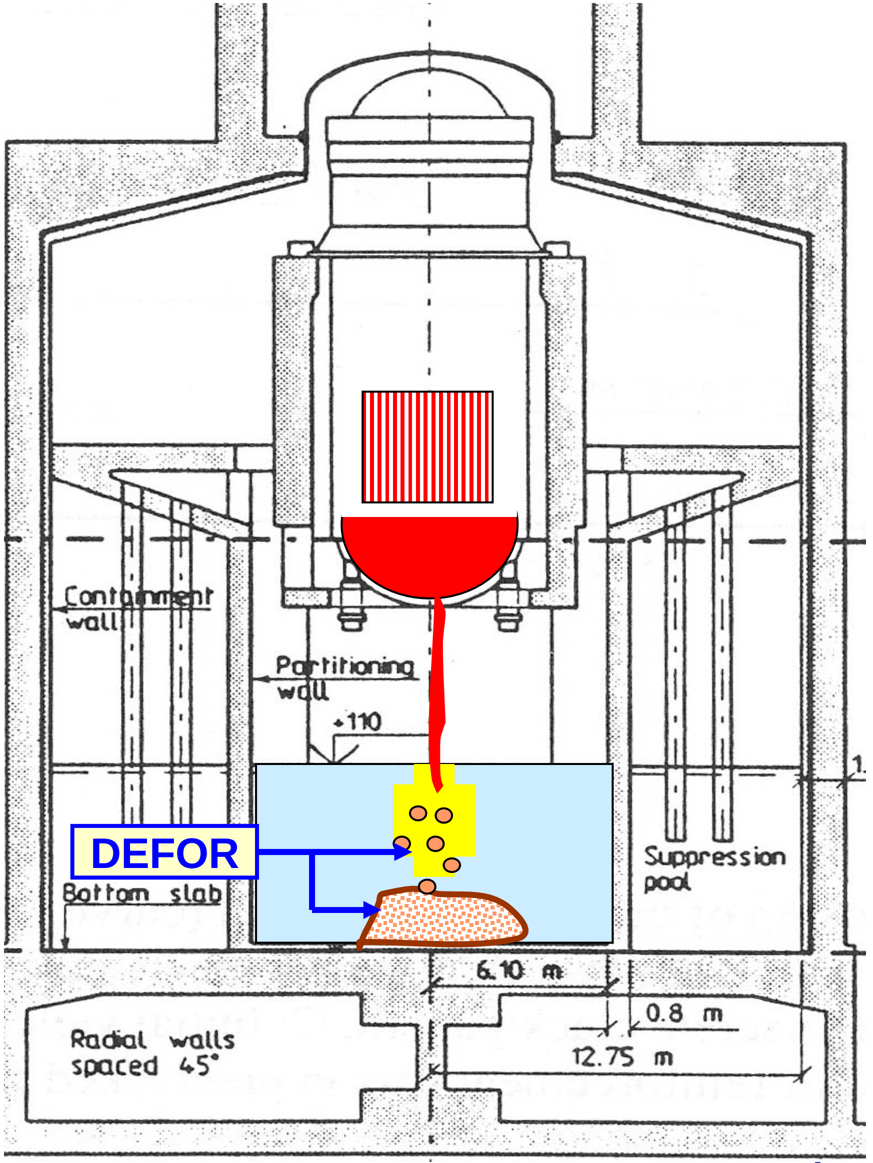
M. Davydov
Electrogorsk Research and Engineering Center on Nuclear Power Plants Safety (EREC), Electrogorsk, Russia

Debris Bed Formation (DEFOR)

- **Severe accident mitigation strategy in Swedish BWRs:**
  – Core melt poured in a deep (7-12m) water pool is expected to fragment quench and form a **coolable debris bed**

- **Is debris bed coolable?**
    - *Spatial configuration of the bed?*
    - *Porosity?*
    - *Particle size distribution?*
    - *Particle morphology?*
    - *Particle agglomeration?*

- **DEFOR program Goal:**
  – Establish methods to predict prototypical debris bed properties important for coolability



Problem Decomposition in Severe Accident Analysis

Fuel Coolant Interaction (FCI)

Debris Bed Coolability

How ?

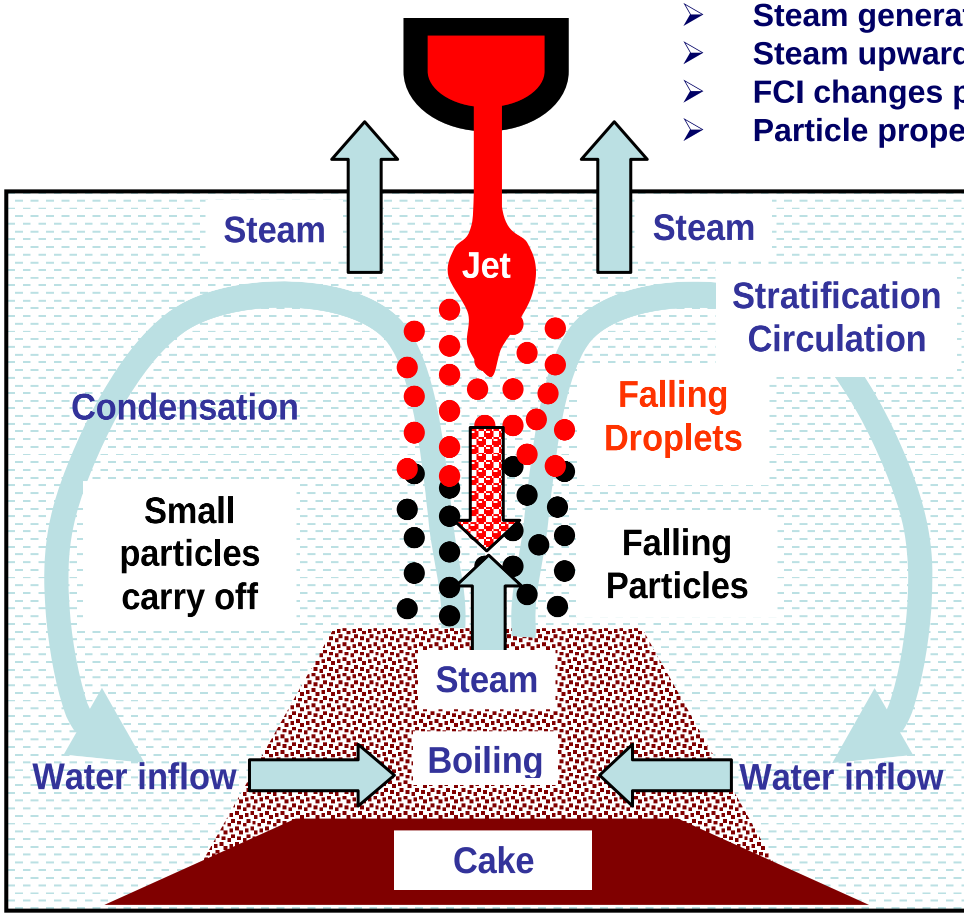
Data

Gap in knowledge

Data

Debris Bed Formation in a LWR Severe Accident

- ➢ The cooling of the debris bed is provided by heat transfer to the water that ingresses into the porous bed interior
- ➢ Steam generated inside debris bed is escaping upwards
- ➢ Steam upward flow changes conditions for FCI
- ➢ FCI changes particle properties (size distribution and morphology)
- ➢ Particle properties affect the debris bed coolability phenomena



Particle formation
- Hydrodynamic fragmentation
- Solidification and fracture
- Size distribution, morphology

Jet fragmentation
- Formation of droplets

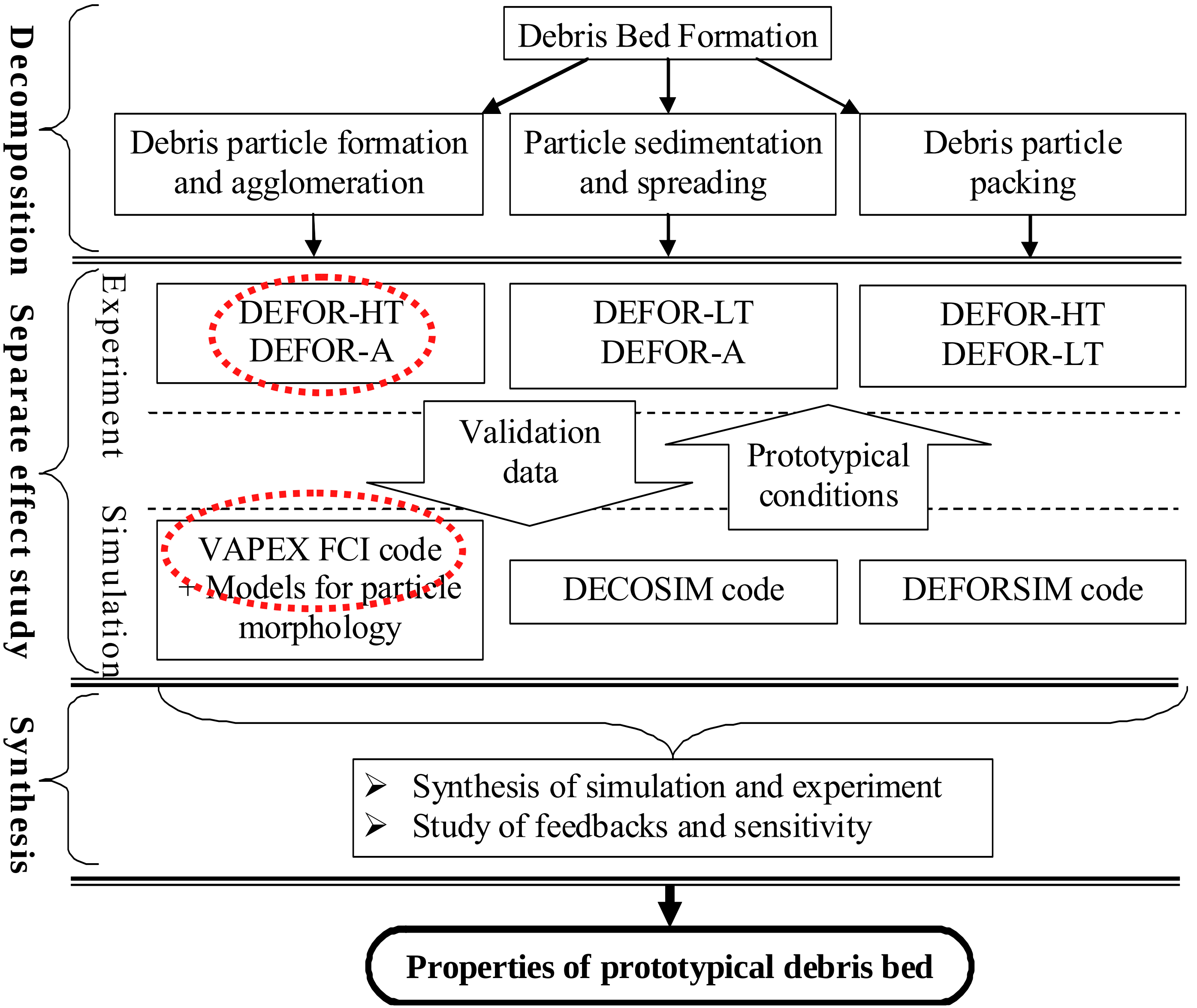
Feedback

DEFOR Phenomena
- Deposition and Packing
- Particle levitation, spreading
- Shape of debris bed
- Agglomerates and cakes
- Porous media properties
  - Porosity (void fraction)
  - Pores size distribution
  - Pore morphology
  - Non-homogeneity
  - Non isotropy

Coolability
- Steam / water flow
  - Inside the debris bed
  - In the pool
- Debris bed dryout

Strong **feedback** between FCI, debris bed formation and coolability

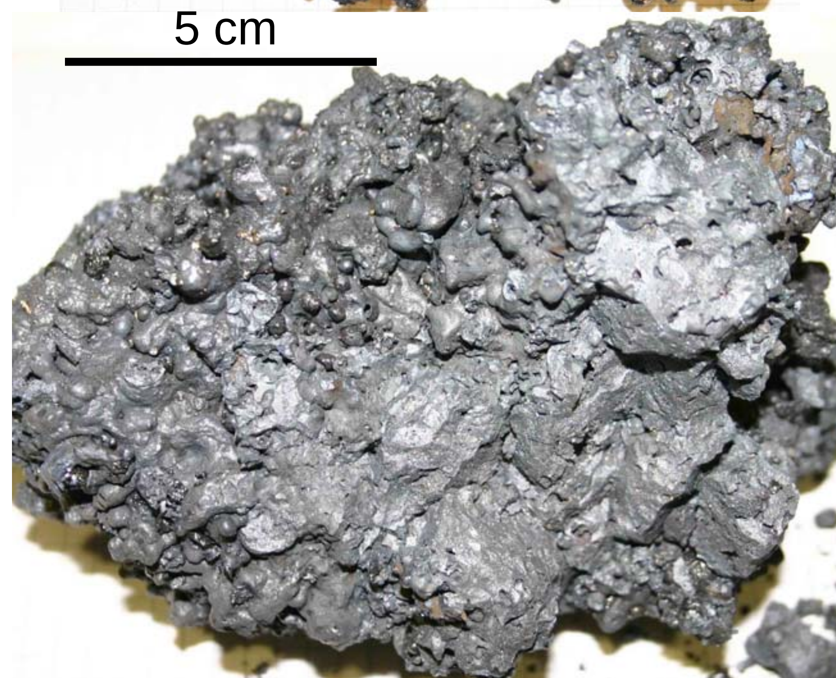
DEFOR Research Program
"To Fill the Gap in Knowledge"



Agglomerates and Cakes

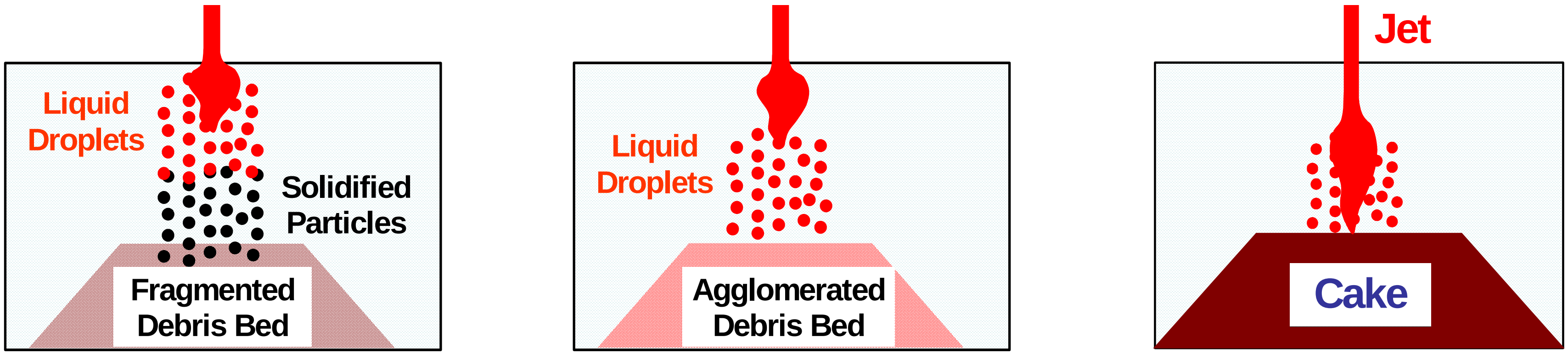
- Agglomerates - “soldered” together groups of particles

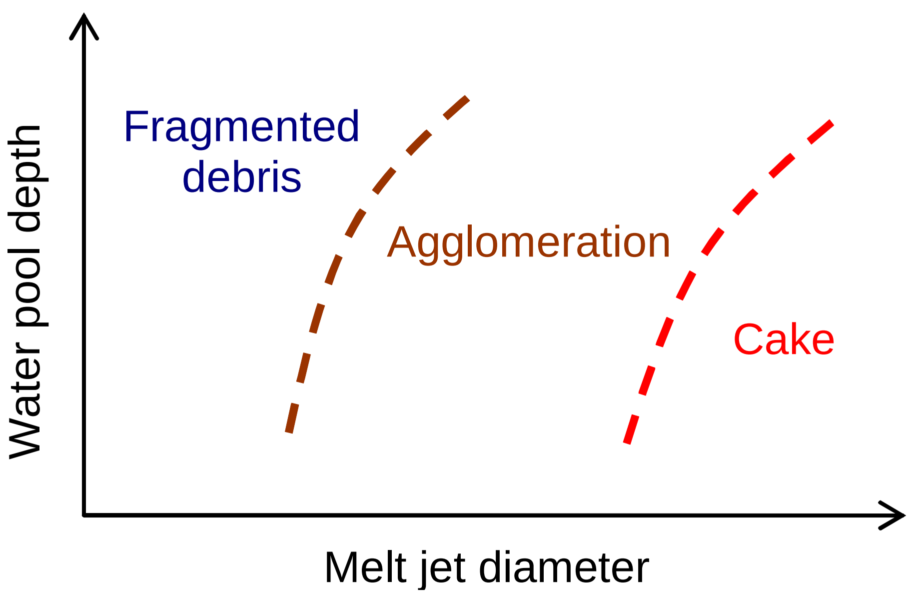
- “Cake” is formed when liquid melt fraction is bigger than fraction of solid particles



Agglomeration Mode Map

- **Development of debris agglomeration mode map for prototypical conditions of a BWR severe accident**





Three modes of debris agglomeration:
- Complete debris fragmentation
- Debris agglomeration
- Cake formation

Simulation Vehicle

- **VAPEX** code developed in Electrogorsk Research and Engineering Center (EREC, Russia) for the analysis of fuel-coolant interaction under severe accident conditions
- **VAPEX** is 2D multiphase/multiflow code considering three phases: water, gas (steam, +air, +hydrogen, +argon) and melt. It has models for the following processes:

  - Thermal hydraulics of water/vapor/non-condensable gas mixtures (Eulerian approach)

$$
\begin{cases}
\dfrac{\partial \alpha_i \rho_i}{\partial t} + \nabla\left(\alpha_i \rho_i \vec{u}_i\right) = \Gamma_i \\
\dfrac{\partial \alpha_i \rho_i \vec{u}_i}{\partial t} + \nabla\left(\alpha_i \rho_i \vec{u}_i \cdot \vec{u}_i\right) = -\alpha_i \cdot \nabla p + \Gamma_i \vec{u}_i + F_i + \alpha_i \rho_i \vec{g} \\
\dfrac{\partial \alpha_i \rho_i e_i}{\partial t} + \nabla\left(\alpha_i \rho_i e_i \vec{u}_i\right) = -p \cdot \left[\dfrac{\partial \alpha_i}{\partial t} + \nabla\left(\alpha_i \vec{u}_i\right)\right] + \Gamma_i h_i + Q_i
\end{cases}
$$

$$
\left.
\begin{aligned}
&\frac{\partial}{\partial t}\left(\rho_f A\right) + \frac{\partial}{\partial z}\left(\rho_f u_f A\right) = -\Gamma_{frag} \\
&\frac{\partial}{\partial t}\left(\rho_f u_f A\right) + \frac{\partial}{\partial z}\left(\rho_f u_f^2 A\right) = -g\left(\rho_f - \rho_a\right)A - \\
&\qquad\qquad -C_D \pi R \rho_a \left(u_f - u_a\right)^2 - u_f \Gamma_{frag}
\end{aligned}
\right\}
$$

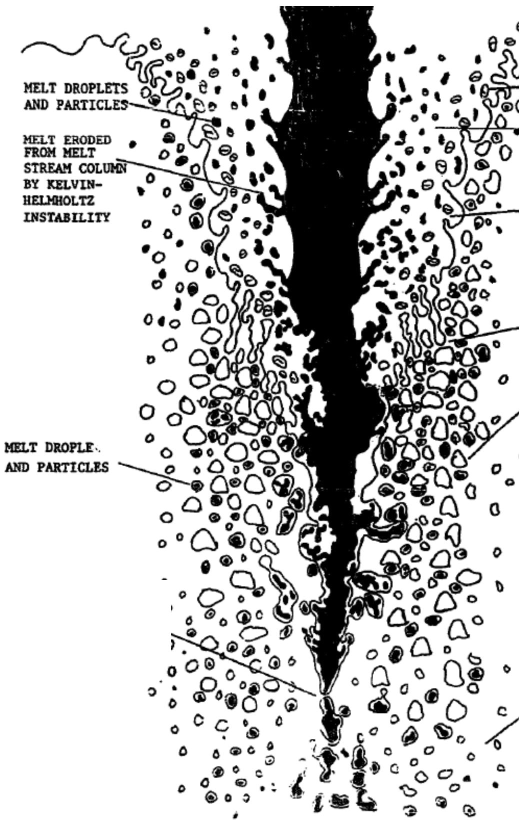
  - Dynamics of melt jet and its fragmentation (Eulerian approach)
  - Droplet sedimentation, debris formation (Lagrangian approach)
  - Radiation heat transfer from droplets to coolant
  - Droplet temperature profile, crust formation

$$
\begin{cases}
\rho_f \dfrac{d\vec{u}_k}{dt} = -\vec{F}_{lf} - \vec{F}_{vf} - g\left(\rho_d - \rho_a\right) \\
\rho_f c_f \dfrac{dT_f}{dt} = -R_{lf}\left(T_f - T_l\right) - R_{vf}\left(T_f - T_v\right) - \psi \varepsilon_f T_f^4
\end{cases}
$$

Jet Breakup Mode Influence on Pre-Deposited State of the Debris

Epstein & Fauske (1985)

Stripping of Kelvin-Helmholtz instabilities from jet lateral surface



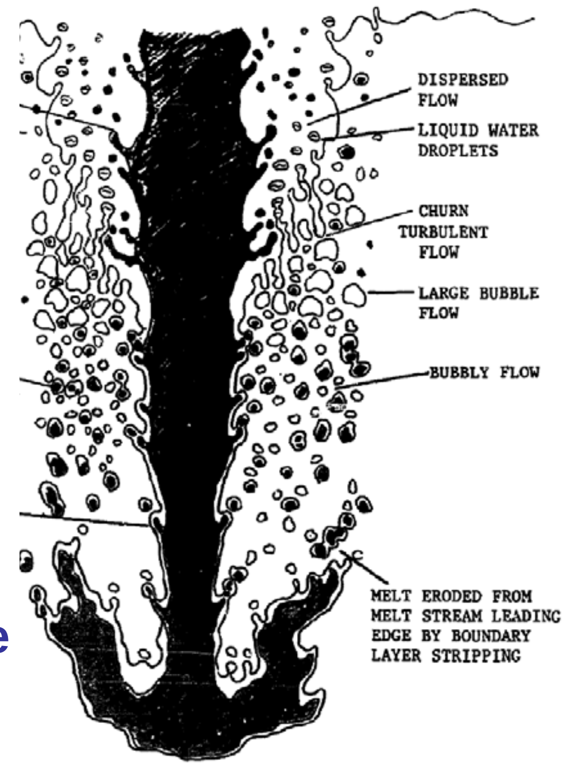
I)

?

How jet breakup mode can affect pre-deposition state of the debris?

Chu & Corradini, (1989)

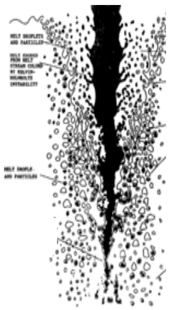
jet leading edge breakup

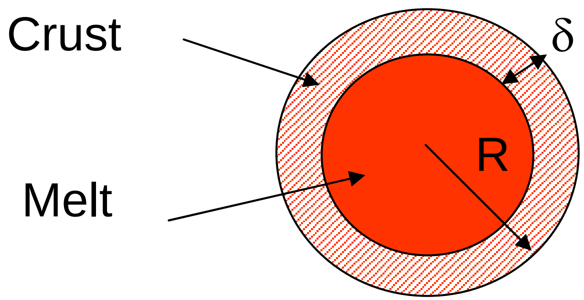


II)

Influence of Jet Breakup Mode on Pre-Deposited State of the Debris

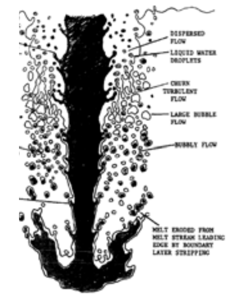
I)



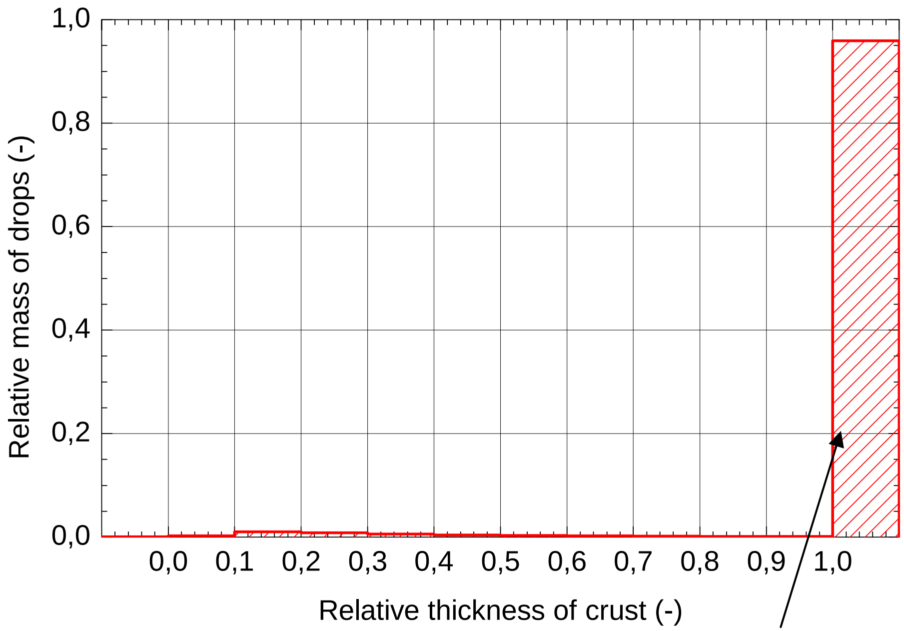


$$\delta_{rel} = \frac{\delta}{R}$$

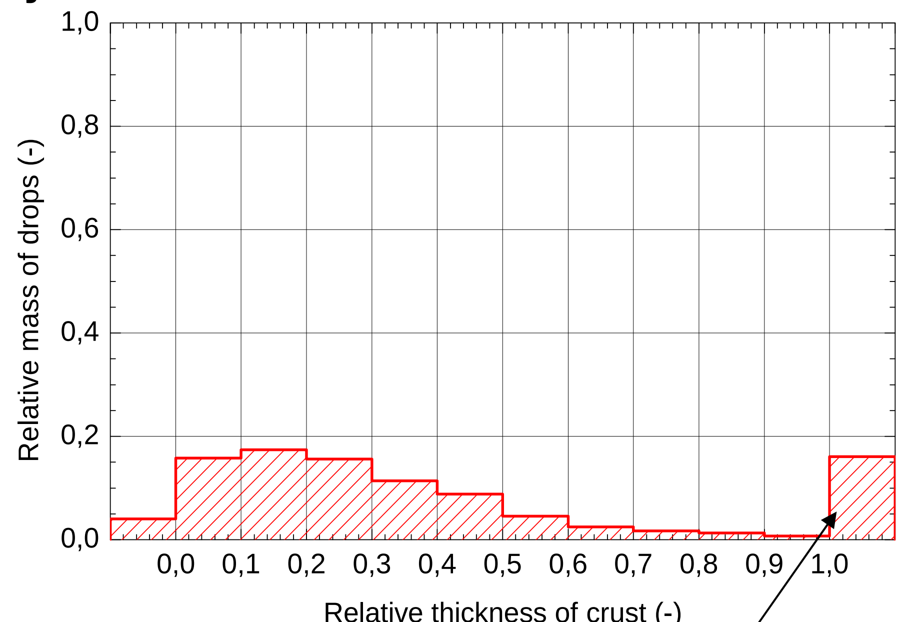
II)



Hpool=7 m, Djet=130 mm

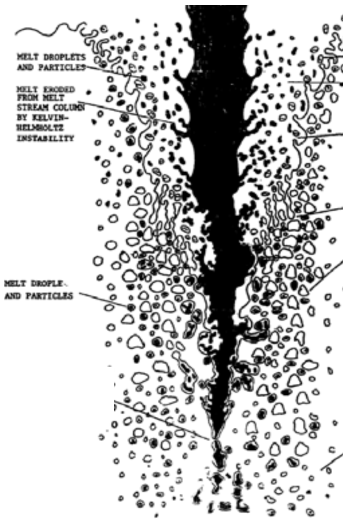


Particles are mostly completely solid

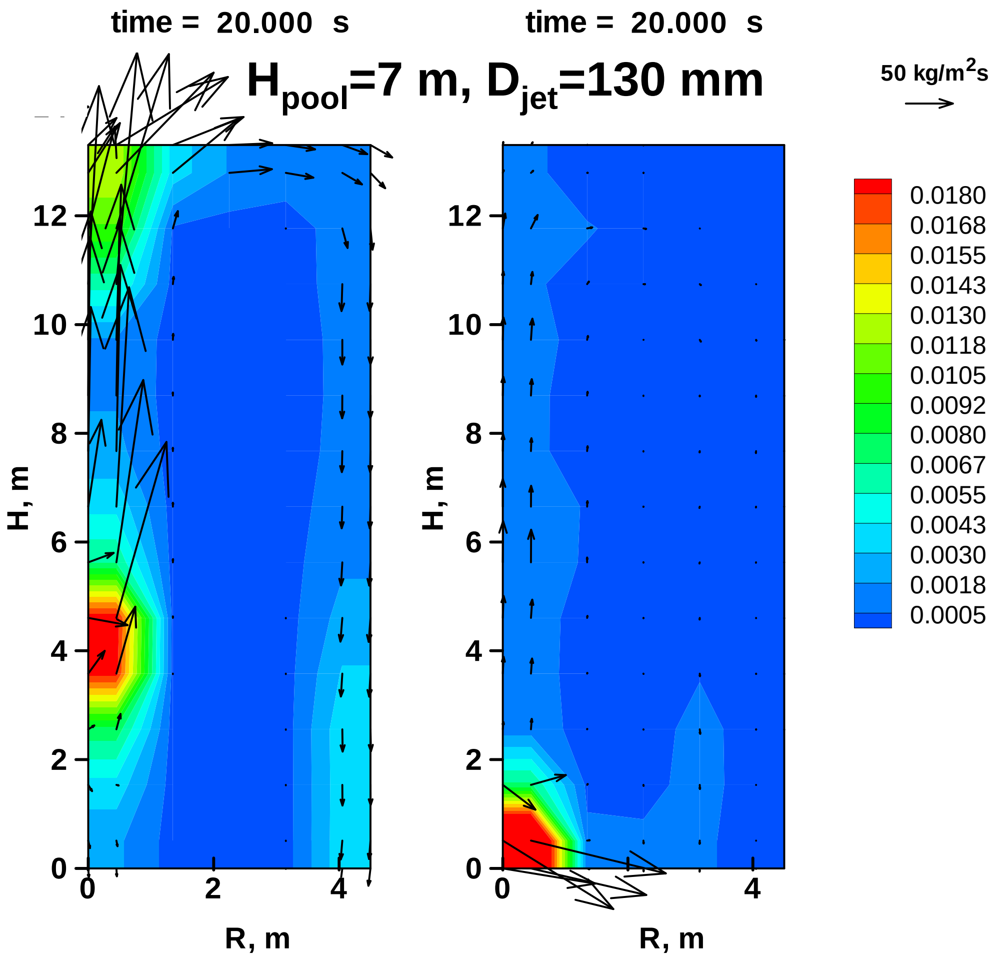


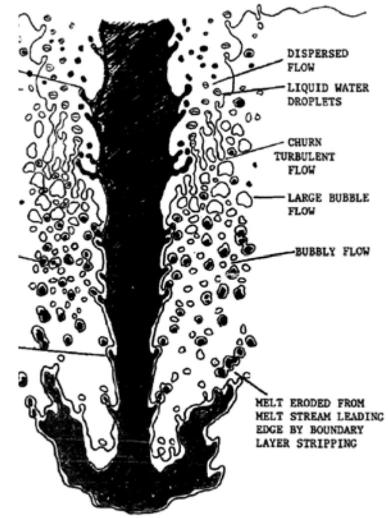
Small fraction of completely solid particles

Influence of Jet Breakup Mode on Pre-Deposition State of the Debris



I)





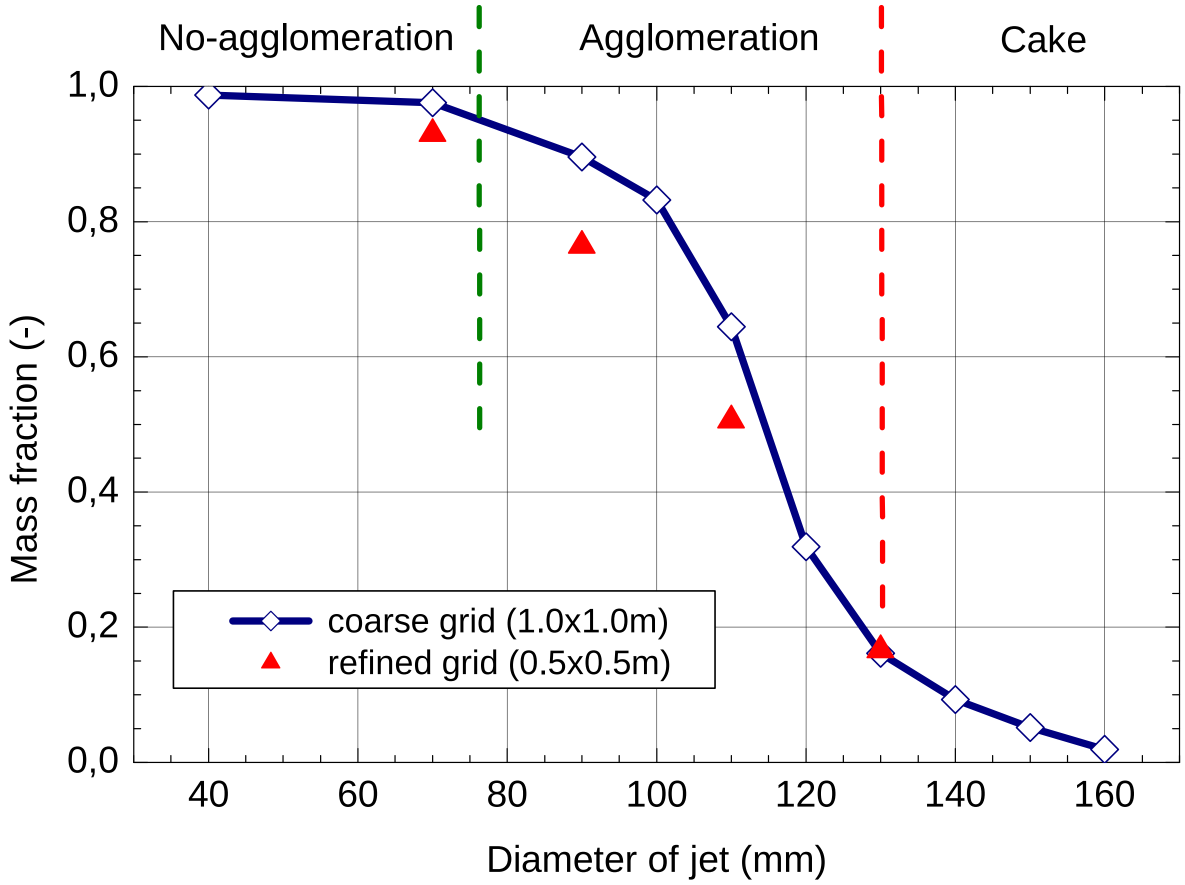
II)

Completely different dynamics of particle motion and solidification:

I) Particles move up first and solidify completely before they settle down

II) Particles move down and have no time to solidify completely before settlement

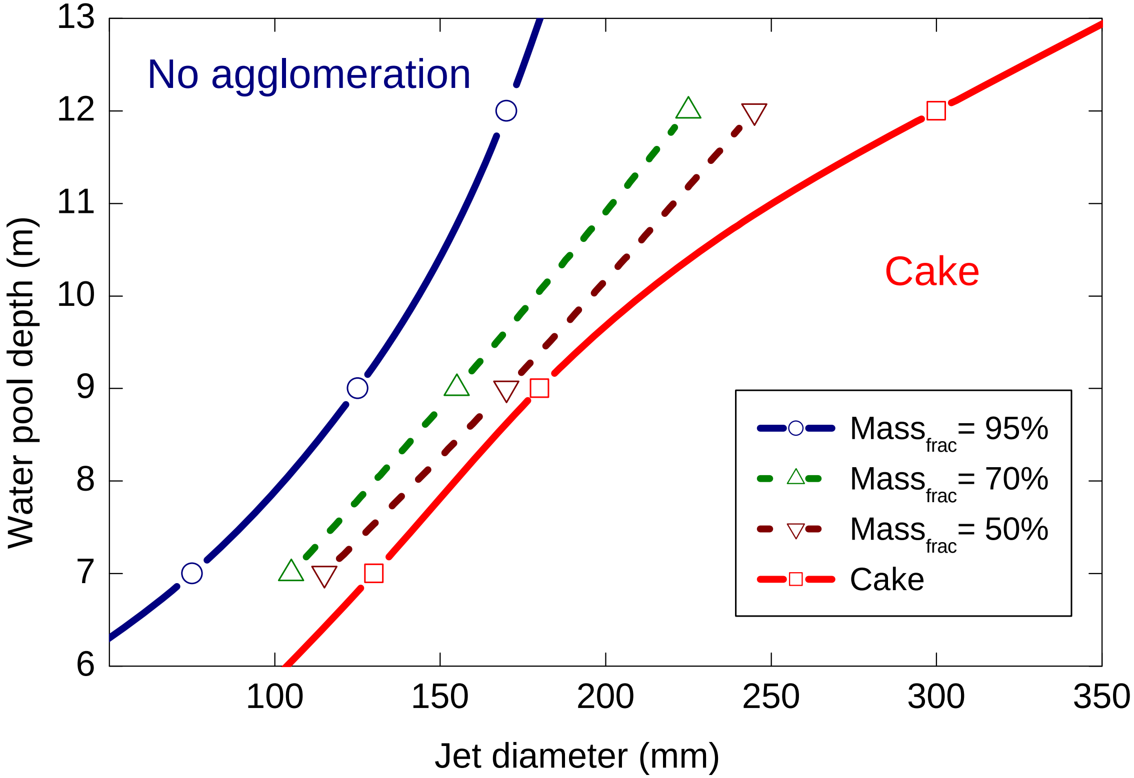
Particle Pre-deposited State



Rapid decrease of completely solid particle fraction with increase of the jet diameter

- **Comparison of mass fraction of completely solidified particles calculated with coarse and fine grids for pool depth Hpool =7 m.**

Agglomeration Mode Map



Pool parameters:

Water	saturated
Pressure	1 bar

Melt parameters:

Total mass	180 t
Jet release height	6 m
Initial temperature	3000 K
Composition	Eutectic

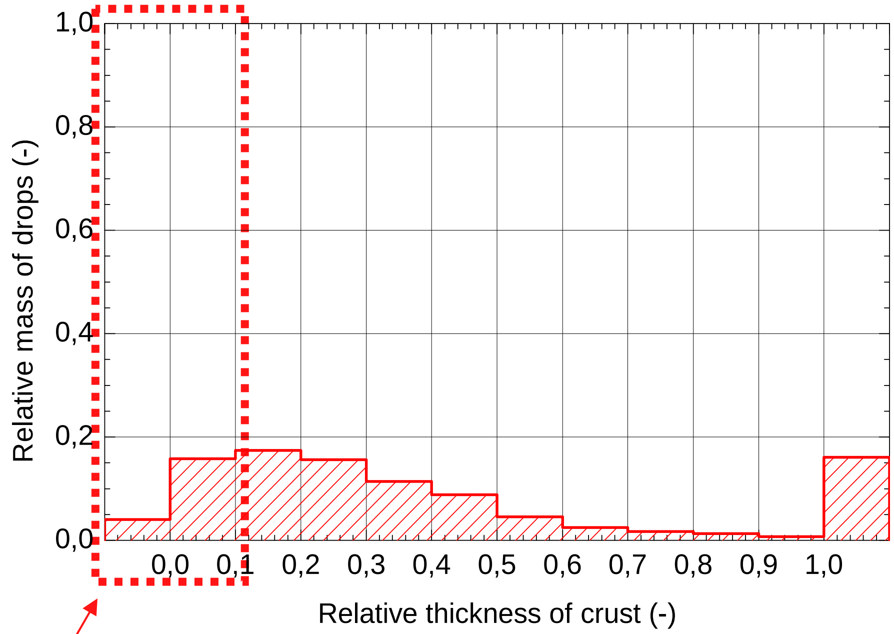
Agglomeration mode map based on the mass fraction of completely solidified particles.

