Implementation of Severe Accident Management Measures, ISAMM 2009

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

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

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The mission of the NEA is:

– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as

– to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

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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.
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)
Implementation of Severe Accident Management Measures, ISAMM 2009

Workshop Proceedings
Schloss Böttstein, Switzerland
October 26 - 28, 2009

in collaboration with
KKB, KKL, KKM, KKG and PSI

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M. Jermann, Vice Director, PSI
U. Weidmann, Director, NPP-Beznau
S. Guentay, Vice Chair, CSNI/WGAMA

**Session 1: Current Status & Insights of SAM, Part 1**

*Session chair/co-chair: N. Suh/A. Torri*

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- **Technical Challenges in Applying SAMG Methodology to Operating CANDU Plants**
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- **Accident Management in German NPPs: Status of Implementation and The Associated Role of PSA Level 2**
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Chairs/Co-Chairs: J. Primet / V. Dang

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  S. Guentay (PSI, Switzerland),  
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  D. Helton (US-NRC, USA)  
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  N. Dessars (Westinghouse Electric Belgium S.A, USA)

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  C. G. Tinkler (US.NRC, USA),
  K.C. Wagne (SNL, USA),

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

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- **Effectiveness of current SAMG implementation - How can consequence analyses be used to improve the effectiveness of SAM?**,
  Mark Leonard (Dycoda, US)
Opening and Introduction of the Workshop
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
# Key Figures 2009

<table>
<thead>
<tr>
<th>Category</th>
<th>Value</th>
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</thead>
<tbody>
<tr>
<td>PSI funds (global budget)</td>
<td>244 MCHF</td>
</tr>
<tr>
<td>External funding</td>
<td>55 MCHF</td>
</tr>
<tr>
<td>Staff / FTE</td>
<td>~ 1350 PJ</td>
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<tr>
<td>Of which externally financed</td>
<td>~ 320 PJ</td>
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<tr>
<td>Doctoral students</td>
<td>~ 300</td>
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<td>Apprentices</td>
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<tr>
<td>External users</td>
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<tr>
<td>Number of scientific publications</td>
<td>~ 900</td>
</tr>
<tr>
<td>PSI-employees with teaching duties at ETH and universities</td>
<td>~ 75</td>
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</tbody>
</table>
Ressource Distribution 2009

c.a. 300 MCHF (incl. external funding)

- Particle Physics: 10%
- Nuclear Energy and Safety: 17%
- Solid State Physics and Materials Sciences: 40%
- General Energy: 13%
- Life Sciences: 20%
Non-nuclear Energy Research

- Use of Biomass
- SNG: Synthetic Natural Gas
- Combustion
- High Temp Solarchemistry
- Hydrogen
  - Electrochemistry
  - Fuel cells, Batteries
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!!
The MEG experiment searching for the decay $\mu^+ \rightarrow e^+ \gamma$ at the $10^{-13}$ level! Assembly now complete.

$\mu^+$ stop and decay inside 175 $\mu$m target

$e^+$ are measured in drift chambers built by PSI’s Detector Group

EDM Experiment
SLS - Layout
Structure of Proteins measured with Synchrotron Light

Diffraction pattern

Membran

Zytoplasma

Rhesus Protein

Amonium transport

JMD1 2009
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

<table>
<thead>
<tr>
<th></th>
<th>SLS</th>
<th>SINQ</th>
<th>S$_\mu$S</th>
<th>LTP</th>
<th>PSI total</th>
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<tr>
<td>Beamlines</td>
<td>14</td>
<td>13</td>
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<td>Instrument Days</td>
<td>1657</td>
<td>1895</td>
<td>655</td>
<td>660</td>
<td>4867</td>
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<td>Experiments</td>
<td>1036</td>
<td>446</td>
<td>168</td>
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<tr>
<td>User Visits</td>
<td>2912</td>
<td>677</td>
<td>185</td>
<td>180</td>
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<tr>
<td>Individual Users</td>
<td>1616</td>
<td>447</td>
<td>151</td>
<td>120</td>
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<td>New Proposals</td>
<td>656</td>
<td>275</td>
<td>156</td>
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<td>1088</td>
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</tbody>
</table>
Geographic distribution of SLS users 2008,

- EU: 50%
- Switzerland except PSI: 20%
- PSI: 21%
- Others: 9%

Geographic distribution of SμS users

- EU: 37
- Switzerland except PSI: 30%
- PSI: 28%
- Others: 27%

Geographic distribution of SINQ users 2008

- EU: 40%
- Switzerland except PSI: 30%
- PSI: 23%
- Others: 10%
Comparison of Characteristics of Photons und Protons for Radiation Therapy

![Graph comparing photon and proton characteristics](image)
OPTIS: Proton Therapy for Eye Melanomas (since 1984)

Cooperation with Univ. Lausanne

>5100 patients treated
>98 % tumor control

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

> 500 patients treated with deep-seated tumors

> 40 % of patients are below 40 years old

Design E. Pedroni, PSI
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
Strategic Initiative: SwissFEL (XFEL)

Low-Emittance Gun

High-Voltage Pulser
SwissFEL performance

• Gain in brilliance

$10^4$ (Mittel) - $10^{10}$ (Peak)
comparable to EU, US, JP X-FEL

• Investments $\approx$ 250-280 MCHF
SwissFEL: what for?

How fast and small can magnetic writing be?

Determination of protein structures and interactions

Catalytic reactions (time)

http://fel.web.psi.ch
Thank you for the attention
Welcoming Speech of Dr. U. Weidmann, Director, NPP-Beznau is not available
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.

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.

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:
  - Different practices are displayed
  - 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
- 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+W Risk, 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.
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
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

Safety Fundamentals

Safety Requirements

Safety Guides

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
1

SR
15

SG
104
SAFETY STANDARDS RELATING TO NPPs

Safety Fundamentals (SF-1)

- Safety Requirements
  - Safety of NPPs: Design
    - No. NS-R-1
  - Safety of NPPs: Operation
    - No. NS-R-2
- Safety Guide
  - Fire Safety in the Operation of NPPs
    - No. NS-G-2.1
  - Operational Limits and Conditions and Operating Procedures for NPPs
    - No. NS-G-2.2
  - Periodic Safety Review of NPPs
    - No. NS-G-2.10
- 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)
  - Development and Application of Level-2 PSA for NPPs
    - (DS-393)
SAFETY STANDARDS DEVELOPMENT PROCESS

1. Outline and work plan prepared by Secretariat; Review by the Committees and the Commission

2. Secretariat and consultants: drafting or revising of safety standard

Draft

3. Review by Safety Standards Committee(s)

Draft

Final draft

5. Endorsement by COMMISSION ON SAFETY STANDARDS (CSS)

4. Member States

Approved for publication

6. Publication Committee

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

Nuclear Safety Standards Committee (NUSSC) - reviewing Safety Standards for nuclear facilities and activities

DS393

DS394
SAFETY GUIDE
Severe Accident Management Programmes for Nuclear Power Plants
(NS-G-2.15)
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
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
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
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
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
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
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 SAMG’s 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”
SAFETY SERVICES

International Atomic Energy Agency

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.

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.

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.

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 occurred 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 occurred at research reactors.

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.
IPSART SERVICE

-established in 1988

-conducted in accordance with dedicated guidelines
More than 50 IPSART missions 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

QA, internal review

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
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)

- 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
IAEA has prepared ‘User Manual’ for RAMP service: “GUIDELINES FOR THE REVIEW OF ACCIDENT 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 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:
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)
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
## Table 1: SAMG Entry Conditions

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<thead>
<tr>
<th>Condition</th>
<th>Parameter</th>
<th>Typical Instrumentation</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Loss of core cooling</td>
<td>No subcooling margin in inlet headers for &gt;15 minutes</td>
<td>Heat transport system (HTS) temperature and pressure instrumentation</td>
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<td><strong>AND</strong> either</td>
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<tr>
<td>2. Loss of moderator cooling to fuel channels</td>
<td>Moderator level below top of highest channels</td>
<td>Moderator level instrumentation</td>
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<td>or</td>
<td></td>
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<tr>
<td>3. Major release of fission products from the fuel</td>
<td>Plant radiation levels &gt; setpoints</td>
<td>Ex-containment gamma measurements</td>
</tr>
</tbody>
</table>
Onset of Severe Accident
• 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)
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
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 differentials 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.
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
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

— Consulting mandate to the RSK and resulting recommendations (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

Guide Probabilistic Safety Analysis

Guide Security Analysis

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

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

Data for PSA of NPPs (Daten zur Quantifizierung von Ereignis- ablaufdiagrammen und Fehlerbäumen)
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

<table>
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<tr>
<th>Measure</th>
<th>KWBA</th>
<th>KWBB</th>
<th>GKN1</th>
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<th>KKG</th>
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<th>KKP2</th>
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<td>Emergency power supply from neighbouring plant</td>
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<td>Restoration of off-site power supply</td>
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<td>Additional off-site power supply (underground cable)</td>
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<td>Sampling system in the containment</td>
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</tr>
<tr>
<td>✓ design</td>
<td>•</td>
<td>•</td>
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<td>•</td>
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<td>•</td>
</tr>
</tbody>
</table>

- • realised through backfitting measures
- g license granted
- □ not applicable

| Verantwortung für Mensch und Umwelt | □ □ □ □ □ □ □ |

Bundesamt für Strahlenschutz
# Implementation of AM-measures in BWRs

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

- √: 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 $\rightarrow$ 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 H₂ 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*

* 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

<table>
<thead>
<tr>
<th>Release category</th>
<th>Containment (C) failure mode</th>
</tr>
</thead>
<tbody>
<tr>
<td>RC-A</td>
<td>LOCA outside C</td>
</tr>
<tr>
<td>RC-B</td>
<td>Uncovered SGTR</td>
</tr>
<tr>
<td>RC-C</td>
<td>Early C rupture</td>
</tr>
<tr>
<td>RC-D</td>
<td>C isolation failure</td>
</tr>
<tr>
<td>RC-E</td>
<td>Covered SGTR</td>
</tr>
<tr>
<td>RC-F</td>
<td>Sump line failure</td>
</tr>
<tr>
<td>RC-G</td>
<td>Late C rupture</td>
</tr>
<tr>
<td>RC-H</td>
<td>Basemat melt-through</td>
</tr>
<tr>
<td>RC-I</td>
<td>Unfiltered C venting</td>
</tr>
<tr>
<td>RC-J</td>
<td>FCV</td>
</tr>
<tr>
<td>RC-K</td>
<td>No failure</td>
</tr>
</tbody>
</table>
PSA results on SAM efficiency: GKN 1*

* 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*

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

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

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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)
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1. Background and history
2. Accident management measures implemented to the operating NPPs
3. Accident management measures implemented to the recent NPPs
4. Conclusions
# 1. Background and history

<table>
<thead>
<tr>
<th>Date</th>
<th>Major events for AM</th>
</tr>
</thead>
<tbody>
<tr>
<td>May, 1992</td>
<td>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.</td>
</tr>
<tr>
<td>July, 1992</td>
<td>MITI encouraged utilities to establish AM implementation plans, using benefit of insights obtained from PSA.</td>
</tr>
<tr>
<td>March, 1994</td>
<td>The utilities submitted AM implementation plans to MITI. MITI reviewed utilities plans.</td>
</tr>
<tr>
<td>October, 1994</td>
<td>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.</td>
</tr>
</tbody>
</table>
1. Background and history (cont’d)

<table>
<thead>
<tr>
<th>Date</th>
<th>Major events for AM</th>
</tr>
</thead>
<tbody>
<tr>
<td>February, 2002</td>
<td>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.)</td>
</tr>
<tr>
<td></td>
<td>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.</td>
</tr>
<tr>
<td>March, 2004</td>
<td>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.</td>
</tr>
<tr>
<td>Up to now</td>
<td>Besides fifty-two operating NPPs, AM have been studied and implemented to four newly constructed NPPs.</td>
</tr>
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</table>
2. Accident management measures implemented to the operating NPPs
## AM measures for BWR

<table>
<thead>
<tr>
<th>Safety function</th>
<th>Purpose</th>
<th>Prevention of core damage</th>
<th>Mitigation of core damage</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor shutdown</td>
<td>Alternate reactivity control</td>
<td>● ARI (except ABWR)</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>● RPT (except ABWR)</td>
<td></td>
</tr>
<tr>
<td>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></td>
<td>Alternate coolant injection</td>
<td>● MUWC</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>● Fire extinguishing system or filtrate water system</td>
<td></td>
</tr>
<tr>
<td>Heat removal from CV</td>
<td>Hard vent system</td>
<td>● Hard vent system</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Alternate cooling</td>
<td></td>
<td>● Alternate cooling by dry-well cooler or CUW</td>
</tr>
<tr>
<td>Recovery of RHR</td>
<td></td>
<td>● Recovery of RHR</td>
<td></td>
</tr>
<tr>
<td>Supporting function</td>
<td>Electric power supply</td>
<td>● Electric power supply from adjacent unit</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>● Electric power supply from HPCS-DG (Single-unit site)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>● Installation of dedicated EDG</td>
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<tr>
<td>Recovery of EDG</td>
<td></td>
<td>● Recovery of EDG</td>
<td></td>
</tr>
</tbody>
</table>

AM measures for alternate coolant injection (BWR)
AM measures for CV heat removal (BWR)

- Stack
- Reactor building ventilation system
- Dry-well cooler
- Containment vessel
- Dry-well
- PLR pumps
- Containment vessel
- Clean up water system

Alternate heat removal:
- Heat removal by hard vent
- Heat removal by SGTS
- Fan

To feedwater system

Residual heat removal system (pumps, Hx, seawater system, etc.)

From reactor, containment

To reactor, containment
## AM measures for PWR

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

<table>
<thead>
<tr>
<th>Alternative</th>
<th>Recirculation Line</th>
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<tbody>
<tr>
<td>CV spray header</td>
<td>M</td>
</tr>
<tr>
<td>RWSP</td>
<td>M</td>
</tr>
<tr>
<td>CV spray Hx</td>
<td>M</td>
</tr>
<tr>
<td>CV spray pumps</td>
<td>M</td>
</tr>
<tr>
<td>Tie-line between LPI and CSI</td>
<td>M</td>
</tr>
<tr>
<td>RHR Hx</td>
<td>M</td>
</tr>
<tr>
<td>Low press. injection pumps</td>
<td>M</td>
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<tr>
<td>Alternative Recirculation Pump</td>
<td>M</td>
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Alternative Recirculation Line
Power supply from the adjacent unit
## Reactor types and safety systems (BWR)

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

# Reactor types and safety systems (PWR)

<table>
<thead>
<tr>
<th>Safety systems</th>
<th>type A</th>
<th>type B</th>
<th>type C</th>
<th>type D</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Type</td>
<td>Two-loop</td>
<td>Three-loop</td>
<td>Four-loop with ice condenser</td>
<td>Four-loop</td>
</tr>
<tr>
<td>ECCS</td>
<td>HPI (2), Boosted by LPI during recirculation</td>
<td>CHSI (3), Boosted by LPI during recirculation</td>
<td>CHSI (3), HPI (2), Boosted by LPI during recirculation</td>
<td>HPI (2)</td>
</tr>
<tr>
<td>LPI</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>Acc.</td>
<td>2</td>
<td>3</td>
<td>4</td>
<td>4</td>
</tr>
<tr>
<td>CV spray</td>
<td>2</td>
<td>2</td>
<td>2</td>
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</tr>
</tbody>
</table>

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)

Type A, BWR2, 3
Type B, BWR4
Type C, BWR5
Type D, ABWR

CDF or Reduction ratio

CDF (before AM implementation)  CDF (after AM implementation)  Reduction ratio
CFF results before and after AM implementation (BWR)

- Type A, BWR2, 3
- Type B, BWR4
- Type C, BWR5
- Type D, ABWR
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)

<table>
<thead>
<tr>
<th>CDF (before AM implementation)</th>
<th>CDF (after AM implementation)</th>
<th>Reduction ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type A, 2-loop</td>
<td>Type B, 3-loop</td>
<td>Type C, 4-loop with ice condenser CV</td>
</tr>
<tr>
<td>Type A, 4-loop</td>
<td></td>
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</table>

HFPP results: 0.01 0.10 1.00 10.00
CFF results before and after AM implementation (PWR)

- Type A, 2-loop
- Type B, 3-loop
- Type C, 4-loop with ice condenser CV
- Type D, 4-loop

Legend:
- □ CFF (before AM implementation)
- □ CFF (after AM implementation)
- ■ Reduction ratio
• 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)

<table>
<thead>
<tr>
<th>Case</th>
<th>Conditions for analysis</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td><strong>Base case</strong>&lt;br&gt;Basic design&lt;br&gt;(without automatic CCWS train separation, alternative recirculation)</td>
</tr>
<tr>
<td>2</td>
<td><strong>Basic design with AMs by operation manuals</strong>&lt;br&gt;Basic design&lt;br&gt;(with automatic CCWS train separation, alternative recirculation)&lt;br&gt;AMs by operation manuals (no hardware modifications)&lt;br&gt;  - Use of turbine bypass system&lt;br&gt;  - Cooldown and recirculation&lt;br&gt;  - Forced RCS depressurization</td>
</tr>
<tr>
<td>3</td>
<td><strong>All AMs implemented</strong>&lt;br&gt;with AM measures&lt;br&gt;  - Natural convection cooling in CV&lt;br&gt;  - Coolant injection to CV&lt;br&gt;  - Electric power supply from adjacent unit</td>
</tr>
</tbody>
</table>
Component cooling water system
automatic isolation

CCWS surge tank

Header A
Header C
Header B

CCWS pumps

Sea water system

Header A (safety systems)
Header C (non-safety systems)
Header B (safety systems)
• 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.
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>
<thead>
<tr>
<th>Table 1 – Some features of the French PWRs in operation</th>
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<tbody>
<tr>
<td></td>
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<tr>
<td>Loops</td>
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<tr>
<td>Safety injection</td>
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<tr>
<td>Accumulators</td>
</tr>
<tr>
<td>Specific procedures for additional water injection means</td>
</tr>
<tr>
<td>Containment</td>
</tr>
</tbody>
</table>
1 - French PWRs in operation

- 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
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 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
Perspectives on Severe Accident Mitigation Alternatives for US Plant License Renewal

Tina Ghosh, Robert Palla, Donald Helton*
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 (IPEEEEs)
  – 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
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

