Session II-B
PSA Softwares
This is a blank page.
Dr. Jeong presented “Development of a PSA Tool for an Interim Storage Facility of Spent Fuels.” In Korea, the spent fuel will be stored in the centralized interim storage facility until the national policy for the spent fuel is determined, probably at the end of 2009. The PSA is selected as the safety assessment methodology of the facility in the design and operation stage and the development plan for the PSA tool are described in detail. The PSA tool will be used for (1) safety assessment of the operational phase of the facility; (2) checking the compliance with the performance objective of the facility; and (3) design modification by identifying SSCs important to safety. Importance of obtaining the reference data for the PSA tool development and the selection of the complete initiating event list are discussed.

Dr. Homma of JAEE presented “Development of GSALab Computer Code for Global Sensitivity Analysis (GSA).” GSA is a very important technique for identifying the effect of uncertain inputs on the uncertainty of the model output. The GSALab computer code is developed for this purpose. The GUI design of GSALab makes it very easy to use and to see the results. It seems that GSALab is the first software which aims at comparing different GSA methods. The updating of GSALab is now being in process. From the floor, there is a question concerning the availability of the software. Currently being in the development stage, it will be a free open software after the completion.

Dr. Choi presented “Development of condition monitoring and diagnosis system for standby diesel generator.” Condition monitoring and diagnosis system has been developed for standby diesel generator at Wolsung nuclear unit 3&4. Condition monitoring module has functions of real-time monitoring for operation parameter’s condition status on 8 systems. It can also provide operating key parameters to performance monitoring function of EDG reliability program. Diagnosis module provided information on causes of function failure prediction of abnormal condition based on rule base and inference engine in JESS computer program. This system is expected to help standby diesel generator to improve the reliability of SDG. A question was raised if it can be applied to other NPPs. Essentially there is no problem in broader application other than Wolsung plant.

Mr. Kim presented “Development of a Web-based Risk Monitoring System.” They are developing web-based risk monitoring system in order to make it easy to share the risk information. After developing W-RIMS, ORION, SPV will be integrated in a couple of years.

Mr. Oh presented “The Pilot Application of ORION Program.” They applied ORION program to evaluate and manage risk during refueling outage. They found that the ORION program is useful to maintain or reduce the shutdown risk.
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Development of a PSA Tool for an Interim Storage Facility of Spent Fuels

May. 18 2009
Jongtae Jeong

Radioactive Waste Technology Development Division
Korea Atomic Energy Research Institute

Contents

I Introduction
II Development Plan for the PSA Tool
III Concluding Remarks
Introduction

- The nuclear power plays an important role in Korea.
- The problem of the radioactive waste including the spent fuels for the sustainable development of nuclear power industry.
- The KRMC (Korea Radioactive Waste Management Corporation) was established in January 1 of 2008.
  - Transport, storage, treatment, and disposal of radioactive waste
  - Site selection, construction, operation, and management of disposal facility
  - Transport and interim storage of spent fuel
- The first R&D program organized by the KRMC:
  - A Technology development for the transport and storage system for the spent fuels

Nuclear Power Plants in Korea

<table>
<thead>
<tr>
<th>Site</th>
<th>In Operation</th>
<th>Under Const.</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kori</td>
<td>4(3,137)</td>
<td>2(2,000)</td>
<td>a</td>
</tr>
<tr>
<td>Wolsong</td>
<td>4(2,779)</td>
<td>2(2,000)</td>
<td>6(4,779)</td>
</tr>
<tr>
<td>Yonggwang</td>
<td>6(5,900)</td>
<td>-</td>
<td>6(5,900)</td>
</tr>
<tr>
<td>Ulchin</td>
<td>6(5,900)</td>
<td>-</td>
<td>6(5,900)</td>
</tr>
<tr>
<td>Total</td>
<td>20(17,716)</td>
<td>4(4,000)</td>
<td>24(21,716)</td>
</tr>
</tbody>
</table>

Units (MW), As of Dec. 2006
Introduction

Spent Fuel Stored in NPPs

As of December 2008 (unit: tU)

<table>
<thead>
<tr>
<th>Plant Site</th>
<th>Capacity</th>
<th>Accumulation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kori</td>
<td>2,253</td>
<td>1,685</td>
</tr>
<tr>
<td>Yong-gwang</td>
<td>2,686</td>
<td>1,623</td>
</tr>
<tr>
<td>Ulchin</td>
<td>2,328</td>
<td>1,294</td>
</tr>
<tr>
<td>Wolsong</td>
<td>5,980</td>
<td>5,481</td>
</tr>
<tr>
<td>Total</td>
<td>13,247</td>
<td>10,083</td>
</tr>
</tbody>
</table>

- Resolution of the insufficient spent fuel storage capacity of NPP
  - Expansion of the site storage capacity by re-racking and transshipment (PWR)
  - Transshipment to the dry concrete silo since 1991 (CANDU)
- Long-term National Policy for the SF management
  - Will be established based on the results from R&D program and public involvement
  - Store and manage until 2016

Introduction

Establishment of system concept

Safe transport and storage of SF

Basic technology development

- DB for Spent Fuels
- Burn-up Effect
- Test Facility

Safety and long-term integrity

- Integrity of Spent Fuels
- Integrity of Structures
- Safety Assessment (PSA)

The purpose of this research is to develop a PSA tool for the interim storage facility of spent fuels in Korea
Development Plan for the PSA Tool

Collect Information for Safety Analysis

- Site Description
- Facility Design and Operations
- Waste Stream
- Natural and Human-induced Hazard

PSA Sub-modules Performing Specific Analysis

- Operational Hazard Analysis
- Event Sequence Frequency Analysis
- Public and Worker Radiological Case Consequence Analysis

Application Area

- Regulatory compliance Analysis
- Total System Risk Analysis
- Structures, Systems, and components Important to Safety Analysis

Conceptual Module Structure for the PSA Tool

Identify and verify sufficient information to perform safety analysis in the form of a database

Subjects

- Site specific data
  - Meteorology, Population distribution
- Facility data
  - Description and design details of structures, systems, and components
  - Characterization of spent fuels and source terms
  - Description of facility operational processes and procedures
  - Understanding of the component and facility functions and sequence of operations.
Development Plan for the PSA Tool

Naturally occurring and human-induced hazards

Identify hazards for the identification of initiating events

Subjects

- Site-specific, naturally occurring hazard
  - Seismic events
  - High winds
  - Lightning Strikes
- Site-specific, human-induced hazard
  - Mechanical events such as drop
  - Thermal events such as fire

Operational Hazard Analysis

- Identify initiating events
- The what-if analysis will be used for identifying process hazards
- The FMEA or HAZOP addresses hardware and equipment failures
**Development Plan for the PSA Tool**

**Event Sequence Analysis**
- Event scenarios will be developed using the ETA method.
- FTA method will be used to find failure probabilities for each node in the ET.
- FTREX tool will be used in the event sequence analysis.

**Consequence Analysis**
- Calculates the radiological doses to the members of the public and workers.
- Point estimate or probabilistic calculations can be performed for the public dose determination.
- Point estimate calculations of the worker dose will be also available.
In Korea, the spent fuel will be stored in the centralized interim storage facility until the national policy for the spent fuel is determined.

The safety assessment is very important during the operation of the facility as well as in the design stage.

We adopted the PSA as a safety assessment methodology of the facility.

We presented the development plan for the PSA tool which can be applied to the safety assessment of the interim storage facility of spent fuels in Korea.

The PSA tool will be used:
- Safety assessment of the operational phase of the facility.
- Checking the compliance with the performance objective of the facility.
- Design modification by identifying SSCs important to safety.
감 사 합 니 다!
Development of GSALab Computer Code for Global Sensitivity Analysis

Qiao LIU, Toshimitsu HOMMA
Nuclear Safety Research Center
Japan Atomic Energy Agency (JAEA)

Background

- Uncertainty is an integral part of risk assessment problems (Apostolakis, 1990).
- Given a PSA model $Y = g(X_1, X_2, \ldots X_n)$.
Sensitivity Analysis (SA)

- **Sensitivity Analysis (SA):**
  - It is "the study of how the uncertainty in the output of a model (numerical or otherwise) can be apportioned to different sources of uncertainty in the model input" (Saltelli et al, 2004).

- **Why SA is needed?**
  - For identifying main factors that should be focused on
    - Identify main contributors to output uncertainty, in which resources should be invested.
  - For model simplification
    - For those input parameters contributing little to output uncertainty, they can be fixed at any value within their distributions.

Local SA Method vs Global SA Method

- **Local SA method:**
  - Interested in some kind of derivative of the model output with respect to the model input.
  - \( S_i = \frac{\partial Y}{\partial X_i} \)
    - The model output itself is focused.
    - Only small perturbation of the input under study is permitted, while keeping others at nominal values.
    - Effective only if the model is linear.