Excessive conservatism!

Coefficient of Agglomeration

Assumptions:

- Agglomeration is a result of particle scale physical processes
- Crust thickness distribution gives initial conditions for onset of agglomeration
- Mass fraction of agglomerates m_{aggl} is proportional to the total mass fraction of completely liquid droplets and thin-crust particles m_{liq}



$\delta_{rel} = \delta_{crust} / R_{drop} \leq 0.1$

$$m_{aggl} = \alpha \cdot m_{liq} \qquad (1)$$

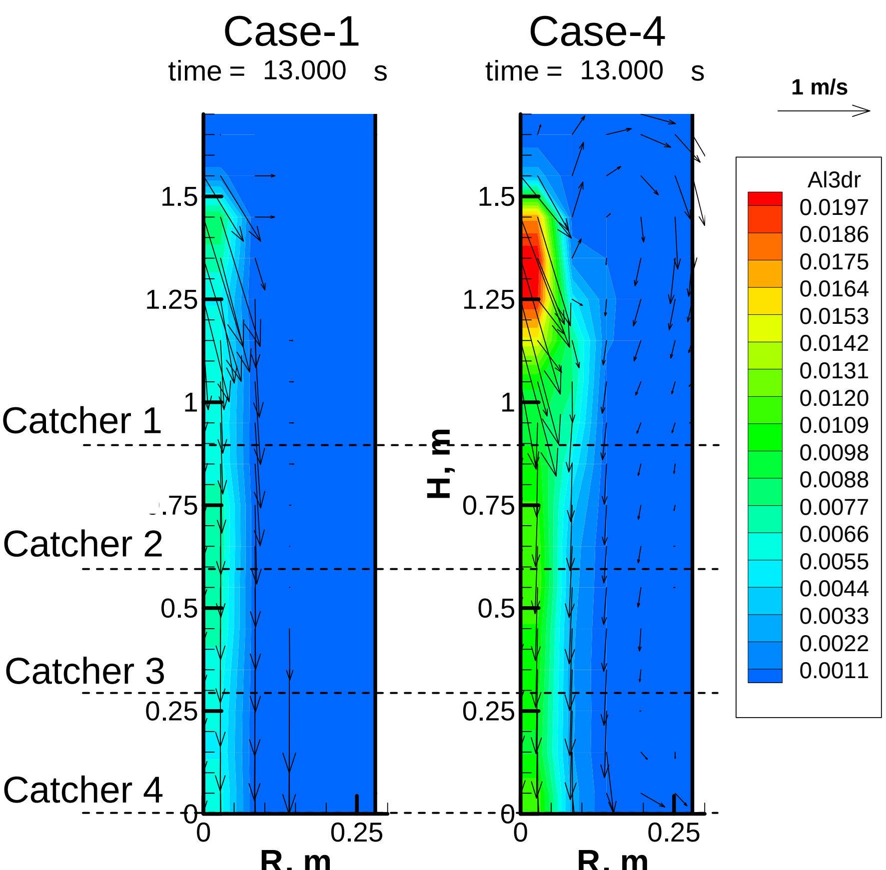
α – coefficient of agglomeration.

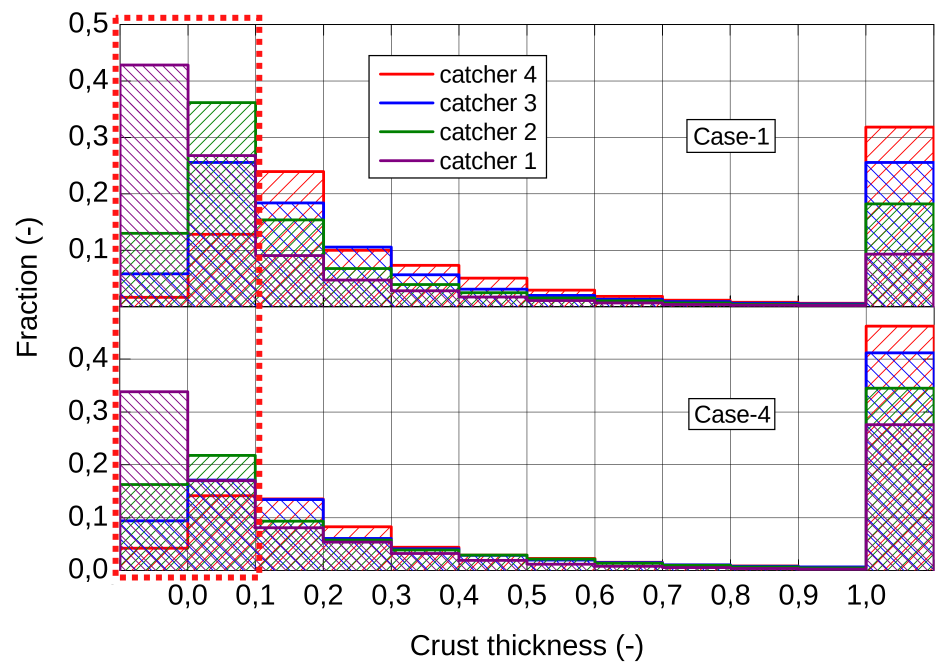
To take into account intrinsic uncertainties in the agglomeration phenomena α has to be rather conservative than best estimation

Water Subcooling and Jet Diameter Influence on Fraction of Liquid Particles m_{liq}

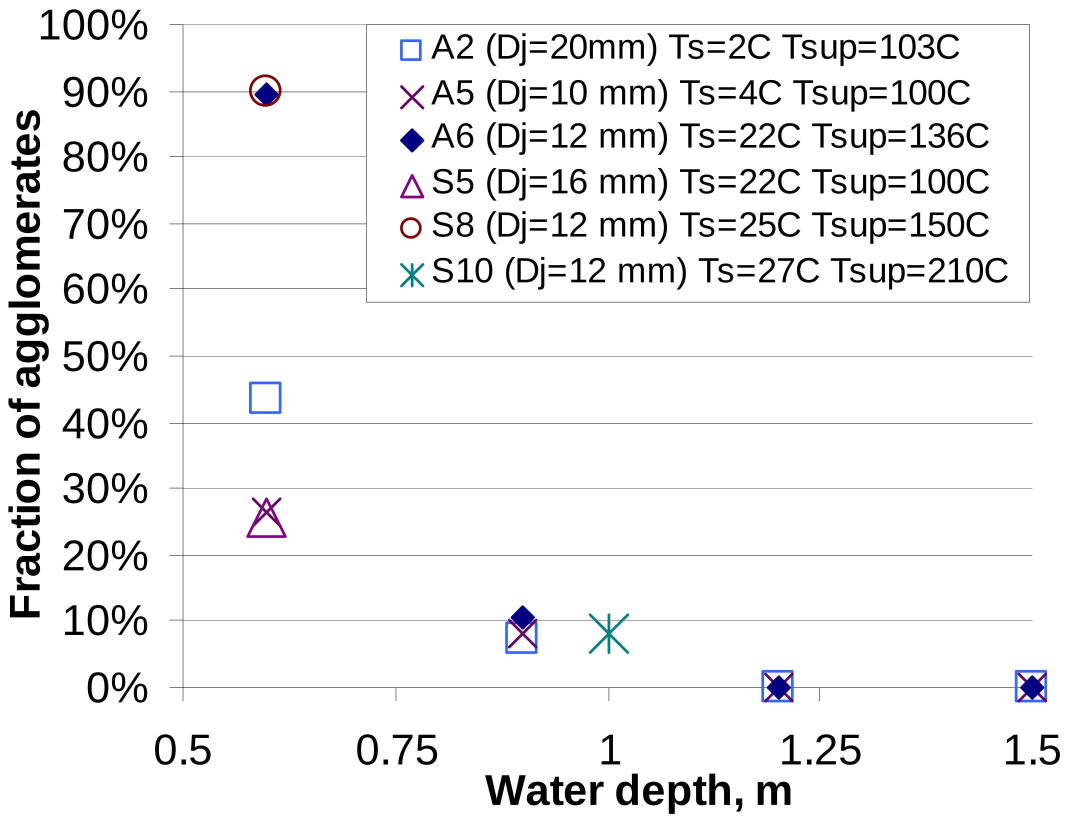
- Lower fraction of liquid particles in saturated coolant with 20 mm jet (Case-4)
- Considerable influence of steam generation on particle spreading in the tests section

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





DEFOR-A data: Fraction of Agglomerates m_{aggl} versus Pool Depth



Fraction of agglomerated debris is lower in tests with low water subcooling and with bigger jet diameter

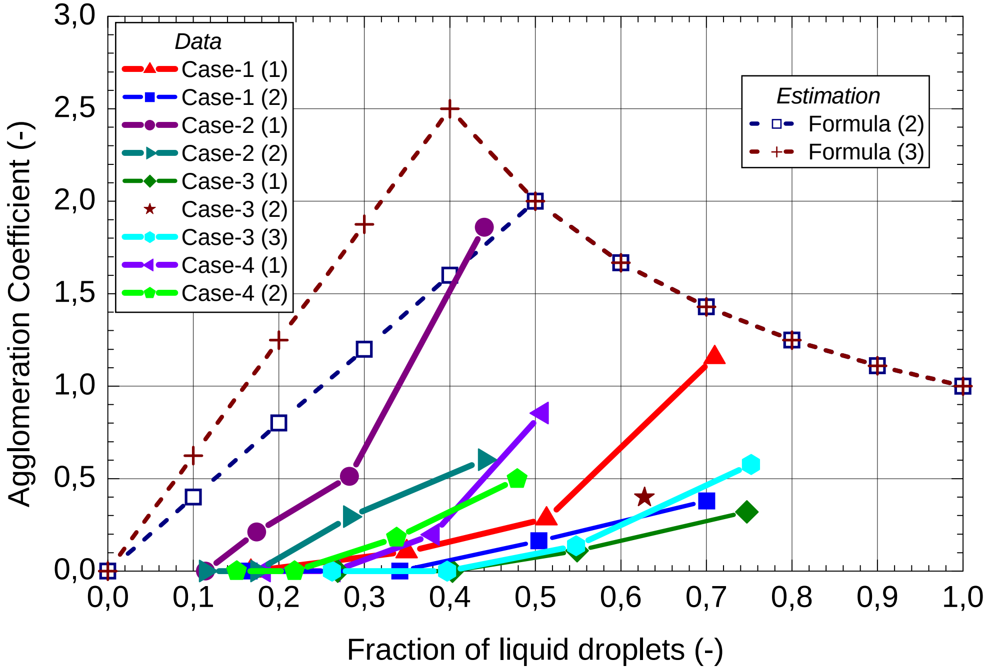
Fraction of agglomerated debris reduces rapidly with increasing of the pool depth

Conservative-Mechanistic Estimation of α

$$\alpha_{ij} = m_{agl}^{j} / m_{liq}^{i}$$

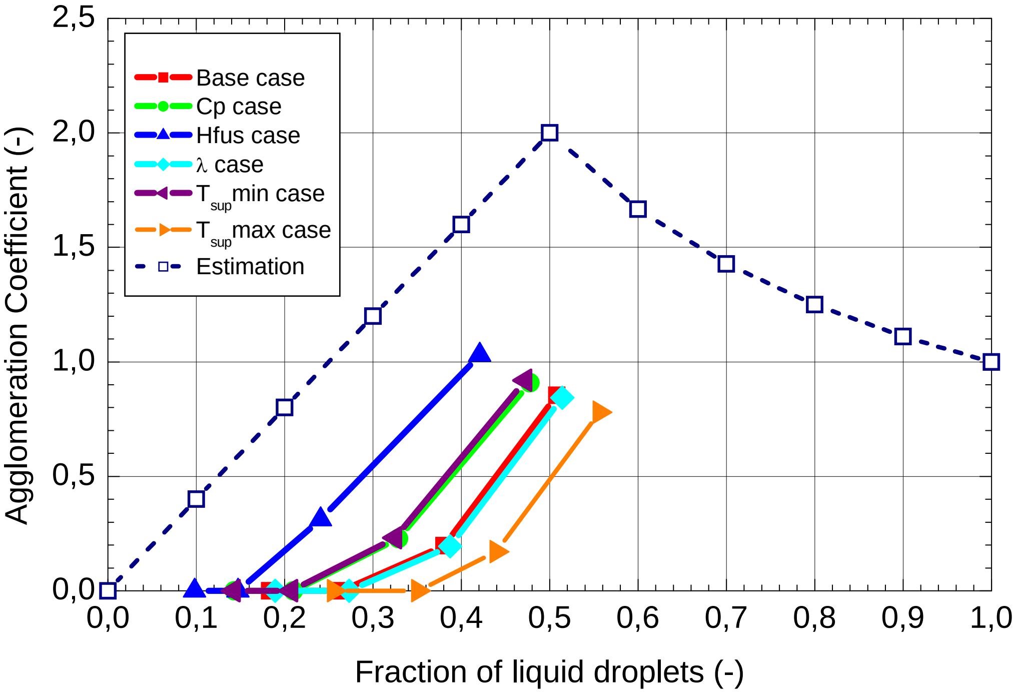
$$\alpha(m_{liq}) = \begin{cases} 4 \cdot m_{liq}, & m_{liq} \le 0.5 \\ 1/m_{liq}, & m_{liq} > 0.5 \end{cases} \quad (2)$$

$$\alpha(m_{liq}) = \begin{cases} 25/4 \cdot m_{liq}, & m_{liq} \le 0.4 \\ 1/m_{liq}, & m_{liq} > 0.4 \end{cases} \quad (3)$$



- Highest values for α come from simulations with smallest predicted fraction of liquid particles combined with biggest experimentally observed fraction of agglomerates
- Formula (2) provide enveloping estimation for α obtained from different combinations of experimental and analytical data
- In the limiting case of large mass fraction of liquid particles all solid particles will be glued with and eventually absorbed by the liquid particles
- Formula (3) is for sensitivity analysis

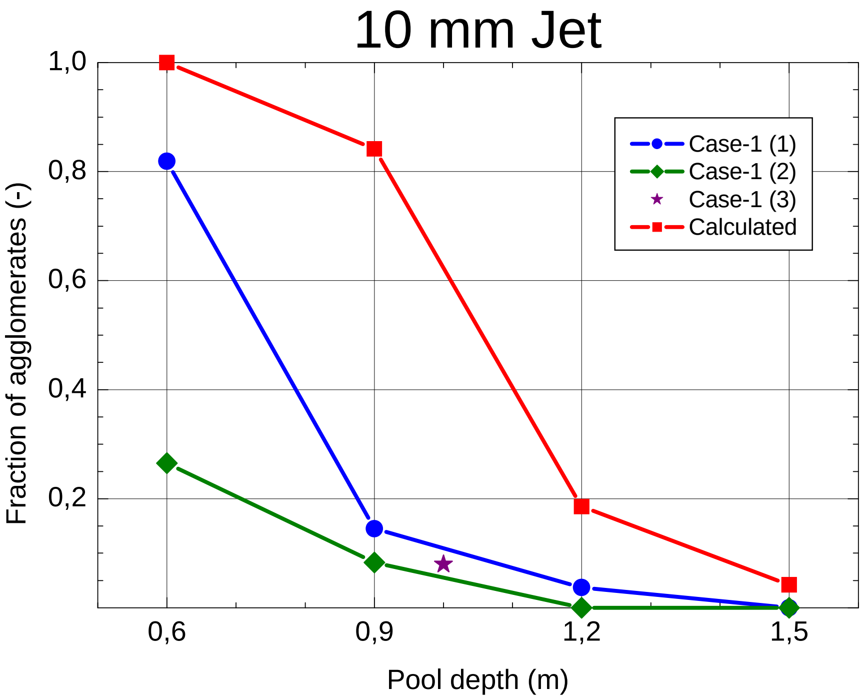
Sensitivity to Melt Properties

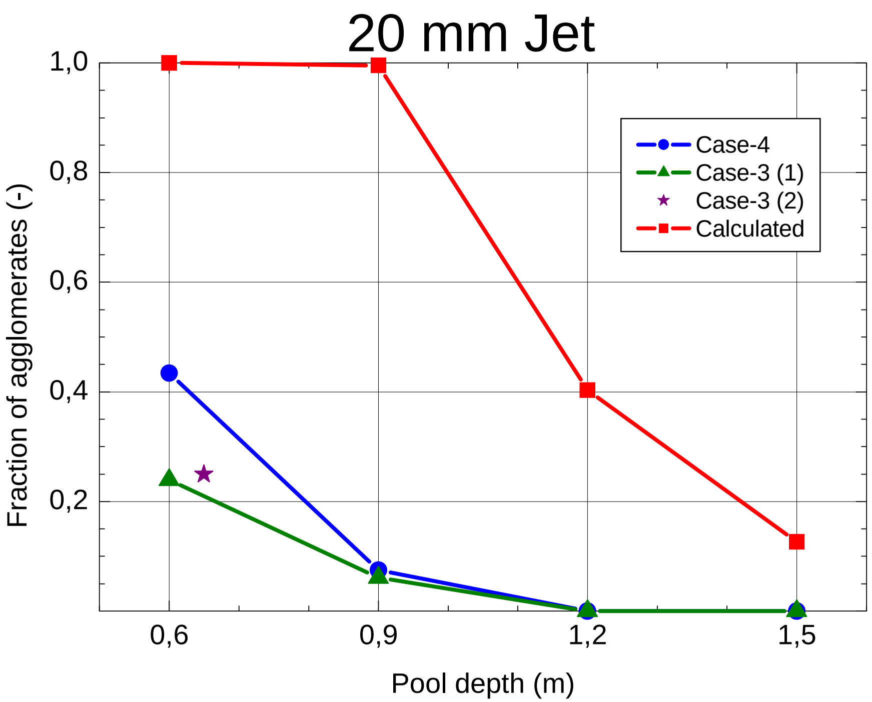


Case	Melt thermo-physical properties			
	C_p, J/(kg·K)	H_{fus}, J/kg	λ, W/(m·K)	T_{sup}, C
Baseline Case-4	**280**	**83**	**5.3**	**100**
"C_p case"	**200**	83	5.3	100
"H_{fus} case"	280	**25**	5.3	100
"λ case"	280	83	**3**	100
"T_{sup} min case"	280	83	5.3	**70**
"T_{sup} max case"	280	83	5.3	**150**

- Melt composition (properties) and melt superheat are intrinsically uncertain elements in the plant accident scenario.
- Results of sensitivity study to thermo-physical properties and melt superheat suggest that formula (2) provides bounding estimate coefficient of agglomeration

Conservative-Mechanistic Prediction of Agglomeration

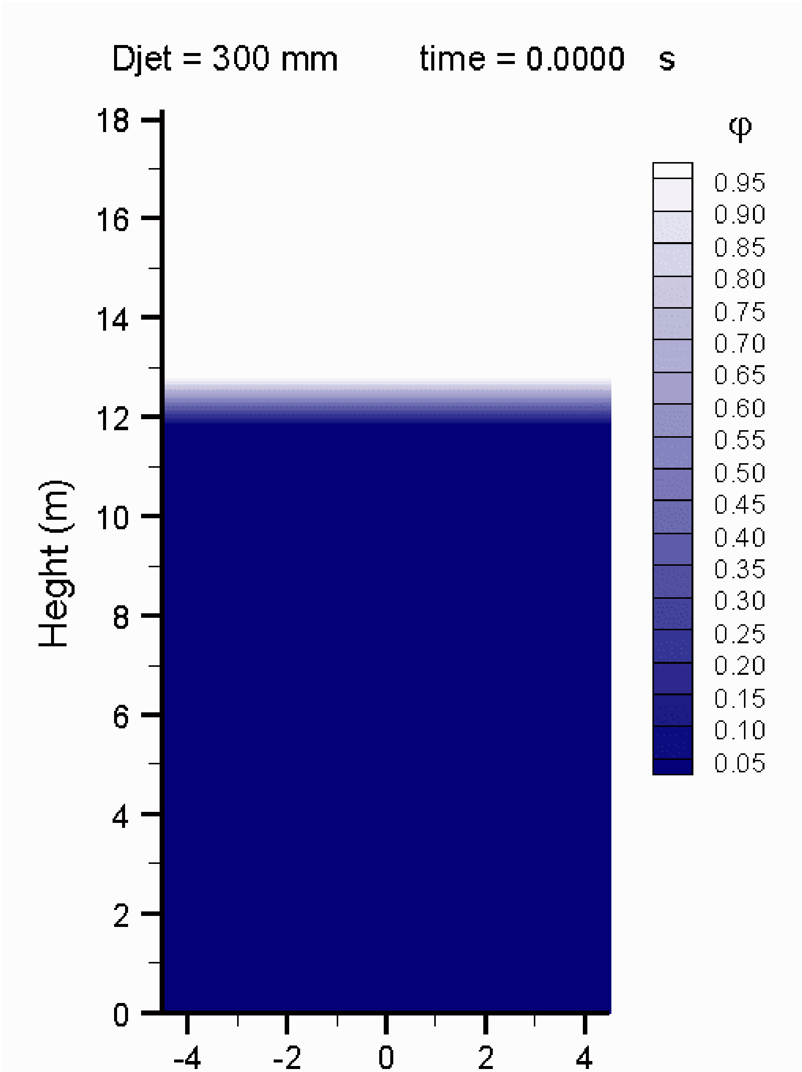




- Conservative-mechanistic approach provides both necessary degree of conservatism and, at the same time, takes into account mechanistic limiting mechanisms in the system behavior
  - Predicted values of the mass fraction of agglomerates are well above the experimentally measured ones
  - Predicted fraction of the agglomerated debris decreases rapidly with increasing of the water pool depth

Plant Scale Simulations

Sensitivity study of pre-deposited state of debris to FCI parameters



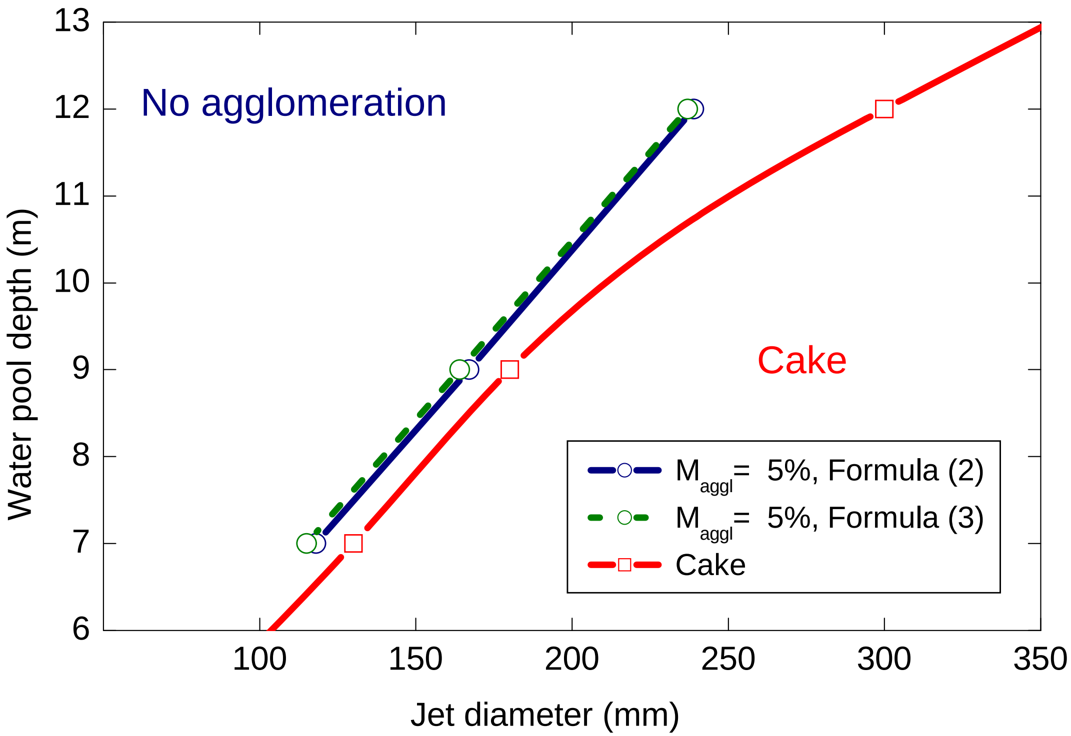
<u>Pool:</u>

Diameter	9 m
<u>Depth</u>	**7-12m**
Pressure	1 bar
Water	saturated

<u>Melt:</u>

Total mass	180 t
<u>Jet diameter</u>	**70 – 300 mm**
Temperature	3000 K
Composition	Eutectic corium

Agglomeration Map Sensitivity to α



Pool parameters:

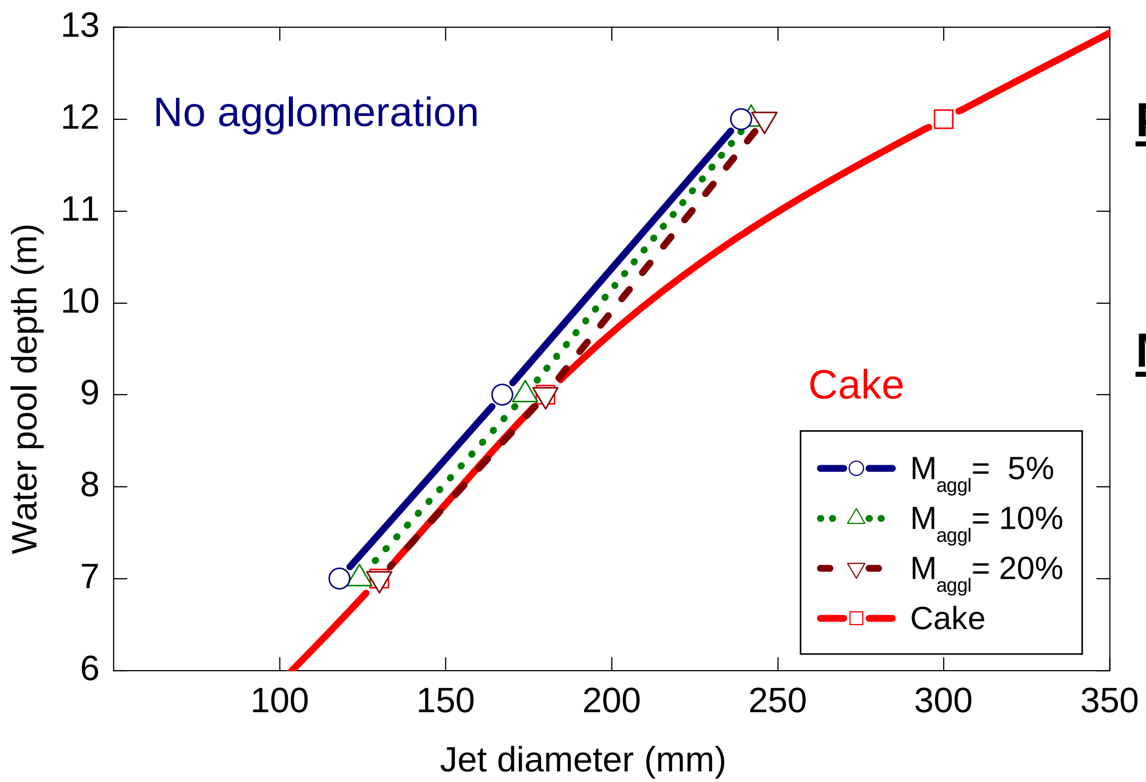
Water	saturated
Pressure	1 bar

Melt parameters:

Total mass	180 t
Jet release height	6 m
Initial temperature	3000 K
Composition	Eutectic

- Results of prediction are robust and insensitive to small variations of bounding closure for α

Agglomeration Mode Map



Pool parameters:

Water	saturated
Pressure	1 bar

Melt parameters:

Total mass	180 t
Jet release height	6 m
Initial temperature	3000 K
Composition	Eutectic

- Conservative-mechanistic quantification of the agglomeration mode map and mass fraction of agglomerated debris

Summary and Outlook

- Approach for conservative-mechanistic quantification of the debris agglomeration mode map is proposed
- Simulation data confirms that it is possible in principle to achieve completely fragmented debris bed within the present design of Swedish BWRs
- 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

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Experimental

1. Karbojian A., Ma W.M., Kudinov P., Davydov M., Dinh N., “A scoping study of debris formation in DEFOR experimental facility”, 15th International Conference on Nuclear Engineering, Nagoya, Japan, April 22-26, 2007, Paper number ICON15-10620.
2. Kudinov P., Karbojian A., Ma W.M., and Dinh T.-N., “An experimental study on debris formation with corium stimulant materials,” Proc. ICAPP’08, Anaheim, CA USA, June 8–12, 2008, paper 8390.
3. Karbojian A., Ma W., Kudinov P., and Dinh T.-N., “A Scoping Study of Debris Bed Formation in the DEFOR Test Facility”, Nuclear Engineering and Design, 239, 2009, 1653-1659.
4. Kudinov P., Karbojian A., Tran C.-T., “Experimental Investigation of Melt Debris Agglomeration with High Melting Temperature Simulant Materials,” Proceedings of ISAMM-2009, Böttstein, Switzerland, October 26 - 28, 2009.
5. Kudinov P., Karbojian A., Ma W., and Dinh T.-N. “The DEFOR-S Experimental Study of Debris Formation with Corium Simulant Materials,” Nuclear Technology, 2009 (accepted, in press).

Analytical

6. Kudinov P., Karbojian A., Ma W.M., Davydov M., and Dinh T.-N., “A Study of Ex-Vessel Debris Formation in a LWR Severe Accident”, Proceedings of ICAPP 2007, Nice, France, May 13-18, 2007, Paper 7512.
7. Kudinov P. and Dinh T.-N., “An analytical study of mechanisms that govern debris packing in a LWR severe accident”, The 12th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-12), Sheraton Station Square, Pittsburgh, Pennsylvania, U.S.A. September 30-October 4, 2007. Paper 247.
8. Kudinov P. and Dinh T.-N., “A Computational Study of Debris Bed Formation,” *ANS Transactions*, 2008, paper 193463.
9. Yakush S., Kudinov P., and Dinh T.-N., “Modeling of Two-Phase Natural Convection Flows in a Water Pool with a Decay-Heated Debris Bed,” Proc. ICAPP’08, Anaheim, CA USA, June 8–12, 2008, paper 8409.
10. Dombrovsky L.A., Davydov M.V., and Kudinov P., Thermal radiation modeling in numerical simulation of melt-coolant interaction, Proc. Int. Symp. Adv. Comput. Heat Transfer (CHT-08), May 11–16, 2008, paper 155.
11. Dombrovsky L.A., Davydov M.V., Kudinov P., “Thermal radiation modeling in numerical simulation of melt-coolant interaction,” Comp. Therm. Sci. 1 (1) 2009, pp. 1-35.
12. Kudinov P., Davydov M., “Development of Ex-Vessel Debris Agglomeration Mode Map for a LWR Severe Accident Conditions,” The 17th International Conference on Nuclear Engineering, July 12-16, 2009, Brussels, Belgium, Paper ICONE17-75080.
13. Yakush S., Kudinov P., and Dinh T.-N., “Multiscale Simulations of Self-organization Phenomena in the Formation and Coolability of Corium Debris Bed,” Proc. The 13th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-13), September 27-October 2, 2009. Kanazawa City, Ishikawa Prefecture, Japan, Paper N13P1143.
14. Yakush S. and Kudinov P., “Transient Phenomena of Ex-vessel Debris Bed Formation in a LWR Severe Accident,” ANS Transactions, 2009, paper 210830.
15. Kudinov P. and Davydov M., “Approach to Prediction of Melt Debris Agglomeration Modes in a LWR Severe Accident,” Proceedings of ISAMM-2009, Böttstein, Switzerland, October 26 - 28, 2009.
16. Yakush S. and Kudinov P., “Simulation of Ex-Vessel Debris Bed Formation and Coolability in a LWR Severe Accident,” Proceedings of ISAMM-2009, Böttstein, Switzerland, October 26 - 28, 2009.
17. Kudinov P. and Kudinova V., “Influence of Water Subcooling on Fracture of Melt Debris Particle,” ANS Transactions, 2009, paper 210646.

