Session III-A

PSA & Applications

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Session Summary III-A: PSA & Applications

Chair: Jin-Hee PARK (KAERI), Jong-Soo CHOI (KINS)

Three presentations introduced in PSA & Application session and one presentation not introduced. One was presented by TEPCO Japan and two were presented by domestic speakers, KHNP and KEERI.

Mr. Eisuke Sato from TEPCO SYSTEMS Corp. Japan presented for the Upgrade of internal events PSA model using the AESJ PSA standard. He presented the procedure of the upgrade PSA model, the PSA main elements analysis and the results of each analysis such as system Reliability, Human Reliability, Quantification, Sensitivity analysis. He concluded that this upgrade of PSA applied into new maintenance program in TEPCO and the enhancement of PSA will be performed in the future.

Mr. Hwang, Seok Won from KHNP(NETEC)-Korea, introduced the background, Data management strategy and current status for Development of Nuclear Reliability Database System(PRinS) and its application for PSA or Maintenance Rule in Korea. He concluded that this Database System could enhance plant safety and efficiency, plant economical efficiency and apply the prompt and reasonable action for regulatory inspections also.

Mr. Park, Jin Hee from KEARI-Korea introduced development of regulatory PSA Model for Graded Regulation based RIPB for KINS. He presented the background, status, plan and insight of the regulatory PSA Model.

The attendants showed the interests on PSA upgrade in Japan and Regulatory PSA model development and asked several questions about Database System development in Korea.

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III-A-1

Upgrade of internal events PSA model using the AESJ Level-1 PSA Standard for operating state

The 10th Korea-Japan PSA Workshop

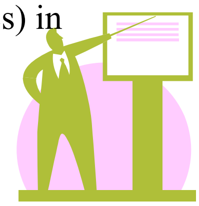
2009.5.18-20

Seogwipo, Jeju Island, Korea

Eisuke Sato TEPCO SYSTEMS Corp.
Mitsuru Yoneyama TEPCO SYSTEMS Corp.
Shigeatsu Tomizawa TEPCO

Presentation Overview

I. Introduction and Background
II. Overview of the upgraded PSA model
III. Quantification results of the upgraded PSA model
  III.A. Quantification of Accident Sequences
  III.B. Uncertainty analysis
  III.C. Sensitivity Analysis
IV. Adopting PSA results for maintenance of Structures systems and components (SSCs) in TEPCO Nuclear Power Plants (NPPs)
V. Conclusion



I. Introduction and Background

- In Japan
  - Interest in PSA has been growing
    - Nuclear Safety Commission (NSC)
      - "the policy statement of risk informed nuclear safety regulation"
    - Nuclear and Industrial Safety Agency (NISA)
      - "the basic guideline of utilizing risk information to nuclear safety regulation"
    - Japanese industrial code of maintenance of nuclear power plant (JEAC 4209-2007)
      - Perform maintenance effectively and efficiently by utilizing risk information.
    - Atomic Energy Society of Japan (AESJ)
      - "The Level-1 PSA standard of internal events for operating state (AESJ standard)" was published at April 2009.
- In TEPCO
  - Using the AESJ standard, PSA model has been upgraded.
    - Ensured high quality of the PSA model (not yet peer reviewed).
    - Important categorization of SSCs will be performed to increase nuclear plant safety using information of its risk importance.

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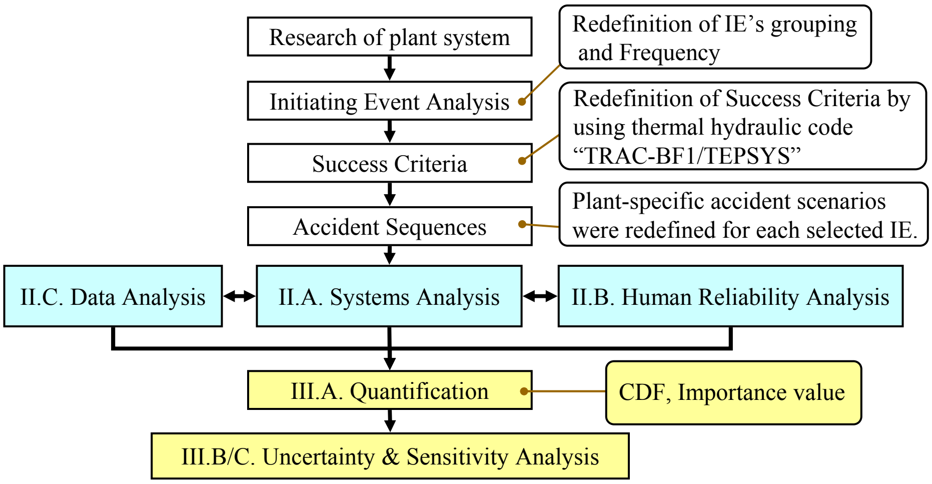
II. Overview of the upgraded PSA model

Procedure of the internal events Level-1 PSA model for operating state explained in the AESJ standard.
(The concept of capability category like the ASME Std. was not made in the AESJ Std.)



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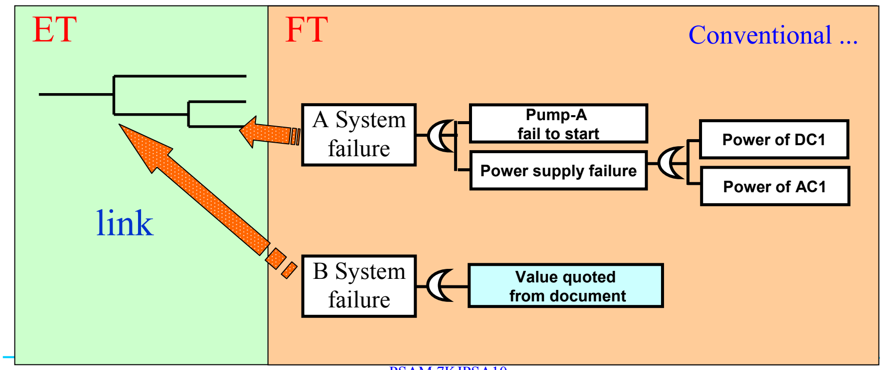
II.A. Systems Reliability Analysis (1/6)

- AESJ requirement (For example,)
  - **Modeling of system**
    - Recommend to use fault tree method.
      (Also another appropriate methods are allowed.)
    - If the capabilities of SSCs analyzed by design or other safety analysis are insufficient for relevant accident sequence, these are addressed as unavailable.
  - **Modeling of dependency**
    - Identify and model dependencies derived from dependency analysis of initiating events and different intersystem and intra-system.
      - Dependencies between front line system and support system.
      - Establish common cause failure groups and failure modes by considering design, malfunction, environment, and maintenance activity.

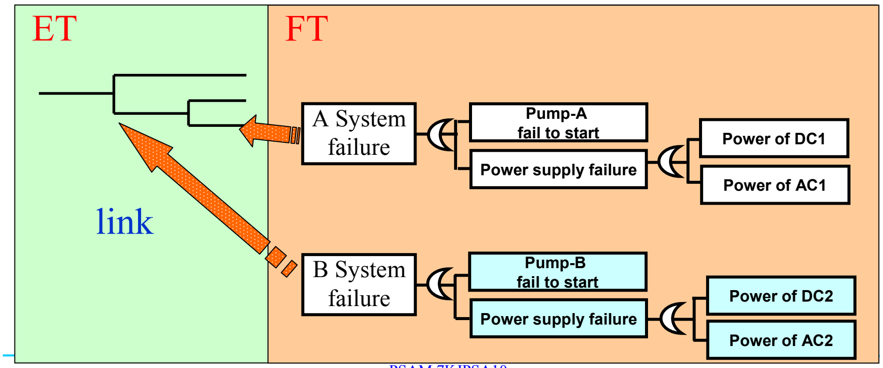
II.A. Systems Reliability Analysis (2/6)

- **Modeling of system**
  - Recommend to use fault tree method.
  - If the capabilities of SSCs analyzed by design or other safety analysis are insufficient for relevant accident sequence, these are addressed as unavailable.



II.A. Systems Reliability Analysis (3/6)

- **Modeling of system**
  - Recommend to use fault tree method.
  - If the capabilities of SSCs analyzed by design or other safety analysis are insufficient for relevant accident sequence, these are addressed as unavailable.



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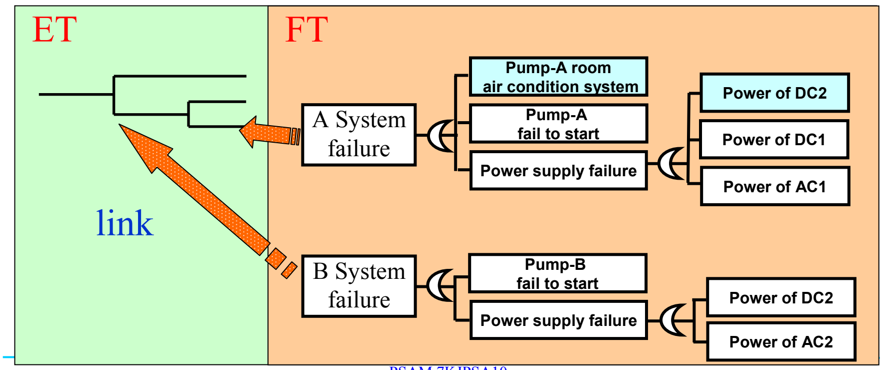
II.A. Systems Reliability Analysis (4/6)

- **Modeling of system**
  - Recommend to use fault tree method.
  - If the capabilities of SSCs analyzed by design or other safety analysis are insufficient for relevant accident sequence, these are addressed as unavailable.



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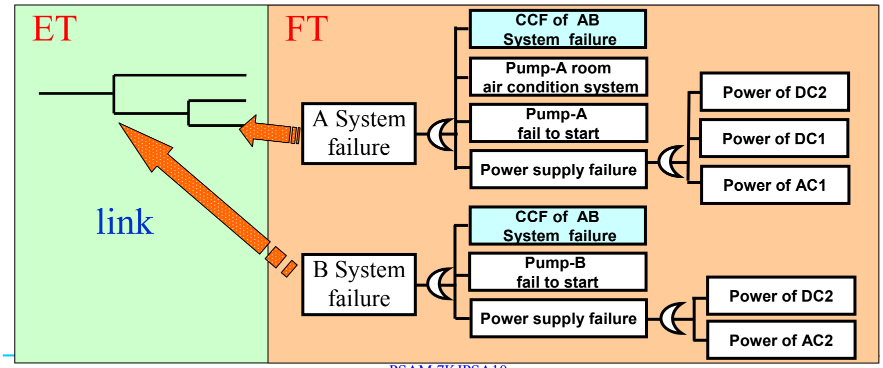
II.A. Systems Reliability Analysis (5/6)

- **Modeling of dependency**
  - Identify and model dependencies derived from dependency analysis of initiating events and different intersystem and intra-system.
    - Dependencies between front line system and support system.
    - Establish common cause failure groups and failure modes by considering design, malfunction, environment, and maintenance activity.



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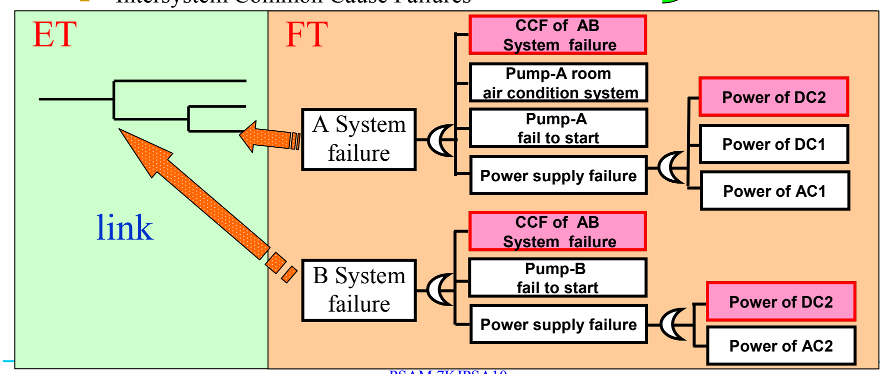
II.A. Systems Reliability Analysis (6/6)

- Upgrade Systems Reliability Analysis
  - **Modeling of system**
    - Condensate and feed water system
    - PCV vent with Stand-by Gas Treatment System (SGTS).
    - Deterioration of environmental condition
  - **Modeling of dependency**
    - Intersystem Common Cause Failures

Dependencies can be addressed appropriately



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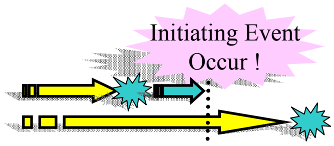
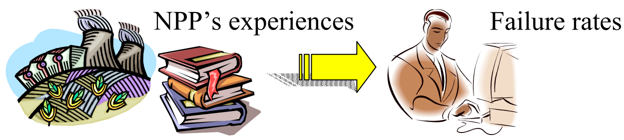
II.B. Human Reliability Analysis



- AESJ requirement (For example,)
  - **Latent errors** (pre-initiator errors)
    - The assessment of probabilities shall be performed by using a systematic process that addresses the plant-specific , activity-specific influences on human performance.
  - **Dynamic errors** (post-initiator errors)
    - Measure the impact of the plant-specific , scenario-specific Performance Shaping Factors (**PSF**).
- Upgrade HRA (For example,)
  - THERP method and parameter
  - Referring to NUREG/CR-6883 and NUREG-1792
    - Performance Shaping Factors (**PSF**) are considered systematically, for each corresponding procedure, available time, stress, complexity, dependency between operator action, and so on.

II.C. Preparation of Parameter Dataset

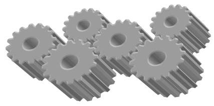
- Conventional…
  - NUREG/CR-1278, IEEE Std-500 …etc
- AESJ requirement (For example,)
  - If generic parameters are used as these are , the parameters shall be consistent with plant's attribute.
- Upgrade Parameter (For example,)
  - Component failure rates are prepared by general parameters estimated from Japanese NPP's experiences.
  - The generic parameters are issued by Japan Nuclear Technology Institute (JANTI).



  - The failure rates are generally smaller than conventional ones.

III.A. Quantification of Accident Sequences (1/4)

- CDF is quantified.
- Important contributors to CDF are identified.
  - "Safety Watcher" code (succession code of FT-FREE).
    - Developed uniquely by TEPSYS
    - fault tree linking method
    - Introduces ARALIA which is generalized tool based on Binary Decision Diagram (BDD)
    - Dependencies among all basic events (**1793**) are addressed appropriately.



III.A. Quantification of Accident Sequences (2/4)

Influence on CDF

Increase factor (II.A)		Decrease factor (II.C)
➢Impact by detail CCF model & Dependencies among systems	<	➢The domestic component failure rates are relatively smaller.

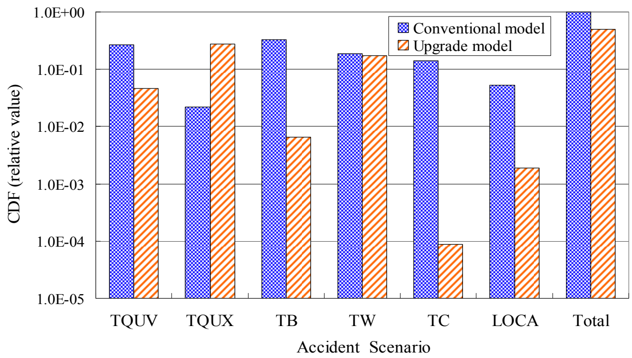


Fig.1 The relative CDF of each Accident Scenario

➢The total CDF evaluated by the upgraded model is smaller than that of conventional model

III.A. Quantification of Accident Sequences (3/4)

- The risk dominant accident scenario is TQUX brought by manual shutdown by loss of DC-A power as shown in Fig.3.

This is ･･･

- Several systems are unavailable due to loss of DC-A power.

The reason is ...

- Following systems are modeled in detail.
  - Condensate and feed water system (upgraded)
  - High Pressure Injection system
  - Depressurization system

⇩

Dependencies on DC-A power can be expressed.

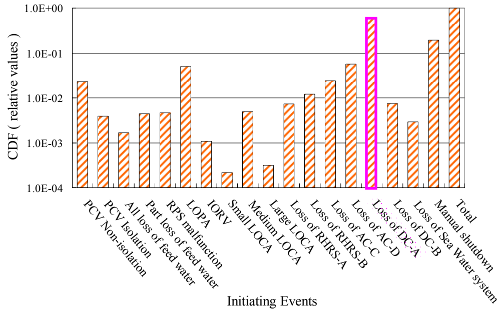


Fig.2 The relative CDF results of each initiating events

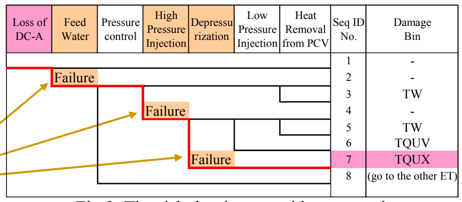


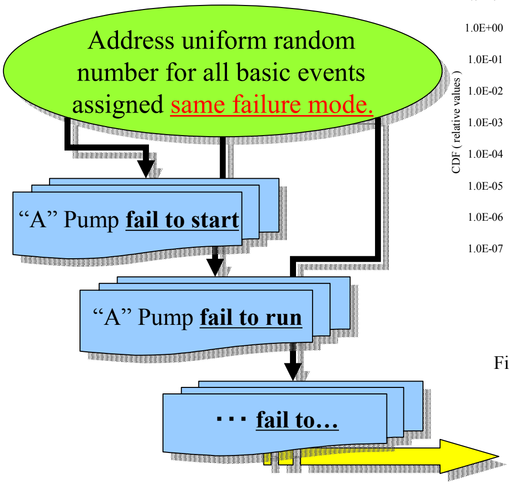
Fig.3 The risk dominant accident scenario

III.A. Quantification of Accident Sequences (4/4)

- Importance values of all basic events are evaluated automatically by Safety Watcher.
- Importance values of systems are evaluated by the importance values of basic events.
  - FV (Fussell- Vesely)
    - Emergency Diesel Generator
    - Operation human errors associated with coolant injection
    - Power supply systems
  - RAW (Risk Achievement Worth)
    - Control rod drive system
    - Power supply systems
    - Sea Water systems
  - Power supply systems have high dependency.

III.B. Uncertainty analysis

- Uncertainty analysis is performed by Monte Carlo method of Safety Watcher.



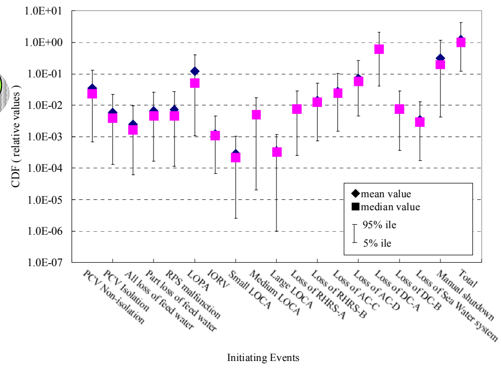


Fig.3. Uncertainty Analysis results in each initiating events

III.C. Sensitivity Analysis

- Candidates of analysis
  - a. Frequency of manual shutdown due to failure of DC-A power
    - ✓ The frequency is estimated by assuming 0.5 events occurred since the initiating event has never occurred in Japan.
  - b. Human Error Probability (HEP) to leave a handy valves inappropriate state. (latent human error)
    - ✓ The HEP can be decreased by updating actual plant procedure/management.
  - c. Operation human error probability to depressurize RPV by SRV
    - ✓ This is dynamic human errors, it may have less sensitivity than that of latent.
- Sensitivity Analysis is performed about a & b.

Case a. *Frequency of manual shutdown due to failure of DC-A power*

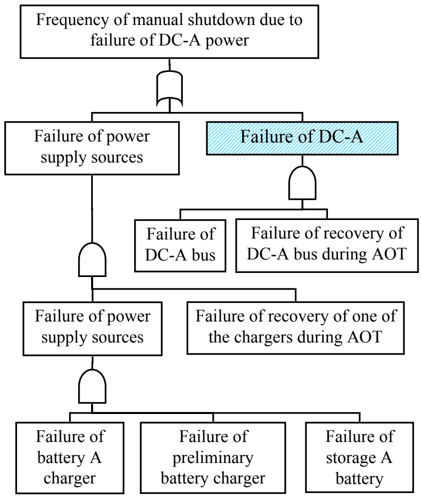


Fig.5 Fault tree model of manual shutdown due to the failure of DC-A power (case2).

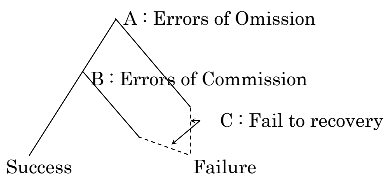
- Base case
  - Assumed 0.5 event occurred
- Two other methods as example in the AESJ std.
  - Assumed 1 event occurred (case1)
  - Fault Tree system analysis (case2)

Table.1 Sensitivity analysis results about case a.

Case	Calculation methods	IE frequency	CDF
Base	Assumption of 0.5events occurred	–	–
Case1	Assumption of 1 event occurred	200%	160%
Case2	Fault Tree system analysis	2%	41%

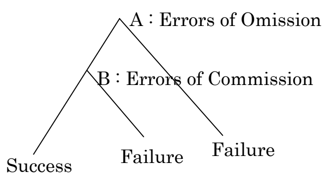
The CDF is decreased by adopting frequency evaluated by fault tree analysis.

Case b. *Latent HEP of forgetting to open handy valves*



HEP = (A + B) × C

Fig.6 HRA tree of recovery expected



HEP = A + B

Fig.7 HRA tree of recovery not expected

- Base case
  - Expect recovery
- Two other methods
  - Not expecting recovery (case1)
  - Not occurred the HE (case2)

Table.2 Sensitivity analysis results about case b.

Case	Condition	HEP	CDF
Base	Expecting recovery	–	–
Case1	Not expecting recovery	5882%	8336%
Case2	Not occurred the HE	0%	70%

It may be effective to use valve checklist, plant patrol and so on

IV. Adopting PSA results for maintenance of SSCs in TEPCO NPPs

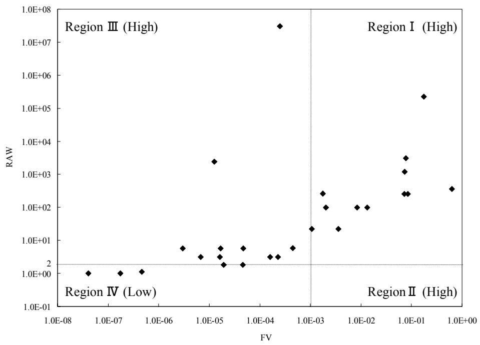


Fig.8. Risk importance based on the new maintenance program

- Recently
  - “The new maintenance program”
  - Risk importance values are considered as supplementing to classification of SSCs.
  - Refer these risk importance to decide maintenance importance and performance criteria.
- In the future
  - Peer review will be performed.
  - Studying applicability to implement reasonable maintenances and plant operations.

V. Conclusions

- Using the AESJ standard
  - Upgrade internal events Level-1 PSA model for operating state
  - Introduce the PSA results into the new maintenance program in TEPCO.
- In the future
  - Enhance quality of the PSA model through peer review and so on.
  - Expected that PSA results will be used for effective maintenance and plant operation.

Thank you for your attention !