- **Global SA method (GSA):**
  - Interested in assessing the effect of the entire distribution of the model input on the output uncertainty.
    - The entire distribution of the model input is considered.
    - The contribution of the uncertainty in a model input is evaluated with the values of all the other inputs varying as well.
GSA Methods (1)

- Variance-based methods
  - First Order Sensitivity Index (Sobol', 1990)
    \[ S_1 = \frac{\mathbb{E}(\mathbb{E}(Y|X_{\neq i}))}{\mathbb{E}(Y)} \]
  - Total Effect Sensitivity Index (Homma & Saltelli, 1996)
    \[ S_T = \frac{\mathbb{E}(\mathbb{E}(Y|X_{\neq i}))}{\mathbb{E}(Y)} \]

- Distribution-based methods
  - Liu & Homma method (2007)
    \[ S_X = \frac{E_X(A_X)}{|E(Y)|} \]
    Where
    \[ E_X(A_X) = \int f_X(x_i) \cdot A_X(x_i) \, dx_i \]
    \[ A_X = \int [F_{Y|X_{\neq i}}(y) - F_Y(y)] \, dy \]

GSA methods (2)

- Distribution-based methods
  - Borgonovo method (2007)
    \[ \delta_i = \frac{1}{2} E_X[s(X_i)] \]
    Where
    \[ E_X[s(X_i)] = \int f_X(x_i)s(X_i) \, dx_i \]
    \[ s(X_i) = \int f_Y(y) - f_{Y|X_{\neq i}}(y) \, dy \]
  - Park & Ahn method (1994)
    \[ I(i:o) = \int_{R(x)} f_i(x) \ln \left( \frac{f_i(x)}{f_0(x)} \right) \, dx \]
  - Chun, Han & Tak method (2000)
    \[ MD(i:o) = \left( \int (y'_i - y'_p)^2 \, dp \right)^{1/2} \frac{E(Y^0)}{E(Y^p)} \]

- Other methods, such as entropy...
Development of GSALab Code

- **Objective**
  - Implementing the state of the art GSA methods in one computer code to compare them to each other

- **Development Environment**
  - CodeGear C++ Builder 2007

- **Functions**
  - **Random Samples Generation**
    - Uniform, log-uniform, normal, lognormal, triangle, …
  - **Uncertainty Analysis**
    - Statistics (eg, mean, median, variance, …)
    - CDF, PDF
  - **Sensitivity Analysis**
    - Calculate and rank the uncertainty importance of model inputs using different GSA methods.

Random Sample Generation

- Generate samples of model inputs based on Monte Carlo method

(Tabulated Sample values)
Model Edition and Calculation

- Input model of interest and calculate model output by using generated samples of model inputs.

Uncertainty Analysis

- Calculate CDF(PDF), mean, median, variance, ...
Sensitivity Analysis

Calculate the importance of each model input based on different GSA methods.

( Ranking of Model Inputs )

Tabulated Values of the Importance of Each Model Input

Linking with External Model

For complex models, the user should:

1. Generate the samples of model inputs using GSALab;
2. Calculate the model output by using the generated samples of model input;
3. Read the model outputs and input samples into GSALab to perform uncertainty analysis and sensitivity analysis.
Summary

- GSA is a very important technique for identifying the effect of uncertain inputs on the uncertainty of the model output.
- The GSALab computer code is developed for performing GSA. Multiple GSA methods have been implemented in GSALab.
- The GUI design of GSALab makes it very easy to use and to see the results.
- To our knowledge GSALab is the first software which aims at comparing different GSA methods.
- The updating of GSALab is now being in process.
Development of condition monitoring and diagnosis system for standby diesel generator

KJPSA Joint Workshop
2009. 5. 18~20
Kwanghee Choi, Jonghyuck Park, Jongsun Park

Korea Electric Power Research Institute

Contents

- Background
- Condition monitoring and diagnosis system
  - Condition monitoring module
  - Diagnosis module
  - Relationship with Reliability Program
- Conclusion
Introduction

Background

- The Emergency Diesel Generator (EDG) of the nuclear power plant is designed to supply the power to the nuclear reactor on Station Black Out (SBO) condition.
- The reliability program of EDG had been developed and applied on Korean nuclear power plant from 2005. But condition monitoring and diagnosis system was not developed.
- Performance monitoring function of reliability program has only trending display by manual input.
- The operation reliability of onsite emergency diesel generator should be ensured by real time condition monitoring system designed to monitor and analysis the condition of diesel generator.

Introduction

Background

- For this purpose, we have developed the online condition monitoring and diagnosis system for the wolsong unit 3&4 standby diesel generator including diesel engine performance.
- In this paper, technologies of condition monitoring and diagnosis system (SDG MDS) for the wolsong standby diesel generator are described.
- By using the condition monitoring module of the SDG MDS, performance monitoring function for major operating parameters of EDG reliability program required by Reg. guide 1.155 can be operated as on line monitoring system.
Condition monitoring and Diagnosis system

- A. Data acquisition System (DAS)
- B. Condition Monitoring and Diagnosis System Sever
- C. E-Book
- D. Engine condition monitoring system
Condition monitoring module

- Condition monitoring module composed of
  - Monitoring function
  - Trending function
  - Monitoring function
- Operating parameters stored in system server through DAS are total 161 data included 77 alarm signals as shown in Table

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Operating parameter</th>
<th>Alarm parameter</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel oil</td>
<td>1</td>
<td>5</td>
</tr>
<tr>
<td>Lub oil</td>
<td>3</td>
<td>9</td>
</tr>
<tr>
<td>Low Temp Cooling</td>
<td>4</td>
<td>7</td>
</tr>
<tr>
<td>Hot Temp Cooling</td>
<td>18</td>
<td>7</td>
</tr>
<tr>
<td>Engine</td>
<td>22</td>
<td>18</td>
</tr>
<tr>
<td>Generator</td>
<td>10</td>
<td>6</td>
</tr>
<tr>
<td>Exhaust/Intake air</td>
<td>25</td>
<td>0</td>
</tr>
<tr>
<td>Starting/Control air</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>Etc.</td>
<td>0</td>
<td>22</td>
</tr>
<tr>
<td>Sub-total</td>
<td>84</td>
<td>77</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td><strong>165</strong></td>
</tr>
</tbody>
</table>

Condition monitoring module

- Input data from alarm module composed of
  - 77 alarm input data
  - 4 combination data
- This alarm monitoring function
  monitor all alarm and logic signal data status in one screen as “Trip, Interlock, Alarm, Control, and Status” shown on below Table.

<table>
<thead>
<tr>
<th>Category</th>
<th>Signals</th>
<th>Display</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Single Input</td>
<td>Combination</td>
</tr>
<tr>
<td>1 Trip</td>
<td>13</td>
<td>3</td>
</tr>
<tr>
<td>2 Interlock</td>
<td>8</td>
<td>1</td>
</tr>
<tr>
<td>3 Alarm</td>
<td>35</td>
<td></td>
</tr>
<tr>
<td>4 Control</td>
<td>13</td>
<td></td>
</tr>
<tr>
<td>5 Status</td>
<td>8</td>
<td></td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>77</td>
<td>4</td>
</tr>
</tbody>
</table>
Condition monitoring module

- Condition monitoring module (sample)

![Condition monitoring module](image)

Condition monitoring module

- Alarm monitoring module

![Alarm monitoring module](image)
Condition monitoring module

- Condition monitoring module (Trending function)

This module is developed by JESS (Java Expert System Shell) program.

It is required to develop the diagnosis module that bring the data of condition monitoring module by DDE (Dynamic Data Exchange) communication from condition monitoring.

Data used in diagnosis module is real time data from condition monitoring module as average values calculated in three times.

Diagnosis module is divided into functional failure diagnosis and prediction diagnosis.

This average value is used in functional failure diagnosis for distinguished condition of alarm as HH (High High), H (High), N (Normal), L (Low), LL (Low Low).
Diagnosis Module

- The process of diagnosis module is composed of working memory, inference engine and rule base as shown on below figure.