OECD SERENA: A Fuel Coolant Interaction Programme (FCI) devoted to reactor case

KAERI

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Commissariat à l'Énergie Atomique, Cadarache, France

S.W. Hong
Korea Atomic Energy Research Institute, Korea

Implementation of Severe Accident Management Measures (ISAMM-2009)

Paul Scherrer Institute Villigen, Switzerland October 26 - 28, 2009

ISO 9001
ISO 14001

CADARACHE

Outlines

- **Fuel Coolant Interaction: state of the art**

- **SERENA-Phase 1**
  - **In-vessel reactor case**
  - **Ex-vessel reactor case**
  - **Conclusion**

- **SERENA-Phase 2**
  - **Main objectives**
  - **Organisation**
  - **Experimental facilities: TROI –KROTOS**
  - **Experimental grid**

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State of the art

- **More than 30 years of a complex physic study…**
  - **Bottom-up approach from elementary mechanism investigation at microscopic scale ⇨ TREPAM, MICRONIS…**
  - **Top-down approach from the measurement of the consequences at macroscopic scale ⇨ KROTOS, FARO, ANAIS, TROI…**

- **Evaluation from thermodynamic approach**
  - **Bounding conversion energetic efficiency, but the shape of the pressure load is not calculated ⇨ too much conservatism in the approach…**

- **Specific simulation tools for FCI**
  - **Two steps : pre-mixing calculation and explosion calculation**
  - **Still on-going validation related to sensitivity problem at initial condition for explosion ⇨ phase distribution within pre-mixing**
  - **Coupled phenomena not easy to discriminate : heat transfer coefficient, heat transfer surface, partition between steam and water…**
  - **Material effects still unexplained : corium efficiency ~0.1% and ~1% for alumina...**

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State of the art

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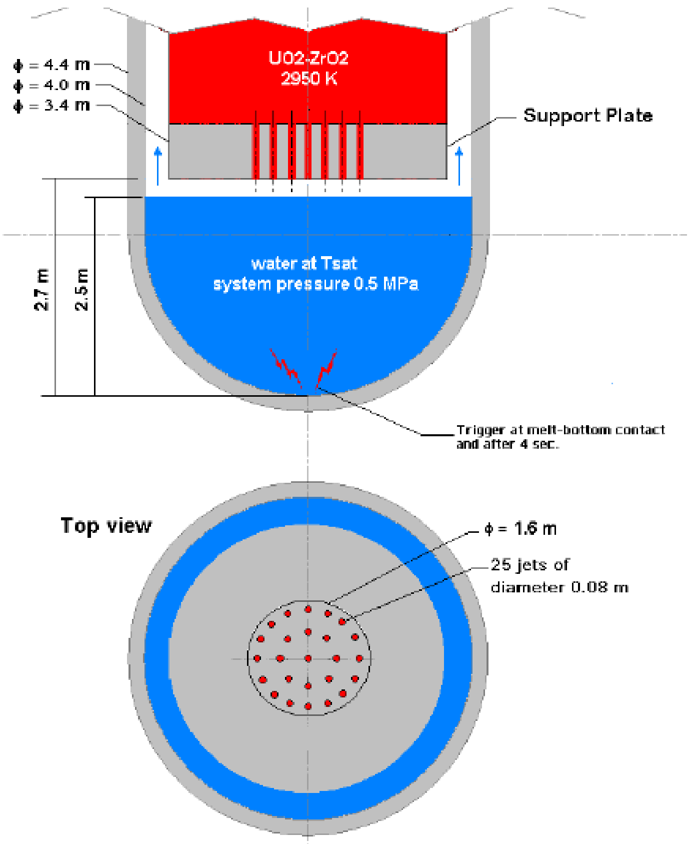
- **OCDE SERENA program**
  - **Code benchmark applied to reactor situation to evaluate the remaining uncertainties**
  - **Approach**
    - **Better fit of models against experimental results for pre-mixing and explosion phases**
    - **In vessel and ex-vessel reactor applications**
    - **Results analyze and comparison with admissible margins**
    - **Conclusion and recommendation in terms of R&D**

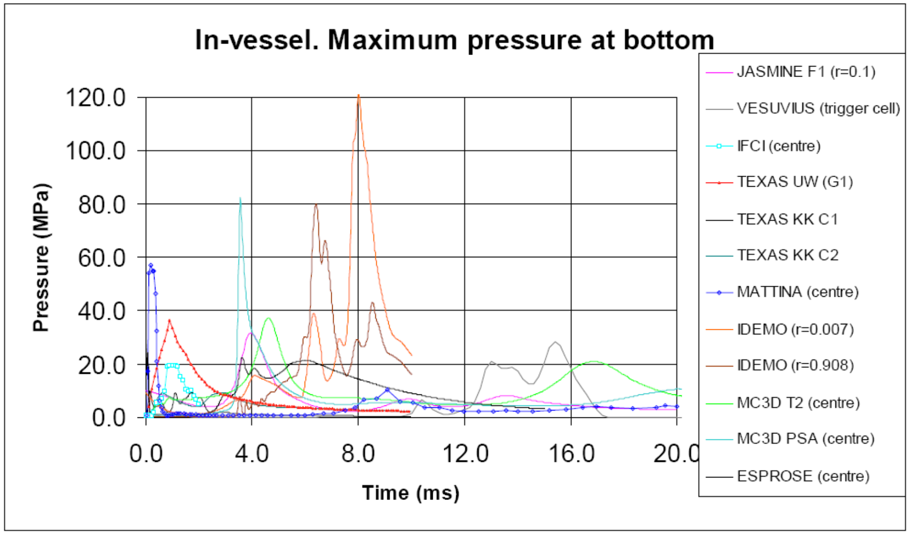
⇨ **Close issue for in-vessel situation**

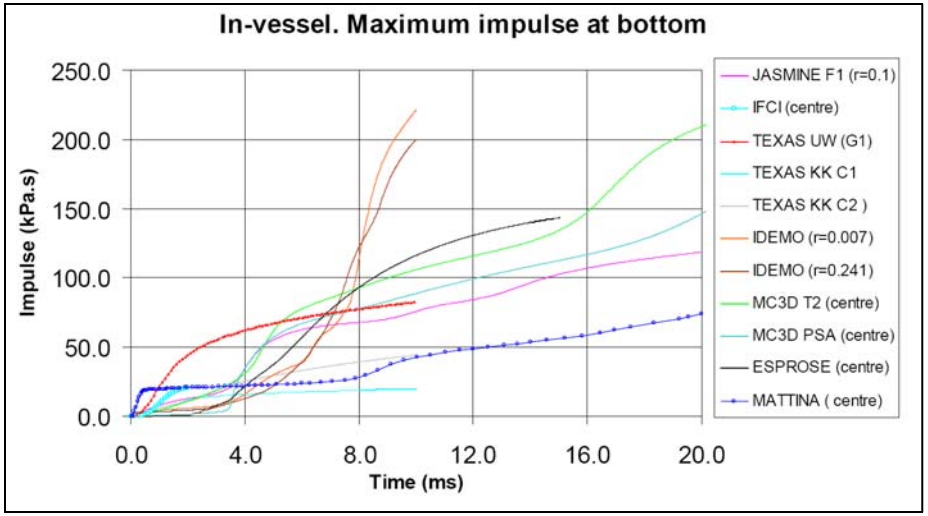
⇨ **For ex-vessel situation the results dispersion does not allow safety margins evaluation**

Serena (phase 1): In-vessel reactor case



In-vessel





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Serena (phase 1): Ex-vessel reactor case

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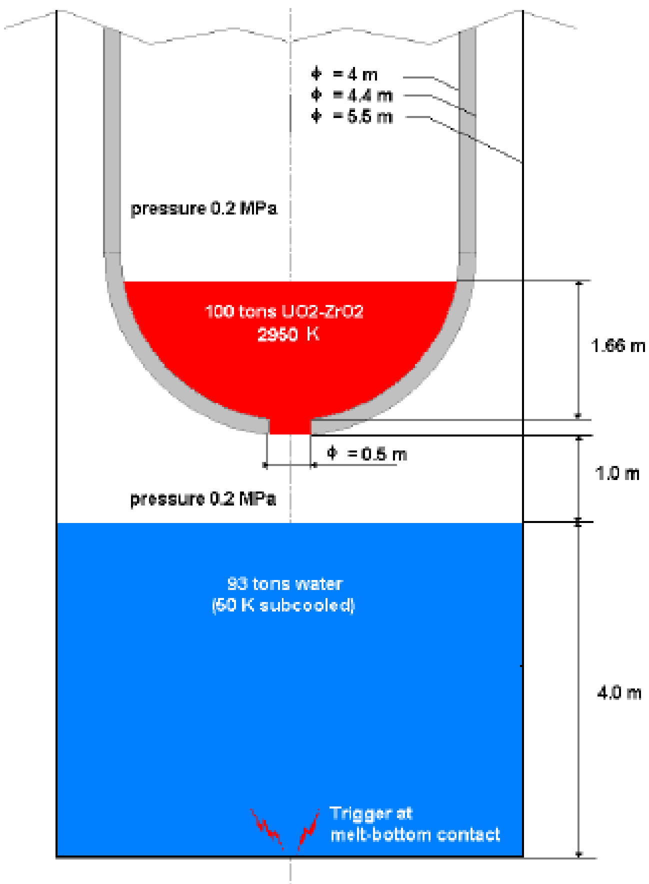

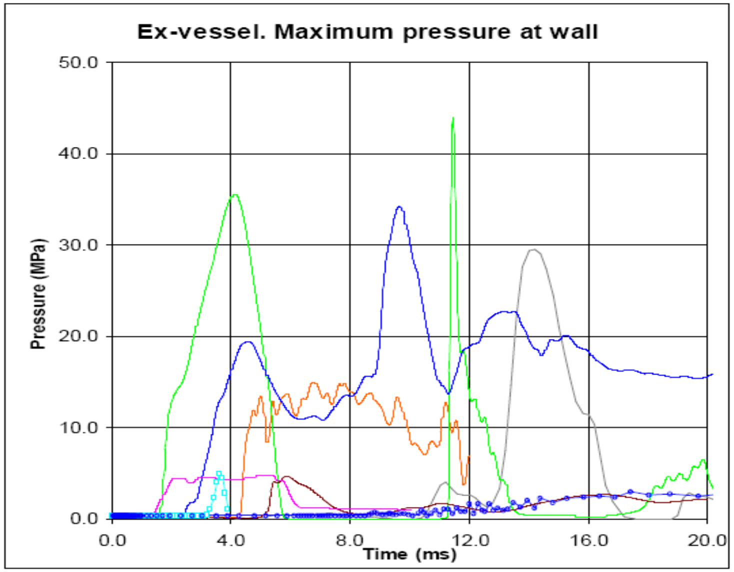

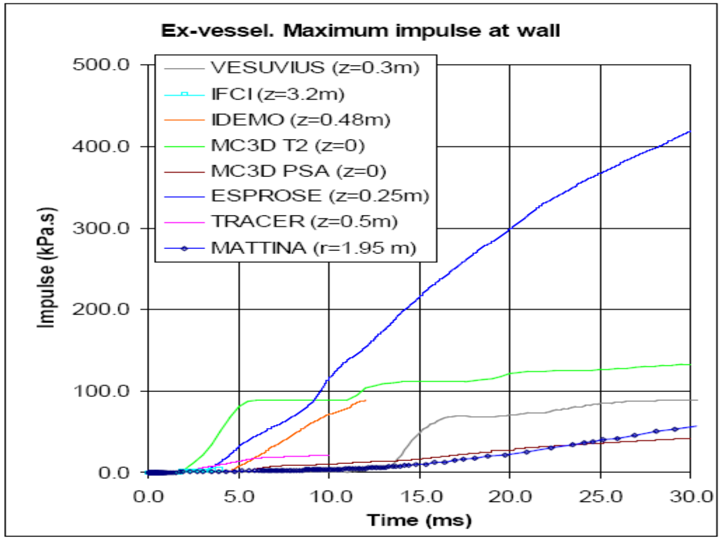

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Serena (phase 1): Main conclusions

- FCI code applications to reactor situations showed that:
  - In the absence of pre-existing loads, in-vessel steam explosion would not challenge the integrity of the vessel
  - Damage to the cavity is to be expected for ex-vessel explosion
    - May challenge the integrity of the containment
    - But, the level of the loads cannot be predicted due to a large scatter of the results
    - Action is required to bring the scatter of the predictions to acceptable levels

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Safety significance of ex-vessel SE

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- Flooding of reactor cavity is considered as SAM measures for new PWRs like APR-1400 and AP1000 to assure IVR of core melt. Flooding of drywell is considered for BWR also
- Flooding of reactor cavity is not considered for existing PWR as SAM strategy. However, presence of water in the reactor cavity, caused by the use of spray and/or by a primary circuit rupture, cannot be excluded
- Consequently, there is a need to be able to establish containment safety margins to ex-vessel explosion
- **This is the scope of the SERENA Phase 2 program**

Major uncertainties

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1. **The component distribution in the pre-mixture at the time of the explosion, especially the level of void**
   – Induced by large uncertainties that affect existing experimental data in the absence of detailed information of the pre-mixing zone internals
     - Only global void fraction available form level swell measurements
2. **The explosion behaviour of corium melts**
   - What are the very reasons why corium melts exhibit low energetics?
   - Impossibility to obtain explosive melt-rich, void-poor mixtures (due to, e.g., density, temperature, hydrogen production,…)?
   - Effect of corium properties directly on the energetics (effect of viscosity; thermodynamic, chemical and mechanical behaviour,…)?
   - Can this behaviour be generalised?

Serena (phase 2)

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- **Purpose: To carry out confirmatory research required to reduce uncertainties in Fuel Coolant Interaction phenomena to acceptable level for risk assessment**
- **Expected Outcome:**
  - **Remove uncertainties on void distribution by providing detailed data of internal structure of pre-mixing**
  - **Confirm low explosivity of corium**
  - **Both by using a large spectrum of corium melts and conditions in KROTOS and TROI facilities**
  - **Bring the scatter of the predictions for ex-vessel steam explosion to acceptable limits for risk evaluation of containment failure**
  - **By improving modelling and code performance on the basis of the new data**

Serena (phase 2)

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- **Operating Agents: CEA and KAERI**
- **4 Year Project (Started late 2007)**
- **Participating organizations: 16 members**
  AECL, CEA, EDF,GRS,IKE, IRSN, JNES, JSI, KAERI, KINS, KMU, KTH, NRC,PSI, Tractebel, VTT

Serena (phase 2): organisation

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- **Analytical Program**
  - Increasing the capabilities of FCI models/codes for use in reactor analyses by complementing the work performed in Phase-1 through integrating the results of the Phase-2 Experimental program
  - Work oriented at fitting for purpose for safety analyses and elaboration of the major effects which reduce the explosion strength
- **Applied codes**
  - IKEMIX/IKEJET+IDEMO: IKE
  - JASMINE: JNES
  - MC3D: AECL, CEA, IKE, JSI, KAERI, KINS, Tractebel
  - TEXAS-V: UWM, VTT

Serena (phase 2): TROI facility

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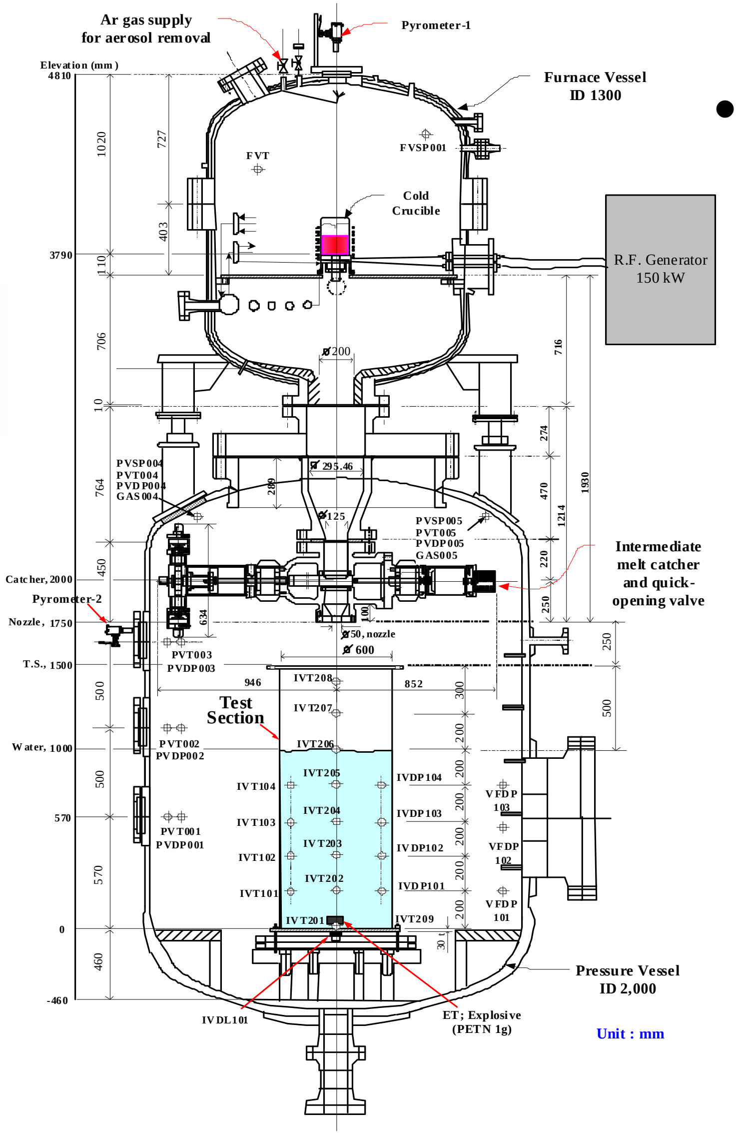
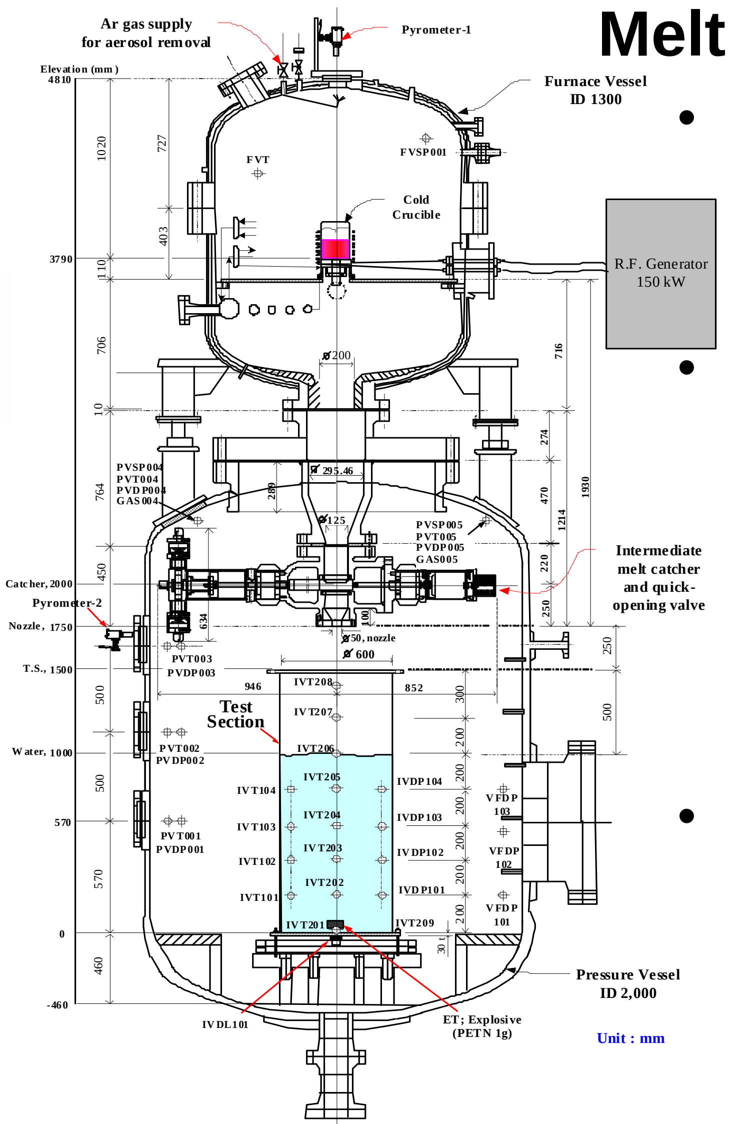


- **Furnace vessel**
  - **Cold crucible melting method**
  - **Pressure vessel**
  - **Quick-opening valve with an intermediate melt catcher**
  - **Wide interaction chamber**
  - **Trigger device**

Serena (phase 2): TROI facility



Melt Release

- **At a required melt temperature, a plug (Temporal device to plug melt release hole of the cold crucible) is removed.**
- **And a puncher is actuated pneumatically to perforate the crust, formed at the bottom of the crucible.**

  Then, the melt is discharged into an intermediate catcher by gravity.

  - **Melt is accumulated in the intermediate melt catcher for around 2 seconds.**
- **Melt is delivered into the water in the interaction vessel by opening the valve located below the melt catcher.**

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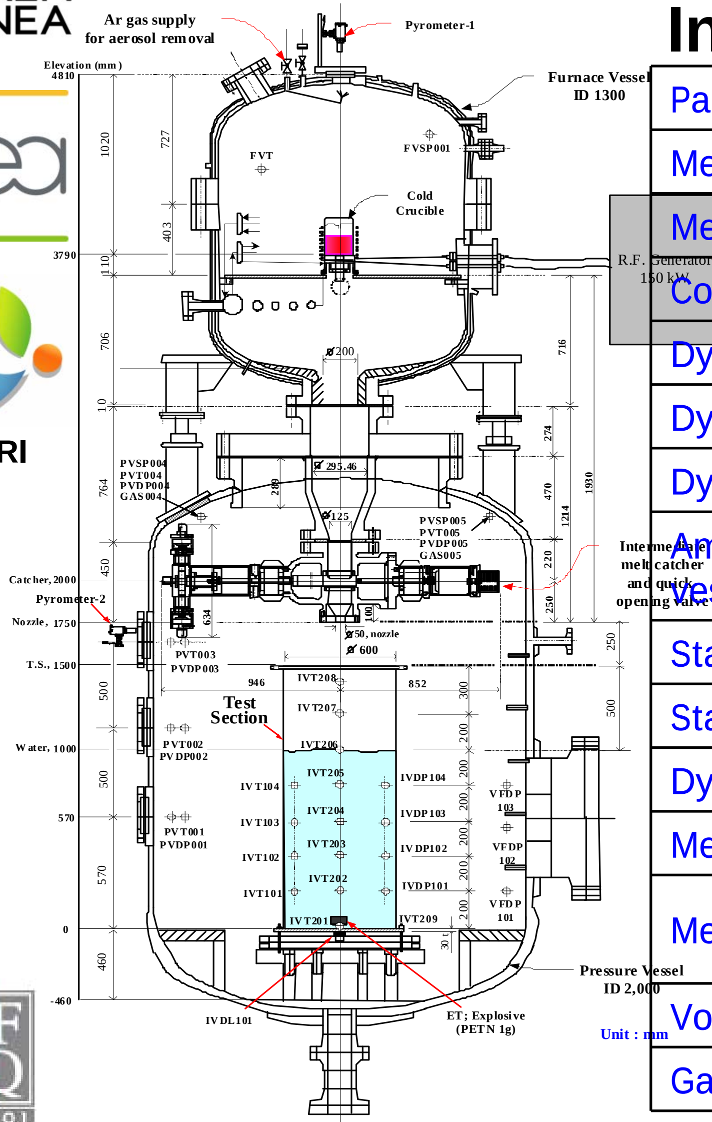
Serena (phase 2): TROI facility



Instrumentation

Parameter	Sensing location
Melt temperature 1	Top window
Melt temperature 2	Melt delivery path
Coolant temperature	IVT101 ~ IVT104
Dynamic pressure in the coolant (1)	PIVDP101~ PIVDP104
Dynamic pressure in the coolant (2)	KIVDP101 ~KIVDP104
Dynamic load at the test section bottom	IVDL101
Ambient temperature in the pressure vessel	PVT001 ~ PVT005
Static pressure in the furnace vessel	FVSP001
Static pressure in the pressure vessel	PVSP004, PVSP005
Dynamic pressure in the pressure vessel	PVDP004, PVDP005
Melt velocity in water	IVT201 ~ IVT209
Melt entry velocity (video)	at the above of the interaction vessel
Void fraction	VFDP101 ~VFDP103
Gas sampling for hydrogen detection	GAS005

Serena (phase 2): KROTOS facility

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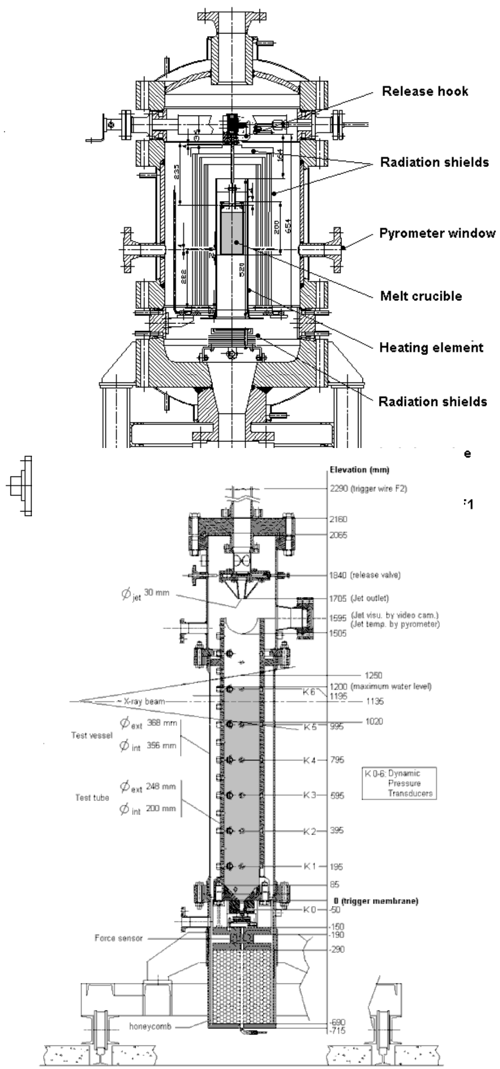

- **Furnace vessel**
  - **Hot crucible melting method**
  - **Pressure vessel**
  - **Puncher**
  - **Trigger device**

Serena (phase 2): KROTOS facility

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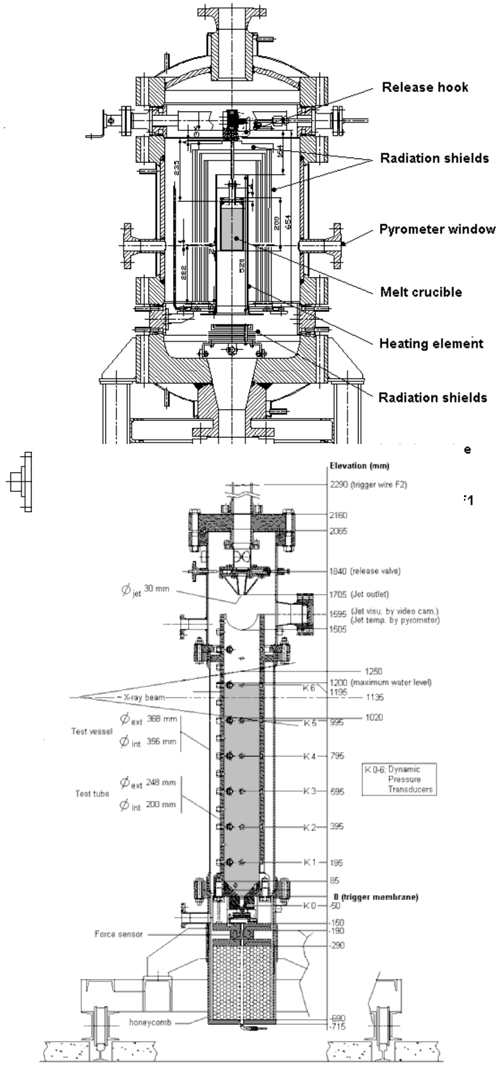

- **Melt release**
  - **Gravity fall of the crucible**
  - **Break-up on puncher**
  - **Opening of the crucible**
  - **Release of the corium**
  - **Stop of the corium**
  - **Release of the corium at 0-velocity**

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Serena (phase 2): KROTOS facility

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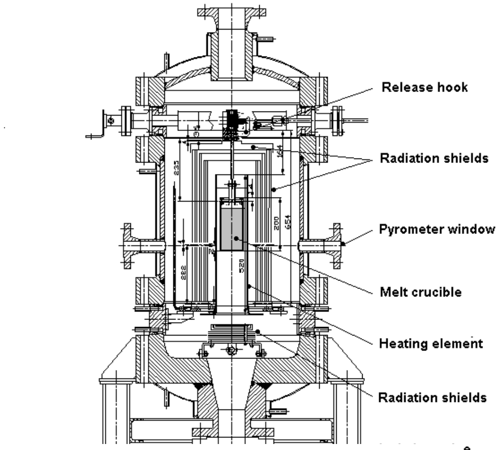
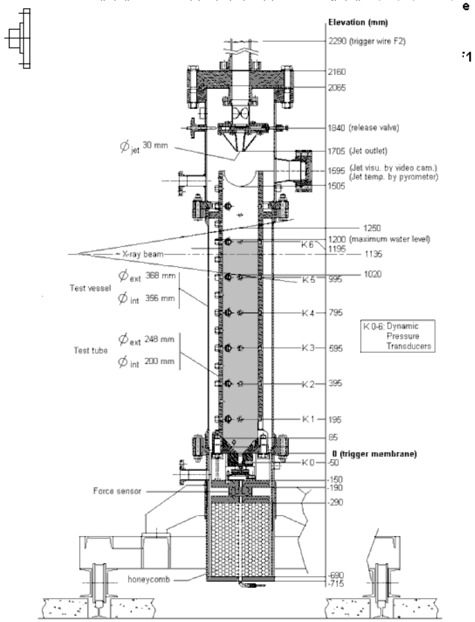
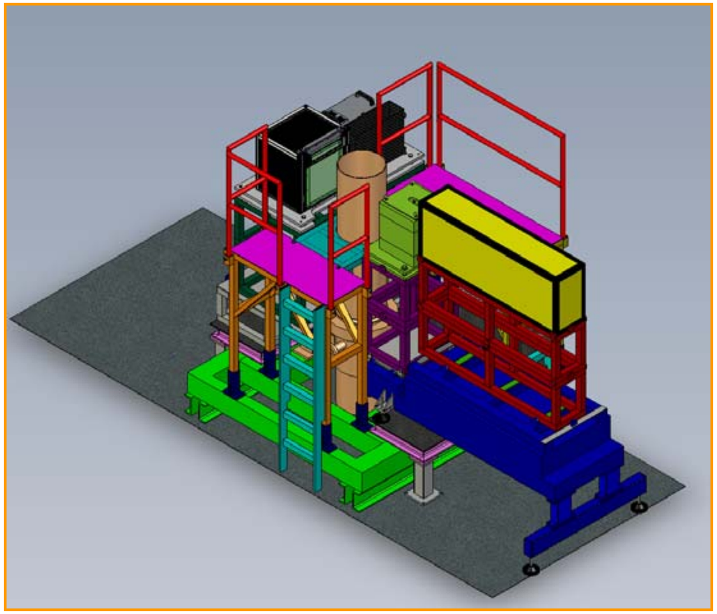




X-Ray beam

1. Provide quantitative data from KROTOS experiments with prototypic materials on the melt fragmentation for the development and validation of the modeling for FCI codes including:
    - Corium fragments velocity, size distribution and volume
    - Void velocity and volume
2. Analysis of the fuel fragmentation mechanism within water



Serena (phase 2): experimental grid

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		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

Improved Molten Core Cooling Strategy in a Severe Accident Management Guideline

J. H. Song[1], C. W. Huh[2], N. D. Suh[3]

[1]Korea Atomic Energy Research Institute
[2]Korea Institute of Nuclear Safety

CONTENTS

Introduction

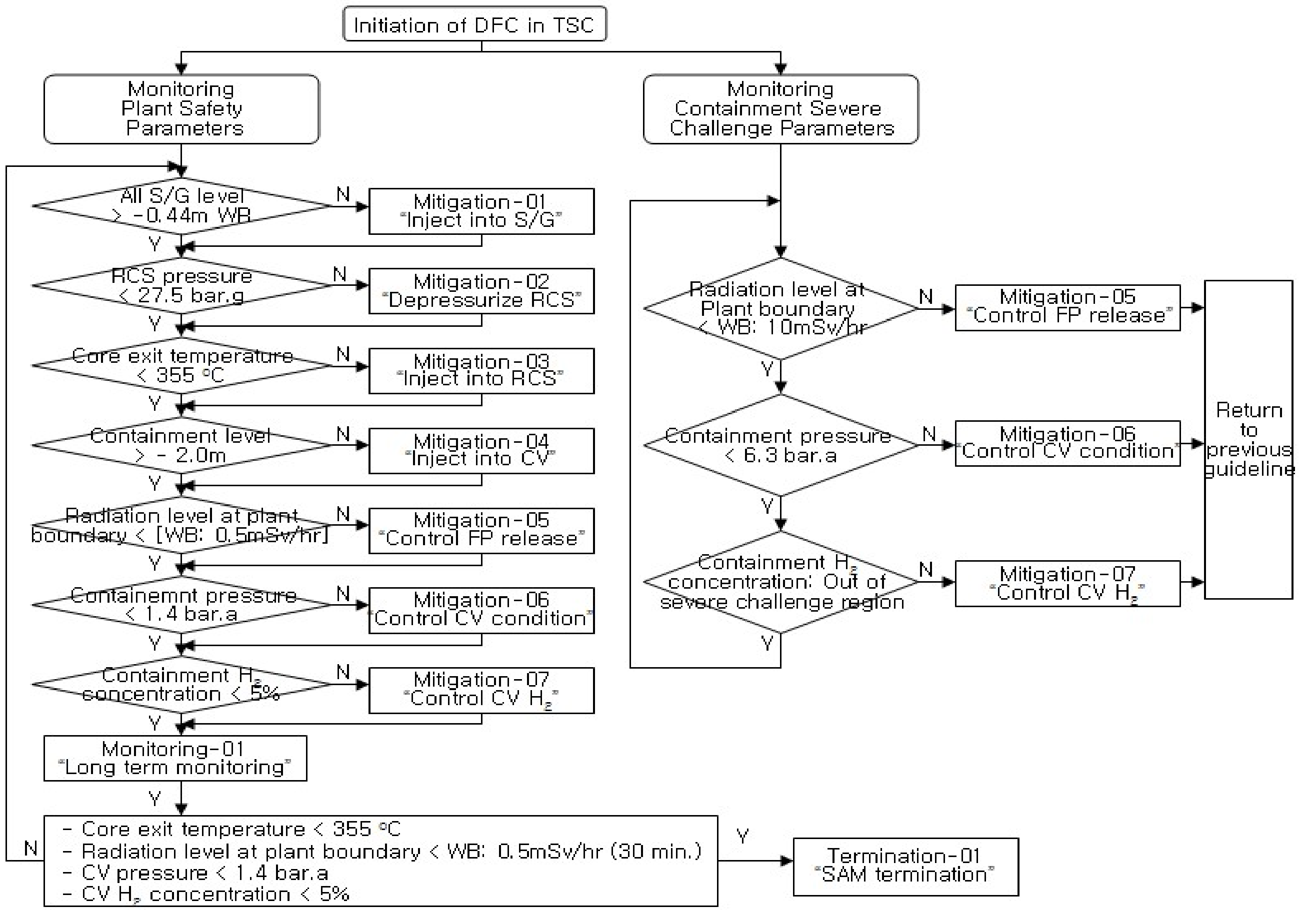
- Topic of Paper is Molten Core Cooling Strategy

- Focus is on Ex-Vessel Debris Coolability
  - For new reactor, various core catchers are proposed
    – EPR, VVER, EBSWR..
  - Operating plants rely on SAMG to handle the issue
  - Effectiveness of current SAMG needs to be evaluated
  - Plant specific analysis using MELCOR code was performed for Kori unit 1 and Ulchin 1&2
  - Recent research results from OECD/MCCI applied

- Improvement suggested for SAMG

Introduction

Strategy Flow Chart



Introduction

KSNP Severe Accident Management Guideline

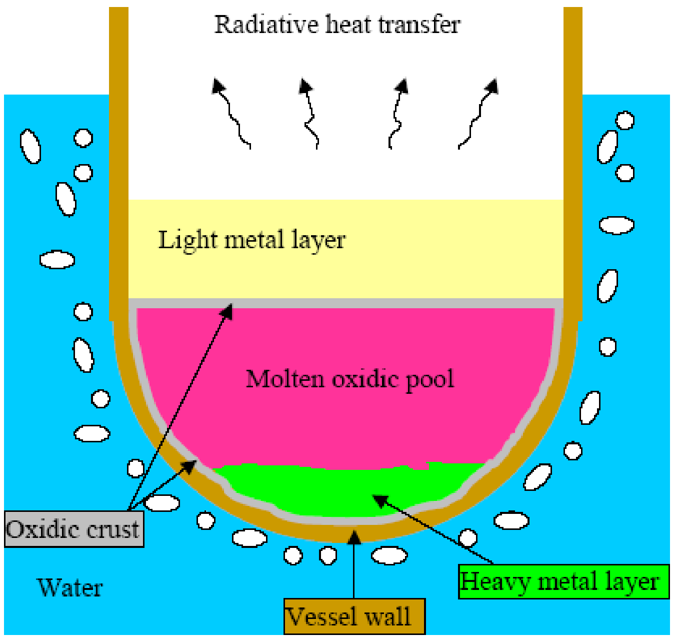
- Prevention of Reactor Vessel Failure
  M-01(Mitigation-1) : Inject into the S/G
  M-02(Mitigation-2) : Depressurize the RCS
  M-03(Mitigation-3) : Inject into the RCS
  M-04(Mitigation-4) : Inject into Containment

- Mitigation of Fission Product Release
  M-05(Mitigation-5) : Mitigate Fission Product Release

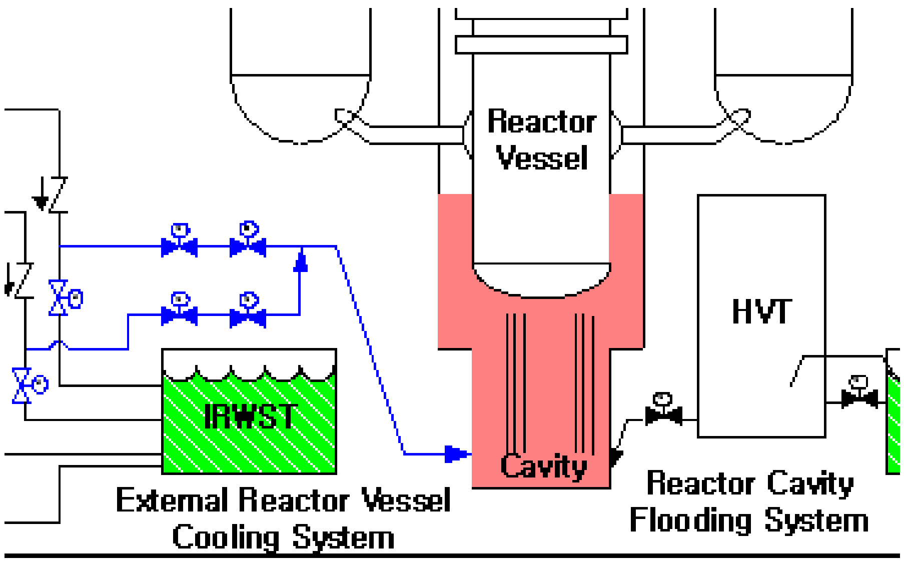
- Prevention of Containment Failure
  M-06(Mitigation-6) : Control Containment Condition
  M-07(Mitigation-7) : Control Containment Hydrogen

Introduction

Objective of M-04 ; In Vessel Retention by Ex-Vessel Cooling



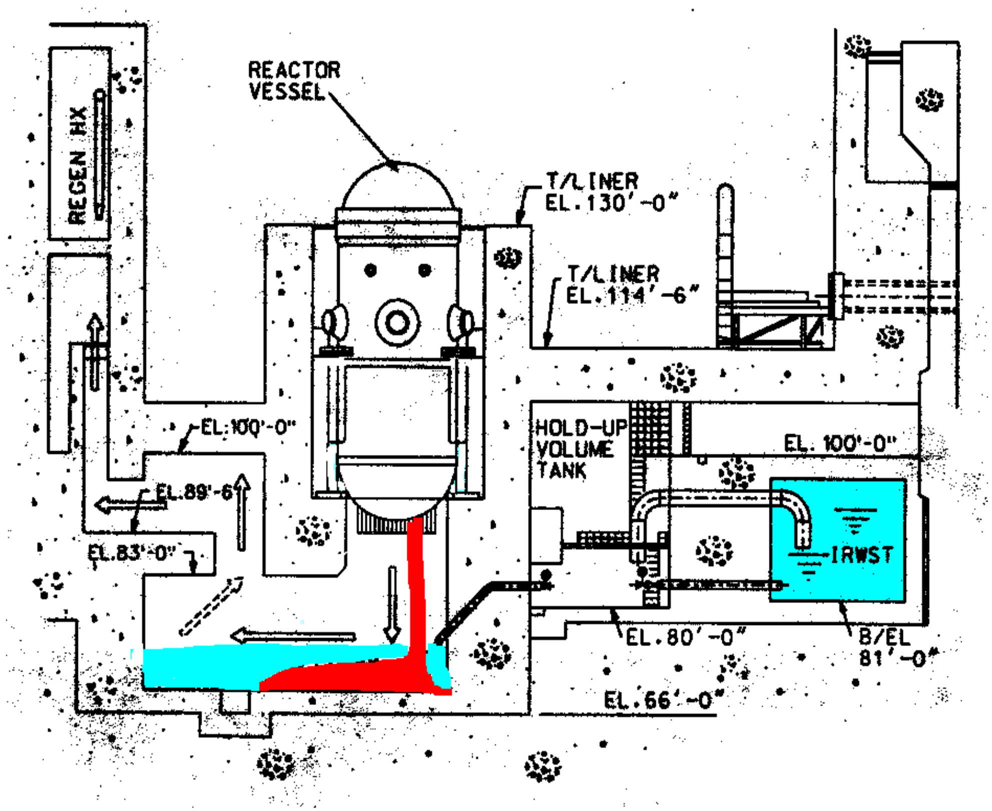
New reactor:
Passive Flooding
in case of AP1000

