Session V-A

Risk Informed Application (II)

Summary of Session V-A: Risk Informed Application (II)

Chair: Akihide HIDAKA (JAEA), Myung-Ki KIM (KEPRI)

At this section chaired by Akihide Hidake of JAEA and Myung-Ki Kim of KEPRI, four papers are presented in the Diamond Hall A on the second of the workshop. About 20 people took part in the section and made interesting questions and good comments. Through the section Korea participants and Japan participants have an opportunity of knowing each side of RIR status.

The first paper is "Current Status of RIR Implementation in Korea" presented by Namduk Suh worked at KINS of Korea. It deals with Korea RIR implementation status and future activities. He expresses the role of the utility and more aggressive implementation for successful settlement of RIR. He make a conclusion that the objective and outcomes of RIR is not clearly shown until now so that more systematic and integrated way is needed and deregulation and enhanced safety are in parallel considered in RIR.

The second paper is "Shutdown Risk Monitoring in TEPCO" presented by Hidetaka Imai of TEPCO of Japan. He presents the risk monitoring system during shutdown using PSA results and the lessons learned by introducing risk monitoring system. He make known that even though its initial stage, the staff and engineers of the plant have positive mind set.

The third paper is "Increase Safe and Operating Reliability by Development of Planned Outage Process Standardization" presented by Jonghyuck Park of KEPRI of Korea. This paper deals with the plant outage process standardization considering the time-lines and adequacy of pre-outage planning and outage scheduling. He explained the standardized schedule program the so-called, NPOMS, is currently applied in KHNP all plants.

The last paper is "Safety and Economic Results of Risk-Informed Inservice Inspection Program at Ulchin Units 3&4" presented by Bag Soon Chung of KEPRI of Korea. He explained the status of RIR, that the RI-ISI research was started 10 year ago, and through long term regulatory institute's review the topical report of RIR has been approved and RIR for plants are being prepared with topical report. He presents the estimating costs saving is 1000,000 USD for one unit by applying RIR program and strongly expressed RIR yields both enhancing safety and reducing radiation exposure.

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KJPSA, Jeju, May 18 ~19, 2009

Current Status of RIR Implementation in Korea

May 18~19, 2009

Presented by Namduk SUH

RIR P.M.
Korea Institute of Nuclear Safety

KJPSA, Jeju, 2009

Contents

KINS

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I. Introduction

KINS

❑ Historical Background

- Sept. 1994, policy statement on nuclear safety
  - ✓ recommend to introduce risk informed regulation
- Aug. 2001, policy statement on severe accident of NPP
  - ✓ safety goal
  - ✓ probabilistic safety assessment
  - ✓ severe accident prevention and mitigation capability
  - ✓ severe accident management program
- Dec.2002, NSC (Nuclear Safety Commission) recommends introduction of 「maintenance program」 to enhance safety

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I. Introduction

KINS

❑ Items in Implementation

- RIR items currently on going in Korea
  - ✓ RI Licensing Amendment
  - ✓ RI Integral Leak Rate Test
  - ✓ RI In-Service Inspection of Piping
  - ✓ Maintenance Rule
  - ✓ RI Periodic Inspection
  - ✓ Risk Monitoring System

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I. Introduction

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❑ Needs to evaluate current status of RIR Implementaion

- Utility side
  - ✓ utility wants expansion of RIA , given the efforts for PSA of all operating plants in Korea
- Regulatory body side
  - ✓ efforts to pursue effectiveness and efficiency
- More than 15 yrs of activities need to be reviewed and evaluated to step forward with confidence
  - ✓ are we on right way?
  - ✓ optimization through RIA seems good, but what about the safety?

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II. Current Status of RIR Implementation

KINS

II.1 Risk-Informed Licensing Amendment

❑ **Introduction**

- Integrated regulatory decision-making on the application of risk-informed licensing amendment : Extension of STI / AOT

❑ **Comparison with current approach**

Current Approach	Risk-Informed Approach
- Engineering judgment - Deterministic safety assessment (Defense-in-depth, safety margin)	- Engineering judgment - Deterministic safety assessment (Defense-in-depth, safety margin) - Probabilistic safety assessment and operating experience - Performance monitoring after change

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II. Current Status of RIR Implementation

KINS

II.1 Risk-Informed Licensing Amendment

- **Current Status**
  - ESFAS/STI extended from 1 month to 3 months for 4 NPPs
  - STI/AOT extension for Kori-1 allowed recently
  - AOT of essential Invertor for Uljin 3&4 extended from 24 hrs to 7 days
  - Engineering judgement based on availability of the system, and then the risk information are integrally taken into account
  - Performance monitoring after change is specified in technical specifications
  - MR is not credited yet for performance monitoring

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II. Current Status of RIR Implementation

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II.2 Risk-Informed Integral Leak Rate Test

- **Current Status**
  - Interval of ILRT extended from 5 yrs to 10 yrs
  - 2 consecutive tests of 5 yr interval should satisfy the performance criteria before applying for the extension of test interval
  - Interval extended for 11 units among 20 operating units
  - Other units are waiting for the the condition to be fulfilled

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II. Current Status of RIR Implementation

KINS

II.3 Risk-Informed Piping In-Service Inspection (ISI)

❑ Introduction

- Developing integrated regulatory decision-making on the application of risk-informed piping ISI licensing amendment
- Allowing risk-informed alternate methods to select areas for ISI

❑ Comparison with current approach

	Current Approach	Risk-Informed Approach
Selection of inspection points	- Section XI of ASME code	- Based on high risk significance by PSA - Selecting points by integrated information (piping failure probability, risk impacts, expert panel, etc.)
Inspection method	- Depending on piping size, welding type	- Depending on failure mechanism considering piping failure probability, risk impact of piping failure,

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II. Current Status of RIR Implementation

KINS

II.3 Risk-Informed Piping In-Service Inspection (ISI)

❑ Current Status

- Technical Report for ISI reviewed and accepted on July, 2008
- Utility is preparing for a plant specific application based on the allowed TR methodology
- First application is expected for Uljin 3&4 plants

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II. Current Status of RIR Implementation

KINS

II.4 Maintenance Rule

- **Introduction**
  - NSC recommends introduction of a 「maintenance system」 on 2002. Reduction of reactor trip or power transient caused by BOP in U.S referenced
  - Utility opposes a rule-making for MR, but has developed standard implementation programs
  - Pilot application since Jan., 2007, beginning with Uljin 3&4 and Kori 3&4
- **Current Status**
  - Nov.,2008, KINS evaluated the status of pilot application of Uljin 3&4, with a team of 11 staffs for 5 days
  - General conclusion is as follows

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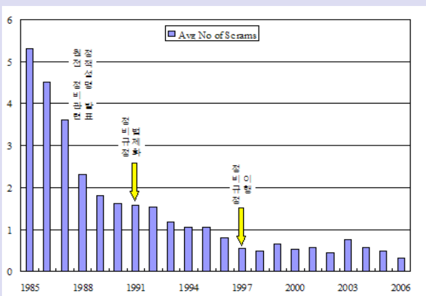
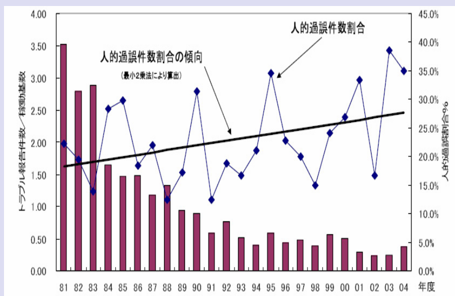
II. Current Status of RIR Implementation

KINS

II.4 Maintenance Rule

- **Main Conclusion on Pilot Application**
  - Direct relation between MR implementation and the reduction of Rx trip or power transient is not clear.
    - ✓ Capacity factor seems sacrficed for safety in case of Japan, but proof of causality in real life is not an easy job anyway !





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II. Current Status of RIR Implementation

KINS

II.4 Maintenance Rule

- Capability of system engineers in coping the safety issues enhanced
- MR enables reliability data to be gathered more systematically and reliably
- MR contributes in enhancing PSA quality
- MR provides tool for performance monitoring after LB changes

❑ Still, lack of clear quantitative effects of implementing MR

❑ MR in Korea is not mature enough yet for a full scope rule-making as in U.S.