Diagnosis module has 565 rules for diagnosis on symptom

<table>
<thead>
<tr>
<th>System</th>
<th>Rule for Diagnosis of Operation Parameter</th>
<th>Rule for Diagnosis of Alarm</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Simple factor</td>
<td>Combination factor</td>
</tr>
<tr>
<td>Fuel Oil System</td>
<td>1</td>
<td>4</td>
</tr>
<tr>
<td>Lubricating Oil System</td>
<td>5</td>
<td>2</td>
</tr>
<tr>
<td>Low Temperature Cooling Water</td>
<td>6</td>
<td>12</td>
</tr>
<tr>
<td>High Temperature Cooling Water</td>
<td>10</td>
<td>123</td>
</tr>
<tr>
<td>Engine</td>
<td>30</td>
<td>39</td>
</tr>
<tr>
<td>Generator</td>
<td>12</td>
<td>-</td>
</tr>
<tr>
<td>Exhaust Gas &amp; Intake Air</td>
<td>74</td>
<td>150</td>
</tr>
<tr>
<td>Starting &amp; Control Air</td>
<td>1</td>
<td>-</td>
</tr>
<tr>
<td>Status etc.</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Sum</td>
<td>110</td>
<td>340</td>
</tr>
<tr>
<td>Total</td>
<td>565</td>
<td></td>
</tr>
</tbody>
</table>
Relationship with reliability program

- Reliability program of EDG composed of performance monitoring module that has trending function on major operation parameters required by IEEE 387.
- Performance monitoring data of reliability program is almost covered by condition monitoring module’s data.
- The automatic input program for connecting data between monitoring system and reliability program of EDG at interval of 30 minutes, 60 minutes, 90 minutes, 120 minutes after startup has developed.

Conclusion

- Condition monitoring and diagnosis system has developed for standby diesel generator at Wolsung nuclear unit 3&4.
- Condition monitoring module has functions of real-time monitoring for operation parameter’s condition status on 8 system by shown as numbers, bar-chart and trending diagrams.
- Condition monitoring module can also provide operating key parameters to performance monitoring function of EDG reliability program.
- Diagnosis module provided information on causes of function failure and prediction of abnormal condition based on rule base and inference engine in JESS computer program.
- This system is expected to help standby diesel generator to improve the reliability of SDG.
KJPSA, Jeju
May 18-19, 2009

W-RIMS Development

Myungsu Kim
pro@khnp.co.kr

Korea Hydro & Nuclear Power
Nuclear Engineering & Technology Institute

Risk Monitor  RIMS  W-RIMS  Future plans
Risk Monitor
Scheduling with the risk information

Risk Monitoring for the current configuration
Development for all 20 operating NPPS

2003 ~ 2007
e-P&ID

RIMS → W-RIMS

Web interface
- Easy to share the risk information → Risk culture

ERP connection
- Enhance the data integrity
- No need to input maintenance data only for RIMS

More Functions
- e-P&ID, Engineering Work Station, Plant Information...
- Documentation, e-mail notification...
Integration for the Risk monitoring systems

Keywords for the successful W-RIMS implementation

MR
OLM
RIR
RIA
Q&A

Development of a web-based risk monitoring system in Korea

Myungsu Kim

Thank you
Pilot Application of ORION program

2009. 5.17

Hae-Cheol Oh (haech@kepri.re.kr)
Myung Ki Kim
Bag Soon Chung
Miro Seo

Contents

I. Introduction
II. ORION Description
III. Qualitative Assessment Model during Refueling Outage
IV. Results of Application
V. Conclusions
I. Introduction

- The nuclear utility in Korea implemented the many outage R&D activities

- The objective of the shutdown R&D effort is to provide technical products that address near-term & long term needs for enhancing refueling outage management

- The R&D activities include the Revision of Abnormal Procedures, the Development of Planned Outage Process Standardization, and Software Development such as ORION (Outage Risk Indicator of NPP) and NPOMS (Nuclear Power Plant Outage Management System)

II. ORION Description

- In 2007, KEPRl developed the ORION Program to evaluate and manage risk during shutdown operations

- This program has the capability to provide the plant staff and management personnel with understandable results of deterministic evaluations of plant safety

- One major element of this program is the interpretation of risk results and insights by a series of colors and associated guidance

- ORION identifies the logical flow paths for a selected safety function as part of guidelines for managing risks during shutdown operations
II. ORION Description

 Scheduler Interface

- POS
- Order
- Maintenance Schedule
- Overall Risk
- Safety Function Status
- Duration Selection
- Multiple O/H Schedule
- Information Showing

 Maintenance Schedule Import from DREAMS(ERP)

- ORION Work Order
- ORION Work Order Automatic or Manual Selection

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II. ORION Description

- Tracing Function

III. Qualitative Assessment Model during Refueling Outage

- The ORION model using the qualitative Defense-In-Depth (DID) methods given in NUMAR 91-06
  - 8 SFATs (Safety Function Assessment Trees) for key safety functions of plant are used, the risk is considered acceptable when key safety functions and plant activities are managed
    - Reactivity control, Decay Heat Removal, Inventory control
    - SFP cooling, AC & Dc power, Cooling water, Containment integrity
  - The result of safety function assessment is a set of colors indicating the level of DID (the safety margin)
  - The level of DID is decided with only the number of mitigating system or train
III. Qualitative Assessment Model during Refueling Outage

◆ The following color definitions are typical for a SFAT

DID Index

<table>
<thead>
<tr>
<th>Color</th>
<th>Definition</th>
</tr>
</thead>
<tbody>
<tr>
<td>Green</td>
<td>&gt; N+1 Full Support</td>
</tr>
<tr>
<td>Yellow</td>
<td>N+1 Adequate Support</td>
</tr>
<tr>
<td>Orange</td>
<td>N Reduced Support</td>
</tr>
<tr>
<td>Red</td>
<td>N-1 Minimum Support</td>
</tr>
</tbody>
</table>

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III. Qualitative Assessment Model during Refueling Outage

Decay Heat Removal (DHR) SFAT

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III. Qualitative Assessment Model during Refueling Outage

Safety Function Color Basis Document

- Defense-In-Depth Evaluation Criteria at POS B (DHR SFAT)
- Defense-In-Depth Evaluation consider the Redundancy and Diversity of Safety Function Tree

<table>
<thead>
<tr>
<th>SYSTEM</th>
<th>Steam Generator (Including MBBV, MSAV)</th>
<th>High Pressure Safety Injection</th>
<th>Safety Color</th>
<th>Defense-In-Depth Index</th>
<th>(Description of OID Index)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 1</td>
<td>2</td>
<td>2</td>
<td>GREEN</td>
<td>4</td>
<td>(4=SG 2 Train + HPSI 2 Train)</td>
</tr>
<tr>
<td>Case 2</td>
<td>2</td>
<td>1</td>
<td>YELLOW</td>
<td>3</td>
<td>(3= SG 2 Train + HPSI 1 Train)</td>
</tr>
<tr>
<td>Case 3</td>
<td>2</td>
<td>0</td>
<td>RED</td>
<td>1</td>
<td>(2= SG 2 Train + HPSI 0 Train)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>(OID Index=1 (-1 due to loss of diversity))</td>
</tr>
<tr>
<td>Case 4</td>
<td>1</td>
<td>2</td>
<td>YELLOW</td>
<td>3</td>
<td>(3= SG 1 Train + HPSI 2 Train)</td>
</tr>
<tr>
<td>Case 5</td>
<td>1</td>
<td>1</td>
<td>ORANGE</td>
<td>2</td>
<td>(2= SG 1 Train + HPSI 1 Train)</td>
</tr>
<tr>
<td>Case 6</td>
<td>1</td>
<td>0</td>
<td>RED</td>
<td>0</td>
<td>(1= SG 1 Train + HPSI 0 Train)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>(OID Index=0 (-1 due to loss of diversity))</td>
</tr>
<tr>
<td>Case 7</td>
<td>0</td>
<td>-</td>
<td>RED</td>
<td>0</td>
<td>Primary Heat Sink is not available (SG should be available to enter the next POS)</td>
</tr>
</tbody>
</table>

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IV. Results of Application

❖ POS Define

O/H Block Schedule

Evaluation Results for the Planned Schedule

Evaluation Results for the Actual O/H Schedule
IV. Results of Application

- Association with NPOMS (Real-Time Schedule Management)

V. Conclusions

- With clearly defined safety functions and the methods which can evaluate the level of defense-in-depth according to maintenance schedules, low power and shutdown risk in nuclear power plant can be managed more easily and more objective way.

- The ORION (Outage Risk Indicator of NPP) program will be useful to maintain or reduce the shutdown risk as shorter outage are implemented.
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Session Summary II-C: Severe Accident Management & Level -3 PSA (II)

Chair: Mitsuhiro KAJIMOTO (JNES), JinHo SONG (KAERI)

Five papers were presented and discussed about Severe Accident Management (SAM) and Level-3 PSA in this session.

II-C-1. Mitsuhiro KAJIMOTO (JNES-Japan): Severe Accident and Accident Management Study at JNES
This presentation focused on the applications of the remarkably progressed Computer Fluid Dynamics (CFD) to the Severe Accident and Accident Management analysis. Use of CFD for the analysis of the current issues in Severe Accident Management Guideline or Guidance (SAMG) Studies was discussed.

II-C-2. Tae-hyeong KIM (KINS-Korea): Recent Regulatory Activities Related to Severe Accident Management in Korea
“Policy on Severe Accident of NPPs,” in 2001, stated “An owner of nuclear power reactor should establish and implement severe accident management programs.” This presentation introduced the recent regulatory activities related to SAM in Korea. Some weaknesses of strategies due to the plant design features in the current SAMG for plant specific applications were discussed.