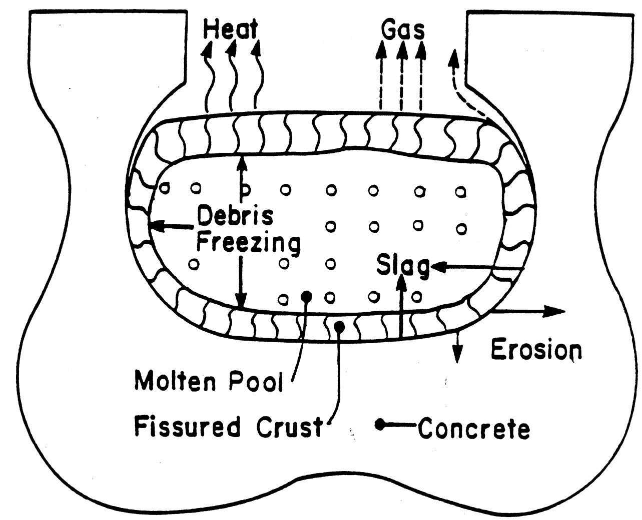


Existing Reactor SAMG:
Use available pumps and water
resource to flood the reactor cavity

Introduction

Objective of M-04 ; MCCI and Debris Coolability





Introduction

Kori -1 Plant, Ulchin 1&2

❑ Kori-1

- W 2-loop PWR
- first operation in 1978
- power : 587 MWe
- 2 RCPs, 2 SGs , 1 PZR

❑ Ulchin 1 &2

- Framatom 3-loop PWR
- first operation in 1988, 1989
- power : 900 MWe
- 3 RCPs, 3 SGs , 1 PZR

Inject Into Containment Strategy

- First objective of M-04 strategy

  - 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

  - a simple calculation for Kori-1 plant shows whether this strategy is possible or no

Inject Into Containment Strategy

available water inventory

RCS (including PZR)	6,109 ft^3
SIT(2)	2,544 ft^3
RWT	34,756 ft^3
Boron Tank (2)	534 ft^3

	43,943 ft^3

cavity free volume

height (ft)	area(ft^2)	free vol.(ft^3)
below 6(sump)		4,942
7.83	4,960	9,093
18.0	4,795	57,843

thus, water could be filled up to 14.07 ft

Inject Into Containment Strategy

- RV bottom is at 12.05 ft

  - theoretically, the water can immerse the very low part of RV
  - practically, we should consider the water remaining in the RCS, in containment as steam. Thus it is more reasonable to say that the water inventory is not sufficient enough to immerse the RV
  - also RV insulator design is crucial for establishing steam venting. Kori-1 insulator is not designed for that purpose

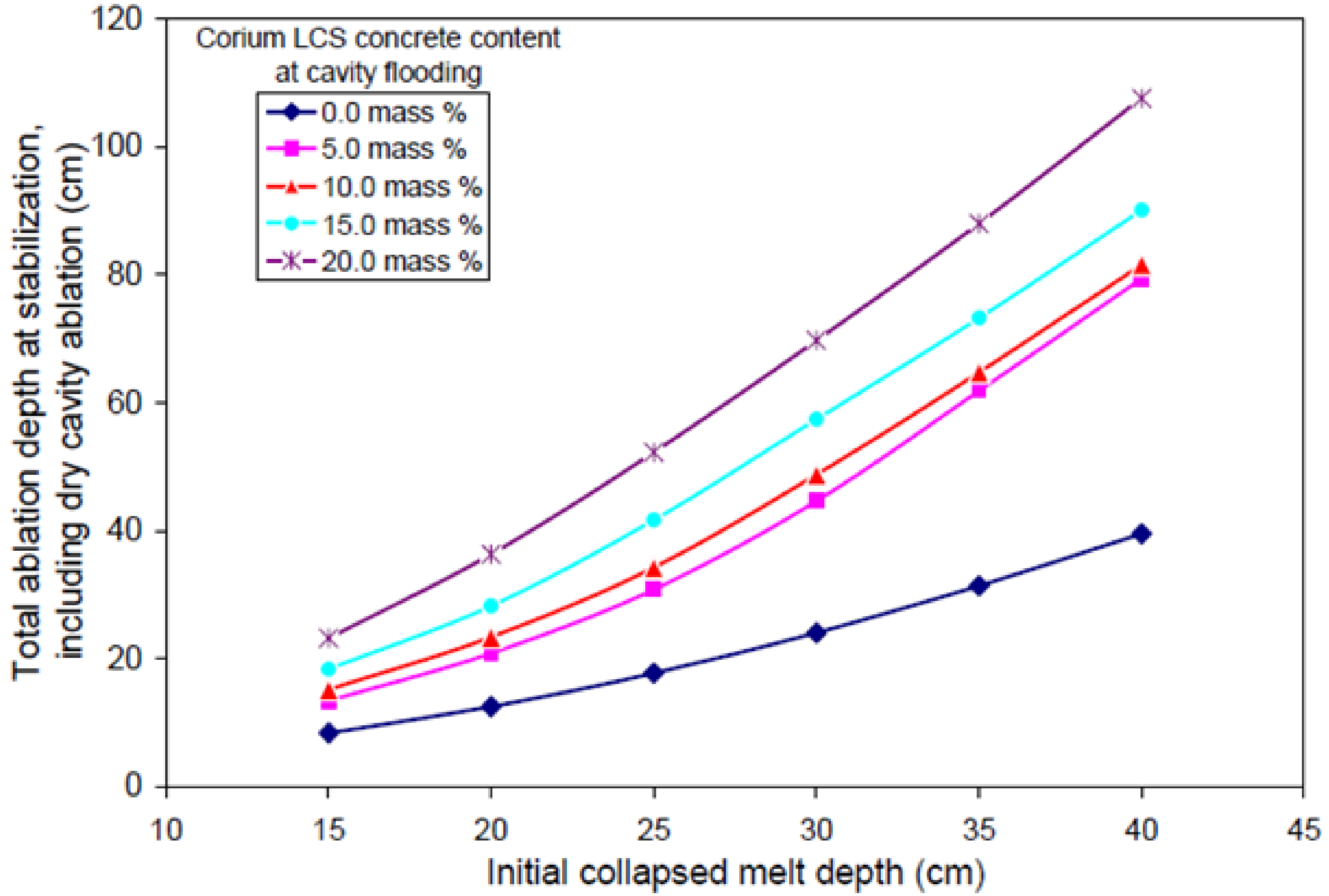
- Thus, ex-vessel cooling is not achievable for Kori-1

Inject Into Containment Strategy

- Second objective of M-04 strategy
  - cool the debris by injecting water into the cavity.
  - ex-vessel debris coolability by top-flooding was an un-resolved issue
  - recent results of OECD/MCCI phase 2 program shed some light on this issue. Incorporating the OECD/MCCI results, an integral analysis for a typical PWR for the MCCI process was performed and produced a figure which could be applied to reactor calculation

Inject Into Containment Strategy

Total ablation depth at stabilization as a function of initial collapsed melt depth (OECD/MCCI-2005-TR06, OECD MCCI Project Final Report, 2006)



Inject Into Containment Strategy

✓MELCOR computer code was used to analyze the typical severe accident scenario and to evaluate the effectiveness of operator action. SBO without any operator action is considered to accelerate accident progression

✓ Sequence of Top Events for Kori-1, RV failure at T=23,650 s

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

Inject Into Containment Strategy

MCCI condition at 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 code)	*0.47m*	*1.17m*
Remaining Base mat Depth	*1.953m*	*1.333m*

✓ At 10 hrs after SBO, the ablation depth to stabilization is ~1.1m from the previous figure

✓ The remaining depth is 1.953m at this time

✓ Thus we have 0.853 m of margin before melt through

✓ Even with the uncertainties considered, we have sufficient margin to say that the corium will be cooled if we top-flood at 10 hrs after SBO

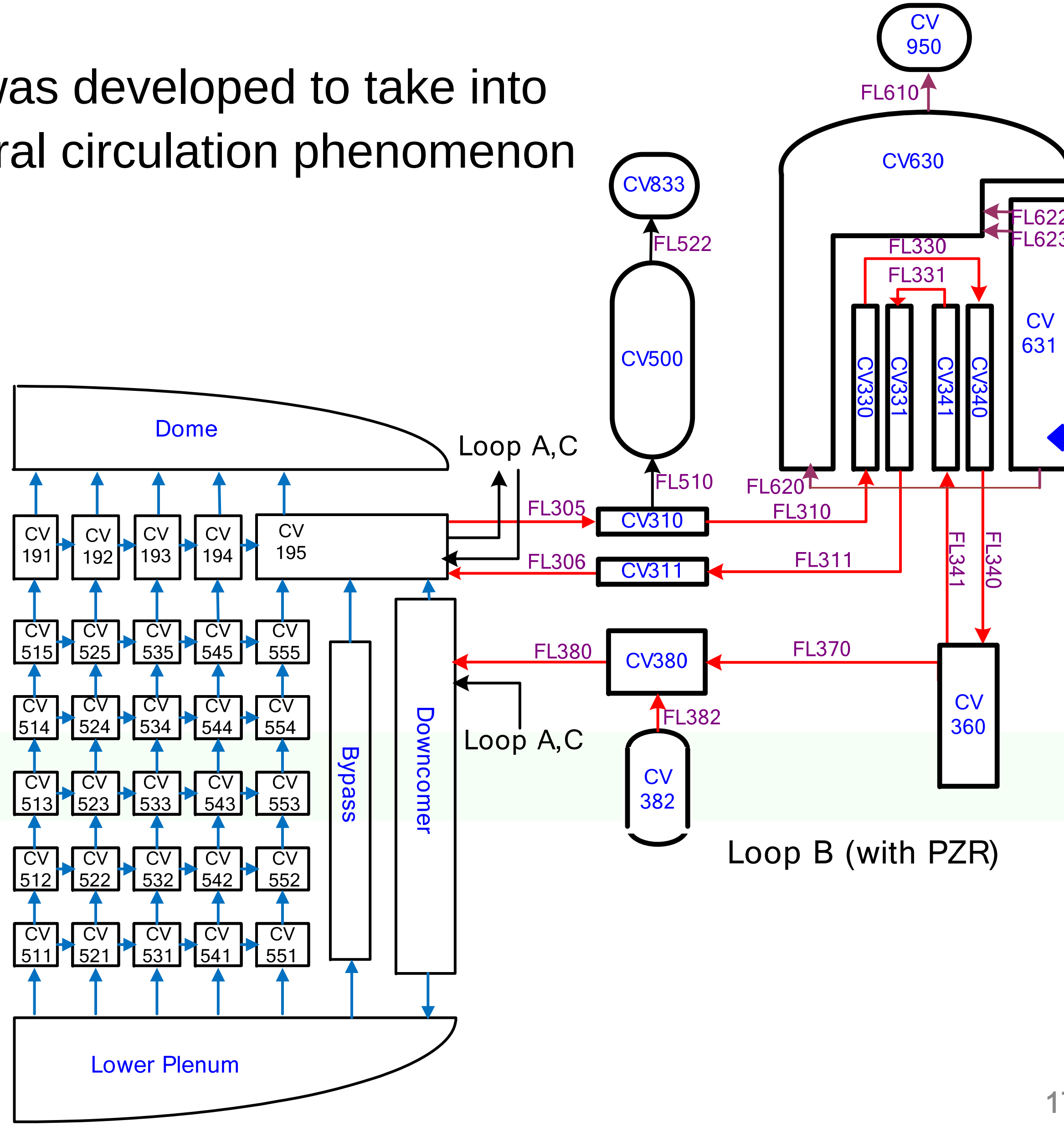
RCS Depressurization Strategy

- RCS Depressurization Strategy affects the debris coolability
- RCS Depressurization Strategy (M-02) is
  - Top priority strategy for high pressure accident
  - To establish core cooling with safety injection and to alleviate HPME/DCH
  - Depressurize RCS below a setpoint of 2.75MPa using all POSRVs

- Sensitivity Analysis is to assess
  - Feasibility and efficiency in mitigating severe accident progression
  - **SBO with only SIT using MELCOR 1.8.5 code for Uljin Unit 1**
  - ➢ Depressurization Timing
    - Based on CET temperature considering core damage condition
  - ➢ Depressurization Capacity
    - PZR POSRVs : 3 POSRVs (49.5 kg/s per valve at 17MPa)

RCS Depressurization Strategy

- MELCOR input was developed to take into account the natural circulation phenomenon



RCS Depressurization Strategy

- Evaluation Results
  - All cases satisfy the set point (2.75MPa) at the time of RPV failure
    - No need to open 3 POSRVs as is recommended in current SAMG
    - Depressurization Timing could be delayed ➔ provides operator time margin
  - The lower the depressurization rate, the later the RPV fails

➔ The time delay of RPV failure is important in managing the severe accident **and it also affects the debris coolability**

RCS Depressurization Strategy

Top Events of SBO Accident

	Low Depressurization Capacity (1 POSRV)			High Depressurization Capacity (3 POSRVs)		
Time of Events (sec)	**Case 1 (ref.)**	**Case 2**	**Case 3**	**Case 1 (ref.)**	**Case 2**	**Case 3**
Depressurization Initiation (CET)	**9845 (973K)**	**10410 (1100K)**	**12520 (1500K)**	**9845 (973K)**	**10410 (1100K)**	**12520 (1500K)**
SIT Injection	**11262**	**11791**	**13646**	**10263**	**10846**	**12877**
SIT Empty	**24827**	**31169**	**37146**	**11210**	**11816**	**14230**
Initial Vessel Breach	**32451**	**37954**	**40822**	**22336**	**22870**	**25240**
Pressure at RPV Failure (<2.75MPa)	**0.75**	**1.27**	**0.25**	**0.39**	**0.35**	**0.38**

RCS Depressurization Strategy

The effect of Optimized RCS depressurization strategy on the ex-vessel debris coolbility.

- probability of power recovery at 13hrs is 98%

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

RCS Depressurization Strategy

- The probability of power recovery at 13 hrs is 98%

- The optimized depressurization delays the vessel breach time and guarantees the debris coolability with a probability of 98%
  - more analyses for other accident scenarios are in need

- On the other hands, results show that the current depressurization does not guarantee coolability of ex-vessel debris by top-flooding in case power does not recover early

- It justifies our efforts of developing an optimum depressurization strategy in conjunction with M-04.

Summary

- Conclusion and Improvement Suggested

- If there is little chance of delaying the failure of a reactor vessel by a pre-flooding, there is no reason to pre-flood the reactor cavity
- According to the OECD/MCCI result, top-flooding should be done early
  - plant specific timing, thus plant specific SAMG needs to be analyzed
- Different depressurization for **M-02** can affect the timing of vessel breach and coolability of the molten corium
  - a plant specific optimum M-02 in conjunction with M-04 can be developed
- Appropriate instrumentation to detect either the breach of the reactor vessel or discharge of corium into the reactor cavity is needed
  - Thermocouples in the RPV insulation in case of EPR.

IRSN
INSTITUT
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ET DE SÛRETÉ NUCLÉAIRE

AEN NEA

SARnet

OECD/SARNET WORKSHOP ON IN-VESSEL COOLABILITY

Main Outcomes

B. Clément [1,2,3] (IRSN), J. Birchley [2,3] (PSI), H. Löffler [2] (GRS), W. Tromm [2,3] (KIT), A. Amri [3] (OECD/NEA)

[1] General chair
[2] Session chair
[3] Member of Organising Committee

INTRODUCTION

- In-Vessel Coolability issue identified as most important by OECD/NEA/CSNI/WGAMA and EC-SARNET, both because of its safety significance and of lacks in knowledge
  - WGAMA Work-plan for Severe Accidents NEA/SEN/SIN/AMA/2008(3)
  - SARP Final Report (SARNET-SARP-D96)
- In 2008, recommendation by WGAMA, endorsed by CSNI, to organise a workshop on the issue in fall 2009
- Joint OECD/SARNET workshop held at NEA headquarters in October 2009 (12^{th} to 14^{th})
- Main preliminary outcomes presented today, final conclusions to be included in Workshop Proceedings to be issued as a CSNI report in 2010

TECHNICAL BACKGROUND (1/3)

Severe Accident Management Guidelines give priority to containment integrity as compared with core integrity after some progression in the course of a severe accident, e.g. when indications from Core Exit Thermocouples exceed a given threshold

However, trying to cool the degrading fuel and/or corium within the Reactor Pressure Vessel is a way to slowdown/stop the progression of the accident and to delay/avoid Reactor Pressure Vessel rupture that may endanger the containment integrity by dynamic loads (Direct Containment Heating, ex-vessel Steam Explosion) and/or static loads (Corium/Concrete Interaction)

Once a water source has been recovered, different strategies can be used: send water in the core and/or cool the RPV externally. They might in some cases conflict with other uses of available water, e.g. activating spray systems in the containment

TECHNICAL BACKGROUND (2/3)

However, sending water in a degrading core (the reflooding issue) is not straightforward as:

– The efficiency of reflooding for significantly delaying or stopping core degradation is not demonstrated for all situations;

– It may result in high hydrogen production rates that may threaten the containment integrity by dynamic loading (H2 combustion); it may also result in a pressure peak that may endanger the containment integrity by DCH if the reactor pressure vessel has been previously weakened by corium slumps.

Given these adverse considerations, some Severe Accident Management Guidelines (SAMGs) consider cautions in how and when to send water in the core. In addition, reflooding models used for evaluation of SAM suffer from a lack of validation that makes it difficult to assess the suitability of different accident management strategies

TECHNICAL BACKGROUND (3/3)

Trying to cool the RPV externally to assure In-Vessel Retention is also not straightforward as:

– This accident management measure was not taken into account in the original design of existing reactors

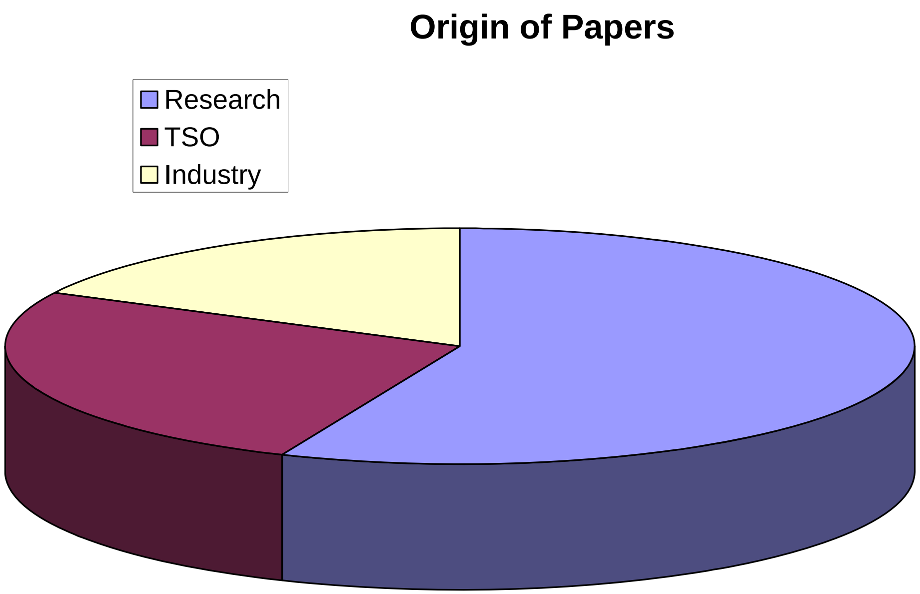
– The probability of success strongly depends on the reactor detailed specific features (reactor pit geometry, type of heat insulation, connections to the reactor dome…). It also decreases with the reactor power

Also, if external cooling turns out to be inefficient, the occurrence of an ex-vessel steam explosion cannot be ruled out and this is still considered as a non resolved issue

As for in-vessel reflooding, the models used for evaluation of SAM suffer from a lack of validation that makes it difficult to assess the suitability of different accident management strategies

GENERAL OVERVIEW (1/4)

- 66 participants from Belgium, Bulgaria, Canada, Czech Republic, Finland, France, Germany, Hungary, Italy, Korea, Slovak Republic, Spain, Sweden, Switzerland, United Kingdom, United States + OECD/NEA
- 22 papers presented and discussed in 4 technical sessions
- Final session for summary by technical sessions chairs and discussion



GENERAL OVERVIEW (1/4)

- 22 papers in 4 technical sessions
  - General studies

    General safety studies on in-vessel coolability including PSA level 2
  - Experimental work

    Review of recent, ongoing and planned experimental programmes
  - Phenomenological and modelling work

    Review of models used or under development for severe accident calculation tools
  - Specific reactor studies

    Analyses of specific cases for in-vessel coolability
- Conclusions by sessions' chairs and discussion

GENERAL OVERVIEW (2/4)

❑ SESSION 1 (general studies)

- One paper by KIT (FZK) synthesising existing knowledge on degraded core reflood and identifying global influential parameters

- Two papers by IRSN and GRS on results and main lessons learnt from PSA level 2 studies for French and German reactors

- One paper by CEA presenting the development of a new tool to be used by EDF for PSA level 2 studies

❑ SESSION 2 (experimental work)

- One paper by KIT on QUENCH (reflooding of bundles)

- Two papers by IKE and IRSN on debris bed coolability (DEBRIS and PEARL)

- One paper by KIT on molten pools (LIVE)

- One paper by CEA on RPV external cooling (CNU)

GENERAL OVERVIEW (3/4)

- SESSION 3 (phenomenological and modelling work)

- One overview by CEA of melt dynamics treating strong coupling between material property effects and thermal-hydraulics

- Three papers by IRSN and IKE on debris characterisation and modelling of reflooding for a severely damaged core including debris cooling

- Two papers by RUB and IRSN on the simulation of two QUENCH experiments conducted under conditions adverse to quenching

- One paper by GRS on simulation of TMI-2 accident by ATHLET-CD

- One paper on the results of the OECD benchmark exercise on an alternative TMI-2 scenario (authors = participants to benchmark)

GENERAL OVERVIEW (4/4)

- SESSION 4 (specific reactor studies)

- Two papers by IVS and Paks NPP on RPV external cooling for VVER-440/213 showing good prospects

- One paper by AMEC and British Energy about the optimal use of water after core degradation has started (Sizewell B)

- Two papers by RIT and AREVA NP about RPV external cooling for BWRs

PRELIMINARY CONCLUSIONS (1/6)

➢ The present studies reinforce the view that sending water in a degrading core (the reflooding issue) is not straightforward 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 was given as a guideline figure for minimum water flow rate

- In addition to the phenomenological issues related to cooling a degraded core, the probability for recovery of water sources has to be addressed

PRELIMINARY CONCLUSIONS (2/6)

- Similarly, presented results reinforce the view that trying to cool the RPV externally to assure In-Vessel Retention is also not straightforward

- The maximum amount of molten corium that can be retained in the RPV lower head has been estimated by different methods at 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 there are good prospects for success

PRELIMINARY CONCLUSIONS (3/6)

➢ 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 a severe accident, and there is no guarantee of successful cooling; the measures are very plant specific

- In addition to the phenomenological issues related to cooling 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 melt-down accident by water injection is generally considered as high and depends on reactor specific features

- This need calls for a sustained R&D effort, both on experimental and analytical point of views

PRELIMINARY CONCLUSIONS (4/6)

- 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

- Still a difficulty with present models is to predict reliably if reflooding during early core degradation would or not trigger a cladding oxidation runaway – oxidation of melts? thermal-hydraulics?

- 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-D effects are expected to become important needs to be evaluated – larger scale experiments are probably not feasible

PRELIMINARY CONCLUSIONS (5/6)

- The questions of uncertainty and adequacy of the codes was discussed, revealing 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 the way to needs for new data

PRELIMINARY CONCLUSIONS (6/6)

➢ 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 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

SUGGESTIONS FOR FUTURE WORK

- It is expected that ongoing experimental programmes and analytical efforts will help making progress in the coming years – it would then be valuable to issue a State of the Art Report as foreseen in the WGAMA work plan

- This SOAR should include a status on the ability of simulation tools to predict reliably fuel/corium coolability, planned benchmarks being useful for that purpose – their precise definition should take this objective into account

- Organising follow-up workshops, as suggested by some participants, could be discussed at the next WGAMA meeting

- Benchmark exercises will continue to play a role in helping to understand and place estimates on code uncertainties

Simulation of Ex-Vessel Debris Bed Formation and Coolability in a LWR Severe Accident

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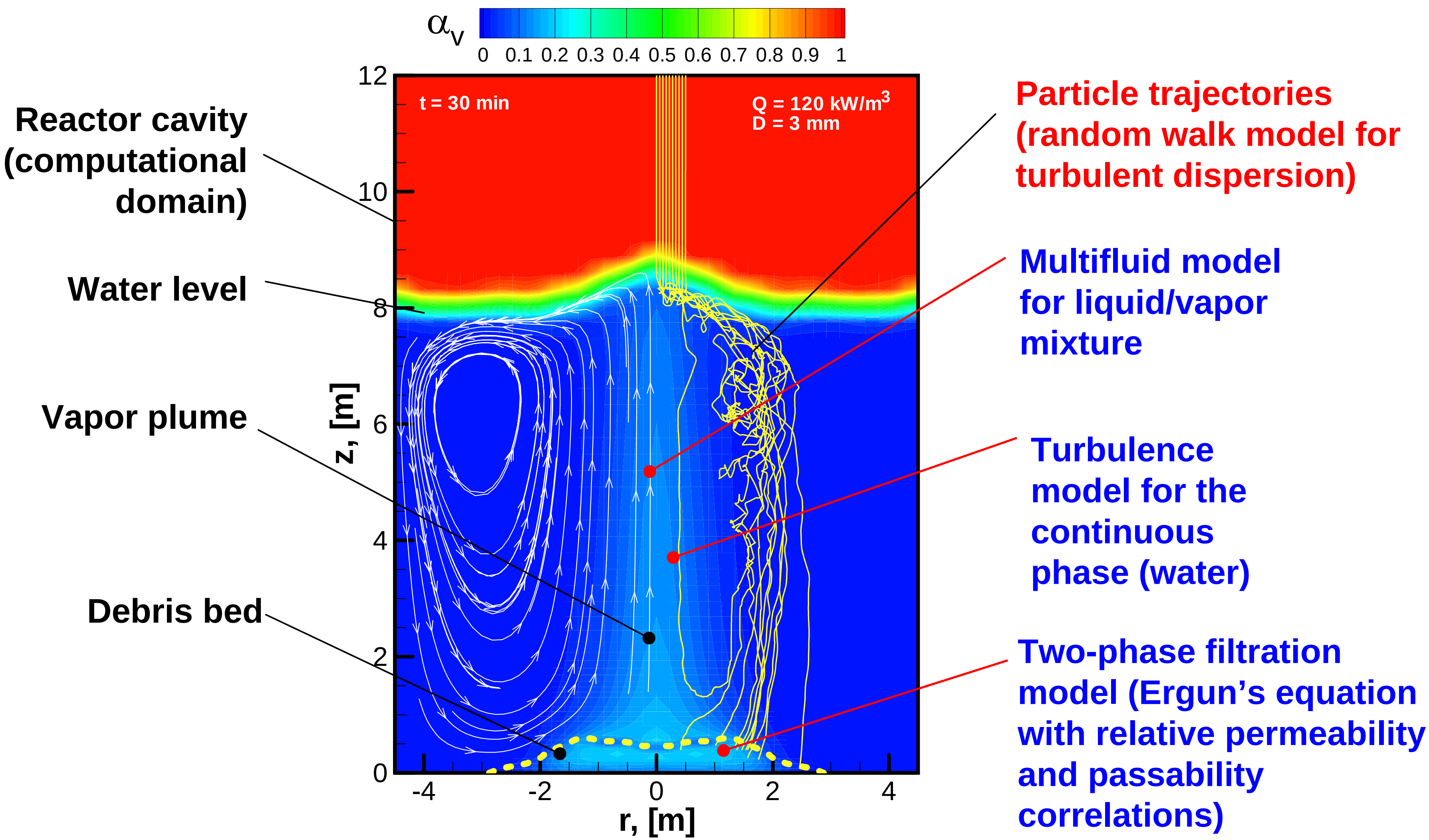
Introduction

- Severe accident management strategy for Swedish type BWRs adopts reactor cavity flooding for termination of ex-vessel accident progression
- Core melt materials ejected from the reactor vessel into a deep pool in the reactor cavity are fragmented, quenched and form a porous debris bed which should be coolable by natural circulation
- Criterion generally accepted for successful long-term cooling: no local dryout should occur
- DECOSIM (DEbris COolability SIMulator) code is developed for simulation of debris bed formation and coolability

Motivation and Objectives

- We consider two different scenarios of the debris bed formation:
  - i) gradual melt release (dripping mode)
  - ii) rapid melt release (jet, massive melt release)
- In the dripping melt release mode, the shape of debris bed can be affected by “self-organization” phenomena. Namely, convective flows driven by vapor release in the already existing debris bed may affect particle sedimentation and determine the ultimate shape of the bed
- In case of massive melt release coolabilty depends on
  - debris bed shape and heat release rate
  - porosity and particle size
  - encapsulated particle porosity
  - presence of a low-permeability “cake”

DECOSIM Physical Models



Governing Equations

Debris Bed

$$\frac{\partial(\varepsilon\rho_i\alpha_i)}{\partial t}+\nabla(\varepsilon\rho_i\alpha_i\mathbf{U}_i)=-\Gamma_i$$

$$\Gamma=\begin{cases}\dfrac{\dot{Q}}{\Delta H_{ev}} & \text{for}\quad \alpha_l>0\\ 0 & \text{for}\quad \alpha_l=0\end{cases}$$

$$\varepsilon\alpha_i\nabla P=\varepsilon\alpha_i\rho_i\mathbf{g}-\mathbf{F}_{is}$$

$$\mathbf{F}_{is}=\varepsilon\alpha_i\left(\frac{\mu_i}{KK_{ri}}\mathbf{j}_i+\frac{\rho_i}{\eta\eta_{ri}}|\mathbf{j}_i|\mathbf{j}_i\right)$$

$$K_{rl}=(1-\alpha)^3,\quad \eta_{rl}=(1-\alpha)^5$$

$$K_{rv}=\alpha^3,\quad \eta_{rv}=\alpha^5$$

Ambient Flow

$$\frac{\partial(\rho_i\alpha_i)}{\partial t}+\nabla(\rho_i\alpha_i\mathbf{U}_i)=0$$

$$\rho_i\alpha_i\left(\frac{\partial\mathbf{U}_i}{\partial t}+(\mathbf{U}_i\cdot\nabla)\mathbf{U}_i\right)=-\alpha_i\nabla P$$

$$+\alpha_i\nabla\tau_i+\alpha_i\rho_i\mathbf{g}-\mathbf{F}_{ij}$$

$$\mathbf{F}_{ij}=\frac{3}{4}\alpha_i\frac{\rho_j}{D_i}C_D^{ij}|\mathbf{U}_j-\mathbf{U}_i|(\mathbf{U}_j-\mathbf{U}_i)$$

$$\frac{\partial\rho_l\alpha_l k_t}{\partial t}+\nabla(\rho_l\alpha_l k_t\mathbf{U}_l)=$$

$$=\nabla\left(\alpha_l\frac{\mu}{\sigma_k}\nabla k_t\right)+\alpha_l(G-\varepsilon_t)$$

$$\frac{\partial\rho_l\alpha_l\varepsilon_t}{\partial t}+\nabla(\rho_l\alpha_l\varepsilon_t\mathbf{U}_l)=$$

$$=\nabla\left(\alpha_l\frac{\mu}{\sigma_\varepsilon}\nabla\varepsilon_t\right)+\alpha_l\frac{\varepsilon_t}{k_t}(C_1G-C_2\varepsilon_t)$$

Model for the Debris Sedimentation

- **Lagrangian approach: equations of motion are solved for a number of discrete particles with empirical correlations for the drag force**
- **One-way coupling only: particles are affected by the flow, but not vice versa (no account for "collective effects")**
- **Random walk model accounts for turbulent dispersion of particles**
- **"Gap-Tooth" numerical algorithm developed for efficient simulation of long transients**

$$\frac{d\mathbf{r}^k}{dt} = \mathbf{U}_m^k \qquad \rho_m \frac{d\mathbf{U}_m^k}{dt} = -\mathbf{F}_{lm} - \mathbf{F}_{vm} - (\rho_m - \rho_a)\mathbf{g}$$

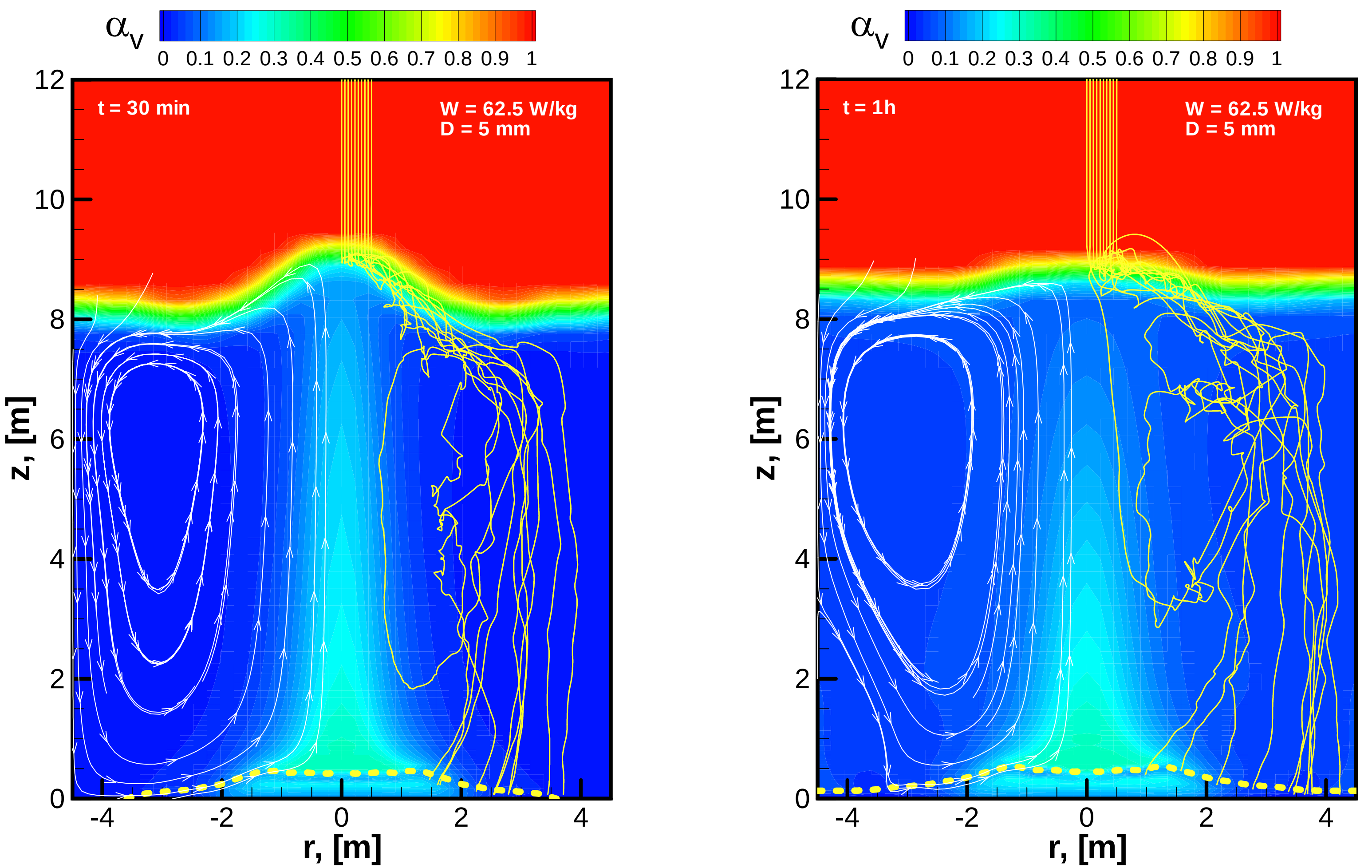
$$\mathbf{F}_{im} = \alpha_i \frac{3}{4} \frac{\rho_i \left|\mathbf{U}_i - \mathbf{U}_m\right|}{D_m} C_D(\mathrm{Re}_m)\left(\mathbf{U}_i - \mathbf{U}_m\right) \qquad \mathbf{U}_l = \tilde{\mathbf{U}}_l + \mathbf{u}'(k, \varepsilon)$$

$$C_D(\mathrm{Re}_m) = \frac{24}{\mathrm{Re}_m}(1 + 0.15\,\mathrm{Re}_m^{0.687}) \qquad \mathrm{Re}_m = \rho_i D_m \left|\mathbf{U}_i - \mathbf{U}_m\right| / \mu_i$$