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The 10th Korea-Japan Joint Workshop on PSA
May 18-20, 2009 • Jeju, Korea

Development of Web-Based Plant Reliability Data Information System and its Application

2009. 5. 18

Hwang, Seok Won (swhwang@khnp.co.kr)

KHNP (NETEC)
Reliability Engineering team

KHNP

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Contents

- Background
- Data Management Strategy
- Current Reliability DB (PRinS)
- Conclusion

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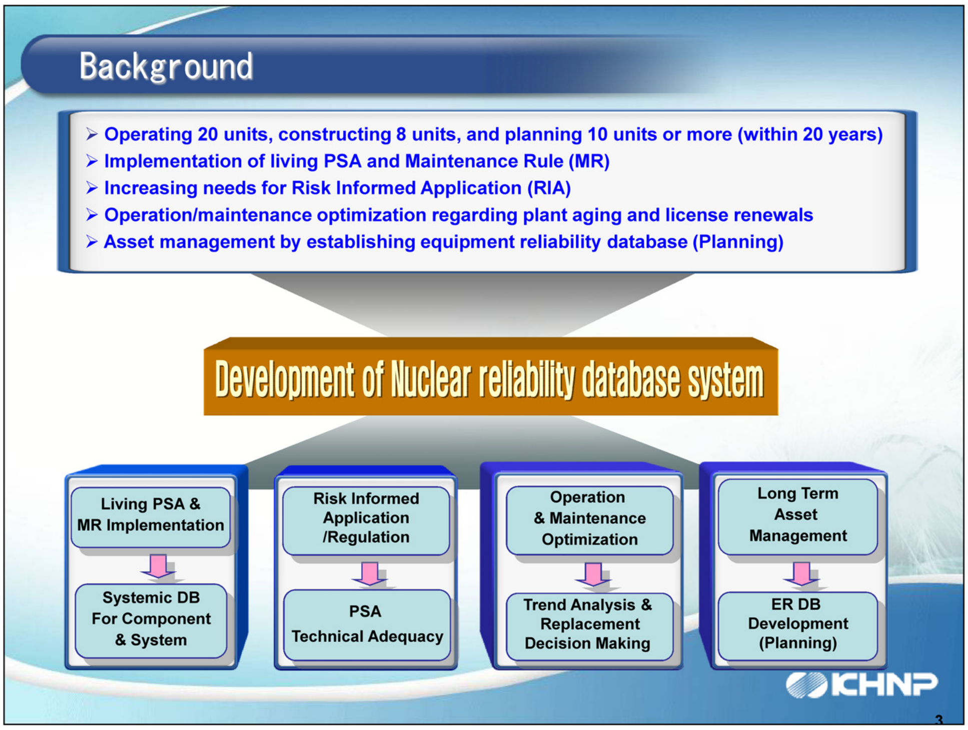
Background
➢ Operating 20 units, constructing 8 units, and planning 10 units or more (within 20 years)
➢ Implementation of living PSA and Maintenance Rule (MR)
➢ Increasing needs for Risk Informed Application (RIA)
➢ Operation/maintenance optimization regarding plant aging and license renewals
➢ Asset management by establishing equipment reliability database (Planning)
Development of Nuclear reliability database system
Living PSA & MR Implementation
Systemic DB For Component & System
Risk Informed Application /Regulation
PSA Technical Adequacy
Operation & Maintenance Optimization
Trend Analysis & Replacement Decision Making
Long Term Asset Management
ER DB Development (Planning)
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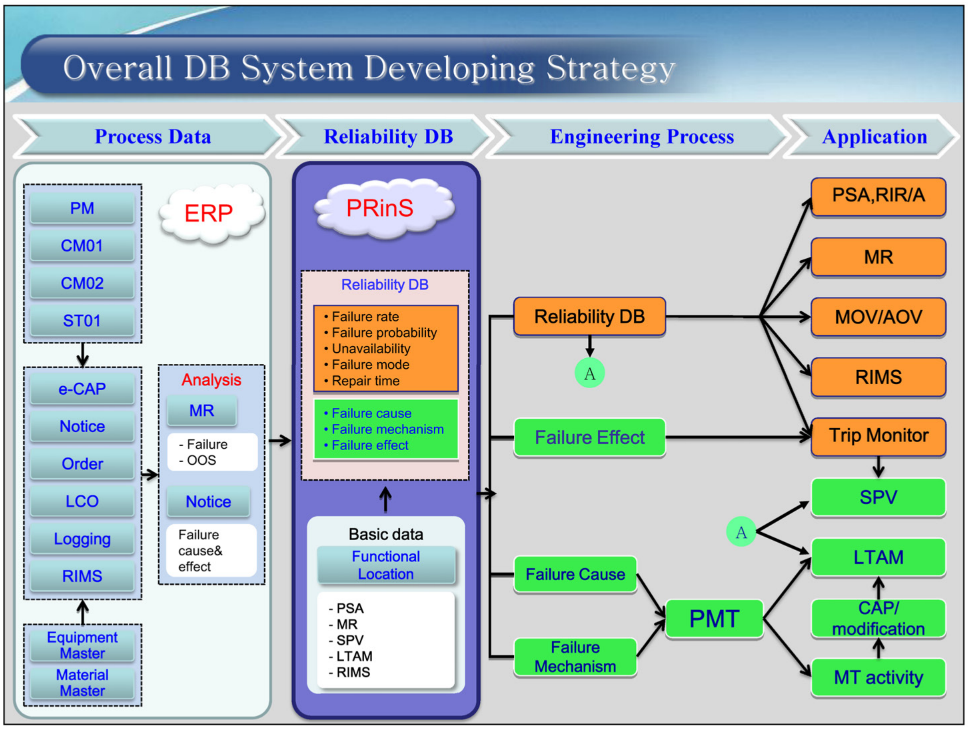
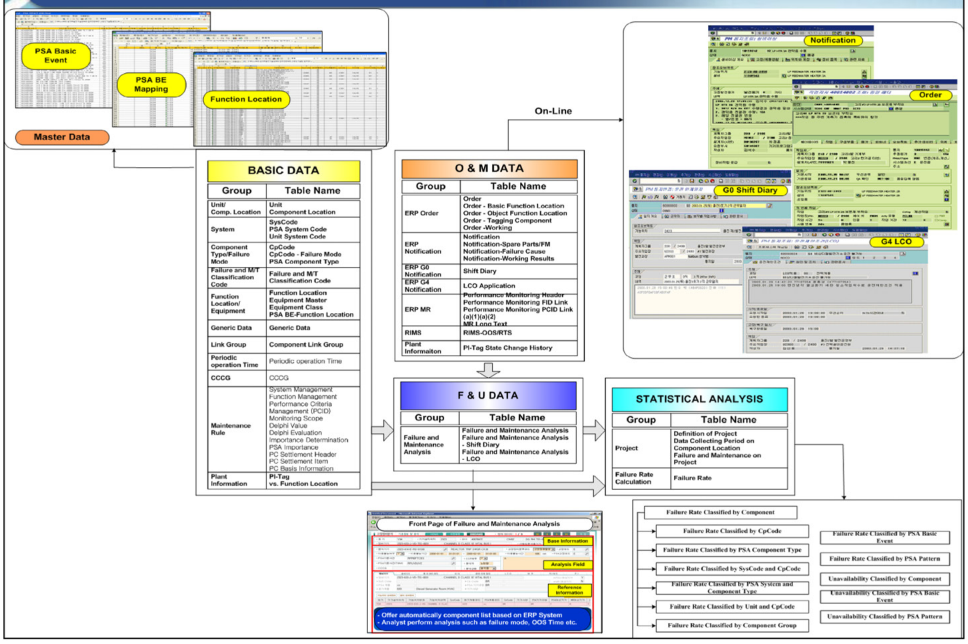
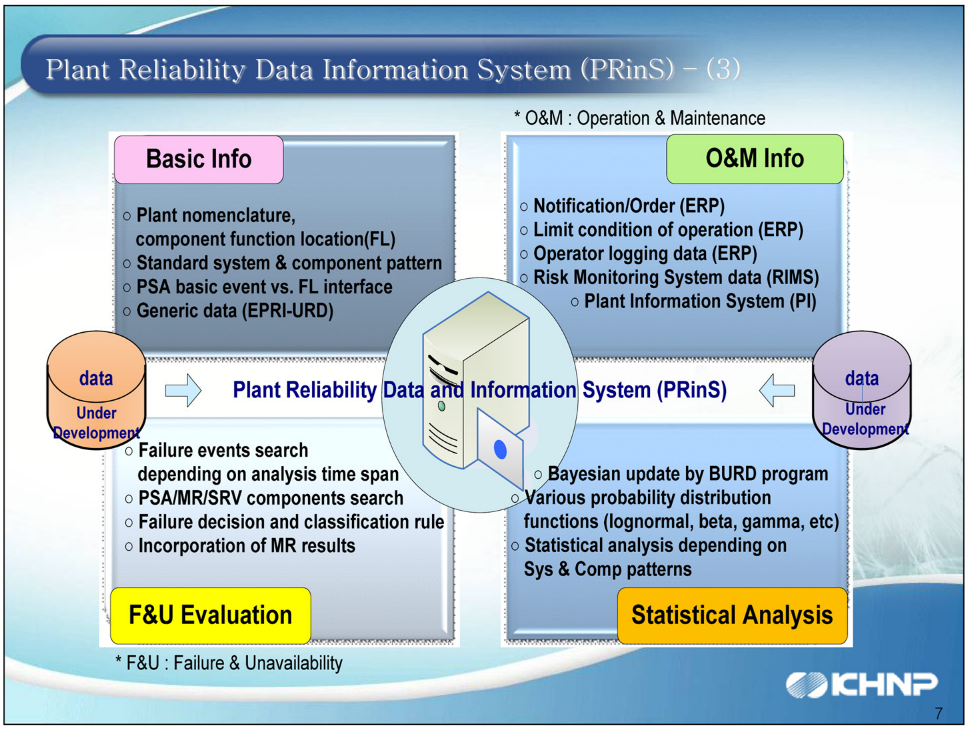

Overall DB System Developing Strategy
Process Data
Reliability DB
Engineering Process
Application
PM
CM01
CM02
ST01
ERP
e-CAP
Notice
Order
LCO
Logging
RIMS
Equipment Master
Material Master
Analysis
MR
- Failure
- OOS
Notice
Failure cause& effect
PRinS
Reliability DB
• Failure rate
• Failure probability
• Unavailability
• Failure mode
• Repair time
• Failure cause
• Failure mechanism
• Failure effect
Basic data
Functional Location
- PSA
- MR
- SPV
- LTAM
- RIMS
Reliability DB
A
Failure Effect
Failure Cause
Failure Mechanism
PMT
A
PSA,RIR/A
MR
MOV/AOV
RIMS
Trip Monitor
SPV
LTAM
CAP/ modification
MT activity

Plant Reliability Data Information System (PRinS) – (1)

Main Frame of Plant Reliability Data Information System

Plant Reliability Data Information System (PRinS) – (2)




Plant Reliability Data Information System (PRinS) – (3)
* O&M : Operation & Maintenance
Basic Info
○ Plant nomenclature, component function location(FL)
○ Standard system & component pattern
○ PSA basic event vs. FL interface
○ Generic data (EPRI-URD)
O&M Info
○ Notification/Order (ERP)
○ Limit condition of operation (ERP)
○ Operator logging data (ERP)
○ Risk Monitoring System data (RIMS)
○ Plant Information System (PI)
data
Under Development
Plant Reliability Data and Information System (PRinS)
data
Under Development
○ Failure events search depending on analysis time span
○ PSA/MR/SRV components search
○ Failure decision and classification rule
○ Incorporation of MR results
F&U Evaluation
○ Bayesian update by BURD program
○ Various probability distribution functions (lognormal, beta, gamma, etc)
○ Statistical analysis depending on Sys & Comp patterns
Statistical Analysis
* F&U : Failure & Unavailability
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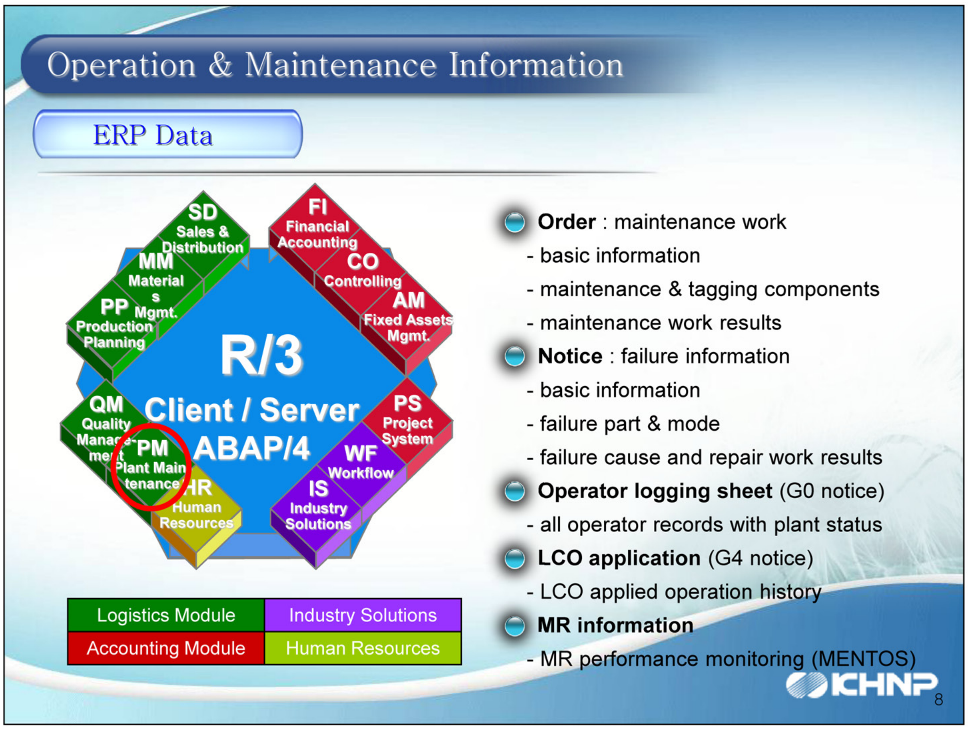
Operation & Maintenance Information
ERP Data
SD Sales & Distribution
MM Materials Mgmt.
PP Production Planning
FI Financial Accounting
CO Controlling
AM Fixed Assets Mgmt.
R/3
Client / Server
ABAP/4
QM Quality Management
PM Plant Maintenance
HR Human Resources
PS Project System
WF Workflow
IS Industry Solutions
Logistics Module
Industry Solutions
Accounting Module
Human Resources
Order : maintenance work
- basic information
- maintenance & tagging components
- maintenance work results
Notice : failure information
- basic information
- failure part & mode
- failure cause and repair work results
Operator logging sheet (G0 notice)
- all operator records with plant status
LCO application (G4 notice)
- LCO applied operation history
MR information
- MR performance monitoring (MENTOS)
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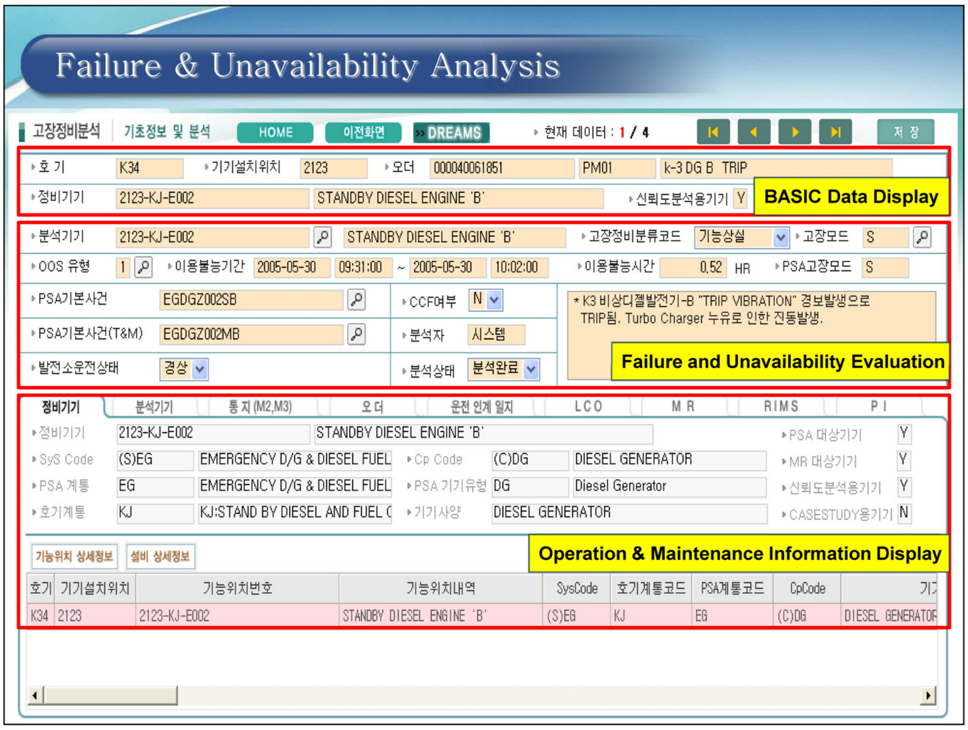
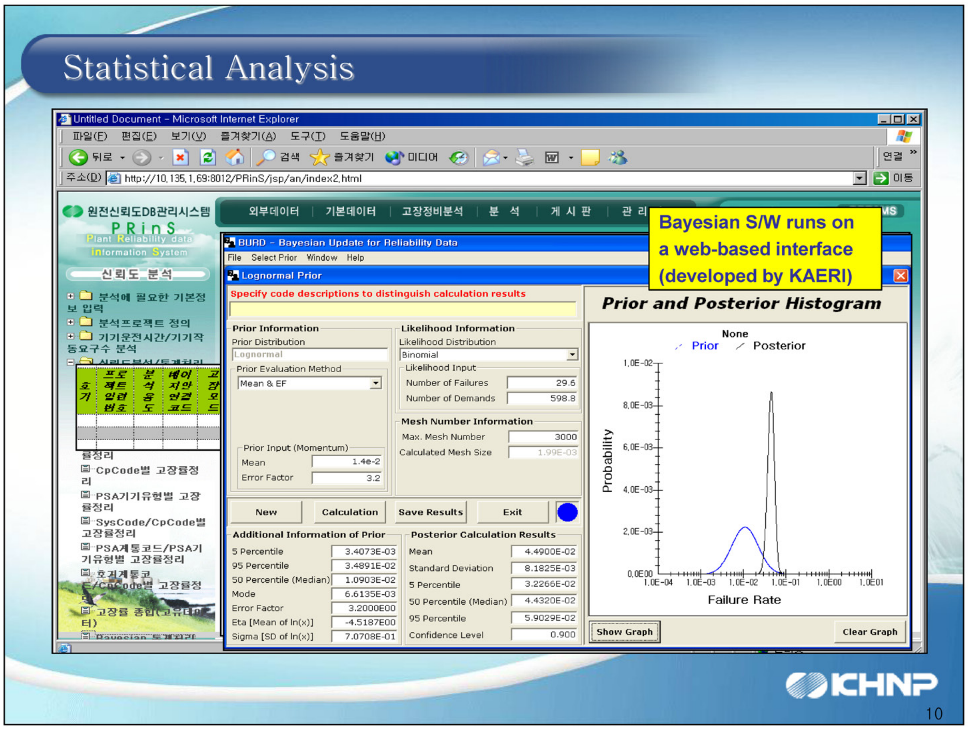

Failure & Unavailability Analysis
고장정비분석
기초정보 및 분석
HOME
이전화면
DREAMS
BASIC Data Display
Failure and Unavailability Evaluation
Operation & Maintenance Information Display


Statistical Analysis
Bayesian S/W runs on
a web-based interface
(developed by KAERI)
BURD - Bayesian Update for Reliability Data
Lognormal Prior
Prior and Posterior Histogram
Prior
Posterior
Probability
Failure Rate
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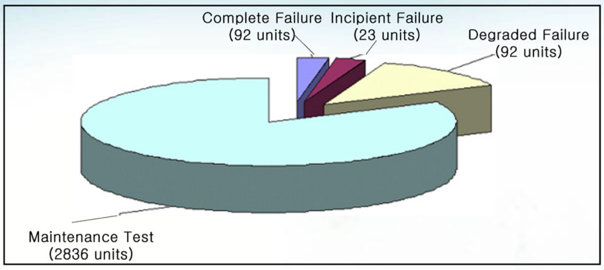
Data Application to PSA for Reference Plant



Name	Current CDF	Updated CDF	Decrease Rate
Component	4.75E-06	3.81E-06	19.7%
RPS/ESFAS		4.71E-06	0.73%
Component Unavailability		4.54E-06	4.24%
Total	4.75E-06	3.64E-06	23.4%

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Reliability DB Application Status

Unit	Function Location	Equipment Master	Basic Data	BD Upload	F&U Evaluation	PSA Application
K1	39,999	46,429	○	○	○	
K2	50,322	46,429	○	○	○	
K34	100,778	106,444	○	○	○	○
Y12	96,492	100,653	○	○	○	
Y34	96,821	119,999	○	○	○	○
Y56	117,487	122,342	○	○	○	
U12	76,440	82,012	○	○	○	
U34	107,720	114,708	○	○	○	○
U56	118,218	121,073	○	○	○	
W1	54,922	123,724	○	○	○	
W2	52,023	-	○	○	○	
W34	118,601	123,581	○	○	○	
SUM	1,029,823	1,107,394				

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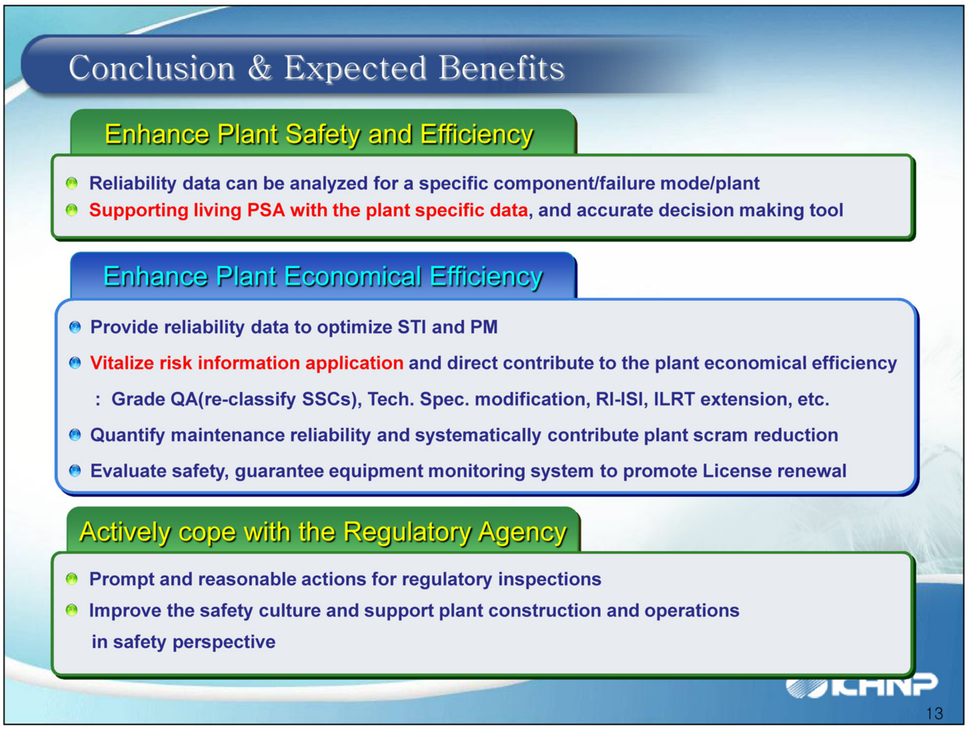
Conclusion & Expected Benefits
Enhance Plant Safety and Efficiency
Reliability data can be analyzed for a specific component/failure mode/plant
Supporting living PSA with the plant specific data, and accurate decision making tool
Enhance Plant Economical Efficiency
Provide reliability data to optimize STI and PM
Vitalize risk information application and direct contribute to the plant economical efficiency
: Grade QA(re-classify SSCs), Tech. Spec. modification, RI-ISI, ILRT extension, etc.
Quantify maintenance reliability and systematically contribute plant scram reduction
Evaluate safety, guarantee equipment monitoring system to promote License renewal
Actively cope with the Regulatory Agency
Prompt and reasonable actions for regulatory inspections
Improve the safety culture and support plant construction and operations
in safety perspective
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Thank you for Listening
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Ⅲ-A-3

10th Korea-Japan Workshop on PSA, May 18-20, 2009, Jeju, Korea

ISSA Technology

Safety Implications of Maintenance Rule with TMI-2 Perspective

Inn Seock Kim
ISSA Technology, Inc.
Maryland, USA

Nam Duk Suh and Sung Hun Oh
Korea Institute of Nuclear Safety (KINS)
Daejeon, Korea

Work performed under the sponsorship of KINS

PRESENTATION OVERVIEW

ISSA Technology

- **PART I: Introduction**
  - Status of Korea in Connection with Potential Introduction of Maintenance Rule
  - US Experience with MR Implementation
  - Should Korea Introduce Maintenance Rule?
  - Improving Regulatory Effectiveness through RIPBR
  - Evolution of the Maintenance Rule in the USA
- **PART II: Effects of Maintenance on Equipment and Safety with a TMI-2 Perspective**
- **PART III: Suggestions on the Implementation of Maintenance Rule in Korea**
- **PART IV: Concluding Remarks**

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PART Ⅰ

INTRODUCTION

Status of Korea in Connection with Potential Introduction of Maintenance Rule

- Korean Nuclear Safety Committee
  - 2002.12 : Recommended Introduction of Maintenance Rule in Korea
  - 2007.12: Required Investigation of the Necessity of Maintenance Rule through a Pilot Regulatory Evaluation

- Korea Hydro and Nuclear Company (KHNP)
  - Developed the MR Implementation Procedure
  - Performed Pilot MR Implementation in Ulchin 3&4, and Kori 3&4 Units

- Korea Institute of Nuclear Safety (KINS)
  - Evaluated the KHNP's MR Pilot Program at Ulchin 3&4 in Nov. 2008
  - Findings Include Deficiencies in the Determination of Scope, Performance Criteria, and Periodic Evaluation, etc.
  - Planning Regulatory Evaluation of Pilot MR Implementation at Kori 3&4

US Experience with MR Implementation

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- Both the NRC and the Nuclear Industry Tried to Improve Maintenance Especially Since the TMI Accident
- Plant Availability Considerably Increased Since Early 1990s
  - MR Initially in Effect from July 1991
  - ~70% in early 1990s to ~90% in early 2000s
- Average Number of Scrams Also Decreased Since Early 1990s
  - From almost 2 per year in early 1990s to ~0.5 per year in early 2000s
- Hence, MR is Considered as One of the First and Most Successful RIPBR Implementations in the USA

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Should Korea Introduce Maintenance Rule?