II-C-3. Tao LIU (INET-China): Consideration of Emergency Source Terms for Pebble-bed High-Temperature Gas-Cooled Reactor
A pebble-bed high-temperature gas-cooled power plant as an advanced Nuclear Power Plant is being constructed in China. Insights on emergency source terms selection obtained from the PSA and pebble-bed HTGR emergency source term suggestion were proposed and discussed.

II-C-4. Do Sam KIM (KINS-Korea): Probabilistic Estimation of the Early and Cancer Fatality Risks at the Korean Nuclear Power Plants
As large early release frequency (LERF) could be used as a measure of early fatality, four LERF definitions or evaluation methodologies were presented and discussed to estimate early fatality.

II-C-5. JinHo SONG (KAERI-Korea): Implementation of a Molten Core Cooling Strategy in a Severe Accident Management Guideline
KAERI and KINS researched jointly the molten core cooling strategy in SAMG. Suggestion for the improvements of SAMG were discussed.
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Severe Accident and Accident Management Study at JNES

Masao Ogino and Mitsuhiro Kajimoto
Severe Accident Evaluation Group
Nuclear Safety Analysis and Evaluation Office
Nuclear Energy System Safety Division
Japan Nuclear Energy Safety Organization

The 10th KJPSA Workshop, May 18-20, 2009
Haevichi Hotel & Resort, Jeju, Korea

Outline

Background
- Severe Accident Research Activities
- Mitigative Accident Management
- CFD Severe Accident Analysis
  3.1 PSI-ARTIST Analysis
  3.2 CFD Application to RCS analysis
  3.3 CFD Application to CV analysis

Summary and conclusions
The application area of risk-informed regulation has been expanded and used to improve safety decision making and improve regulatory effectiveness in Japan.

For those applications, severe accident codes with higher quality and reliability are needed.

JNES has been using lumped parameter code for integrated severe accident analysis, and CFD code for local mechanistic analysis to complement the lumped parameter code.

This presentation focuses on the applications of the remarkably progressed CFD to the SA/AM analysis.

1. Severe Accident Research Activities

PSA/SA/AM analysis, Risk informed regulation

- Lumped parameter codes
  - Model validation with experimental data and CFD analysis

- CFD Applications
  - To complement lumped parameter codes

- OECD Projects
  - OECD MCCI-2
  - OECD SETH-2
  - OECD ROSA
  - OECD SERENA
  - OECD BIP

- Cooperative Projects
  - PSI ARTIST
  - NRC CSARP
  - PHEBUS-FP
  - JAEA etc.

- Computational fluid dynamics (CFD)
  (Mechanistic analysis)

- Resolutions of SA key phenomena (FP behavior, core debris coolability, containment integrity, etc.) which significantly affect on environment.

- Mechanistic analysis with computational fluid dynamics (CFD) to complement experimental data for plant analysis.
International cooperative experimental research for key phenomena and modeling

<table>
<thead>
<tr>
<th>Items</th>
<th>Cooperation Projects</th>
<th>Objectives</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fission product behavior</td>
<td>PSI ARTIST (with PSI)</td>
<td>Aerosol deposition in the broken steam generator</td>
</tr>
<tr>
<td></td>
<td>Iodine Experiments (with JAEA)</td>
<td>Iodine behavior in late phase severe accident</td>
</tr>
<tr>
<td></td>
<td>OECD BIP</td>
<td>Iodine behavior in containment</td>
</tr>
<tr>
<td></td>
<td>PHEBUS-FP</td>
<td>Integrated experiment for FP behavior in severe accident</td>
</tr>
<tr>
<td>Debris / water interaction</td>
<td>OECD MCCI</td>
<td>Debris-concrete interaction</td>
</tr>
<tr>
<td></td>
<td>OECD MASCA</td>
<td>In-vessel thermal behavior of debris</td>
</tr>
<tr>
<td></td>
<td>OECD SERENA</td>
<td>Debris-coolant interaction (steam explosion)</td>
</tr>
<tr>
<td>Containment behavior</td>
<td>OECD SETH2</td>
<td>Thermal hydraulics in the containment</td>
</tr>
<tr>
<td>Out side containment</td>
<td>OECD SFP</td>
<td>LOCA (fuel heat up and zirconoy ignition) at storage fuel pool.</td>
</tr>
</tbody>
</table>

2. Mitigative Accident Management

(Example of PWR)
Effectiveness of SAM to prevent CV failure

- With the natural convection containment vessel cooling SAM, the frequency of the overpressure sequence decreased to one-twelfth.

\[
\begin{align*}
\text{Total Reduction} &= 1/5 \\
6.9E-08 &\quad 1.5E-08
\end{align*}
\]

**Containment Failure Mode** (Dry large CV PWR)

3. CFD Severe Accident Analysis

**Needs of CFD Severe Accident Analysis Models**

- Containment bypass (SGTR-SA) has a large risk of LER
- CV cooling by natural convection is an important SAM

- To evaluate the detail of flow velocity field and FP aerosol deposition in the broken SG
- To evaluate the detail of flow velocity field and temperature distribution in the RCS
- To evaluate the detail of the multi-compartment (space) effects.
3.1 PSI-ARTIST Analysis

SG secondary CFD model

ARTIST Separator and Dryer

Dryer (CAD data)  Dryer panel (Shevron vanes)

Separator (CAD data)  Swirl vane
CFD model of separator & dryer section
(Base Model)

- 3D model with 1,730,000 hexagonal and tetra meshes.
- Shevron vanes: parallel flat panels.
- Maximum mesh size: 13 mm.

Velocity fields in Separator & Dryer

- Detail of flow velocity field at separator and dryer.
- Small down flow through lid and higher velocity at the outlet of the separator.

Flow velocity (m/s)

- Inlet mass flow rate: 600 kg/h
**Flow velocity at outlet side of separator**

- Flow simulation by CFD model well agreed with the experimental data at MP4 90deg.

**Flow velocity (MP4 90deg)**

**Tracking of droplets in Dryer Panel**

- Droplet diameter: 30 $\mu$m
- Inlet flow velocity: 0.1 m/s (at the inlet of the panel)
- All droplets were trapped on the projected walls.
3.2 CFD Application to RCS analysis

RCS CFD model

- Three dimensional RCS thermal hydraulics and aerosol behavior
- Application of ARTIST CFD methodology to the RCS analysis base on the experience in the ARTIST project.

<table>
<thead>
<tr>
<th>RCS CFD model</th>
<th>System</th>
<th>One loop model</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mesh model</td>
<td>870 k meshes</td>
</tr>
<tr>
<td></td>
<td>Turbulence flow model</td>
<td>k-ε model</td>
</tr>
<tr>
<td>Initial conditions</td>
<td>Pressure</td>
<td>157 kg/cm²</td>
</tr>
<tr>
<td></td>
<td>Temperature</td>
<td>800 K</td>
</tr>
<tr>
<td>Boundary conditions</td>
<td>SG</td>
<td>Temperature: 600 K</td>
</tr>
<tr>
<td></td>
<td>Core</td>
<td>Power: 120000 W/m³</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Thermal conductivity: 1000 W/m·K</td>
</tr>
</tbody>
</table>
High temperature RCS natural convection (1/2)

- Application to the reflux cooling mode during a severe accident
- Prediction of the multi-dimensional flow velocity field and temperature distribution during a reflux cooling mode
- Coolant temperature at SG tube was maintained very low which prevented the induced SGTR.

Reflex cooling mode during a SA

High temperature RCS natural convection (2/2)

- High temperature RCS natural convection during a severe accident.
- SAM of the reinforced depressurization with pressurizer relief valve
- High temperature gas flow into SG tube from RV was mitigated by the depressurization with the pressurizer relief valve forced opening

SAM: Forced opening of Pz relief valve during a high temperature RCS natural convection

RCS natural convection
(800 sec after forced open of Pz relief valve)
3.3 CFD Application to CV analysis

Containment vessel CFD model

Containment cooling SAM by natural convection

- Containment vessel cooling by natural convection during a CV spray failure.
- SAM of supplying the component cooling water to the air-conditioning cooler

Ex. BWR cooler

PWR SAMs for mitigation
Containment vessel CFD model

- Containment vessel of PWR plant consists of multi-compartment (~24).
- FLUENT 3D model
  - 50,000 tetra meshes
  - $K-\omega$ turbulence model
  - Steam condensation model
  - Aerosol tracking model

Applied the “containment vessel CFD model” to study the SAM of PWR.
(SAM: containment natural convection cooling by recirculation cooler)

Thermal hydraulic analysis case and aerosol tracking case

<table>
<thead>
<tr>
<th>Case</th>
<th>Injection point (LOCA location)</th>
<th>Initial conditions</th>
<th>Injection flow conditions</th>
<th>Note</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Pressure/Temperature (Air/steam/H₂)</td>
<td>Mass fraction (steam/H₂)</td>
<td>Mass flow rate (steam/H₂)</td>
</tr>
<tr>
<td>1</td>
<td>SG compartment</td>
<td>0.49MPa/413K</td>
<td>0.26/0.74/0.0</td>
<td>10kg/s/0.0kg/s</td>
</tr>
<tr>
<td>2</td>
<td>↑</td>
<td>↑</td>
<td>↑</td>
<td>↑</td>
</tr>
</tbody>
</table>
Flow pattern in the containment vessel

Steam gas, injected into a lower compartment, mixed almost completely with the air in whole CV by natural convection.