Simulation of Debris Bed Formation

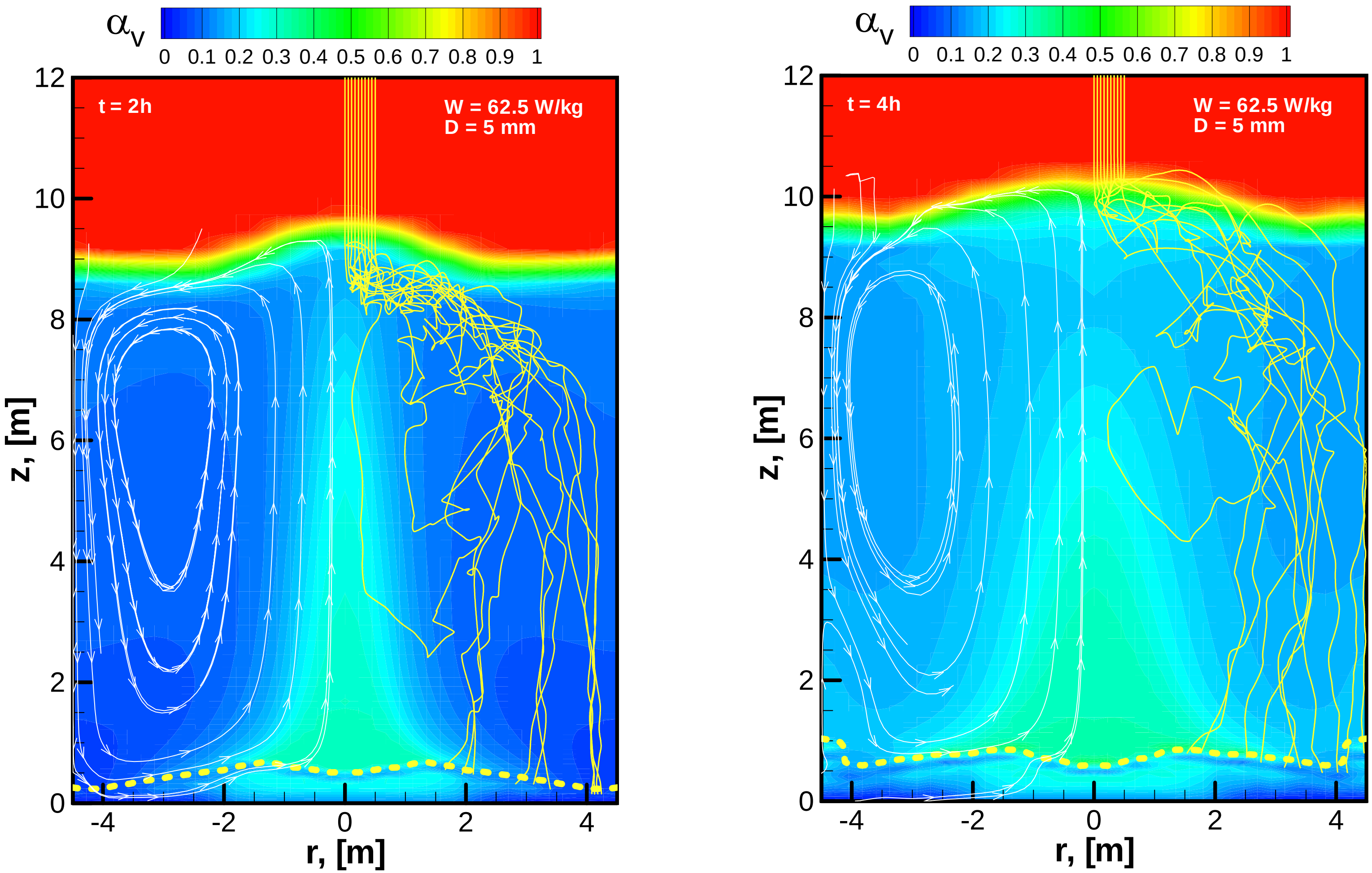
- Water pool of 9 m diameter and 12 m height is considered
- Water is filled to the level of 8 m
- Total mass of melt supplied: M_0 = 200 t
- Total melt supply time: t_M= 4 hours
- Melt particle diameter: 3 to 10 mm (same particle diameter used for debris bed)
- Porosity of the debris bed: 0.4
- Specific heat release rate: 25 W/kg of corium (120 kW/m^3 of debris bed) and 62.5 W/kg (300 kW/m^3) to study strong and weak convection

Baseline Scenario: W=62.5 W/kg
Particles: D = 5 mm



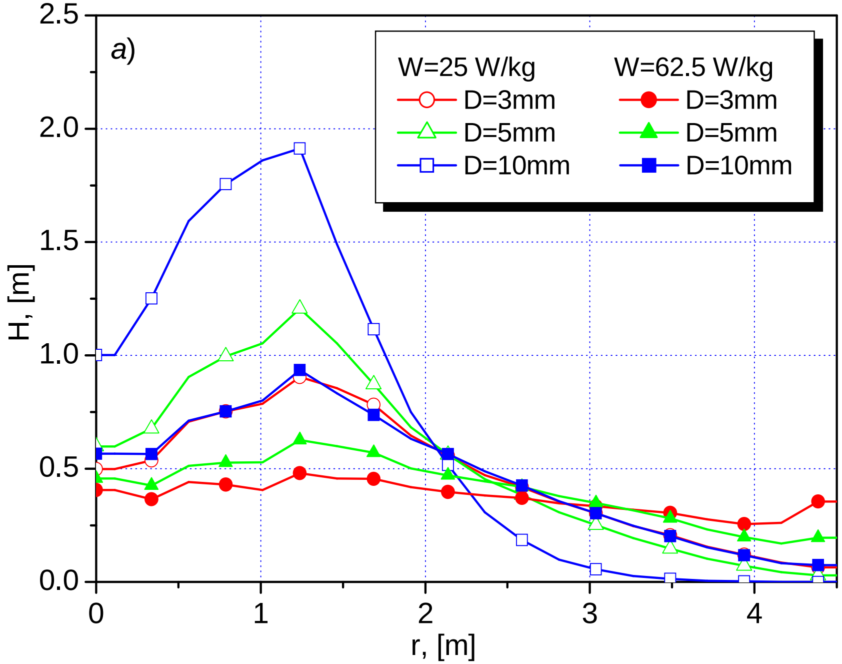
Flowfields and debris bed shapes (shown by yellow dashed line) at 30 min and 1 hour

Baseline Scenario: W=62.5 W/kg
Particles: D = 5 mm

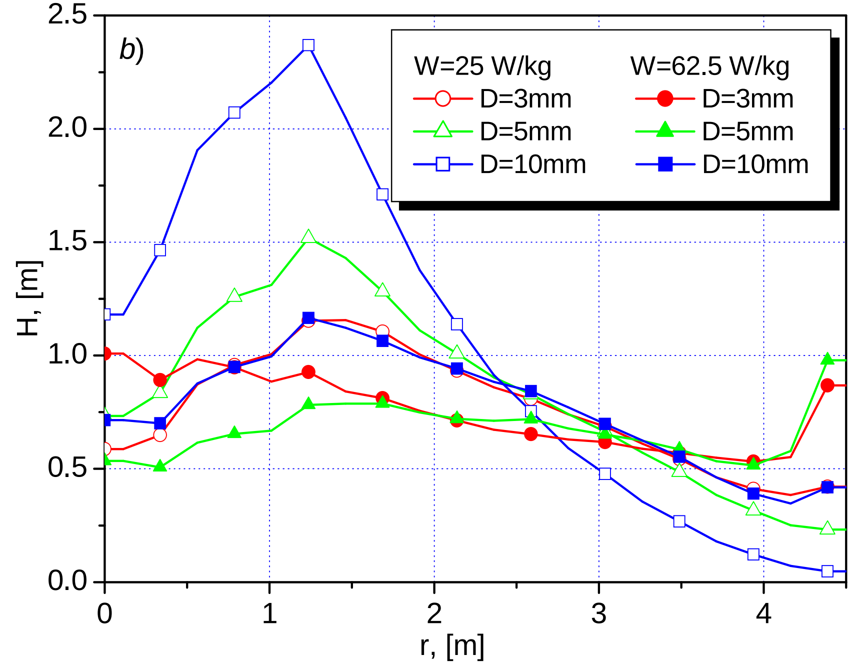


Flowfields and debris bed shapes (shown by yellow dashed line) at 2 and 4 hours

Debris Bed Shape



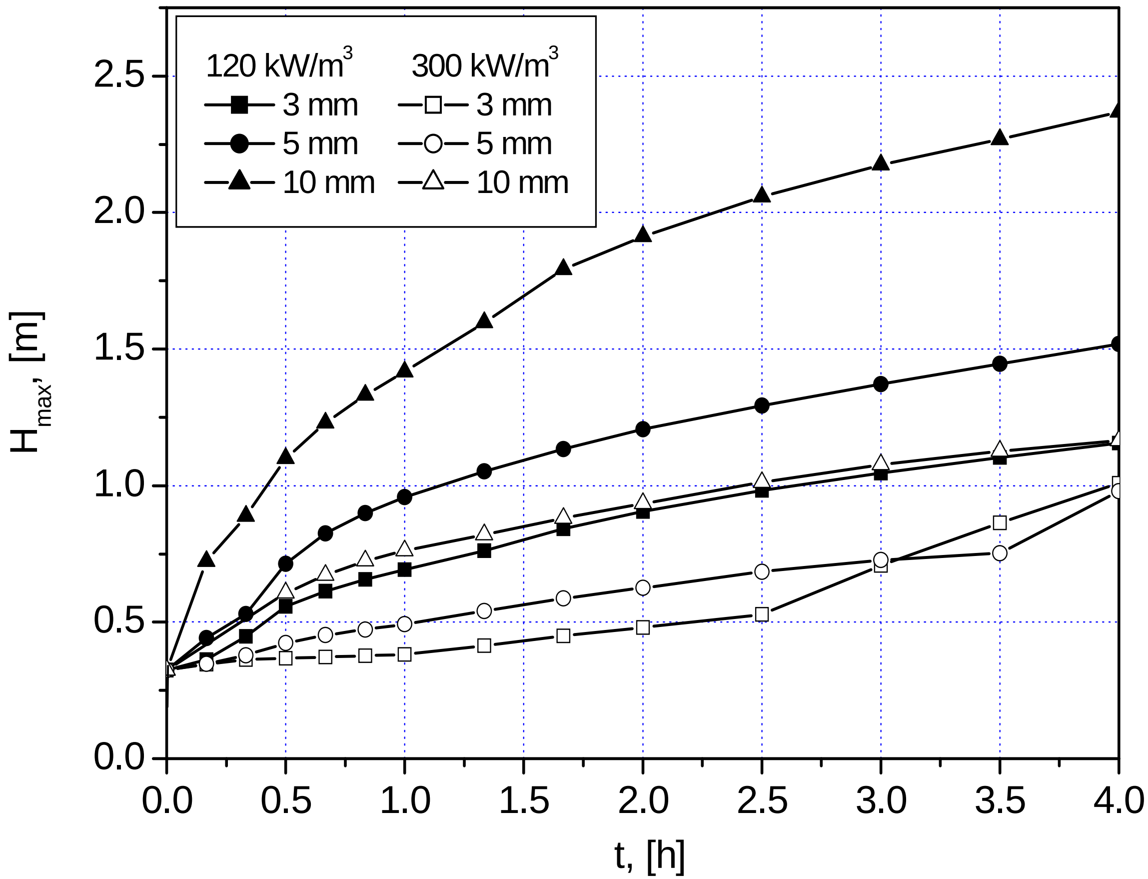
2 hours



4 hours

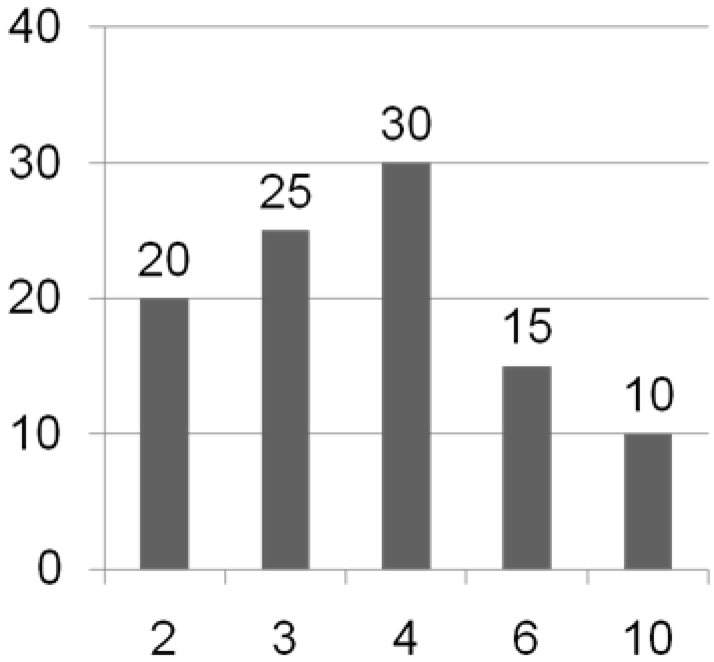
- ➢Natural convection flows promote flattening of debris bed, especially for fine particles
- ➢ For melt particles with size distribution, non-homogneous debris bed is expected

Maximum Height of Debris Bed

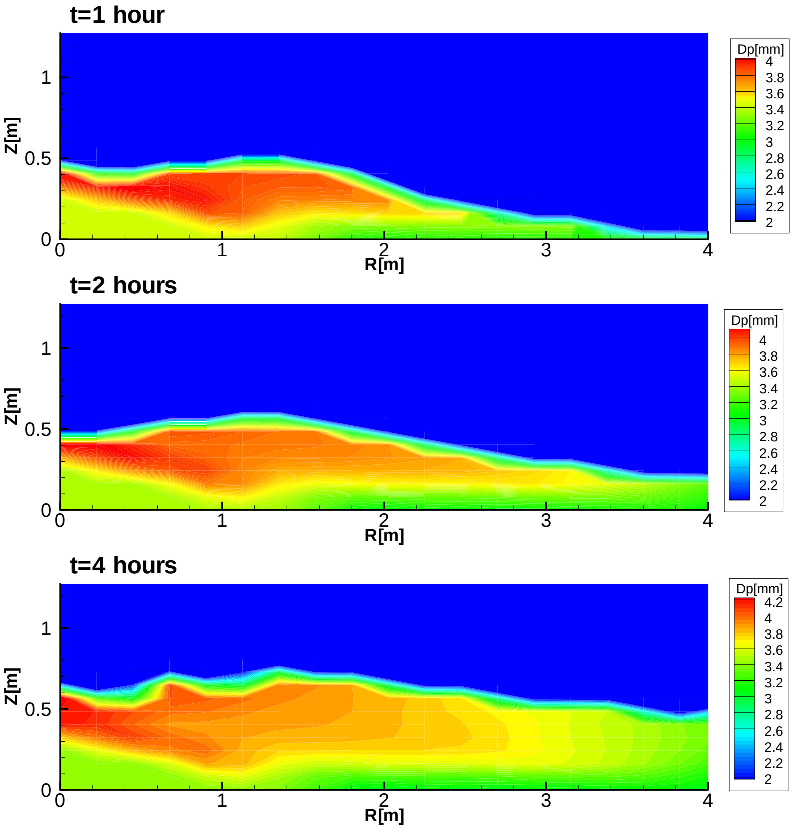


Stronger convective flows result in particle spreading over the pool bottom, especially for fine particles

Formation of Debris Bed from Particles with Size Distribution



Particle Size Distribution

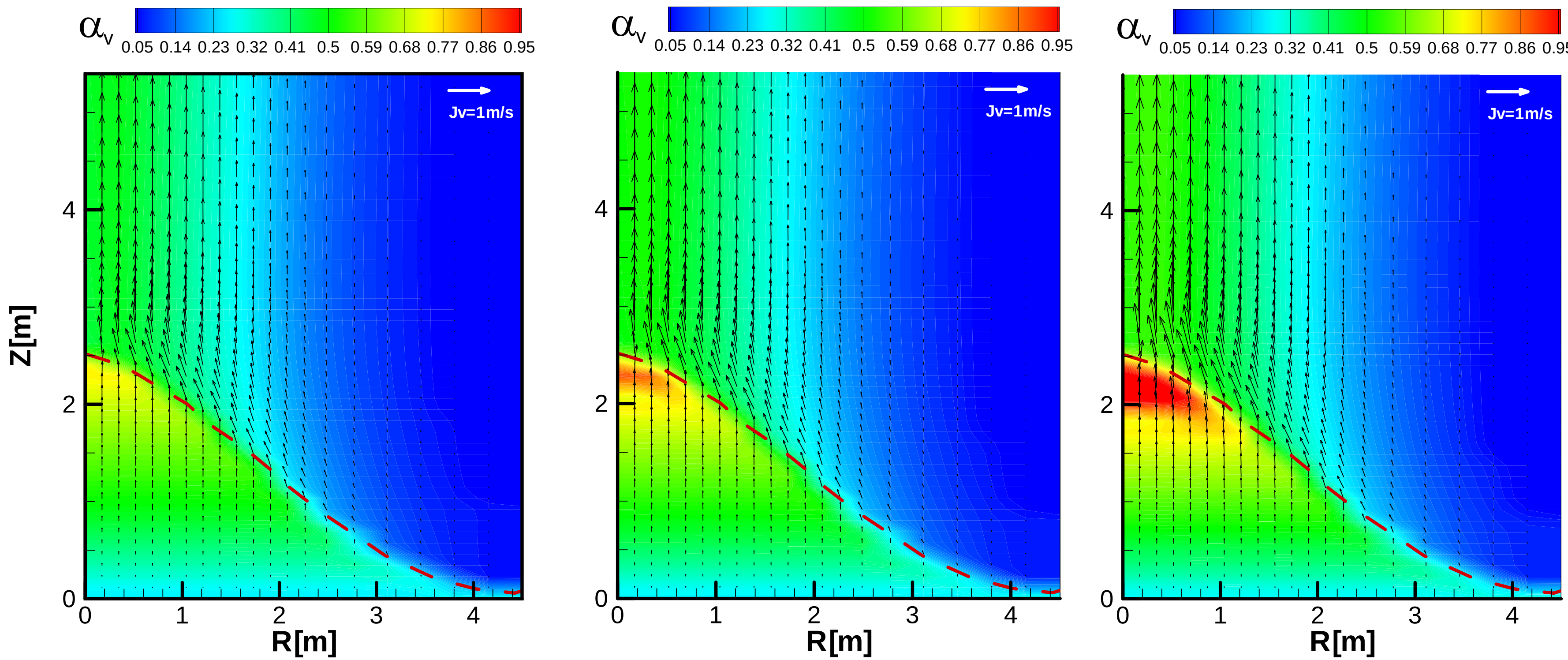


Mean Particle Size Distribution in Debris Bed (W=120 W/kg)

Parametric Studies of Debris Bed Coolability

- Water pool of 9 m diameter and 12 m height
- Water is filled to the level of 8 m
- Total mass of corium: M_0 = 200 t
- System pressure: 3 bar
- Gaussian-shaped debris bed, H=2.5 and 2 m
- Particle diameter: 2 and 3 mm
- Porosity of the debris bed: 0.4
- Specific heat release rate: up to W=350 W/kg
- "Cake": permeability is reduced to 1/2 and 1/5 of its debris bed value
- Encapsulated porosity: 25% (equivalent to 15% increase in the overall void fraction)

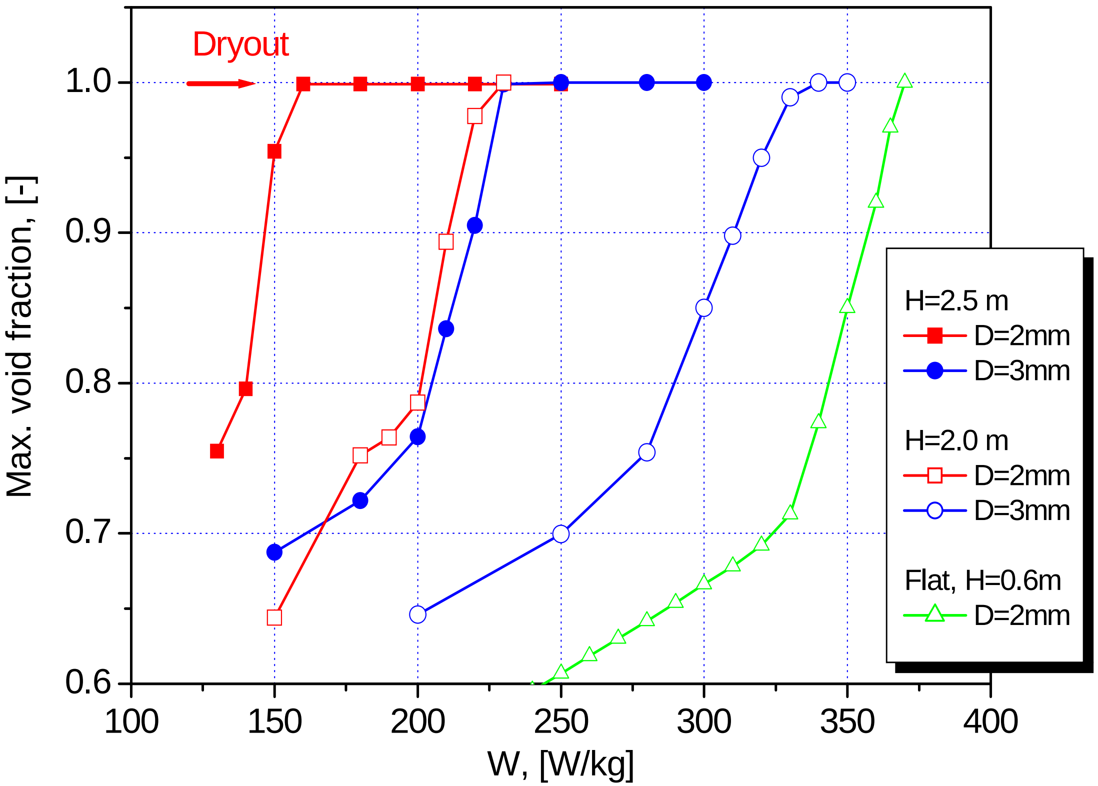
Dryout Development in a Gaussian-Shaped Debris Bed (D = 2 mm)



130 W/kg 150 W/kg 180 W/kg

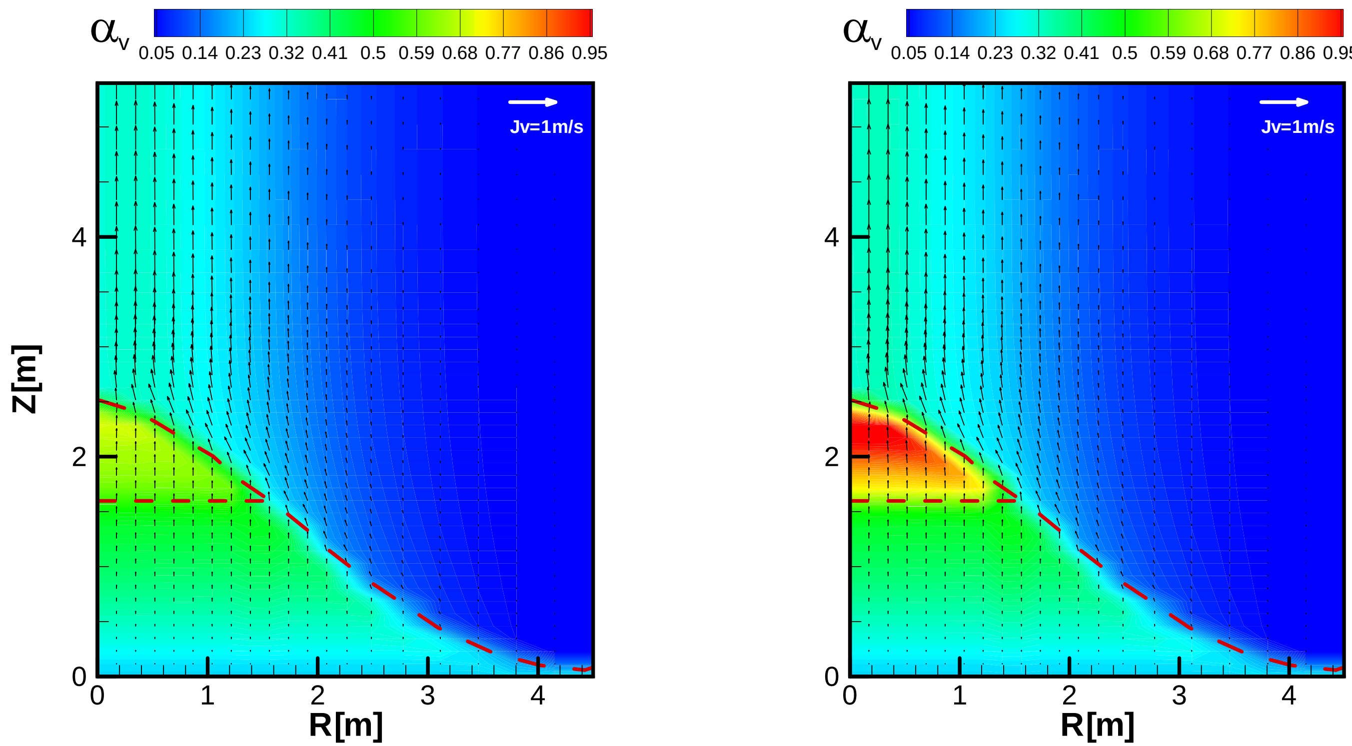
Void fraction distributions in the debris bed for different specific heat release rates

Summary of Coolability Results



Maximum void fraction vs Specific heat release rate for different particle diameters and debris bed heights, as well as for a 0.6 m high flat layer (with the same total mass of debris)

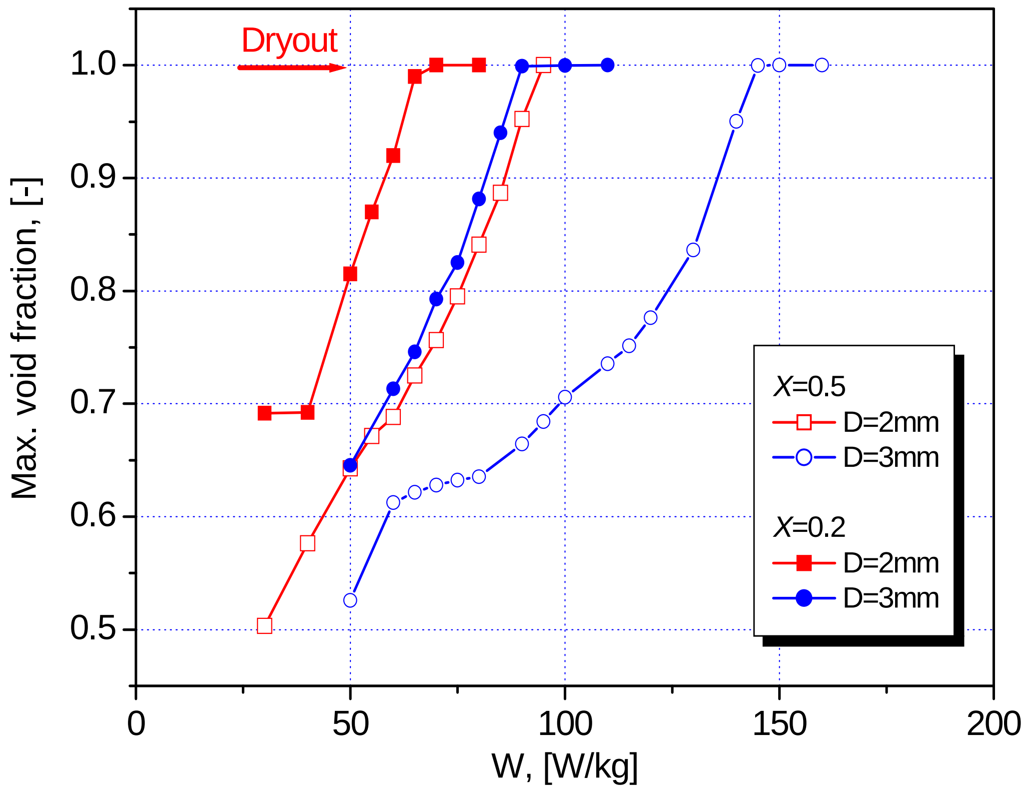
Dryout Development in a Debris Bed with a “Cake” (D = 2 mm, W=65W/kg)



Permeability in the “cake”: 1/2 (left) and 1/5 (right)

Void fraction distributions in the debris bed with a “cake” occupying top 5% (by volume)

Coolability of Debris Bed with “Cake”



Maximum void fraction vs Specific heat release rate for different “cake” permeability reduction factors

Conclusions

- For the scenario of gradual melt release (dripping mode), self-organization mechanism due to natural convective flows plays an important role in distributing the melt particles over the bottom of pool
- The resulting debris bed can be non-homogeneous
- Dryout in a heap-shaped debris bed occurs more readily than in a flat layer with the same mass of debris
- Debris bed height affects significantly its coolability
- Formation of a low-permeability “cake” on the top of debris bed has a pronounced negative effect
- Effects of encapsulated particle porosity require further studies (system pressure dependent)

Acknowledgements

- The work is performed in the framework of MSWI project, funded by the
  - APRI group (Swedish Nuclear Power Inspectorate SSM and power generating companies)
  - Swiss ENSI
  - EU SARNET Project
  - Nordic Nuclear Safety Program (NKS)

OECD/NEA Workshop on Implementation of Severe Accident Management Measures

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”, Moscow, RF

Böttstein, Switzerland, October 26-28, 2009

SAMG DEVELOPMENT FOR VVER-1000/V-320

- Development of generic SAMG for operating VVER-1000/V-320 plants (2003-2006) – revision 1 – under sponsorship of RF Utility “Concern Energoatom”
- Comments of Balakovo NPP and Kalinin NPP specialists
- Development of generic SAMG, revision 2 based on comments from the plants (2007)
- Development of SAMG for Unit 4 of Balakovo NPP (2008)

Participants: Institute for Nuclear Reactors of RRC “Kurchatov Institute”, Russian Minatom International Nuclear Safety Center (RMINSC)

DESCRIPTION OF BALAKOVO NPP, UNIT 4 SAMG

The package of SAMG documents:

- the set of SAMG guidelines and computational aids
- the set of documents “Rules of accident management”
- the document “Executive volume”

The documents “Rules of accident management” have been developed for each guideline or computational aid of the SAMG

BALAKOVO NPP, UNIT 4 SAMG COMPOSITION

The SAMG composition 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.

BALAKOVO NPP, UNIT 4 SAMG COMPOSITION (continued)

The DFC guidelines:

- 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”

BALAKOVO NPP, UNIT 4 SAMG COMPOSITION (continued)

The SCST guidelines:

- SCG-1, “Mitigate Fission Product Releases”
- SCG-2, “Depressurize Containment”
- SCG-3, “Control Hydrogen Flammability”
- SCG-4, “Control Containment Vacuum”

The guidelines for MCR:

- SACRG-1, “Severe Accident Control Room Initial Response”
- SACRG-2, “Severe Accident Control Room Guideline for Transients After the TSC is Functional”

BALAKOVO NPP, UNIT 4 SAMG COMPOSITION (continued)

The severe accident exit guidelines:

•SAEG-1, “TSC Long Term Monitoring Activities”,

•SAEG-2, “SAMG Termination”.

The auxiliary computational aids:

•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 BALAKOVO NPP, UNIT 4 SAMG GUIDELINES

The guidelines associated with hydrogen management are based on the computational aid CA-3. At the Balakovo plant there are no hydrogen concentration instrumentation available during accidents.

The SAG-4 guideline has been designed for application after core melt release from the reactor vessel. 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.

The SAG-1 guideline provides different ways of feeding the SG secondary side including passive water delivery from the feedwater trains and water supply from fire engines.

VALIDATION OF THE BALAKOVO NPP, UNIT 4 SAMG

Purpose of validation:

- evaluation of SAMG elements applicable for mitigation of SA consequences during the in-vessel phase of severe accidents

Stages of validation of the Balakovo NPP, Unit 4 SAMG:

- training of the Balakovo NPP specialists in the field of severe accident management,
- preparation of scenarios and computer analyses,
- validation exercises.

VALIDATION OF THE BALAKOVO NPP, UNIT 4 SAMG (continued)

The training 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.

VALIDATION OF THE BALAKOVO NPP, UNIT 4 SAMG

(continued)

The scenarios for 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.

VALIDATION OF THE BALAKOVO NPP, UNIT 4 SAMG

(continued)

Results of validation:

•12 elements of SAMG have been evaluated (DFC, DFC guidelines, guidelines for MCR, SAMG exit guidelines),
•28 comments have been made by specialists of Balakovo NPP

Main conclusion:
•SAMG of Balakovo NPP, Unit 4 is considered as acceptable,
•Comments if the plant specialists will be taken into account; SAMG corrections needed will be performed.

FURTHER ACTIVITIES ASSOCIATED WITH SAMG DEVELOPMENT FOR VVER-1000 PLANTS

The Balakovo NPP, Unit 4 SAMG documentation is being 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.

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.

THE STRATEGY “INJECT INTO THE STEAM GENERATORS”

Purposes of the strategy:

- 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.

Consequences of the strategy non-usage:

- induced hot leg and SG tubing failures due to creep,
- potential for mass and energy and FP release into containment or into secondary circuit and further into environment

THE STRATEGY "INJECT INTO THE STEAM GENERATORS" (continued)

The ways of water supply into steam generators in VVER-1000/V-320 plants:

- three groups of feedwater pumps (main feedwater, auxiliary feedwater and emergency feedwater),
- passive feeding steam generators by water from feedwater trains and deaerators,
- feeding steam generators from mobile pumps (fire engines).

Passive SG feeding: SG depressurization is needed by means of BRU-A (steam dump to atmosphere) opening

Water supply from fire engines: the modernization needed was performed; the main element of the modernization was installation of special pipeline Dn100 into the feedwater pipeline system.

THE STRATEGY "INJECT INTO THE STEAM GENERATORS" (continued)

In the Balakovo NPP the following pumps of fire 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 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.

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES

Accident scenario: total loss of feedwater

Assumptions:

- safety systems are available and able to supply borated water into the primary circuit when primary pressure becomes low enough due to AM measures;
- fire engine pumps supply water from the source of large enough volume

The following AM measures are simulated:

- opening of BRU-As (steam dump to atmosphere) at certain time moment after initial event,
- 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) when secondary pressure decreases enough for fire engine pump operation.

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES
(continued)

Accident progression without AM measures (it is assumed that 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 rise 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.

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES (continued)

Time moment of beginning of AM measures in computer analysis (BRU-A opening) was taken at the phase of the core heatup. 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 and decrease of primary coolant temperature in the core inlet and outlet.

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.