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- Regulatory Burden on the Side of KINS
  - Lack of Manpower in the Midst of Increasing Number of NPPs
- Economic Benefits Expected for KHNP Especially in Connection with On-Line Maintenance to be Allowed with Introduction of MR
- What are the Safety Implications of MR Implementation?
  - Korean NPPs Already in Operation with Good Performance without MR
  - If No Safety Benefits, No Need to Introduce MR in Korea

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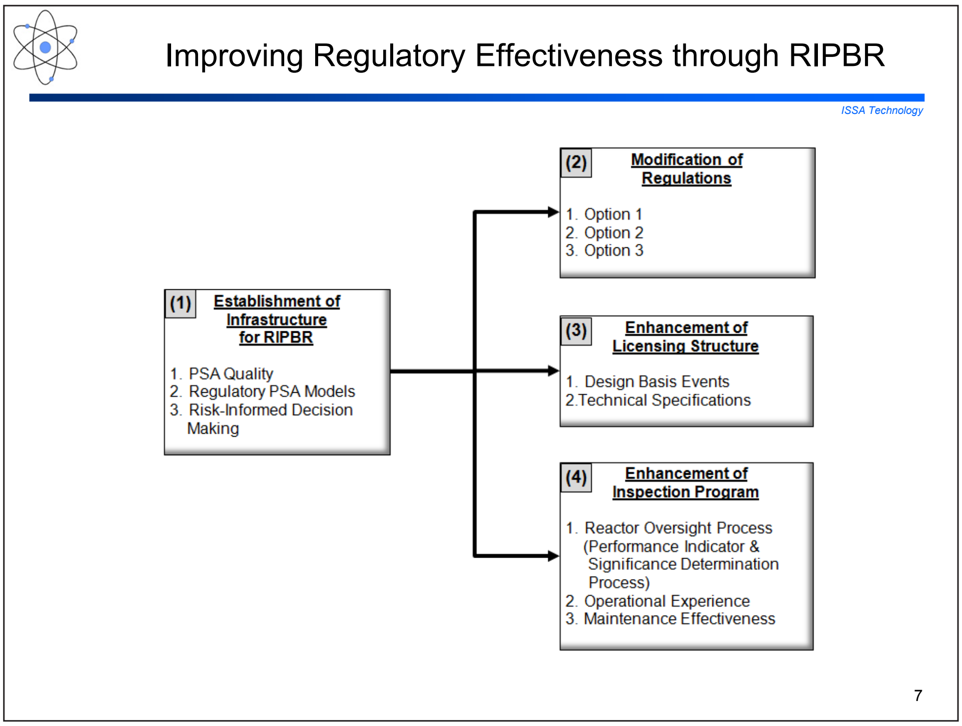
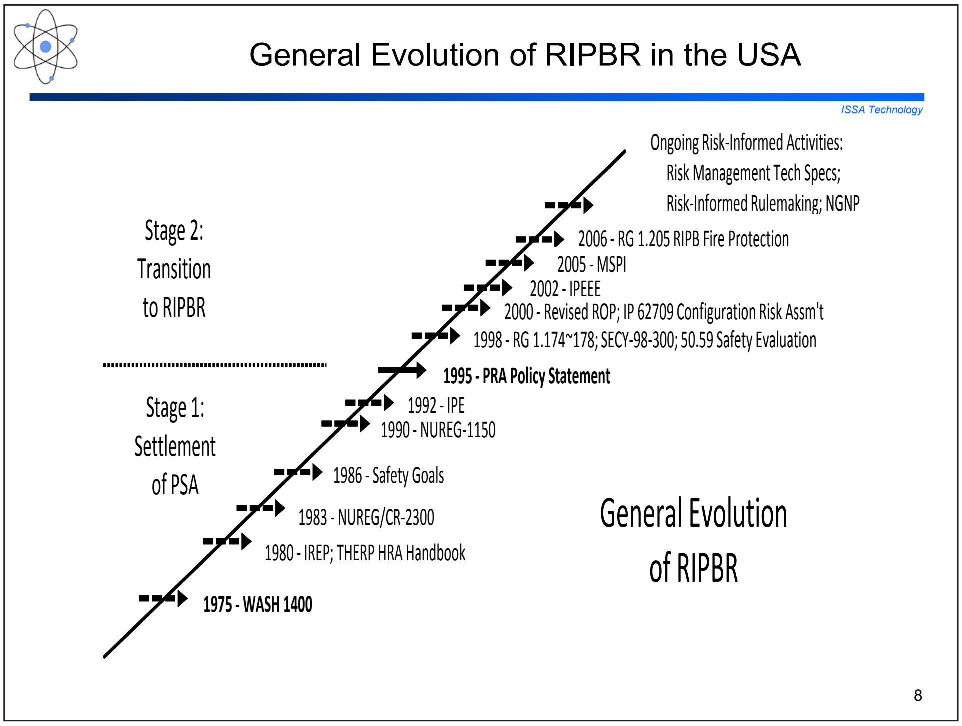

Improving Regulatory Effectiveness through RIPBR
ISSA Technology
(1) Establishment of Infrastructure for RIPBR
1. PSA Quality
2. Regulatory PSA Models
3. Risk-Informed Decision Making
(2) Modification of Regulations
1. Option 1
2. Option 2
3. Option 3
(3) Enhancement of Licensing Structure
1. Design Basis Events
2.Technical Specifications
(4) Enhancement of Inspection Program
1. Reactor Oversight Process (Performance Indicator & Significance Determination Process)
2. Operational Experience
3. Maintenance Effectiveness
7


General Evolution of RIPBR in the USA
ISSA Technology
Ongoing Risk-Informed Activities:
Risk Management Tech Specs;
Risk-Informed Rulemaking; NGNP
2006 - RG 1.205 RIPB Fire Protection
2005 - MSPI
2002 - IPEEE
2000 - Revised ROP; IP 62709 Configuration Risk Assm't
1998 - RG 1.174~178; SECY-98-300; 50.59 Safety Evaluation
1995 - PRA Policy Statement
1992 - IPE
1990 - NUREG-1150
1986 - Safety Goals
1983 - NUREG/CR-2300
1980 - IREP; THERP HRA Handbook
1975 - WASH 1400
Stage 2: Transition to RIPBR
Stage 1: Settlement of PSA
General Evolution of RIPBR
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Evolution of Maintenance Rule in the USA

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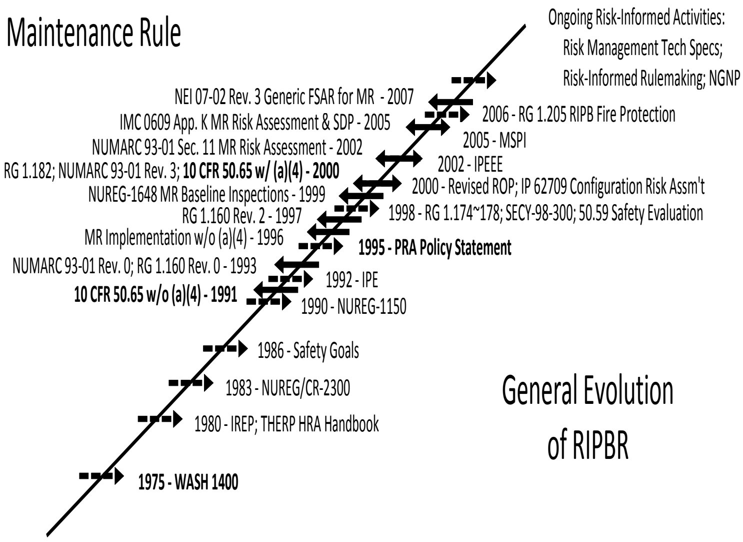

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PART II

EFFECTS OF MAINTENANCE ON EQUIPMENT AND SAFETY WITH A TMI-2 PERSPECTIVE

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Types of Maintenance and Safety Implications

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Type	Positive Impact	Negative Impact
Surveillance	o Detection of Equipment Failure or Degradation (Large)	o Potential Occurrence of Test-Caused Transients (Medium) o Equipment Unavailable during Surveillance Act (Medium)
Preventive Maintenance	o Prevention of Equipment Failure (Large)	o Equipment Unavailable during Preventive Maintenance o Potential Human Error (Medium)
Corrective Maintenance	o Restoration of Equipment Function (Large)	o Potential Human Error (Medium)
Post-Maintenance Testing	o Verification of Equipment Functionality (Large)	o Potential Occurrence of Test-Caused Transients (Medium) o Equipment Unavailable during Test (Medium)

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Effect of Maintenance on Equipment

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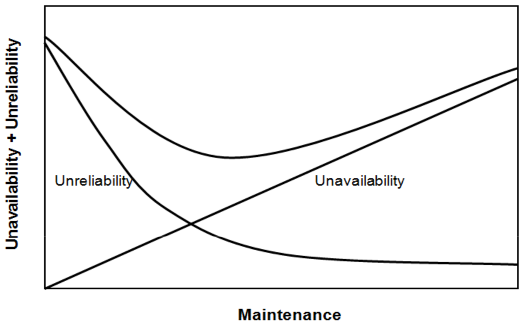

$$F(t) = 1 - e^{-\lambda t}$$

F(t) = Unreliability
λ = Failure Rate

$$U = \frac{d}{u + d}$$

U = Unavailability
u = Up Time
d = Down Time

D.A. Dube, "Technical Basis for the Mitigating Systems Performance Index (MSPI)," USNRC RES Seminar, February 2006

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Maintenance Errors Were Principal Causes of Several Major Accidents in a Wide Range of Technologies

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- The Apollo 13 Oxygen Tank Blow Out (1970)
- The Flixborough Cyclohexane Explosion (1974)
- **The Accident at the TMI-2 Nuclear Power Plant in Pennsylvania (1979)**
- The Crash of a DC10 at Chicago O'Hare (1979)
- The Explosion on the Piper Alpha Oil and Gas Platform in the North Sea (1988)
- The Explosion at the Phillips 66 Houston Chemical Complex in Pasadena, Texas (1989)
- A Blocked Pitot Tube Contributing to the Total Loss of a B757 at Puerto Plata in the Dominican Republic (1996)
- The Oxygen Generator Fire in the Hold of a DC9 over Florida (1996)

From "Managing Maintenance Error – A Practical Guide," James Reason , 2003

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Maintenance Errors Were Also Principal Causes in Risk-Significant Operational Events in NPPs

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- In 28 Events (76%) Out of 37 Risk-Significant Operational Events between 1992-1997 in the USA, Maintenance Error Was One of the Major Causes
  - Inadequate Maintenance and Maintenance Practices
  - Error in Work Package Development , QA and Use
  - Inadequate Post-Maintenance Testing
  - Inadequate Technical Knowledge

- ***Lessons Learned***
  - ***Maintenance Needs to be Performed Effectively***
  - ***Maintenance Errors Could Become Major Threats to Safety***
  - ***Maintenance Rule Could Help Reduce Maintenance Errors***

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Impact of Preventive Maintenance on Safety from an Accident Sequence Perspective

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IE * HW1 * HW2 * ..... * HE * RF

	Positive Impact	Negative Impact
Initiating Event	Avoid or Reduce Initiators (Large)	Cause Initiators to Occur (Small)
Hardware Failure	Increase Equipment Availability (Large)	Equipment Unavailable during Maintenance (Small)
Human Error	Decrease Human Error Potential (Large)	N/A
Recovery Failure	Increase Recovery Potential (Large)	N/A

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Impact of On-Line Maintenance on TMI-2 Accident

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	Likely Impact on TMI-2 Accident (Assumption of On-Line Maintenance)
Initiating Event	Prevention of the Loss of Main Feedwater and the Small-Break LOCA: The loss of main feedwater initiating event could have been prevented if the condensate polishing system had been maintained in a good condition by implementation of the maintenance rule. In addition, the induced occurrence of a small-break loss of coolant through the stuck-open PORV also could have been prevented if the cyclic operation of the PORV had been improved by an effective maintenance program.
Hardware Failure	o Prevention of Failure of the Condensate Polishing System: The two failures in the condensate polishing system (i.e., failure of the air-operated polisher bypass valve and a faulted valve on one of the polishers) might have been fixed by a good maintenance program. If it had been the case, the condensate pumps would not have lost their suction pressure because of the improved margin due to the bypass valve, even though some water had leaked into the instrument air system. o Prevention of the PORV Failure: If the reliability of the PORV (especially, the cyclic on-off operation) had been improved by a good maintenance program, the loss of main feedwater incident at TMI-2 would not have led to a small LOCA. It is very likely that the plant could achieve a safe shutdown condition following the loss of main feedwater initiator since the operators were well trained for this inicident and there was an event-oriented EOP for this specific transient.

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Impact of On-Line Maintenance on TMI-2 Accident

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	Likely Impact on TMI-2 Accident (Assumption of On-Line Maintenance)
Human Error	o Less Likely Error on the EFW Block Valve: The emergency feedwater system is a safety-related system; as such, the increase in risk would have been assessed for this system before performing the test. The operators might have been more careful to ensure the system functionality, and so they might not have committed a restoration error for the block valve. o Decreased Likelihood of Operator Misdiagnosis: One of the most important causes of the TMI-2 accident is that the control room operators misdiagnosed the condition of the primary system such that they thought it were going solid, although the coolant was being lost through the stuck-open PORV. If the EFW block valve was not mispositioned, then the operators might not have committed the misdiagnosis error because more time would have been available to them before core damage occurs.
Recovery Failure	Not directly relevant to the TMI-2 accident

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K-HFACS Analysis of the TMI-2 Accident

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Human Factor Layers	Human Factor Categories	Number of Instances and Description		
Unsafe Acts	Decision Error	2	2	persistent trial of removing clogged resin from the condensate polishing system
				misjudgment of the primary system status following the PORV stuck-open failure
Preconditions for Unsafe Acts	TE - Human-Machine Interface	2	12	poor alarm system design
				key indicators on the back panel
	TE - Procedural or Diagram Error	1		poor operating procedures
	TE - Equipment Deficiencies	3		faulty valve on one of the polishers that allowed water intrusion into IAS
				chronically leaking PORV or some other valve
				PORV failure to close upon a command signal
	TE - Configuration Error	1		misalignment of the EFW system
	Personal Readiness	1		insufficient training
	Adverse Mental State	1		extremely high stress especially due to too many alarms
	Physical/Mental Limitation	3		information overload, insufficient reaction time, and complexity
Unsafe Supervision	Inadequate Work Control	1	3	inadequate work package for the condensate polishing system
	Problem Identification & Resolution	2		lack of corrective action on two previous occasions of water introduction into IAS
				failure of operating experience feedback from the Davis-Besse event
Organizational Influences	Resource Management	2	2	maintenance and engineering backlog of the problem of water introduction to IAS
				inadequate training program

From I.S. Kim, "Feasibility Study for Development of Human Error Pattern Analysis Methodology for Operational Experience Feedback," Prepared for Korea Institute of Nuclear Safety, July 2008.

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Impact of Maintenance Rule on TMI-2 Accident

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Item	Content	Likely Impact of Maintenance Rule onTMI-2 Accident (Implementation Assumed)
(a)(1)	Performance Monitoring	o If an effective maintenance program had been in place at TMI-2, the loss of main feedwater due to failure of the condensate polishing system could have been averted because of the appropriate corrective action potentially taken as a result of the system degradation o If the operating experience of the PORV stuck failure at Davis Besse (a similar B&W plant) in September 1977 had been properly accounted for at TMI-2, the TMI-2 operators might have diagnosed the stuck-open failure of the PORV during the accident evolution
(a)(4)	Risk Evaluation	o The assessment of the increase in risk would have been carried out for the emergency feedwater system before performing the test, and the operators might have been more careful to ensure the system functionality o It is very likely that the risk impact might not have been evaluated for the condensate polishing system under (a)(4) because it is typically not a risk-significant system. However, more effective maintenance would have been performed for this system, because it is included in the scope of maintenance rule by item (b)(2)
(b)(1)	Inclusion of Safety-Related SSCs in (a)(1)	The emergency feedwater system would have been included in the scope of the maintenance rule at TMI-2 because it is safety related
(b)(2)	Non-Safety-Related SSCs to be Included in (a)(1)	The condensate polishing system would have been included in the scope of the maintenance rule at TMI-2 because its failure can cause a reactor scram or actuation of a safety-related system

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PART III

SUGGESTIONS ON THE INTRODUCTION OF MAINTENANCE RULE IN KOREA

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Suggestions on the MR Implementation in Korea

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- Safety Principles To be Observed When Voluntarily Entering LCOs for On-Line Maintenance (NRC Inspection Manual Part 9900)
  - On-Line Maintenance should be able to increase the system/plant reliability, and reduce SSC condition deficiencies
  - Should not abuse the allowance to perform a PM action on-line by repeatedly entering and exiting LCO action statements . The licensee should carefully plan the PM action to prevent such abuse.
  - In deciding whether to remove equipment from service, confidence in the OPERABILITY of the independent equipment that is redundant (or diverse) to the affected equipment should be high and risk insights should be used
  - While performing an on-line PM action, the licensee should avoid performing other testing or maintenance that would increase the likelihood of a transient
- Appropriate Implementation of (a)(4) Vital for MR Success
  - Configuration Risk Management with Technically Adequate PRA Models

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PART IV

CONCLUDING REMARKS

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Concluding Remarks

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- The Essence of Maintenance Rule is to Keep the Plant in a Healthier Condition Overall As Compared to Otherwise Possible
- Utility Pilot Implementations of Maintenance Rule Are Underway in Korea
- Considerable Efforts Expected from Not Only the Utility But the Regulatory Side As Well
- Confirmation of Benefits Through Pilot Implementations Needed
  - Increase in Equipment Reliability, Reduction in the Number of Scrams
  - Contribution to PSA Quality, Increase in Operational Flexibility
  - Better Configuration Management, Risk Reduction
  - Reduced Maintenance Errors, Enhanced Safety Culture, etc.

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감사합니다

QUESTIONS?

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III-A-4

The 10th Korea-Japan PSA Workshop,
May 18-20, 2009, Jeju

Development of Regulatory PSA Model (MPAS) in KOREA

2009. 5. 19

Jin Hee Park
Integrated Risk Assessment Division
Korea Atomic Energy Research Institute

한국원자력연구원
Korea Atomic Energy Research Institute

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Introduction

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❑ Background

- The regulatory paradigm change has been worldwide trend from Prescriptive Regulation to Risk-informed & Performance-based (RIPB)
- Regulatory Body (MEST & KINS) has prepared the RIPB regulation frame work and several prototype RIPB regulation has applied for testing in Korean NPPS
- This project initiated to provide the quantitative Index for RIPB regulation
  - National long-mid term R&D project for new regulation policy

❑ Scope and schedule of this project

- Level-1 PSA (full power, internal events) Model development for each reactor type
  - Modify the utility PSA model not create new model
- Regulatory PSA models for Each type of Reactor has being developing a year
  - WH 900MWe(2007) & WH 600MWe (2008) type Model developed
  - FR 900MWe Model is developing(2009)
  - CANDU 600MWe(2010), OPR-1000(2011) type would be developed
  - Level-2, external PSA Model development (planning)

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Introduction

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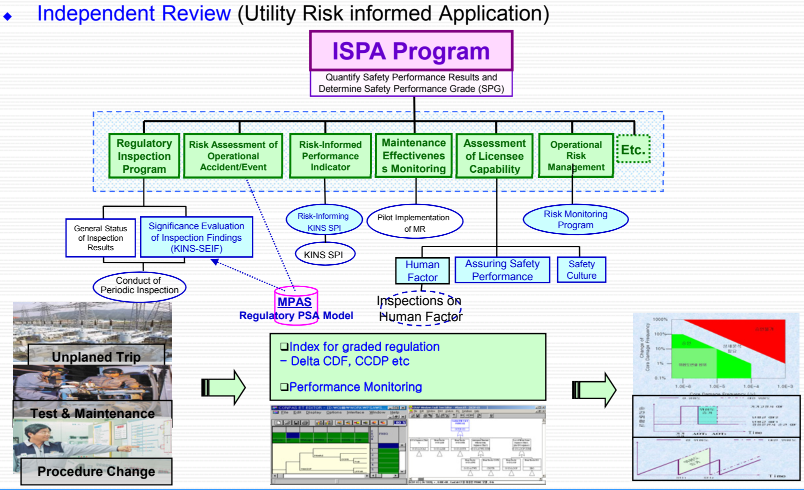
❑ Roles of MPAS model in RIPB regulation framework

- To support Decision Making for Regulatory Body
  - Quantitative Index for Graded Regulation for each Unit (based on Risk & Performance)
  - Independent Review (Utility Risk informed Application)



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Model Development 5

❑Main Focus for Regulatory PSA Model

- Realistic & conservative PSA Model
  - As-built & as operated condition
  - Best estimated T/H analysis
  - HRA Standardization
- Consistency among each unit type
  - Identical approach
    - Accident Sequence Analysis
    - Assumption
    - CCF
    - HRA
    - IE, Component Data (under developing)

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Model Development 6

❑Review the utility PSA model for each reactor type

- To Identify the items be needed improvement for MPSA model
  - **Plant operation information**
  - **Design document**
  - **EOP & AOP**
  - **Design change items**
  - **Interview with plant operation staff**
- Develop list of items be needed improvement for esch PSA element

Element	Items for Improvement	remarks
IE	Transfer between IE after transient	Clear transfer logic develop in quantification
	ATWS IE analysis	UET reanalysis & Clear transfer in Accident analysis
	Specific IE analysis	LOOP, Transient IE reanalysis using Korean Specific Data
AS	SGTR Accident analysis	Reanalysis to consistent with Regulatory PSA models
	Seal LOCA reanalysis during LOCCW, SBO	Using recent information
	Respect the plat design change items (AAC D/G)	Reanalysis needed
SC	Best estimated calculation	BE Code for T/H analysis • Reanalysis for LOCA, SBO, ATWS and so on

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Model Development

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❑Redefine the success criteria for each Event tree

- To respect more realistic plant response for accident sequence
- To support the HRA analysis
- Recalculation using BE code
  - MAPP for utility PSA model
  - MARS for regulatory PSA model

ET	Accident Scenario	Time to Core damage (min)	
		MAAP model	MARS model
Small LOCA	1' pipe rupture (Cold Leg) w/o recovery action	105.1	125.3
	21' pipe rupture (Cold Leg) w/o recovery action		37.8
SGTR	1 tube rupture w/o recovery action	156.6	208.3
SBO	1. Plant response after AFWS operation for 4 hours during SBO accident	147.8	435.8
	2.Seal LOCA analysis for leak rate **180, 480 gpm** during SBO	-	**74.8** **61.3**
General Transient	2. Total Loss of secondary cooling	85.9	57.8
	2.1 elapse time for Feed & Bleed during Transient	-	75.8

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Model Development

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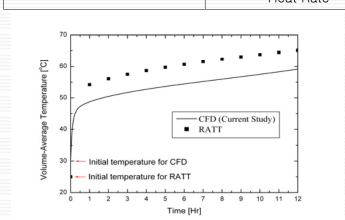
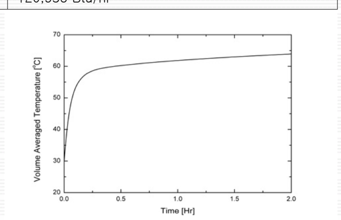
❑Re-calculation for room Heat-up rate

- **To assess the component operability during Loss of Room cooling**
  - **Perform walkdown to respect room real condition and realistic model**
  - **Simulated two sample cases using CFD (Computational Fluid Dynamics) code**
- **CSS, AFW Pump Room heat-up rate not exceeded the limiting room temperature**

Room condition		
CS Pump Room	Volume	4,680 ft^3 ($13.0 \times 18.0 \times 20.0$ ft^3)
	Heat Rate	61,800 Btu/hr
AFW MD Pump Room	Volume	4,312 ft^3
	Heat Rate	120,956 Btu/hr



CS Pump room Heat-up rate



AFWS pump room Heat-up rate

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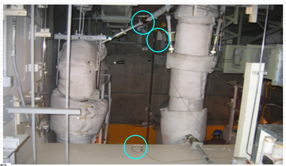
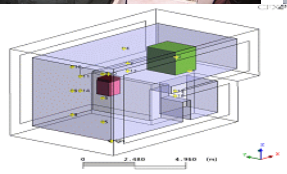
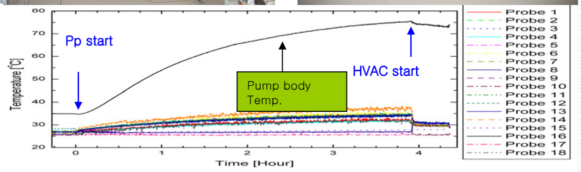
Model Development 9

❑On-site test to measure pump room heat-up rate

- To verify the simulation using CFD code
  - Measure the heat-up rate at 18 T/Cs for 4 hours without HVAC
  - Unexpected result is identified (draft result)
    - Room temperature heat-up rate is not exceeded to the room temperature limit
    - Pump body temperature raised rapidly (un-expected situation & not considered in PSA)
      - It could damage insulator of pump motor windings
    - Long-term research item (under calculation in detail)

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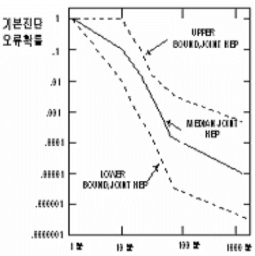
Model Development 10

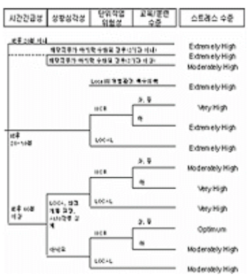
❑Re-analysis HRA for MPAS model

- K-HRA calculator
  - KAERI developed cooperated with KINS/KHNP/KOPEC
  - Standard HRA methodology
- Operation personnel interview

K-HRA calculator flow chart



Basic HRA value



Decision Tree

작업성격	스트레스 수준	기본 오류 확률 (mean)	THERP 단위직무 HEP 분포 (median)
Simple Response	Low	0.002	
	Optimum / Moderately High	0.001	
	Very High / Extremely High	0.003	
Step-by-Step	Low	0.01	0.001~0.01
	Optimum	0.005	0.0005~0.005
	Moderately High	0.01	0.001~0.01
	Very High	0.02	
	Extremely High	0.05	0.0025~0.025
Dynamic	Low	-	0.001~0.01
	Optimum	0.01	0.0005~0.005
	Moderately High	0.03	0.0025~0.025
	Very High	0.08	
	Extremely High	0.25	0.25

Input for Decision Tree

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Model Development

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❑Comparison table for HRA

Name of HRA event	Event description		Utility Model	MPAS Model
HRTCFRPC1-10	Core cooling recovery	OPERATOR FAILS TO RECOVER CORE COOLING RE C-1	1.15E-03	1.64E-02
HRTCFRPC1-14	LPSI operation	OPERATOR FAILS TO MAKE-UP RCS INVENTORY (LOCCW)	2.56E-02	1.22E-02
HRFRPH1-10S	Feed & Bleed	OPERATOR FAILS TO INITIATE FEED-AND- BLEED (Small LOCA/SGTR) RE H-1	6.77E-03	1.15E-02
HRTAFRPS1-4	Emergency Boration	OPERATOR FAILS TO INJECT BORATION WATER USING CHRAGING PUMP	2.54E-03	1.63E-03
HRESP1_3-3_4	High Pressure C/L HPSI Recirculation	Operator fails to initiate high pressure cold leg recirculation (HPCR-SLOCA) SUP 1-3	2.13E-03	1.61E-03
HRESP1_3-3_4	High Pressure C/L HPSI Recirculation	Operator fails to initiate high pressure cold leg recirculation SUP 1-3	2.13E-03	1.61E-03

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Model Development

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❑MPAS model development

- **Interview with plant personnel (MCR operator) in detail**
  - **Compare accident sequence with plant EOP & AOP**
  - **Compare system fault tree with as-built & as-operated condition**
- **Final comparison results**
  - **To apply into MPAS model**

Review item	Model developer's opinion	Plant's opinion	remarks
Loss of Vital Bus	Plant response confirm	RCP trip occur	Add in IE
ATWS	UET analysis needed	agree	UET reanalysis needed
General TR	Considering PORV stuck open during transient	agree	Add in ET
SBO	Considering Specific data & Seal LOCA	agree	Add in ET

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Summary

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❑Two types of MPAS model developed

- WH 900Mwe & WH 600MWe Type reactor
- FR 900MWe type is under developing

❑Future Tasks

- Model Standardization is still demanded
  - to provide more reasonable & fair index for degraded regulation
    - Number of I.E, Component failure probability, Model depth & etc.
- User interface Module
  - for inspector's friendliness
- Level-II & External Events Model
  - Extend the MPAS model scope

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Session III-B

Fire PSA

Summary of Session III-B: Fire PSA

Chair: Tsuyoshi UCHIDA (JNES), Moon-Hak JEE (KEPRI)

1. “Development of fire PSA database system” presented by Dong-Kyu Kim at KOPEC

Dr. Kim’s presentation for the topic related with fire PSA database system development proposes an advanced fire PSA approach for new nuclear power plants by application of fire PSA database system. This system accommodates the information of plant partitioning, fire frequencies, PSA equipment, and cable arrangement. In addition, fire PSA D/B can utilize a couple of KOPEC’s A/E software such as IPIMS, KCMS, OPMS. We are sure that the fire PSA DB system can reduce a lot of time consuming for fire-induced PSA work and enhance the efficiency of analysts’ solution for fire safety analysis.

2. “Development of the fire PSA methodology and the fire analysis computer code system” projected by Katsunori OGURA at JNES

It is quite impressive that JNES has developed its own fire PSA methodology to figure out the CDF status with the risk-significant fire scenarios. It was also presented that JNES has integrated the zone and CFD fire model that was introduced as CFAST/FDS network. In Japan, JNES conducts fire PSA for LPSD as well as full-power operation nuclear power plants. It is expected that at next KJPSA meeting the verification and validation result of CFAST/FDS network and the development status of seismic-induced fire PSA methodology will be introduced.

3. “A comparative study of two quantification algorithms and importance measures in fire PRA model” by presenter, Kil-Yoo KIM at KAERI

Dr. Kim presented improved algorithm for the importance measure in FV for the fire PRA model. His conclusion is that in FV calculation the failure probability of component K due to target room fire should be used as determinant factor to choose quantification method 1 that is conventional approach or quantification method 2 that is the advanced calculation. This presentation means that it is prequisite to find the failure probability of each risk-significant component in target room and it is expected the certainty of fire CDF quantification can be improved a great deal.