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II. Current Status of RIR Implementation

KINS

II.5 Risk-Informed Periodic Inspection (RIPI)

❑ Regulatory Periodic Inspection

- Utility shall receive the RPI during refuelling outage
  - ✓ 18 months currently
- Standard inspection items for KSNP are 80
- Inspection items include ;
  - ✓ Surveillance tests specified in Tech. Spec. (~50%)
  - ✓ In-service inspection and in-service testing (~20%)
  - ✓ BOP (~20%) ; introduced into the RPI since May, 2005

❑ Concept of current RIPI

- Developed and applied since 2006

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II. Current Status of RIR Implementation

KINS

II.5 Risk-Informed Periodic Inspection (RIPI)

- Objectives of RIPI
  - ✓ enhance safety and inspection effectiveness by focusing on risk significant systems
  - ✓ enhance regulatory efficiency by allocating regulatory resources to risk significant systems
  - ✓ graded inspection was in mind
- ❑ Current Status
  - 54 major failure events were evaluated. 24 inspection items were selected and added to the current inspection checklist
  - Selected items came mainly from CCF because CCF is a main contributing factor in the fault trees of PSA

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II. Current Status of RIR Implementation

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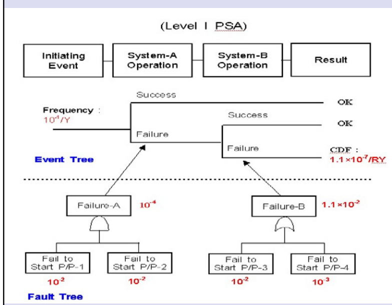
II.5 Risk-Informed Periodic Inspection (RIPI)



Area	Inspection Items
1. Reactor Assembly (5)	∘ Fuel Integrity ∘ Zero Power Rx. Physics Test ∘ Rx. Physics Test during Power Operation ∘ CPC/COLSS Test ∘ Rx and Internal Structures
2. Reactor Coolant Sys.(8)	∘ RCPB Components and Piping ISI ∘ Steam Generator Tube ECT ∘ Water Chemistry ∘ PZR Valves ∘ Rx. Coolant Pumps ∘ RCPB Leak Rate Measurement ∘ RCS Flow Rate Measurement
3. I&C Sys. (8)	∘ RPS Response Time Measurement ∘ ESFAS Response Time Measurement ∘ ESF Slave Relay Functional Test ∘ Control Rod Drop Time Test ∘ Control Rod Position Indication System ∘ Safety-related I&C Sys. Calibration ∘ Seismic Monitoring System ∘ NSSS Integrity Monitoring System
4. Fuel Handling(2)	∘ Fuel Transfer System ∘ SFP Cooling/Purification System
5. Rad. Waste(3)	∘ Rad. Waste Management System ∘ HVAC Performance Test ∘ Radiation Chemistry Management
6. Radiation Control (4)	∘ Health Physics Program ∘ Radiation Measurement and Monitoring ∘ Meteological Monitoring System ∘ Environmental Radioactivity Monitoring
7. Rx. Containment (7)	∘ Containment Isolation System ∘ Combustible Gas Control ∘ Containment Spray System ∘ Containment LLRT ∘ Containment ILRT ∘ Integrity of Containment Liner Plate ∘ Containment Post-tensioning ISI
8. Safety Sys. (3)	∘ Emergency Core Cooling System ∘ Shut Down Cooling System ∘ Refueling Water Storage Tank

Area	Inspection Items
9. Electric Power Supply (7)	∘ EDG Performance Test ∘ Battery/Charger ∘ Safety-related Electric Equipment ∘ Generator System ∘ Transformer System ∘ Switchyard System ∘ EDG Mechanical System
10. Power Conversion (19)	∘ Auxiliary Feed Water System ∘ Main Feed Water system ∘ Main Feeder and Vacuum System ∘ Demineralizing and Circulating Water Sys. ∘ Condensate Storage and Transfer Sys. ∘ Turbine/Generator System ∘ TBN Auxiliary System ∘ Moisture Separator/Reheater/Extraction Sys. ∘ TBN Bypass Sys. ∘ Generator Stator Cooling Sys. ∘ Generator Gas Control/Hydrogen Sealing ∘ MSSV Test ∘ MSIV/MFIV ∘ Integrity of TBN Rotor ∘ Main Feed Water Control System ∘ TBN Control/Protection System ∘ Secondary System I&C Calibration ∘ Condensate and Feed Water Sys. NDE ∘ Carbon Steel Piping Thinning
11. Others (14)	∘ Primary Component Cooling System ∘ Primary Sea water cooling System ∘ Essential Chilled water System ∘ Compressed Air System ∘ Fire Protection System ∘ Chemical and Volume Control System ∘ Safety and Relief Valve Set point ∘ Safety-class Pump and Valve IST ∘ Secondary Sea water cooling System ∘ HVAC System (Mechanical) ∘ Supports and Snubbers ∘ Integrity of Seismic Category I Structures ∘ Safety-related Protective Coating ∘ Safety-class 2, 3 Comp. and Piping ISI

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II. Current Status of RIR Implementation

KINS

II.5 Risk-Informed Periodic Inspection (RIPI)

- RIPI has been applied for more than 3 yrs, but the acceptance of RIPI by KINS inspectors are still low
- Difficulties of CCF related inspection
  - ✓ not clear how and what to inspect
  - ✓ CCF was already checked during design review and why we should confirm it every outage period ?
- Current RIPI methodology needs to be revised
- PSA can provide new insights on the meaning of regulatory inspection.
- Efforts to reduce the number of inspection items, thus enhancing a regulatory efficiency while maintaining the current safety level

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II. Current Status of RIR Implementation

KINS

II.6 Risk Monitoring System

❑ Introduction

- Developed and implemented in compliance with the policy statement on severe accident of 2001
- RIMS for all operating plants were developed and implemented on site

❑ Current Status

- Use is not active at this time
- Use is not officially credited yet
- Utility expects RIMS to be a tool for on-line maintenance

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III. Perspectives of Future Activities

KINS

- Conclusion from evaluation of current RIR implementation
  - RIA is implemented per item and per plants
  - Objectives and outcomes are not clearly shown
  - RIR needs to be pursued in more systematic and integrated way
    - ✓ Korean regulatory environment requires to show that the RIA does not reduce the overall risk of a specific plant
    - ✓ Efforts to enhance safety in compensation of risk reduction following RIA are needed
- Systematic and integrated strategy for RIR implementation is in need
- Efforts to enhance safety are needed in parallel
- Hope to establish a strategic plan of RIR implementation before the end of this year

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V-A-2

Shutdown Risk Monitoring in TEPCO

Hiroki Sato※1, Takahiro Masuda※1, Yasutaka Denda※1,
Mitsuru Yoneyama※2,
Hidetaka Imai※3, Shun-ichi Imai※3, Koichi Miyata※3

※1：Fukushima-Daiichi NPS, TEPCO
※2：Nuclear Engineering Department,TEPSYS
※3：Nuclear Asset Management Department, TEPCO

東京電力
TEPCO

Contents

- Risk evaluation trial
  - ✓Fukushima-Daiichi #5 22nd refueling outage risk evaluation
  - ✓Maintenance schedule modification
- Lessons learned from our trial

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Introduction

◆Background

Risk-informed safety management

✓has been discussed among Japanese utilities and regulatory parties
✓is accumulating industry-wide interest
✓is expected to advance further in the future

◆Aiming of our trial

TEPCO introduced risk monitors in its 3 nuclear power stations; Fukushima-Daiichi, Fukushima-Daini and Kashiwazaki-Kariwa

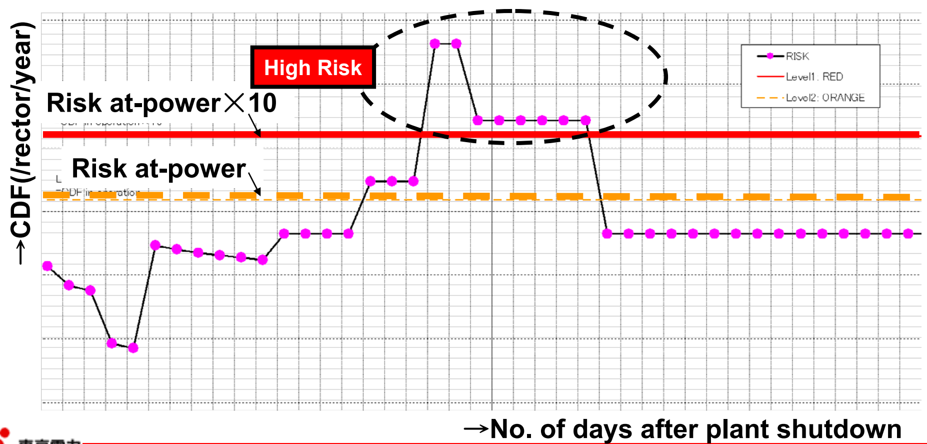
✓**To optimize plant risks during refueling outages**

✓**To raise personnel awareness for reactor safety**

Risk evaluation result

✓Example case
→Fukushima-Daiichi #5 22nd refueling outage

✓Risk at the day 19th to 26th exceeds the CDF at-power ×10



Reason of risk peaking

At the risk peak

Plant Status	
➢All Fuels unloaded ➢Pool Gate opened ➢1 Water intake channel out-serviced	
Heat Removal	**Injection**
RHR(A) ~~RHR(B)~~ FPC(A) ~~FPC(B)~~	LPCI(A) ~~LPCI(B)~~ CS(A) ~~CS(B)~~ ~~MUWC(A)~~ ~~MUWC(B)~~

✗ out serviced by maintenance

Reason of risk peaking

✓2 heat removal systems
✓4 injection systems
} unavailable

✓If RHR(A) is lost, decay heat cannot be removed enough

Decay heat＞Heat Removed by FPC(A)