<table>
<thead>
<tr>
<th>Cost Factor</th>
<th>Significance</th>
<th>NUREG/BR-0184 Section</th>
<th>Related Parameters</th>
<th>Average (and Ranges) of MAB from Submittals for All Approved License Renewals</th>
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<tbody>
<tr>
<td>APE</td>
<td>Offsite exposure</td>
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<td>Offsite economic</td>
<td>5.7.5</td>
<td>$\Delta$Offsite Economic Cost (from Level 3 PRA) and accident frequency (from Level 2 PRA)</td>
<td>$400K ($10K – $2,700K)</td>
</tr>
<tr>
<td>AOE</td>
<td>Onsite exposure</td>
<td>5.7.3</td>
<td>Immediate occupational dose (33 person-Sv)</td>
<td>$17K ($1K – $130K)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Long term occupational dose (200 person-Sv)</td>
<td></td>
</tr>
<tr>
<td>ACC</td>
<td>Onsite economic</td>
<td>5.7.6.1</td>
<td>Onsite cleanup and decontamination cost ($1.1 \cdot 10^9 single event, present worth)</td>
<td>$870K ($37K – $6,300K)</td>
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<tr>
<td>ARPC</td>
<td>Onsite economic</td>
<td>5.7.6.2</td>
<td>Plant power level</td>
<td></td>
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<tr>
<td>Total</td>
<td></td>
<td></td>
<td></td>
<td>$1,700K ($110K - $8,700K)</td>
</tr>
</tbody>
</table>
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

<table>
<thead>
<tr>
<th></th>
<th>Average</th>
<th>Ranges</th>
</tr>
</thead>
<tbody>
<tr>
<td>CDF (/yr)</td>
<td>$4.0 \times 10^{-5}$</td>
<td>$1.9 \times 10^{-6} – 3.3 \times 10^{-4}$</td>
</tr>
<tr>
<td>Population Dose</td>
<td>$0.15$</td>
<td>$0.006 – 0.69$</td>
</tr>
<tr>
<td>(person-Sv/year)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$/\text{event}$</td>
<td>$2.8 \text{ billion}$</td>
<td>$49 \text{ million} – $12 \text{ billion}$</td>
</tr>
<tr>
<td>$/\text{person-Sv}$</td>
<td>$220,000$</td>
<td>$69,000 – $670,000$</td>
</tr>
<tr>
<td>Total MAB</td>
<td>$1.7 \text{ million}$</td>
<td>$110,000 – $8.7 \text{ million}$</td>
</tr>
</tbody>
</table>
Typical Cost Benefit Threshold
3% Discount, 20 year term

Reduction in Person-Sv [1 Person-Sv = 100 Person-rem] (per year)

Reduction in CDF (per year)

$4 Million
$3 Million
$2 Million
$1 Million
$500K
$100K
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
• 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

Youngho Jin
Korea Atomic Energy Research Institute

OECD/NEA ISAMM, 2009.10.26~28
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 5&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

<table>
<thead>
<tr>
<th>Back ground</th>
<th>Requirements for CP/OL</th>
<th>severe accident policy implementation</th>
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</thead>
<tbody>
<tr>
<td>Post-TMI actions</td>
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<tr>
<td>YEAR</td>
<td>89</td>
<td>90</td>
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<tr>
<td>K # 1</td>
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<tr>
<td>K # 2</td>
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</tr>
<tr>
<td>K # 3,4</td>
<td>9</td>
<td>L1-N</td>
</tr>
<tr>
<td>Y # 1,2</td>
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<tr>
<td>Y # 3,4</td>
<td>4</td>
<td>L2-N</td>
</tr>
<tr>
<td>Y # 5,6</td>
<td></td>
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<td>U #1,2</td>
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<tr>
<td>U #3,4</td>
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<td>L2-N</td>
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<td>U #5,6</td>
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<tr>
<td>W #1</td>
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<tr>
<td>W #2,3,4</td>
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<tr>
<td>11</td>
<td>L2-N</td>
<td>11</td>
</tr>
<tr>
<td>1</td>
<td>L2-N</td>
<td>12</td>
</tr>
<tr>
<td>9</td>
<td>L2-U1/RM</td>
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<td>7</td>
<td>L2-U1/RM</td>
<td>12</td>
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<td>L2-U1/RM</td>
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<td>12</td>
</tr>
<tr>
<td>1</td>
<td>L2-U1/RM</td>
<td>12</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Period</th>
<th>Years</th>
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<tbody>
<tr>
<td></td>
<td>'00</td>
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</tr>
<tr>
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<td>'09</td>
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</table>

<table>
<thead>
<tr>
<th>YGN 5,6</th>
<th>’99.12~’01.06</th>
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<tr>
<td>YGN 3,4 UCN3,4,5,6</td>
<td>’02.04~’02.12</td>
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<tr>
<td>Kori 1</td>
<td>’02.09~’03.12</td>
</tr>
<tr>
<td>Kori 2,3,4 YGN 1,2</td>
<td>’03.01~’04.12</td>
</tr>
<tr>
<td>UCN 1,2</td>
<td>’05.06~’07.05</td>
</tr>
<tr>
<td>WS 1,2,3,4</td>
<td>’08.01 ~ ’09.12</td>
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<tr>
<td>Training Program</td>
<td>’01.07~</td>
</tr>
<tr>
<td>Implementation</td>
<td>’02.04~</td>
</tr>
</tbody>
</table>
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: $5.30 \times 10^{-6}$/ry
- Containment Failure Frequency: $1.66 \times 10^{-6}$/ry
- Containment Bypass (SGTR): $7.99 \times 10^{-7}$/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
Frequency of SGTR-37

- **Evaluation of operator available time**
  - Computer code: MARS
  - RCS cooldown rate: 100 °F/hr
  - Operator available time: about 40 minutes

- **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 minutes
  - HEP: 0.0256 (cf. 0.59)

- **Frequency of SGTR-37**: $2.734 \times 10^{-8}$/ry  (cf. 5.20E-7)
### Result of HEP Analysis

<table>
<thead>
<tr>
<th></th>
<th>Before EOP revision</th>
<th>After EOP revision</th>
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</thead>
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<tr>
<td><strong>Available Time</strong></td>
<td>30 min.</td>
<td>30 min.</td>
</tr>
<tr>
<td></td>
<td>40 min.</td>
<td>40 min.</td>
</tr>
<tr>
<td></td>
<td>50 min.</td>
<td>50 min.</td>
</tr>
<tr>
<td></td>
<td>60 min.</td>
<td>60 min.</td>
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<tr>
<td><strong>MMI</strong></td>
<td>Medium</td>
<td>Medium</td>
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<tr>
<td></td>
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<td></td>
<td>Medium</td>
<td>Medium</td>
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<tr>
<td><strong>Procedure</strong></td>
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<td>Medium</td>
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<tr>
<td></td>
<td>Low</td>
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</tr>
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<td></td>
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<td>Medium</td>
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<tr>
<td></td>
<td>Low</td>
<td>Medium</td>
</tr>
<tr>
<td><strong>Training</strong></td>
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<td>Medium</td>
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<tr>
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<td>Low</td>
<td>Medium</td>
</tr>
<tr>
<td></td>
<td>Low</td>
<td>Medium</td>
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<tr>
<td><strong>HEP</strong></td>
<td>5.90E-1</td>
<td>2.56E-2</td>
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<tr>
<td></td>
<td>3.05E-1</td>
<td>1.42E-2</td>
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<tr>
<td></td>
<td>2.39E-1</td>
<td>1.16E-2</td>
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<tr>
<td></td>
<td>6.80E-2</td>
<td>5.44E-3</td>
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<tr>
<td><strong>Ratio</strong></td>
<td>0.043</td>
<td>0.046</td>
</tr>
<tr>
<td></td>
<td>0.046</td>
<td>0.048</td>
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<tr>
<td></td>
<td>0.08</td>
<td>0.08</td>
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</table>
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 equipment
  - 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

<table>
<thead>
<tr>
<th>Containment Failure Mode</th>
<th>Base Case</th>
<th>Fan Coolers</th>
<th>Spray Recovery</th>
<th>Fan Coolers &amp; Spray Recovery</th>
</tr>
</thead>
<tbody>
<tr>
<td>Intact</td>
<td>3.635E-06 (0.686)*</td>
<td>3.856E-06 (0.728)</td>
<td>4.067E-06 (0.768)</td>
<td>4.091E-06 (0.772)</td>
</tr>
<tr>
<td>Early containment Failure</td>
<td>1.192E-08 (0.002)</td>
<td>1.192E-08 (0.002)</td>
<td>1.192E-08 (0.002)</td>
<td>1.192E-08 (0.002)</td>
</tr>
<tr>
<td>Late containment Failure</td>
<td>5.376E-07 (0.101)</td>
<td>2.992E-07 (0.056)</td>
<td>5.938E-08 (0.011)</td>
<td>3.259E-08 (0.006)</td>
</tr>
<tr>
<td>Basemat Meltthrough</td>
<td>1.286E-07 (0.024)</td>
<td>1.462E-07 (0.028)</td>
<td>1.750E-07 (0.033)</td>
<td>1.775E-07 (0.034)</td>
</tr>
<tr>
<td>Containment Bypass</td>
<td>9.841E-07 (0.186)</td>
<td>9.841E-07 (0.186)</td>
<td>9.841E-07 (0.186)</td>
<td>9.841E-07 (0.186)</td>
</tr>
<tr>
<td><strong>Total Frequency (ryo)</strong></td>
<td><strong>5.297E-06</strong></td>
<td><strong>5.298E-06</strong></td>
<td><strong>5.297E-06</strong></td>
<td><strong>5.297E-06</strong></td>
</tr>
</tbody>
</table>

* Fraction of the total frequency
Change of Containment Failure Frequency

- NoCF
- ECF
- LCF
- BMT
- Bypass
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

OECD/NEA ISAMM, 2009.10.26~28
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
- 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

<table>
<thead>
<tr>
<th>Initiating Event Group (Number of Initiators)</th>
<th>CDF Contribution [1/a]</th>
<th>CDF Contribution, [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td>LOCAs (10)</td>
<td>2.61E-7</td>
<td>40.4</td>
</tr>
<tr>
<td>Transients (40)</td>
<td>8.07E-9</td>
<td>1.2</td>
</tr>
<tr>
<td>SGTRs (6)</td>
<td>2.87E-9</td>
<td>0.4</td>
</tr>
<tr>
<td><strong>Internal Events (Total)</strong></td>
<td>2.77E-7</td>
<td>42.9</td>
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<tr>
<td>Aircraft crashes (7)</td>
<td>1.13E-8</td>
<td>1.8</td>
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<tr>
<td>External Floods (1)</td>
<td>1.42E-8</td>
<td>1.8</td>
</tr>
<tr>
<td>Fires (23, more than 300 scenarios)</td>
<td>2.37E-8</td>
<td>0.4</td>
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<tr>
<td>Cooling Water Intake Plugging (2)</td>
<td>2.66E-9</td>
<td>0.4</td>
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<tr>
<td>Internal Floods (20)</td>
<td>1.34E-9</td>
<td>0.2</td>
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<tr>
<td>Seismic Events (41)</td>
<td>3.37E-7</td>
<td>52.1</td>
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<tr>
<td><strong>External Events (Total)</strong></td>
<td>3.65E-7</td>
<td>56.5</td>
</tr>
<tr>
<td>Other</td>
<td>3.87E-9</td>
<td>0.5</td>
</tr>
<tr>
<td><strong>Total CDF</strong></td>
<td>6.46E-7</td>
<td>100%</td>
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Are the results too optimistic??
## Comparison with other studies, CDF

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<tr>
<th>KKG (GPSA2009)</th>
<th>KKG (GPSA2003)</th>
<th>KKG (GPSA1994)</th>
<th>Convoy/Preconvoy (GRS studies)</th>
<th>KKB (2009), Modernized Westinghouse plant</th>
</tr>
</thead>
<tbody>
<tr>
<td>6.5E-7/a Plant State 2008</td>
<td>1.4E-6/a (Model of 2009 Plant State 2003)</td>
<td>2.3E-6/a (Model of 2009 Plant State 1994)</td>
<td>1.7E-6/a (without seismic) (preconvoy–4.6E-6/a with seismic)</td>
<td>1.7E-5/a (preliminary)</td>
</tr>
</tbody>
</table>

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
The CDF is dominated by external events, Seismic events ca. 52%, total 56.5% of CDF
Results of Level 2 PSA, power operation

<table>
<thead>
<tr>
<th>DENOTATION</th>
<th>RELEASE CATEGORY</th>
<th>RELEASE FREQUENCY, [1/A]</th>
</tr>
</thead>
<tbody>
<tr>
<td>VENTF</td>
<td>Release via filtered venting</td>
<td>1.70E-07</td>
</tr>
<tr>
<td>LERF</td>
<td>Large early release (within 10 hours after core damage)</td>
<td>5.08E-08</td>
</tr>
<tr>
<td>LLR</td>
<td>Large late (offsite) release</td>
<td>5.01E-08</td>
</tr>
<tr>
<td>SMREL</td>
<td>Small and moderate releases</td>
<td>2.92E-07</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Initiating Event Group</th>
<th>FDF per year</th>
<th>% of External Events Group</th>
<th>% of Total FDF</th>
</tr>
</thead>
<tbody>
<tr>
<td>External Events &amp; Internal Hazards</td>
<td>1.96E-06</td>
<td>81.8%</td>
<td>81.8%</td>
</tr>
<tr>
<td>Fires</td>
<td>1.31E-06</td>
<td>66.9%</td>
<td>54.8%</td>
</tr>
<tr>
<td>Seismic Events</td>
<td>5.47E-07</td>
<td>27.9%</td>
<td>22.8%</td>
</tr>
<tr>
<td>External Floods</td>
<td>5.34E-08</td>
<td>2.7%</td>
<td>2.2%</td>
</tr>
<tr>
<td>Internal Floods</td>
<td>2.86E-08</td>
<td>1.5%</td>
<td>1.2%</td>
</tr>
<tr>
<td>Cooling Water Intake Plugging</td>
<td>1.88E-08</td>
<td>1.0%</td>
<td>0.8%</td>
</tr>
<tr>
<td>Aircraft Crashes</td>
<td>3.85E-10</td>
<td>&lt; 0.1%</td>
<td>&lt; 0.1%</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Internal Initiating Event Group</th>
<th>FDF per year</th>
<th>% of Internal Event Group</th>
<th>% of Total FDF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Internal Events</td>
<td>4.36E-07</td>
<td>18.2%</td>
<td>11.1%</td>
</tr>
<tr>
<td>LOCAs</td>
<td>2.67E-07</td>
<td>61.1%</td>
<td>7.1%</td>
</tr>
<tr>
<td>Transients</td>
<td>1.70E-07</td>
<td>38.9%</td>
<td>7.1%</td>
</tr>
</tbody>
</table>

All Initiating Events 2.40E-06

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

Venting is also efficient for shutdown modes

Villigen, October 26-28, 2009

OECD/NEA “Implementation of Severe Accident Management”
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

<table>
<thead>
<tr>
<th>Event</th>
<th>Time (Case 1: 3 S/Gs Filled)</th>
<th>Time (Case 2: 2 S/Gs Filled)</th>
<th>Time (Case 3: No S/Gs Filled)</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>Swollen level at Top of active fuel (TAF)</td>
<td>1081s (0.3 h)</td>
<td>1463s (0.4 h)</td>
<td>1366s (0.4 h)</td>
<td>Local boiling starts earlier</td>
</tr>
<tr>
<td>Swollen level oscillates between 75% and 100% TAF</td>
<td>1886s (0.5 h)</td>
<td>1642s (0.5 h)</td>
<td>1571s (0.4 h)</td>
<td>Heat removal via the secondary side of the filled SGs, reflux-condenser mode of heat transfer</td>
</tr>
<tr>
<td>Evaporation of water in SGs completed (&lt;5m), start of pressure increase</td>
<td>27000-27300s (7.5 to 7.6h)</td>
<td>22356-22573s (6.2 to 6.3h)</td>
<td>0</td>
<td>Level below 5 m, some heat transfer is possible until water level drops below 1-1.5 m</td>
</tr>
<tr>
<td>Setpoint of safety Valve THxS090 achieved</td>
<td>44600s (12.4h)</td>
<td>33615s (9.3h)</td>
<td>8080s (2.2h)</td>
<td>Induced LOCA in the containment (40 cm²)</td>
</tr>
<tr>
<td>Start of Gap release</td>
<td>49473s (13.7h)</td>
<td>38457s (10.7h)</td>
<td>13424s (3.7h)</td>
<td>Onset of core damage, core damage state A according to SAMG</td>
</tr>
<tr>
<td>First clad melting</td>
<td>50860s (14.1h)</td>
<td>39638s (11.0h)</td>
<td>14724s (4.1h)</td>
<td>Core damage state B according to SAMG</td>
</tr>
<tr>
<td>Reactor vessel rupture</td>
<td>66880s (18.6h)</td>
<td>51891s (14.4h)</td>
<td>32795s (9.1h)</td>
<td>Core damage state C according to SAMG</td>
</tr>
</tbody>
</table>
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

<table>
<thead>
<tr>
<th>Case</th>
<th>Total</th>
<th>Noble Gas (NG)</th>
<th>Cs</th>
<th>CsI</th>
<th>Ba</th>
<th>Te</th>
<th>I</th>
<th>Ru</th>
<th>Mo</th>
<th>Ce</th>
<th>La</th>
<th>Cd</th>
<th>Sn</th>
</tr>
</thead>
<tbody>
<tr>
<td>2 SGs (mean)</td>
<td>6.54E18</td>
<td>5.55E18</td>
<td>3.56E16</td>
<td>3.19E17</td>
<td>7.67E16</td>
<td>6.67E16</td>
<td>1.79E14</td>
<td>2.84E15</td>
<td>3.90E17</td>
<td>7.49E16</td>
<td>9.75E14</td>
<td>8.68E15</td>
<td>1.57E16</td>
</tr>
<tr>
<td>3 SGs (lower limit)</td>
<td>5.98E18</td>
<td>5.17E18</td>
<td>3.16E16</td>
<td>2.91E17</td>
<td>8.44E16</td>
<td>8.57E16</td>
<td>1.393E14</td>
<td>3.57E15</td>
<td>7.14E16</td>
<td>2.22E07</td>
<td>6.56E15</td>
<td>5.60E15</td>
<td>8.35E15</td>
</tr>
<tr>
<td>0 SGs (upper limit)</td>
<td>6.68E18</td>
<td>5.70E18</td>
<td>2.61E16</td>
<td>2.79E17</td>
<td>7.51E16</td>
<td>1.03E17</td>
<td>1.91E14</td>
<td>2.53E15</td>
<td>3.26E17</td>
<td>1.39E17</td>
<td>3.89E15</td>
<td>6.32E15</td>
<td>1.14E16</td>
</tr>
</tbody>
</table>

The radioactivity releases are very similar despite the differences in accident progression.
## Insights from Accident and Source term analysis, shutdown, SBO, vessel head removed

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

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**

<table>
<thead>
<tr>
<th>EVENT</th>
<th>TIME</th>
<th>COMMENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Swollen level drops to top of active fuel</td>
<td>159400s (44.3h)</td>
<td>Fuel cooled by oscillating water/steam mixture until this point in time, start of fuel heat-up and oxidation</td>
</tr>
<tr>
<td>Start of gap release</td>
<td>180265 (50.1h)</td>
<td>“Fuel damage state B” “approximate” definition according SAMG</td>
</tr>
<tr>
<td>Exceedance of large release threshold (2.0E14 Cs)</td>
<td>183320s (50.9h)</td>
<td>Unfiltered release of this amount of Cs may lead to a radiation dose of 100 mSv</td>
</tr>
<tr>
<td>Failure of fuel racks, start of core (melt) -concrete interaction</td>
<td>203080s (56.4h)</td>
<td>Molten fuel starts to progress towards the sump area (possible bypass scenario)</td>
</tr>
</tbody>
</table>

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
Introduction

• Loviisa SAM strategy
  \(\rightarrow\) originally designed to cope with severe accidents starting from power operation

• Risk profile from PRA level 1* shows the importance of the shutdown states \(\rightarrow\) 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
For Loviisa NPP

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

**In-vessel retention of corium**
- Inlet valves
- Outlet valves
- Lowering of the bottom part of the thermal insulation/neutron shield
- Screening of impurities

**Containment isolation**
- Manual backup
- Local control centres
- Monitoring of the isolation success
- Monitoring of the leakages
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)
**H₂ burn**

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

![Graph showing maximum absolute pressure after combustion vs. initial absolute pressure before combustion](image)
H$_2$ burn

- $\rho_{\text{AICC}}$

- Well mixed upper comp.