4. Improved fire-PSA with quantitative fire risk assessment by Moon-Hak JEE at KEPRI

The final presenter was Mr. Jee who is professional engineer of fire protection. According to his material, the contribution of fire-induced CDF to CDF in toal is very high due to the conventional approach that is prone to conservatism and to cover uncertainty in fire-induced CDF factors such as fire frequency, severity factor, non-suppression probability, and CCDP itself. He proposed that more fire compartments can be screened and fire-induced CDF will be declined by use of performance-based fire modeling and the recent fire PSA methodology. Particularly, the active fire suppression strategy with the ventilation-controlled fire and the advanced fire fighting strategy with purpose to control fire CDF at normal power operation as well as shutdown period was suggested.

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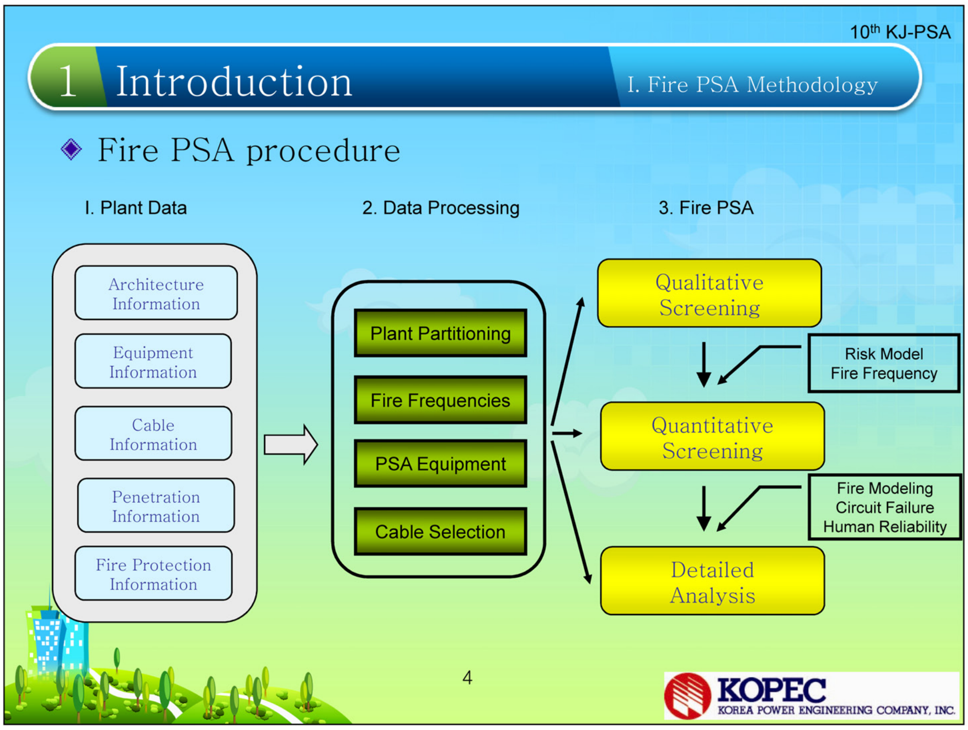
Ⅲ-B-1


Development of Fire PSA Database System
10th Korea-Japan Joint Workshop on PSA
2009. 05. 19
Dong-kyu Kim, Hak-Kyu Lim, Kwang-Nam Lee
Integrated Safety Assessment Group
KOPEC
KOPEC
KOREA POWER ENGINEERING COMPANY, INC.


Contents
I Fire PSA Methodology
II Fire PSA Database
III Development Plan
IV Conclusion


I. Fire PSA Methodology


10th KJ-PSA
1 Introduction
I. Fire PSA Methodology
Fire PSA procedure
I. Plant Data
2. Data Processing
3. Fire PSA
Architecture Information
Equipment Information
Cable Information
Penetration Information
Fire Protection Information
Plant Partitioning
Fire Frequencies
PSA Equipment
Cable Selection
Qualitative Screening
Risk Model
Fire Frequency
Quantitative Screening
Fire Modeling
Circuit Failure
Human Reliability
Detailed Analysis
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10th KJ-PSA

2 Conventional Data Processing Problems

I. Fire PSA Methodology

- Too many data to handle by hand
  - Repeated compilation for Fire PSA
    - Fire Frequency Calculation
    - PSA Equipment Selection
    - PSA Cable Selection
    - Damaged Target Selection
  - Drawbacks
    - Low Efficiency
    - High Potential Human Error
    - Low Traceability

➔ Need for Fire PSA Database System

5

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II. Fire PSA Database

10th KJ-PSA

1 General

II. What is Fire Database

- Purpose
  - To support the assessment of potential equipment failures for fire scenarios quickly and accurately.
    - Total failure of all circuits in a fire compartment
    - Failure of cable within a specific raceway
    - Failures based on specific equipment failure mode
- Manual compilation is nearly prohibited. (NUREG/CR-6850)
  - Due to the low efficiency
  - Due to human errors that can be introduced while repetitively manipulating large amounts of data.
- Fire PSA Database will enable analyst to conduct the necessary analyses efficiently and accurately.

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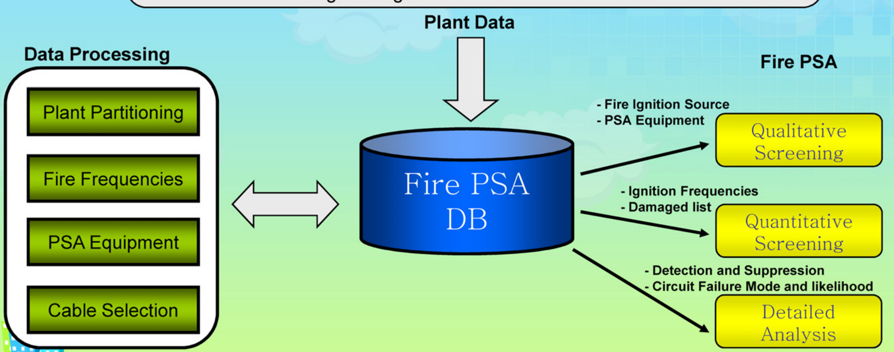
10th KJ-PSA

2 Concept

II. What is Fire Database

- Database in Fire PSA

KOPEC's A/E Software
IPIMS : Integrated Plant Information Management System
- KCMS : KOPEC Cable Management System
- OPMS : Opening And Penetration Management System
- EDB : Engineering Database



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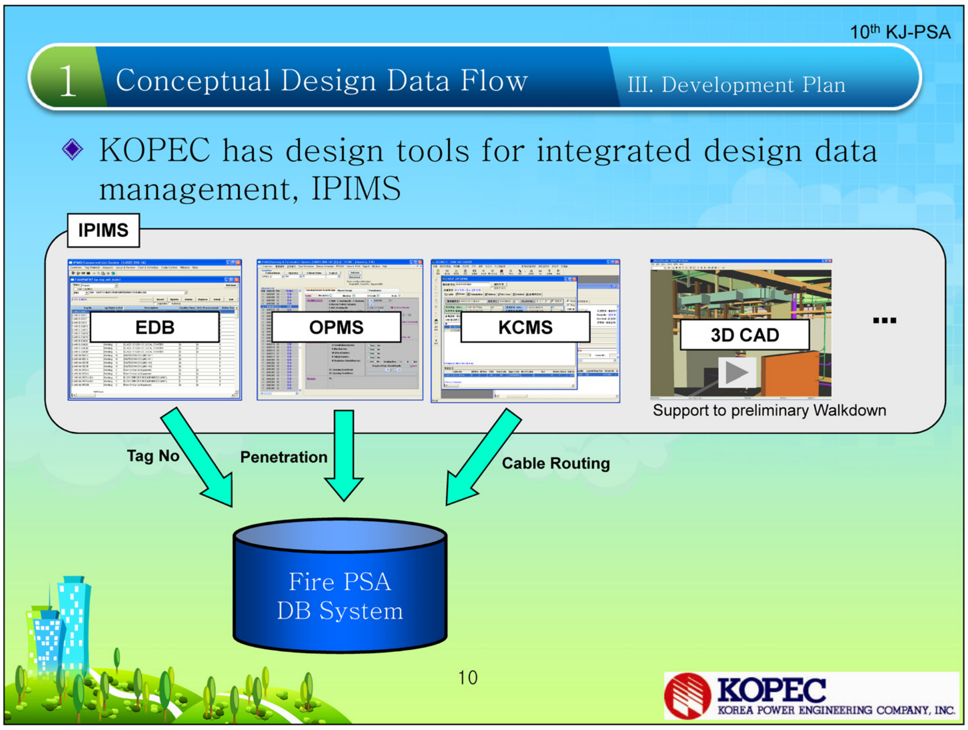
III. Development Plan


10th KJ-PSA
1 Conceptual Design Data Flow
III. Development Plan
◆ KOPEC has design tools for integrated design data management, IPIMS
IPIMS
EDB
OPMS
KCMS
3D CAD
…
Support to preliminary Walkdown
Tag No
Penetration
Cable Routing
Fire PSA
DB System
10
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10th KJ-PSA

2 Development Scopes (1)

III. Development Plan

	Description	KOPEC Fire PSA DB System	NUREG/CR-6850 Requirements
Task 1	Plant Partitioning	◎	◎
Task 2	Component Selection	◎	◎
Task 3	Cable Selection	◎	◎
Task 4	Qualitative Screening	◎	◎
Task 5	Risk Model	○	X
Task 6	Fire Ignition Freq.	◎	X
Task 7	Quantitative Screening	◎	◎
Task 8	Scoping Fire modeling	○	X
Task 9	Circuit Failure Analysis	◎	◎

◎ : Fully support
○ : Partial support

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10th KJ-PSA

2 Development Scopes (2)

III. Development Plan

	Description	KOPEC Fire PSA DB System	NUREG/CR-6850 Requirements
Task 10	Circuit Failure Mode Likelihood	◎	◎
Task 11	Detailed Fire Modeling	X	X
Task 12	Human Reliability	X	X
Task 13	Seismic-Fire interaction	X	X
Task 14	Fire-PSA Quantification	○	X
Task 15	Uncertainty	X	X
Task 16	Documentation	○	X
Task 17	Walkdown	◎	X
Task 18	Fire PSA Database	◎	◎

◎ : Fully support
○ : Partial support

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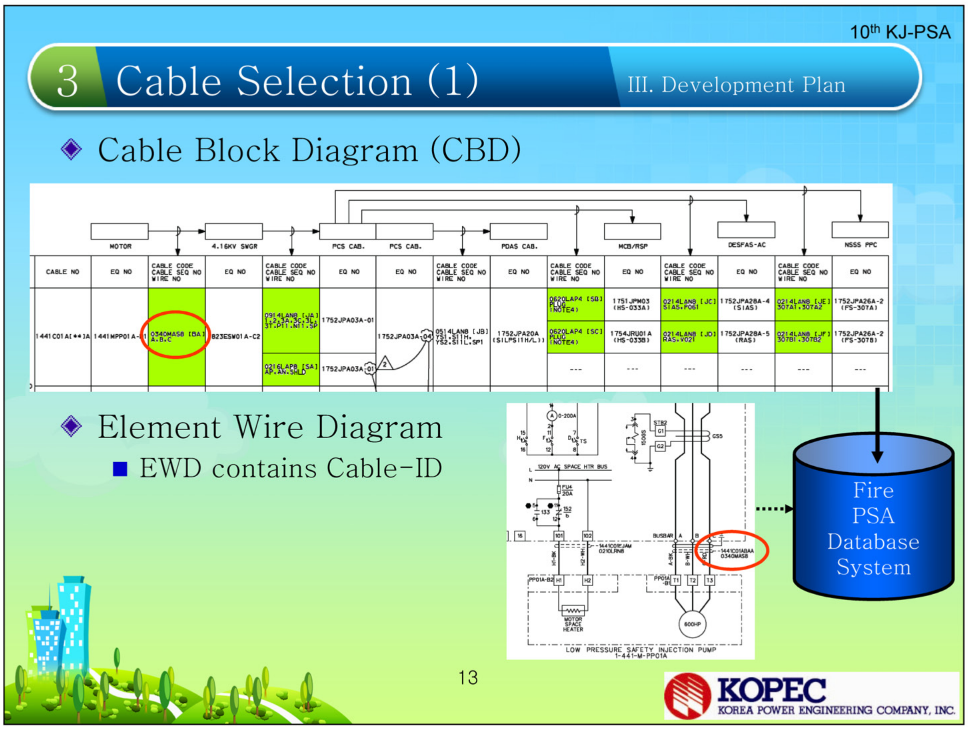
10th KJ-PSA
3 Cable Selection (1)
III. Development Plan
Cable Block Diagram (CBD)
Element Wire Diagram
EWD contains Cable-ID
Fire PSA Database System
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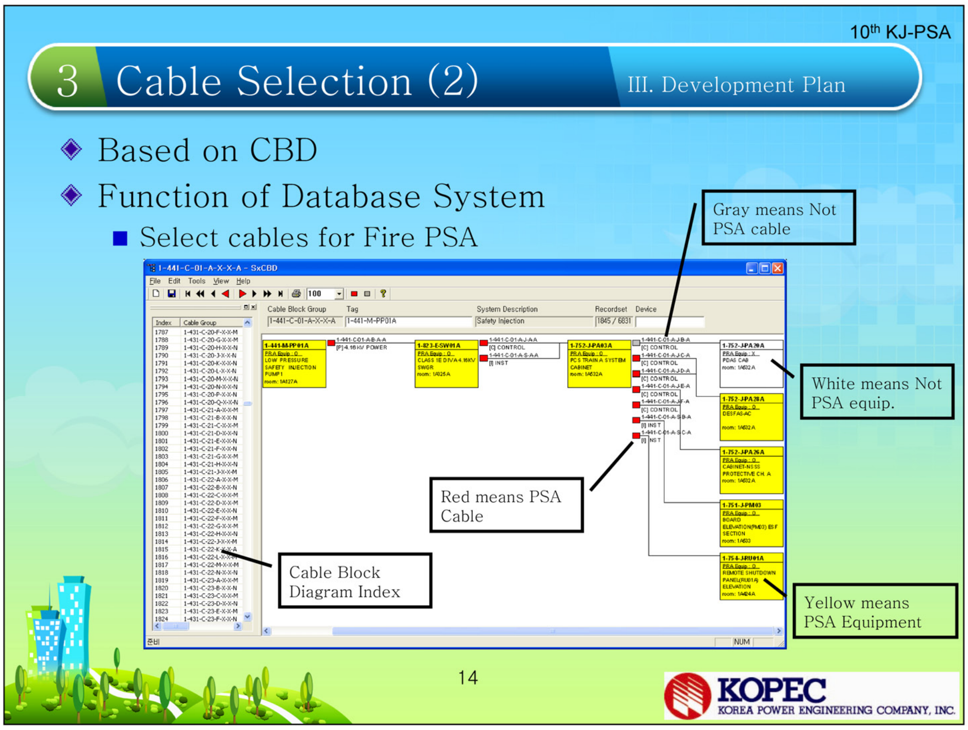
10th KJ-PSA
3 Cable Selection (2)
III. Development Plan
Based on CBD
Function of Database System
Select cables for Fire PSA
Gray means Not PSA cable
White means Not PSA equip.
Red means PSA Cable
Cable Block Diagram Index
Yellow means PSA Equipment
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10th KJ-PSA

4 Fire Frequency Calculation

III. Development Plan

- Base on EDB (Engineering Database)
  - Equipment
    - Count by compartment
    - Apportion of ignition frequencies (Bin)
  - Location weighting factor (ex: Battery room)
    - Check all compartments
- Function of Database System
  - Calculate Fire Frequency in a Fire Compartment
  - If enough data for equipment and compartments are given, fire ignition frequency is calculated automatically.

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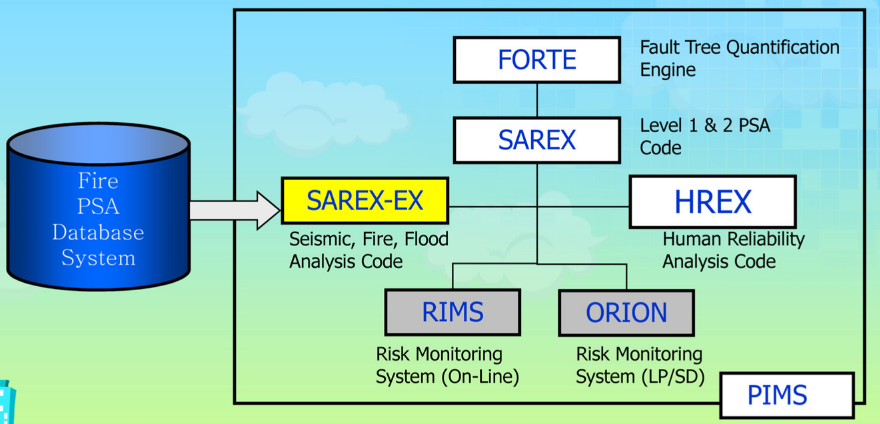
10th KJ-PSA

5 Connection with PSA Software

III. Development Plan

- PSA Software developed by KOPEC



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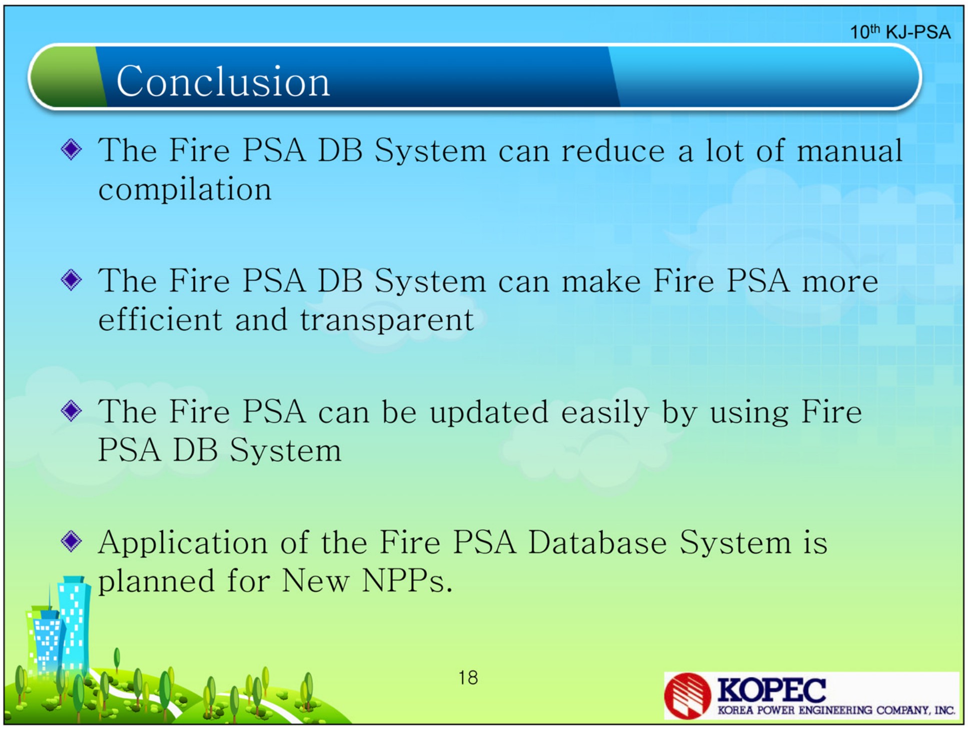
IV. Conclusion


10th KJ-PSA
Conclusion
◆ The Fire PSA DB System can reduce a lot of manual compilation
◆ The Fire PSA DB System can make Fire PSA more efficient and transparent
◆ The Fire PSA can be updated easily by using Fire PSA DB System
◆ Application of the Fire PSA Database System is planned for New NPPs.
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Thank you

Ⅲ-B-2

Development of the Fire PSA Methodology and the Fire Analysis Computer Code System

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

Katsunori OGURA, Tomomichi ITO, Tsuyoshi UCHIDA and Yusuke KASAGAWA

Probabilistic Safety Assessment Group
Nuclear Safety Analysis and Evaluation Office
Nuclear Energy System Safety Division

Incorporated Administrative Agency
Japan Nuclear Energy Safety Organization

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JNES

Outline

1. Introduction
2. Fire PSA Methodology
   (1) Outline of Fire PSA Methodology
   (2) Key Analysis Technique
3. Preliminary Fire PSA results
4. Development of Fire Analysis Code System
5. Conclusions & Future Plan

JNES

2

1. Introduction

- JNES has developed Fire PSA methodology to understand the CDF level with dominant fire scenarios.
  - The following methods have been developed in order to quantify CDFs due to fire scenarios established :
    - ✓ Quantification of fire frequencies based on operational experiences of NPPs in Japan with Baysian update technique
    - ✓ Identification of initiating events such as loss of secondary side cooling, loss of AC power, loss of coolant accident with Event Tree technique
    - ✓ Development of fire severity factors with fire propagation analyses
    - ✓ Integration of elements above and quantification of fire scenarios
  - JNES has implemented the following Fire PSA:
    - ✓ Fire PSA for typical PWRs and a BWR during power operation
    - ✓ Trial Fire PSA for a typical 1100MWe PWR during low power and shutdown operation (Limited Scope)
  - Seismic-induced fire PSA method is being developed.

JNES 3

2. Fire PSA Methodology

(1) Outline of Fire PSA Methodology

- The methodology is composed of “Spatial Interaction Analysis” to identify fire zones, “Quantitative Screening” under the conservative manner and “Detailed Analysis” to quantify fire scenarios. [Next Page]
- The risk significant fire scenarios are quantified as follows;

$$CDF_{fire} = \sum_i (F_{fire} \times P_{IE} \times F_s (\times P_{prop}) \times CCDP)_i$$

CDF_{fire} = Total CDF due to the risk significant fire scenarios

F_{fire} = Fire frequency for the risk significant fire scenario "i"

P_{IE} = Probability of initiating event in case of fire scenario "i"

F_s = Fire severity factor in case of fire scenario "i"

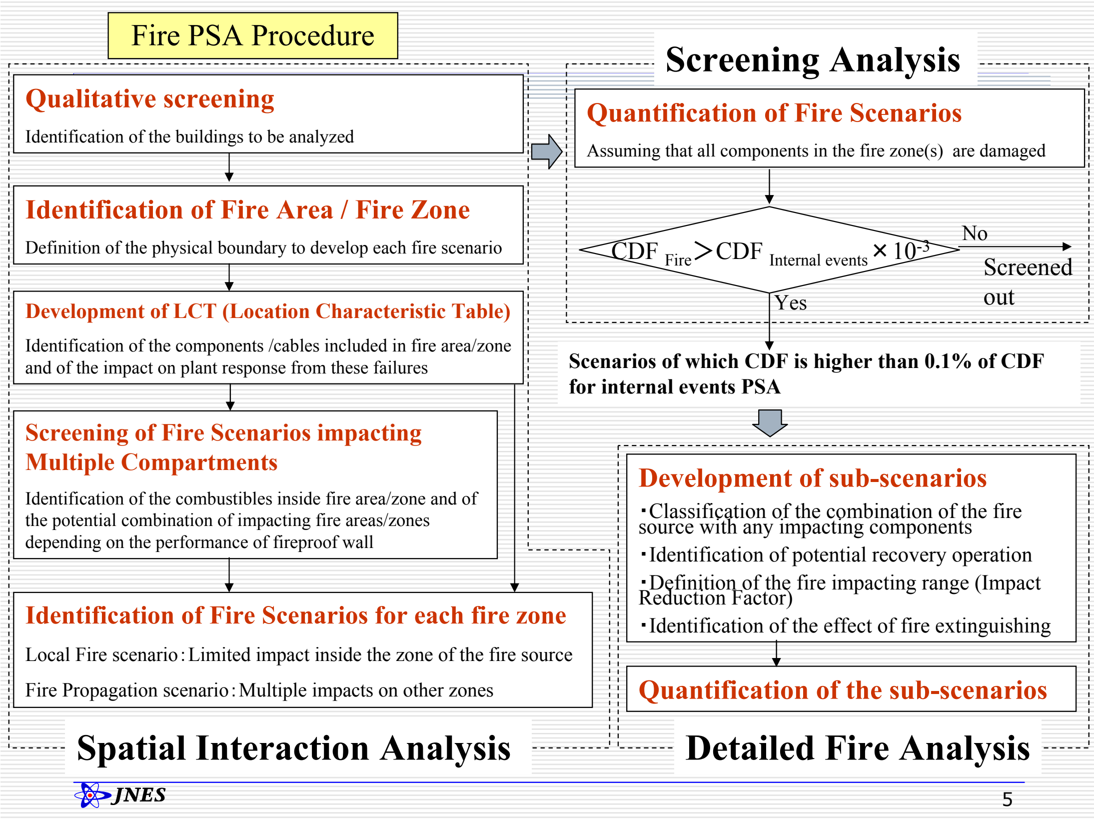
P_{prop} = Probability of fire propagation to other Fire Zones

(Applicable only to fire propagation scenarios)

$CCDP$ = Conditional Core Damage Probability for an initiating event in case of fire scenario "i"

- Fire Severity is based on phenomenological fire propagation analyses. Fire Severity Factors include the effect of fire suppression as well as the effect of the physical separation among the structure, system and components (SSCs).

JNES 4



2. Fire PSA methodology

(2) Key Analysis Technique

- Fire-induced initiating events were identified and quantified, applying the event tree and fault tree technique and considering SSCs installed in each Fire Zone.
- Bayesian update technique to fire events of operational experiences of NPPs in U.S. and in Japan was applied to quantify the fire frequencies because NPPs in Japan had experienced a small amount of fire events.
- Fire frequencies developed were apportioned to each Fire Zone, based on the number of component.
- The following computer codes were applied to develop the Fire Severity Factors.
  - α-Flow: Field Model
  - COMPBRN-III: Zone Model
  - HAWKS (Thermal Conduction): FEM Model
- These computer codes are being replaced with FDS and CFAST codes.
- ET and FT models developed for internal events were applied to quantify fire-induced accident sequences.

JNES 6

3. Preliminary Fire PSA Results

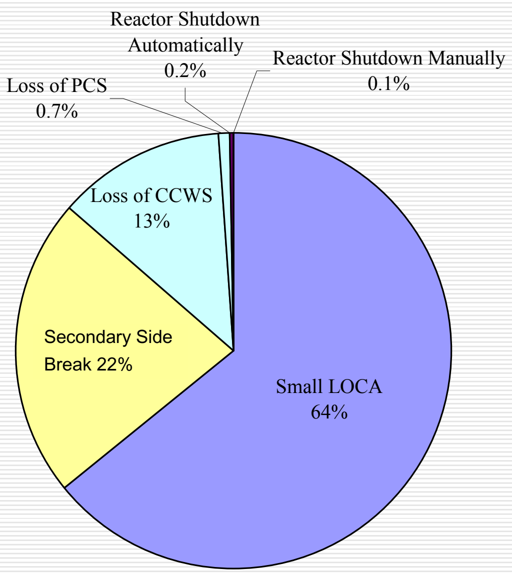
Dominant Accident Sequences in Power Operation

RED: Fire source, BLUE: Initiating Events

Reactor Types	Dominant Sequence	Contribution to total CDF
PCCV 4 Loop PWR	**Fire in emergency switchgear room (ESGR)** ＋Loss of AFW and spurious open of PORV due to **hot-short** in a control cable (**SLOCA**) ＋Loss of Feed & Bleed	**Approx. 23%** Contribution of CDF due to the scenarios relevant to cable / cabinet fires in switchgear room was more than about 30% of total CDF.
2 Loop PWR	**Fire of a control cable in CCW pump room** ＋Loss of CCWS, LPIS, CV-Spray, Instrumental Air system and Emergency Low Voltage Bus due to fire（**Loss of CCWS**） ＋Loss of MD-AFWP due to power cable failure caused by fire (Degradation of secondary side cooling) The power cables failed were just above the fire source.	**Approx. 49%** Contribution of CDF due to fires in CCW pump room was about 78% of total CDF.