✓MUWCs(highly reliable injection systems) are unavailable

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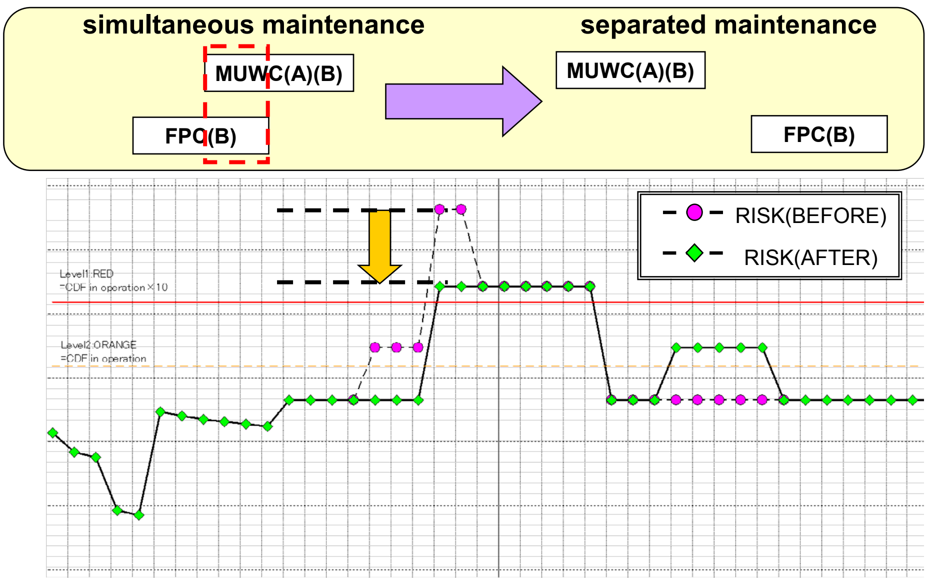
To reduce peak level



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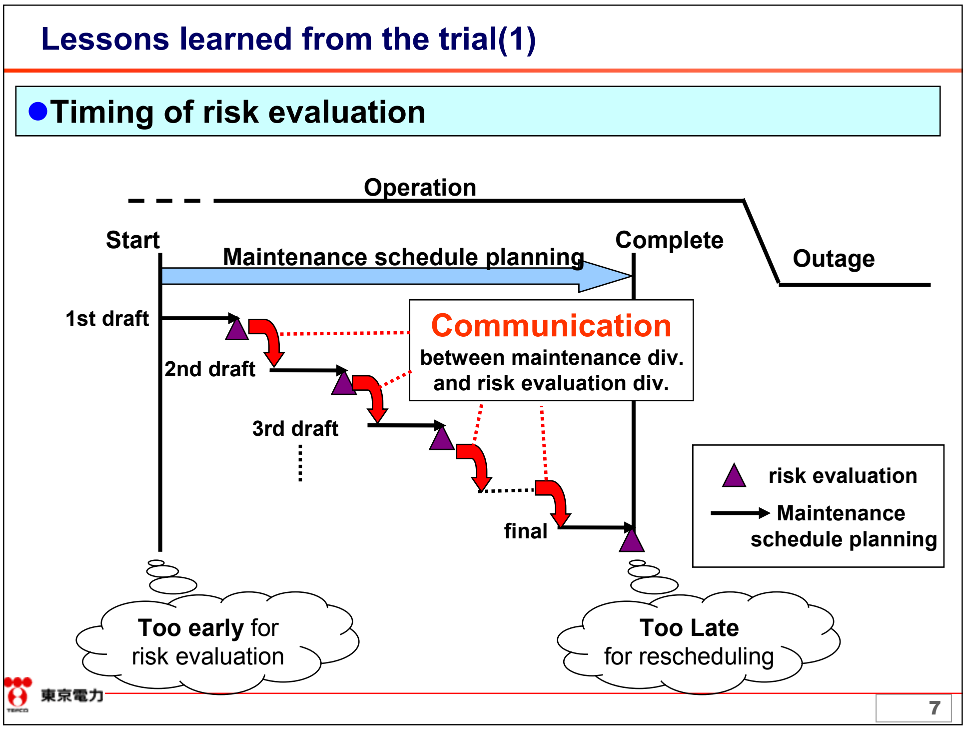
Lessons learned from the trial(1)
●Timing of risk evaluation
Operation
Start
Maintenance schedule planning
Complete
Outage
1st draft
2nd draft
3rd draft
final
Communication
between maintenance div. and risk evaluation div.
risk evaluation
Maintenance schedule planning
Too early for risk evaluation
Too Late for rescheduling
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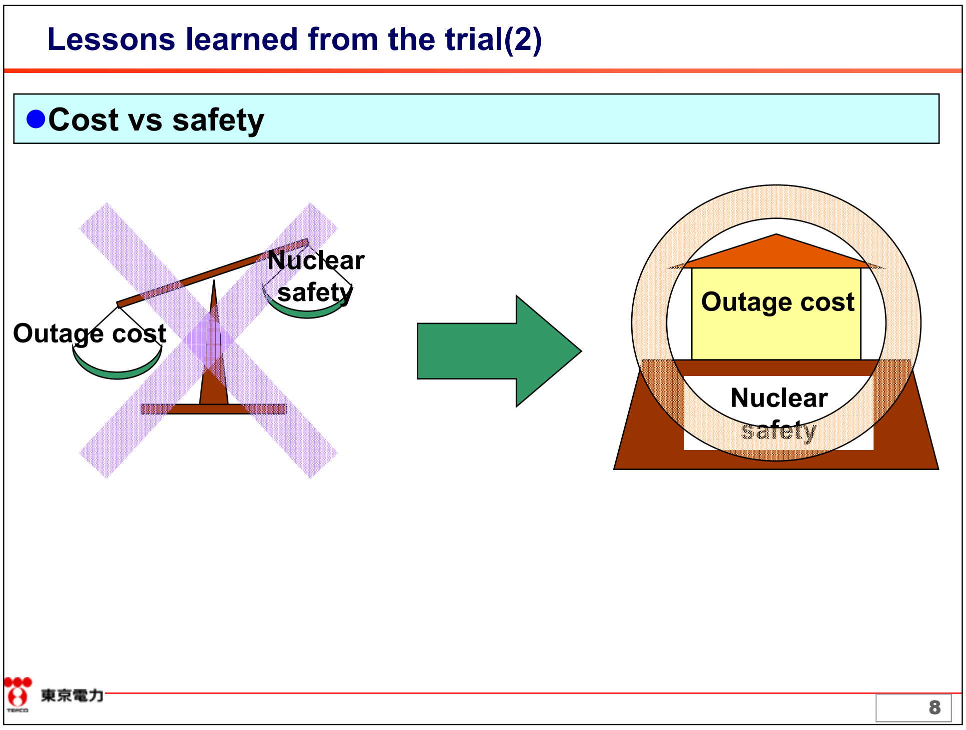
Lessons learned from the trial(2)
●Cost vs safety
Nuclear safety
Outage cost
Outage cost
Nuclear safety
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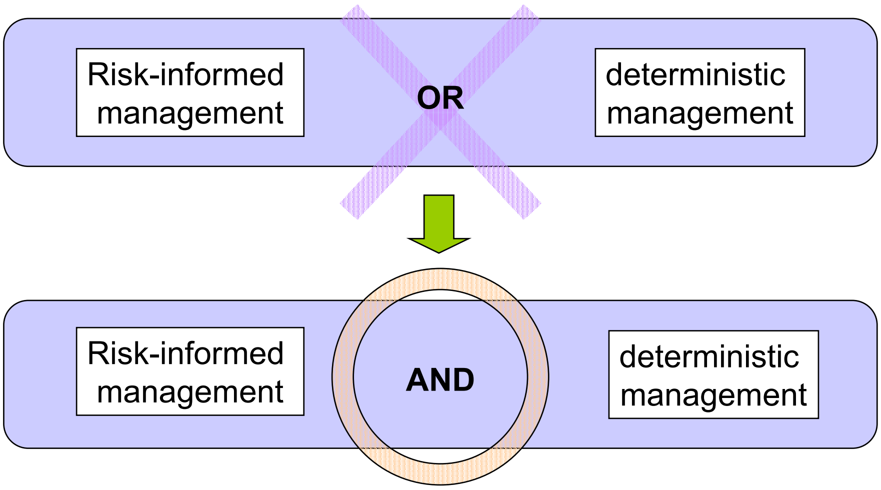
Lessons learned from the trial(3)

● **Coexistence of risk-informed management and deterministic management**



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Conclusion

✓ We have introduced a risk monitor and carried out shutdown risk monitoring trial for about a year.

✓ We already have some examples of maintenance schedule modification using risk information.

✓ We obtained some lessons learned through the trial.