- Typical for shutdown states

<table>
<thead>
<tr>
<th>Initial absolute pressure before combustion (bar)</th>
<th>Maximum absolute pressure after combustion (bar)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.1</td>
<td>1.0</td>
</tr>
<tr>
<td>1.2</td>
<td>1.1</td>
</tr>
<tr>
<td>1.3</td>
<td>1.2</td>
</tr>
<tr>
<td>1.4</td>
<td>1.3</td>
</tr>
<tr>
<td>1.5</td>
<td>1.4</td>
</tr>
<tr>
<td>1.6</td>
<td>1.5</td>
</tr>
<tr>
<td>1.7</td>
<td>1.6</td>
</tr>
<tr>
<td>1.8</td>
<td>1.7</td>
</tr>
<tr>
<td>1.9</td>
<td>1.8</td>
</tr>
<tr>
<td>2.0</td>
<td>1.9</td>
</tr>
</tbody>
</table>

Design pressure

Failure

OK
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

- Very large release
- Large release
- Leakage
- No release
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?
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
- Summary of Level 2 PSA results
- AM Strategies and their components
- Plant modifications
  - 2 phase schedule
- Conclusions
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 $\Rightarrow$ eventual outside cooling more effective
- nominal pressure in primary: 123 bar $\Rightarrow$ 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 m³ water reserve on the bubble condenser trays
## Summary of Level 2 PSA results

### Containment failure modes and their reasons

<table>
<thead>
<tr>
<th>Containment failure modes</th>
<th>Main reason of the cont. failure (physical phenomena)</th>
</tr>
</thead>
<tbody>
<tr>
<td>High pressure RPV rupture</td>
<td>Failure of primary depressurization (human error, valve failure)</td>
</tr>
<tr>
<td>By-pass</td>
<td>Steam generator tube/collector rupture</td>
</tr>
<tr>
<td>Early containment rupture</td>
<td>Hydrogen burn</td>
</tr>
<tr>
<td>Early enhanced containment leakage</td>
<td></td>
</tr>
<tr>
<td>Late containment rupture</td>
<td>Containment slow overpressurization</td>
</tr>
<tr>
<td>Late enhanced containment leakage</td>
<td>Cavity door seal failure due to high temperature (corium near to the door)</td>
</tr>
<tr>
<td>Early containment rupture with spray</td>
<td>Hydrogen burn</td>
</tr>
<tr>
<td>Early enhanced containment leakage with spray</td>
<td></td>
</tr>
<tr>
<td>Late containment rupture with spray</td>
<td></td>
</tr>
<tr>
<td>Late enhanced containment leakage with spray</td>
<td></td>
</tr>
<tr>
<td>Intact containment</td>
<td></td>
</tr>
<tr>
<td>Intact containment with spray</td>
<td></td>
</tr>
</tbody>
</table>
### Summary of Level 2 PSA results

#### Possible Accident Management Measures

<table>
<thead>
<tr>
<th>Main reason of the cont. failure (physical phenomena)</th>
<th>Possible accident management measures</th>
</tr>
</thead>
<tbody>
<tr>
<td>Failure of primary depressurization</td>
<td>SAMG</td>
</tr>
<tr>
<td>Steam generator tube/collector rupture</td>
<td>Bleed from ruptured SG to the containment</td>
</tr>
<tr>
<td>Hydrogen burn</td>
<td>Hydrogen recombiner, igniter or inerting</td>
</tr>
<tr>
<td>Cavity door seal failure</td>
<td>Isolation of room A004 or prevention of RPV failure</td>
</tr>
<tr>
<td>Containment late overpressurization</td>
<td>Filtered venting and/or spray</td>
</tr>
</tbody>
</table>
## Summary of Level 2 PSA results
### AM strategies and their components

<table>
<thead>
<tr>
<th></th>
<th>Base case</th>
<th>Strategy I</th>
<th>Strategy II</th>
</tr>
</thead>
<tbody>
<tr>
<td>Prevention of RPV failure</td>
<td>ECCS recovery</td>
<td>ECCS recovery</td>
<td>ECCS recovery + reactor cavity flooding</td>
</tr>
<tr>
<td>Hydrogen treatment</td>
<td>-</td>
<td>30 recombiners</td>
<td>30 recombiners</td>
</tr>
<tr>
<td>Limitation of radioactive releases</td>
<td>Spray recovery</td>
<td>Spray recovery</td>
<td>Spray recovery</td>
</tr>
<tr>
<td>Prevention of cont. overpressurization</td>
<td>-</td>
<td>Filtered venting</td>
<td>Filtered venting</td>
</tr>
<tr>
<td>Safe integrity of the reactor cavity</td>
<td>-</td>
<td>Isolation of room A004</td>
<td>Solved by cavity flooding</td>
</tr>
<tr>
<td>(External cooling of the molten material)</td>
<td>-</td>
<td></td>
<td>(Not challenged)</td>
</tr>
</tbody>
</table>

### Selection of AM procedures:
- **Release into the atmosphere:** no significant differences between 2 strategies
- **Basemat melt-through frequency:**
  - with isolation of room A004: $1.83 \cdot 10^{-5}$ 1/unit/year
  - with cavity flooding: $1.6 \cdot 10^{-7}$ 1/unit/year (in case of success)
### Summary of PSA Level 2 results

#### Frequency of the release categories

<table>
<thead>
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<td>Loss of coolant</td>
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Key elements of the SAM strategies

- Release management
- Prevention of core damage
- In-vessel retention or ex-vessel debris cooling
- Release and containment management

SAMG
AM strategies and their components

- **Measures to prevent core damage:**
  - Strictly perform the adequate EOPs (EOPs for shutdown 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 → not challenged

- Measures to safe confinement integrity:
  - Confinement isolation
  - Hydrogen treatment: application of 30 large passive recombiners (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 \(\Rightarrow\) 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 features ⇒ 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
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

• 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 confirmatory assessment 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

• 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.
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.
• 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

• 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 confirm 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 confirmatory 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

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<th>END USERS GROUP</th>
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## The Partners

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### 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 should 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 ...)

- **Meeting 2 - 01st of December 2008 - (IRSN -Fontenay-aux-Roses)**
  - General methods (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
  - Discussion on the End-Users workshop follow-up

- **Meeting 3 - 1st, 2d July 2009 - (VTT - Helsinki)**
  - Discussion on the first draft of guideline on general methods
  - Initial exchange of information on the following issues (some outcomes from SARNET will have to be considered)
  - 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

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

- **Meeting 5 - May 2010 (2 days - date to define)**
  - 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)*

- **Meeting 6 - December 2010**
  - 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].


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

“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 Probabilistic Safety Assessment Review Team (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
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

1. Beznau Plant
2. Beznau Accident Management Program
3. Realistic Evaluation of Shutdown Risk
4. Results of Beznau Shutdown PSA
5. Conclusions
Beznau NPP today
Oldest operating PWR worldwide
Plant extensively backfitted
Backfits and Frequency of Core Damage
Total of Internal and External Events

Safety Benefit by Factor of 100

Backfits:
- Seismic Restraints, 3rd DC Feed, EOP Changes
- Addition of Bunker System
- 2nd Hydro Supply, Feed-and-Boil, Startup Solutions
- Additional Emerg. Feedwater Train

Relaxations:
- Reduction of bunker system design to 1 train based on PSA results
2. Beznau Accident Management Program

Hardware for AM

Feed into Reactor (e.g. from fuel pool)

Feed into Spent Fuel Pit

Feed to Cont. Spray

Feed to Cont. Sump

Cont. Filtered Vent

Feed to Cont. Fan Coolers

Feed into SGs
Beznau Accident Management Procedures

Prevention of Core Damage

Mitigation of Core Damage

Core Damage

Accident Severity

Emergency Staff

Operating Crew

Severe Accident Management Guidelines

Emergency Operating Procedures

Operating Crew

EOPs (PWROG)

Link from Shutdown EOPs to AM Proc.

Full-Power SAMGs (PWROG)

Shutdown SAMGs (Westingh. Europe)

Alignment of Mobile Equip. (Beznau Invention)
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

- Long time windows during shutdown
- Alternate indications from TET for core uncovery
- Human errors dominate shutdown CDF
- Independent responsibility of emergency staff for shutdown SAMGs
- Alternate hardware such as fire water pumps

Comprehensive shutdown AM program may significantly reduce shutdown CDF
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

Core stays cooled until water level is mid of fuel length

3000Secondsbetween core uncovery and latest recovery time
- 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

---

### Modeling of AM Measures in Shutdown PSA

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<th>AAA6</th>
<th>LREL</th>
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</thead>
</table>

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Graphical representation of AM measures in shutdown PSA with nodes X1 to X6 connected in a hierarchical structure.
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

Considerable shutdown LERF from non-seismic events (failure to close cont. hatch)

Shutdown Risk < Full-Power Risk
Beznau has all modes, all events, Level 2 PSA

Seismic and fires cannot be neglected during shutdown

Area Events

Beznau has all modes, all events, Level 2 PSA

Seismic and fires cannot be neglected during shutdown
Human errors dominate shutdown CDF.
Cold Shutdown CDF and LERF without AM Measures

Frequency [per calendar year]

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<th>1.50E-05</th>
<th>2.00E-05</th>
<th>2.50E-05</th>
</tr>
</thead>
</table>

Without AM

With AM

Benefit of AM measures at least factor of 2

- All Initiators except Seismic
- Seismic
- Core/Fuel Damage
- Large Release

Accident Management and Risk Evaluation of Shutdown Modes at Beznau NPP | Axpo AG
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 into subjective probability assignment
- Practical need to minimize number 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
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
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 approaches (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)
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
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 into 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 preparedness
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

Nuclear Energy Law

Accompanying Ordinance

Regulatory Guidelines
Regulatory Basis

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

- PSA: Quality and Scope
- PSA: Applications
- 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

<table>
<thead>
<tr>
<th></th>
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<th>KKG</th>
<th>KKL</th>
<th>KKM</th>
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<td>SAMG (Full-Power)</td>
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OECD/NEA Workshop on „Implementation of SAM Measures“, October 2009
Status of Implementation: PSA

4 plant-specific PSA Models
Status of Implementation: PSA

Scope of the PSA Models:

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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 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

<table>
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<th>Types of Actions</th>
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<td>Cases</td>
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<td>19</td>
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</table>

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

<table>
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<tr>
<th>Types of Actions</th>
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<td>Cases</td>
<td>34</td>
<td>19</td>
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Modelling approach

- HRA-type analysis
- APET model

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**

- 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

<table>
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<td>KKM</td>
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## 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-situaton
- 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
Summary

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

<table>
<thead>
<tr>
<th>MERMOS/PSA</th>
<th>Simplified approach</th>
<th>Statistical approach</th>
<th>Detailed approach</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pre initiator</td>
<td>MERMOS pre initiator simplified (from FH7, under development)</td>
<td>MERMOS pre initiator statistique</td>
<td>MERMOS pre initiator détaillé (under development)</td>
</tr>
<tr>
<td>Initiator</td>
<td></td>
<td>(FH7) Future development</td>
<td></td>
</tr>
<tr>
<td>Post initiator</td>
<td>MERMOS post initiator forfaitaire</td>
<td>Observation guide for MERMOS Time related curves</td>
<td>MERMOS post initiator detailed</td>
</tr>
<tr>
<td>Crisis organization</td>
<td>MERMOS crisis team simplified (PSA level 1) (under development)</td>
<td>-</td>
<td>MERMOS detailed (PSA level 2)</td>
</tr>
<tr>
<td>Fire (under development)</td>
<td>MERMOS Fire screening</td>
<td>MERMOS Fire fighting statistical</td>
<td>MERMOS Fire fighting detailed</td>
</tr>
<tr>
<td>Seism, Flood ...</td>
<td></td>
<td></td>
<td>MERMOS Fire operation detailed</td>
</tr>
</tbody>
</table>

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

**Identification**

**Human Factor mission (HFE)**

- **Task analysis**
- **Modes of failure**
- **Qualitative analysis**
  - Operational stories leading to failure
- **Quantification**
  - HEPs

**WHERE**

**HOW**

**HOW MANY**

**WHAT**

**WHY**

- **Data analysis**
  - EDF’s Databank Knowledge + Statistics + Operational Expressions
  - Observations + Actual Events
  - Knowledge on failing operation

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

4 - P. Le Bot – ISAMM Workshop – oct. 2009
LEVEL 2 SPECIFICITY: THE NATIONAL CRISIS TEAM

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

**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 environment
  - 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?

**Identification**

- **Human Factors**
  - Emergency Operating System extended to Emergency Response Teams
  - Situation diagnosis extended to Prognosis

- **Emergency Operating System**
  - Severe Accident Experts analyses
  - Judgments of HFEs

- **WHAT**
  - Observations
  - Analysis

- **WHY**
  - Knowledge on failing operation

- **WHERE**
  - Description of organization
  - Requirements

- **HOW**
  - Task analysis
  - Qualitative analysis
  - Operational stories leading to failure
  - Quantification

- **HOW MANY**
  - Model of Safe Operation (context, CICAS, reconfiguration, strategy, diagnosis, action)
  - Severe Accident Experts analyses and judgments of HFEs

**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

\[ P(\text{HFE failure}) = P_{\text{residual}} + \sum_{i=1}^{n} P(\text{scenario } i) \]
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)

<table>
<thead>
<tr>
<th>Probability of mission failure (HEP):</th>
<th>1.1 E-1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Residual Probability</td>
<td>6. E-5</td>
</tr>
</tbody>
</table>

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

No scenario identified for Wrong Strategy, No Action, Wrong Prognosis, No Prognosis
**Scenario structure / quantification**

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

- **Context (or situation)**
  - Conjunction 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

<table>
<thead>
<tr>
<th>SCENARIO n° 9</th>
<th>Probability: 2.2 E-2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Description:</td>
<td>Team is waiting for confirmation from the Site Main Control Center to depressurize (Site MCC is not available in time)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Situation features</th>
<th>Probability</th>
</tr>
</thead>
<tbody>
<tr>
<td>Following its procedure, the Safety Engineer needs the confirmation of Site MCC</td>
<td>0.9</td>
</tr>
<tr>
<td>Site MCC decision id delayed</td>
<td>0.3</td>
</tr>
<tr>
<td>Local Crisis Team does not decide instead of Site MCC</td>
<td>0.9</td>
</tr>
<tr>
<td>National Technical Center is not able to help</td>
<td>1</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>CICA (Main features of the emergency operation)</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Delegation of decision to Site MCC</td>
<td>0.9</td>
</tr>
</tbody>
</table>

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

- BWR LTSBO (CDF 3E-6/yr)
- BWR STSBO w/RCIC (CDF 3E-7/yr)
- BWR STSBO (CDF 3E-7/yr)
- PWR LTSBO (CDF 2E-5/yr)
- PWR STSBO (CDF 2E-6/yr)
- PWR TISGTR (CDF 5E-7/yr)
- PWR ISLOCA (CDF 3E-8/yr)

- loss of inventory
- core degradation

- * = lower head failure
- ▲ = start of release to environment

Time after accident initiation (hours)
Cesium Release for Unmitigated Sensitivity Cases

![Graph showing cesium release for different scenarios](image-url)
Cesium Release for Unmitigated Sensitivity Cases

1982 Siting Study

Surry steam generator tube rupture

Surry station blackouts

Surry ISLOCA

Peach Bottom station blackouts

Time (hr)

Fraction of Initial Core Inventory
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)

Conclusions

• 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 affect 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)
Deterministic Evaluation
of Quantitative Health
Objective and Target of
Severe Accident
Management

Changwook HUH, Namduk SUH
Gunhyo JUNG
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

![Chart showing acceptable risks for different countries]
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-MACCS2

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 in² 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 \( P_{\text{DBA}} \)
  - ILRT is performed at this \( P_{\text{DBA}} = 57 \) psig
- Thus, source terms are modeled to be released either through 6.0 in² area when the pressure reaches 169 psig, or at 0.1 vol%/day when the pressure reaches \( P_{\text{DBA}} \)
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 m³ and air density of 1.2 kg/m³
- MELCOR flow path area corresponding to this leak rate at P_{DBA} is calculated to be 1.0E-5 m²
- 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

![Graph showing population weighted risk for different scenarios]
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 ft² area.