JNES 7

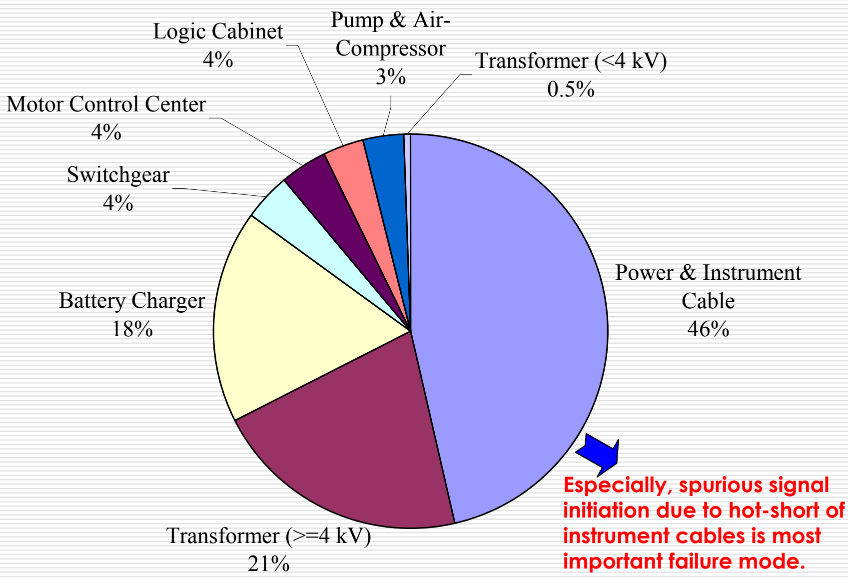
=Results of PCCV- 4 Loop PWR (1) =



- Initiating Event Contribution to CDF -

JNES 8

=Results of PCCV- 4 Loop PWR (2) =



- Fire Source Contribution to CDF -

JNES 9

4. Development of Fire Analysis Code System

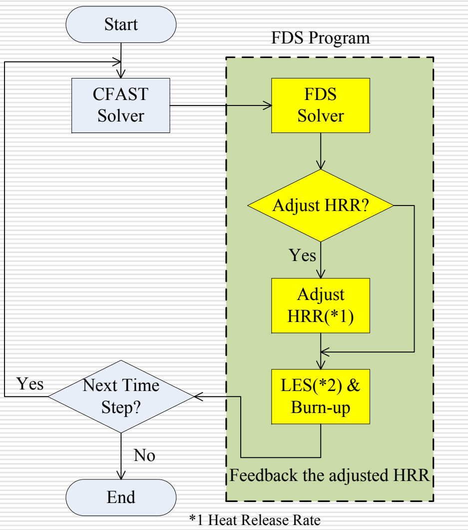
- Now the following computer codes are installed and enhanced in JNES.
  - FDS: CFD Model, Ver. 4 (Latest: Ver. 5.3.12)
  - CFAST: Zone Model, Ver. 5 (Latest: Ver. 6.11)
- Both codes have been integrated into one computer code, that is being reviewed and verified through the OECD-PRISME Project.

  - CFAST (Consolidated Fire and Smoke Transport Model) and FDS (Fire Dynamics Simulator) are being developed in the NIST (National Institute of Standards and Technology)
  - CFAST is a zone model that predicts the effect of a specified fire on temperatures, various gas concentrations and smoke layer heights in a multi-compartment structure.
  - FDS is a computational fluid dynamics (CFD) model of fire-driven fluid flow. The software solves numerically a form of the Navier-Stokes equations appropriate for low-speed, thermally-driven flow, with an emphasis on smoke and heat transport from fires.

JNES 10

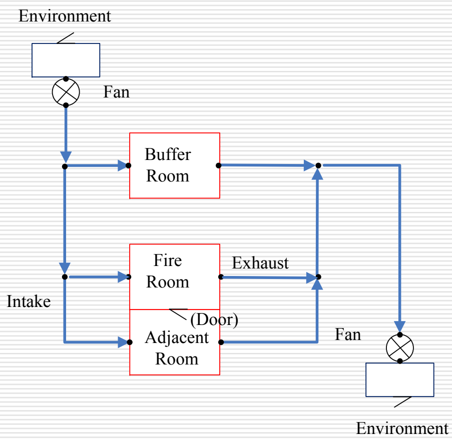
4. Development of Fire Analysis Code System

◆ CFAST/FDS Network



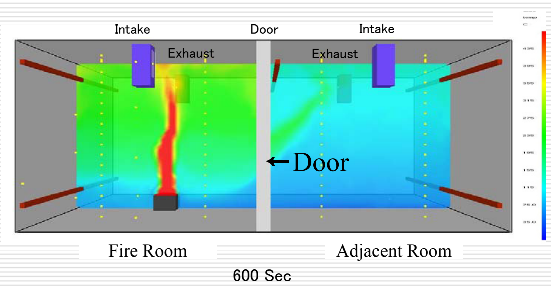
*1 Heat Release Rate
*2 Large Eddy Simulation for turbulent flow.

◆ Sample Analysis Model

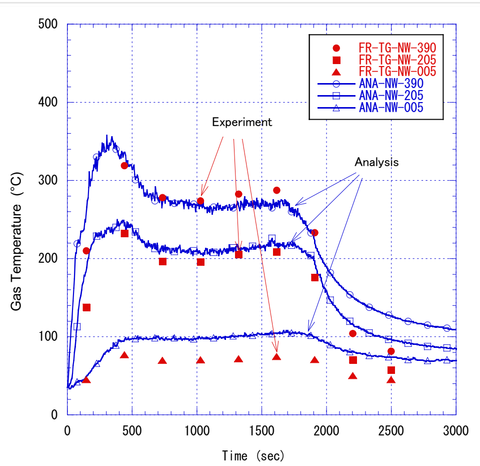


4. Development of Fire Analysis Code System

◆ Sample Analysis Results



- Gas Temperature Distribution at 600 (sec) -



- Time History of Gas Temperature -

5. Conclusion and Future Plan

- Fire PSA methodology has been developed and was applied to NPPs in Japan for power operation and LPSD states. CDFs of preliminary fire PSA for power operation were the higher than that of internal events.
- Fire propagation analysis code system (CFAST/FDS Network) was being developed and verified thru OECD-PRISME Project.

- Extension of the scope for LPSD state is planned to figure out the risk level.
- In order to figure out the fire risk level precisely, the enhancement of the methodology is planned.
  - Verification and validation of phenomenological fire propagation analysis code (CFAST/FDS Network) in the context of Fire PSA.
  - Enhancement of the methodology such as an application of "Electric Circuit Analysis" in NUREG/CR-6850 and related tests in order to quantify the hot-short effect precisely
- Development of seismic-induced fire PSA method being integration of existing seismic PSA and fire PSA methods is ongoing.
- Fire PSA will be applied to review the validity of fire prevention and mitigation measures.

JNES 13

Ⅲ-B-3

The 10th KJ PSA May 18–19, 2009

A Comparative Study of Two Quantification Algorithms and Importance Measures in a Fire PRA Model

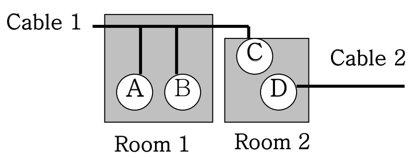
May 18, 2009

Kilyoo Kim, Sang H. Han, Dae Il Kang

Korea Atomic Energy Research Institute

Example Case

Internal Event CDF $=\{IE_1 \cdot A \cdot B \cdot C \cdot E,\ \ IE_2 \cdot A \cdot C \cdot D \cdot F\}$



Fire Occurrence Events	Basic Events	Initiating Events
R_1	A_1, B_1, C_1	IE_1
R_2	C_2, D_2	IE_1, IE_2

Event Descriptions

Table 1.

Fire Occurrence Events	Basic Events	Initiating Events
R_1	A_1, B_1, C_1	IE_1
R_2	C_2, D_2	IE_1, IE_2

Table 2.

Event Name	Event Description	Frequency	Probability
R_1	Fire Occurrence Event in Room 1	0.15/yr	
R_2	Fire Occurrence Event in Room 2	0.2/yr	
A_1	Component A Failure due to Room 1 Fire		1
B_1	Component B Failure due to Room 1 Fire		1
C_1	Component C Failure due to Room 1 Fire		1
C_2	Component C Failure due to Room 2 Fire		1
D_2	Component D Failure due to Room 2 Fire		1
A	Component A Failure due to Random Failure		0.001
B	Component B Failure due to Random Failure		0.005
C	Component C Failure due to Random Failure		0.007
D	Component D Failure due to Random Failure		0.0008
E	Component E Failure due to Random Failure		0.004
F	Component F Failure due to Random Failure		0.003

Conventional Quantification Method

$$\mathrm{CDF} = \sum_i (R_i \text{ x } CCDP_i)$$

where,

R_i = the i^{th} room fire occurrence event having a frequency unit,

$CCDP_i$ = conditional core damage probability given R_i

$\mathrm{CDF} = IE_1 \cdot A \cdot B \cdot C \cdot E + IE_2 \cdot A \cdot C \cdot D \cdot F$

→ $IE_1 \cdot (A+A_f) \cdot (B+B_f) \cdot (C+C_f) \cdot E + IE_2 \cdot (A+A_f) \cdot (C+C_f) \cdot (D+D_f) \cdot F$

→ $CCDP_1 = \Omega \cdot (A+\Omega) \cdot (B+\Omega) \cdot (C+\Omega) \cdot E + \Phi = E$

→ $CCDP_2 = \Omega \cdot (A+\Phi) \cdot (B+\Phi) \cdot (C+\Omega) \cdot E + \Omega \cdot (A+\Phi) \cdot (C+\Omega) \cdot (D+\Omega) \cdot F$

$= A \cdot B \cdot E + A \cdot F$

→ $\mathrm{CDF} = R_1 \cdot E + R_2 \cdot A \cdot B \cdot E + R_2 \cdot A \cdot F$ ------------------ (1)

New Quantification Method 1

CDF = $IE_1 A \cdot B \cdot C \cdot E + IE_2 \cdot A \cdot C \cdot D \cdot F$

$\rightarrow (R_1+R_2) \cdot A \cdot B \cdot C \cdot E + R_2 \cdot A \cdot C \cdot D \cdot F$

$\rightarrow R_1 \cdot A \cdot B \cdot C \cdot E + R_2 \cdot A \cdot B \cdot C \cdot E + R_2 \cdot A \cdot C \cdot D \cdot F$

$\rightarrow R_1 \cdot A_1 \cdot B_1 \cdot C_1 \cdot E + R_2 \cdot A \cdot B \cdot C_2 \cdot E + R_2 \cdot A \cdot C_2 \cdot D_2 \cdot F$ ---------(2)

cf) CDF = $R_1 \cdot E + R_2 \cdot A \cdot B \cdot E + R_2 \cdot A \cdot F$ --------- (1)

Different FVs between Eq.(1) and (2)

Eq.(2)→ $FV_{A\_total} = (R_1 \cdot A_1 \cdot B_1 \cdot C_1 \cdot E + R_2 \cdot A \cdot B \cdot C_2 \cdot E + R_2 \cdot A \cdot C_2 \cdot D_2 \cdot F)/CDF = 1$

Eq.(1)→ $FV_{A\_total} = (R_2 \cdot A \cdot B \cdot E + R_2 \cdot A \cdot F)/CDF = 0.001$

Eq.(2)→ $FV_{D\_total} = (R_2 \cdot A \cdot C_2 \cdot D_2 \cdot F)/CDF = 0.001$

Eq.(1)→ $FV_{D\_total} = (0)/CDF = 0.0$

Event Descriptions– Partial Failure

Fire Occurrence Events	Basic Events	Initiating Events
R_1	A_1, B_1, C_1	IE_1
R_2	C_2, D_2	IE_1, IE_2

Table 3.

Event Name	Event Description	Frequency	Probability
R_1	Fire Occurrence Event in Room 1	0.15/yr	
R_2	Fire Occurrence Event in Room 2	0.2/yr	
A_1	Component A Failure due to Room 1 Fire		1
B_1	Component B Failure due to Room 1 Fire		0.1
C_1	Component C Failure due to Room 1 Fire		1
C_2	Component C Failure due to Room 2 Fire		1
D_2	Component D Failure due to Room 2 Fire		1
A	Component A Failure due to Random Failure		0.001
B	Component B Failure due to Random Failure		0.005
C	Component C Failure due to Random Failure		0.007
D	Component D Failure due to Random Failure		0.0008
E	Component E Failure due to Random Failure		0.004
F	Component F Failure due to Random Failure		0.003

New Quantification Method 2

$CDF = IE_1 A \cdot B \cdot C \cdot E + IE_2 \cdot A \cdot C \cdot D \cdot F$

$\rightarrow (R_1+R_2) \cdot A \cdot B \cdot C \cdot E + R_2 \cdot A \cdot C \cdot D \cdot F$

$\rightarrow (R_1+R_2) \cdot (A+A_1) \cdot (B+B_1) \cdot (C+C_1+C_2) \cdot E$
$\quad + R_2 \cdot (A+A_1) \cdot (C+C_1+C_2) \cdot (D+D_2) \cdot F$,

$= R_1 \cdot (A+A_1) \cdot (B+B_1) \cdot (C+C_1) \cdot E$
$\quad + R_2 \cdot (A+A_1) \cdot (B+B_1) \cdot (C+C_2) \cdot E$
$\quad + R_2 \cdot (A+A_1) \cdot (C+C_2) \cdot (D+D_2) \cdot F$ ---------------------------- (3)

$= R_1 \cdot (AB+A_1B+AB_1+A_1B_1) \cdot (C+C_1) \cdot E$
$\quad + R_2 \cdot (AB+A_1B+AB_1+A_1B_1) \cdot (C+C_2) \cdot E$
$\quad + R_2 \cdot (AC+A_1C+AC_2+A_1C_2) \cdot (D+D_2) \cdot F$
$= R_1 \cdot (ABC+A_1BC+AB_1C+A_1B_1C+ABC_1+A_1BC_1+AB_1C_1+A_1B_1C_1) \cdot E$
$\quad + R_2 \cdot (ABC+A_1BC+AB_1C+A_1B_1C+ABC_2+A_1BC_2+AB_1C_2+A_1B_1C_2) \cdot E$
$\quad + R_2 \cdot (ACD+A_1CD+AC_2D+A_1C_2D+ACD_2+A_1CD_2+AC_2D_2+A_1C_2D_2) \cdot F$

If A_1, B_1, C_1, C_2, D_2 are 1, from 1st term in Eq.(3), the followings are overestimated.
$R_1 \cdot (ABC+BC+AC+C+AB+B+A+1) \cdot E - R_1 \cdot E$
$= R_1 \cdot (ABC+BC+AC+C+AB+B+A) \cdot E$

.....

cf) $CDF = R_1 \cdot A_1 \cdot B_1 \cdot C_1 \cdot E + R_2 \cdot A \cdot B \cdot C_2 \cdot E + R_2 \cdot A \cdot C_2 \cdot D_2 \cdot F$ ---------------(2)

New Quantification Method 3

Let's define;
The failure prob. of K component due to room i fire = K_i

If $K_i \geq 0.5$, $K \rightarrow K_i$,
otherwise, $K \rightarrow K + K_i$,

ex) If $B_1 \geq 0.5$, $B \rightarrow B_1$,
Otherwise, $B \rightarrow B + B_1$

From Table 3,
$CDF = IE_1 A \cdot B \cdot C \cdot E + IE_2 \cdot A \cdot C \cdot D \cdot F$
$\rightarrow (R_1+R_2) \cdot A \cdot B \cdot C \cdot E + R_2 \cdot A \cdot C \cdot D \cdot F$
$\rightarrow R_1 \cdot A \cdot B \cdot C \cdot E + R_2 \cdot A \cdot B \cdot C \cdot E + R_2 \cdot A \cdot C \cdot D \cdot F$
$\rightarrow R_1 \cdot A_1 \cdot (B+B_1) \cdot C_1 \cdot E + R_2 \cdot A \cdot B \cdot C_2 \cdot E + R_2 \cdot A \cdot C_2 \cdot D_2 \cdot F$

Ex) Errors When Methods Applied to B_1 Only

Method 1: $R_1 \cdot A_1 \cdot B_1 \cdot C_1 \cdot E + R_2 \cdot A \cdot B \cdot C_2 \cdot E + R_2 \cdot A \cdot C_2 \cdot D_2 \cdot F$

Method 2: $R_1 \cdot A_1 \cdot (B+B_1) \cdot C_1 \cdot E + R_2 \cdot A \cdot B \cdot C_2 \cdot E + R_2 \cdot A \cdot C_2 \cdot D_2 \cdot F$

Method 3: If $B_1 \geq 0.5$, Method 1, elseif Method 2

Reference: $R_1 \cdot A_1 \cdot (B+B_1-B \cdot B_1) \cdot C_1 \cdot E + R_2 \cdot A \cdot B \cdot C_2 \cdot E + R_2 \cdot A \cdot C_2 \cdot D_2 \cdot F$

B_1 Value		Method 1	Method 2	Method 3	Reference
1	CDF	6.006E−4	6.036E−4	6.006E−4	6.006E−4
	Error	0	3E−6	0	
0.7	CDF	4.206E−4	4.236E−4	4.206E−4	4.215E−4
	Error	9.0E−7	2.1E−6	9.0E−7	
0.5	CDF	3.006E−4	3.036E−4	3.036E−4	3.021E−4
	Error	1.5E−6	1.5E−6	1.5E−6	
0.1	CDF	6.060E−4	6.360E−5	6.360E−5	6.330E−5
	Error	2.7E−6	3.0E−7	3.0E−7	

Conclusions

- New method 1 is simple but sometimes inaccurate (i.e., when $K_i < 0.5$)
- New method 2 is complicated and sometimes inaccurate. (i.e., when $K_i \geq 0.5$)
- New method 3 is best.
  If $K_i \geq 0.5$, $K \rightarrow K_i$,
  If $K_i < 0.5$, $K \rightarrow K + K_i$
- New method 3 is good in FV calculation, and quantification.

Ⅲ-B-4

10th KJPSA

Improved Fire-PSA with Quantitative Fire Risk Assessment

May 19, 2009.

Korea Electric Power Research Institute

Moon-Hak Jee , PE (fire protection)

KEPCO KEPRI

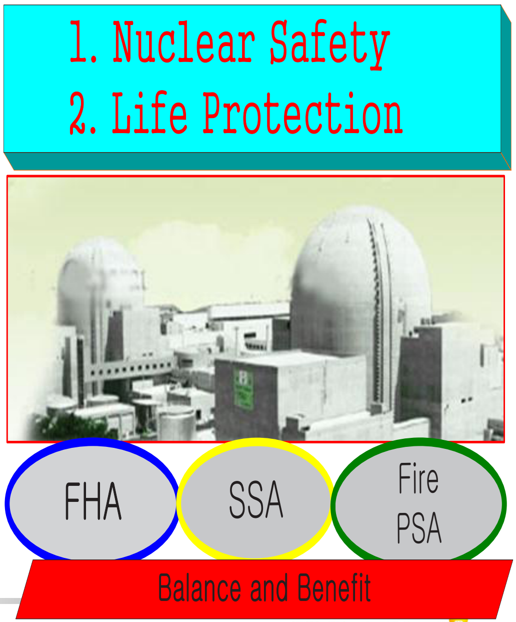
Topics

- Fire-induced CDF and IE CDF
- Fire PSA Improvement
- Fire PSA Factors
- Conclusion

KEPCO -2- KEPRI

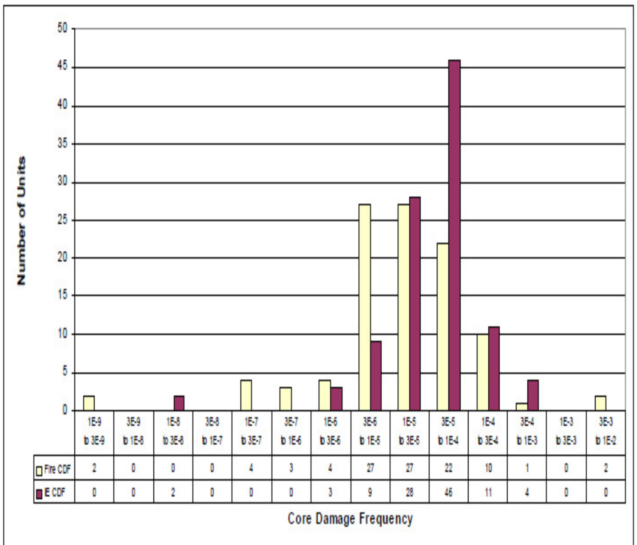
Fire risk management at NPPs

- **Fire Hazard Analysis**
  - Prevention
  - Detection and suppression
  - Mitigation at minimal
- **Safe Shutdown Analysis**
  - Isolation
  - Separation
  - Special Measurements (RSP, SCA)
- **Fire PSA**
  - fire frequency, NSP, HEP
  - CCDP
  - $CDF = \sum (FFxNSPxOtherxCCDP)_i$



KEPCO 3 KEPRI

Fire and IE CDFs Histogram



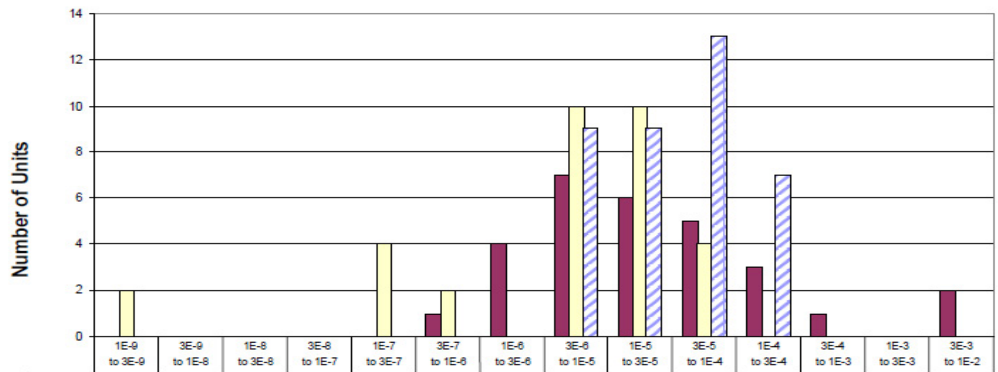
	Fire CDF	IE CDF
Mean	1.46E-04 (0.68)	6.74E-05 (0.32)
Median	1.73E-05	4.63E-05
Minimum	1.30E-09	1.30E-08
Maximum	5.40E-03	4.62E-04
5th percentile	2.44E-07	3.72E-06
95th percentile	2.21E-04	2.37E-04
Count	102	103
Mean (remove outliers)	4.43E-05 (0.39)	7.01E-05 (0.61)

Reference : EPRI TR-112933 (June 1999) : page 4-6 (Figure 4-1)

KEPCO 4 KEPRI

Fire CDF Histogram by Method

- FIVE : Fire Induced Vulnerability Evaluation screening methodology
- Various FPRA : EPRI TR-105928/NUREG/CR-4840/NUREG-2300
- Combination of FIVE screening, fire IPEEE and so on



	1E-9 to 3E-9	3E-9 to 1E-8	1E-8 to 3E-8	3E-8 to 1E-7	1E-7 to 3E-7	3E-7 to 1E-6	1E-6 to 3E-6	3E-6 to 1E-5	1E-5 to 3E-5	3E-5 to 1E-4	1E-4 to 3E-4	3E-4 to 1E-3	1E-3 to 3E-3	3E-3 to 1E-2
5+FPRA	0	0	0	0	0	1	4	7	6	5	3	1	0	2
fire PRA	2	0	0	0	4	2	0	10	10	4	0	0	0	0
FIVE	0	0	0	0	0	0	0	9	9	13	7	0	0	0

Core Damage Frequency

Reference : EPRI TR-112933 (June 1999) : page 4-8 (Figure 4-3)

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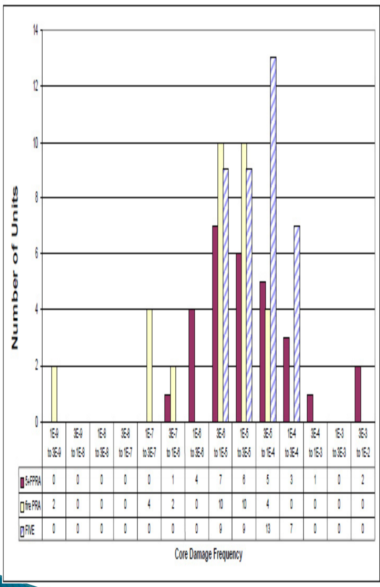
Fire CDF Histogram by Method



Reference : EPRI TR-112933 (June 1999) : page 4-8 (Figure 4-3)

[Raw data (before removing outliers)]

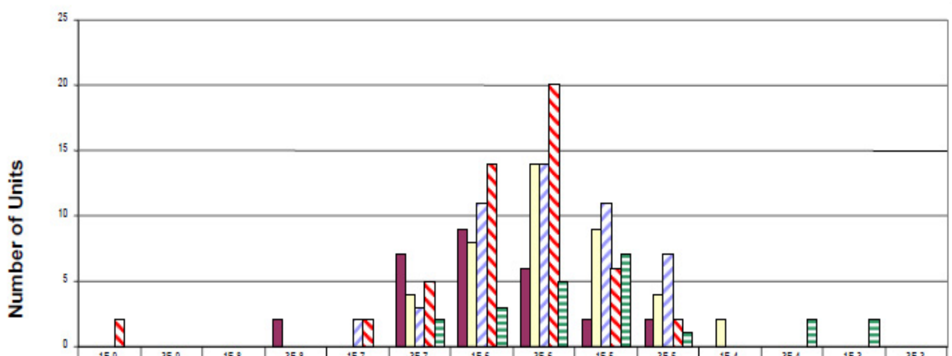
	5+FPRA	fire PRA	FIVE
Mean	**4.12E-04 (0.85)**	**1.22E-05 (0.03)**	**5.72E-05 (0.12)**
Median	1.78E-05	5.80E-06	3.56E-05
Minimum	1.00E-06	1.30E-09	3.91E-06
Maximum	5.40E-03	5.20E-05	2.00E-04
5th percentile	2.45E-06	8.12E-08	4.06E-06
95th percentile	3.04E-03	3.21E-05	2.00E-04
Count	30	31	38