- Timing of risk evaluation
- Cost vs safety
- Coexistence of risk-informed management and deterministic management

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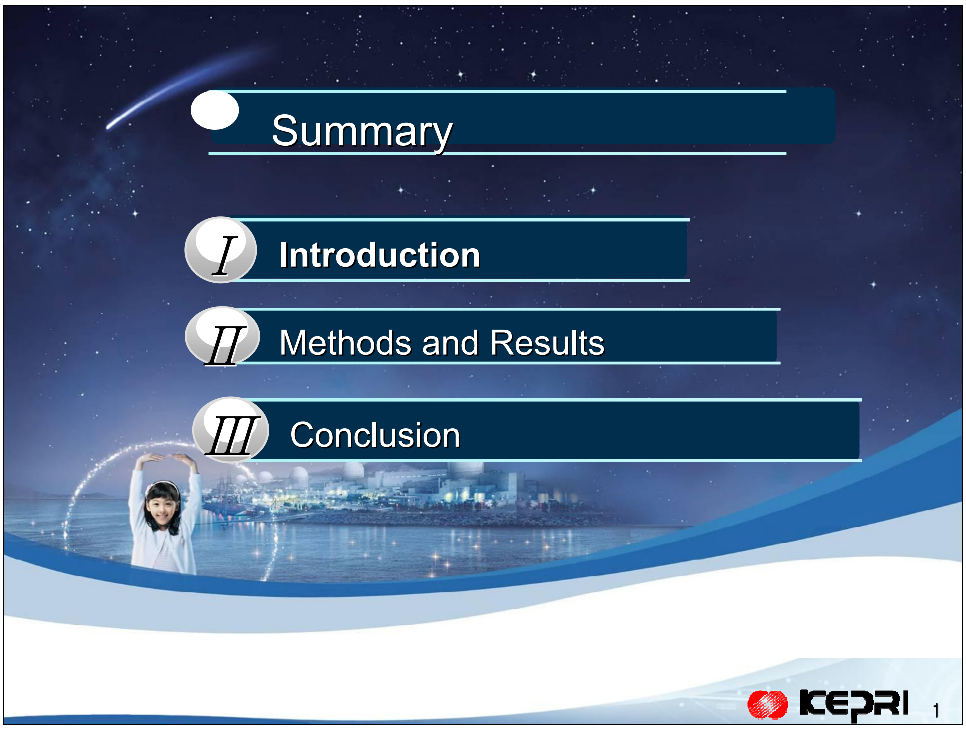
V-A-3


Increaase Safe and Operating Reliability by Development of Planned Outage Process
Standardization for Nuclear Power Plant
2009.05.19
Jong-Hyuck Park
KEPRI


Summary
I Introduction
II Methods and Results
III Conclusion
KEPRI
1

Introduction

◆ I study to make standard outage process from the preparation stage to outage performance monitoring and evaluation stage .

◆ The first critical factors is the timelines and adequacy of pre-outage planning and preparation.

- ✓ Developed the effective pre-outage planning to manage high levels of nuclear safety and performance

◆ The second critical factors is outage scheduling which reflects the plant' s expectation for the conduct of outage.

- ✓ Outage performance cannot be optimized if a schedule is not available to control and properly sequence outage task.
- ✓ Developed 5 standard groups which can be applied 20 power plants in Korea
- ✓ The integration of outage activities is key to a successful schedule.
  - ▪ Preventive and corrective maintenance, modification, surveillance testing, post-maintenance and post modification testing and in-service

KEPRI 2

Introduction

◆ The third critical factors is outage performance monitoring and evaluation

- ✓ Monitor the progress toward meeting outage goals and gather lessons learned from outages.
- ✓ Management uses information about problems and successes from recent outages to improve performance of subsequent outages.

◆ In this paper, we apply the several concept of plant outage process standardization to develop the nuclear power plant outage management system(NPOMS).

◆ The main contribution of this paper is an standard outage management system development to use 20 nuclear power plant in korea.

KEPRI 3

Methods and Results

1. Selection of the Outage Standardization Objects

◆ The quality of work performed during an outage is directly impacts plant reliability during the next operating cycle

◆ Prior to select the outage standardization objects, the design requirements and equipments suppliers and electrical power capacity for 20 operating plants are reviewed and classified to maximize the effects of standardization.

◆ The selected standardization objects
- ✓ 5 groups of operating plants classification
- ✓ outage schedule structure
- ✓ work breakdown structure
- ✓ work group and task list
- ✓ outage planning and performance monitoring
- ✓ outage management system development

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KEPRI 4

Methods and Results

2. Group of Operating Plant Classification

◆ According to the reactor type and power capacity the 5 group of operating plants is classified
- ✓make the bases of standardization.
- ✓classify the other fields of standardization objects

Group	Reactor Type	Capa. (Mw)	Plant Supplier	Plant Site	Plant No.
1	PWR 600	687 660	WH	KORI1	1 2
2	PWR900	950	WH	KORI2 YGN1	3&4 1&2
3	PWR900	950	FRAMATOME	EULJIN1	1&2
4	PWR1000	1,000	DOOSAN	YGN2 YGN3 EULJIN2 EULJIN3	3&4 5&6 3&4 5&6
5	PHWR600	679 700	AECL	WOL-SONG1,2	1&2 3&4

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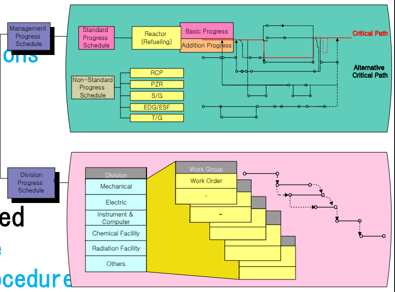
KEPRI 5

Methods and Results

3. Outage Schedule Structure Standardization

◆ The integrated outage schedule is built on details developed during the planning process.
- ✓work scope and priorities
- ✓estimated job duration
- ✓requirements and activity interrelations
- ✓resources and constraints



◆ To make more efficient use of resources the standardization of outage schedule structure developed
- ✓definition of each levels of schedule
- ✓terminology and schedule handling procedure
- ✓activity and milestone numbering system
- ✓standard template of each level schedule

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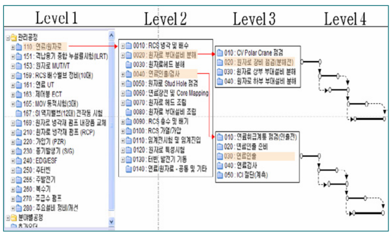
KEPRI 6

Methods and Results

4. Work Breakdown Structure(WBS) Standardization

◆ The standard template of outage preparation and management schedule was developed to perform the function of progress monitoring and control of each level of schedule.

➢The WBS of outage preparation schedule
- ✓ WBS description and structure , numbering system
- ✓ start and finishing date of individual work
- ✓ performance measuring methods
- ✓ progress input methods



➢The WBS of outage management schedule
- ✓ WBS work group code and structure
- ✓ WBS numbering system and description
- ✓ work relation code and relation type and value
- ✓ essential control point

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KEPRI 7

Methods and Results

5. Work Group and Task List Standardization

◆ The work group code which is installed in the present SAP system and task list of main and supplementary outage schedule is standardized by reactor types and power capacity.

◆ The work group is standardized by the WBS level 2 of work group numbering system ... of other department ... is dev... n.

Classification		Work Group No.	Description
Management Schedule	Standard Schedule	110 151	Fuel / Reactor ILRT
		167 169	SI Check V/V RCP Internal Replacement
	Non-standard Schedule	210 230	RCP Steam Generator
		260 270	Condencer Main Feedwater Pump
Division Schedule	Mechnical	310 320	Pump Vlave
	Electrical	510 520	Main Transformer Motor
	Others	951 994	NDT Test

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KEPRI 8

Methods and Results

6. Outage Planning and Performance Monitoring Standardization

➢ The design features of the outage management schedule

✓ give a weight factor to the each WBS

✓ activity progress rolling up process

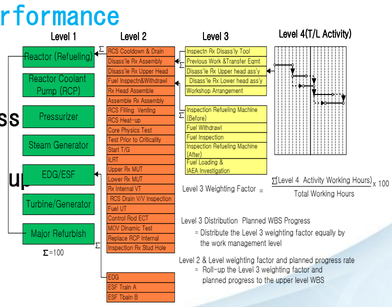
✓ total weight factor of WBS level 1 is 100.

➢ The basic principal of pre-outage performance monitoring design

✓ grouping of activities

✓ each activity' s performance progress calculation

✓ automatic activity progress rolling up process



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KEPRI 9

Methods and Results

7. Computerized Work Control System Standardization

◆ An efficient computerized outage management system is a key tool to optimize and standardize of the outage schedule control.

➢ the nuclear power plant outage
- ✓ Web based system
- ✓ connect to the ERP system
- ✓ upgraded system with updating the existing system's problems
- ✓ real time based management.
- ✓ cover from the outage preparation stage to the post outage review sta

KEPRI 10

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Conclusion

◆ Developed the effective and standard nuclear power plant outage management system(NPOMS) applicable to operating 20 plant in korea.

◆ The developed outage management system(NPOMS) is much more practical method to apply operating power plants then the one currently in use by the other utilities and industries.
- ✓ reduce the outage planning and scheduling time
- ✓ save labor cost
- ✓ optimize real- time outage process using computerized system.

◆ The NPOMS is composed 3 parts such as
- ✓ 1 standard pre-outage planning
- ✓ 5 standard outage schedule
- ✓ 1 standard post-outage evaluation program.

◆ We expect that the safety and operating reliability will be increased by using the developed real time management system.