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES
(continued)

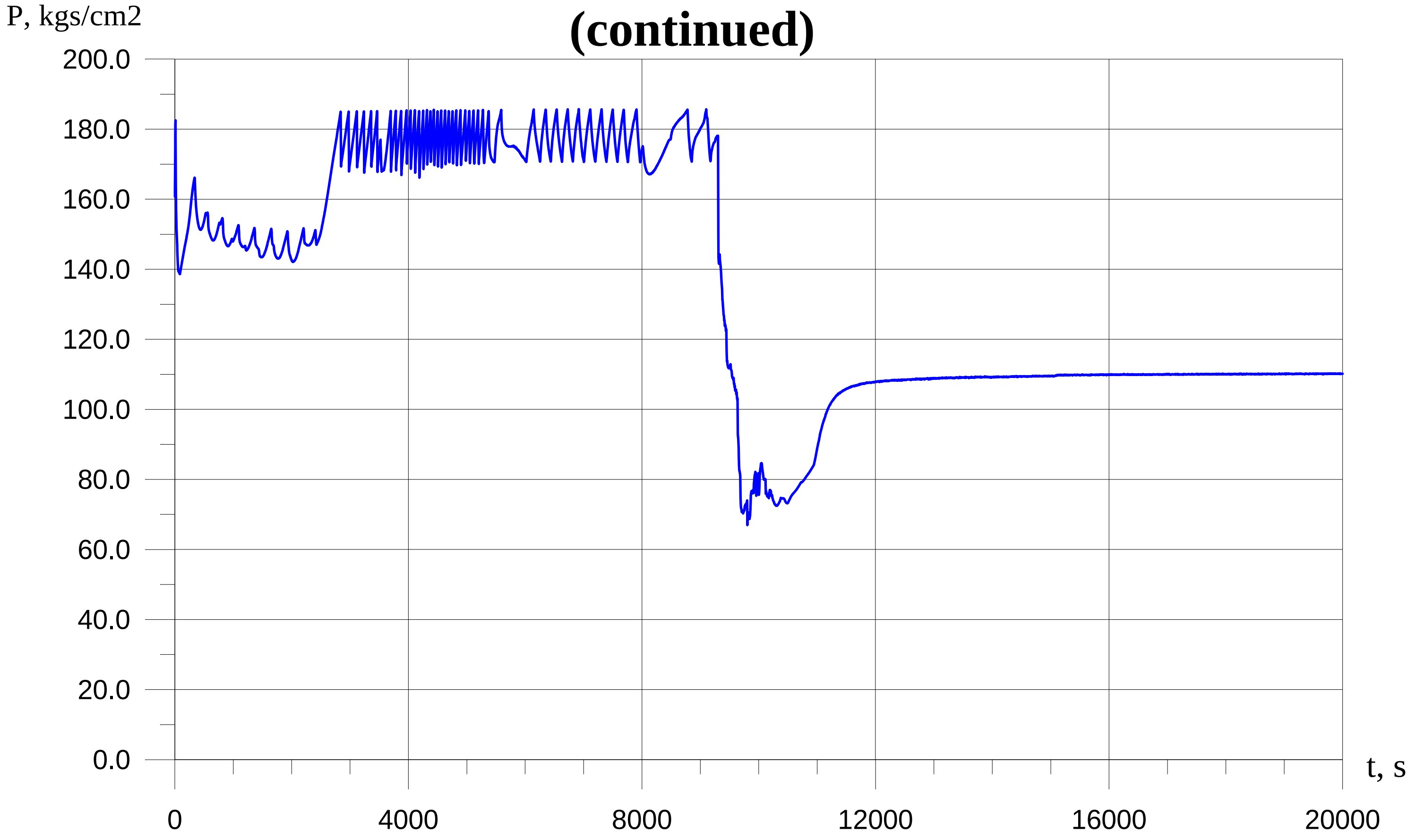


Figure 1. Primary pressure

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES (continued)

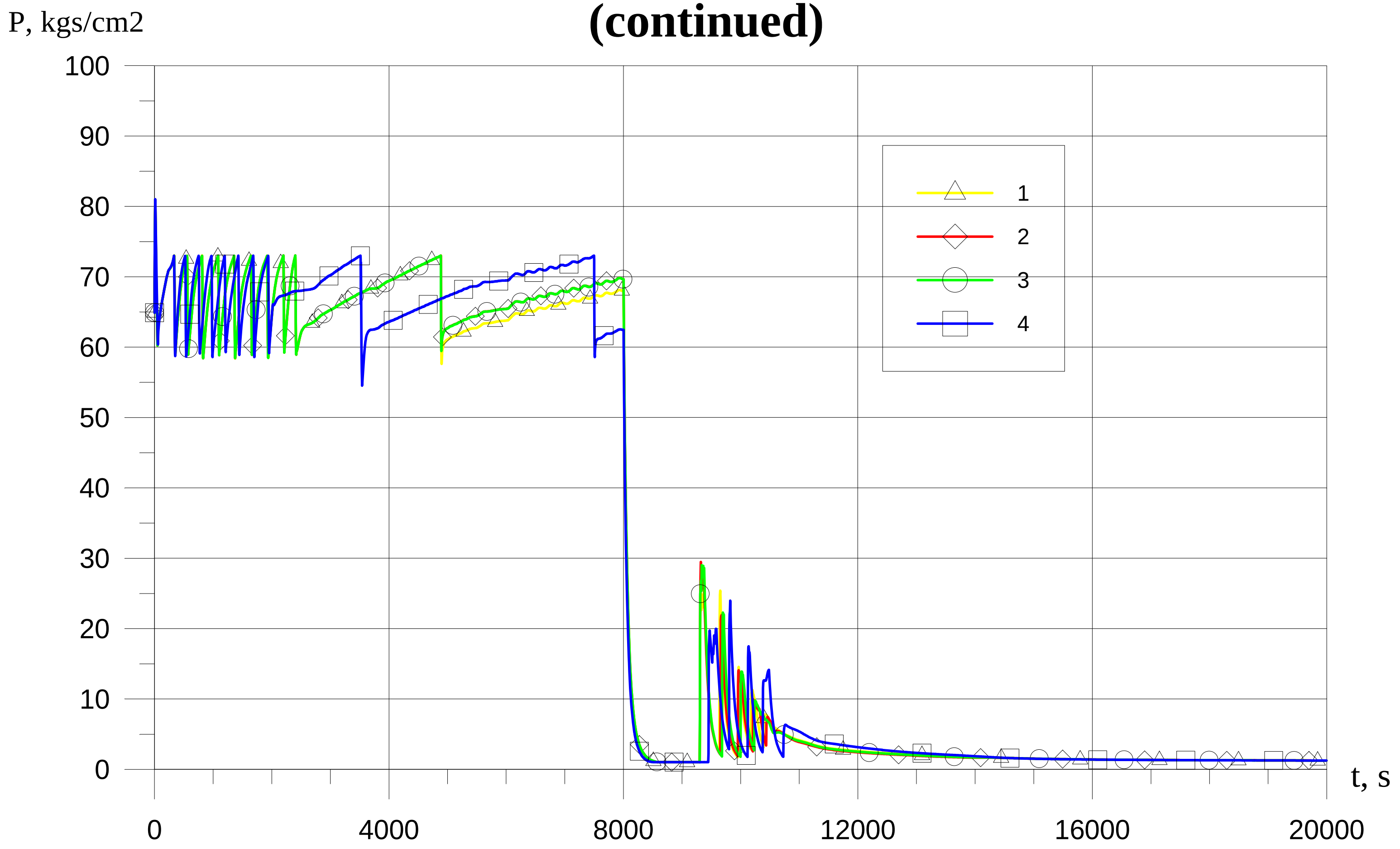


Figure 2. Pressure in steam generators

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES (continued)

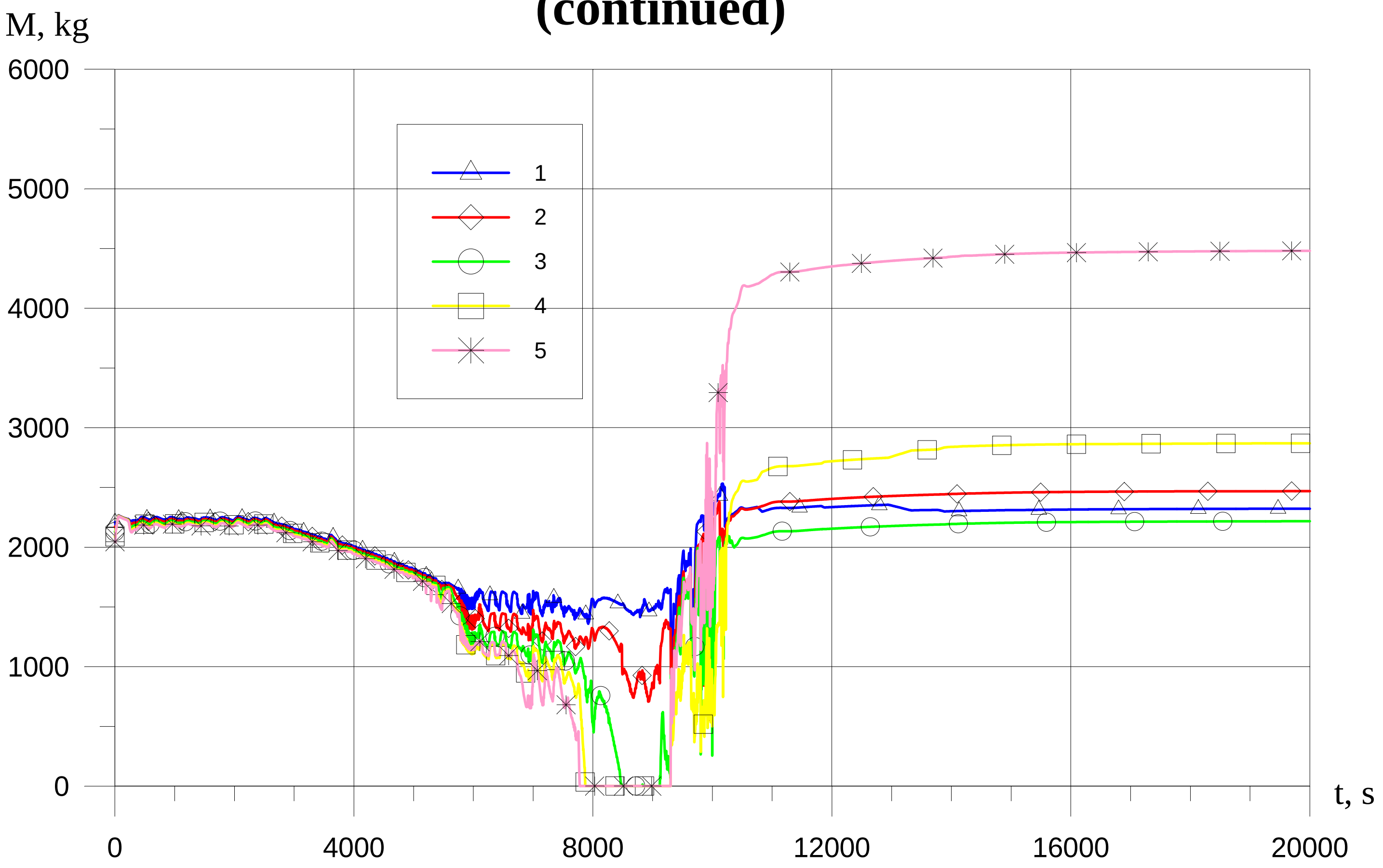


Figure 3. Coolant mass in the core axial nodes 1 to 5

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES
(continued)

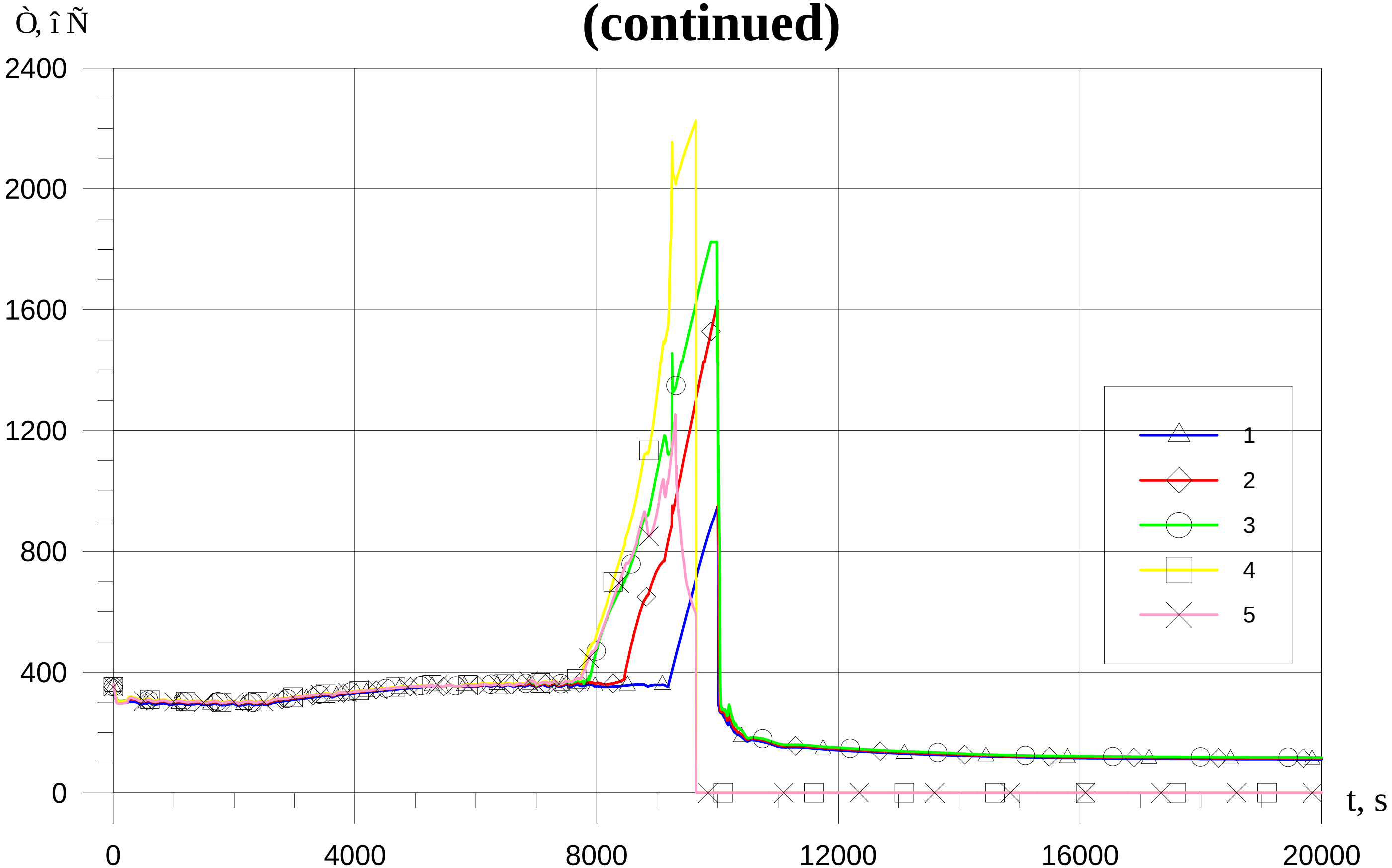


Figure 4. Fuel rod cladding temperatures in the core axial nodes 1 to 5

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES
(continued)

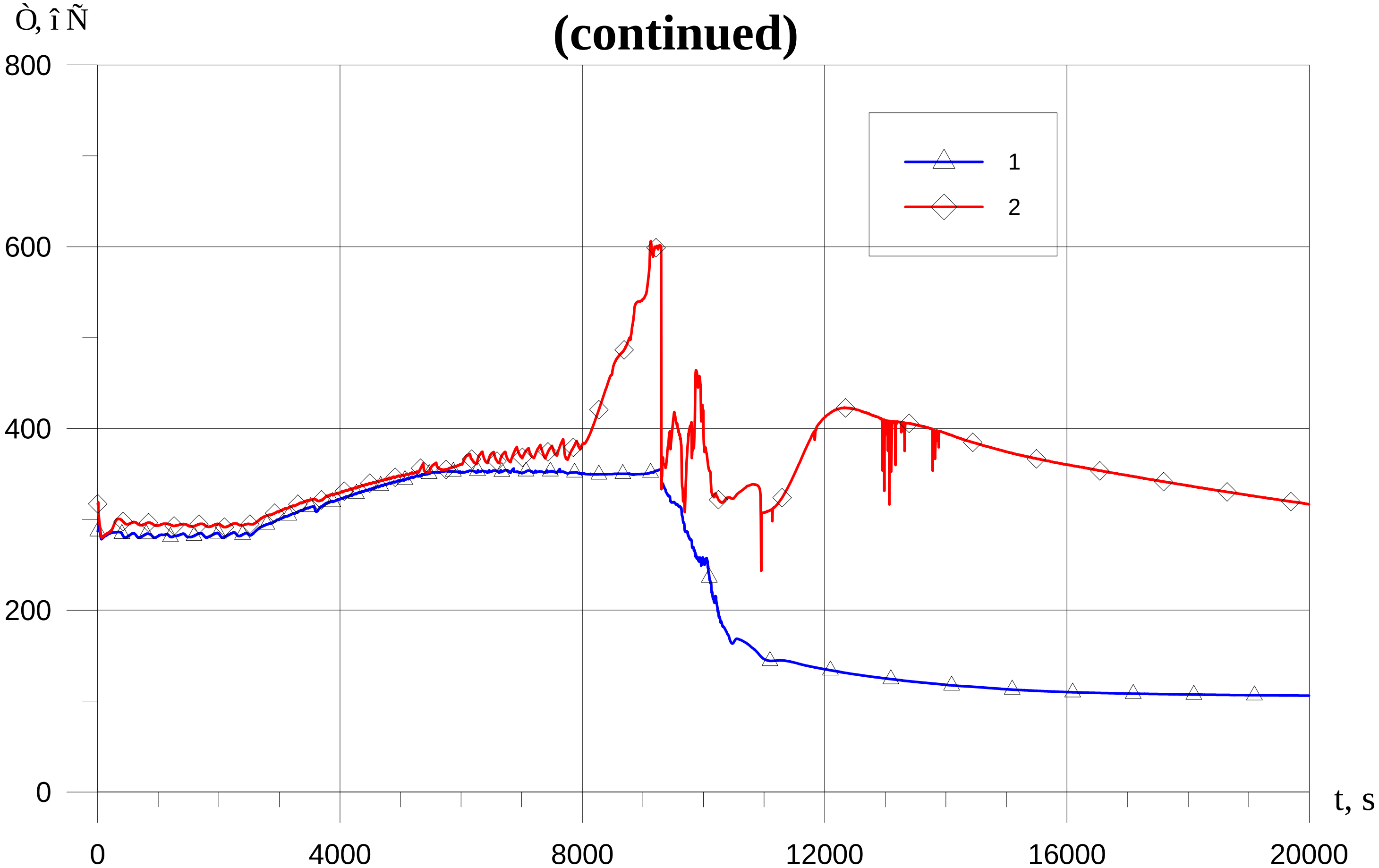


Figure 5. Temperature of coolant at core inlet and outlet

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES (continued)

Summary of results for case without AM actions:

- beginning of core heatup – about 7740 s;
- reactor vessel failure – after 19000 s;

Base case with AM actions:

- BRU-A opening – 8000 s;
- beginning of water supply into SGs – 8500 s.

Additional variants considered:

- beginning of water supply into SGs – 13500, 15500, 17500 s (in the last case the reactor vessel failure occurs).

RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES (continued)

Conclusions:

- capability to prevent reactor vessel failure in case of water supply into SGs from fire engine pumps is shown;
- scenarios realistic with respect to duration of water supply with fire engine pumps are to be analyzed based on information from Balakovo NPP
- nodalization scheme of VVER-1000 steam generator for MELCOR is to be improved;
- some MELCOR model parameters are to be adjusted based on analyses with mechanistic codes (e.g. ATHLET-CD)

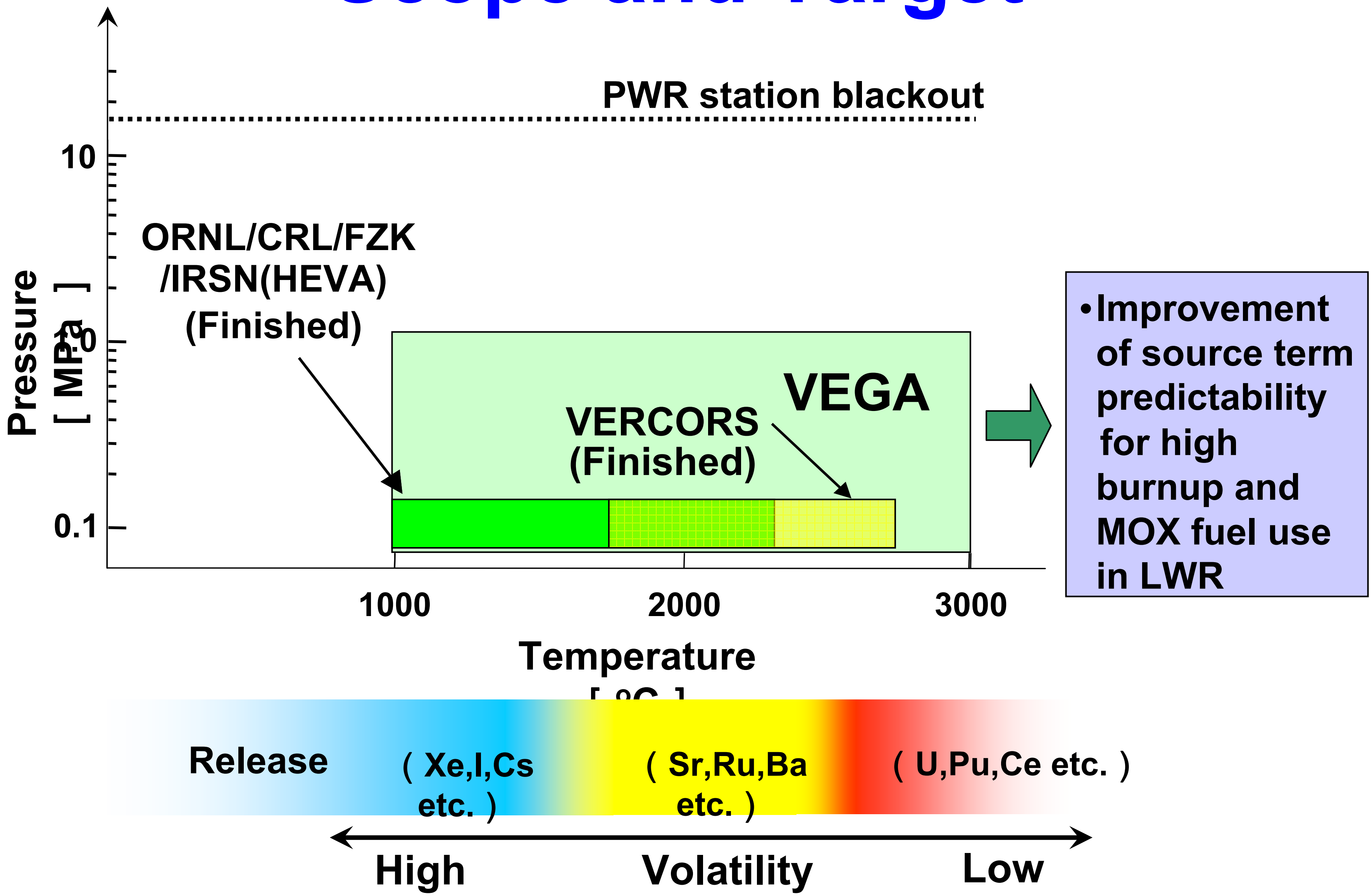
Ambient Pressure-dependent Radionuclide Release from Fuel Observed in VEGA Tests under Severe Accident Condition and Influence on Source Term Evaluation

Akihide HIDAKA

Japan Atomic Energy Agency

Presented at OECD/NEA Workshop on Implementation of Accident Management Measures (ISAMM-2009) on October 26-28, 2009, Switzerland

Scope and Target

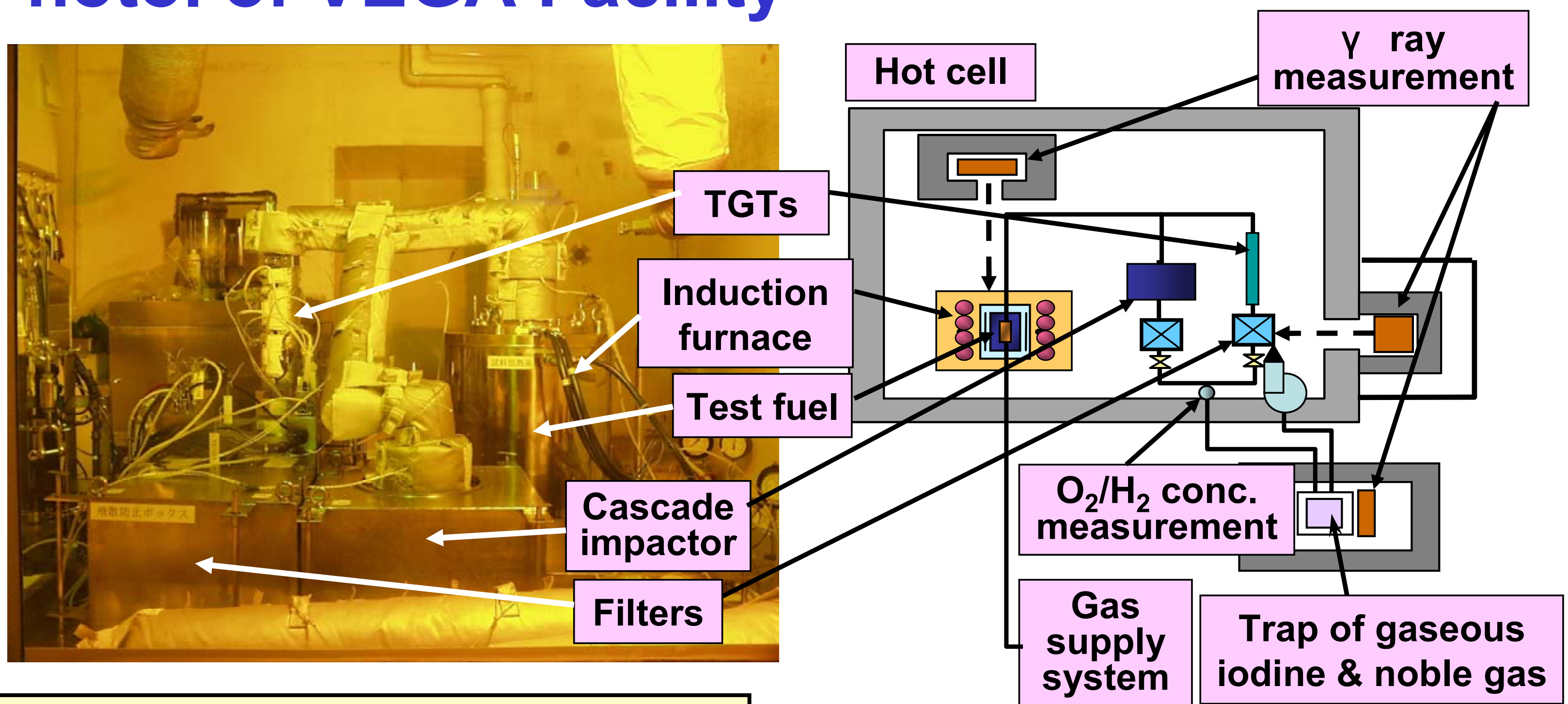


VEGA (Verification Experiment of radionuclide Gas/Aerosol Release)

Schematic of VEGA Test Apparatus

Radionuclide released from 2 pellets heated by induction coil is delivered by steam or He to downward piping and quantified by gamma ray measurement or chemical analyses.

Photo. of VEGA Facility



Required performance and efforts

- Max. pressure >1.0MPa ➡ Installation of furnace inside chamber to minimize radionuclide leakage
- Max. temp. > 3150K ➡ Development of ThO_2 crucible that is stable under oxidizing and high temperature conditions

VEGA Test Matrix

PWR, BWR, MOX reference

BWR pressure effect

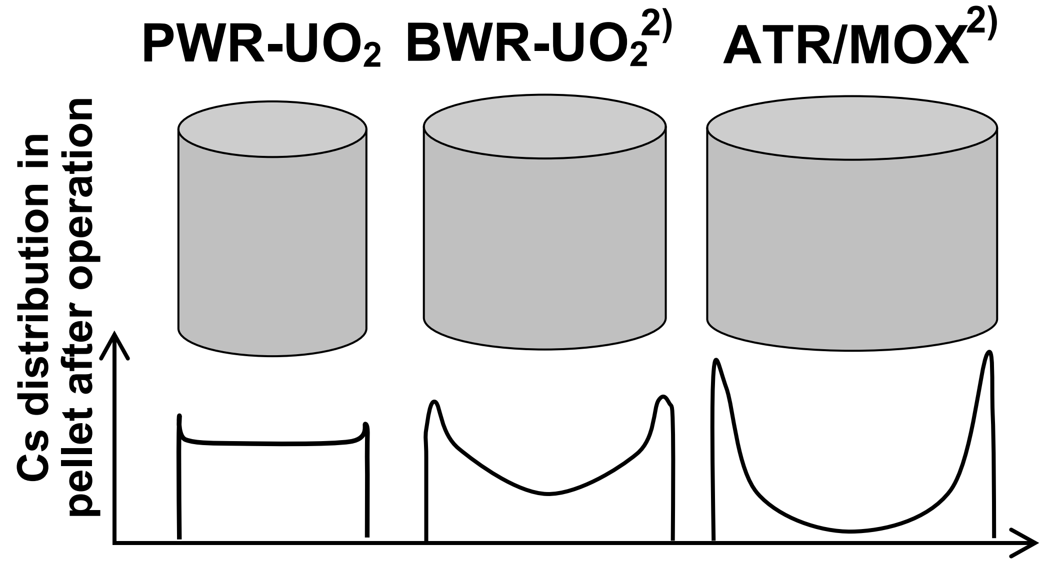
Test No.	1	2	3	4	5	M1	6	M2	7	8
Date of heat up	1999 Sep.	2000 Apr.	2000 Oct.	2001 Jun.	2002 Jan.	2002 Aug.	2002 Dec.	2003 Aug.	2003 Dec.	2004 Oct.
Fuel specimen	UO_2 PWR	UO_2 PWR	UO_2 PWR	UO_2 PWR	UO_2 PWR	MOX ATR	UO_2 BWR	MOX ATR	UO_2 BWR	UO_2 BWR
Cladding	No	No	No	Yes	No	No	Yes	No	yes	No
Burnup (GWd/tU)	47	47	47	47	47	43	56	43	56	56
Re-irra-diation	No	No	No	No	NSRR	No	JRR-3	No	JRR-3	No
Max Temp. (K)	2773	2773	3123	2773	3123	3123	2773	3123	2773	3123
Carrier gas	He	He	He	Steam	He	He	Steam	He	Steam	He
Pessure (MPa)	0.1	1.0	0.1	0.1	1.0	0.1	0.1	1.0	1.0	1.0
Remarks	PWR 1st test	High press.	PWR ref.	Steam ref.	Re-irradia-tion	MOX ref.	Short-life / Steam	MOX high press.	Short-life / High press.	BWR ref.

PWR pressure effect

MOX pressure effect

Fuel Specimens Used in VEGA

- PWR, BWR : Irradiated at Japanese commercial reactors
- MOX : Fabricated by JAEA and irradiated at ATR Fugen
  - ⇨ ^{239}Pu burns mainly (Slight difference in fission yield between ^{235}U and ^{239}Pu) ,
  - ⇨ Fabricated to minimize the size of Pu rich spot



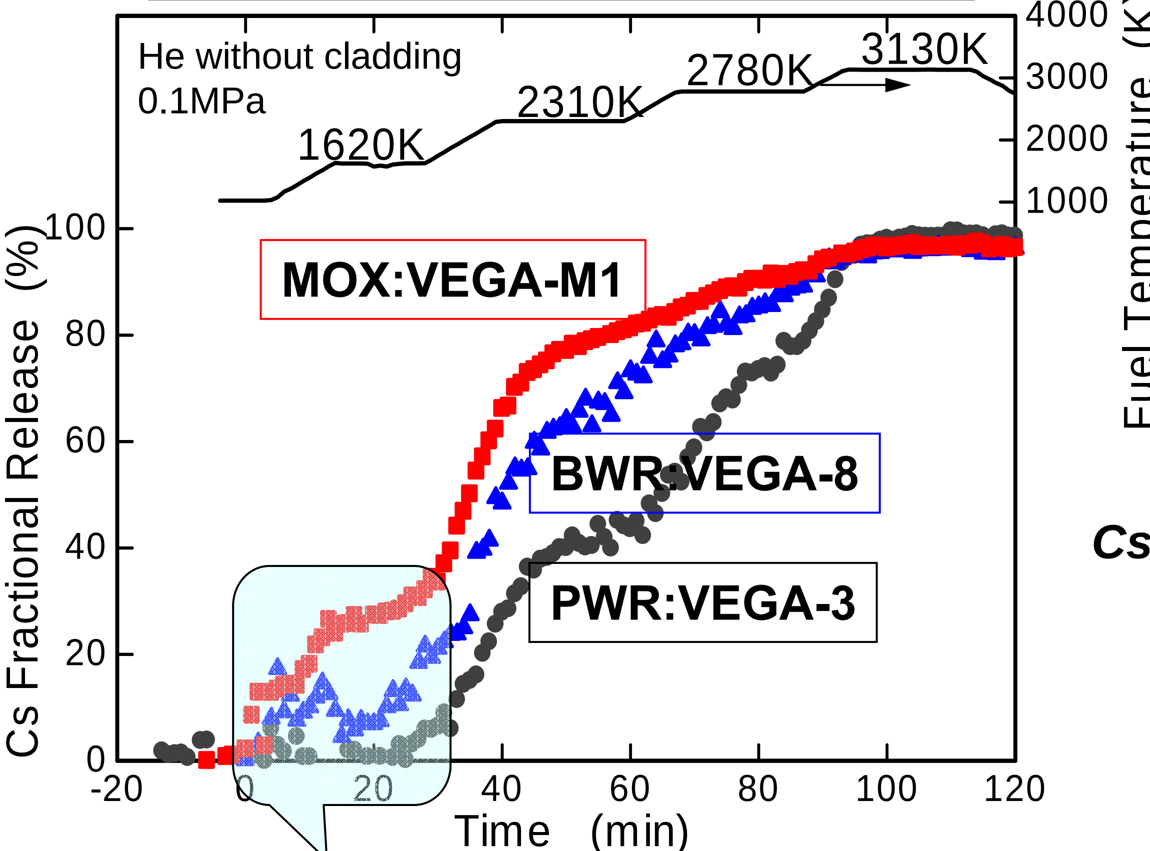
Pellet diameter (m)	0.0081	0.0104	0.0124
Burn-up (GWd/t)	47	56	43
Theoretical density (%)	95	97	95
Pu amounts after operation (wt%)	1.1	1.2	2.9
Linear heat rate during operation (kW/m)	18	26	28
Temperature during operation (center/periphery, K) 1)	1000/660	1500/870	1700/900
FP gas release during operation (%)	0.4	12	20

1) Averaged temperature calculated by FRAPCON-2

2) Movement of Cs to pellet periphery due to high temperature at center region during operation

Reference Tests with Different Fuels

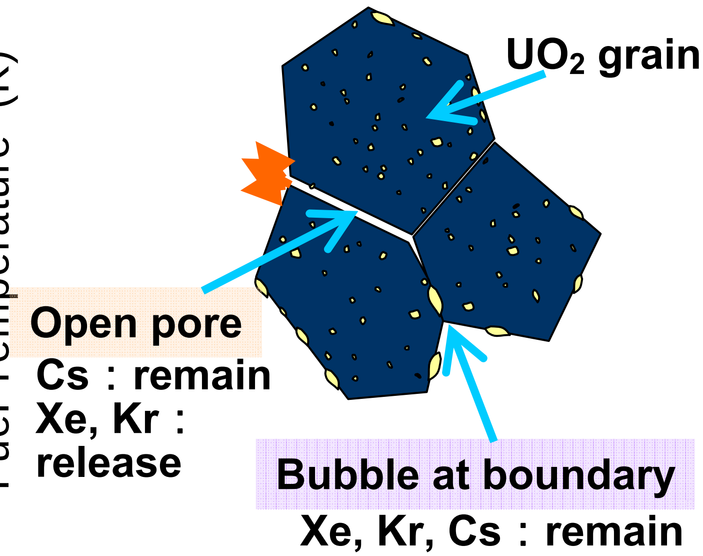
Purpose: Understanding of basic release behavior from different fuels



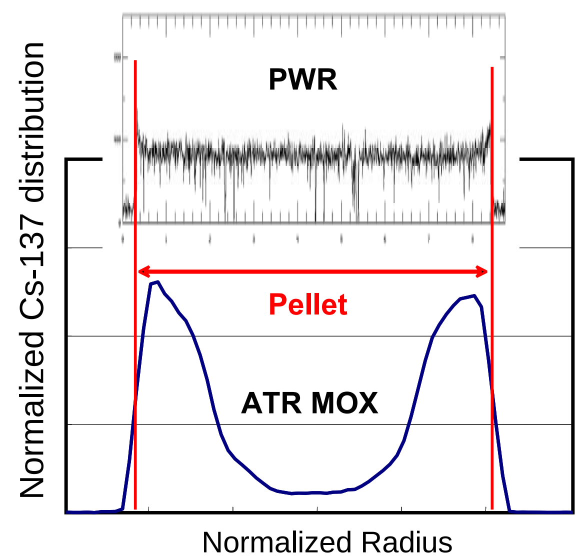
Difference in Cs release among fuels
→ Difference in Cs moved from center pellet to periphery during power operation

No large difference in Cs release above 2300K among PWR, BWR and MOX

Fuel after irradiation



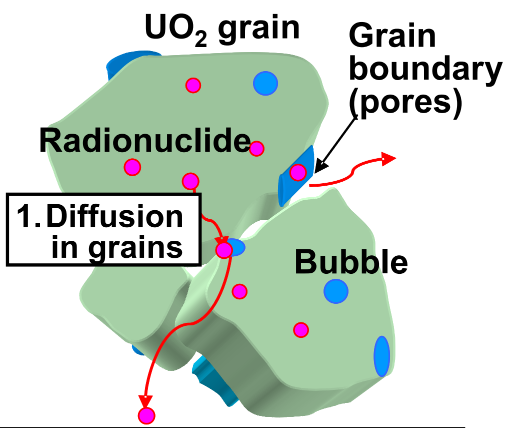
Cs distribution in pellet diameter



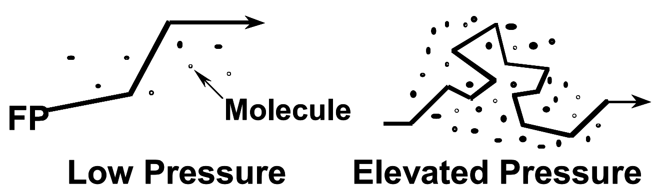
Effect of Pressure on Release (1/5)

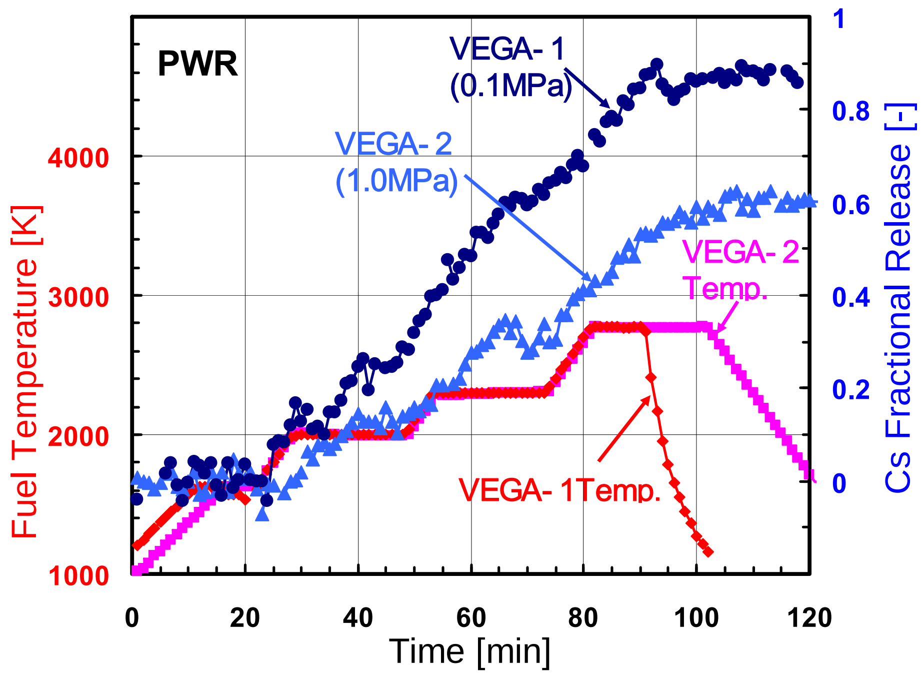
- Experimentally first observation -
Cesium fractional release from PWR fuel at 1.0MPa suppressed by 30% compared with that at 0.1MPa

Schematic of UO_2 grains & pores and process of radionuclide release



Pressure dependent gaseous diffusion



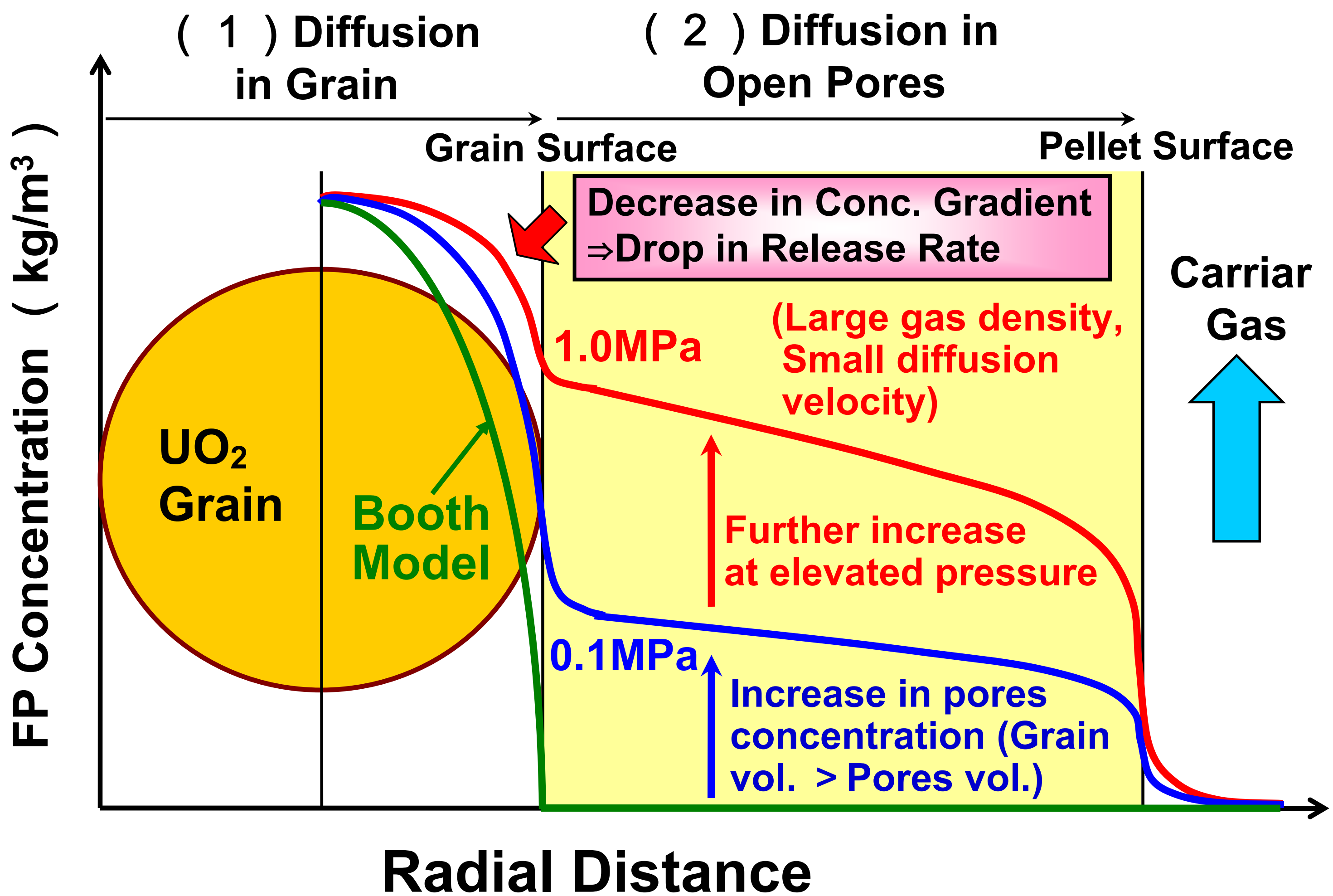


Diffusion in grain is much slower than other processes at 0.1MPa. Previous models consider only diffusion in grains ($D_1 \propto \exp(-1/T)$) .