[statistical data (removing outliers)]

	5+FPRA	fire PRA	FIVE
Mean	**6.25E-05 (0.47)**	**1.31E-05 (0.10)**	**5.72E-05 (0.43)**
Median	1.73E-05	7.50E-06	3.56E-05
Minimum	1.00E-06	1.61E-07	3.91E-06
Maximum	4.04E-04	5.20E-05	2.00E-04
5th percentile	2.44E-06	2.44E-06	2.44E-06
95th percentile	2.70E-04	2.70E-04	2.70E-04
Count	28	29	38

6

Significant Locations Fire CDF Histogram



	1E-9 to 3E-9	3E-9 to 1E-8	1E-8 to 3E-8	3E-8 to 1E-7	1E-7 to 3E-7	3E-7 to 1E-6	1E-6 to 3E-6	3E-6 to 1E-5	1E-5 to 3E-5	3E-5 to 1E-4	1E-4 to 3E-4	3E-4 to 1E-3	1E-3 to 3E-3	3E-3 to 1E-2
CSR	0	0	0	2	0	7	9	6	2	2	0	0	0	0
CR	0	0	0	0	0	4	8	14	9	4	2	0	0	0
Other	0	0	0	0	2	3	11	14	11	7	0	0	0	0
SWGR	2	0	0	0	2	5	14	20	6	2	0	0	0	0
TB	0	0	0	0	0	2	3	5	7	1	0	2	2	0

Core Damage Frequency

Reference : EPRI TR-112933 (June 1999) : page 4-11 (Figure 4-6)

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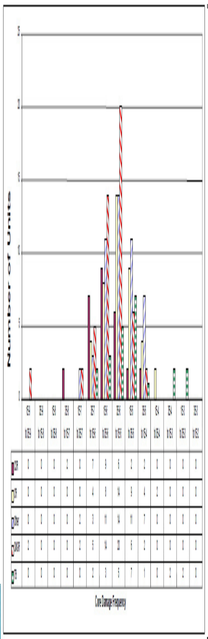
Significant Locations Fire CDF Histogram



[Raw data (before removing outliers)]

	CSR	CR	Other	SWGR	TB
Mean	**7.26E-06** (0.02)	**2.13E-05** (0.06)	**1.49E-05** (0.04)	**6.31E-06** (0.02)	**3.19E-04** (0.86)
5th percentile	1.94E-07	7.01E-07	6.20E-07	1.55E-07	7.70E-07
95th percentile	4.02E-05	7.41E-05	5.88E-05	2.45E-05	2.69E-03
Count	28	41	48	51	22

[statistical data (removing outliers)]

	CSR	CR	Other	SWGR	TB
Mean	**7.26E-06** (0.11)	**2.13E-05** (0.34)	**1.49E-05** (0.24)	**6.57E-06** (0.10)	**1.33E-05** (0.21)
5th percentile	1.94E-07	7.01E-07	6.20E-07	3.34E-07	7.43E-07
95th percentile	4.02E-05	7.41E-05	5.88E-05	2.53E-05	3.19E-05
Count	28	41	48	49	18

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Indications from Fire-induced CDF review

- **Fire-induced CDF changes depending on analysis methods**
  - FIVE, fire-PRA, IPEEE application methods
  - then, what if the newly advanced technology is approved?
  - then, what if the state-of-the-art quantitative fire risk tool is applied?
- **A few compartments contribute most of fire-induced CDFs**
  - Is the way to classify fire ignition sources still valid for NPPs?
  - Can fire frequency data be effective to NPPs up to date?
- **Fire CDFs are diverse from plants, reactor types, and time**
  - Conventional risk analysis ways are too conservative and uncertain
  - Newly developed fire modeling tools with V&V are applicable
  - FIVE, COMPBRN-II/III should be replaced with new quantitative tools

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Fire PRA methodology in Korea

- **Identical fire PRA model based on FIVE and EPRI TR-105928**
  - FIVE methodology is used for fire compartment screening purpose
  - It combines deterministic and probabilistic approaches
  - Technique at FIVE is highly conservative to include uncertainty
  - Fire modeling analysis is quite bounding rather than smart engineering
  - All Korean NPPs are still using FIVE and EPRI fire PRA methodology
- **Major fire compartments contribute fire-induced CDFs**

Kori-2	CR > Inverter room > SWGR > TB > CSR
Kori-3,4	CR > SWGR > TB > Others

  - Major contribution compartments : different values at EPRI TR-112933
  - Fire-induced CDF : order of E-05
  - IE CDF : order of E-06

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Fire PSA improvement with fire risk assessment

- NFPA-803 : fire protection standard for LWR (up to 2004)
- NFPA-805 : PB fire protection standard for LWR (now)
- Transition from NFPA-803 to NFPA-805 in US
  - US NRC revised 10CFR50.48 in June 2004
    - (c) National Fire Protection Association Standard NFPA 805 - approved
    - (c) (3) compliance with NFPA 805
- NUREG/CR-6850 : Fire PRA methodology
  - RI-PB approach for fire risk management
  - introduction of fire modeling (zone and CFD model)
  - HEP and HRA (Performance affecting/shaping factor)
  - Uncertainty analysis and Sensitivity analysis
  - 3-dimentional risk distribution and configuration

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Fire PSA improvement with fire risk assessment

- FPRA : **FIVE, COMPBRN-Ⅱ/Ⅲ, fire PRA, IPEEE, etc**
- Recent Tools : Zone Model, Field Model
  - FPEtool : Zone Model / Computer Program by NIST
  - **CFAST : Zone Model / Computer Program by NIST**
  - **FDS : CFD Model of Fire Driven Fluid Flow**
    - Smoke and Heat transport from Fire
    - Display FDS result by SMOKEVIEW
  - Others : Magic, Flame, LES, Jasmine, **Engineering tools**, etc

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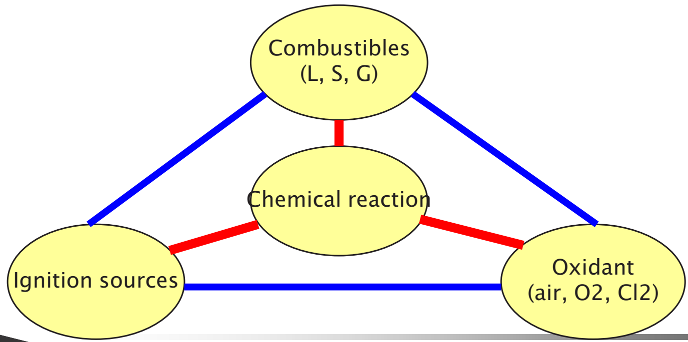
Fire PSA improvement with fire risk assessment

- $CDF = \sum (\lambda_i * SF_i * NSP_i * CCDP_i)$
  - Λ_i : fire frequency
  - SF_i : severity factor
  - NSP_i : non-suppression probability $_i$
  - $CCDP_i$: conditional core damage probability



KEPCO -13- KEPRI

Fire PSA factors : fire ignition frequency

- Fire ignition frequency for fire PSA in Korea
  - EPRI fire events data for conventional and generic database (NSAC 178L)
  - US NPPs between 1965 and 1988 (updated data from 1965 to 2000)
- New fire PRA methodology (NUREG/CR-6850)
  - Classification fire frequency on US NPPs : 37 bins with split fractions

Table 6-1
Fire Frequency Bins and Generic Frequencies

ID	Location	Ignition Source (Equipment Type)	Mode	Generic Freq (per rx yr)	Split Fractions for Fire Type: Electrical	Oil	Transient	Hotwork	Hydrogen	HEAF[1]
1	Battery Room	Batteries	All	7.5E-04	1.0	0	0	0	0	0
2	Containment (PWR)	Reactor Coolant Pump	Power	6.1E-03	0.14	0.86	0	0	0	0
3	Containment (PWR)	Transients and Hotwork	Power	2.0E-03	0	0	0.44	0.56	0	0

- Expansion of fire data to cover similar fire events (OECD and industry data)
- Incorporation of fire modeling results for fire initiation and propagation
- Fire control activities for advanced fire protection program (in Korea)

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Fire PSA factors : SF and NSP

- **Conventional fire PSA**
  - Complete failure of safety function at compartment
  - Deterministic fire suppression probability
  - No benefit of incipient fire detection and response by procedures

Automatic suppression system reliability (FIVE)	
Wet pipe sprinkler	2.0E-02
Preaction sprinkler	5.0E-02
Deluge sprinkler	5.0E-02
CO2	4.0E-02
Halon	5.0E-02

- **Advanced severity factor and non-suppression probability**
  - Severity factor : based on gamma distribution function with discretization
  - NSP : estimated fire event tree for fire detection and suppression capability

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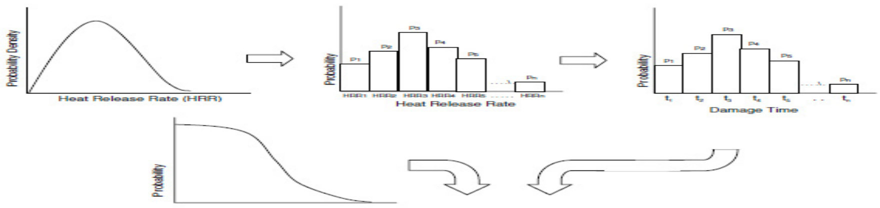
Fire PSA factors : SF and NSP

- **Equation** : $SF_k \cdot P_{ns,k} = \int_{All\,t} p_{damage}(t) P_{ns}(t)\,dt$
- **Severity factor and Non-suppression probability calculation**



HRR Values	HRR1	HRR2	HRR3	HRR4	....	HRRn
Individual Severity Factor	$P_{k,1}$	$P_{k,2}$	$P_{k,3}$	$P_{k,4}$	....	$P_{k,n}$
Time to damage	$t_{k,1}$	$t_{k,2}$	$t_{k,3}$	$t_{k,4}$	....	$t_{k,n}$
Prob. of supp. after damage	$P_{NS,k,1}$	$P_{NS,k,2}$	$P_{NS,k,3}$	$P_{NS,k,4}$	....	$P_{NS,k,n}$
$[SF_k \cdot P_{ns,k}]_i$	$P_{k,1} \cdot P_{ns,k,1}$	$P_{k,2} \cdot P_{ns,k,2}$	$P_{k,3} \cdot P_{ns,k,3}$	$P_{k,4} \cdot P_{ns,k,4}$	....	$P_{k,n} \cdot P_{ns,k,n}$

- **Quantitative fire risk assessment by fire model (program)**
  - HRR based on fire size, MLR, combustion amount, ventilation effect
  - Fire detection and suppression capability and activation time
  - Fire suppression capability and extinguishment and so many quantities

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Fire PSA factors : CCDP

- **CCDP : based on IE PSA model consideration of fire events**
- **Advanced fire risk assessment : more rooms will be screened**
  - For PWR plants in Korea, 80% of rooms screened-out
  - A few fire compartment occupied most of fire-CDF
  - Conventional and generic approach is satisfactory the present goal
- **More challenge for the improvement fire-induced CDF**
  - Recent scientific fire modeling tools can improve fire CDF more
  - Newly modified fire protection program in Korean NPPs
  - Credited fire protection system and managerial fire control procedure

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Ventilation control for confined compartment

- **Designed ventilation conditions for fire PSA**
  - Natural or forced ventilation flow rate and condition : unchanged
  - FIVE and COMPBRN : as aggravated situation for bounding conditions
- **Predominant ventilation effect to fire growth and propagation**
  - At Initial and without control , combustible control fire is governing
  - If controlled under-ventilated condition is easily created
  - With fire suppression aid, fire growth and propagation is stopped
- **More research and development for ventilation control**
  - Fire fighting strategies for manual fire suppression
  - Pressure variation due to ventilation-controlled fire
  - Oxygen depletion, back-draft, flash-over, radiation concentration

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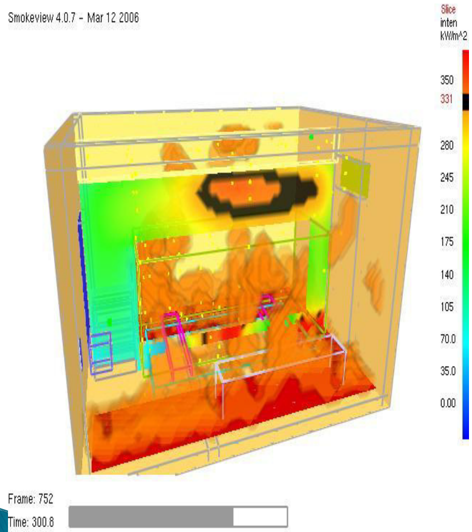
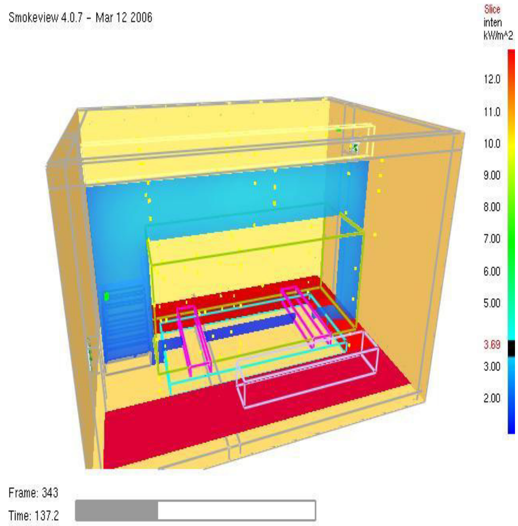
Ventilation control for confined compartment

Well-ventilated condition

Under-ventilated condition





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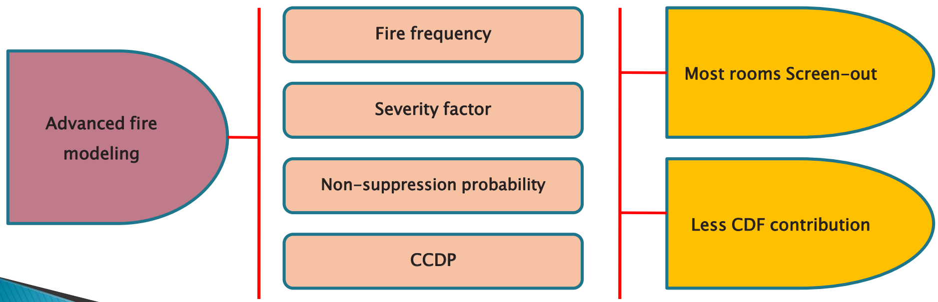
Conclusion

- More fire compartments can be screened out
  - With performance-based fire modeling
  - Recent fire PSA methodology
- Fire-induced and/or total CDF can be declined
- Advance fire fighting strategies for each fire compartment
- Active fire suppression tactics with ventilation-control fire



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Session III-C

Severe Accident & Safety Analysis (I)

Session III-C Severe Accident & Safety Analysis (I)

Chair: Tao LIU (INET), Han-Chul KIM (KINS)

Four papers, contributed by KAERI, KINS and JNES, are collected into Session III-C. They deal with hydrogen issue, severe accident code validation, iodine chemistry, and safety depressurization. Main issues are severe accident code validation and adequate application.

The first one of these papers is “Evaluation of THAI-HM2 Test with MELCOR Code” presented by Dr. Jung-Jae Lee. THAI-HM2 is one of the OECD/NEA-THAI international programme. A brief introduction of THAI-HM2 test is given first including the background, test facility, test procedure, thermal hydraulic phenomena and instrumentation, and test results. MELCOR analysis follows two steps. Step1: A basic case study for several conditions in the analysis; Step2: Sensitivity study of nodalization in axial direction. The applicability of MELCOR code for a dynamic thermal hydraulic process where a stratified light gas cloud is broken-up by steam plume is verified in the paper. Some pieces of limitation and suggestion on MELCOR code application are also obtained from the work.

The second paper is about “MELCOR improvement and applications” given by Mitsuhiro Kajimoto from JNES. Severe accident research activities in Japan are summarized first, and then model development based on existing experimental data and analytical approach is described in detail. Subsequently, ongoing international cooperative research projects are also introduced. All these work show that MELCOR code improvement is necessary and is advancing towards a higher quality and reliability.

The third one is “Validation of MELCOR Iodine chemistry model with BIP Test Data”. An intercomparison of MELCOR calculation and RTP experiment data is done to validate the MELCOR iodine pool chemistry model. The paper gives a detail description of pool chemistry model of MELCOR and RTF tests analysis and then presents the results of comparison. The conclusions from the work are: the MELCOR code can properly simulate the molecular iodine gas formation against sump pH in accident condition, while the calculation result shows model development needs because there is a tendency to underestimate I2(gas) concentration.

The last one in this secession is “Analysis of RCS feed & bleed operation to mitigate a severe accident for OPR1000” presented by Rae-Joon Park. A feed and bleed operation of the reactor coolant system (RCS) to prevent reactor vessel failure has been analyzed in an optimized power reactor (OPR) 1000. SCDAP/RELAP5 code is used to evaluate the Feed and Bleed operation for the Total LOFW in the severe accident of the OPR1000. Effect of operator action timing on the consequences and operator action capacity on the consequences are key elements for the accident mitigation. Several cases concerning a total LOFW with and without RCS feed and bleed are introduced in the presentation. Suggestions of operator action timing and action capacity are obtained from the work.

As a summary, every paper in Secession III-C is related to severe accident calculation codes, three papers of which are refers to MELCOR and the remainder is related to SCDAP/RELAP5. These codes are important tools for severe accident analysis which has become an indispensable part in NPP safety issue. Therefore, further development in this area was called for by the presenters.

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Ⅲ-C-1

The 10th Korea-Japan Joint Workshop on PSA

Evaluation of MELCOR Code for THAI-HM2 Test

May 19, 2009
Jeju, Korea

Lee, Jung-Jae
Risk Assessment Department, KINS

Korea Institute of Nuclear Safety

Contents

Slide 1

I. Introduction

❑ Needs

- Reduce the uncertainties in the analysis of hydrogen distribution in containment for regulatory review of nuclear power plants
- Establish a reliable analysis methodology for hydrogen risk assessment

❑ Objectives

- To verify MELCOR code against OECD/THAI-HM2 experiment
- To establish a better analysis methodology using MELCOR 1.8.5
  - There is no large change in MELCOR 1.8.6 related to this study.

Slide 2

I. Introduction

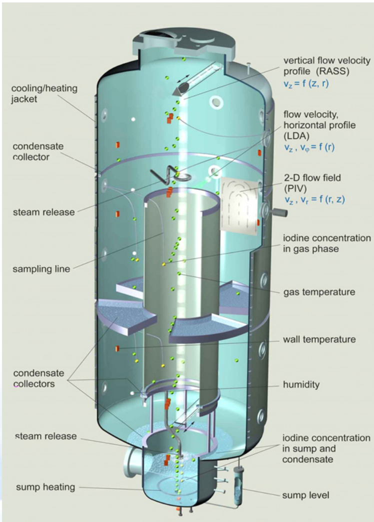
❑ OECD/NEA THAI Program

- THAI: Thermal hydraulics, Hydrogen, Aerosols, Iodine
- Sponsored by Federal German Ministry of Economy and Labor ('07-'09)
- Experimental programs consist of:
  - HM tests: Material scaling of H2-He and mixing
    (Blind & Open calculations executed for V&V of various CFD and LP codes)
  - HD tests: H2 deflagration
  - HR tests: H2-PAR
  - IW tests: Wash down of I2 deposition from walls
  - IO tests: Impact of I2-O3 reaction with aerosols
  - AW tests: Wash down of aerosol deposits from structures
- KINS has been joining the THAI international program in cooperation with KAERI.

Slide 3

II. THAI-HM2 Test

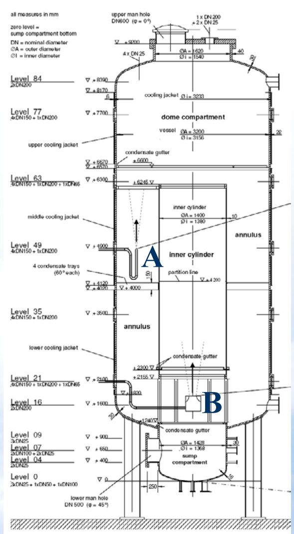
- THAI Facility for HM2 test
  - Located at Eschborn, Germany
  - Downscaled containment test facility made of stainless-steel
  - H=9.2m, D=3.2m → V=60m^3
  - Inner cylinder in central region (dividing the volume into the inner region and the annulus) and condensate trays in the middle of annulus
  - Several small condensate guts
  - Over 350 channels for measurement including P, T, v, C, and etc.



Slide 4

II. THAI-HM2 Test

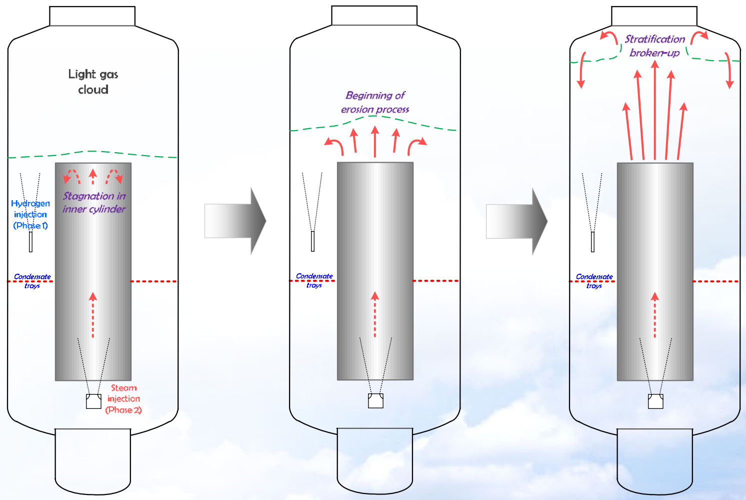
- Test Procedure
  - Phase 1: 0 to 4,200 sec
    - Hydrogen injection at point 'A'
    - Stratification of atmosphere in vessel
  - Phase 1a: 4,200 to 4,320 sec
    - No injection period of 2 minutes
  - Phase 2: 4,320 to 6,820 sec
    - Steam injection at point 'B' and steam plume rising up through the inner cylinder
    - Erosion of stratification by steam plume and generation of natural circulation loop occurred



Slide 5

II. THAI-HM2 Test

❑ Thermal Hydraulic Phenomena in Phase 2

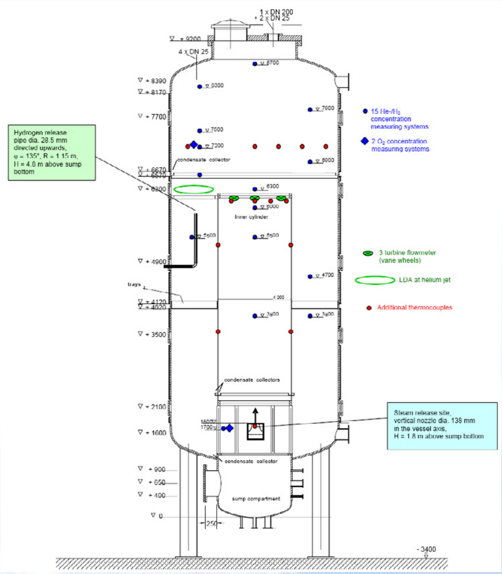


Slide 6

II. THAI-HM2 Test

❑ Instrumentation

- The measurements of 33 sensors were used in HM-2 test for the benchmark;
  - P
  - T_{atm}, T_{wall}
  - C_{H2}
  - V_{steam}
- LDA was equipped for 2-D velocity profile

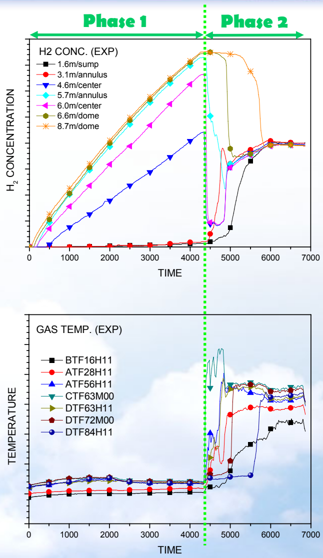


Slide 7

II. THAI-HM2 Test

Test Results

- Phase 1 (hydrogen injection)
  - P and C_{H2} increase linearly with atmospheric stratification.
  - The highest C_{H2} in the dome region and the lowest one (~0) in the bottom region were measured.
- Phase 2 (steam injection)
  - C_{H2} in the dome decreased by steam plume in orderly manner while that in bottom region increases.
  - Steam plume destroyed the gas stratification of the dome region in the vessel.
  - Finally the natural circulation was generated and the atmosphere in the vessel was fully mixed.

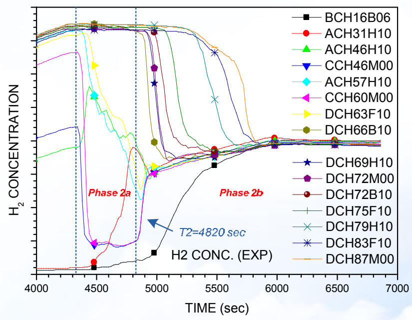


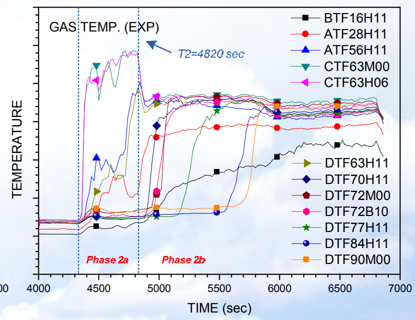
Slide 8

II. THAI-HM2 Test

Erosion Process in Phase 2

- Profiles of C_{H2} and T_{atm} show the break-up of stratification.
  - Orderly erosion process occurred from the lower region to the higher region
- Atmosphere in the vessel was fully mixed after the erosion process.