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Thank You
KEPRI

V-A-4

Safety and Economic Results of RI-ISI Program at Units 3 & 4

Korea Electric Power Corporation
Korea Electric Power Research Institute

Bagsoon Chung

May 18-19, 2009 10th Korea-Japan PSA Workshop

Contents

Ⅰ. Introduction & Background

Risk-Informed Inservice Inspection

- An alternative to ASME Section XI periodic inspections that will focus on safety while reducing cost & radiation exposure. This approach is a blend of probabilistic & deterministic methods

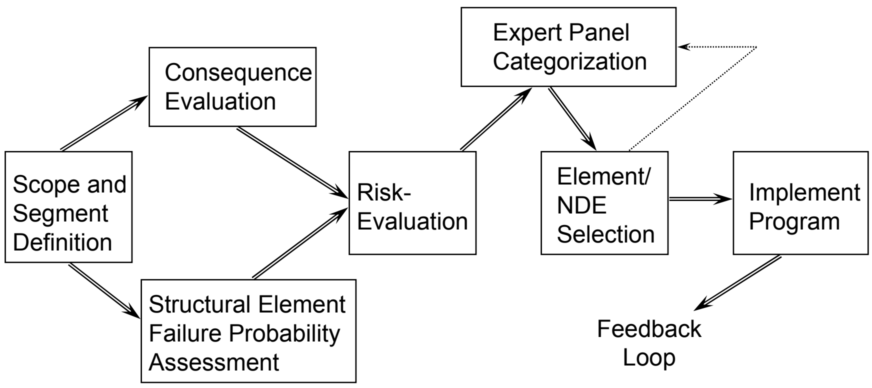
Ⅰ. Introduction & Background

- **Change in NRC Policy** - PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods & data & in a manner that complements the NRC's deterministic approach & supports the NRC's traditional defense-in-depth philosophy
- **KINS/GT-N24 -** KINS developed General Regulatory Guide for using probabilistic risk assessment in risk-informed deci - sions on plant-specific changes to the licensing basis (2007)

ASME Section XI Enhanced by Risk-Informed ISI

	ASME Section XI Process	Risk-Informed ISI
Consequence	Class 1, 2 and 3	Exercising of PSA model (CDF, LERF, others)
Failure Probability	High design stress and fatigue locations augmented by random selection	Exercising of SRRA model, including design, experience and operations - focused on areas of highest failure potential

Ⅱ. Summary of Risk-Informed ISI Process



Ⅲ. Exams Reduction Rate

Plant	ASME Class	Current NDE Exams	RI-ISI Exams	Reduction Rate (%)
Ulchin 3	Cl. 1	491	310	36.9
	Cl. 2	591	314	46.9
	Total	1,082	624	**42.3**
Ulchin 4	Cl. 1	491	321	34.6
	Cl. 2	591	314	46.9
	Total	1,082	635	**41.3**

IV. Change in Risk Calculations

Cases	Current NDE EXAMS	RI-ISI EXAMS	Reduction (%)	Change in Risk
CDF w/o	1.14E-07	8.90E-08	**21.9**	Risk Reduction
CDF with	6.83E-08	6.22E-08	**8.9**	
LERF w/o	2.02E-08	8.96E-09	**55.6**	
LERF with	9.12E-10	4.87E-10	**46.6**	

Change in Risk Calculations Method

- Impact on Risk Calculation Process
  - Identify segments addressed by current Section XI program
  - Identify segments addressed by risk-informed ISI program
  - For segments being inspected by NDE, use failure probability that credits ISI effectiveness (probability of detection and frequency)
  - Compare change in risk for current Section XI program and risk-informed ISI program by system and by total plant piping

    NOTE: Assumption is made that current location within the segment addresses the risk associated with that segment

V. Estimated Costs Saving

Description	Considerations	Saving per Inspection
Direct Costs		
Actual Inspection Costs	NDE (Including Scaffolding, Insulation Removal etc.) (42% Reduction)	**US$70,000**
ALARA Costs	Using $10,000/Rem Saved (3 Person-Rem/inspection Estimated)	**US$30,000**
Indirect Costs		
Administrative Costs	Paper work including work orders Surveillance and Clearance	Not Estimated
Outage Critical Path	Reduction of 1 days of Outage time anticipated as outages become shorter	Not Estimated
Analysis Costs	From flaw indication evaluations in low safety significant piping	Not Estimated
Total Estimated Costs		**US$100,000**

VI. Conclusions

- Implementation of Risk-Informed ISI methodology will yield costs saving while enhancing safety and reducing radiation exposure.

Session V-B

Seismic PSA

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Summary of Session V-B: Seismic PSA

Chair: Katsunori OGURA (JNES), In-Kil CHOI (KAERI)

1. "Study on Effects of Correlative Degree of Component Damages on Seismic PSA During Component Outage"

Cancelled

2. "Recent R&D Activity on Seismic PSA in KAERI" presented by Dr In-Kil Choi of KAERI

This paper presents resent research and development activity on seismic PSA in Korea Atomic Energy Research Institute. This five year project, from 2007 to 2011, consists of four main categories.

- Uncertainty reduction of probabilistic seismic hazard analysis
- Development of seismic fragility analysis methodology for degraded components
- Experimental studies on the fragility level of safety significant components
- Development of plant specific seismic risk quantification model

The detailed R&D activity related to the main topics and the interim results of this project were introduced.

3. "Radiological Consequence Analysis for Seismic Events in BWR Plants" presented by Ms. Kyoko FUNAYAMA of JNES

The purpose of this study is to examine the effect of evacuation on individual risk for a typical BWR plant, and identify dominant accident sequences and those individual risks. The conditional probability of individual risk was reduced about 1/10 - 3/5 by evacuation. For the realistic evaluation of individual risk by evacuation, the evacuation model should be realistic enough to reflect the possibility of evacuation under strong earthquake condition.

4. "A Study on the Uncertainty of Seismic Hazard in the PSHA for a Korean NPP Site" presented by Ms. Hyun Me Rhee of KAERI.

Korean peninsula is a low and intermediate seismicity area. The PSHA study has been performed using historical and instrumental earthquake records. But, there is no strong earthquake data. And the historical earthquake is recorded with a simple description in the old documents. So, the seismic hazard curves have a large uncertainty due the lack of strong motion data. The sensitivity analysis was performed for the input parameters. The final results show that the Gutenberg-Richter parameter which expresses the recurrence frequency was identified as a dominant input parameter to reduce the uncertainty of the seismic hazard of Korean NPP sites.

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Study on Effects of Correlative Degree of Component Damages on Seismic PSA During Component Outage

Y. Narumiya[*1] T. Ohya[*1]
Y. Katagiri[*2] T. Kuramoto[*2] K. Toyoshima[*2] T.Higashiyama[*2]

[*1] Kansai Electric Power Co., Inc.
[*2] Nuclear Engineering, Ltd. (NEL)

Contents

1. Background

Risk-informed Decision Making about Plant Safety Management Activities is under discussion.

For the purpose of those activities, All Risks from Internal and External Events that affect those activities should be evaluated.

Seismic Effects are particularly important in Japan

Effects of **Risk Increase attributable to an Earthquake during Component Outage** are evaluated in viewpoints below.

- **Correlative Degree of Component Damages** caused by Earthquake.
- **Difference of Component** which is set Out of Service.

2.1 Conditions

- Using the Event Tree Linking Approach
- Evaluation is performed by RISKMAN
- Evaluation Items are CDF and RAW

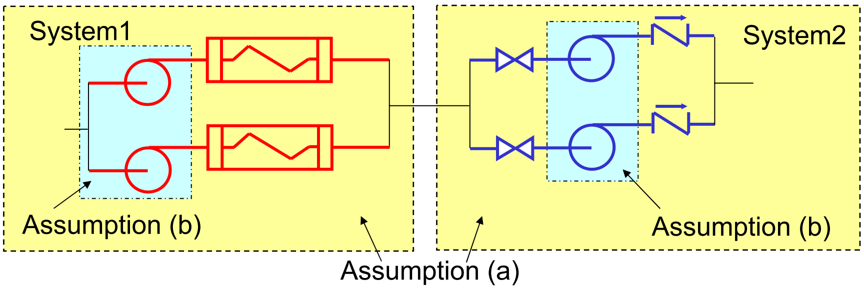
RAW's Definition in this study

$$\text{RAW}=\frac{\text{CDF with Component Outage (Train-A is set Out of Service)}}{\text{Baseline CDF}}$$

Baseline CDF＝CDF without Component Outage

2.2 Assumptions of Correlation of Component Damages

To identify the conservative condition, Different Correlative Degrees of Component Damages during Component Outage are considered.



(a) Between Different Systems

- Completely Independent correlation is assumed.

(b) Among Redundant Components in Same System

- Different correlative degrees of component damages are assumed. （Completely Dependent, Partially Dependent, Completely Independent）

2.3 Outline Of System Analysis Model for Seismic PSA

Initiator (Seismic Level)

SEISINI

- Seismic induced Initiating Events are considered, e.g. LLOCA.
- Split Fractions are created from Component Fragility.

SEISPRE

- Seismic Component Damages and Correlative degree of Component Damages are considered.
- Split Fractions are created from Component Fragility.
- Top Events are divided by Train.

SUPPORT

SEISEARLY

SEISLATE

- Based on Internal Event PSA Model.
- Linking to the Top Event of SEISPRE is modeled.
- Component Outage is considered.

End State [CDF]

2.4 Selection of Systems to be Evaluated

Viewpoints of Selection

- AOT-applicable Redundant Systems selected according to the Safety Preservation Rules.
- Systems showing a high RAW on Internal Events PSA.
- Systems for which effects caused by different functions (Electrical Support, Mechanical Support and Front Line Systems) can be identified separately.