Since gaseous diffusion ($D_2 \propto T^{1.5}/P$) becomes slow at elevated pressure, expected process is not governed only by diffusion in grain but also by diffusion in pores.

Proposal of 2 stage diffusion model considering diffusion in grains & pores

Effect of Pressure on Release (2/5)



Despite large difference in diffusion coeff. between grains and pores, small difference in diffusion time between them at elevated pressure. Pressure effect could appear in case of the rate-determining step located at diffusion in both of grains and pores.

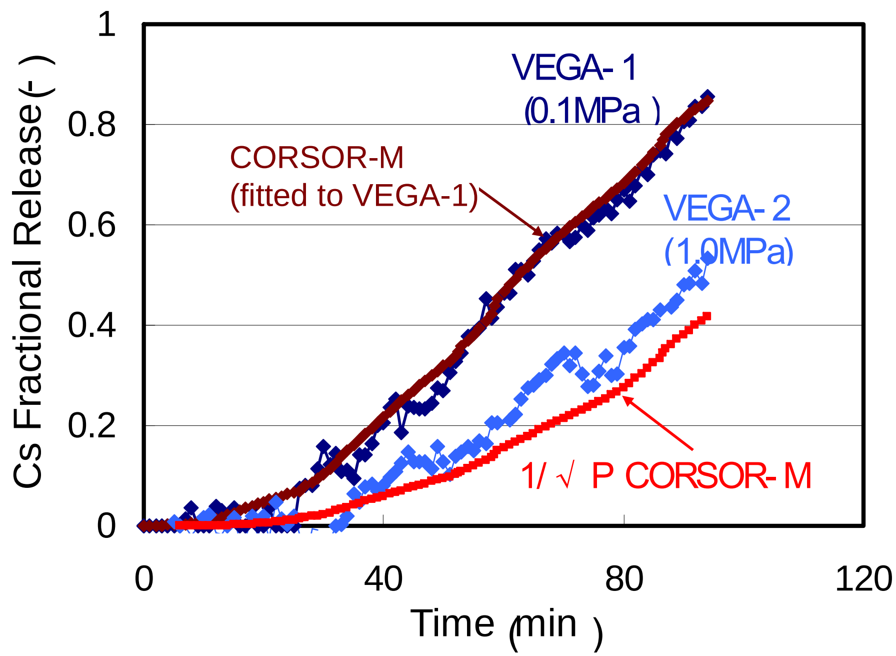
Effect of Pressure on Release (3/5)

- Confirmed reproducibility of observed pressure effect by 2 stage diffusion model solved by numerical calculation

⇨ Too complicated for source term analysis

- Derivation of a simplified model considering a part of the rate determining step located at gaseous diffusion in pores

The multiplier $\sqrt{P_0}/\sqrt{P}$ comes from the pressure dependency of gaseous diffusion flux in pores at pellet surface



Previous release model (CORSOR-M)

$$k = k_0 \exp\left(-\frac{Q}{RT}\right)$$

Proposed model with pressure effect ($1/\sqrt{P}$ CORSOR-M)

$$k = k_0 \sqrt{\frac{P_0}{P}} \exp\left(-\frac{Q}{RT}\right) \quad (P \geqq 0.1\text{MPa})$$

- Modified CORSOR-M so that calculation at 0.1MPa might agree with the measurement
- $1/\sqrt{P}$ CORSOR-M agrees reasonably with measurement at 1.0MPa although model slightly overestimates the pressure effect.

Hidaka, et al., "Proposal of Simplified Model of Radionuclide Release from Fuel under Severe Accident Conditions Considering Pressure Effect," J. Nucl. Sci. Technol. 41 [12], 1192-1203 (2004).

Effect of Pressure on Release (4/5)

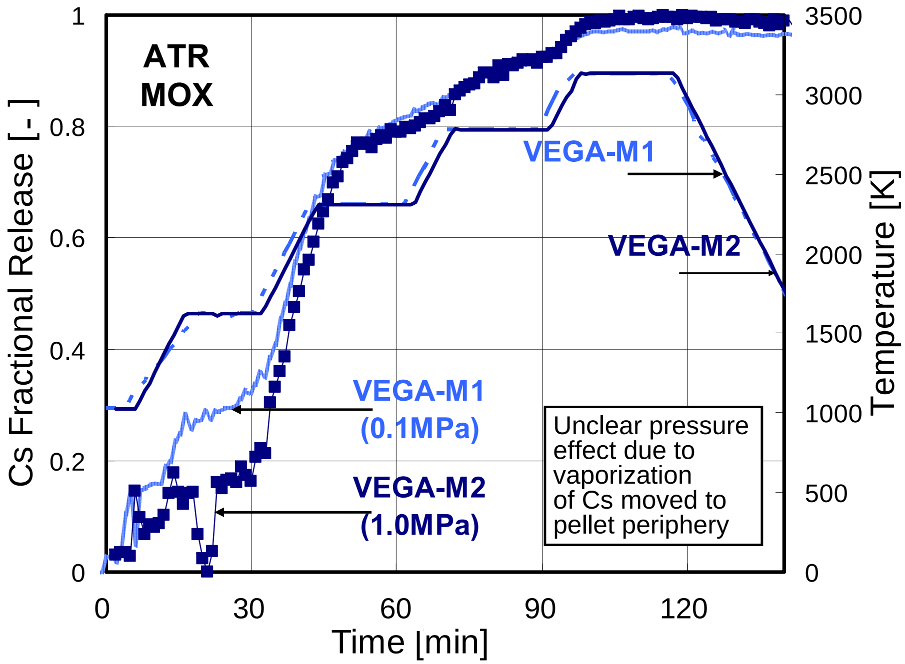
Final Releases of Pressure Effect Tests in VEGA

Test No.	Fuel	Test conditions	Fractional release (%)							
			γ ray measurement (half-life)							
			^{137}Cs 30yr	^{125}Sb 3yr	^{131}I 8d	^{132}Te 3d	^{140}Ba 13d	^{106}Ru 1yr	^{103}Ru 39d	^{140}La 2d
1	PWR 47GWd/tU No cladding	2,773K, 0.1MPa, He	86	89	–	–	–	5	–	–
2		2,773K, 1.0MPa, He	61	68	–	–	–	0	–	–
M1	ATR/MOX 43GWd/tHM No cladding	3,123K, 0.1MPa, He	97	95	–	–	–	6	–	–
M2		3,123K, 1.0MPa, He	98	96	–	–	–	3	–	–
6	BWR 56GWd/tU with cladding	624hr JRR-3 Re-irradiation H_2O, 2,773K, 0.1MPa	93	–	97	98	49	14	16	3
7		624hr JRR-3 Re-irradiation H_2O, 2,773K, 1.0MPa	98	83	96	98	34	6	7	4

- The pressure effect was observed in PWR fuel but not clearly in BWR & MOX fuels. Possible reason is a difference in Cs moved to the pellet periphery during operation and domination of the vaporization from periphery that is not affected so much by pressure.
- BWR fuel were re-irradiated before heat-up test at JRR-3 with low thermal neutron flux to accumulate short-life radionuclide. In BWR fuel tests, the pressure effect was observed in release of low-volatile radionuclide in spite of no observation of effect in volatile one.

Effect of Pressure on Release (5/5)

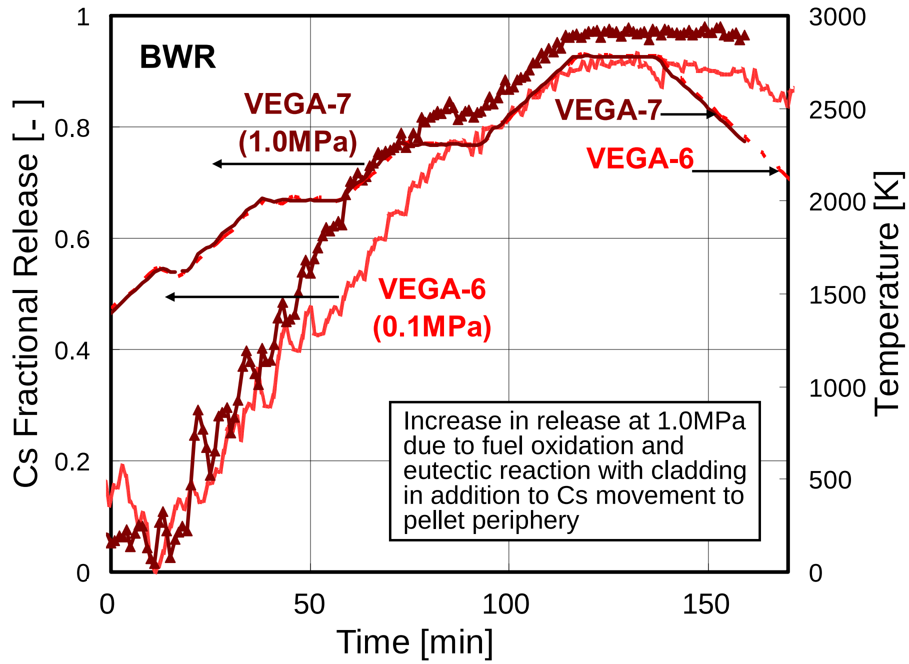
Pressure Effect in MOX Fuel



- In MOX, release below 1700K at 1.0MPa slightly became smaller than that at 0.1MPa.
- Possible reason is an increase in boiling point of Cs at elevated pressure.
- Almost no pressure effect was observed in release above 1700K.

Pressure Effect in BWR Fuel

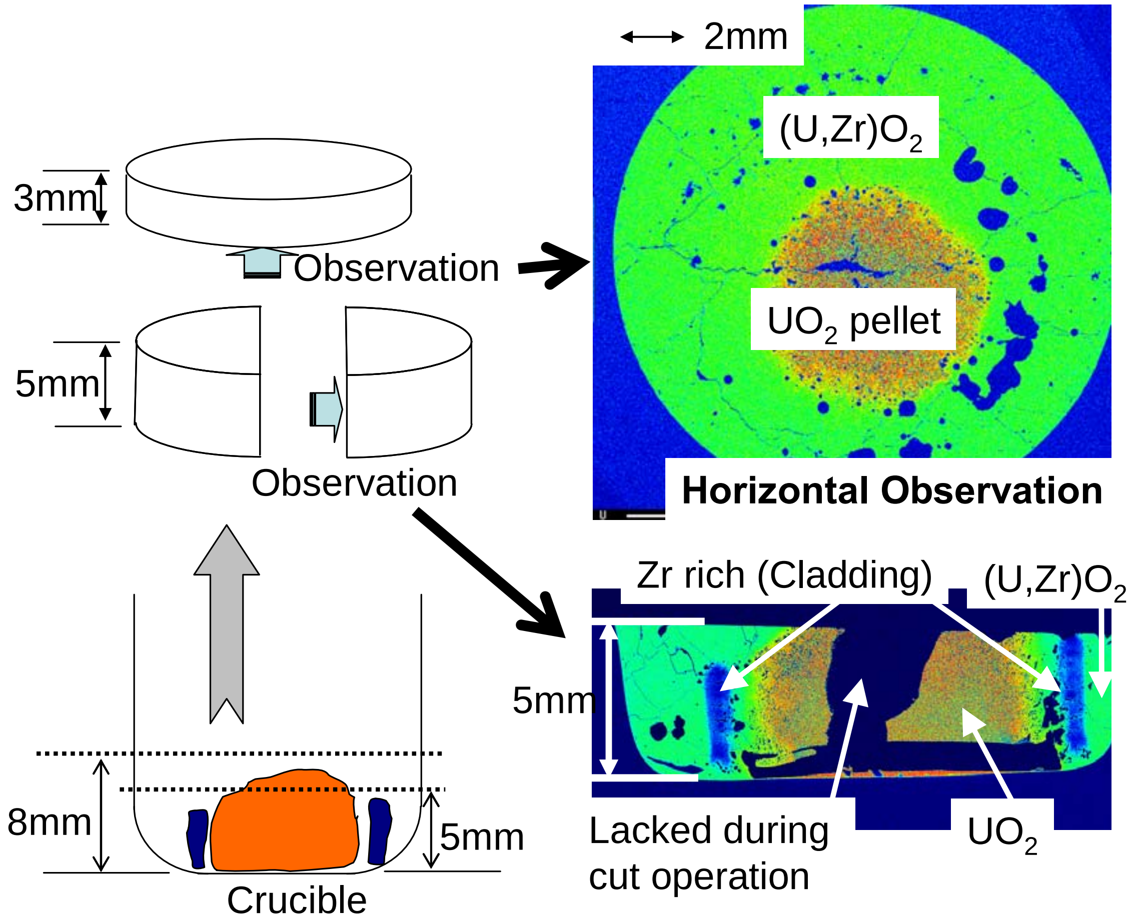
(with cladding under steam condition)



- In BWR, to the contrary, release at elevated pressure slightly increased.
- Possible reason is that steam conc. at 1.0MPa was higher by a factor of 4 than that at 0.1MPa from the test condition.
- This could have enhanced fuel oxidation at 1.0MPa compared with that at 0.1MPa.

Phenomena Affecting Release

Crucible After Oxidizing Condition Test



Vertical Observation

- Fuel oxidation causes increase in defect in UO_2 grain matrix.
- Contact of liquefied Zr with fuel causes reduction of UO_2 and decrease in melting point of fuel

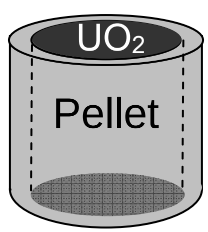
Increase in Release

Fuel pellet height
Before: 20 mm
After: 5～8 mm

Liquefaction of 60 - 75% of pellet

Liquefied （U, Zr）O_2 reached 8mm in height from crucible bottom

Possible overestimation of eutectic reaction in VEGA



Fuel oxidation depends on the ratio of surface area/volume. Fuel oxidation area could be limited near the cladding rupture point in real NPPs while both of top and bottom faces of pellets exposed to steam in VEGA would result in overestimation of fuel oxidation.

Summary of Pressure Effect

Fuel (center / peripheral temp. K)	Carrier gas	Temp.	Noble gas	Cs / iodine	Low volatile radionuclide
PWR without cladding (1000/660)	He	< 2300K	O	O	-
		> 2300K	O	O	O
BWR with cladding (1500/870)	Steam + He	< 2300K	X	X	-
		> 2300K	X	X	O
MOX without cladding (1700/900)	He	< 2300K	X	X	-
		> 2300K	O	o	O

O Effect measured by test, O Not measured but mechanistically possible, o Small effect measured, X No effect measured, - Mechanistically impossible (Not measured)

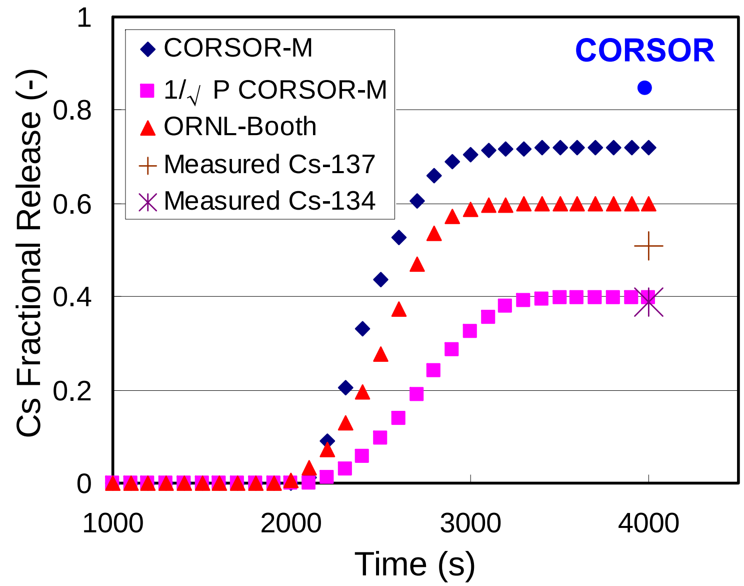
Important Results

- The pressure effect could appear when 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.
- Release of low-volatile radionuclide depends on neither the irradiation history nor fuel liquefaction while the pressure effect could appear because the form of low-volatile radionuclide at time of release from grains could be vapor.
- Relationship between the pressure effect and the irradiation history, fuel oxidation, eutectic reaction is expected to be further examined in other future tests that simulate better the real conditions during severe accidents.

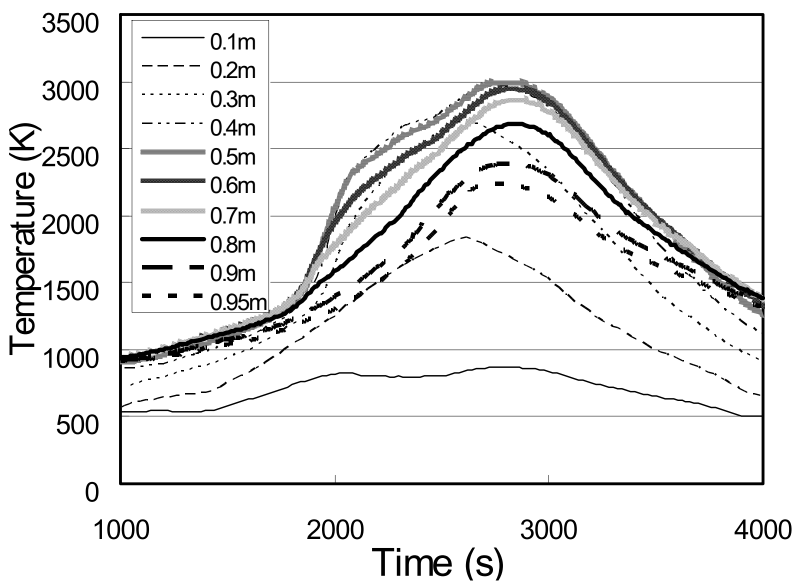
Verification of Proposed $1/\sqrt{P}$ CORSOR-M

- To verify effectiveness of proposed model, application of the model to other experiment at elevated pressure.
- Severe Fuel Damage (SFD) Test 1-4 at Power Burst Facility in USA in 1985
  - A test bundle: 26 irradiated (36GWd/t) PWR type fuel rods, 2 fresh instrumented rods, and 4 Ag-In-Cd control rods
  - Test simulated S_2D sequence at 6.95MPa
  - Finally 18% of fuel liquefaction

Calculated Cs release during SFD 1-4



Temperature distribution for SFD 1-4 bundle



- Measured ^{137}Cs and ^{134}Cs fractional releases at the end of the test was 51% and 39%, respectively.
- At the time, best estimate analysis with CORSOR model predicted fractional release of 83%.
- $1/\sqrt{P}$ CORSOR-M model gave more reasonable prediction compared with the conventional ones.

Influence on Source Term Evaluation (1/3)

Issues: Decrease in radionuclide release under elevated pressure may affect PWR source term evaluation and AM measures such as intentional primary system depressurization.

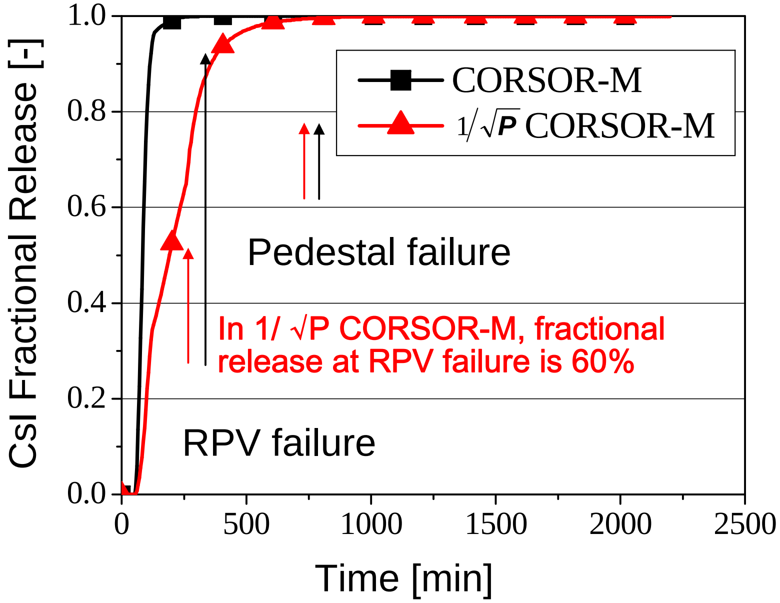
- Analyses using JAEA's THALES-2 with CORSOR-M and 1/ $\sqrt{P}$CORSOR-M
  - Reference plant :
    BWR5 with Mark-II containment
    Rated power: 3,300 MWt
    Pressure of RCS: 7.5 MPa
  - Accident sequence: TQUX (Loss of feed water followed by failures of both HPI and ADS)

⇨ Perspectives obtained from BWR analyses can be also applied to PWR.

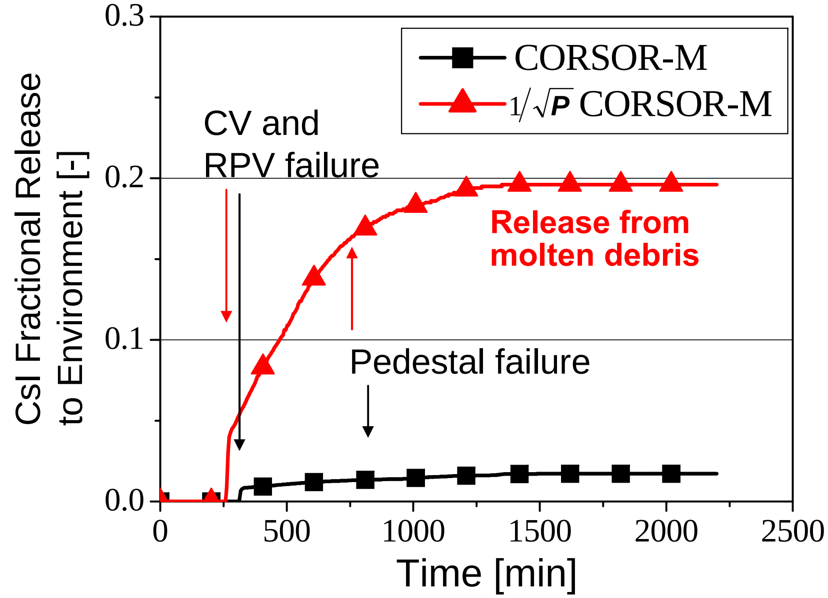
- Two sensitivity calculations on timing of CV failure
  1) Early CV failure : Simultaneous failures of RPV and CV
  2) Late CV failure : CV overpressure due to accumulation of non-condensable gases

Influence on Source Term Evaluation (2/3)

CsI Release from Fuel



Source Term in case of early CV failure



Event Timings and Source Terms

Events	TQUX (late CV failure)		TQUX (early CV failure)	
(min)	CORSOR-M	$1/\sqrt{P}$ CORSOR-M	CORSOR-M	$1/\sqrt{P}$ CORSOR-M
Core melt initiation	53	53	53	53
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

Pressure effect ⇨ **Acceleration of accident progression, Increase in source term at early CV failure**

Influence on Source Term Evaluation (3/3)

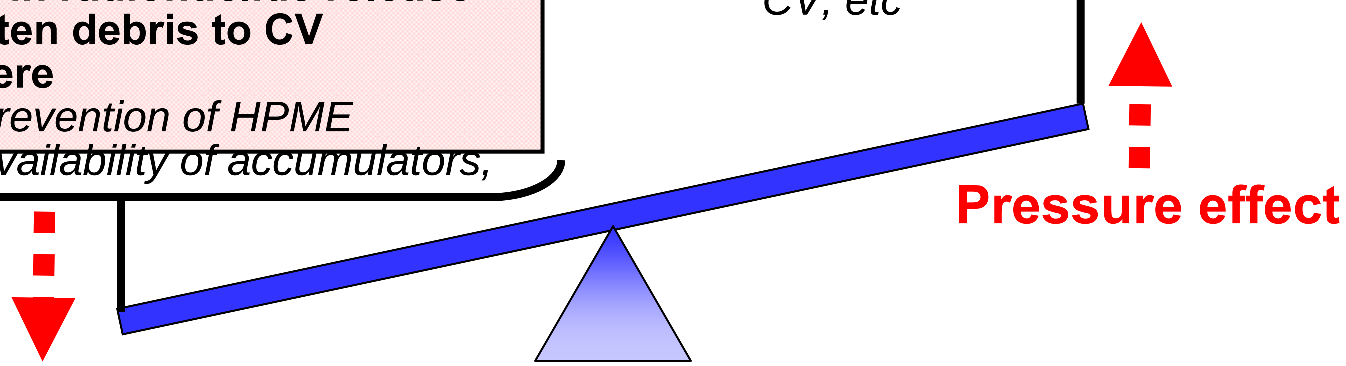
Study on intentional primary system depressurization considering pressure effect on release based on present THALES-2 analyses

Advantages

- **Delay in accident progression before and after RPV failure due to reduction of decay heat from fuel or molten debris**
- **Mitigation of source terms in case of early CV failure due to decrease in radionuclide release from molten debris to CV atmosphere**
- *Prevention of HPME*
- *Availability of accumulators, etc.*

Disadvantages

- **Enhancement of radionuclide release into primary system**
- *Discharge of primary coolant including radionuclide into CV, etc*



Future issues

- Present analyses with the pressure effect showed increase or decrease in source terms depending on the timing of CV failure. Detailed analyses are further needed.
- Systematic evaluations with the pressure effect are desirable for various accident sequences considering combination of AM measures.

Conclusions

- Totally 10 tests were performed in VEGA under the highest pressure or temperature conditions among previous studies from 1999 to 2004.
- Tests with PWR fuel at 1.0MPa showed experimentally first that Cs release was suppressed by about 30% compared with that at 0.1MPa.
- Observed pressure effect could be explained by 2-stage diffusion model and predicted by a proposed $1/\sqrt{P}$ CORSOR-M model.
- In BWR and MOX, however, the pressure effect was not observed clearly due to domination of vaporization from Cs deposited at peripheral pellet as a result of higher linear heat rate during operation and differences in conditions such as fuel oxidation and eutectic reaction.
- Relationship between the pressure effect and the factors described above is desirable to be further examined by other future tests considering better the actual conditions and irradiation history of fuel.
- The decrease in release under elevated pressure may affect PWR source terms and AM measures. Present analyses with the pressure effect suggested that the intentional depressurization has more advantages such as delay in accident progression and mitigation of source terms at early CV failure despite increase in release into RCS.
- The effect of pressure on consequences needs to be evaluated systematically for various accident sequences and AM measures.

Backup Slides

Formulation of Pressure Effect (1/2)

Proposed 2 Stage Diffusion Model

1. Diffusion in UO_2 Grain

$$\frac{\partial}{\partial t}C_1(R,i)=\frac{1}{r^2}\frac{\partial}{\partial r}\left[r^2 D_1(i)\frac{\partial}{\partial r}C_1(R,i)\right] \qquad (1)$$

a) Diffusion time in grain = a^2/D_1
b) Diffusion time in pore = $\alpha L^2 / \beta' D_2$
a)=b) for Kr at 2300K

2. Diffusion in Open Pores

Coefficient related to change in coordinate system

$$\alpha\frac{\partial}{\partial t}C_2(i)=\frac{1}{R}\frac{\partial}{\partial R}\left[\underbrace{R\beta' D_2(i)\frac{\partial}{\partial R}C_2(i)}_{\text{Diffusion in pellet diameter}}\right]+(1-\alpha)\underbrace{\frac{3}{a}D_1(i)\frac{\partial C_1(R,i)}{\partial r}\bigg|_{r=a}}_{\text{FP inflow rate}} \qquad (2)$$

Boundary condition: Continuity of concentration from grain surface to pores

Porosity of fuel (-)	α	0.05
Porosity of open pore (10^{-4} ×Resistance of diffusion from closed to open pore (-)	β'	1.1×10^{-6} Unknown parameter
Radius of UO_2 grain (m)	a	6.0×10^{-6}
Pellet diameter (m)	L	4.0×10^{-3}

	Grain	Pore
Radial coordinate	r	R
Concentration(kg/m^3)	C_1	C_2
Diffusion coeff.(m^2/s) Kr @2300K	D_1 10^{-13}	D_2 10^{-3}

2 stage diffusion model reproduced well decrease in Cs release at 1.0MPa.

Formulation of Pressure Effect (2/2)

Derivation of simplified model

Approximation by one-dimensional diffusion under steady state

Inflow rate from grain to pore

$$D_1 \cdot \frac{\partial C_1}{\partial R}\Big|_{R=a} \cdot S_1 \approx D_1 \cdot \frac{\overline{C_1} - \overline{C_2}}{\Delta R} \frac{V_1}{a} \quad (1)$$

Release rate from pore to fuel outside

$$D_2 \cdot \frac{\partial C_2}{\partial R}\Big|_{R=L} \cdot S_2 \approx D_2 \cdot \frac{\overline{C_2}}{\sqrt{\pi D_2 t}} \cdot S_2 = \sqrt{\frac{D_2}{\pi t}} \cdot \frac{V_2}{L} \overline{C_2} \quad (2)$$

$$k(P) = \frac{\sqrt{\frac{D_2}{\pi t}} \frac{V_2}{L} \overline{C_2}}{\underbrace{\overline{C_1} V_1 + \overline{C_2} V_2}_{\text{Inventory}}} \quad (3)$$

Pressure dependency of k can be expressed by Eq.(4) using conditions : Eq(1)=Eq.(2) and Eq.(3)

Hidaka, et al., "Proposal of Simplified Model of Radionuclide Release from Fuel under Severe Accident Conditions Considering Pressure Effect," J. Nucl. Sci. Technol. 41 [12], 1192-1203 (2004).

$$\frac{k(P)}{k(P_0)} = \sqrt{P_o/P} \cdot \frac{1 + \boxed{qT^{3/4}\exp(\theta/T)}}{1 + qT^{3/4}\exp(\theta/T)\sqrt{P_o/P}} \quad (4)$$

Where q does not depend on temperature.

$$\left(D_1 = D_o \exp(-\theta/T),\ D_2(P) = dT^{3/2}/P\right)$$

Under the test conditions, this term <<1

$$\frac{k(P)}{k(P_0)} \cong \sqrt{\frac{P_0}{P}} \quad (5)$$

Existing release model（CORSOR-M）

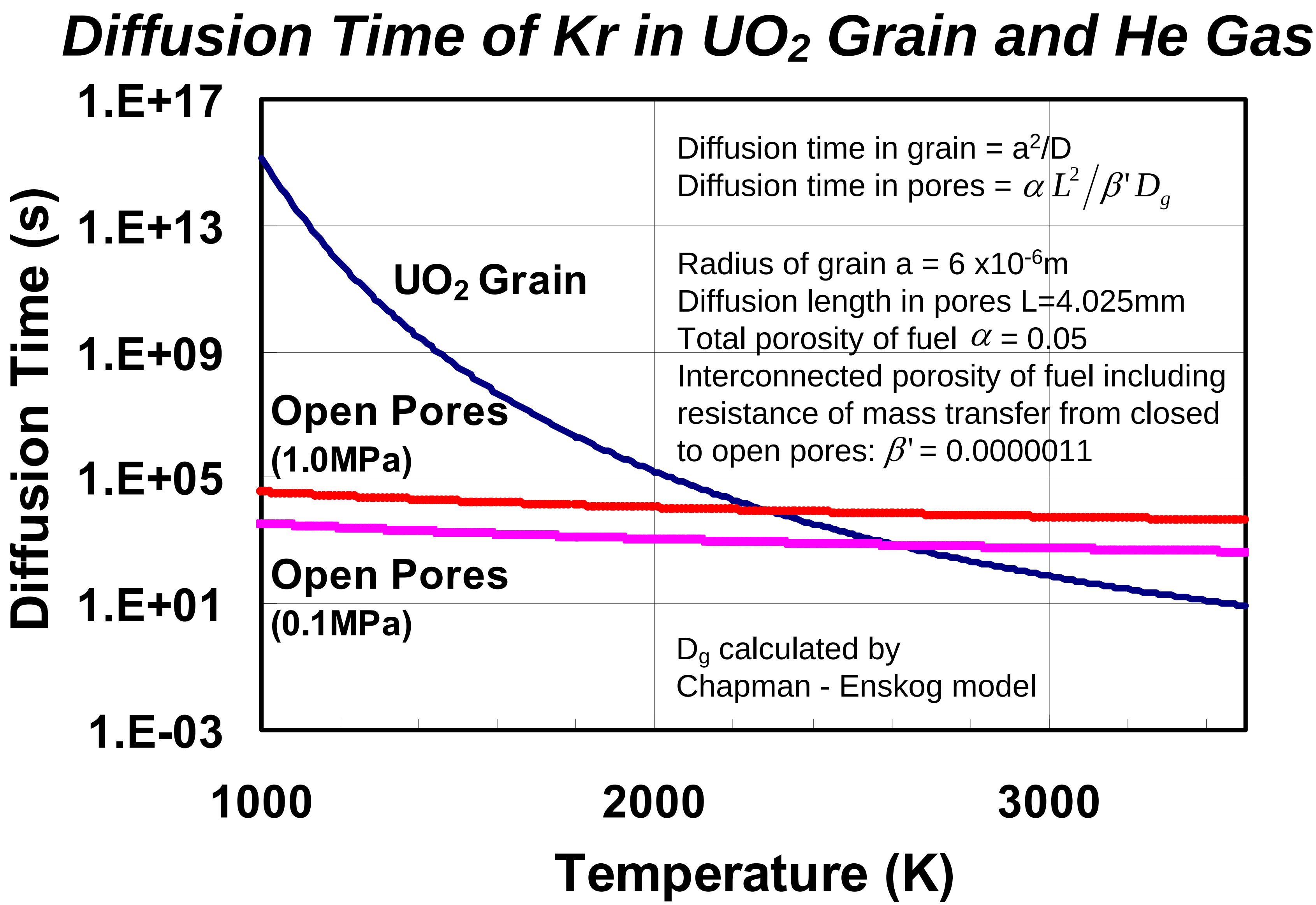
$$\mathbf{k = k_0 \exp\left(-\frac{Q}{RT}\right)}$$

Proposed model with pressure effect

$$\mathbf{k = k_0 \sqrt{\frac{P_0}{P}} \exp\left(-\frac{Q}{RT}\right)} \quad (\mathbf{P \geqq 0.1MPa})$$

$\sqrt{P_0}\ /\sqrt{P}$ comes from the pressure dependency of gaseous diffusion flux in pores at pellet surface, $\sqrt{D_2}$ of Eq.(2)

Study on Diffusion Time in Grain and Pores



Phenomena Affecting Release

Possible Phenomena that affect release in Steam Atmosphere

Eutectic reaction between UO_2-Zr cladding

- Melting point of Zr （M.P. of Zr：2123K, M.P. of ZrO_2：2993K）

↓

- Liquefied Zr in contact with UO_2

↓

- Diffusion of oxygen from UO_2 to liquefied Zr

↓

- Reduction of UO_2：Decrease in M.P. of UO_2 （M.P. of UO_2：3123K, M.P. of U：1405K）

↓

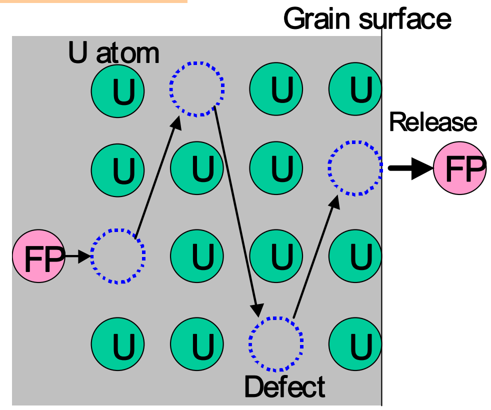
- Liquefaction of UO_2（UO_2+Zr→(U, Zr)O_2）

↓

Increase in radionuclide release

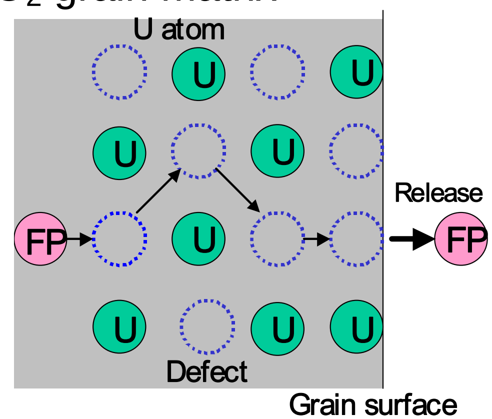
UO_2 oxidation $UO_2+O_x \rightarrow UO_{2+x}$