Temperature distribution in containment vessel

Atmospheric temperature was almost homogeneous due to natural convection.

Effective cooling was achieved by CV recirculation cooler.
Aerosol behavior in containment vessel

- Released aerosol particles (FP) are locally deposited on the floors near walls and duct outlet of the cooler in the containment vessel.

Aerosol concentration
(1000 sec, AMMD=1 μm)

Deposited aerosol distribution
(1000sec, AMMD=1 μm)

Summary and conclusions

- JNES developed a CFD flow simulation approach and aerosol tracking model of SG separator and dryer in the work of the cooperated PSI-ARTIST project.

- JNES applied the above CFD methodology to the RCS and the containment vessel to discuss the detail of the SA phenomena and evaluate the effectiveness of the SAM.

- Useful SA/AM knowledge was accumulated:
  - Detail of the flow pattern in RCS and CV.
  - Effectiveness of the “forced depressurization SAM” for the induced SGTR during a severe accident.
  - Effective cooling by “CV natural convection SAM”.
  - Detail behavior and deposition of the aerosol particles (FP) in a containment vessel, etc.

- Need of further high quality experimental data to validate the RCS-CFD model and the CV-CFD model for SA and SAM mechanistic analysis.
Recent Regulatory Activities Related to Severe Accident Management in Korea

Tae H. Kim, Jae H. Park, Jung J. Lee, Key Y. Sung
Korea Institute of Nuclear Safety (KINS)

The 10th Korea-Japan Joint Workshop on PSA (KJPSA), Jeju, Korea, May 18–20, 2009

Contents

I. Introduction
II. SAMP of Ulchin 1&2
III. Regulatory Review on Ulchin 1&2 SAMP
IV. Development of Safety Requirement for SAMP
V. Conclusions
I. Introduction

Background

• No specific legal requirement against a severe accident
  • 10CFR50.34(f) of USNRC has been applied to licensing of new NPPs since 1989.

• Policy statement: “Policy on Severe Accident of NPPs” (2001.8)
  • Requires 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

I. Introduction

• SAMP of the policy statement
  • “An owner of nuclear power reactor should establish and implement severe accident management programs.”
  • Contents of SAMP
    » Development of accident management strategies and guidelines
    » Assessment of availability of essential instrumentation during the accident
    » Establishment of SAM organization
    » Training and education program, etc.
I. Introduction

History

• Aug. 2001: Policy on Severe Accident of NPPs
• Aug. 2001: Administrative order by MOST
  ➢ KHNP: to submit the action plan for the policy statement on severe accident
  ➢ KHNP submitted the implementation plan for PSA and SAMP (Oct. 2001)
• Oct. 2002~: Submission of PSA and SAMP by KHNP
• Jan. 2003~: Review on PSA and SAMP by KINS

Status of SAMP establishment

• Development of SAMP by utility
  ➢ LWRs (16 units)
  ➢ CANDU (4 units): ongoing
• Review by KINS
  ➢ Finished for 16 units’ SAMPs
  ➢ Documents subject to the review
    ➢ SAMG and technical background report
    ➢ Writer’s guide
    ➢ Training program
    ➢ Validation program
    ➢ Maintenance program

<table>
<thead>
<tr>
<th>Unit</th>
<th>Development of SAMG</th>
<th>KINS' review</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kori 1</td>
<td>2003</td>
<td>2006</td>
</tr>
<tr>
<td>Kori 2</td>
<td>2004</td>
<td>2007</td>
</tr>
<tr>
<td>Kori 3,4</td>
<td>2004</td>
<td>2007</td>
</tr>
<tr>
<td>Yonggwang 1,2</td>
<td>2004</td>
<td>2007</td>
</tr>
<tr>
<td>Yonggwang 3,4</td>
<td>2002</td>
<td>2006</td>
</tr>
<tr>
<td>Yonggwang 5,6</td>
<td>2001</td>
<td>2001</td>
</tr>
<tr>
<td>Ulchin 1,2</td>
<td>2007</td>
<td>2008</td>
</tr>
<tr>
<td>Ulchin 3,4</td>
<td>2002</td>
<td>2006</td>
</tr>
<tr>
<td>Ulchin 5,6</td>
<td>2002</td>
<td>2006</td>
</tr>
</tbody>
</table>
II. SAMP of Ulchin 1&2

- Ulchin units 1&2
  - Type: PWR – 3 Loop, Framatome
  - Capacity: 950 MWe
  - Commercial operation:
    - #1: '88.09
    - #2: '89.09
  - Availability factor ('08):
    - #1: 99.58 %
    - #2: 88.25 %

---

II. SAMP of Ulchin 1&2

- Capabilities against severe accident

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Kori 3&amp;4 (W-3 Loop)</th>
<th>Ulchin 1&amp;2 (Ft-3-Loop)</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Thermal Power (MWt)</td>
<td>2,775</td>
<td>2,775</td>
<td>Decay heat removal</td>
</tr>
<tr>
<td>Total Mass of Zircalloy (lbm)</td>
<td>43,500</td>
<td>43,497</td>
<td>H2, Heat removal</td>
</tr>
<tr>
<td>Containment Fee Volume (ft³)</td>
<td>2.08x10⁶</td>
<td>1.74x10⁶</td>
<td>Containment performance</td>
</tr>
<tr>
<td>Containment Design Pr. (psig)</td>
<td>60</td>
<td>61</td>
<td>Containment performance</td>
</tr>
<tr>
<td>Containment Design Temp. (°F)</td>
<td>295</td>
<td>284</td>
<td>Containment performance</td>
</tr>
<tr>
<td>RCS Depressurization (°, inch)</td>
<td>3 PORV (6)</td>
<td>3 POSRV (6)</td>
<td>Prevention of HPME</td>
</tr>
<tr>
<td>Rx Cavity Floor Area (ft²)</td>
<td>696.6</td>
<td>363.8</td>
<td>Mitigation of MCCI</td>
</tr>
</tbody>
</table>
II. SAMP of Ulchin 1&2

◆ Development of Ulchin 1&2 SAMG

• Based on the WOG SAMG
• Elements of SAMG
  ◦ Diagnostic flow chart (DFC)
  ◦ 10 severe accident guidelines (SAGs)
  ◦ 7 computational aids (CAs)
• Transition from EOP to SAMG
  ◦ When core exit temperature exceeds 700 °C with its increasing trend.
  ◦ Once the SAMG is initiated, it is not allowed to return to the EOP.
Mitigation strategies and actions of SAGs

<table>
<thead>
<tr>
<th>SAG number</th>
<th>SAG name</th>
<th>Mitigation actions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Emergency-1</td>
<td>MCR guideline</td>
<td>In case TSC is not functional,</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Depressurize RCS • Inject into RCS  • Depressurize CV</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Inject into S/G</td>
</tr>
<tr>
<td>Mitigation-1</td>
<td>Inject into S/G</td>
<td>• Inject into S/G</td>
</tr>
<tr>
<td>Mitigation-2</td>
<td>Depressurize RCS</td>
<td>• Depressurize RCS • Depressurize S/G</td>
</tr>
<tr>
<td>Mitigation-3</td>
<td>Inject into RCS</td>
<td>• Inject into RCS</td>
</tr>
<tr>
<td>Mitigation-4</td>
<td>Inject into containment (CV)</td>
<td>• Inject into CV</td>
</tr>
<tr>
<td>Mitigation-5</td>
<td>Control fission products releases</td>
<td>• Depressurize CV • Dump steam to condenser</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Vent Aux. building</td>
</tr>
<tr>
<td>Mitigation-6</td>
<td>Control containment conditions</td>
<td>• Remove heat form CV</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Depressurize or vent CV</td>
</tr>
<tr>
<td>Mitigation-7</td>
<td>Control containment hydrogen</td>
<td>• Recombine H₂ • Burn H₂ intentionally</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Stop active heat sinks in CV</td>
</tr>
<tr>
<td>Monitoring-1</td>
<td>Long term monitoring</td>
<td>N/A</td>
</tr>
<tr>
<td>Termination-1</td>
<td>SAM termination</td>
<td>N/A</td>
</tr>
</tbody>
</table>