Selected Systems

Electrical Support System	6.6kV AC Bus (AC BUS)
	Emergency Diesel Generator (DG)
Mechanical Support System	Sea Water System (SWS)
	CCW System (CCWS)
Front Line System	RHR System (RHRS)
	Auxiliary Feed Water System (AFWS)

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3.1 Effects on Core Damage Frequency

	Completely Dependent	Partially Dependent	Completely Independent
AC BUS	10.0	9.7	9.7
DG	6.2	5.9	5.9
SWS	3.1	2.7	2.6
CCWS	3.1	2.8	2.7
RHRS	1.6	1.1	1.0
AFWS*	1.7	1.2	1.1
Baseline CDF	1.6	1.1	1.0

CDFs are normalized by Baseline CDF of Completely Independent Case.
*Turbine Driven Auxiliary Feed Water Pump is set Out of Service.

- CDF for Completely Dependent Correlation is largest followed by Partial Dependence and Complete Independence in descending order.
- Frequencies of System Damage increase as the Correlative Degree of Component Damages becomes higher.

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3.2 Effects on RAW

	Seismic PSA			Internal Events PSA
	Completely Dependent	Partially Dependent	Completely Independent	
AC BUS	6.3	8.8	9.7	123.9
DG	3.9	5.3	5.9	2.4
SWS	1.9	2.4	2.6	2.7
CCWS	2.0	2.5	2.7	1.1
RHRS	1.0	1.0	1.0	3.4
AFWS*	1.1	1.1	1.1	2.4

*Turbine Driven Auxiliary Feed Water Pump is set Out of Service

- RAW is largest for Completely Independent Correlation, followed by Partially Dependence and Complete Dependence in descending order.
- RAW Dependence on Component Outage increases as Correlative Degree decreases.

3.3 Effects on Component Out of Service, etc

- Both CDF and RAW orders according to the Correlative Degree of Component Damages **do not change depending on the Component** set Out of Service.
- CDF and RAW become **higher as effects for other systems due to Component Outage become larger**.
- RAW for Seismic PSA is **less than Three times** that for Internal Events PSA at the largest.

4. Conclusion

- Conservative condition of Correlative Degree of Component Damages

  For CDF : ***Completely Dependent***
  For RAW : ***Completely Independent***

- Effects on Component Out of Service

  CDF and RAW become higher as effects for other systems due to Component Outage become larger.

- Relation Between Seismic PSA and Internal PSA

  RAW for Seismic PSA is not excessively higher than that for Internal Events PSA.

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V-B-2

Recent R&D Activities on Seismic PSA in KAERI

May 19, 2009

In-Kil Choi

Korea Atomic Energy Research Institute

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Introduction

- Seismic safety of NPP
  - Important after 2007 Chuetsu event
  - Recent large earthquake event causes great damage to structures and a loss of lives
- Degradation of SSCs
  - Aging related degradation of SSCs can cause a reduction of seismic safety margin
    - Individual components
    - Plant level seismic safety – CDF
- Systematic Aging Management
  - Realistic seismic risk evaluation based on the operating conditions
  - Risk based aging management

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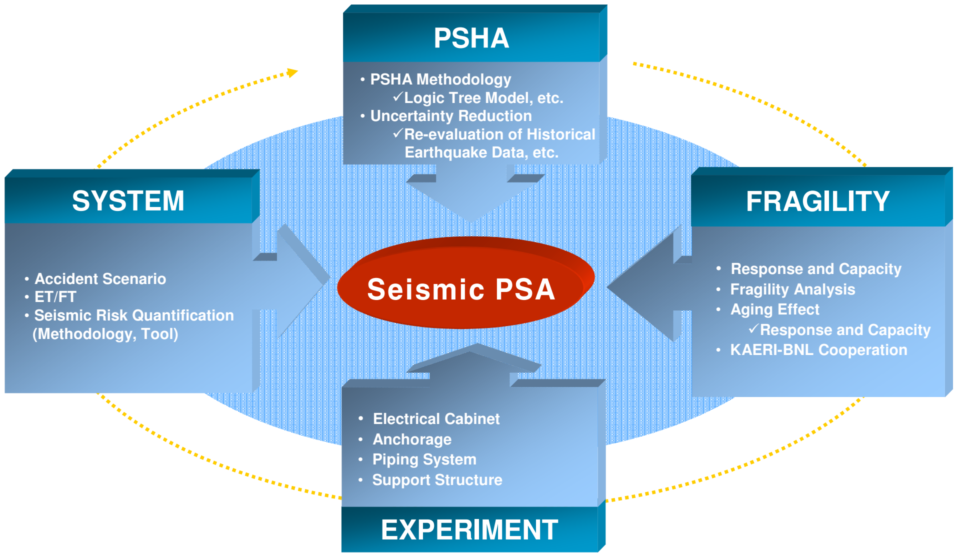
SPRA Research Plan

- 5 year project (2007 – 2011)



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International Collaboration

■ KAERI-BNL Cooperation

FY	KAERI	BNL
2007	• Collection and database of degradation occurrences in Korean NPPs • Evaluation of … n NPPs	• Collection and review of degradation occurrences in US NPPs • Evaluation of a … Ps • Evaluation of a … fragility
2008	• Analysis of de … ean NPPs • Evaluation of … of construction r … Ps • Modeling of lo … of construction materials in K …	• Analysis of deg … NPPs • Evaluation of l … of construction materials for U … • Modeling of lo … f construction materials in US …
2009	• Development … gradation models for sei … omponents	• Development … luation methodology f … tion • Development … capability evaluation me … structures and components
2010	• Evaluation of … criteria for degraded str … • Evaluation … structures an …	• Developm … criteria for structures an …
2011	• Fragility modeling … tructures and components • Case study: Application of seismic capability evaluation methodology to a Korean operating NPP	• Development of seismic fragility evaluation methodology considering age-related degradation

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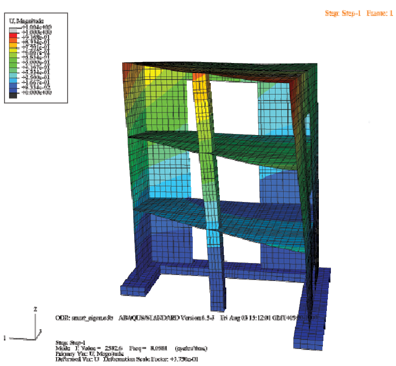
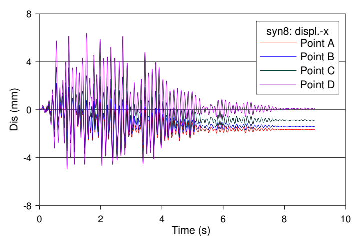
International Collaboration

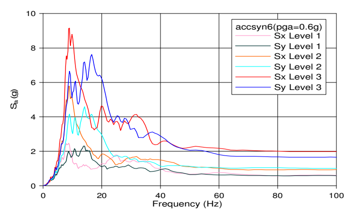
■ SMART 2008



Reinforced Concrete Structure



Acceleration Response



Floor Response Spectra

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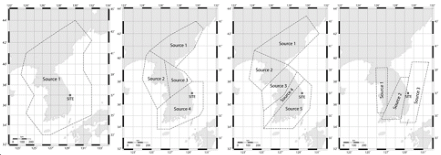
Interim Research Results

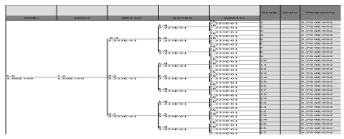
- **PSHA Code Development**
  - EQHAZ + Attenuation Eq.
    - Foreign Country : 4, Korean : 4
  - CONPAS : Pre-processor for LT
    - Automatic Input Generation
  - Post-processor for Statistical Analysis
    - Automatic Analysis for PSHA Results



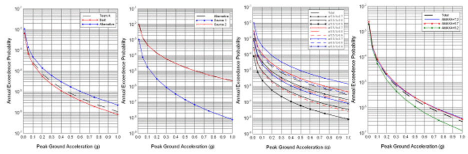
Team A Team B Team C Team D

Seismic Source Model from Expert Teams

- **Sensitivity Analysis for Input Parameter**
  - Organize Expert Panel
    - Team for Source Model : 4 Team
    - Attenuation Equation : 2 Expert
  - Sensitivity Analysis for 5 Input Parameter
    - Richter-a, b, M_{max}, M_{min}, Focal Depth
  - Uncertainty Analysis
    - Uncertainty for Individual Team
    - Total Uncertainty



Example Logic Tree



Source Model Richter Value Control Earthquake

Sensitivity Analysis Results

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Interim Research Result

- Component degradation in Korean NPP
  - Select oldest NPP
  - Plant walkdown for investigation
  - Total of 530 components
  - Develop database system for managing collected data
    - Component specification
    - Location (Building, Floor)
    - Anchorage detail
    - Degradation occurrence

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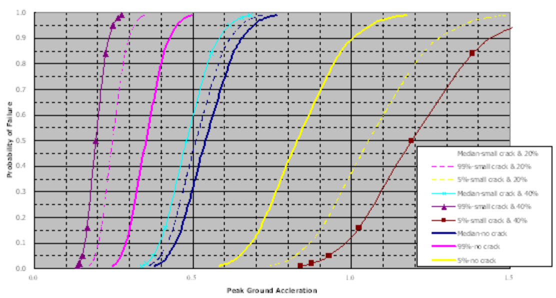
Interim Research Results