<table>
<thead>
<tr>
<th>Cases</th>
<th>Average Individual Risk</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Leak through 0.1 ft² (9E-3 m²) at 132 psig and sprayed at 10 hrs later</td>
<td>Can. Fat. / 0 - 1.6 km 5.54E-2  &lt;br&gt;Can. Fat. / 64 - 80 km 3.13E-5</td>
</tr>
<tr>
<td>2. Leak through 0.01 ft² (9E-4 m²) at 1.82E5 sec.</td>
<td>Can. Fat. / 0 - 1.6 km 4.45E-3  &lt;br&gt;Can. Fat. / 64 - 80 km 1.59E-6</td>
</tr>
<tr>
<td>3. Leak at the rate of 0.1 vol%/day (1E-5 m²) at PDBA and sprayed at 5.E5 sec.</td>
<td>Can. Fat. / 0 - 1.6 km 2.64E-3  &lt;br&gt;Can. Fat. / 64 - 80 km 2.53E-7</td>
</tr>
</tbody>
</table>
Evaluation Results and Target of SAM

- Pressure behaviour for 3 cases
Evaluation Results and Target of SAM

- Sensitivity Evaluation for Case-3

<table>
<thead>
<tr>
<th>Cases</th>
<th>Average Individual Risk</th>
</tr>
</thead>
</table>
| 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%/day 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
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
Thank you very much
Verification of the SAMG for Paks NPP with MAAP code calculations

Gábor Lajtha, Zsolt Téchy
NUBIKI, Hungary
József Elter, Éva Tóth
Paks NPP, Hungary

OECD/NEA Workshop on Implementation of Severe Accident Management Measures
PSI, Villigen, Switzerland, October 26-28, 2009
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- In-vessel melt retention
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- Summary
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.
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.
• 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

- 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

<table>
<thead>
<tr>
<th>60 minutes delay from severe accident signal</th>
<th>Primary system pressure</th>
</tr>
</thead>
<tbody>
<tr>
<td>Letdown cross section</td>
<td>Equivalent diameter (mm)</td>
</tr>
<tr>
<td>1</td>
<td>1.5386E-04</td>
</tr>
<tr>
<td>2</td>
<td>3.0772E-04</td>
</tr>
<tr>
<td>3</td>
<td>4.6158E-04</td>
</tr>
<tr>
<td>4</td>
<td>6.1544E-04</td>
</tr>
<tr>
<td>5</td>
<td>7.6930E-04</td>
</tr>
<tr>
<td>6</td>
<td>9.2316E-04</td>
</tr>
<tr>
<td>7</td>
<td>1.0770E-03</td>
</tr>
<tr>
<td>8</td>
<td>1.2309E-03</td>
</tr>
<tr>
<td>9</td>
<td>1.3847E-03</td>
</tr>
<tr>
<td>10</td>
<td>1.5386E-03</td>
</tr>
<tr>
<td>11</td>
<td>1.6925E-03</td>
</tr>
<tr>
<td>12</td>
<td>1.8463E-03</td>
</tr>
<tr>
<td>14</td>
<td>2.0002E-03</td>
</tr>
</tbody>
</table>

Green: effective depressurization, red: failure to reduce pressure
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

<table>
<thead>
<tr>
<th>LPIS restoration time (h)</th>
<th>Lower support plate failure (h)</th>
<th>Hydrogen production until the support plate failure and at the end of the calculation (kg)</th>
<th>Debris mass in the bottom of the reactor vessel (t)</th>
</tr>
</thead>
<tbody>
<tr>
<td>7</td>
<td>-</td>
<td>(246)</td>
<td>-</td>
</tr>
<tr>
<td>8</td>
<td>12,7</td>
<td>177 (252)</td>
<td>20</td>
</tr>
<tr>
<td>9</td>
<td>9,4</td>
<td>220 (248)</td>
<td>40</td>
</tr>
<tr>
<td>10</td>
<td>9,4</td>
<td>220 (251)</td>
<td>40</td>
</tr>
<tr>
<td>11</td>
<td>9,4</td>
<td>220 (274)</td>
<td>40</td>
</tr>
<tr>
<td>12</td>
<td>9,4</td>
<td>220 (302)</td>
<td>47</td>
</tr>
<tr>
<td>12,7</td>
<td>9,4 (Vessel failure: 12,7)</td>
<td>220 (282)</td>
<td>80</td>
</tr>
</tbody>
</table>

- 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

<table>
<thead>
<tr>
<th>Event</th>
<th>0 t/h</th>
<th>6 t/h</th>
<th>12 t/h</th>
<th>18 t/h</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core uncovery:</td>
<td>21 s</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core uncovery II:</td>
<td>1962 s</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>( T_{\text{gas at core exit}} &gt; 643 \text{ K} )</td>
<td>2107 s</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>( T_{\text{gas at core exit}} &gt; 825 \text{ K} )</td>
<td>2500 s</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core melt starts</td>
<td>2938 s</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Water injection starts</td>
<td>No injection</td>
<td>3098 s</td>
<td>3098 s</td>
<td>3098 s</td>
</tr>
<tr>
<td>Lower plate failure</td>
<td>6031 s</td>
<td>6000 s</td>
<td>4927 s</td>
<td>3988 s</td>
</tr>
<tr>
<td>Vessel failure</td>
<td>10842 s</td>
<td>10975 s</td>
<td>12038 s</td>
<td>No failure</td>
</tr>
</tbody>
</table>

#### Hydrogen production

<table>
<thead>
<tr>
<th>Event</th>
<th>209 kg</th>
<th>207 kg</th>
<th>213 kg</th>
<th>238 kg</th>
</tr>
</thead>
<tbody>
<tr>
<td>At lower plate failure</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>At vessel failure</td>
<td>240 kg</td>
<td>239 kg</td>
<td>268 kg</td>
<td>243 kg</td>
</tr>
</tbody>
</table>

- 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)
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
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 $T_{\text{EXIT}}=1100$ 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

- 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

- 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.
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 m³/h), designed to cool the containment. The system is equipped with heat exchangers, but no filter is available.
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.
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
– 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
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 – $1 \times 10^{-6}$ to $5 \times 10^{-6}$/year
  – Short-term SBO (seismic initiator) loss of AC, loss of DC – $1 \times 10^{-7}$ to $5 \times 10^{-7}$/year
  – Loss of Vital AC Bus E12 – $\sim 5 \times 10^{-7}$/year
Scenario Selection

• Surry scenarios analyzed in SOARCA
  – Long-term SBO (seismic initiator) loss of AC – $1 \times 10^{-5}$ to $2 \times 10^{-5}$/year
  – Short-term SBO (seismic initiator) loss of AC, loss of DC, gross rupture of ECST – $1 \times 10^{-6}$ to $2 \times 10^{-6}$/year
  – Short-term SBO (seismic initiator) with thermally induced SGTR – $2.5 \times 10^{-7}$ to $5 \times 10^{-7}$/year
  – ISLOCA – $7 \times 10^{-7}$/year (licensee PRA), $3 \times 10^{-8}$/year (SPAR)
  – Spontaneous SGTR – $5 \times 10^{-7}$/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 pressure controlled by periodic opening of hard pipe vent
Peach Bottom Long-Term SBO

RPV Pressure

Operator manually opens 1 SRV

Station batteries exhausted; portable power supply engaged to sustain open SRV

Two-Phase Mixture Level [in]

Automatic RCIC actuation

Operator takes manual control of RCIC

Reactor Water Level

Operator manually opens 1 SRV

RCIC Isolation Signals (assume manual over-ride)

BAF

TAF

RPV pressure above HCTL

High supp. pool temp
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
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
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

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) ...

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

Each coloured cell = one CV node of input deck

ISAM 2009, M. Sonnenkalb, GRS Cologne
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
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 H₂ 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 ...

- Lower drywell gas composition
  - MELCOR - COCOSYS

- Drywell & wetwell pressure
  - MELCOR - COCOSYS
Use visualization tools to check appropriate modelling of relevant plant specifics ...

RCS and core behaviour

combustion limits
Assess the results carefully and determine relevant phenomena ...

**BWR-69:**

- **H₂ release** through RPV-SV into wetwell before RPV failure
  - Low H₂ 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 H₂ generation after melt relocation into lower plenum water pool -> quenching and evaporation
  - Early local RPV failure
  - Very high H₂ 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
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

Assess the results carefully and determine source term relevant phenomena.
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

<table>
<thead>
<tr>
<th></th>
<th>CsI</th>
<th>CsOH</th>
<th>Te</th>
<th>Kr</th>
</tr>
</thead>
<tbody>
<tr>
<td>Building intact</td>
<td>0.005</td>
<td>0.01</td>
<td>0.005</td>
<td>0.24</td>
</tr>
<tr>
<td>Door damaged</td>
<td>0.07 (0.02 - 0.11)</td>
<td>0.07 (0.02 - 0.13)</td>
<td>0.10 (0.02 - 0.21)</td>
<td>0.88 (0.69 - 0.99)</td>
</tr>
<tr>
<td>Door and building damaged</td>
<td>0.34 (0.19 - 0.48)</td>
<td>0.33 (0.20 - 0.46)</td>
<td>0.37 (0.19 - 0.53)</td>
<td>1.0</td>
</tr>
</tbody>
</table>

**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
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
• 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

~520 Mg H₂O in Calandria Vault

~120 Mg D₂O in Heat Transport System

~230 Mg D₂O in Calandria Vessel
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

Uncovered channels deform into contact with first submerged channel row

Weight of suspended debris is supported by first submerged channel row

Liquid level in calandria slowly drops with boil-off

Uncovered channels deform into contact with first submerged channel row

Calandria Vessel
Rupture Disc

Calandria Vessel

Uncovered channels deform into contact with first submerged channel row

Weight of suspended debris is supported by first submerged channel row

Calandria Vessel

Liquid level in calandria slowly drops with boil-off

Calandria Vessel

Reacto Vault
Water
IN VESSEL SCD ACCIDENTS
CHANNEL DISASSEMBLY

CHANNELS BREAK UP BY SAGGING
Analyses & Small Scale Tests

- Tube wall straining mainly at bundle junctions
- Support by channels at lower elevation
- Channel segments around fuel bundles are rigid
- Submerged channels support uncovered channels

MODERATOR LEVEL TRANSIENT GOVERNS RATE OF DISASSEMBLY

UNRESTRICTED / ILLIMITÉ
IN VESSEL SCD ACCIDENTS
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

- Heat is radiated to the walls of the calandria vessel
- Debris is no longer quenched so it begins to melt and penetrate the porous debris bed
- Residual water pool has completely boiled-off
- Heat is conducted through the debris, and the wall of the calandria vessel to the reactor vault water
• 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
• MAAP-CANDU is the primary tool 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

CANDU COMPONENTS
HTS, CALANDRIA
VESSEL

MAAP-CANDU CHANNELS
SYSTEM

MAAP GENERIC ROUTINES
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
Nodalization of CANDU 6 PHTS

Steam Generator 2

Steam Generator 1

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

C : Cold
H : Hot

UNRESTRICTED / ILLIMITÉ
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)

Time (h)

UO2 Mass in intact Core/Loop (kg)

Disassembly Begins
Loop 1
Loop 2
Core Collapse
CANDU 6 Calandria Vessel Behavior

Graph showing:
- Total Debris Mass
- Crust Mass
- Particulate Mass

Core Debris Mass (kg) vs. Time (h)

- Time (h): 0 to 80
- Core Debris Mass (kg): -20000 to 160000

Mass distributions over time.
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
• 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

<table>
<thead>
<tr>
<th>Event/Case</th>
<th>SBO-D1</th>
<th>SLOCA-E</th>
<th>SFB-C</th>
<th>SGTR-B</th>
<th>SSA-A</th>
</tr>
</thead>
<tbody>
<tr>
<td>SG dry (h)</td>
<td>0.8</td>
<td>2</td>
<td>33</td>
<td>10.7</td>
<td>138</td>
</tr>
<tr>
<td>PT/CT rupture (h)</td>
<td>3.8</td>
<td>41</td>
<td>38</td>
<td>13.1</td>
<td>not applicable</td>
</tr>
<tr>
<td>Core disassembly starts (h)</td>
<td>76</td>
<td>17</td>
<td>1.4</td>
<td>52</td>
<td>13.2</td>
</tr>
<tr>
<td>Containment fails (h)</td>
<td>23</td>
<td>47</td>
<td>38</td>
<td>37</td>
<td>37.6</td>
</tr>
<tr>
<td>CV fails (h)</td>
<td>not applicable</td>
<td>81</td>
<td>54.5</td>
<td>120</td>
<td>66</td>
</tr>
<tr>
<td>MCCI begins (h)</td>
<td>not applicable</td>
<td>92</td>
<td>63</td>
<td>not applicable</td>
<td>78</td>
</tr>
<tr>
<td>Calandria Vault floor failure (h)</td>
<td>not applicable</td>
<td>not applicable</td>
<td>137</td>
<td>not applicable</td>
<td>not applicable</td>
</tr>
<tr>
<td>Percentage of initial inventory of the active isotopes (Cs+Rb+I) released to environment at 500,000 s</td>
<td>3.2%</td>
<td>2.7%</td>
<td>6.8%</td>
<td>12.8%</td>
<td>0.55%</td>
</tr>
</tbody>
</table>
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
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

<table>
<thead>
<tr>
<th>Scenario/Case</th>
<th>SBO-D1</th>
<th>SLOCA-E</th>
<th>SFB-C</th>
<th>SGTR-B</th>
<th>SSA-A</th>
</tr>
</thead>
<tbody>
<tr>
<td>Class III Power</td>
<td>No</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>Class IV Power</td>
<td>No</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>High pressure ECC (HP ECC)</td>
<td>Yes</td>
<td>Yes</td>
<td>No</td>
<td>No</td>
<td>No</td>
</tr>
<tr>
<td>Medium pressure ECC (MP ECC)</td>
<td>Yes</td>
<td>Yes</td>
<td>No</td>
<td>No</td>
<td>No</td>
</tr>
<tr>
<td>Low pressure ECC (LP ECC)</td>
<td>Yes</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
<td>No</td>
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<tr>
<td>Loop Isolation</td>
<td>No</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>No</td>
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<tr>
<td>Emergency power supply (EPS)</td>
<td>72 h</td>
<td>Yes</td>
<td>No</td>
<td>Yes</td>
<td>No</td>
</tr>
<tr>
<td>availability</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Emergency core cooling heat exchanger (ECC HX)</td>
<td>No</td>
<td>No</td>
<td>No</td>
<td>No</td>
<td>No</td>
</tr>
<tr>
<td>SG Crash Cool Down</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>No</td>
<td>Not Applicable</td>
</tr>
<tr>
<td>Moderator Drain</td>
<td>4.2 kg/s</td>
<td>No</td>
<td>30 kg/s</td>
<td>No</td>
<td>No</td>
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<tr>
<td>Shield Cooling</td>
<td>No</td>
<td>No</td>
<td>No</td>
<td>No</td>
<td>No</td>
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<tr>
<td>Auxiliary Feed Water (AFW)</td>
<td>No</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>No</td>
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<tr>
<td>Main Steam Safety Valve</td>
<td>available</td>
<td>available</td>
<td>available</td>
<td>available</td>
<td>locked open</td>
</tr>
</tbody>
</table>
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
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
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 $D_{ae}>1\mu m$, and increases sharply with larger $D_{ae}$

- 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.
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 in-service 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 and Flaw depth graphs](image-url)
Degraded tube and temperature distribution at top of tubesheet

- Degraded tube distribution obtained from SG in-service 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 \((T, \sigma)\)
  – 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)

Mean=0.025, EF=3.8
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

- 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

Majumdar, 1999, NED

0.25in
2in
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

Graph showing:
- SG secondary side gas temperature over time
- Key events:
  - Induced SGTR
  - Accumulator injection
  - Core relocated to lower head
  - Lower head failed
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

<table>
<thead>
<tr>
<th>Liquid level in SG and reactor vessel lower head</th>
</tr>
</thead>
<tbody>
<tr>
<td>[Graph showing liquid level over time]</td>
</tr>
</tbody>
</table>

Accumulator injection

Liquid level at top of tubesheet

Time (s)  x 10^4
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

Example of release without mitigation

AM: SG refill

Pool dried

Aerosol deposition onto SG

Pool scrubbing due to AM

Accu. injection

Example of mitigated release
(not based on actual data)
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

Division of Nuclear Power Safety
Royal Institute of Technology (KTH)
Stockholm, Sweden
Outline

- Motivation
- Objectives
- MELCOR modeling of CRGT cooling
- Preliminary results
- Implication to reactor safety
- Outlook
The in-vessel retention (IVR) has been implemented in a few PWRs, but not applied to any BWR so far.

The BWRs have more potential for IRV in terms 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.
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
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.


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.
Objectives

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

- To investigate the efficacy of CRGT cooling, 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).

Analysis at system level  
Mechanistic analysis for LP  

Increase the reliability of the assessment
## Reference reactor

<table>
<thead>
<tr>
<th>No.</th>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Thermal power, MWt</td>
<td>3900</td>
</tr>
<tr>
<td>2</td>
<td>Operating pressure in vessel, bar</td>
<td>70</td>
</tr>
<tr>
<td>3</td>
<td>Reactor vessel outside height, m</td>
<td>21</td>
</tr>
<tr>
<td>4</td>
<td>Internal vessel diameter, m</td>
<td>6.4</td>
</tr>
<tr>
<td>5</td>
<td>Vessel wall thickness, m</td>
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<tr>
<td>6</td>
<td>Effective core height, m</td>
<td>3.68</td>
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<tr>
<td>7</td>
<td>Number of CRGTs</td>
<td>169</td>
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<td>8</td>
<td>Nominal flow rate per CRGT, g/s</td>
<td>62.5</td>
</tr>
<tr>
<td>9</td>
<td>Nominal flow rate in entire CRGT, kg/s</td>
<td>10.5</td>
</tr>
<tr>
<td>10</td>
<td>Initial UO$_2$ mass, kg</td>
<td>146000</td>
</tr>
<tr>
<td>11</td>
<td>Initial Zr mass, kg</td>
<td>52680</td>
</tr>
<tr>
<td>12</td>
<td>Initial steel mass, kg</td>
<td>100400</td>
</tr>
</tbody>
</table>
Nodalization
## Calculation matrix

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Scenario-1</td>
<td>Station blackout (SBO) <strong>without CRGT cooling</strong></td>
</tr>
<tr>
<td>Scenario-2</td>
<td>( SBO + CRGT \text{ cooling at } 10.5\text{kg/s from time 0} )</td>
</tr>
<tr>
<td>Scenario-3</td>
<td>( SBO + CRGT \text{ cooling at } 10.5\text{kg/s from time 1 hr} )</td>
</tr>
<tr>
<td>Scenario-4</td>
<td>( SBO + CRGT \text{ cooling at } 10.5\text{kg/s from time 2 hrs} )</td>
</tr>
<tr>
<td>Scenario-5</td>
<td>( SBO + CRGT \text{ cooling at } 42\text{kg/s from time 2 hrs} )</td>
</tr>
</tbody>
</table>
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.