II. SAMP of Ulchin 1&2

- **Organization and responsibility**
  - Decision making process is of importance from the viewpoint of severe accident management (SAM) capability
  - Role of each branches
    - MCR
      - Perform immediate actions
      - Monitor plant status
      - Implement actions requested by TSC
    - TSC
      - Identify and prioritize candidate actions
      - Evaluate candidate actions
III. Regulatory Review on Ulchin 1&2 SAMP

Objective of review

- The SAMP implementation should be flexibly performed based on the knowledge base.
- Objective of regulatory review
  - To evaluate the validity of SAMP from the viewpoint of comprehensiveness, consistency, and quality.
- Concerns of the review
  - whether each strategy and its basis are technically sound
  - whether plant specific features are taken into account in the SAMP
  - whether the SAMP is appropriately verified and implemented.
- Korean Safety Review Guideline (KSRG) 19.3 for SAMP

Technical soundness and weakness of strategies

- Ulchin 1&2 SAMG and its technical background documents are based on the WOG SAMG and EPRI TBR
- Majority of strategies are acceptable.
- Some weakness of strategies due to the features of plant design
  - Configuration of reactor cavity
  - No equipment for severe accident H₂ control
III. Regulatory Review on Ulchin 1&2 SAMP

- Flooding the containment with coolant
  - Insufficient water source for the RPV cooling
  - Difficult to pour water into reactor cavity before RPV break
  - Small reactor cavity area (0.012 m²/MWt)

<table>
<thead>
<tr>
<th>Name</th>
<th>CV Pressure (bar.a)</th>
<th>H₂ concentration (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>HP-1</td>
<td>9.98e-03</td>
<td>11.3</td>
</tr>
<tr>
<td>HP-2</td>
<td>9.98e-03</td>
<td>11.3</td>
</tr>
<tr>
<td>HP-3</td>
<td>9.98e-03</td>
<td>11.3</td>
</tr>
<tr>
<td>HP-4</td>
<td>9.98e-03</td>
<td>11.3</td>
</tr>
</tbody>
</table>

- Intentional hydrogen burning by turning on/off the fan cooler in containment
  - Deflagration to detonation transition (DDT) may occur due to flame acceleration (FA) which is generated in the ducts or caused by turbulent flow. ([OECD/NEA SOAR (NEA/CSNI/R (2000)7))]
  - Fan cooler should not be operated too long or continuously for the purpose of H₂ control.
III. Regulatory Review on Ulchin 1&2 SAMP

- RCS injection using HPSI pumps or hydraulic test pump under high RCS pressure condition
    - Possibility of hot leg creep rupture
    - Suggestion of a delayed injection (after RCS pressure depressurized to 1800 psig)
  - Domestic and foreign research results related to the SAMP should be continuously followed up and, if necessary, incorporated in the SAMP.

Availability of equipment and instruments

- Assessment of the environmental condition during a severe accident
  - Comparison between bounding temp./pressure and EQ data
  - Some instruments to be potentially unavailable in the late phase (RPV failure to CV failure) of accident sequence
    - CV radiation monitor, CV sump level switch
- Core exit thermocouple
  - WGAMA Task Group on CET Effectiveness in Accident Management (NEA/SEN/SIN/AMA(2008)7)
    - CET values are lower than those of PCT and there is much time delay between thermocouple readings of CET and PCT.
    - Difference between CET and PCT should be considered in the SAM strategy.
III. Regulatory Review on Ulchin 1&2 SAMP

Convenience for SAMG user

- Some SAGs require the SAMG team to refer to a normal operating procedure in order to conduct a mitigation strategy.
  - Burden to the SAMG staff who is under the highly stressed condition of severe accident
- The valves/equipment line-up tables in the SAMG should be replaced with simple drawings.

IV. Development of Safety Requirement for SAMP

Motivation of the development

- Needs for more detailed technical standards
- Change of the developing process of technical standards
  - The KINS was approved for developing its own technical standards and applying it by MEST in 2008.
- Draft safety requirements for KNGR (developed in 2001)
IV. Development of Safety Requirement for SAMP

Requirements and guides related to SAM

- MEST “Policy on Severe Accident of NPPs”, 2001
- IAEA documents
  - INSAG-12 (Basic safety principles for NPP)
  - Safety services series No. 9 (Guidelines for review of AMP)
  - Safety reports series No. 32 (Implementation of AMP in NPPs)
- CNSC G-306 (Regulatory guide for SAMP)
- USNRC NUREG-1462
- EPRI ALWR-URD
- etc.

Identified elements for SAM requirement

- Development and implementation of the SAMP
- Development of the SAM strategies
- Identification, optimum use and availability of systems and equipment necessary to the SAM
- Training program and procedures to cope with SA
- Decision-making organization and responsibilities
IV. Development of Safety Requirement for SAMP

Safety Requirement 16.4 “Severe Accident Management Program” (draft)

- Article 16.4.1: Establishment and Implementation of SAMP
  - Fundamental criteria to be complied with in implementing the safety principle in general aspect
  - Requires the owner of NPP to establish and implement a SAMP in order to cope with a severe accident.
  - Specifies the key elements of SAMP

- Article 16.4.2: Elements of SAMP
  - Paragraph 1: Actions to be taken by the owner of NPP when establishing the SAMP
    - To develop strategies in order to maximize human resources and equipment utilization for prevention and mitigation of severe accidents, and develop procedures and/or guidelines using the strategies
    - To identify I&C equipment or systems required to diagnose and control SA effectively, and evaluate their availability during a SA
    - To establish the AM organization and decision-making system
    - To establish and implement a training program for the plant personnel
    - To maintain and improve, if necessary, the SAMP by reflecting new technologies and information
IV. Development of Safety Requirement for SAMP

- Article 16.4.2 (cont.)
  - Paragraph 2: Documents and programs required for the SAMP
    - Plant specific SAMG and technical background report
    - Plant specific writer’s guide
    - Validation program
    - Training program
    - Maintenance program

V. Conclusions

- The SAMP of Ulchin 1&2 has been successfully developed and implemented in accordance with the policy statement.
- Regulatory review identified some weakness of strategies and made recommendations for improving the Ulchin 1&2 SAMP.
- KINS has been developing the safety requirements to set up the criteria for the SAMP development and implementation.
- Regulatory body is going to monitor the implementation of SAMP through inspection and observation of emergency exercises/drills.
Thank you for your attention!
Consideration of Emergency Source Terms for Pebble-bed High Temperature Gas-Cooled Reactor

Liu Tao, Zhao Jun, Tong jiejuan, Cao jianzhu
INET of Tsinghua University, Beijing, China

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Outline
- Introduction
- LWR Emergency Source Terms (ESTs) characteristics
- Investigations on advanced NPP EP
- Pebble-bed HTGR ESTs consideration
- Pebble-bed HTGR characteristics relating to ESTs
- PSA for Pebble-bed HTGR ESTs
- Conclusions

The 10th Korea-Japan Joint Workshop on PSA, Jeju, Korea, May 18~20, 2009.
1. Introduction

Emergency planning (EP) as an part of the defense-in-depth strategy is to provide adequate protection for the public health and safety.

ESTs is an crucial factor for EP basic elements.

Pebble-bed HTGR ESTs need further investigation.

2. LWR ESTs characteristics

Based on LWR accidents’ features.
- Having potential CD and LER.
- Having rapid accident progress.

Setting EPZ to assure most of accidents consequences not beyond PAG at EPZ boundary.

Considering both DBAs and severe accidents.
3. Investigations on advanced NPP EP

URD “simplification”, “ALWR good neighbor”

- A challenging, quantitative requirement on mitigation:
  whole body dose less than 25rem at the site boundary
  (about 0.5 miles from the reactor) for accident sequences
  with cumulative frequency greater than $10^{-6}$ per year

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3. Investigations on advanced NPP EP

EUR “Emergency planning simplification”

- **DBC:**
  - No action necessary beyond 800 m from the damaged plant

- **DEC:**
  - No emergency protection action beyond 800 m from the reactor;
  - No delayed action at any time beyond about 3 km from the reactor;
  - No long term action at any distance beyond 800 m from the reactor.

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4. Pebble-bed HTGR ESTs consideration

- ESTs should consider all of potential accident sequences.
- ESTs may depend on specific release scenarios.
- ESTs should reflect Pebble-bed HTGR features.
- ESTs is expected to support the EP simplification.