- Degradation Phenomena in Korean NPP
  - Mainly Crack and Corrosion

Interim Research Result

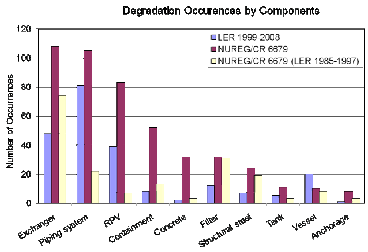
- Fragility Analysis of Expansion Anchor
  - Electrical panel
  - With hairline/small crack in anchorage

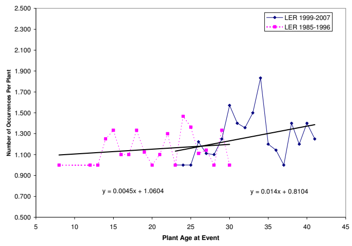


	No Crack	Hairline Crack [<0.25mm]	Small Crack [0.25-0.51mm]
HCLPF	0.27g	0.17g	0.15g
Reduction[%]		37	44

Interim Research Results

- Component Degradation Occurrence in US NPP
  - Under KAERI-BNL collaboration
    - Identify a list of SSCs important for plant seismic risk
    - Collect and review degradation occurrences in US nuclear power plants
    - Assess the trending of degradation of SSCs





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Aging-related Degradation

- Trending analysis results
  - Degradation occurrences slightly grow as the plant age
  - Absolute magnitude of degradation occurrences is relatively low
  - Most degradation occurrences are piping system, exchanger, and RPV
  - Most vulnerable system is RCS
  - Most significant aging effect is cracking
  - Most common aging mechanism is SCC

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Interim Research Results

- Component Fragility Test Plan
  - Electrical Cabinet
  - Anchorage
  - Piping System
  - Support Structure

- Electrical Cabinet Test(2008)
  - 480V MCC for Shin Uljin NPP
  - 3 Kinds of Input Motion
  - Under the 480V Electricity Supplied
  - Measure Structure Response and Relay Signal

Input Motion	Specification
Regulatory Guide 1.60	Design Earthquake (Artificial Motion)
FRS	FRS of Aux. Bldg.
UHS	(Artificial Motion)

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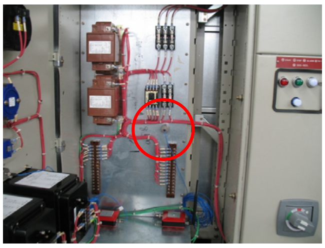
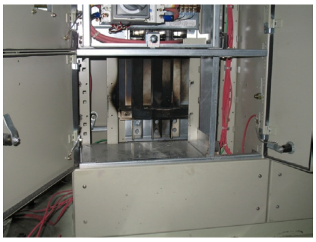
Interim Research Results

- Component Fragility Test Results
  - Identify Failure Mode and Criteria
    - Structural Failure
    - Functional Failure
  - Identify New Failure Mode
  - Identify the In-Cabinet Response
  - Response for Different Input Motion





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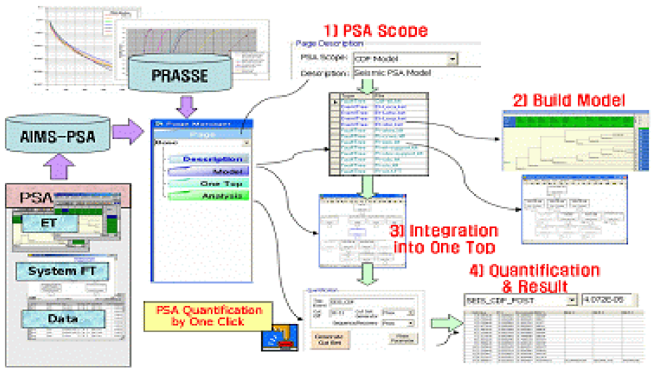
Interim Research Results

- SPRA Model Development
  - AIMS Based One-Top Model
    - Automatic Seismic Risk Quantification



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Interim Research Results

- Necessity of Program Development
  - Quantification Code for Past SPRA Study in Korea
    - EQESRA
    - SEISMIC
  - Need New Program
  - Status of Existing Code
    - Have Only Execution File (DOS Version)
    - Need 2 Steps for Boolean Equations
    - Condensation Method for Uncertainty Analysis
  - SEISMIC Code
    - Need Modification of the Subroutine for Boolean Equation
    - Monte Carlo Simulation Method for Uncertainty Analysis
    - Linear Interpolation for Hazard Curves
  - RISKMAN Code
    - Large ET/Small FT

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Interim Research Results

- **Development of Risk Quantification Program**
  - Language : Fortran 90
  - Uncertainty Analysis : LHS and MCS
  - Specifications
    - Multiple Event Analysis Capability (System)
    - Easy Input for Boolean Equation
    - Logarithmic Interpolation for Hazard Curves
    - Graphic User Interface

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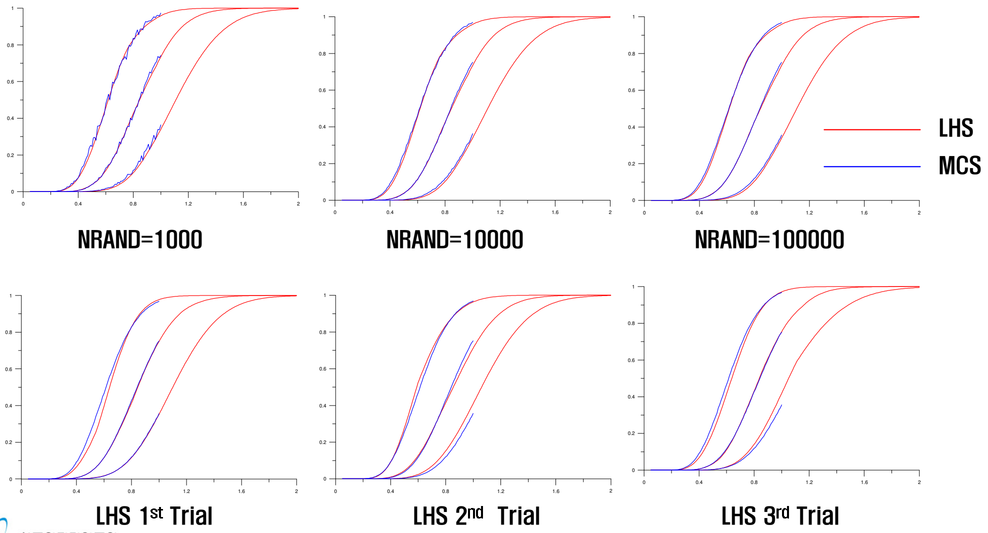
Interim Research Results

- **Comparison MCS vs. LHS Results**



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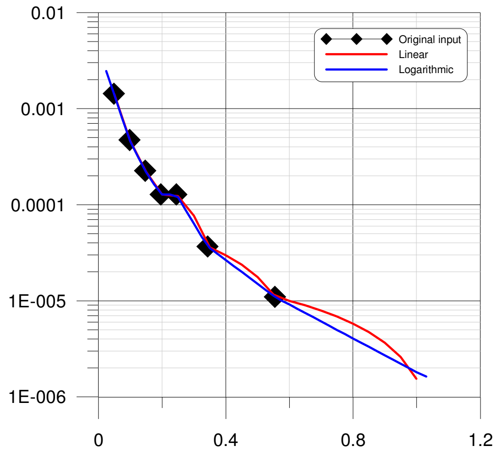
Interim Research Results

■ Hazard Curve Interpolation



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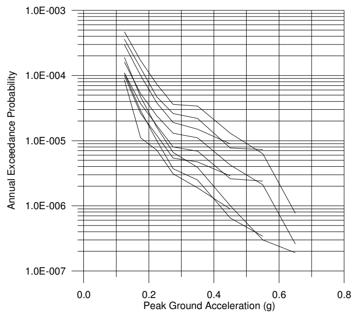
Interim Research Results

■ Verification of Code

□ Simple Problem : A∪B∪C

Component	Median Capacity (g)	β_R	β_U
A	0.63	0.39	0.00
B	0.73	0.41	0.00
C	0.73	0.43	0.00

Code	Plant Damage Frequency(/yr)
PRASSE(LHS)	4.57E-06
PRASSE(MCS)	4.57E-06
EAESRA	4.55E-06



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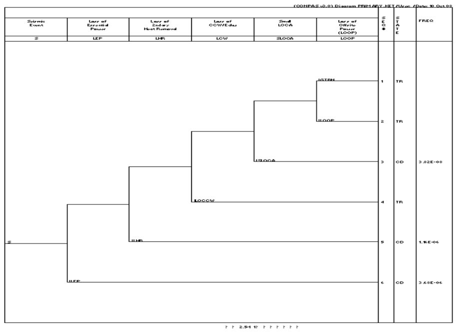
Interim Research Results

- Degradation Effect on Seismic Risk
  - 6 seismic induced initiating events
    - Small LOCA (SLOCA)
    - Loss of essential Power (LEP)
    - Loss of secondary heat removal (LHR)
    - Loss of CCW (LOCCW)
    - Loss of offsite power (LOOP)
    - General transient