Melt mass in the LP and Melt ejection

Pressure evolution
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 = 20°C.

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 10.5 kg/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 10.5 kg/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: \( t = 2 \text{ h} \).
  - 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
**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 $4 \times 10.5 = 42$ kg/s.

![Melt mass in the LP](image)
Results (10)

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

Water level in the core

Temperature in CRGTs
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

**VVER-1000 containment**: modern PWR design
- design pressure: 0.49 Mpa
- free volume: 66000 m3
- leak rate: 0.1 % /24 hours

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

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

**Advantageous design feature**:
- free room on containment floor for corium spreading out of the reactor cavity (over 100 m2)
**Strategies**

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

Corium management: 4 scenarios (strategies) studied:
- 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

**SA scenario:** blackout

**Codes used:**
- CORCON 3.01h (part of MELCOR 1.8.5)
- MEDICIS 1.3.2 (part of ASTEC code)

**Common assumptions:**
- homogeneous melt pool
- siliceous concrete in the cavity

**Initial and boundary conditions 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)

<table>
<thead>
<tr>
<th>Scenario</th>
<th>CORCON</th>
<th>MEDICIS</th>
</tr>
</thead>
<tbody>
<tr>
<td>TB2: no corium spreading</td>
<td>1.802</td>
<td>1.803</td>
</tr>
<tr>
<td>no water injection</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TB3: corium spreading</td>
<td>0.780</td>
<td>0.940</td>
</tr>
<tr>
<td>no water injection</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TB4: no corium spreading</td>
<td>1.797</td>
<td>1.792</td>
</tr>
<tr>
<td>water injection</td>
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<td></td>
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<tr>
<td>TB5: corium spreading</td>
<td>0.805</td>
<td>0.823</td>
</tr>
<tr>
<td>water injection</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Assessment of strategies

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

<table>
<thead>
<tr>
<th>Scenario</th>
<th>CORCON</th>
<th>MEDICIS</th>
</tr>
</thead>
<tbody>
<tr>
<td>TB2: no corium spreading</td>
<td>1.962</td>
<td>1.804</td>
</tr>
<tr>
<td>no water injection</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TB3: corium spreading</td>
<td>1.366</td>
<td>0.921</td>
</tr>
<tr>
<td>no water injection</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TB4: no corium spreading</td>
<td>1.833</td>
<td>1.793</td>
</tr>
<tr>
<td>water injection</td>
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<td></td>
</tr>
<tr>
<td>TB5: corium spreading</td>
<td>0.729</td>
<td>0.804</td>
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<tr>
<td>water injection</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Assessment of strategies

Comparison of MEDICIS and CORCON ablated concrete mass (ton)

<table>
<thead>
<tr>
<th>Scenario</th>
<th>CORCON</th>
<th>MEDICIS</th>
</tr>
</thead>
<tbody>
<tr>
<td>TB2: no corium spreading</td>
<td>471.0</td>
<td>334.3</td>
</tr>
<tr>
<td>no water injection</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TB3: corium spreading</td>
<td>398.1</td>
<td>313.3</td>
</tr>
<tr>
<td>no water injection</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TB4: no corium spreading</td>
<td>439.7</td>
<td>329.7</td>
</tr>
<tr>
<td>water injection</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TB5: corium spreading</td>
<td>285.7</td>
<td>266.3</td>
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<tr>
<td>water injection</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Assessment of strategies

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

!!!!!!! Nevertheless 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:

**Reference scenario** without containment basement melting through
**Modified scenario** with melting through of containment basement

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

Assumptions for modified scenario:
- 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

Fig 1: TOTAL MASS IN CAVALITY

Comparison of REF & MOD Scenarios

- REF
- MOD-CAV1
- MOD-CAV2
Melting through of CNT basement

Fig. 5: AXIAL MELT PENETRATION

Comparison of REF & MOD Scenarios

- REF
- MOD-CAV1
- MOD-CAV2
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:
- reinforcing 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
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
Outline

Symptoms for Transition

Comparison of Transition Temperatures

Influencing Factors

WGAMA CET Working Group

Conclusions
The Transition to Severe Accident Management

Trip or safeguards actuation

- Plant state
- Power operation
- Hot shutdown
- Intermediate shutdown
- Cold shutdown
- RCS open

Onset of core damage

- Preventive measures
- Emergency Operating Procedures
- Shutdown Emergency Operating Procedures

Mitigative measures

- Severe Accident Management Guidelines
- Shutdown Severe Accident Management Guidelines
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)
Comparison of SAMG Entry Conditions

SAM Entry Condition Comparison

Primary Pressure (bar) vs. Core Exit Temperature (°C)

- CEOG
- B&WOG
- Loviisa
- WOG
- OSSA
- GIAG
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’
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.
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

SOURCE TERMS BY RELEASE PATHWAYS
- Steam Side (93.9%)
  - 93.5% probability of SG tube rupture (break covered)
  - 4.20% probability of fuel damage, SG tube rupture (break covered)
  - 0.6% probability of SG tube rupture (dry out)
  - 0.6% probability of SG tube rupture (overfill)
Example of one of the Source Terms

0.27% probability of spike release, SGTR (overfill)

All nuclides

Release Rate (TBq / h)

Total Released Activity (TBq)

Xe-133
I-131
Cs-137

14/12
3 AM
6 AM
9 AM
12 PM
3 PM
6 PM
9 PM
15/12
0 AM
3 AM
6 AM
9 AM
12 PM
3 PM
6 PM
9 PM

NRG-presentation
Example of a Belief Network

- Primary Circuit Integrity
- Secondary Circuit Integrity
- Reactor Building Integrity
- Auxiliary Building Integrity
- Auxiliary Circuit Integrity
- Fuel Integrity
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

- STV2 (RCS) Reactor Coolant System
- STV1 Fuel
- STV3 (SS) Secondary Circuit
- STV4 (PC) Primary Containment
- STV5 (RR) Ring Room
- STV6 (AB) Auxiliary Building

Contains filtration and venting to the environment.
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 °C
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

Alfred Torri, Vladimir Pokorny and Uwe Lüttringhaus
Risk Management Associates, Inc.
(Contact: torri@gorma.com)
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

HARDWARE
- ECCS: Prevent CD
- SAMMs: Prevent ECF and Mitigate Release

EMERGENCY CONTROL TEAM
- OPERATORS - EOPs
- TSC: - SAMGs

Reactor Safety with Severe Accident Management
Is the Deck stacked against Operators?

**PREVENTION: Prevent CD**

**MITIGATION: Prevent ECF**

**HARDWARE**

<table>
<thead>
<tr>
<th></th>
<th>HPI-1</th>
<th>HPI-2</th>
<th>LPIS-1</th>
<th>LPIS-2</th>
<th>RHR-1</th>
<th>RHR-2</th>
<th>RHR-3</th>
<th>RHR-4</th>
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<tbody>
<tr>
<td>DEPRESSURIZE</td>
<td>1 - 16</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Typical Level of Redundancy: 5 to 8 Trains

Typical Reliability: No CCF < 10-8

With CCF < 10-5

**OPERATORS**

<table>
<thead>
<tr>
<th></th>
<th>EOPs</th>
<th>SAMGs</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
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</tr>
</tbody>
</table>

**EMPHASIS THIS WORKSHOP**

Piped Firewater

ESW Injection

Portable Firewater

Cavity Flooding

Containment Spray

Filtered Containment Vent

ETC. …..

Typical Level of Redundancy: 3 to 5 Trains

Typical Reliability: No CCF < 10-4

With CCF < 10-3

80 to 90%

**MCR:** 1 Supervisor, 2 to 3 Operators each with his own Assignment

**TSC:** 5 to 10 each with his own Assignment

Off-site Support

**Typical Level of Redundancy:** 1

**Reliability required to match Hardware:** 10-8

10 to 20%
What is the Evidence?

<table>
<thead>
<tr>
<th>Statistical Evidence TMI/Chernobyl</th>
<th>Evidence from PSA</th>
<th>NRC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Damage Frequency</td>
<td>2x10^-4 / RY</td>
<td>10^-5 to 10^-6/RY</td>
</tr>
<tr>
<td>Conditional Probability of ECF given CD</td>
<td>0.5</td>
<td>0.1 to 0.01</td>
</tr>
</tbody>
</table>

**CAUSE**

- Errors of OMISSION: 0 %
- Errors of COMMISSION: 100 %

**CONCLUSION 1:** The purpose of the (EOP/SAMG)s has to be to make sure that errors of commission 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
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

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Tecnatom

Julio Benavides
C.N. Trillo

José Manuel de Blas
C.N. Garoña, Nuclenor

Miguel Ángel Catena
C.N. Almaraz

Ismael Sol
A.N. Ascó-Vandellós II

Villigen-PSI, Switzerland, 26-28 October 2009
Presentation Contents

- Introduction
- Training Activities
- SAMG Revision
- Simulator models development
- Conclusions
Introduction

- 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

<table>
<thead>
<tr>
<th>Spanish NPP</th>
<th>Concept</th>
<th>Electric Output (MWe)</th>
<th>Startup Date</th>
<th>SAMG implementation date</th>
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<tbody>
<tr>
<td>Santa Mª de Garoña</td>
<td>GE BWR/3 Mark I</td>
<td>465</td>
<td>1971</td>
<td>December 2000</td>
</tr>
<tr>
<td>Cofrentes</td>
<td>GE BWR/6 Mark III</td>
<td>1080</td>
<td>1984</td>
<td>December 2000</td>
</tr>
<tr>
<td>Vandellós II</td>
<td>W PWR 3-L</td>
<td>1080</td>
<td>1987</td>
<td>December 2000</td>
</tr>
<tr>
<td>Trillo</td>
<td>KWU PWR 3-L</td>
<td>1065</td>
<td>1988</td>
<td>2002</td>
</tr>
</tbody>
</table>
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.
Specific retraining programs are addressed to different personnel profiles:

- **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.
Training Activities

✓ 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.**
Severe Accidents Training in Spain: Experiences and Relevant Features

SAMG retraining planning

EOPs domain
- MSLB / SGTR
- LOCA
- SBLO
- FWLB
- LOCA

Transition to severe accident
- FR-C.1
- ECA-1.3
- ECA-0.0
- FR-C.1
- FR-C.1

SAMGs domain
- SAG-5 / SCG-1
- SAG-7 / SCG-3
- SAG-3 / SAG-4
- SAG-1 / SAG-2
- 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
    o Following and evaluating plant parameters
    o Radiological group (source term and doses estimation)
    o Tasks and responsibilities of TSC members
    o SAMG accomplishment.
    o Nowadays in
      ▪ Almaraz NPP (SACAT Project)
      ▪ Garoña NPP (MOCAT Project)
      ▪ Development process in Trillo NPP.
Severe Accidents Training in Spain: Experiences and Relevant Features

ISAMM-2009-15
Radiological Screen (Garoña NPP)
Basic issues in these revisions

- Applicable changes package included in the generic guidelines revision.
- Results and experiences of training courses and exercises carried out during last years (6-7 years of real experience).
- Methodology to make easier the future updating (Maintenance Control Sheets remarking the change cause).
- Increase of applicability and efficiency (PSA revisions, plant design modifications).
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)
  - Passive Autocatalytic Recombiners (PAR), Trillo has this system.
  - Analysis to improve the filling capability of the reactor cavity (“dry cavity”)
  - Analysis to improve the fast filling capability of the Refuelling Water Storage Tank (RWST)

- Changes in Trillo NPP (severe accident RSK recommendations) at implementation date
  - Control Room Air Filtering
  - Secondary “Feed and Bleed”
  - Emergency Power Supply
  - Containment Hydrogen Control (PAR)
Simulator Models Development

✓ Tecnatom has carried out the implementation of a SA module for the full scope simulator of Laguna Verde NPP (Mexico, 2003-2005).

- GE design BWR/5, owned by CFE (Electricity Federal Commission).
- MAAP-4 based “Containment Advanced Model” (MAC) 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

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”
  o Efficient communication between TSC and Control Room
  o Compromising of different plant organisations
  o Decision making process.
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 B₄C 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 B₄C as the control rod showed an unexpected behavior. The use of B₄C 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$_3$I 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$_3$I,

• Establish a process for fast and efficient decomposition of CH$_3$I 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$_3$I 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 CH₃I 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 CH₃I

• 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 patented¹

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 $\beta$ irradiations

In the shielded cell of the Hot Lab… a computerized, remote-operating system has been installed

Photo of reaction vessel in $\gamma$-irradiation chamber and gamma-cell
Effectiveness of PSI Process

T: 22 °C
[CH₃I] 1.10⁻⁷ - 2.10⁻⁴ mol dm⁻³

Decomposition rate (s⁻¹)

Temperature (°C)

Decomposition rate (s⁻¹)

PSI chemical control
no chemical control
pH9
## Quantification of the Enhanced Decomposition of CH$_3$I

<table>
<thead>
<tr>
<th>Reaction mechanism</th>
<th>Enhancement factor in reaction rate with respect to that at 25°C or 80°C</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>25 °C</td>
</tr>
<tr>
<td>No additives (hydrolysis alone)</td>
<td>1</td>
</tr>
<tr>
<td>Radiolysis + Hydrolysis</td>
<td>11·10$^3$</td>
</tr>
<tr>
<td>Hydrolysis + THS alone</td>
<td>15·10$^3$</td>
</tr>
<tr>
<td>Hydrolysis + THS+ALI</td>
<td>200·10$^3$</td>
</tr>
<tr>
<td>Hydrolysis + Radiolysis + THS+ALI</td>
<td>210·10$^3$</td>
</tr>
</tbody>
</table>

*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

![Graph showing the release fraction (%) vs. time (min) and dose (Gy). The graph illustrates the effectiveness of different conditions in suppressing iodide oxidation.

- Very oxidizing conditions by $N_2O$ sparging
- No chemical control
- PSI chemical control
- pH5 continuous sparging with argon during irradiation]
No degradation due to presence of other ions

<table>
<thead>
<tr>
<th>Ion</th>
<th>pH5</th>
<th>pH9</th>
<th>Index</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\text{UO}_2(\text{NO}_3)_2$</td>
<td></td>
<td></td>
<td>(1)</td>
</tr>
<tr>
<td>$\text{NaNO}_3$</td>
<td></td>
<td></td>
<td>(2)</td>
</tr>
<tr>
<td>$\text{NaCl}$</td>
<td></td>
<td></td>
<td>(3)</td>
</tr>
<tr>
<td>$\text{Na}_2\text{SO}_4$</td>
<td></td>
<td></td>
<td>(4)</td>
</tr>
<tr>
<td>$\text{Na}_2\text{C}_2\text{O}_4$</td>
<td></td>
<td></td>
<td>(5)</td>
</tr>
<tr>
<td>$\text{FeSO}_4$</td>
<td></td>
<td></td>
<td>(6)</td>
</tr>
<tr>
<td>Mixture of all ions</td>
<td></td>
<td></td>
<td>(7)</td>
</tr>
<tr>
<td>No other ion</td>
<td></td>
<td></td>
<td>(8)</td>
</tr>
</tbody>
</table>
Implementation of PSI process in NPP containment venting filter systems

- 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 core damage frequency involving scenarios resulting in containment venting, if equipped, via the venting filter. This means, if the core damage frequency (CDF) is roughly $7 \times 10^{-6} \text{ y}^{-1}$ it means that the venting frequency is roughly $4 \times 10^{-6} \text{ y}^{-1}$.