5. Pebble-bed HTGR characteristics relating to ESTs

- Spherical coated particles’ fuel elements
  - TRISO type: pyrolytic carbon, silicon carbide, porous carbon buffer.
  - Defective temperature limitation of silicon carbide is above 1600 °C
- Passive residual heat remove measures.
- Lower power level compares with higher thermal capacity.
6. PSA for Pebble-bed HTGR ESTs

- Screening initiating events (IEs)
- Classifying IEs
- Developing accidents’ spectrum
- Inducing release categories
- Selecting bounding STs.
6. PSA for Pebble-bed HTGR ESTs

- Data Analysis
- Systems and Fault Tree Analysis
- Human Reliability Analysis
- Sequence Quantification

- Initiating Event → Event Trees → Sequence End States → ESTs

- Success Criteria Development
- Thermal and Fluid Flow Analysis
- Source Term Analysis

7. Conclusions

- ESTs is derived from four release categories.
  - RCs P: depressurization accidents
  - RCs S: water ingress accidents
  - RCs T: transient overpressure
  - RCs A: air ingress accidents

- Best estimated with appropriate conservative model is used for bounding Source terms calculation.
END
Thank you for your attention
Probabilistic Estimation of the Early and Cancer Fatality Risks at the Korean NPPs

Do Sam Kim, Key Yong Sung, Jong Soo Choi and Han Chul Kim

Korea Institute of Nuclear Safety

Presented at the 10th Korea-Japan Joint Workshop on PSA (KJPSA)
2009.5.18~20, Jeju Island, Korea

Table of Contents

I. Introduction
II. Data and Methodology
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I. Introduction

1. Background

- Safety goals (The policy on severe accident, August 29, 2001)
  - The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from other accident.
  - The risk to the population in the area of nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risks from all other causes.

To assess health risks to the public, we need to perform Level 3 PSA.

This study was performed as a part of the process of drafting the performance goals in Korea.

2. Level 3 PSA

- To assess the off-site consequences leading, together with the results of Level 2 analysis, to estimates of risks to the public.

- Procedures
  - Establishment of source terms
  - Sampling of Meteorological data
  - Evaluation of atmospheric diffusion and deposition
  - Dose assessment
  - Analysis of radiation-dose reduction by protective measures
  - Evaluation of health effects
3. In this study,

- Site-specific health risks of early and cancer fatalities have been estimated through Level 3 PSA by using
  - Source terms obtained from level 2 PSA,
  - Site-specific meteorological data and population distribution,
  - Emergency response activities modeled based on the licensee’s Radiological Emergency Response Plan including sheltering, evacuation, and dose dependent relocation.

- Sensitivity analysis performed with respect to
  - Early fatalities depending on the multiple definitions of LERF
  - Emergency Response Scenarios

II. Data and Methodology

1. Source Terms

- The information is provided from level 2 PSA,
  (Although it may have large uncertainties.)
  - The amount of released radionuclide
  - Time from the initiation of accidents to the atmospheric release
  - Duration of the release
  - Thermal energy contained in the release
  - Release position (height), etc.

- Level 2 PSA completed for all operating Korean NPPs (~’07)
  - 9–19 source term categories
  - To improve the quality of the source terms, following information should be more investigated:
    - General emergency declaration time to model protective actions.
    - Source term characteristics (such as included thermal energy, release time, etc.)
Table 1. PSA Implementation for operating NPPs

<table>
<thead>
<tr>
<th>Units</th>
<th>PSA Activities</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kori 1,2</td>
<td>2003, 2007(R*)</td>
</tr>
<tr>
<td>Kori 3,4, Yonggwang 1,2</td>
<td>1992(typhoon), 2003(R*)</td>
</tr>
<tr>
<td>Yonggwang 3,4</td>
<td>1994(Design stage), 2004(R*)</td>
</tr>
<tr>
<td>Ulchin 1,2</td>
<td>2005</td>
</tr>
<tr>
<td>Ulchin 3,4</td>
<td>1997(Design stage), 2004(R')</td>
</tr>
<tr>
<td>Wolsong 1</td>
<td>2003, 2007(R')</td>
</tr>
<tr>
<td>Wolsong 2,3,4</td>
<td>1997(Design stage), 2007(R')</td>
</tr>
<tr>
<td>Yonggwang 5,6</td>
<td>2000, 5005(R*)</td>
</tr>
<tr>
<td></td>
<td>Level 1 PSA</td>
</tr>
<tr>
<td>Ulchin 5,6</td>
<td>2002, 2006(R*)</td>
</tr>
<tr>
<td></td>
<td>During shutdown</td>
</tr>
</tbody>
</table>

* Revision

2. Estimation of early fatality for different release definitions

- Estimation of early fatality
  - With no emergency response,
    - We must consider all STCs which can cause early fatalities
  - With assumptions on the emergency response,
    - Large early release frequency (LERF) can be used as a measure of early fatality

⇒ In this paper,
  - we considered the following cases in the estimation of early fatality
    - Containment failure accidents including all STCs (Case 1)
    - 4 different definitions of LERF (Case 2–5)
Case 1: Containment Failure Frequency (CFF)

\[ CFF = CCFP^{1)} \times CDF \approx LRF^{2)} \]

- Early containment failure
- Late containment failure
- Containment bypass
- Containment isolation failure
- Containment bottom melt through

1) CCFP: Conditional containment failure probability
2) LRF: Large Release* Frequency
* Large Release: A collection of all release that would result in one or more early fatalities (NUREG/CR-6094, SECY-89-102)

Case 2: LERF definition 1

⇒ NUREG/CR-6595, Appendix A.2 First definition
- LERF consists of the total frequency of all release classes that occur under the early containment failure or containment bypass categories of the containment failure mode matrix.
  - Early containment failure
  - Containment bypass
  - Containment isolation failure

※ KHNP used this definition in the PSA of domestic NPPs.

Case 3: LERF definition 2

- Accidents with short (<6 hours) delay time from core uncovery to atmospheric release of radioactive materials
  ⇒ Case 2 + Containment failure before reactor vessel break
**Case 4: LERF definition 3**

- **NUREG/CR-6595, Appendix A.2 Second definition**
  
  LERF consists of the frequency of release classes associated with the early failure and bypass containment failure modes which have release fractions of the volatile/semi-volatile fission products (Iodine, Cesium, Tellurium) equal to or greater than about 2.5% to 3%.

**Case 5: LERF definition 4**

- **NUREG/CR-6595, Appendix A.2 third definition**
  
  LERF is the frequency of early failure and bypass containment failure modes that have a release fraction of iodine equal to or greater than about 10%.

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**Table 2. Source term categories to CFF & LERF**

(Ulchin unit 3,4)

<table>
<thead>
<tr>
<th>STC</th>
<th>Description</th>
<th>CFF Case 1</th>
<th>CFF Case 2</th>
<th>CFF Case 3</th>
<th>CFF Case 4</th>
<th>CFF Case 5</th>
<th>LERF Case 1</th>
<th>LERF Case 2</th>
<th>LERF Case 3</th>
<th>LERF Case 4</th>
<th>LERF Case 5</th>
</tr>
</thead>
<tbody>
<tr>
<td>STC 1</td>
<td>CV intact</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 2</td>
<td>CV failure, RB intact</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 3</td>
<td>Early (Leak)</td>
<td></td>
<td>O</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 4</td>
<td>Early (Rupture)</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>STC 5</td>
<td>Late (Rupture), Spray X, corium cooling O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>O</td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>STC 6</td>
<td>Late (Leak), Spray O, corium cooling X</td>
<td>O</td>
<td></td>
<td></td>
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<td></td>
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<tr>
<td>STC 7</td>
<td>Late (Leak), Spray X, corium cooling O</td>
<td></td>
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</tr>
<tr>
<td>STC 8</td>
<td>Late (Leak), Spray X, corium cooling X</td>
<td>O</td>
<td></td>
<td></td>
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<tr>
<td>STC 9</td>
<td>Late (Rupture), Spray X, corium cooling O</td>
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</tr>
<tr>
<td>STC 10</td>
<td>Late (Rupture), Spray X, corium cooling X</td>
<td>O</td>
<td></td>
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</tr>
<tr>
<td>STC 11</td>
<td>Late (Rupture), Spray O, corium cooling X</td>
<td>O</td>
<td></td>
<td></td>
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</tr>
<tr>
<td>STC 12</td>
<td>Late (Rupture), Spray X, corium cooling X</td>
<td>O</td>
<td></td>
<td></td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 13</td>
<td>BMT, Spray X, corium cooling X</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 14</td>
<td>Alpha mode</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 15</td>
<td>Containment failure before Reactor/Vessel Break</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 16</td>
<td>Isolation failure, Spray O</td>
<td>O</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 17</td>
<td>Isolation failure, Spray X</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 18</td>
<td>IS LOCA</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>STC 19</td>
<td>SGTR</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td>O</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
3. Meteorological data and population distribution

- **Meteorological data**
  - MACCS2 Code used for the Level 3 Calculation
  - Meteorological Data
    - Yonggwang: 2000~2004
    - Ulchin, Kori, Wolsong: 2001~2005

- **Population distribution**
  - 10 radial divisions
  - Maximum distance: 80 km
  - Angular division fixed
    - 16 sectors (22.5 degree wide)
  - Population defined at each sector
4. Emergency response scenarios