Slide 9

III. Description of MELCOR Analysis

Procedure

- This study have advanced in step with OECD THAI project, a benchmark of HM-2 test.
- Step 1: A basic case study for several conditions in the analysis
  - Effect of using avg. mass and energy source data and time-dependent raw data
  - Effect of input parameters in CVH, FL and HS packages
  - Effect of wall condensation modeling using film-tracking network
  - Effect of minor division of node in accordance with the test

  ☞ *The BLIND calculation*
- Step 2: Sensitivity study of nodalization in axial direction
  - Effect of the number of axial nodes

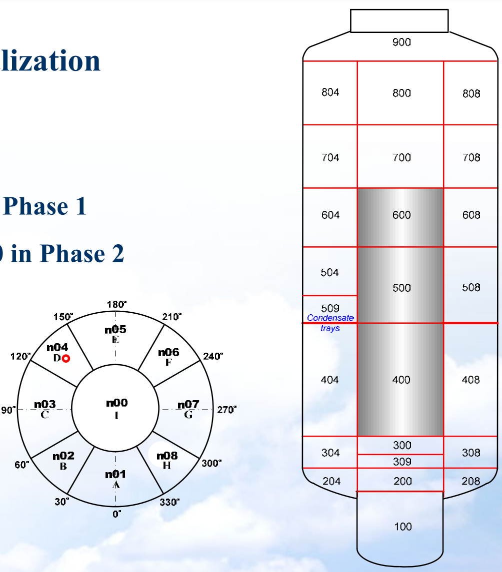
  ☞ *The OPEN calculation*

Slide 10

III. Description of MELCOR Analysis

Blind Calculation - Nodalization

- 9 axial levels
- 67 CVs
- H2 injection @ CV504 in Phase 1
- Steam injection @ CV300 in Phase 2



Slide 11

III. Description of MELCOR Analysis

Efforts Made in Open Calculation

- Effects of HSFT model, radial discretization of inner cylinder nodes and axial discretization have been examined.
- RUN37 for 22-axial nodes showed best agreement with the experiment.

Case	CVH Package (Control Volume)				FL Package (Flow Path)		HS Package (Heat Struct.)	Remark
	# of CVs (# of vertical level)	Gas inj. CV (H2/Steam)	Source data	In-cylinder division	Segment area ratio to real value	Loss Coeff.	HSFT modeling	
RUN20	67 (9-level*)	504 / 300	Time-dep.	No	0.5	5	Yes (partial)	Blind test case
RUN22	67 (9-level*)	504 / 300	Time-dep.	No	0.5	5	Yes (all)	-
RUN26	70 (9-level*)	504 / 300	Time-dep.	Yes** (level 4-6)	0.5	5	Yes (all)	-
RUN29	72 (9-level*)	504 / 300	Time-dep.	Yes** (level 3-7)	0.5	5	Yes (all)	-
RUN33	147 (19-level)	514 / 310	Time-dep.	No	0.5	5	No	-
RUN35	147 (19-level)	514 / 400	Time-dep.	No	0.5	5	No	-
RUN37	174 (22-level)	514 / 310	Time-dep.	No	0.5	5	No	Open test case

* Gas injection nodes were vertically sub-divided by each 2 nodes.

** Inner cylinder nodes were horizontally sub-divided by each 2 nodes.

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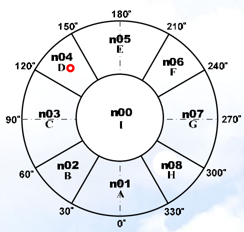
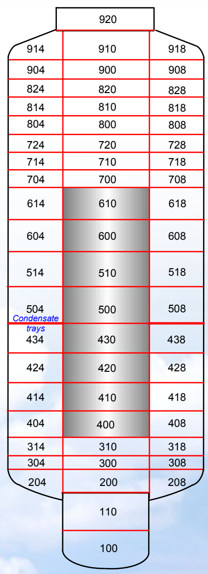
III. Description of MELCOR Analysis

Open Calculation - Nodalization

- 22 axial levels
- 174 CVs
- H2 injection @ CV514 in Phase 1
- Steam injection @ CV310 in Phase 2





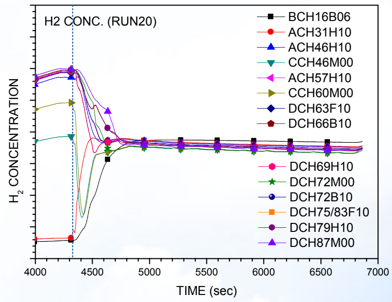
Slide 13

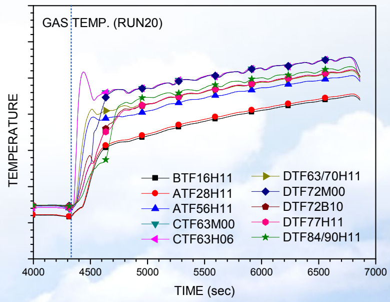
IV. Calculation Results

E = mc²

❑ Blind Calculation

- No large differences were made by variation of input parameters.
- Limitations have been found to simulate H2 stratification.
- The erosion process occurred within very short period after steam inj.





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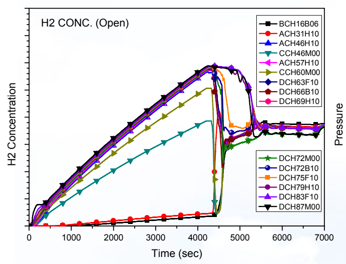
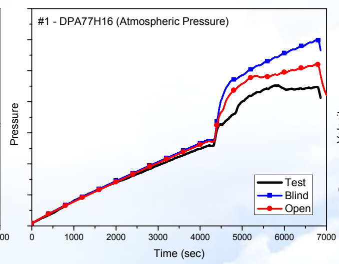
IV. Calculation Results

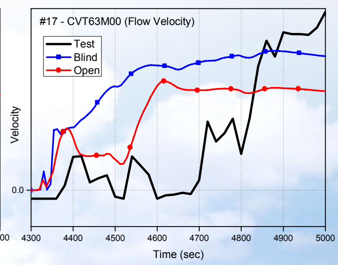
E = mc²

❑ Open Calculation (1)

- H2 concentration in Phase 1 was similar to that of Blind calculation.
- Pressure response and flow stagnation of steam plume at the upper exit of the inner cylinder showed better agreement with the experiment.







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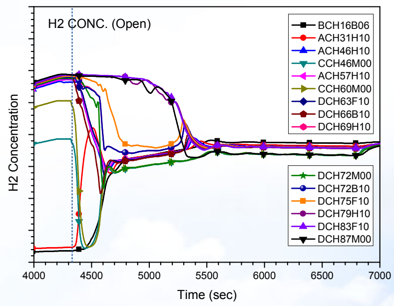
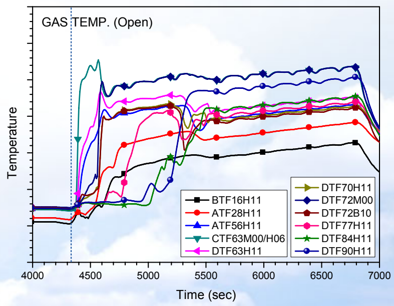
IV. Calculation Results

❑ Open Calculation (2)

- A similar erosion process for H2 cloud could be simulated.
- A discrepancy in the evolution of break-up for H2 cloud was found between the test and calculation.
  - Relatively quick process was found each in the lower dome region and in the upper dome region.





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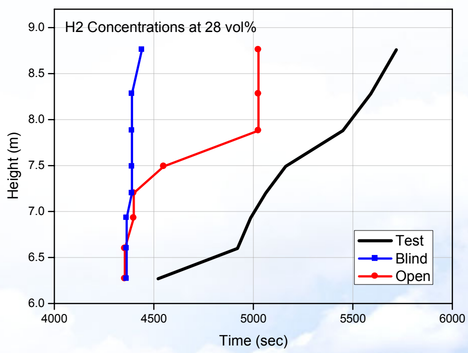
IV. Calculation Results

❑ Improvements in Open Calculation (1)

- Erosion process estimated with each H2 sensor by checking the moment when the C_{H2} was 28 vol.%.



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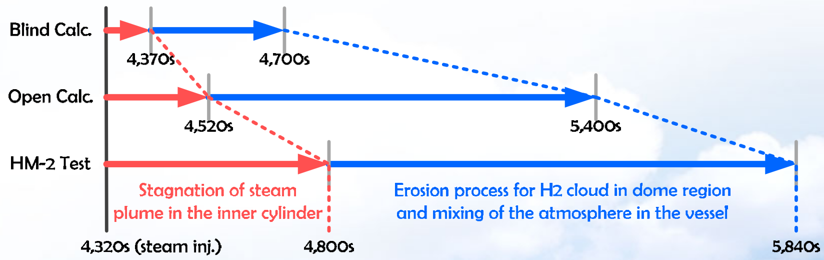
IV. Calculation Results

E = mc²

❑ Improvements in Open Calculation (2)

- Comparison of results among two MELCOR analyses and HM-2 test with the continual processes of the stagnation and erosion.



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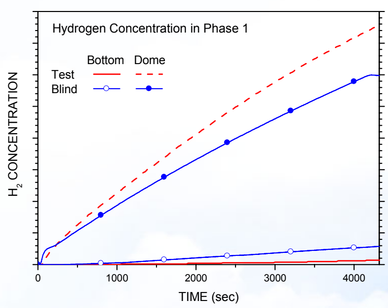
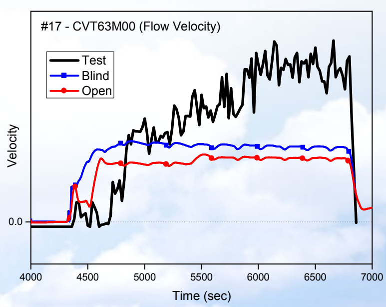
IV. Calculation Results

E = mc²

❑ Some Problems Unresolved in MELCOR Analysis

- Variation of C_{H2} in the vessel is still underestimated.
  - Lower concentration in the dome, higher one at the bottom
- Simulating the velocity and direction of the injection flow is restricted.





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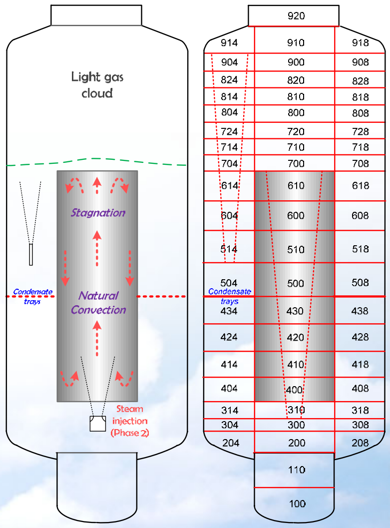
IV. Calculation Results

- Discussion with HM-2 Benchmark Participants (experimentalists and LP & CFD code users)
  - 'Plume zone' should be modeled to simulate the counter-current flow in the inner cylinder.
  - Consideration of the 'fog' in the vessel may give an improved prediction for the pressure and concentration in phase 2 of the experiment.



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V. Concluding Remarks

- The characteristics of MELCOR code has been examined for a dynamic thermal hydraulic process of THAI HM-2 test, as follow;
  - The axial nodalization dominantly affected the simulation of erosion process.
  - Prediction of the stratification was hardly improved even with lager number of nodes.
  - Appropriate modeling for the directional injection of the mass and energy should be used.
- Further study will be done in near future with modeling of plume zone and consideration of the fog phase.
- Insights obtained in this study can be considered in the plant applications using MELCOR code.

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Acknowledgements

"The authors are grateful for the financial support of the Participating Countries to the joint cooperative THAI Project run under the auspices of the Nuclear Energy Agency (NEA), Organization for Economic Cooperation and Development (OECD)."

※ Several interesting presentation for OECD THAI program will be made in NURETH-13 in Japan, this October.

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Ⅲ-C-2

MELCOR Improvement and Applications

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

1. Severe Accident Research Activities
2. Model Development based on Existing Experimental data
3. Model Development based on Analytical Approach
4. International Cooperative Experimental Research for Key phenomena and Modeling

Summary

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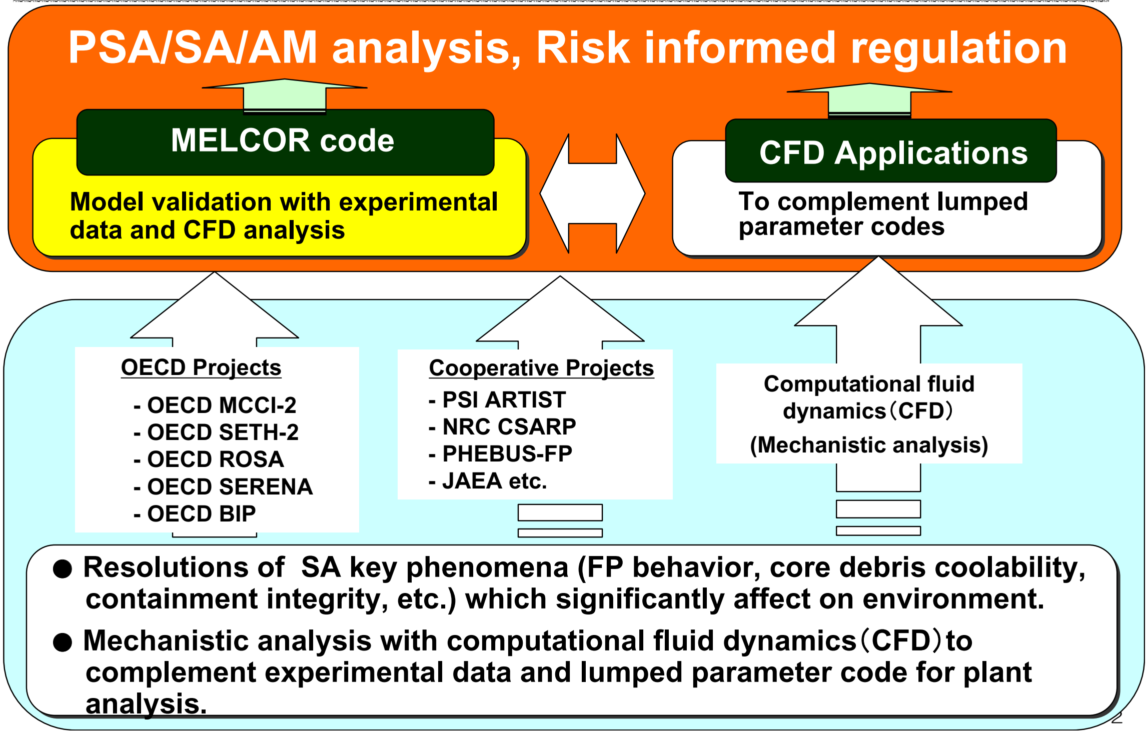
1. Severe Accident Research Activities



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Severe Accident Research Activities

- The application area of risk-informed regulation has been expanded to improve safety decision making and improve regulatory effectiveness in Japan.
- These applications needed for the severe accident codes with higher quality and reliability.
- JNES has been using and improving MELCOR code for the severe accident analysis for NPPs in Japan:
  - (1) Code validation with existing experimental data
  - (2) International cooperative experimental researches for key phenomena
  - (3) Applications of CFD for complementation of MELCOR lumped parameter code.

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Improvement of the MELCOR Code

Categories	Objectives	Resources	Codes
Utilization of Existing Experiments	Code validation with existing experimental data	Experiments at NUPEC（Containment structure behavior test / FP behavior test / Hydrogen mixing and combustion tests） PHEBUS-FP Experiment, etc.	MELCOR
Utilization of Ongoing Experiments	International cooperative experimental research for key phenomena and modeling	International Projects: PSI ARTIST Program, OECD MCCI Program, OECD MASCA Program, OECD SETH2 Program, OECD SERENA Program, OECD SFP Program, NRC CSARP Program, etc.	MELCOR & CFD
CFD Analysis	To complement lumped parameter codes	Computer fluid dynamics（CFD）	MELCOR & CFD

4

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Model Development based on Existing Experimental Results

Items	Contents	Comments
Pool Scrubbing	- Pool scrubbing models for gaseous FPs	Applied two film model
Spray Removal	- Spray removal models for gaseous FPs	Applied two film model
Containment Failure Model	- Failure models based on the experiments at NUPEC	Experiments at NUPEC
FP Leak from Containment	- FP leak models at penetration in containment	Experiments at NUPEC
FP Deposition	- Chemical absorption model at high temperature	Experiments at IRSN
FP Release from Fuel	- Improvements of CORSOR-M & BOOTH	Experiments at ORNL &PHEBUS
Control Rods	- AgI formation Models	PHEBUS-FP Experiments
Spray Droplet	- Droplet size distribution model under low flow rate	Experiments at NUPEC

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Model Development based on Analytical Approach

Items	Contents	Comments
Nodalization	- CV nodalization （change from 4 nodes to 9 nodes)	Multi-compartment CV model
	- Pressurizer relief tank node with pool scrubbing effect	Pool scrubbing effect
	- Crossover-leg (loop-seal model)	Loop seal effect
CFD Analysis	- FP deposition model of steam separator and dryer	Model development based on mechanistic analysis
	- CFD RCS model	
	- CFD containment model	

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2. Model Development based on Existing Experimental Results

Model modification with existing experimental data.

- FP release from fuel based on experiments at ORNL &PHEBUS
- FP release from fuel based on experiment at JAEA (VEGA test)

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FP release model from fuel rod

Original CORSOR-M model over predicted the FP release rate from fuel rod

Modified CORSOR model by comparison with ORNL test results

Modified model predicted agreeably the PHEBUS-FP (FPT-1) test results

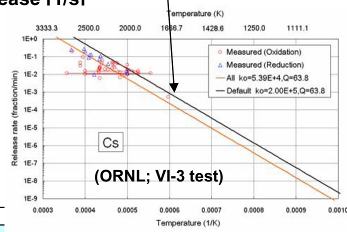
Further comparison with other PHEBUS-FP test results to fix the modified model .

CORSOR-M

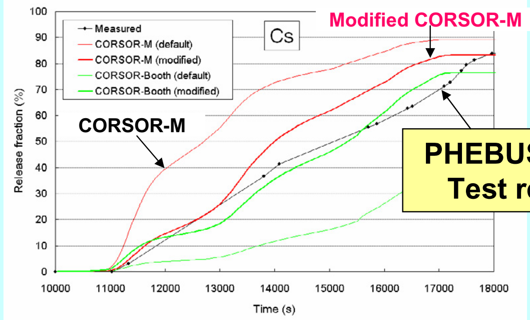
$f = k_0 \cdot \exp(-Q/RT)$

- f : fractional release rate [1/s]
- k_0 : coefficient of fractional release [1/s]
- Q : activation energy [J/mol]
- R : gas constant [J/mol K]
- T : temperature [K]

Over prediction



Cs release rate from fuel rod (FPT-1 Test)



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Pressure dependent FP release model from fuel rod

CORSOR model is independent on back pressure (RCS pressure)

Back pressure affects on the FP release from fuel rod in high pressure sequence

JAEA proposed a pressure dependent FP release model

Applied the new model to MELCOR code

- **Late FP release from fuel rods by the new model**
- **Effects depend on accident sequences**

FP release model depending on back pressure

Original model is independent on back pressure

Installed the pressure dependent new model to MELCOR

【JAEA model】

FP release from fuel rods depends on the back pressure

$$\dot{f}_{corsor} = k_0 \exp(-Q/RT)$$

$$\dot{f} = \frac{1}{\sqrt{P_R}} \times \dot{f}_{corsor} \quad \text{(New model)}$$

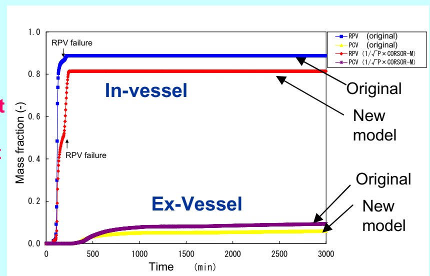
Largely affects on the FP release

(BWR MARK-I TQUX)

- Late FP release from fuel due to high back pressure

In-Vessel: Smaller amount of released FP
Ex-Vessel: Larger amount of released FP

■Smaller amount of CsI released to the environment
(one third of original casein this case)



CsI release rate from fuel rod

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3. Model Development based on Analytical Approach

Improvement of the nodalization

- Crossover-leg (loop-seal model)
- Pressurizer relief tank node with pool scrubbing effect

Model development based on CFD analysis

- FP deposition model of steam dryer
- CFD RCS model

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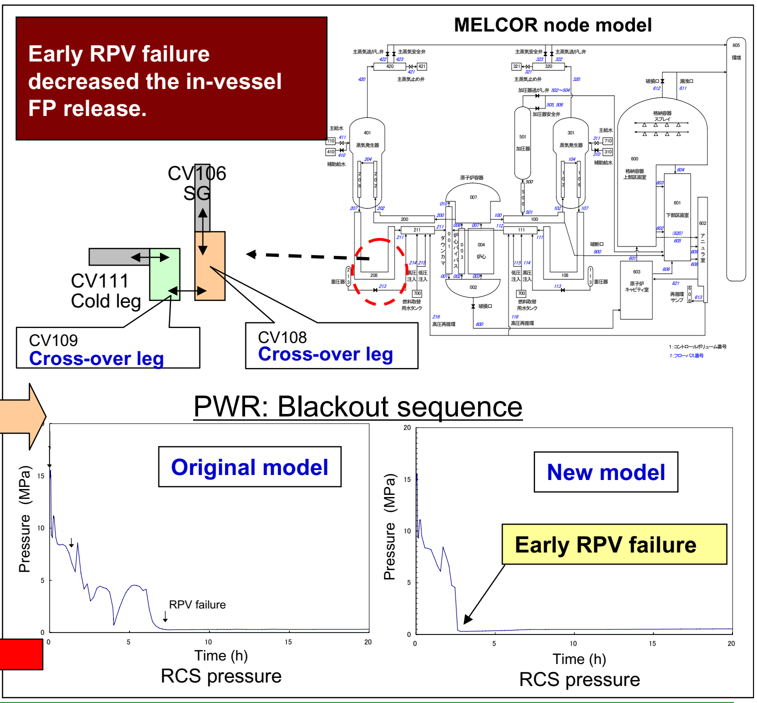
Node model of cross-over leg

One volume node model has been applied to cross-over leg in JNES' MELCOR analysis .

With two control volume model of cross-over leg, the loop seal was formed at cross-over leg.

With the loop seal formation, RPV failed early due to loop flow stagnation and rapid heating of fuel rods.

Applying the two control volumes to the MELCOR cross-over leg node model, thermal hydraulic phenomena and FP behavior were significantly affected.



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Pressurizer Relief Tank Model

Pressurizer relief tank was generally neglected in the system model for level-2 PSA MELCOR analysis.

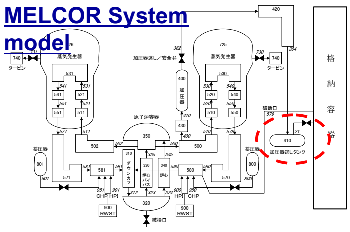
During PWR transient sequence released FP in the core region transferred to containment vess through PORV and Pz relief tank

Pressurizer relief tank may have FP retention effects by pool scrubbing.

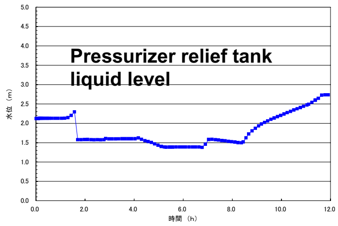
Applying the pressurizer relief tank control volume with aerosol and gaseous FP retention models to MELCOR analysis.

Remarkable amount of aerosol and gaseous FP was captured in the tank by pool scrubbing effects.

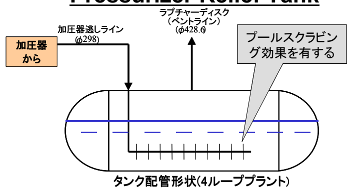
MELCOR System model

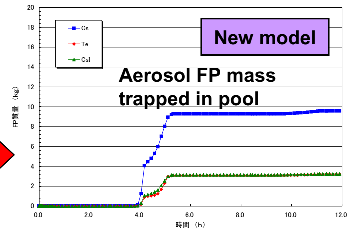


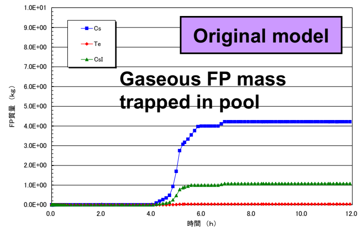
PWR: Station back-out



Pressurizer Relief Tank







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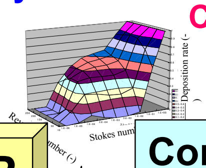
Approach of MELCOR improvement with CFD

- CFD model validation with experimental data
- Modification of MELCOR model with CFD analysis

Ex: SG secondary model

CFD code (Mechanistic analysis) → **Experiment**: **ARTIST Test analysis** — Validation of CFD model → **Actual plant**: **Plant analysis** — Application of validated CFD model → **Comparison CFD vs MELCOR** → **Correlations or Modification of model parameter** — Application of CFD results → **MELCOR code** (System analysis)



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FP Deposition Model of Steam Dryer

SGTR severe accident is a CV bypass sequence and risk dominant

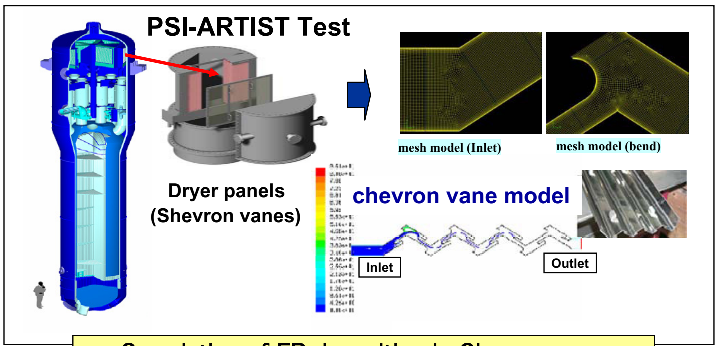
FP deposition in failed SG was almost neglected

FP deposition in failed SG was investigated by PSI-ARTIST program

Validated CFD model with ARTIST test was applied to produce correlations of FP deposition for MELCOR

･Correlations of FP deposition at steam dryer (Shevron vane) with Re number and Stk number

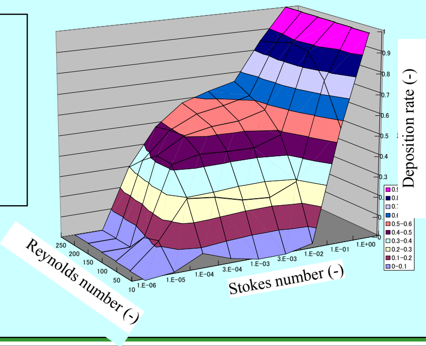
PSI-ARTIST Test



Correlation of FP deposition in Chevron vane

【CFD conditions】

- FLUENT 6.2
- 2D 100K mesh model
- Turbulence model：RSM
- Velocity：0.10～0.30 m/s
- Diameter：0.10～40 μ m



MELCOR improvement with CFD analysis

To study the applicability of CFD to MELCOR lumped-parameter code.

To reflect three dimensional effects to MELCOR models with CFD calculation.