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 unacceptable high containment iodine concentrations during the core melting phase, especially from the cores with $\text{B}_4\text{C}$ 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

- KKG 970 MW (1979)
- KKL 1165 MW (1984)
- KKM 355 MW (1971)
- KKB 365 + 365 MW (1969 + 1971)
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

- Update of the existing Shutdown EOPs (SFA)
- Development of new SSAMG (SFA-AM) Containment Flooding during Shutdown Conditions
- Evaluation and Validation of Accident Management Strategy
- Update of existing KKL Analysis Model (MELSIM/MELCORE)
- Analysis of Severe Accident Phenomenology during Shutdown Conditions
- Input from Leibstadt Shutdown PSA

Kernkraftwerk Leibstadt

OECD/NEA Workshop "Implementation of Severe Accident Management (SAM) Measures, October 26-28, 2009"
### Leibstadt NPP Shutdown PSA Results

<table>
<thead>
<tr>
<th>Event Type</th>
<th>Description</th>
<th>CDF At-Power</th>
<th>FDF Shutdown</th>
<th>Overall CDF</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Internal Events</strong></td>
<td>All LOCA Events</td>
<td>1.04E-07</td>
<td>3.34E-08</td>
<td>1.37E-07</td>
</tr>
<tr>
<td></td>
<td>Transients and special initiators</td>
<td>3.26E-07</td>
<td>5.92E-09</td>
<td>3.32E-07</td>
</tr>
<tr>
<td></td>
<td><strong>Total:</strong></td>
<td><strong>4.30E-07</strong></td>
<td><strong>3.94E-08</strong></td>
<td><strong>4.69E-07</strong></td>
</tr>
<tr>
<td><strong>External Events</strong></td>
<td>Earthquakes</td>
<td>2.14E-06</td>
<td>3.01E-07</td>
<td>2.44E-06</td>
</tr>
<tr>
<td></td>
<td>High winds and tornadoes</td>
<td>6.47E-08</td>
<td>1.21E-08</td>
<td>7.68E-08</td>
</tr>
<tr>
<td></td>
<td>Airplane crash</td>
<td>1.34E-08</td>
<td>5.66E-10</td>
<td>1.40E-08</td>
</tr>
<tr>
<td></td>
<td>Weir failure</td>
<td>3.21E-14</td>
<td>1.96E-13</td>
<td>2.28E-13</td>
</tr>
<tr>
<td></td>
<td><strong>Total:</strong></td>
<td><strong>2.22E-06</strong></td>
<td><strong>3.14E-07</strong></td>
<td><strong>2.53E-06</strong></td>
</tr>
<tr>
<td><strong>Area Events</strong></td>
<td>Fire</td>
<td>7.59E-07</td>
<td>4.07E-07</td>
<td>1.17E-06</td>
</tr>
<tr>
<td></td>
<td>Flood</td>
<td>5.02E-07</td>
<td>5.71E-07</td>
<td>1.07E-06</td>
</tr>
<tr>
<td></td>
<td>Turbine Missile</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td><strong>Total:</strong></td>
<td><strong>1.26E-06</strong></td>
<td><strong>9.78E-07</strong></td>
<td><strong>2.24E-06</strong></td>
</tr>
<tr>
<td><strong>Grand Total</strong></td>
<td></td>
<td><strong>3.91E-06</strong></td>
<td><strong>1.33E-06</strong></td>
<td><strong>5.24E-06</strong></td>
</tr>
</tbody>
</table>

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)

<table>
<thead>
<tr>
<th>Shutdown EOPs (SFA) (prevention)</th>
<th>Shutdown SAMG (SFA-AM) (mitigation)</th>
</tr>
</thead>
<tbody>
<tr>
<td>SFA-1704-43 Loss of Shutdown Cooling (RHR/SEHR) during Shutdown</td>
<td>SFA-1704-AM02 Reactor Vessel and Containment Flooding during Refueling and Shutdown Conditions</td>
</tr>
<tr>
<td>SFA-1704-44 Loss of Coolant during Shutdown</td>
<td></td>
</tr>
<tr>
<td>SFA-1704-46 Loss of Power Supply during Shutdown</td>
<td></td>
</tr>
</tbody>
</table>

- 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

Scenario 1
Loss of RHR by the Station Blackout SBO. The shutdown line is manually isolated

Scenario 3
Station Blackout combined with a break in the common section of the RHR shutdown line
**Analyzed Scenarios – Vessel Open**

**Scenario 5**
Loss of RHR cooling caused by the Station Blackout

**Scenario 7**
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

<table>
<thead>
<tr>
<th>Scenario</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
</tr>
</thead>
<tbody>
<tr>
<td>1) SBO, RHR isolated</td>
<td>1E-6</td>
<td>1E-7</td>
<td>&lt;1E-8</td>
<td>&lt;1E-8</td>
<td>1E-6</td>
<td>&lt;1E-8*</td>
<td>&lt;1E-8*</td>
<td>&lt;1E-8*</td>
</tr>
<tr>
<td>2) SBO, RHR not isolated</td>
<td>1E-6</td>
<td>1E-7</td>
<td>&lt;1E-8</td>
<td>&lt;1E-8</td>
<td>1E-6</td>
<td>&lt;1E-8*</td>
<td>&lt;1E-8*</td>
<td>&lt;1E-8*</td>
</tr>
<tr>
<td>3) SBO, RHR leak, not isolated</td>
<td>1E-6</td>
<td>1E-7</td>
<td>&lt;1E-8</td>
<td>&lt;1E-8</td>
<td>1E-6</td>
<td>&lt;1E-8*</td>
<td>&lt;1E-8*</td>
<td>&lt;1E-8*</td>
</tr>
<tr>
<td>4) SBO, RHR leak, isolated -30'</td>
<td>1E-6</td>
<td>1E-7</td>
<td>&lt;1E-8</td>
<td>&lt;1E-8</td>
<td>1E-6</td>
<td>&lt;1E-8*</td>
<td>&lt;1E-8*</td>
<td>&lt;1E-8*</td>
</tr>
</tbody>
</table>

#### Entry frequency (appr.) [1/calendar year]

- Scenario 1: 1E-6, Scenario 2: 1E-7, Scenario 3: <1E-8, Scenario 4: <1E-8, Scenario 5: 1E-6, Scenario 6: <1E-8*, Scenario 7: <1E-8*, Scenario 8: <1E-8*

#### Scenario Time [hour]

<table>
<thead>
<tr>
<th>Event</th>
<th>Scenario 1</th>
<th>Scenario 2</th>
<th>Scenario 3</th>
<th>Scenario 4</th>
<th>Scenario 5</th>
<th>Scenario 6</th>
<th>Scenario 7</th>
<th>Scenario 8</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Uncovery (TAF)</td>
<td>8.25</td>
<td>3.79</td>
<td>1</td>
<td>0.77</td>
<td>14.29</td>
<td>12.5</td>
<td>1.64</td>
<td>1.3</td>
</tr>
<tr>
<td>Gap release</td>
<td>9.63</td>
<td>4.8</td>
<td>2.29</td>
<td>1.93</td>
<td>15.77</td>
<td>13.2</td>
<td>2.36</td>
<td>1.64</td>
</tr>
<tr>
<td>Start Release to the Environment</td>
<td>10.3</td>
<td>5.32</td>
<td>2.96</td>
<td>2.08</td>
<td>16.45</td>
<td>14.2</td>
<td>3.82</td>
<td>2.44</td>
</tr>
<tr>
<td>Vessel Breach</td>
<td>19.54</td>
<td>16.2</td>
<td>15.66</td>
<td>9.68</td>
<td>38.68</td>
<td>20.6</td>
<td>7.45</td>
<td>18.1</td>
</tr>
<tr>
<td>Start of Core Concrete Interaction</td>
<td>20.42</td>
<td>16.42</td>
<td>16.76</td>
<td>9.74</td>
<td>38.72</td>
<td>20.8</td>
<td>8.26</td>
<td>18.8</td>
</tr>
</tbody>
</table>

- 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

- 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
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 extent 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

Milan CVAN
VUJE, Inc., Slovakia

Dušan Šiko
SE, a.s., Slovakia

October 2009
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
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)
Obr. 7.0.5-6  PODLAŽIE +6,0 m
In-vessel retention of corium

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 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 activities 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

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

- 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”

Debris Bed Formation
- Debris particle formation and agglomeration
- Particle sedimentation and spreading
- Debris particle packing

Validation data

Synthesis of simulation and experiment
- Study of feedbacks and sensitivity

Properties of prototypical debris bed

Experiment
- DEFOR-HT
- DEFOR-A

Simulation
- VAPEX FCI code
- Models for particle morphology
- DECOSIM code
- DEFORSIM code

Synthesis
- DEFOR-HT
- DEFOR-LT
- DEFOR-HT
- DEFOR-LT
Containment: 4x4x4 m, 5 bar max pressure

DEFOR-HT (High Temperature) experimental program
## DEFOR-S (Snapshot) Test Matrix

<table>
<thead>
<tr>
<th>N</th>
<th>Experiment</th>
<th>Simulant</th>
<th>Mixture</th>
<th>Pool depth, m</th>
<th>Melt temp., °C</th>
<th>Melt superheat, °C</th>
<th>Water temp., °C</th>
<th>Porosity</th>
<th>Fraction of agglomerates, %</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>DEFOR-S1</td>
<td>MnO-TiO₂</td>
<td>Eutectic</td>
<td>0.65</td>
<td>1450</td>
<td>81</td>
<td>16</td>
<td>71</td>
<td>0</td>
</tr>
<tr>
<td>2</td>
<td>DEFOR-S2</td>
<td>MnO-TiO₂</td>
<td>Eutectic</td>
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<td>18</td>
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<tr>
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<td>DEFOR-S3</td>
<td>Bi₂O₃-WO₃</td>
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<td>950</td>
<td>70</td>
<td>21</td>
<td>70</td>
<td>0</td>
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<td>4</td>
<td>DEFOR-S4</td>
<td>WO₃-TiO₂</td>
<td>Eutectic</td>
<td>0.65</td>
<td>1400</td>
<td>167</td>
<td>20</td>
<td>69</td>
<td>0</td>
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<tr>
<td>5</td>
<td>DEFOR-S5</td>
<td>Bi₂O₃-WO₃</td>
<td>Eutectic</td>
<td>0.65</td>
<td>980</td>
<td>100</td>
<td>75</td>
<td>59</td>
<td>20</td>
</tr>
<tr>
<td>6</td>
<td>DEFOR-S6</td>
<td>Bi₂O₃-WO₃</td>
<td>Non-eutectic</td>
<td>0.65</td>
<td>1060</td>
<td>-20</td>
<td>20</td>
<td>68</td>
<td>0</td>
</tr>
<tr>
<td>7</td>
<td>DEFOR-S7</td>
<td>Bi₂O₃-WO₃</td>
<td>Non-eutectic</td>
<td>0.65</td>
<td>1010</td>
<td>16</td>
<td>19</td>
<td>62</td>
<td>0</td>
</tr>
<tr>
<td>8</td>
<td>DEFOR-S8</td>
<td>Bi₂O₃-WO₃</td>
<td>Non-eutectic</td>
<td>0.65</td>
<td>1020</td>
<td>26</td>
<td>75</td>
<td>46</td>
<td>90</td>
</tr>
<tr>
<td>9</td>
<td>DEFOR-S9</td>
<td>Bi₂O₃-WO₃</td>
<td>Non-eutectic</td>
<td>1.1</td>
<td>1070</td>
<td>45</td>
<td>11</td>
<td>68</td>
<td>0</td>
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<td>10</td>
<td>DEFOR-S10</td>
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<td>62</td>
<td>8</td>
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<td>DEFOR-S11</td>
<td>Bi₂O₃-WO₃</td>
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<td>1070</td>
<td>200</td>
<td>53</td>
<td>57</td>
<td>0</td>
</tr>
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<td>12</td>
<td>DEFOR-S12</td>
<td>Bi₂O₃-WO₃</td>
<td>Non-eutectic</td>
<td>1.1</td>
<td>1150</td>
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<td>50</td>
<td>65</td>
<td>0</td>
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<td>13</td>
<td>DEFOR-S13</td>
<td>Bi₂O₃-WO₃</td>
<td>Eutectic</td>
<td>1.1</td>
<td>1100</td>
<td>230</td>
<td>35</td>
<td>68</td>
<td>0</td>
</tr>
</tbody>
</table>
DEFOR-S5: Water 75°C, Eutectic melt, $T_{\text{melt}} = 980$ °C
Porosity 59.3%, Mass of agglomerates 20%

Fragile Agglomerates
DEFOR-S8: Water 75°C,
Non-Eutectic melt, $T_{\text{melt}} = 1020$°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
DEFOR-A (Agglomeration) experiment

Debris agglomeration as a function of pool depth. Data for code validation.
## DEFOR-A Test Conditions

<table>
<thead>
<tr>
<th>N</th>
<th>Parameter</th>
<th>A2</th>
<th>A5</th>
<th>A6</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Component 1</td>
<td>Bi2O3</td>
<td>Bi2O3</td>
<td>Bi2O3</td>
</tr>
<tr>
<td>2</td>
<td>Component 2</td>
<td>WO3</td>
<td>WO3</td>
<td>WO3</td>
</tr>
<tr>
<td>3</td>
<td>Component 1 molar fraction, %</td>
<td>27%</td>
<td>27%</td>
<td>27%</td>
</tr>
<tr>
<td>4</td>
<td>Component 2 molar fraction, %</td>
<td>73%</td>
<td>73%</td>
<td>73%</td>
</tr>
<tr>
<td>5</td>
<td>Eutectic mixture</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>6</td>
<td>Density of the mixture, kg/m³</td>
<td>7811</td>
<td>7811</td>
<td>7811</td>
</tr>
<tr>
<td>7</td>
<td>Melt volume, liter</td>
<td>3</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td>8</td>
<td>Melt mass, kg</td>
<td>23.43</td>
<td>23.43</td>
<td>23.43</td>
</tr>
<tr>
<td>9</td>
<td>Melting temperature of the melt, °C</td>
<td>870</td>
<td>870</td>
<td>870</td>
</tr>
<tr>
<td>10</td>
<td>Maximum temperature in the funnel, °C</td>
<td>973</td>
<td>972</td>
<td>1006</td>
</tr>
<tr>
<td>11</td>
<td>Water temperature before melt pouring, °C</td>
<td>94</td>
<td>91</td>
<td>73</td>
</tr>
<tr>
<td>12</td>
<td>Water temperature after melt pouring, °C</td>
<td>98</td>
<td>96</td>
<td>78</td>
</tr>
<tr>
<td>13</td>
<td>Water pool depth, m</td>
<td>1.52</td>
<td>1.52</td>
<td>1.52</td>
</tr>
<tr>
<td>14</td>
<td>Jet free fall height, m</td>
<td>0.18</td>
<td>0.18</td>
<td>0.18</td>
</tr>
<tr>
<td>15</td>
<td>Jet diameter, mm</td>
<td>20</td>
<td>10</td>
<td>12</td>
</tr>
<tr>
<td>16</td>
<td>Maximum melt pool depth in the funnel, m</td>
<td>0.15</td>
<td>0.15</td>
<td>0.15</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Catcher</th>
<th>Depth measured from water surface, m</th>
<th>Elevation from the pool bottom, m</th>
</tr>
</thead>
<tbody>
<tr>
<td>Catcher-1</td>
<td>0.6</td>
<td>0.9</td>
</tr>
<tr>
<td>Catcher-2</td>
<td>0.9</td>
<td>0.6</td>
</tr>
<tr>
<td>Catcher-3</td>
<td>1.2</td>
<td>0.3</td>
</tr>
<tr>
<td>Catcher-4</td>
<td>1.5</td>
<td>0</td>
</tr>
</tbody>
</table>

- 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
  - Melt 24 kg
  - Pool depth 1.52 m
  - Melt superheat 136K
  - Djet=12 mm
  - Water subcooling 22 K

DEFOR-A6: Top view on catchers
Beds are heap-like
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
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
  - $D_{jet}=22$ mm
  - Water subcooling 2 K
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 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.
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

Analytical
High solidification rate + solid fracture due to thermal stresses → rock-type particles

**Are Results Prototypical?**

**Particle morphology**

**DEFOR Simulant**

**Corium**

DEFOR-S7 experiment
WO$_3$-Bi$_2$O$_3$

TROI 13 experiment
Corium (UO$_2$=70%, ZrO$_2$=30%)
Are Results Prototypical?

Particle morphology

Slow solidification rate \(\rightarrow\) 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

P. Kudinov
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M. Davydov
Electrogorsk Research and Engineering Center on Nuclear Power Plants Safety (EREC), Electrogorsk, Russia
Debris Bed Formation (DEFOR)

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How?

Data

Gap in knowledge

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**Particle formation**
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- Solidification and fracture
- Size distribution, morphology

**Jet fragmentation**
- Formation of droplets

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  - Non-homogeneity
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**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”

Debris Bed Formation
- Debris particle formation and agglomeration
- Particle sedimentation and spreading
- Debris particle packing

Validation data

Prototypical conditions

VAPEX FCI code
+ Models for particle morphology

DECOSIM code
DEFORSIM code

- Synthesis of simulation and experiment
- Study of feedbacks and sensitivity

Properties of prototypical debris bed
Agglomerates and Cakes

- Agglomerates - “soldered” together groups of particles

- “Cake” is formed when liquid melt fraction is bigger than fraction of solid particles
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
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)

\[
\frac{\partial}{\partial t}(\rho_f A) + \frac{\partial}{\partial z}(\rho_f u_f A) = -\Gamma_{\text{frag}}
\]

\[
\frac{\partial}{\partial t}(\rho_f u_f A) + \frac{\partial}{\partial z}(\rho_f u_f^2 A) = -g\left(\rho_f - \rho_a\right)A - C_D \pi R \rho_a (u_f - u_a)^2 - u_f \Gamma_{\text{frag}}
\]

- Dynamics of melt jet and its fragmentation (Eulerian approach)

\[
\frac{d\vec{u}_k}{dt} = -\vec{F}_{\text{vis}} - \vec{F}_{\text{vis}} - g\left(\rho_a - \rho_f\right)
\]

\[
\rho_f \frac{dT_f}{dt} = -R_f (T_f - T_i) - R_f (T_f - T_i) - \psi \varepsilon_f T_f^4
\]

- Droplet sedimentation, debris formation (Lagrangian approach)

- Radiation heat transfer from droplets to coolant

- Droplet temperature profile, crust formation
Jet Breakup Mode Influence on Pre-Deposited State of the Debris

Epstein & Fauske (1985)
Stripping of Kelvin-Helmholtz instabilities from jet lateral surface

Chu & Corradini, (1989)
jet leading edge breakup

How jet breakup mode can affect pre-deposition state of the debris?
Influence of Jet Breakup Mode on Pre-Deposited State of the Debris

Particles are mostly completely solid

Small fraction of completely solid particles

\[ \delta_{rel} = \frac{\delta}{R} \]
Influence of Jet Breakup Mode on Pre-Deposition State of the Debris

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

Comparison of mass fraction of completely solidified particles calculated with coarse and fine grids for pool depth $H_{\text{pool}} = 7$ m.

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 $H_{\text{pool}} = 7$ m.
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 $m_{liq}$

\[
\alpha = \frac{m_{aggl}}{m_{liq}}
\]

$\alpha$ – coefficient of agglomeration.

To take into account intrinsic uncertainties in the agglomeration phenomena, $\alpha$ 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

<table>
<thead>
<tr>
<th>Case</th>
<th>Melt jet diameter</th>
<th>Coolant state</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case-1</td>
<td>Dj=10 mm</td>
<td>Subcooling</td>
</tr>
<tr>
<td>Case-2</td>
<td>Dj=10 mm</td>
<td>Saturation</td>
</tr>
<tr>
<td>Case-3</td>
<td>Dj=20 mm</td>
<td>Subcooling</td>
</tr>
<tr>
<td>Case-4</td>
<td>Dj=20 mm</td>
<td>Saturation</td>
</tr>
</tbody>
</table>
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 $\alpha$

$$\alpha_{ij} = \frac{m_{agl}}{m_{liq}}$$

$$\alpha(m_{liq}) = \begin{cases} 4 \cdot m_{liq}, & m_{liq} \leq 0.5 \\ 1/m_{liq}, & m_{liq} > 0.5 \end{cases} \quad (2)$$

$$\alpha(m_{liq}) = \begin{cases} 25/4 \cdot m_{liq}, & m_{liq} \leq 0.4 \\ 1/m_{liq}, & m_{liq} > 0.4 \end{cases} \quad (3)$$

- Highest values for $\alpha$ come from simulations with smallest predicted fraction of liquid particles combined with biggest experimentally observed fraction of agglomerates
- Formula (2) provide enveloping estimation for $\alpha$ 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

- 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 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

**Pool:**
- Diameter: 9 m
- **Depth:** 7-12 m
- Pressure: 1 bar
- Water: saturated

**Melt:**
- Total mass: 180 t
- **Jet diameter:** 70 – 300 mm
- Temperature: 3000 K
- Composition: Eutectic corium

\[ D_{\text{jet}} = 300 \text{ mm} \quad \text{time} = 0.0000 \text{ s} \]
Agglomeration Map Sensitivity to $\alpha$

- Results of prediction are robust and insensitive to small variations of bounding closure for $\alpha$.