- Emergency phase and exposure pathway

<table>
<thead>
<tr>
<th>Release</th>
<th>7 days (1 week)</th>
<th>Max (317 year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Phase</td>
<td>emergency</td>
<td>Long-term</td>
</tr>
<tr>
<td>2. Exposure Pathways</td>
<td>Cloudshine</td>
<td>Groundshine</td>
</tr>
<tr>
<td></td>
<td>Groundshine</td>
<td>Resuspension inhalation</td>
</tr>
<tr>
<td></td>
<td>Resuspension inhalation</td>
<td>Food and water ingestion</td>
</tr>
<tr>
<td>3. Protective Actions</td>
<td>Sheltering</td>
<td>Decontamination</td>
</tr>
<tr>
<td></td>
<td>Evacuation</td>
<td>Temporary</td>
</tr>
<tr>
<td></td>
<td>Dose dependent relocation</td>
<td>interdiction</td>
</tr>
<tr>
<td></td>
<td></td>
<td>condemnation</td>
</tr>
</tbody>
</table>

- MACCS2 code Modeling

- Region
  - Sheltering and Evacuation region
  - Relocation zone

- Evacuation phase
  - Early, middle, late phase with different velocity

- Shielding factor and Breathing rate
  - Evacuees while moving
  - Normal activity (in sheltering and evacuation zone)
  - Sheltered activity

- Dose dependent relocation
  - Hot-spot(500mSv), normal relocation(250mSv)

A. Inside the Evacuation and Sheltering Region

B. Relocation Zone: Outside of the Evacuation and Sheltering Region
Emergency response scenario modeling

- **Evacuation scenario**
  - Sheltering → Evacuation
  - Radial evacuation
  - At time 0, general emergency is declared, release start

![Diagram showing timelines for release, duration of release, alarm time, delay to shelter, delay to evacuation, and evacuation.]

5. Risk of Early and Cancer Fatality

- **Early fatality**

\[
ILR = \sum_{n=1}^{N} F_n \times \frac{EF_n}{TP(1.6km)}
\]

where

- \( F_n \): frequency of the release capable of causing early fatalities for accident sequence “n”
- \( CFF, LERF \) investigated
- \( EF_n \): number of early fatalities within 1.6km in case of accident sequence “n”
- \( TP(1.6km) \): total population within 1.6km from NPP

* Early fatality = H(Acute dose)
  - Acute dose commitment period: 1 year
Cancer fatality

\[ ILR = \sum_{n=1}^{N} F_n \times \frac{LF_n}{TP(8km)} \]

where:
- \( F_n \): frequency of the release capable of causing cancer fatalities for accident sequence \( n \)
- \( LF_n \): number of cancer fatalities within 8 km in case of accident sequence \( n \)
- \( TP(8km) \): total population within 8 km from NPP

\* Cancer fatality = R(Lifetime dose)
  : Lifetime dose commitment period : 50 years

III. Results

1. Cancer Fatality

- Safe goal for the risk of cancer fatality in Korea
  : \( \sim 1.0 \times 10^{-6} \) /year
- In the estimation of cancer fatality
  - All source term categories were considered.
  - Population averaged risks were obtained within 8 km from the plant.
- The results show that,
  - Even though we assume no protective actions, all Korean NPPs satisfy the Safety goal for the cancer fatality
Case 1
- Dose dependent relocation (Hotspot: 500mSv, Normal: 250mSv)
- 95 % evacuation, 5% sheltering
Case 2 : No protective actions

Fig. 1. Cancer fatality in Korean NPPs

2. Early Fatality

- Safe goal for the risk of early fatality : ~ 5.0 x10^{-7} /year
- The risk is estimated for the following cases of risk measures
  - Case 1: Containment failure frequency (CFF)
  - Case 2~5 : Large early release frequency (LERF)
- The results show that,
  - LERF definitions do not make significant differences
  - Wolsung unit 2,3&4 show the smallest risk values due to
    - its lower LERF values and
    - Smaller quantity of released source term
  - Protective actions like sheltering and evacuation decrease the risks one or two orders in all cases
Fig. 2. Early fatalities at Ulchin unit 3,4 NPPs

Fig. 3. Early fatalities with respect to meteorological data (100% evacuation)
Fig. 4. Early fatalities with respect to meteorological data  
(No protective actions)

Fig. 5. Early fatalities for internal and external initiating events  
(No protective actions)
IV. Conclusions

• In this paper,
  ✓ The risks of early and cancer fatality estimated for Korean NPPs
    - based on the up-to-date site-specific data
    - including the Level 2 PSA results.
  ✓ Also, the sensitivity cases analyzed for the followings:
    - Emergency response scenarios, various definitions for the LERF,
      Meteorological data, and Internal & external initiating events

• The results show that,
  ✓ The changes of early fatality risk is not significant for the various
    definitions of LERF.
  ✓ The Risk of cancer fatality have large margins to the safety goals
    than the case of early fatality
  ✓ To improve the estimation, further study should be done in the field
    of the quality enhancement of Level 2 and 3 PSA.
Implementation of a Molten Core Cooling Strategy in a Severe Accident Management Guideline

J. H. Song¹, N. D. Suh²

¹ dosa@kaerlne.kr, ² k220snd@khnre.kr

Molten Core Cooling

- In-vessel re-flooding
  - Water is injected into RV, if available, during a severe accident
  - Hydrogen generation, RCS repressurization, steam explosion...  

- Ex-Vessel Reactor Cooling
  - AP600, AP1000 – Heat load removed by pool boiling
  - Streamlined Insulation Structure – Increase in CHF
  - Passive flooding of reactor cavity
  - Need to consider ex-vessel steam explosion

- Ex-Vessel Debris Coolability
  - OECD/MCCI project – top flooding
  - Core catcher – spreading and cooling issues
New Reactors: AP1000 – In Vessel Retention

New Reactors: EPR – Core Catcher, Containment
**Existing Reactors: MCCI and Debris Coolability**

![Diagram of reactor with labels and images](image)

**KSNP Severe Accident Management Guideline**

- **For Existing Reactors**

  - 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: pre-flooding.

  - 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
Strategy Flow Chart

Inject into the Containment

Available water source, pumps: Can we fill the reactor cavity?
- APR1400, new reactor, use BAMP+SCP to fill up the reactor cavity
- In case of Kori -1, cannot reach the bottom of RV

10th KJPSA, Jeju, Korea, May '09
MCCI and Debris Coolability: OECD/MCCI Program at ANL

Results of OECD/MCCI Program

Figure 6-9. Maximum Basemat Ablation (LCS Concrete).
Kori-1 Plant

- General description of the plant
  - W-2-loop PWR
  - First operation in Apr 1978
  - Power: 1.724 MWth, 587 MWn
  - 2 RCPs, 2 SGs, 1 PZR

- Containment
  - Steel vessel
  - Design pressure: 43.0 psig (398 KPa)
  - P_fiber: 122 psig (943 KPa)
  - Net free volume: $1.45 \times 10^4$ ft$^3$

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MELCOR Analysis for Kori-1

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JAEA-Review 2009-038
## SBO Sequence of Top Events for Kori-1

<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>

- Accident Scenario analyzed is SBO without any operator action to accelerate accident progression
- Sequence of Top Events are on the left table
- Corium is ejected at T=23,650 s

## 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>

- Concrete type of Kori-1 is LCS
- Average design depth of base mat concrete is 2.133 m
- MELCOR provides information on the initial collapsed melt depth, ratio of concrete content.
- Fig.6-9 of OECD/MCCI results provides the ablation depth to stabilization
### MCCI condition at Reactor Cavity - 10hrs

<table>
<thead>
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<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>
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- At 10 hrs after SBO, the ablation depth to stabilization is ~1.1m from Fig.6-9
- 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

### MCCI condition at Reactor Cavity - 24 hrs

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- At 24 hrs after SBO, parameters are outside the applicability of OECD data
- Extrapolating the figure, we can estimate the margin to melt through is less than 0.233m
- Thus taking into account the uncertainties, we should say the coolability is not guaranteed
Improvements Suggested for SAMG

• Appropriate instrumentation to detect either the breach of the reactor vessel or discharge of corium into the reactor cavity.

  - A suggestion for using ex-core indicators for detection of relocation behavior of corium (CEOG, 1995). It only provided indirect information on the behavior of the molten core.

• Need for a calculation aid, which would give us a direction as to whether we should initiate a pre-flooding or a post-flooding.

  - If there is little chance of delaying the failure of a reactor vessel by a pre-flooding, for example, due to a shortage of the water inventory or an inadequacy of the reactor vessel and the insulator geometry in providing steam flow path, there is no reason to pre-flood the cavity, as it can lead to an unexpected damage to the reactor cavity or the containment due to an energetic steam explosion.
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