Inert



FP transport through defect of UO_2 grain matrix

Steam



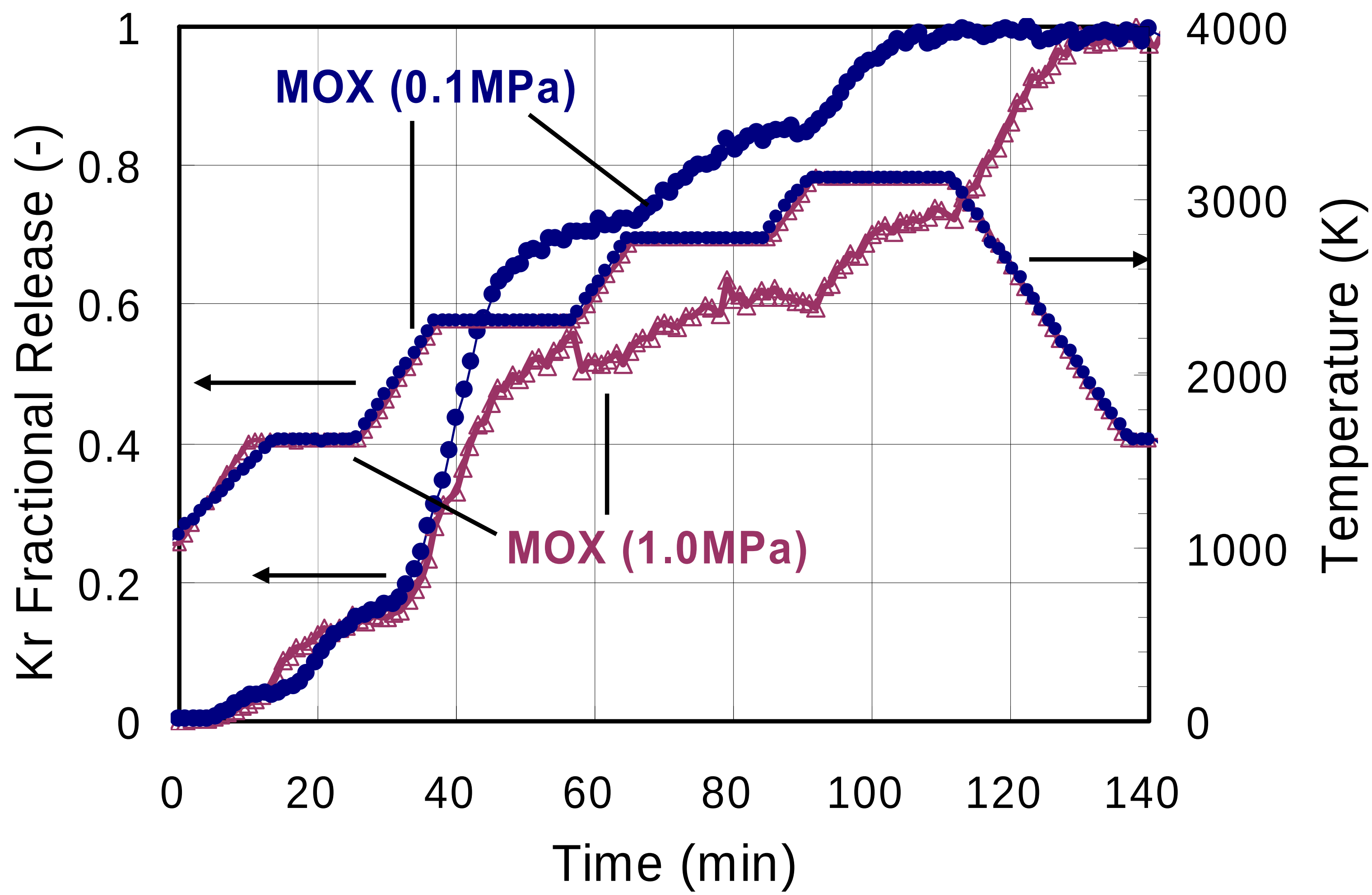
Increase in defect in UO_2 grain matrix due to oxidation

↓

Easier FP transport

Increase in release rate

Effect of Pressure on Kr Release from MOX



Panel Discussions: Part 2

Human and Organizational Aspects of SAM;

their importance vs. technical issues

Oct. 28. 2009

Changwook, HUH
Korea Institute of Nuclear Safety
at
OECD/NEA Workshop on ISAMM-2009

Schloss Böttstein, Switzerland

Contents

Status of SAMG Development in Korea

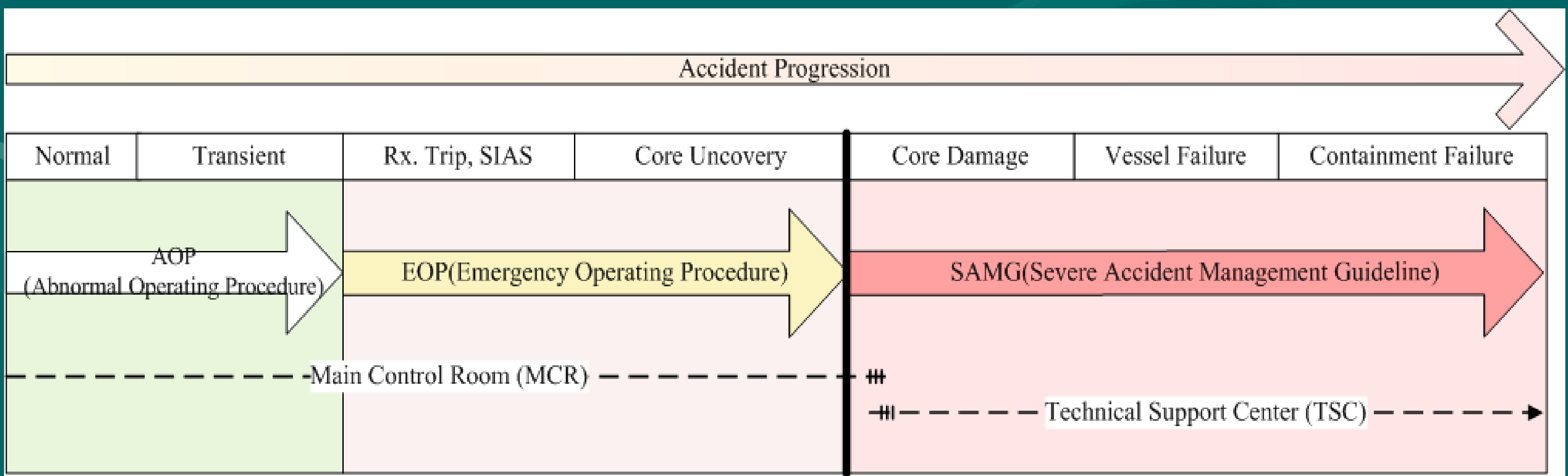
Introduction

- SAMG Development in Korea
  - Policy Statement on Severe Accident of NPPs (2001.8)
    - Require the license holder to take measures to minimize the possibility of severe accident and, if it should occur, to take proper measures to minimize the risk of radiation exposure to the public
    - Major elements of the policy
      - Safety goal
      - PSA
      - SA prevention and mitigation capability
      - SAMP
  - Review and implementation of PWR SAMG was completed by 2008

Introduction

➢ Framework of SAMG

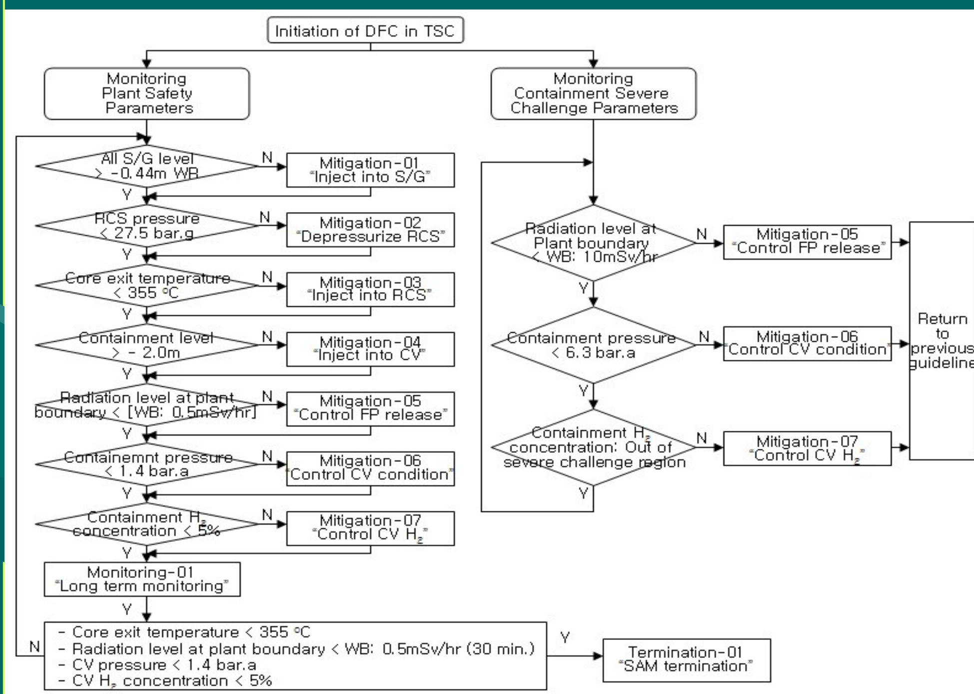
- Designed to fill the gap between EOP and Emergency Plan
- Clear cut between EOP and SAMG with regard to human factors
  - No concurrent usage of EOP and SAMG to prevent the conflicts
    - – Effects of using spray both for pressure control and for hydrogen control
  - Once entering into the SAMG, it's not allowed to return to the EOP
  - Provides an opportunity to clearly focus on the goals associated with each guidance ; preventive vs. mitigative



Introduction

➢ SAMG Structure

- Developed referencing the WOG SAMG
- Basic Philosophy
  - •Be complemented with EOPs
  - •Maximize the use of existing equipments
- Diagnostic Flowchart and 7 Mitigative guideline



SAG	SAG	Mitigation actions
M-1	Inject into S/G	• Inject into S/G • Depressurize S/G
M-2	Depressurize RCS	• Depressurize RCS • Depressurize S/G
M-3	Inject into RCS	• Inject into RCS
M-4	Inject into containment (CV)	• Inject into CV
M-5	Control fission products releases	• Depressurize CV • Dump steam to condenser • Vent Aux. building
M-6	Control containment conditions	• Remove heat form CV • Depressurize or vent CV
M-7	Control containment hydrogen	• Recombine H_2 • Burn H_2 intentionally • Stop active heat sinks in CV

Review of Current SAMG

Review of Current SAMG

◆ Technical Aspects of Current SAMG

➢ M-01 (Injection into SG) and 03 (Injection into RCS)

- are similar to EOP actions and are introduced into SAMG to bridge the clear cutting of EOP and SAMG

➢ M-06 (Control Containment), and M-07 (Control Hydrogen)

- Spray and FCL are main components relied on in these strategies and also mainly used in EOP
  - For H_2 control, deliberate ignition and steam-inert are models not proven.
  - For a reliable control of H_2, ESFs (PAR, ignitor) are in need.

➢ M-02 (RCS Depressurization), and M-04 (Injection into Containment)

- M-02 for severe accident needs to be different from that in EOP.
- In implementing M-04,
  - Objective of IVR is not possible for most of operating plants in Korea
  - Pre-flooding and Top-flooding strategy give quite different results

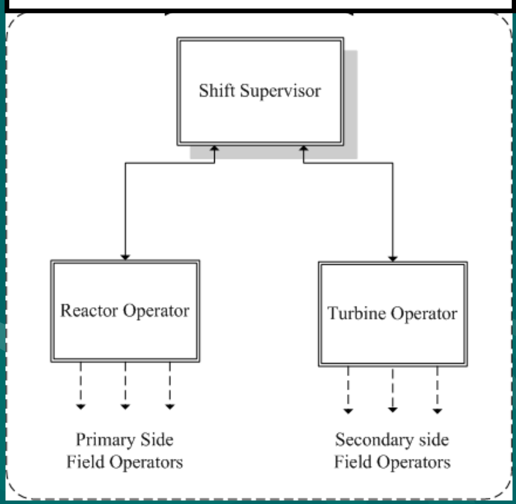
Review of Current SAMG

- Remarks on Technical Aspects of Current SAMG
- Recent results of SA research need to be applied
  - Need to reflect more insights and knowledge learned from recent severe accident researches since late 1980s (EPRI TBR)
    - Ex-vessel debris coolability is one of the main un-resolved issues, but recent OECD/MCCI results may make plant application possible
- Focus seems to be lost in the current SAMG while cycling the diagnostic flow chart to check the restoration of once failed component

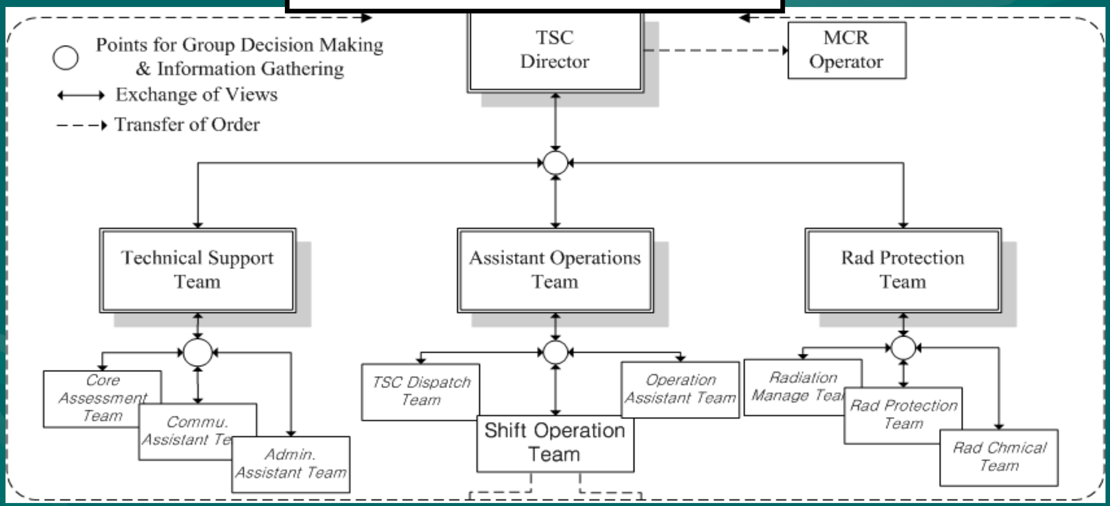
Review of Current SAMG

- Organizational Aspects of current SAMG
  - Responsibility of plant control shifts from MCR to TSC
  - TSC with several teams decides SAM strategy
    - MCR : by shift supervisor based on prescribed procedures (EOP)
    - TSC : via group discussion using guidelines (SAMG)

Organization of MCR



Organization of TSC



- Effectiveness of TSC Decision Making has not been evaluated in real and risky severe accident conditions.

Decision Making Process in TSC

Decision Making Process in TSC

- Group Decision Making
  - Common belief ;
  - Group is, when compared with individual
    - •more knowledge, more ideas, better memories
    - •Evaluate alternatives better, catch errors
    - •more rational and more moderate decision making

➔ Group decision in TSC is generally believed to be more effective for an optimal decision making during a severe accident condition with high risk and uncertainty,

Decision Making Process in TSC

- Are Group Decisions Always Good?
  - Of course not.
  - In the early 1960s researches questioned this assumption, especially under risky and uncertain situations

    "During the Civil War, the councils of war were abandoned because the group process yielded excessively cautious decisions… and fighting wars requires taking risks"

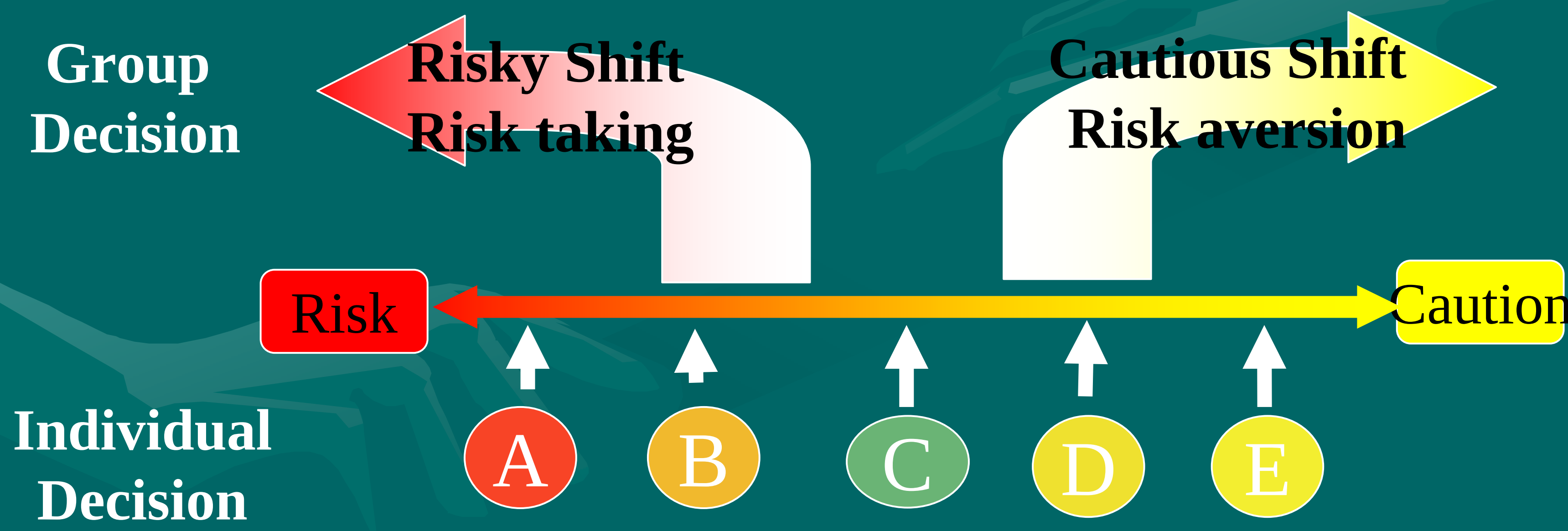
•It is mentioned by General James M Gavin, president of ADI (the world's first management consulting service firm)

Decision Making Process in TSC

- This is the **RISKY SHIFT** Phenomena
  - Stoner (1961) observed that individuals, when placed in group, take more risks than they would otherwise
    - Diffusion of Responsibility : Don't worry about possible negative consequences because group can diffuse responsibility for the decision.
    - Familiarization : Anxiety about possible consequences of a risky decision decreases as people become familiar with choice dilemma.
    - Leadership Theories : Focus is on how specific members influence groups (power, conformity, deviance)
    - Value Theory : Individuals take more chances in the presence of others than they would take alone.
  - Some researchers found a cautious shift.
- Groups make either riskier or more cautious decisions than would have been made by individual members acting alone

Decision Making Process in TSC

- The direction of shift depends on the members' original viewpoints.
  - Risk taking attitude ➔ Risky shift
  - Risk aversion attitude ➔ Cautious shift



An Illustrative Simulations of Group Decision Making

Illustrative Simulation

- Plant Conditions
  - High Pressure Sequence due to Station Blackout
    - •No human error
    - •Only passive or manual activation available
    - •All safety system could be activated when power is recovered
  - SGs were dried out and CET exceeds 650 ℃
  - Plant staffs are trying to restore the power
  - RCS pressure is still around 17 MPa
    - •Reference Plant: UCN 1&2
- M-02 (RCS depressurization Strategy) should be considered according to SAMG

Illustrative Simulation

- RCS Depressurization strategy mainly aims
  - To establish core cooling with safety injection (RPV integrity)
  - To prevent HPME/DCH (Containment integrity)
- Recommended Actions
  - Depressurize RCS below a set point of 2.75 MPa using all relief pathway including PZR safety and relief valves
  - UCN 1 plant has 3 PZR safety and relief valves
- TSC concerning points
  - Recovery Possibility of failed equipments (AC power)
  - Available relief pathway
  - Positive and negative impacts

Illustrative Simulation

- Decision Process in TSC
  - Identify the available means for depressurizing the RCS
    - open PZR Relief Valves manually or using battery power if available
  - Identify the impacts of depressurizing the RCS

Positive Impacts	Negative Impacts
▪ Initiation of Low Pressure Injection ▪ Prevention of High Pressure Melt Ejection ▪ Prevention of Creep Rupture of RCS Piping	▪ Steam Explosions in RPV ▪ Loss of RCS inventory due to PZR PORV Use ▪ Containment Overpressure ▪ Containment Challenge from a Hydrogen Burn ▪ Fission Product Release from SGs

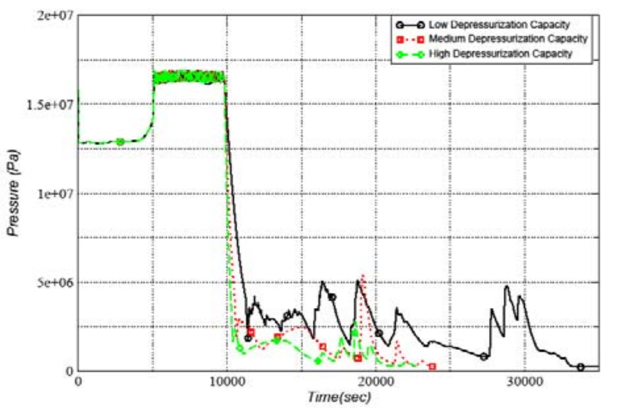
- TSC should choose which action is most appropriate.
  - If PZR valves only are available, TSC should decide to open

  ➔ When? How many valves?

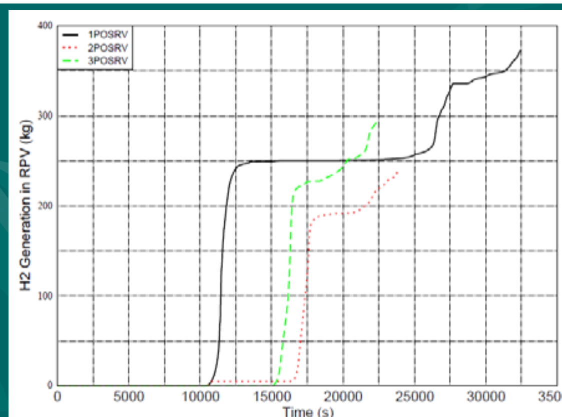
Illustrative Simulation

- Identify the Effects of Depressurizing the RCS

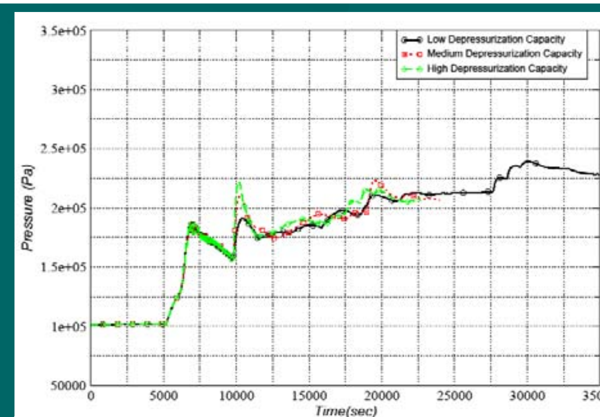
		Case 1 Open 1 Valves	Case 2 Open all Valves
RCS Depressurization Accumulator Injection	No. Dep.	slow	fast
Rapid Hydrogen Generation	-	earlier	later
Containment Pressure	-	A little lower	a little higher
RCS Pressure	-	Higher but below set point	Lower
Initial Vessel Breach	4.2 hr	9.1 hr	6.2 hr
Pressure at RPV Failure	16.7 MPa	0.75 MPa	0.39 MPa



RCS Pressure



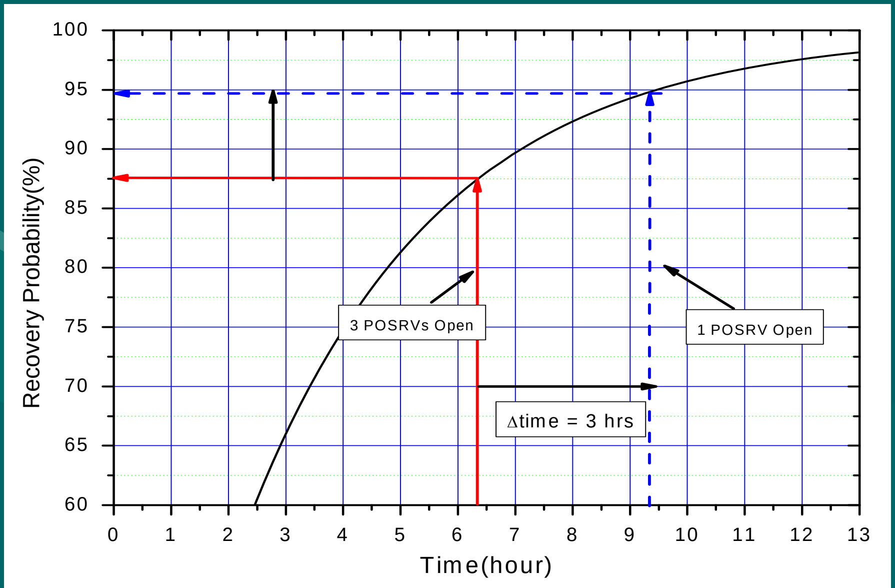
H2 Generation in Core



Containment Pressure

Illustrative Simulation

- Decide which choice is more effective
  - Two cases: RCS pressure at the time of RPV failure is below 1MPa
    - HPME/DCH might be not an issue
  - If you open one valve, RPV failure can be delayed to 9 hours w/ only SIT.
    - AC Recovery Prob. will increase,
    - But the potential risk is getting increased before AC power recovers.



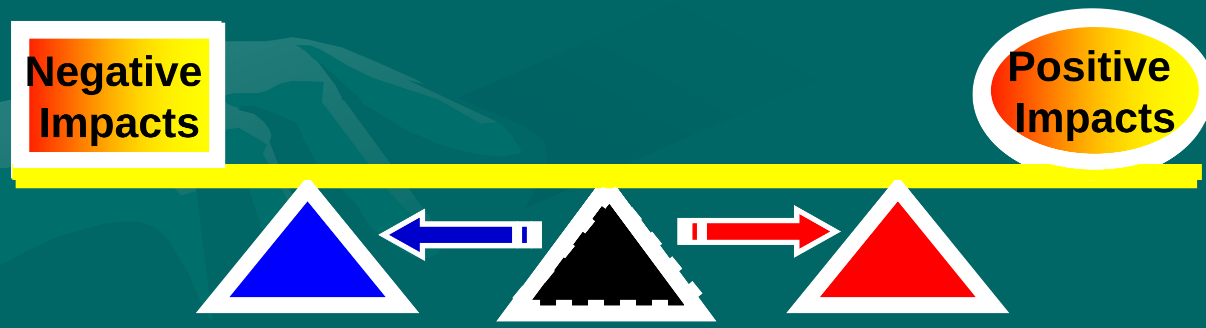
- ✓ The core melting progression can be terminated without RPV failure?
  - It can be possible if AC power is recovered in time.
- ✓ Then, can you take risks causing by opening 1 valve until AC power recover?

Illustrative Simulation

- Possible Results of Decision Making
  - Choice A (Risk aversion attitude)
    - Early Containment Failure (HPME/DCH) should be prevented first of all
    - No wait for AC power recovery
    - Decision may be made with focusing on fast depressurization
  - Choice B (Risk taking attitude)
    - Case 1(1 valve open) may delay RPV failure, even though the RCS pressure is decreased slower than in case 2
    - If AC power is recovered in time, RPV integrity can be maintained even though the risk is getting increased before AC power recovery
    - Decision may be made with focusing on RPV integrity
- Then, what do you think is an optimal choice?
  - Consequence only can answer after the accident is terminated.

Decision Making in TSC

- In current SAMG, TCS decision is supposed to be optimal balancing positive and negative impacts of various actions.
- What is the problem caused by the polarization of group decision (risky shift or cautious shift)?
  - The results of decision making could not be ensured to be consistent
- Decision making should be a lever, but if a fulcrum is moving to one or the other side, what will happen?



Summary

Summary

- Technical Aspects of Current SAMG
  - SAMG is need to be revisited reflecting recent results of severe accident research
    - Some strategies are not feasible for operating plants, some issues are near to be resolved
  - SAM strategies need to focus more on mitigating severe accident specific phenomena
    - M-01, M-03 and M-05 can be implemented in MCR, if appropriate procedure is available, even though core damage progresses
    - Ex-vessel debris coolability, source term management, containment venting seem to be main phenomena to mitigate the effects

Summary

- Organizational Aspects of Current SAMG
  - Current framework of SAMG needs to be revisited
    - Clear cut of EOP and SAMG, accepting H2 control should be done by ESFs
    - Shift of plant control from MCR to TSC, considering the effect of risky (cautious) shift phenomena
- Effectiveness of Decision Making in TSC
  - Polarization of group decision under risky situation is a proven human behavior
    - Optimal accident management by TSC is somewhat doubtful
  - Procedurization of SAM needs to be considered seriously
    - Accumulation of analyses experiences and research results make it possible, in a certain sense

Thank you very much
for your attention!

-- Effectiveness of current SAMG implementation --

How can consequence analyses be used to improve the effectiveness of SAM?

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OECD/NEA Workshop on
Implementation of Severe Accident
Management Measures
(ISAMM 2009)

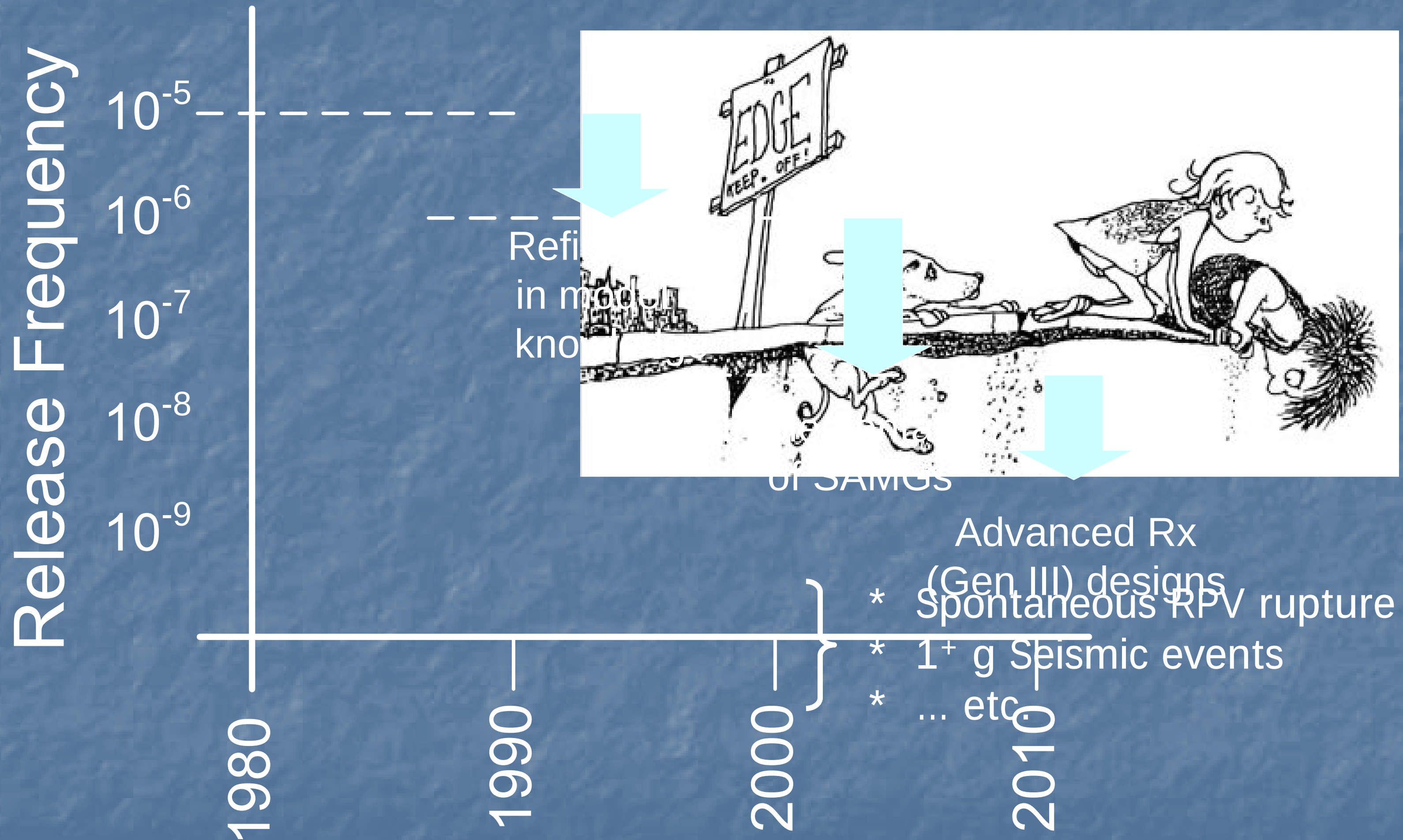
Premise for Discussion:

- Current SAM measures were developed with three principles in mind:
  - Terminate damage to reactor fuel,
  - Maintain containment integrity for as long as possible,
  - Minimize the magnitude of fission product released to the environment.
- SAM measures developed with an “Inside-Out” perspective on risk management
  - PSA was the primary tool for identifying the plant conditions to be addressed,
  - Goal: transform a hazardous situation into stable condition that can be maintained in the long term
  - “Success” measured in terms <u>internal</u> to plant

Limitations of ‘Inside-Out’

- Metrics for success are in-direct
  - Delay in time of containment failure
  - Reduction in activity released
- Scenarios not always a realistic representation of plant behavior
  - Level 1 PSA forbids credit for ‘benevolent failures’
  - Component/system ‘failures’ are binary (success/fail) with no intermediate conditions (degraded operation)
- Advancements in modeling tools and implementation of SAM measures have driven nominal estimates of risk toward values that were once considered ‘remote and speculative’

Limitations of 'Inside-Out'

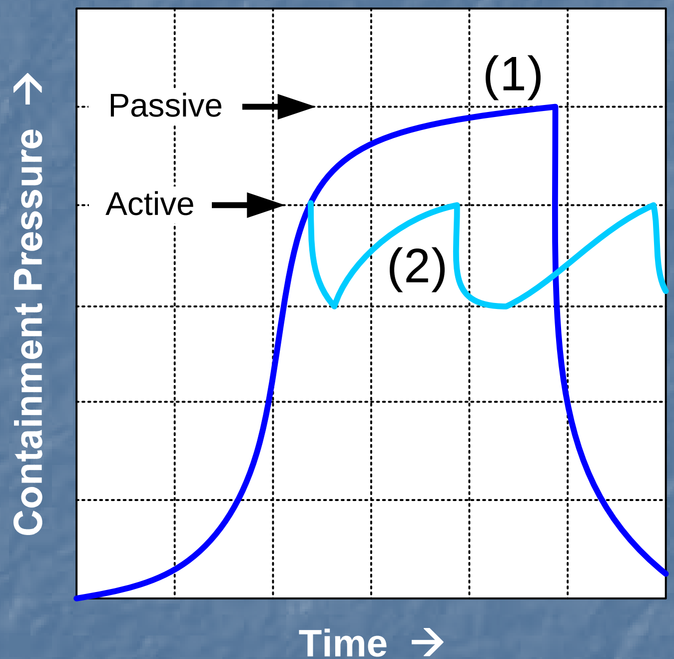


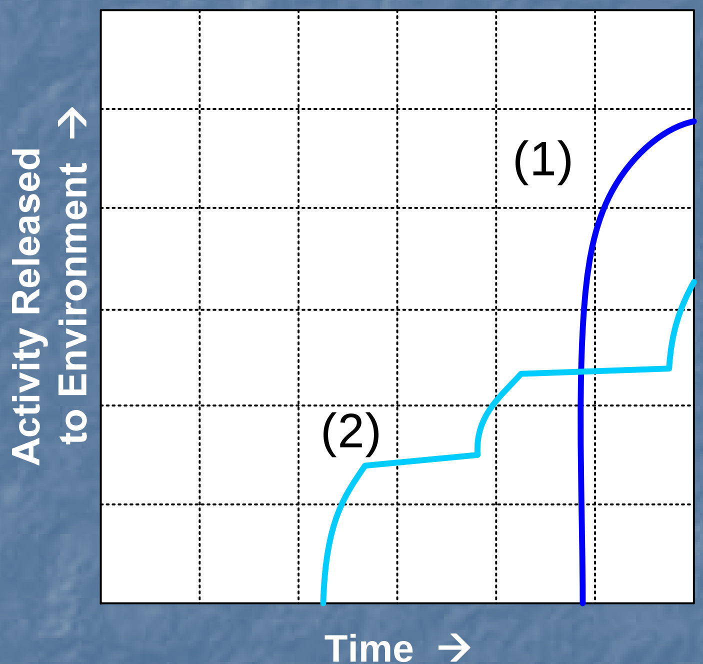
The challenge: can we reverse our perspective and look ‘Outside-In’?

- Why not evaluate SAM effectiveness where the outcome is ultimately measured?
  - Direct calculation of offsite risk measure
    - Dose, land contamination, etc.
    - Can we abandon risk (QHO) ‘surrogates’ to account for local environmental factors and eliminate effects of scale
  - Optimize current strategies for minimum offsite ‘consequence’
    - Consider factors beyond time/magnitude of release
    - Links to offsite consequence analysis tools

One simple example: Containment venting strategies optimized for radiological consequences

Consider two strategies:
(1) Passive actuation of rupture disk
(2) Manual vent with re-isolation





Is the value of reducing/controlling a release greater or less than the value of a delay in the start of a release?

The challenge: can we reverse our perspective and look ‘Outside-In’?

- Has PSA been used/abused to a point where we’re not asking the right questions?
  - Are the effects of plant behavior <u>not</u> captured in PSA important?
  - Is residual risk in ‘remote and speculative’ events adequately covered by current SAMGs?
  - Can the characteristics of a radiological release be used to understand what’s going on inside the plant?

Discussion