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

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
DEFOR Publications
2007-2009

Experimental

Analytical
OECD SERENA: A Fuel Coolant Interaction Programme (FCI) devoted to reactor case

P. Piluso
Commissariat à l'Énergie Atomique, Cadarache, France

S.W. Hong
Korea Atomic Energy Research Research Institute, Korea

Implementation of Severe Accident Management Measures (ISAMM-2009)
Paul Scherrer Institute Villigen, Switzerland October 26 - 28, 2009
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
State of the art

- More than 30 years of a complex physics 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...
State of the art

**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**

- Water at Test: system pressure 0.5 MPa
- Support Plate
- Trigger of melt-bottom contact and after 4 sec.

**In-vessel. Maximum pressure at bottom**

- JASMINE F1 (r=0.1)
- VESUVIUS (trigger cell)
- FCI (centre)
- TEXAS UW (G1)
- TEXAS KK C1
- TEXAS KK C2
- MATTINA (centre)
- IDINO (r=0.007)
- IDINO (r=0.008)
- MC3D T2 (centre)
- MC3D PSA (centre)
- ESFROSE (centre)

**In-vessel. Maximum impulse at bottom**

- JASMINE F1 (r=0.1)
- FCI (centre)
- TEXAS UW (G1)
- TEXAS KK C1
- TEXAS KK C2
- IDINO (r=0.007)
- IDINO (r=0.241)
- MC3D T2 (centre)
- MC3D PSA (centre)
- ESFROSE (centre)
- MATTINA (centre)
Serena (phase 1): Ex-vessel reactor case
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
Safety significance of ex-vessel SE

- 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

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 from 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)

• **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)

• 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

• **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

- 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.
## Instrumentation

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Sensing location</th>
</tr>
</thead>
<tbody>
<tr>
<td>Melt temperature 1</td>
<td>Top window</td>
</tr>
<tr>
<td>Melt temperature 2</td>
<td>Melt delivery path</td>
</tr>
<tr>
<td>Coolant temperature</td>
<td>IVT101 ~ IVT104</td>
</tr>
<tr>
<td>Dynamic pressure in the coolant (1)</td>
<td>PIVDP101 ~ PIVDP104</td>
</tr>
<tr>
<td>Dynamic pressure in the coolant (2)</td>
<td>KIVDP101 ~ KIVDP104</td>
</tr>
<tr>
<td>Dynamic load at the test section bottom</td>
<td>IVDL101</td>
</tr>
<tr>
<td>Ambient temperature in the pressure vessel</td>
<td>PVT001 ~ PVT005</td>
</tr>
<tr>
<td>Static pressure in the furnace vessel</td>
<td>FVSP001</td>
</tr>
<tr>
<td>Static pressure in the pressure vessel</td>
<td>PVSP004, PVSP005</td>
</tr>
<tr>
<td>Dynamic pressure in the pressure vessel</td>
<td>PVDP004, PVDP005</td>
</tr>
<tr>
<td>Melt velocity in water</td>
<td>IVT201 ~ IVT209</td>
</tr>
<tr>
<td>Melt entry velocity (video)</td>
<td>at the above of the interaction vessel</td>
</tr>
<tr>
<td>Void fraction</td>
<td>VFDP101 ~ VFDP103</td>
</tr>
<tr>
<td>Gas sampling for hydrogen detection</td>
<td>GAS005</td>
</tr>
</tbody>
</table>
Serena (phase 2): KROTOS facility

- Furnace vessel
  - Hot crucible melting method
  - Pressure vessel
  - Puncher
  - Trigger device
Serena (phase 2): KROTOS facility

- 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
Serena (phase 2): KROTOS facility

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

<table>
<thead>
<tr>
<th></th>
<th>KROTOS</th>
<th>TROI</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Challenging conditions (to be finalised through discussion with the partners)</strong></td>
<td>Standard geometrical conditions</td>
<td>High system pressure (0.5 MPa)</td>
</tr>
<tr>
<td></td>
<td>High melt superheat</td>
<td>Reduced free fall (Melt jet velocity)</td>
</tr>
<tr>
<td></td>
<td>High system pressure (0.5 MPa)</td>
<td>and thick melt jet</td>
</tr>
<tr>
<td>Mat:</td>
<td>to be decided</td>
<td></td>
</tr>
<tr>
<td><strong>Geometry effect</strong></td>
<td><strong>Effect of geometry by comparison between KROTOS and TROI</strong></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Standard conditions: jet of diameter 3 cm</td>
<td>Large jet at penetration (5 cm)</td>
</tr>
<tr>
<td></td>
<td>Mat 1: 70%UO₂-30%ZrO₂</td>
<td></td>
</tr>
<tr>
<td><strong>Material effect</strong></td>
<td>Oxidic composition</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Standard conditions</td>
<td>Large jet at penetration (5 cm)</td>
</tr>
<tr>
<td></td>
<td>Mat 2: 80%UO₂-20%ZrO₂</td>
<td></td>
</tr>
<tr>
<td><strong>Material effect</strong></td>
<td>Oxidation/composition</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Standard conditions</td>
<td>Large jet at penetration (5 cm)</td>
</tr>
<tr>
<td></td>
<td>Mat 3: 70%UO₂-30%ZrO₂ +steel +Zr</td>
<td></td>
</tr>
<tr>
<td><strong>Material effect</strong></td>
<td>Large solidus/liquidus ΔT</td>
<td>Large jet at penetration (5 cm). Failure at the bottom, considering layer inversion. (2-5 cm)</td>
</tr>
<tr>
<td></td>
<td>Mat 4: 70%UO₂-30%ZrO₂ +FP+iron oxide+absorber materials</td>
<td></td>
</tr>
<tr>
<td><strong>Reproducibility tests</strong></td>
<td>Idem Test 3 or 4</td>
<td>Idem Test 3 or 4</td>
</tr>
</tbody>
</table>
Improved Molten Core Cooling Strategy in a Severe Accident Management Guideline

J. H. Song¹, C. W. Huh², N. D. Suh³

¹Korea Atomic Energy Research Institute
²Korea Institute of Nuclear Safety

OECD/NEA Workshop (ISAMM-2009)
CONTENTS

- Introduction
- Inject into Containment Strategy
- RCS Depressurization Strategy
- Summary
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
OECD/NEA Workshop (ISAMM-2009)

Introduction

Strategy Flow Chart

Initiation of DFC in TSC

Monitoring Plant Safety Parameters

- All S/G level > 0.4 m
  - Y
  - N
- RCS pressure > 2.6 bar
  - Y
  - N
- Core exit temperature ≤ 365 °C
  - Y
  - N
- Containment level ≥ -2.0 m
  - Y
  - N
- Radiation level at plant boundary ≤ (WBI: 0.5 mSv/hr)
  - Y
  - N
- Containment pressure ≤ 1.4 bar
  - Y
  - N
- Containment H₂ concentration ≤ 5%
  - Y
  - N

- Monitoring-01: "Long term monitoring"

Monitoring Containment Severe Challenge Parameters

- Monitoring-01: "Inject into S/G"
- Mitigation-02: "Depressurize RCS"
- Mitigation-03: "Inject into RCS"
- Mitigation-04: "Inject into CV"

Radiation level at Plant boundary ≤ WBI: 15 mSv/hr

- Y
- N

Containment pressure ≤ 6.3 bar

- Y
- N

Mitigation-05: "Control FP release"

Mitigation-06: "Control CV condition"

- Y
- N

Containment H₂ concentration: out of severe challenge region

- Y
- N

Mitigation-07: "Control CV H₂"

Return to previous guideline

Termination-01: "SAM termination"

- Y
- N

- Core exit temperature ≤ 365 °C
- Radiation level at plant boundary ≤ WBI: 0.5 mSv/hr (30 min.)
- CV pressure ≤ 1.4 bar
- CV H₂ concentration ≤ 5%
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 not
Inject Into Containment Strategy

available water inventory

- RCS (including PZR) 6,109 ft³
- SIT(2) 2,544 ft³
- RWT 34,756 ft³
- Boron Tank (2) 534 ft³

-------------
43,943 ft³

cavity free volume

<table>
<thead>
<tr>
<th>height (ft)</th>
<th>area(ft²)</th>
<th>free vol.(ft³)</th>
</tr>
</thead>
<tbody>
<tr>
<td>below 6(sump)</td>
<td></td>
<td>4,942</td>
</tr>
<tr>
<td>7.83</td>
<td>4,960</td>
<td>9,093</td>
</tr>
<tr>
<td>18.0</td>
<td>4,795</td>
<td>57,843</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Time (sec)</th>
<th>Top Events</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>Reactor Trip</td>
</tr>
<tr>
<td>5,350</td>
<td>SG Dry out</td>
</tr>
<tr>
<td>9,852</td>
<td>Core Uncover</td>
</tr>
<tr>
<td>10,510</td>
<td>Core Dry out</td>
</tr>
<tr>
<td>14,530</td>
<td>Clad Melting</td>
</tr>
<tr>
<td>23,640</td>
<td>UO$_2$ Relocation to Lower Head</td>
</tr>
<tr>
<td>23,650</td>
<td>Lower Head Failure</td>
</tr>
</tbody>
</table>
Inject Into Containment Strategy

MCCI condition at Reactor Cavity

<table>
<thead>
<tr>
<th></th>
<th>At 10 hrs after SBO</th>
<th>At 24 hrs after SBO</th>
</tr>
</thead>
<tbody>
<tr>
<td>Corium Mass in Cavity</td>
<td>102.8 ton</td>
<td>166.5 ton</td>
</tr>
<tr>
<td>Concrete Mass Eroded</td>
<td>38.2 ton</td>
<td>122 ton</td>
</tr>
<tr>
<td>Ratio of Concrete Content</td>
<td>27%</td>
<td>42%</td>
</tr>
<tr>
<td>Melt Depth (by MELCOR code)</td>
<td>0.47m</td>
<td>1.17m</td>
</tr>
<tr>
<td>Remaining Base mat Depth</td>
<td>1.953m</td>
<td>1.333m</td>
</tr>
</tbody>
</table>

- 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)
• 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

<table>
<thead>
<tr>
<th>Time of Events (sec)</th>
<th>Low Depressurization Capacity (1 POSRV)</th>
<th>High Depressurization Capacity (3 POSRVs)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Case 1 (ref.)</td>
<td>Case 2</td>
</tr>
<tr>
<td>Depressurization Initiation (CET)</td>
<td>9845 (973K)</td>
<td>10410 (1100K)</td>
</tr>
<tr>
<td>SIT Injection</td>
<td>11262</td>
<td>11791</td>
</tr>
<tr>
<td>SIT Empty</td>
<td>24827</td>
<td>31169</td>
</tr>
<tr>
<td>Initial Vessel Breach</td>
<td>32451</td>
<td>37954</td>
</tr>
<tr>
<td>Pressure at RPV Failure (&lt;2.75MPa)</td>
<td>0.75</td>
<td>1.27</td>
</tr>
</tbody>
</table>
## RCS Depressurization Strategy

The effect of Optimized RCS depressurization strategy on the ex-vessel debris coolbility.
- Probability of power recovery at 13hrs is 98%

<table>
<thead>
<tr>
<th>Depressurization using Current Strategy</th>
<th>Depressurization using Optimum Discharge</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Initial Condition in Cavity</strong></td>
<td></td>
</tr>
<tr>
<td>corium mass in cavity ; 169 ton</td>
<td>corium mass in cavity ; 152 ton</td>
</tr>
<tr>
<td>corium height ; 0.6m</td>
<td>corium height ; 0.472 m</td>
</tr>
<tr>
<td>concrete content ; 32%</td>
<td>concrete content ; 24%</td>
</tr>
<tr>
<td>remaining thickness of basemat ; ~2.67m</td>
<td>remaining thickness of basemat ; 2.8m</td>
</tr>
<tr>
<td><strong>Out of range of data applicability, extrapolation gives rough estimation</strong></td>
<td>within the marginal point of data</td>
</tr>
<tr>
<td><strong>Results</strong></td>
<td></td>
</tr>
<tr>
<td>ablation depth to stabilization ; &gt;2.0m</td>
<td>ablation depth to stabilization ; ~ 1.2m</td>
</tr>
<tr>
<td>Uncertainty renders the coolability not guaranteed</td>
<td>Sufficient margin for the coolability guaranteed</td>
</tr>
</tbody>
</table>
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.
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 (12th to 14th)
- 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.
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.
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
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)
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)
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
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
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.
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
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.
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

Sergey Yakush

Institute for Problems in Mechanics, Russian Academy of Sciences, Moscow, Russia

Pavel Kudinov

Division of Nuclear Power Safety, Royal Institute of Technology (KTH), Stockholm, Sweden
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
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 coolability 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

- Reactor cavity (computational domain)
- Water level
- Vapor plume
- Debris bed

Particle trajectories (random walk model for turbulent dispersion)
Multifluid model for liquid/vapor mixture
Turbulence model for the continuous phase (water)
Two-phase filtration model (Ergun’s equation with relative permeability and passability correlations)

$Q = 120 \text{kW/m}^3$
$D = 3 \text{mm}$
$t = 30 \text{min}$
Governing Equations

**Debris Bed**

\[
\frac{\partial (\varepsilon \rho_i \alpha_i)}{\partial t} + \nabla (\varepsilon \rho_i \alpha_i U_i) = -\Gamma_i
\]

\[
\Gamma = \begin{cases} 
\frac{\dot{Q}}{\Delta H_{ev}} & \text{for } \alpha_l > 0 \\
0 & \text{for } \alpha_l = 0
\end{cases}
\]

\[
\varepsilon \alpha_i \nabla P = \varepsilon \alpha_i \rho_i g - F_{is}
\]

\[
F_{is} = \varepsilon \alpha_i \left( \frac{\mu_i}{KK_{ri}} j_i + \frac{\rho_i}{\eta_{ri}} |j_i| j_i \right)
\]

\[
K_{rl} = (1 - \alpha)^3, \quad \eta_{rl} = (1 - \alpha)^5
\]

\[
K_{rv} = \alpha^3, \quad \eta_{rv} = \alpha^5
\]

**Ambient Flow**

\[
\frac{\partial (\rho_i \alpha_i)}{\partial t} + \nabla (\rho_i \alpha_i U_i) = 0
\]

\[
\rho_i \alpha_i \left( \frac{\partial U_i}{\partial t} + (U_i \cdot \nabla)U_i \right) = -\alpha_i \nabla P
\]

\[
+ \alpha_i \nabla \tau_i + \alpha_i \rho_i g - F_{ij}
\]

\[
F_{ij} = \frac{3}{4} \alpha_i \frac{\rho_j}{D_i} C^{ij}_{D} |U_j - U_i| (U_j - U_i)
\]

\[
\frac{\partial \rho_i \alpha_i k_t}{\partial t} + \nabla (\rho_i \alpha_i k_t U_i) =
\]

\[
= \nabla \left( \alpha_i \frac{\mu}{\sigma_k} \nabla k_t \right) + \alpha_i (G - \varepsilon_t)
\]

\[
\frac{\partial \rho_i \alpha_i \varepsilon_t}{\partial t} + \nabla (\rho_i \alpha_i \varepsilon_t U_i) =
\]

\[
= \nabla \left( \alpha_i \frac{\mu}{\sigma_\varepsilon} \nabla \varepsilon_t \right) + \alpha_i \frac{\varepsilon_t}{k_t} (C_1 G - C_2 \varepsilon_t)
\]
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

\[
\frac{dr^k}{dt} = U^k_m \\
\rho_m \frac{dU^k_m}{dt} = -F_{lm} - F_{vm} - (\rho_m - \rho_a)g
\]

\[
F_{im} = \alpha_i \frac{3}{4} \frac{\rho_i |U_i - U_m|}{D_m} C_D(Re_m) (U_i - U_m) \quad U_l = \bar{U}_l + u'(k, \varepsilon)
\]

\[
C_D(Re_m) = \frac{24}{Re_m} (1 + 0.15 Re_m^{0.687}) \quad Re_m = \rho_i D_m |U_i - U_m| / \mu_i
\]
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
Natural convection flows promote flattening of debris bed, especially for fine particles.

For melt particles with size distribution, non-homogeneous 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)

Void fraction distributions in the debris bed for different specific heat release rates

130 W/kg

150 W/kg

180 W/kg
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

**Graph:**
- **Y-axis:** Max. void fraction, [-]
- **X-axis:** W, [W/kg]
- **Legend:**
  - X=0.5
  - D=2mm
  - D=3mm
  - X=0.2
  - D=2mm
  - D=3mm

**Key:**
- Data points indicate the dryout conditions for different permeability reduction factors and particle diameters.
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
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.
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”
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”
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”.
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.
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

Figure 1. Primary pressure

(continued)
RECOVERY OF WATER SUPPLY INTO STEAM GENERATORS FROM FIRE ENGINES

(continued)

Figure 2. Pressure in steam generators

P, kgs/cm²

1 2 3 4

0 10 20 30 40 50 60 70 80 90 100

0 4000 8000 12000 16000 20000

t, s

Figure 2. Pressure in steam generators
Figure 3. Coolant mass in the core axial nodes 1 to 5
Figure 4. Fuel rod cladding temperatures in the core axial nodes 1 to 5
Figure 5. Temperature of coolant at core inlet and outlet

(continued)
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).
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

Scope and Target

ORNL/CRL/FZK /IRSN(HEVA) (Finished)

VERCORS (Finished)

VEGA

PWR station blackout

Pressure [MPa]

0.1

10

Temperature [°C]

1000 2000 3000

Release

( Xe,I,Cs etc.)

( Sr,Ru,Ba etc.)

( U,Pu,Ce etc.)

High Volatility Low

VEGA (Verification Experiment of radionuclide Gas/Aerosol Release)

• Improvement of source term predictability for high burnup and MOX fuel use in LWR
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.