To apply CFD to the FP behavior prediction during a reflux cooling mode with a loop seal

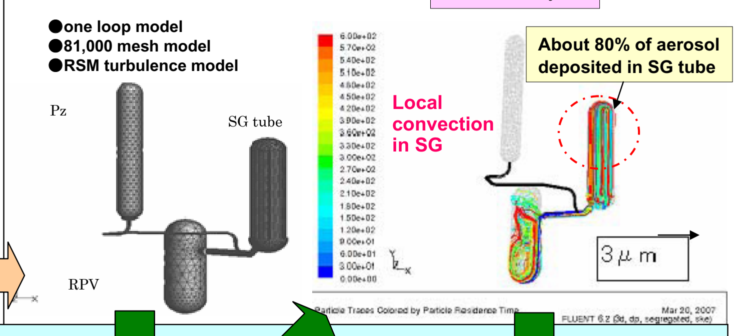
CFD calculation showed that large amount of FP deposited on hot-leg, SG plenum, and SG tube

CFD RCS model need to be validated with experimental data

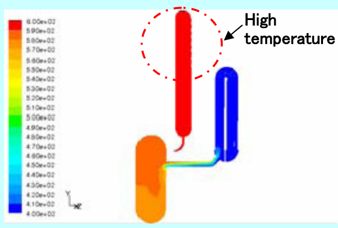
CFD RCS model

- one loop model
- 81,000 mesh model
- RSM turbulence model

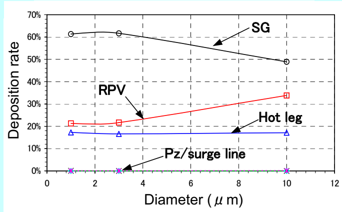
Aerosol analysis



Thermal hydraulic analysis



Temperature in RCS



Distribution of deposited aerosol

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4. International cooperative experimental research for key phenomena and modeling

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

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Summary

- With the expansion of application area of risk-informed regulation, severe accident codes need to have higher quality and reliability.
- To Improve and validate the integrated severe accident code MELCOR:
  - (1) Code validation with experimental data
  - (2) Applications of CFD to complement the lumped parameter code MELCOR
- Need of further experimental data and CFD analysis to complement the MELCOR and reduction of the uncertainty bound in severe accident analysis.

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Ⅲ-C-3

Validation of MELCOR Iodine chemistry model with BIP Test Data

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Korea Institute of Nuclear Safety

The 10th Korea-Japan Joint Workshop on PSA (KJPSA)
Jeju, Korea, May 19, 2009

Introduction (1/3)

❑ **Background**

- **Understanding of iodine behaviour in the containment on accident condition is essential because Iodine is a critical source term for early fatality**
- **For consequence analysis, KINS uses the MELCOR code**
  - MELCOR code has 'Iodine pool chemistry model', with which we can simulate the formation of radioactive molecular iodine gas ($I_{2(gas)}$) from containment sump under the severe accident condition
  - To validate the MELCOR iodine pool chemistry model, calculation results are compared with RTF experiment data
  - These experimental data were obtained from BIP project and ISP41

10th Korea-Japan Joint workshop on PSA Jeju, Korea, May 19, 2009 -2-

Introduction (2/3)

❑ OECD/NEA BIP (Behaviour of Iodine Project)

- **An international research program('07~'10) for analyzing the behavior of iodine in containment with the RTF (Radioiodine test facility) experiment facility**
- **Korea is taking part in BIP (Japan as well)**
- **AECL of Canada, the operation agent**
  - releases five experimental data about iodine behavior to participants and,
  - conducts new experiments about organic iodine reactions with painted surface (absorption rate, organic iodine formation rate etc.)
- **Summary of Progress**
  - '07. 9 : 1st BIP PRG (Program Review Group) meeting
  - '08. 2 : First data(BIP-1) release
  - '08. 5 : 2nd BIP PRG meeting
  - '08. 6 : Second data(BIP-2) release
  - '08.11 : 3rd BIP PRG meeting
  - '09. 1 : Third data(BIP-3) release
  - '09. 5 : 4th BIP PRG meeting

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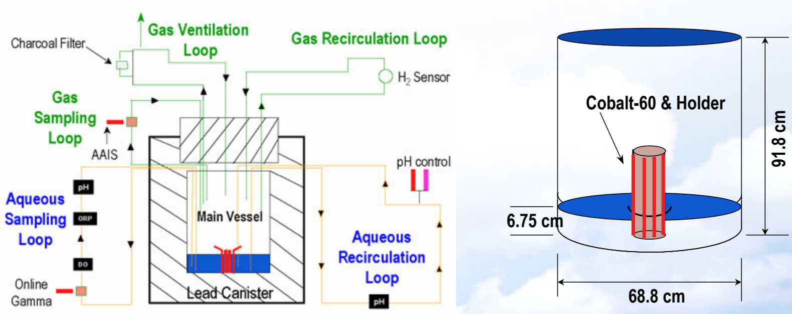
Introduction (3/3)

- **RTF (Radioiodine Test Facility)**
  - The RTF is a test facility for simulating containment in accident conditions
  - Located in CRNL (Chalk River Nuclear Laboratories) of AECL, Canada
  - It was Operated during 1986~1999
  - Re-operated by BIP project since 2007



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Pool Chemistry Model of MELCOR (1/3)

- NUREG-1456 assumes that 95% of the iodine reaching the containment is in aerosol form (principally, CsI)
  - Initially most of the iodine aerosol will be trapped in sump water
  - However, there are processes regenerating gaseous forms of iodine that release into the containment atmosphere, thus becoming available for release to the environment
- MELCOR PCM uses following chemistry to predict the iodine concentration in the containment atmosphere
  - The radiolysis of the air and cable insulations
  - Aqueous phase pH
  - Molecular iodine ($I_{2(aq)}$) generation in aqueous phase
  - Molecular iodine ($I_{2(gas)}$) partitioning to gas phase

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Pool Chemistry Model of MELCOR (2/3)

- Molecular iodine(I_2) generation chemistry
  - In aqueous phase, molecular iodine(I_2) is generated from iodide ions

$$2I^- + 2H^+ + \frac{1}{2}O_2 \Leftrightarrow I_2 + H_2O$$

$$2I^- \xleftrightarrow{radiation} I_2$$

  - Important parameters of these reactions are temperature, radiation dose and pH level
    - Hydrogen ion concentration↑ (pH↓) : Hydrolysis of I_2 will be suppressed, and $I_{2(gas)}$ concentration↑
    - pH control system of NPP
      - It is also important to prevent corrosion of components
      - Adding sodium hydroxide(NaOH) to containment spray water
      - TSP(Tri-sodium phosphate, Na_3PO_4) container in containment floor

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Pool Chemistry Model of MELCOR (3/3)

- $I_{2(gas)}$ concentration is calculated by $I_{2(aq)}$ concentration of pool using partition coefficient
  - Partition coefficient : Ratio of concentration between atmosphere and pool

$$\frac{d[I_{2atm}]}{dt} = k_{pool}\frac{A_{pool}}{V_{atm}}\left(\frac{[I_{2aq}]}{PC_{I_2}} - [I_{2atm}]\right)$$

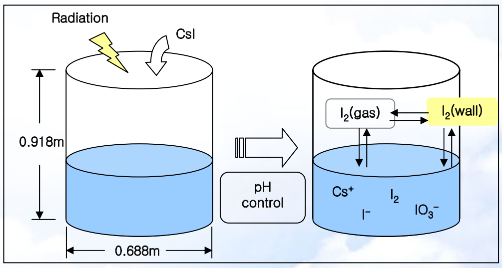
- $[I_{2atm}]$ = atmospheric iodine concentration (kmole/m^3)
- $[I_{2aq}]$ = bulk pool iodine concentration (kmole/m^3)
- k_{pool} = mass transfer coefficient from pool surface to atmosphere (m/s)
- PC_{I2} = partition coefficient of iodine (function of Temp. in PCM)

$$\log_{10} PC_{I2} = 13.5467 - 0.0605142T + 7.166\times10^{-5}T^2$$

Analysis of RTF tests (1/5)

❑ Common condition of RTF tests

- **Water phase 25 L , Gas phase 315 L**
- **Iodine source : ~10^{-5}M CsI labeled with ^{131}I**
- **Tests started at pH 10 (CANDU HTS condition)**
- **^{60}Co source (0.5~2kGy/hr)**



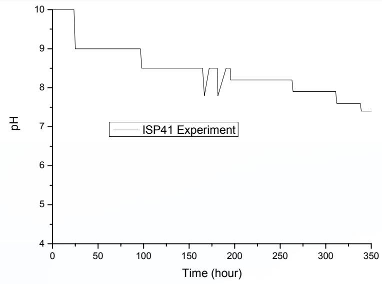
Analysis of RTF tests (2/5)

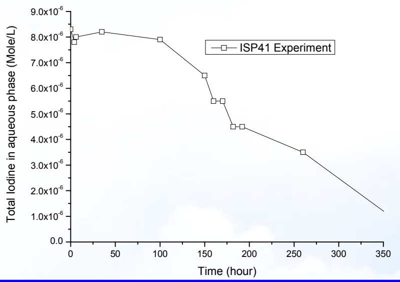
❑ ISP41 experiment

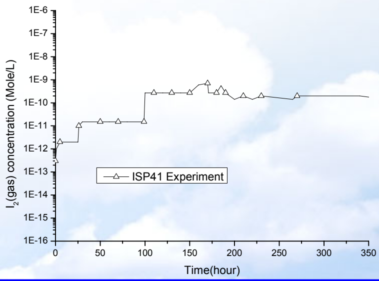
- **ISP41(International Standard Problem no. 41) is suitable for validating the ability of a code about the behaviour of iodine volatility with pH change**
- **Dose rate and temperature were constant, and pH was controlled**

Dose rate	1.36kGy/hr
Temperature	25℃
Initial I_2	9 x 10^{-6} mol/L CsI
Aqueous Volume	25L
Gas Volume	315L
Aqueous Surface Area	5200cm^2
Interfacial Surface Area	3700cm^2
Gas Surface Area	22000cm^2
Pool pH	*Fully Controlled*

Analysis of RTF tests (3/5)







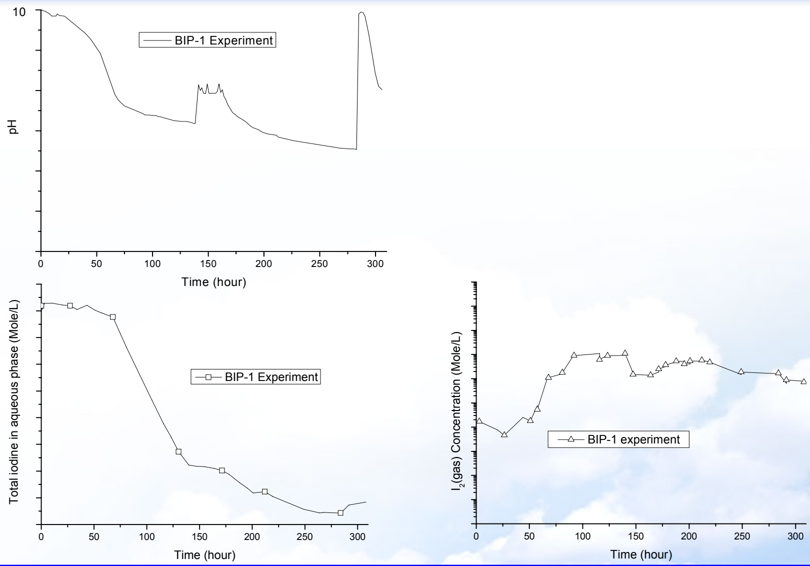
Analysis of RTF tests (4/5)

❑ BIP-1 experiment

- Actually, BIP-1 is an organic iodine formation experiment with organic impurity, but $I_{2(gas)}$ formation data is available to our purpose
- Dose rate and temperature were constant, and pH was partially controlled
- Another conditions are similar to ISP41

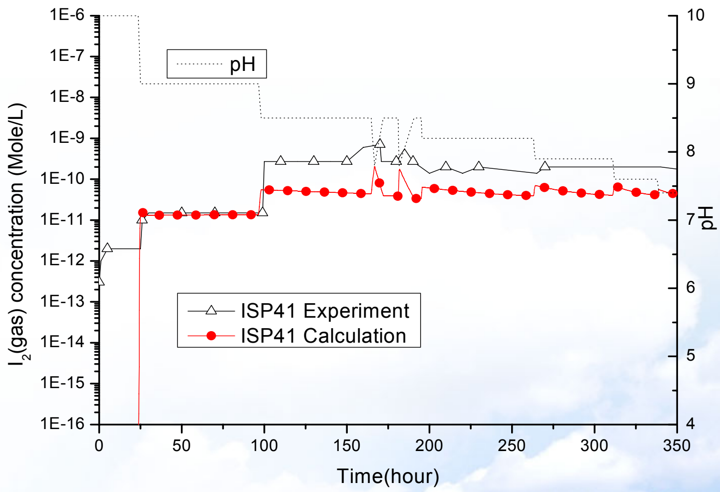
Dose rate	*Less than ISP41*
Pool pH	*Partially Controlled*
Organic impurity	*MIBK (Methyl Isobutyl Ketone)*

Analysis of RTF tests (5/5)



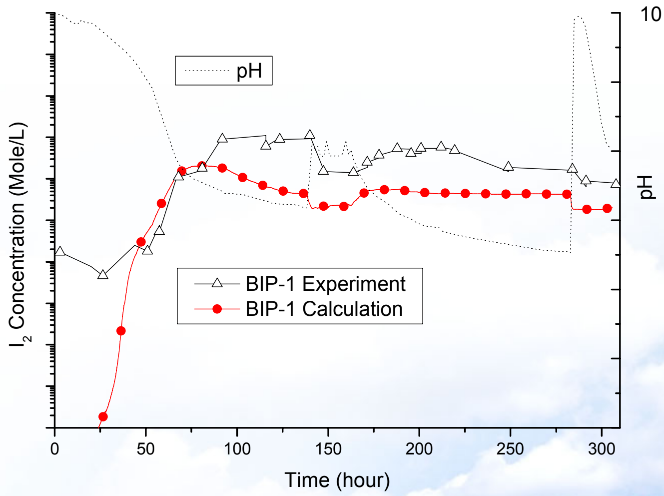
Results of Calculation (1/4)

❑ Result of ISP41 calculation



Results of Calculation (2/4)

❑ Result of BIP-1 calculation



Results of Calculation (3/4)

- Generally, overall trend of $I_{2(gas)}$ concentration is similar to the experimental result in both calculation cases
  - In base phase (pH 7~10), the more pH decrease, the more $I_{2(gas)}$ concentration increases
- MELCOR extremely underestimates $I_{2(gas)}$ concentration when pH is greater than 9.0
  - It may be insignificant result because maximum sump pH of PWR in accident condition is about 8.5
  - For example, in case of Yonggwang 3,4 DBA analysis, pH range is 7.0~8.5 in 49~54℃ considering operation of hydrazine(N_2H_4) and TSP

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Results of Calculation (4/4)

- MELCOR estimates $I_{2(gas)}$ concentration accurately when pH is about 8.0~9.0
- MELCOR code underestimates $I_{2(gas)}$ concentration when pH is lower than 8.0
  - $I_{2(gas)}$ concentration in calculation result is about ten times lower than experimental results
  - In this range, it is necessary to improve the pool chemistry model of MELCOR

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Conclusions

- ❑ **The MELCOR code can properly simulate the molecular iodine gas formation against sump pH in accident condition**
- ❑ **The calculation result shows some model development needs because there is a tendency to underestimate $I_{2(gas)}$ concentration**

Ⅲ-C-4

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

Analysis of RCS Feed & Bleed Operation to Mitigate a Severe Accident for OPR1000

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Content

- Introduction
  - Research Needs & Backgrounds
  - Objective
- General Plant Data of the OPR1000 (=KSNP)
- SCDAP/RELAP5 Input Model
- SCDAP/RELAP5 Results and Discussion
- Conclusions

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Research Needs & Backgrounds(1/3)

- **Objective of the RCS Feed & Bleed Operation**
  - to prevent core damage in the high RCS pressure sequence, such as total LOFW (in EOP)
  - to prevent reactor vessel failure during a severe accident (in SAMG)
- **Feed & Bleed Timing and Capacity is very Important.**
  - positive effect: to prevent or mitigate a severe accident
  - negative effect: to contaminate the containment

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Research Needs & Backgrounds(2/3)

- **Feed & Bleed Operation of the RCS during a Severe Accident**
  : very important *conjugate In-Vessel strategy* to mitigate a severe accident in SAMG.
- **Coolant Injection into the In-Vessel (RCS Feed Operation): Operation of SI (HP&LP)**
- **RCS Bleed of Depressurization**
  - Direct: SDS (Safety Depressurization System)
- **Positive Effect of Feed & Bleed Operation in SAMG**
  - to prevent the reactor vessel failure and the RCS pipe line failure

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Research Needs & Backgrounds (3/3)

- **Negative Effects of Feed & Bleed Operation in SAMG**
  - Increase of the in-vessel steam explosion possibility
  - Increase of the hydrogen generation
  - Over-pressurization of the containment
- **Two factors on the consequences**
  - Operator action timing
  - Operator action capacity
- **Detailed evaluation by using the best estimate computer code is necessary to develop the SAMG Procedure.**

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Objective

- **To evaluate the feed and bleed operation for the total LOFW by using the SCDAP/RELAP5 in the severe accident (SAMG) of the OPR1000**
  - Effect of Operator Action Timing on the Consequences
  - Effect of Operator Action Capacity on the Consequences

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OPR1000 (=KSNP)

- To develop by incorporating the latest technologies and the experiences in construction and operation gained from previous NPPs, including the EPRI ALWR requirements.

- General Plant Data of OPR (Optimized Power Reactor) 1000
  - Two Hot Legs & Steam Generators, Four Cold Legs & RCPs
  - Power: 1,000 MW_e
  - Pressurizer Safety Relief Valve (SRV) to control RCS pressure
  - 2 SDS (Safety Depressurization System) Valves for a RCS bleed
  - 2 low & high pressure safety injections to 4 cold legs

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SAMG of OPR1000

- Entry Condition: 650 ˚C of Core Exit Temperature

- In-Vessel Strategies
  . Coolant injection into the Steam Generator to 63 % NR of Water Level
  . Depressurization of the RCS to approximately 2.9 MPa (RCS bleed operation)
  . Coolant injection into the In-Vessel to approximately 371.1 ˚C of Core Exit Temperature (RCS feed operation)

- Other Strategies
  . Coolant injection into the Containment
  . Control of FP Release
  . Control of Containment Pressure
  . Control of Hydrogen in the Containment

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Dominant Severe Accident Sequences in OPR1000

- SBLOCA(Small Break Loss Of Coolant Accident) without SI(Safety Injection) : 22.4 %
- SBO(Loss of AC and/or DC power): 14.4 %
- STGR(Steam Generator Tube Rupture): 13.8 %
- Total LOFW (Loss of Feed Water): 13.8 %
- LBLOCA without SI: 12.7 %
- MBLOCA without SI: 7.7 %

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Analysis Method of Total LOFW

- Base Case Total LOFW
  - Transient initiated at 0 seconds
  - Reactor & Turbine are tripped by a high Pressurizer pressure signal
  - RCS pressure is controlled by the Pressurizer SRV
  - Secondary pressure is controlled by the Steam Generator SRV
- RCS Bleed Operation: 1 or 2 Pressurizer SDS Valves

  (SAMG Strategy: Depressurization of the RCS)
- RCS Feed Operation:

  4 Trains of Safety Injections (Low & High)

  (SAMG Strategy: Coolant injection into the RCS)

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SCDAP/RELAP5/MOD3 Computer Code

- To develop as a best estimate computer code for severe accident analysis from initiating event to reactor vessel failure at the INL under the sponsorship of the U.S.NRC.
- To merge the RELAP5/MOD3, SCDAP, and COUPLE models.
- The RELAP5/MOD3 model: the severe accident situations of the overall RCS thermal hydraulic response.
- The SCDAP model: core damage progression and fission product release from fuel rods to coolant.
- The COUPLE model: the heat-up of the debris bed and surrounding structures in the reactor lower head vessel after the melted core material relocates to the lower plenum

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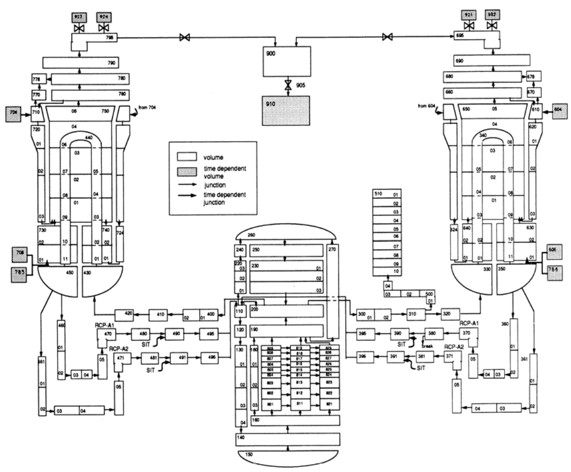
SCDAP/RELAP5/MOD3 Input Model(1/2)

- RELAP5 Part
  - Simulation of Reactor Vessel, Reactor Coolant System, Steam Generator, Pressurizer, RCP, Feed & Steam Lines, Four Safety Injection Tanks, and Three Trains of Safety Injection (High & Low)
  - Core: 3 Channels with Cross Flow Junctions



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SCDAP/RELAP5/MOD3 Input Model(2/2)

- RELAP5 Part for Coolant Injection into the In-Vessel with RCS Depressurization
  - Secondary Pressure Control: S/G Safety Valves
  - RCS Bleed: Two Pressurizer SDS Valves
  - RCS Feed (Coolant Injection into the In-Vessel): 4 Trains of the Safety Injection
- SCDAP Part: 6 components(3 fuel rods, 3 control rods), 10 axial nodes, 6 radial nodes for fuel rods, 2 radial nodes for control rods
- COUPLE Part: 234 nodes & 204 elements
- Creep Input: S/G Tube, Pressurizer Surge Line, Hot Leg, Reactor Vessel Wall

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Total LOFW

- Base Case Total LOFW
  - Initial SRV Opening Time = 1,606 sec
  - Reactor Vessel Failure Time = 6,115 sec
  - Available Time after SRV Initial Opening = 4,509 sec (75 minutes)
- Only RCS Bleed Case
  - RCS Bleed Time: 5-60 minutes after the SRV opened initially
  - RCS Bleed Capacity: 1 or 2 SDS Valves
- RCS Feed & Bleed Operation (Coolant Injection into the In-Vessel with RCS Depressurization)
  - RCS Bleed Time: 40 minutes after the SRV opened initially (SAMG entering point)
  - RCS Bleed Capacity: 2 SDS Valves
  - RCS Feed Capacity: 1- 4 trains of HPSI or LPSI

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S/R5 Results in Only RCS Bleed Case

Case	SIT Actuation Time (s)	RV Failure Time (s)	RCS Pressure at RV Failure (MPa)
Base	-	6,115	15.2
SDS2- 40 minutes	4,142	23,995	0.81 < 2.9 MPa
SDS2- 50 minutes	5,090	5,995	3.10
SDS1- 5 minutes	4,904	6,438	4.08
SDS1- 30 minutes	4,930	10,655	3.17

(Unit = sec)

The opening of two SDS valves till 40 minutes (SAMG entering time) after initial opening of SRV can depressurize the RCS sufficiently. The opening of one SDS valve cannot depressurize the RCS sufficiently enough.

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S/R5 Results in RCS Feed and Bleed Operation Case

Case	SIT or HPSI Act. Time (s)	RV Fail. Time (s)	RCS Pressure at RV Fail. (MPa)
Base	-	6,115	15.2
SDS2- 40 min (*)	4,572	23,388	(0.41)
SDS2-40 min, HPSI1-20208	20,200	No RV Failure to 50,000 sec	
SDS2-40 min, HPSI1-21000	21,000	22,225	0.9
SDS2-40 min, HPSI3-21000	21,000	22,285	1.0

(Unit = sec)

Only one train operation of the HPSI at 20,200 seconds with a RCS depressurization by using two SDS valves at 40 minutes after an initial opening of the SRV prevents a reactor vessel failure. The safety injection time of 21,000 seconds is too late.

(*) 40 minutes: SAMG entering point

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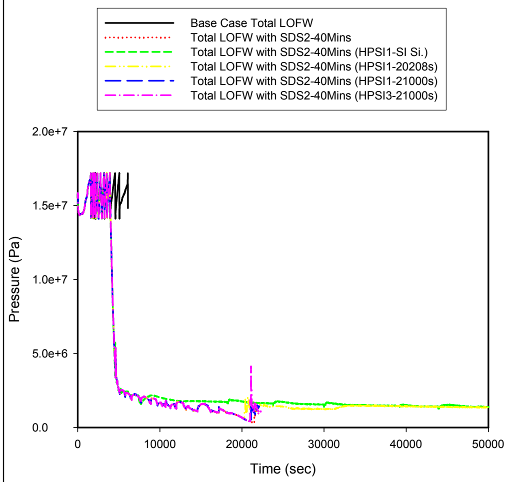
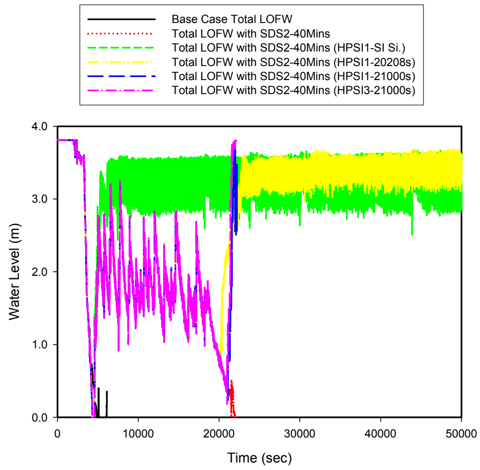
RCS Feed and Bleed Operation



Pressurizer Pressure



Collapsed Water Level in the Core

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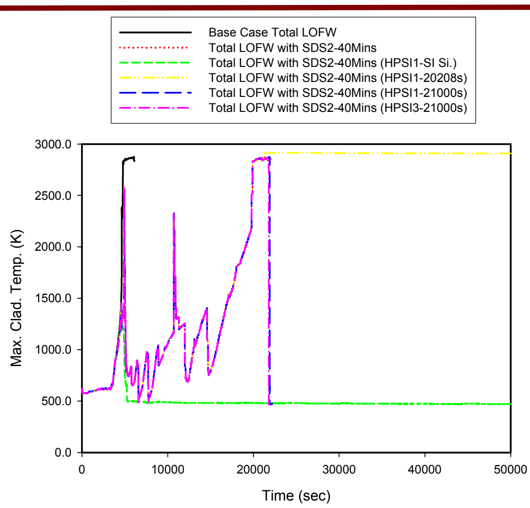
RCS Feed and Bleed Operation



Maximum Temperature of Fuel Cladding

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Conclusions

- Only one train operation of the HPSI at 20,200 seconds with a RCS depressurization by using two SDS valves at 40 minutes after an initial opening of the SRV (SAMG entering point) prevents a reactor vessel failure for the total LOFW in the OPR1000.
- Safety Injection time at 21,000 seconds is too late for prevention of reactor vessel failure.
- These results will be used for SAMG development (detailed operator action procedure) in the OPR1000.

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Thank you very much for your attention !

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