**Schematic of VEGA Test Apparatus**

**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 ThO₂ crucible that is stable under oxidizing and high temperature conditions
## VEGA Test Matrix

<table>
<thead>
<tr>
<th>Test No.</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>M1</th>
<th>6</th>
<th>M2</th>
<th>7</th>
<th>8</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel specimen</td>
<td>UO₂ PWR</td>
<td>UO₂ PWR</td>
<td>UO₂ PWR</td>
<td>UO₂ PWR</td>
<td>MOX ATR</td>
<td>UO₂ BWR</td>
<td>MOX ATR</td>
<td>UO₂ BWR</td>
<td>UO₂ BWR</td>
<td></td>
</tr>
<tr>
<td>Cladding</td>
<td>No</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
<td>No</td>
<td>No</td>
<td>Yes</td>
<td>No</td>
<td>yes</td>
<td>No</td>
</tr>
<tr>
<td>Burnup (GWd/tU)</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>47</td>
<td>43</td>
<td>56</td>
<td>43</td>
<td>56</td>
<td>56</td>
<td>56</td>
</tr>
<tr>
<td>Re-irradiation</td>
<td>No</td>
<td>No</td>
<td>No</td>
<td>No</td>
<td>NSRR</td>
<td>No</td>
<td>JRR-3</td>
<td>No</td>
<td>JRR-3</td>
<td>No</td>
</tr>
<tr>
<td>Max Temp. (K)</td>
<td>2773</td>
<td>2773</td>
<td>3123</td>
<td>2773</td>
<td>3123</td>
<td>2773</td>
<td>3123</td>
<td>2773</td>
<td>3123</td>
<td></td>
</tr>
<tr>
<td>Carrier gas Pressure (MPa)</td>
<td>He 0.1</td>
<td>He 1.0</td>
<td>He 0.1</td>
<td>Steam 0.1</td>
<td>He 1.0</td>
<td>He 0.1</td>
<td>Steam 0.1</td>
<td>He 1.0</td>
<td>Steam 1.0</td>
<td></td>
</tr>
</tbody>
</table>
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

<table>
<thead>
<tr>
<th></th>
<th>PWR-UO$_2$</th>
<th>BWR-UO$_2$</th>
<th>ATR/MOX$^2$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pellet diameter (m)</td>
<td>0.0081</td>
<td>0.0104</td>
<td>0.0124</td>
</tr>
<tr>
<td>Burn-up (GWd/t)</td>
<td>47</td>
<td>56</td>
<td>43</td>
</tr>
<tr>
<td>Theoretical density (%)</td>
<td>95</td>
<td>97</td>
<td>95</td>
</tr>
<tr>
<td>Pu amounts after operation (wt%)</td>
<td>1.1</td>
<td>1.2</td>
<td>2.9</td>
</tr>
<tr>
<td>Linear heat rate during operation (kW/m)</td>
<td>18</td>
<td>26</td>
<td>28</td>
</tr>
<tr>
<td>Temperature during operation (center/periphery, K) $^1$</td>
<td>1000/660</td>
<td>1500/870</td>
<td>1700/900</td>
</tr>
<tr>
<td>FP gas release during operation (%)</td>
<td>0.4</td>
<td>12</td>
<td>20</td>
</tr>
</tbody>
</table>

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

He without cladding 0.1MPa

MOX: VEGA-M1

BWR: VEGA-8

PWR: VEGA-3

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

Cs distribution in pellet diameter

Fuel after irradiation

UO₂ grain

Open pore

Bubble at boundary

Xe, Kr, Cs: remain

Xe, Kr: release

Fuel Temperature (K)

Cs Fractional Release (%)
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 UO₂ grains & pores and process of radionuclide release

1. Diffusion in grains
   - UO₂ grain
   - Grain boundary (pores)

2. Gaseous diffusion in pores
   - Radionuclide
   - Bubble

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

Previous release model (CORSOR-M)

\[ k = k_0 \exp \left( -\frac{Q}{RT} \right) \]

Proposed model with pressure effect (1/√P CORSOR-M)

\[ k = k_0 \sqrt{\frac{P_0}{P}} \exp \left( -\frac{Q}{RT} \right) \quad (P \geq 0.1 \text{MPa}) \]

The multiplier \( \sqrt{\frac{P_0}{P}} \) comes from the pressure dependency of gaseous diffusion flux in pores at pellet surface

Effect of Pressure on Release (4/5)

**Final Releases of Pressure Effect Tests in VEGA**

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Fuel</th>
<th>Test conditions</th>
<th>Fractional release (%)</th>
<th>γ ray measurement (half-life)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>$^{137}$Cs $^{125}$Sb $^{131}$I $^{132}$Te $^{140}$Ba $^{106}$Ru $^{103}$Ru $^{140}$La</td>
</tr>
<tr>
<td>1</td>
<td>PWR</td>
<td>2,773K, 0.1MPa, He</td>
<td>86</td>
<td>89</td>
</tr>
<tr>
<td>2</td>
<td>PWR</td>
<td>2,773K, 1.0MPa, He</td>
<td>61</td>
<td>68</td>
</tr>
<tr>
<td>M1</td>
<td>ATR/MOX</td>
<td>3,123K, 0.1MPa, He</td>
<td>97</td>
<td>95</td>
</tr>
<tr>
<td>M2</td>
<td>ATR/MOX</td>
<td>3,123K, 1.0MPa, He</td>
<td>98</td>
<td>96</td>
</tr>
<tr>
<td>6</td>
<td>BWR</td>
<td>624hr JRR-3 Re-irradiation $^{1/2}$O, 2,773K, 0.1MPa</td>
<td>93</td>
<td>97 98</td>
</tr>
<tr>
<td>7</td>
<td>BWR</td>
<td>624hr JRR-3 Re-irradiation $^{1/2}$O, 2,773K, 1.0MPa</td>
<td>98</td>
<td>83 96 98</td>
</tr>
</tbody>
</table>

- 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**
(with cladding under steam condition)

- 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**

- 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

- Fuel oxidation causes increase in defect in UO₂ grain matrix.
- Contact of liquefied Zr with fuel causes reduction of UO₂ 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.

Pellet oxidation increases in defect in UO₂ grain matrix. Contact of liquefied Zr with fuel causes reduction of UO₂ and decrease in melting point of fuel.

Crucible After Oxidizing Condition Test

- Fuel oxidation causes increase in defect in UO₂ grain matrix.
- Contact of liquefied Zr with fuel causes reduction of UO₂ and decrease in melting point of fuel.

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## Summary of Pressure Effect

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR without cladding (1000/660)</td>
<td>He</td>
<td>&lt; 2300K</td>
<td>O</td>
<td>O</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>&gt; 2300K</td>
<td>O</td>
<td>O</td>
<td>O</td>
</tr>
<tr>
<td>BWR with cladding (1500/870)</td>
<td>Steam + He</td>
<td>&lt; 2300K</td>
<td>X</td>
<td>X</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>&gt; 2300K</td>
<td>X</td>
<td>X</td>
<td>O</td>
</tr>
<tr>
<td>MOX without cladding (1700/900)</td>
<td>He</td>
<td>&lt; 2300K</td>
<td>X</td>
<td>X</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>&gt; 2300K</td>
<td>O</td>
<td>O</td>
<td>O</td>
</tr>
</tbody>
</table>

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$_2$D sequence at 6.95MPa
  - Finally 18% of fuel liquefaction

**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 $1/\sqrt{\text{CORSOR-M}}$
  - 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**

- CORSOR-M
- $1/\sqrt{P}$ CORSOR-M

**Event Timings and Source Terms**

<table>
<thead>
<tr>
<th>Events</th>
<th>TQUX (late CV failure)</th>
<th>TQUX (early CV failure)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(min)</td>
<td>CORSOR-M</td>
<td>$1/\sqrt{P}$ CORSOR-M</td>
</tr>
<tr>
<td>Core melt initiation</td>
<td>53</td>
<td>53</td>
</tr>
<tr>
<td>Vessel failure</td>
<td>313</td>
<td>260</td>
</tr>
<tr>
<td>Pedestal failure</td>
<td>768</td>
<td>745</td>
</tr>
<tr>
<td>Containment failure</td>
<td>1996</td>
<td>2038</td>
</tr>
<tr>
<td>Release fraction to</td>
<td>17.8</td>
<td>14.8</td>
</tr>
<tr>
<td>environment (%)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Source Term in case of early CV failure**

- CV and RPV failure
- Release from molten debris

**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 \text{UO}_2\text{ Grain}**

\[
\frac{\partial}{\partial t} C_1(R,i) = \frac{1}{r^2} \frac{\partial}{\partial r} \left[ r^2 D_1(i) \frac{\partial}{\partial r} C_1(R,i) \right]
\]

(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**

\[
\alpha \frac{\partial}{\partial t} C_2(i) = \frac{1}{R} \frac{\partial}{\partial R} \left[ R \beta' D_2(i) \frac{\partial}{\partial R} C_2(i) \right] + (1-\alpha)\frac{3}{a} D_1(i) \frac{\partial C_1(R,i)}{\partial r} \bigg|_{r=a}
\]

(2)

Diffusion in pellet diameter
FP inflow rate

**Boundary condition:** Continuity of concentration from grain surface to pores

<table>
<thead>
<tr>
<th>Porosity of fuel (-)</th>
<th>( \alpha )</th>
<th>0.05</th>
<th>Grain</th>
<th>Pore</th>
</tr>
</thead>
<tbody>
<tr>
<td>Porosity of open pore (10^{-4}\times)Resistance of diffusion from closed to open pore (-)</td>
<td>( \beta' )</td>
<td>( 1.1\times10^{-6} )</td>
<td>Radial coordinate</td>
<td>( r )</td>
</tr>
<tr>
<td>Radius of \text{UO}_2\text{ grain (m)}</td>
<td>( a )</td>
<td>( 6.0\times10^{-6} )</td>
<td>Concentration(kg/m(^3))</td>
<td>( C_1 )</td>
</tr>
<tr>
<td>Pellet diameter (m)</td>
<td>( L )</td>
<td>( 4.0\times10^{-3} )</td>
<td>Diffusion coeff.(m(^2)/s) Kr @2300K</td>
<td>( D_1 ) (10^{-13})</td>
</tr>
</tbody>
</table>

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

\[ D_1 \cdot \frac{\partial C_1}{\partial R} \bigg|_{R=a} \cdot S_1 \approx D_1 \cdot \frac{C_1 - C_2}{\Delta R} \cdot \frac{V_1}{a} \quad (1) \]

Release rate from pore to fuel outside

\[ D_2 \cdot \frac{\partial C_2}{\partial R} \bigg|_{R=L} \cdot S_2 \approx D_2 \cdot \frac{C_2}{\sqrt{\pi D_2 t}} \cdot S_2 = \frac{D_2}{\pi t} \cdot \frac{V_2}{L} \cdot C_2 \quad (2) \]

\[ k(P) = \frac{\sqrt{D_2}}{\pi t L} \cdot \frac{V_2}{C_2} \quad (3) \]

Inventory

Pressure dependency of \( k \) can be expressed by Eq.(4) using conditions: Eq.(1)=Eq.(2) and Eq.(3)

\[ \frac{k(P)}{k(P_0)} = \sqrt{\frac{P_0}{P}} \cdot \frac{1 + q T^{3/4} \exp(\theta/T)}{1 + q T^{3/4} \exp(\theta/T) \sqrt{P_0/P}} \quad (4) \]

Where \( q \) does not depend on temperature.

\[ (D_1 = D_o \exp(-\theta/T), D_2(P) = dT^{3/2}/P) \]

Under the test conditions, this term \( <<1 \)

\[ \frac{k(P)}{k(P_0)} \approx \sqrt{\frac{P_0}{P}} \quad (5) \]

Existing release model (CORSOR-

\[ k = k_0 \exp\left(-\frac{Q}{RT}\right)^M \]

Proposed model with pressure effect

\[ k = k_0 \sqrt{\frac{P_0}{P}} \exp\left(-\frac{Q}{RT}\right) \quad (P \geq 0.1 \text{MPa}) \]

\[ \sqrt{P_0/P} \quad \text{comes from the pressure dependency of gaseous diffusion flux in pores at pellet surface,} \quad \sqrt{D_2} \quad \text{of Eq.(2)} \]

Study on Diffusion Time in Grain and Pores

**Diffusion Time of Kr in UO₂ Grain and He Gas**

- **Diffusion time in grain**: $a^2/D$
- **Diffusion time in pores**: $\alpha L^2 / \beta' D_g$

- Radius of grain $a = 6 \times 10^{-6} \text{m}$
- Diffusion length in pores $L = 4.025 \text{mm}$
- Total porosity of fuel $\alpha = 0.05$
- Interconnected porosity of fuel including resistance of mass transfer from closed to open pores: $\beta' = 0.0000011$

- $D_g$ calculated by Chapman - Enskog model
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
- Liquefaction of UO$_2$ (UO$_2$+Zr→(U, Zr)O$_2$)

Increase in radiation release

UO$_2$ oxidation
UO$_2$+O$_x$→UO$_{2+x}$

Inert
- Release
- Easier FP transport

Steam
- Increase in defect in UO$_2$ grain matrix due to oxidation
- Increase in release rate
Effect of Pressure on Kr Release from MOX

The graph shows the fractional release of Kr over time for MOX at different pressures. The curves illustrate the release at 0.1 MPa and 1.0 MPa, with the latter showing a more rapid release rate. The y-axis represents the Kr fractional release, while the x-axis shows the time in minutes. The temperature (K) is indicated on the right side of the graph.
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
- Review of Current SAMG
  - Technical Aspects
  - Organizational Aspects
- Group Decision Making in TSC
- An Illustrative Simulation on Group Decision
- Summary
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

<table>
<thead>
<tr>
<th>SAG</th>
<th>Mitigation actions</th>
</tr>
</thead>
<tbody>
<tr>
<td>M-1</td>
<td>Inject into S/G</td>
</tr>
<tr>
<td></td>
<td>• Inject into S/G</td>
</tr>
<tr>
<td></td>
<td>• Depressurize S/G</td>
</tr>
<tr>
<td>M-2</td>
<td>Depressurize RCS</td>
</tr>
<tr>
<td></td>
<td>• Depressurize RCS</td>
</tr>
<tr>
<td></td>
<td>• Depressurize S/G</td>
</tr>
<tr>
<td>M-3</td>
<td>Inject into RCS</td>
</tr>
<tr>
<td></td>
<td>• Inject into RCS</td>
</tr>
<tr>
<td>M-4</td>
<td>Inject into containment (CV)</td>
</tr>
<tr>
<td></td>
<td>• Inject into CV</td>
</tr>
<tr>
<td>M-5</td>
<td>Control fission products releases</td>
</tr>
<tr>
<td></td>
<td>• Depressurize CV</td>
</tr>
<tr>
<td></td>
<td>• Dump steam to condenser</td>
</tr>
<tr>
<td></td>
<td>• Vent Aux. building</td>
</tr>
<tr>
<td>M-6</td>
<td>Control containment conditions</td>
</tr>
<tr>
<td></td>
<td>• Remove heat form CV</td>
</tr>
<tr>
<td></td>
<td>• Depressurize or vent CV</td>
</tr>
<tr>
<td>M-7</td>
<td>Control containment hydrogen</td>
</tr>
<tr>
<td></td>
<td>• Recombine H₂</td>
</tr>
<tr>
<td></td>
<td>• Burn H₂ intentionally</td>
</tr>
<tr>
<td></td>
<td>• Stop active heat sinks in CV</td>
</tr>
</tbody>
</table>
Human and Organizational Aspects of Severe Accident Management

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 H₂ control, deliberate ignition and steam-inert are models not proven.
  • For a reliable control of H₂, 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
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 unresolved 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

- Shift Supervisor
- Reactor Operator
  - Primary Side Field Operators
- Turbine Operator
  - Secondary side Field Operators

Organization of TSC

- TSC Director
- Assistant Operations Team
- Technical Support Team
- TSC Dispatch Team
- Shift Operation Team
- Core Assessment Team
- Commu. Assistant Team
- Admin. Assistant Team
- Op. Assistant Team
- Rad Protection Team
- Radiation Management Team
- Rad Chemical Team
- MCR Operator

Effectiveness of TSC Decision Making has not been evaluated in real and risky severe accident conditions.
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)
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.
The direction of shift depends on the members’ original viewpoints.

- Risk taking attitude ➔ Risky shift
- Risk aversion attitude ➔ Cautious shift

Decision Making Process in TSC

Group Decision

Risky Shift
Risk taking

Cautious Shift
Risk aversion

Individual Decision

Risk

Caution

A B C D E
Human and Organizational Aspects of Severe Accident Management

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 °C
  ▪ 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
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
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

<table>
<thead>
<tr>
<th>Positive Impacts</th>
<th>Negative Impacts</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initiation of Low Pressure Injection</td>
<td>Steam Explosions in RPV</td>
</tr>
<tr>
<td>Prevention of High Pressure Melt Ejection</td>
<td>Loss of RCS inventory due to PZR PORV Use</td>
</tr>
<tr>
<td>Prevention of Creep Rupture of RCS Piping</td>
<td>Containment Overpressure</td>
</tr>
<tr>
<td></td>
<td>Containment Challenge from a Hydrogen Burn</td>
</tr>
<tr>
<td></td>
<td>Fission Product Release from SGs</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th></th>
<th>Case 1</th>
<th>Case 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Accumulator Injection</td>
<td>slow</td>
<td>fast</td>
</tr>
<tr>
<td>Rapid Hydrogen Generation</td>
<td>earlier</td>
<td>later</td>
</tr>
<tr>
<td>Containment Pressure</td>
<td>A little lower</td>
<td>a little higher</td>
</tr>
<tr>
<td>RCS Pressure</td>
<td>Higher but below set point</td>
<td>Lower</td>
</tr>
<tr>
<td>Initial Vessel Breach</td>
<td>4.2 hr</td>
<td>9.1 hr</td>
</tr>
<tr>
<td>Pressure at RPV Failure</td>
<td>16.7 MPa</td>
<td>0.75 MPa</td>
</tr>
<tr>
<td></td>
<td>0.39 MPa</td>
<td></td>
</tr>
</tbody>
</table>

- **RCS Pressure**
  - Case 1: Low Depressurization Capacity
  - Case 2: Medium Depressurization Capacity
  - Case 3: High Depressurization Capacity

- **H2 Generation in Core**

- **Containment Pressure**
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?
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 internal 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’

- Spontaneous RPV rupture
- 1+ g Seismic events
- etc.

Refinement in model & knowledge

Implementation of SAMGs

Advanced Rx (Gen III) designs

- Spontaneous RPV rupture
- 1+ g Seismic events
- … etc.

Release Frequency

10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}

1980 1990 2000 2010
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 not 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