Example Logic Tree

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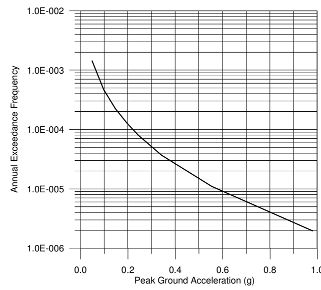
Interim Research Results

- Component Degradation
  - 7 component
  - 10% to 50% reduction of median acceleration capacity (with same logarithmic standard deviation for uncertainty and randomness)
  - One seismic hazard curve

Component	Initiating Event
Instrumentation Tube	SLOCA
CS Tank	LHR
Battery Charger	LEP
Battery Rack	LEP
Diesel Generator	LEP
ECW Compression Tank	LOCCW
LOOP	LOOP



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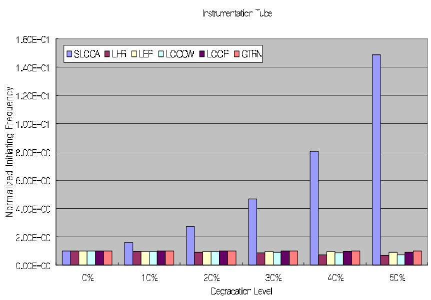
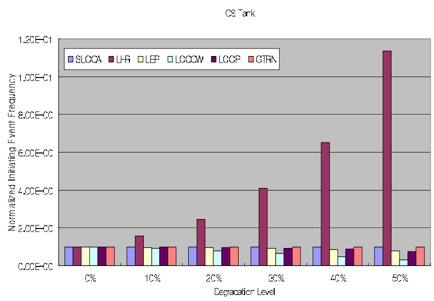
Interim Research Results

- Degradation Effect on IEF
  - Instrumentation Tube (max. 15)
  - Condensate Storage Tank (max. 11)





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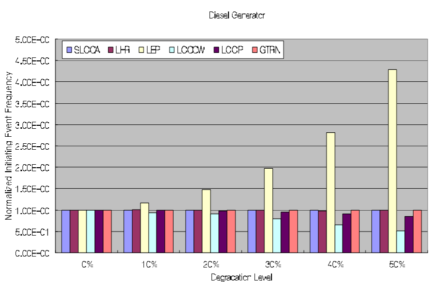
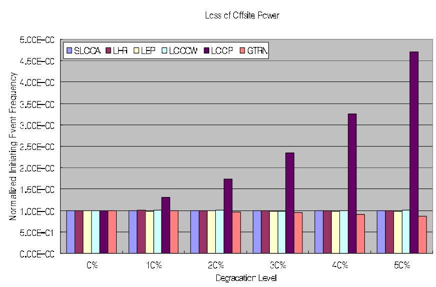
Interim Research Results

- Degradation Effect on IEF
  - Diesel Generator (max. 4.3)
  - Loss of Offsite Power (max. 4.7)





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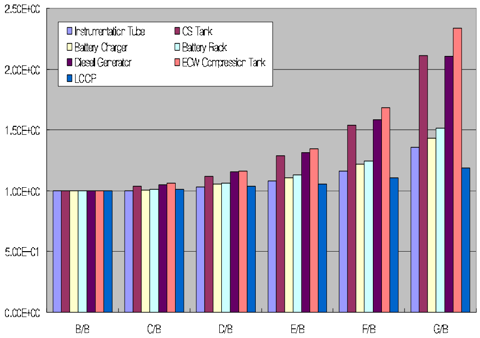
Interim Research Results

- Degradation Effect on CDF
  - CDF is affected by the failure probability of mitigating system
  - Biggest seismic risk contributor due to the degradation
    - ECW Compression Tank
    - Condensate Storage Tank
    - Diesel Generator



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Interim Research Results

- Degradation Effect Analysis Using SPRA
  - Identification of significant seismic risk contributor by aging-related degradation is possible from the Seismic PSA study
  - Possible to identify the safety significant components to secure a long term seismic safety of a plant

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Conclusion

- Introduce Research Activity on SPRA in KAERI
  - Hazard, Fragility and Risk Quantification
  - Develop Plant Specific SPRA Model
  - Development of SPRA Tools
  - Realistic Seismic Risk Evaluation
- International Cooperation

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Thank You!!!

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JNES

Radiological Consequence Analysis for Seismic Events in BWR plants

K-J PSA, May 18-20, 2009

Kyoko FUNAYAMA, Toru TACHINO and Mitsuhiro KAJIMOTO

Severe Accident Evaluation Group,
Nuclear Safety Analysis and Evaluation Office,
Nuclear Energy System Safety Division,
Japan Nuclear Energy Safety Organization (JNES)

The 10th Workshop on K-J PSA at Jeju, Korea , May 18-20, 2009

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Contents

- Background and Objectives
- Analytical Conditions
- Calculated Results
  - Reduction of Individual Risk by Evacuations
  - Risk Dominant Sequences in Individual Risks
- Conclusions

2

Background and Objectives

- Background
  - Japan Nuclear Energy Safety Organization (JNES) in Japan has been promoting the Level 3 PSA program.
  - In the program, the MACCS2 code has been extensively applied to analyze radiological consequences for typical BWR and PWR plants in Japan.
  - In addition, effects of accident management (AM) on radiological consequence for a typical BWR plants based on Level 2 PSA results have been discussing at JNES.
- Objectives
  - To examine effect of evacuation on individual risks for typical BWR plants in Japan, and
  - Identify dominant accident sequences and these Individual risks

3

Analytical Conditions

Level 2 PSA

- Frequencies of accident sequences that lead to large release
- Source terms of accident sequences

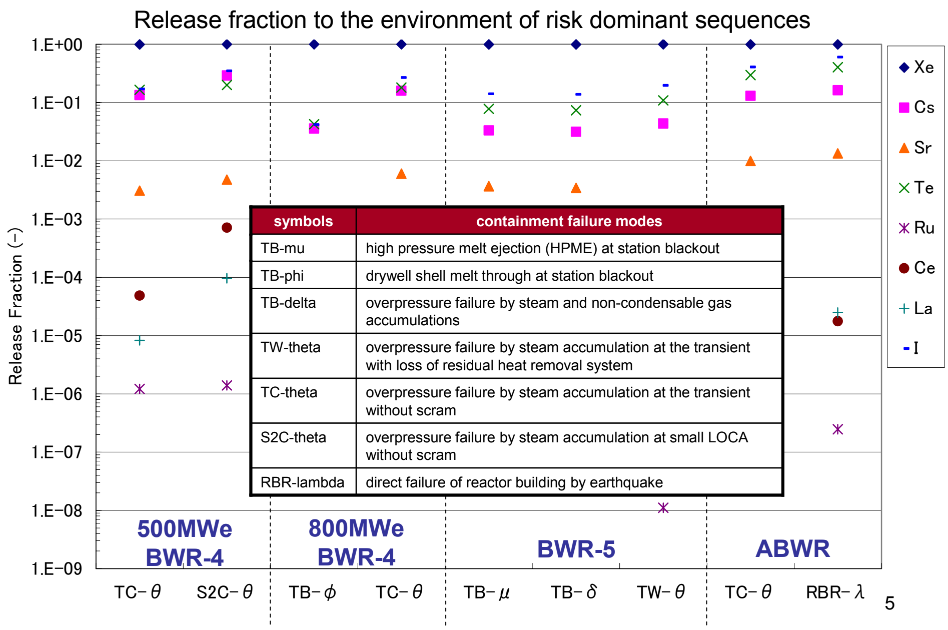
Level 3 PSA

- Site conditions
- Evacuation
  effect of evacuation
- Individual risk
  dominant accident sequences

- Reference Plants
  a typical BWR-4 (540MWe) with a Mark-I Containment
  a typical BWR-4 (840MWe) with a Mark-I Containment
  a typical BWR-5 (1,100MWe) with a modified Mark-II Containment
  a typical ABWR (1,380MWe) with a RCCV Containment

4

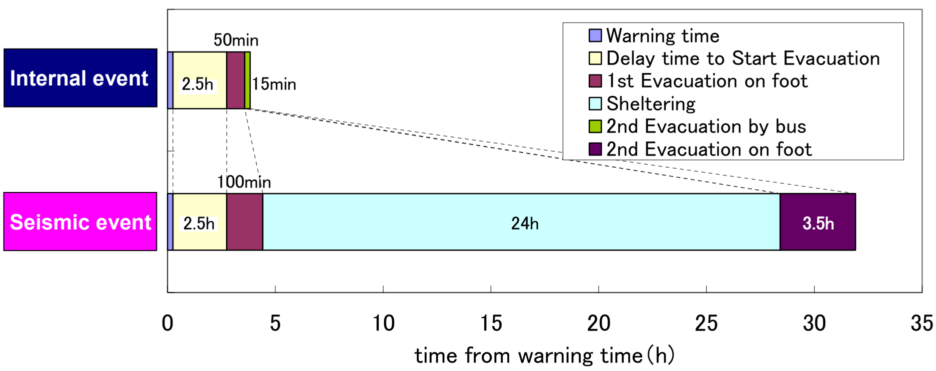
Level 2 PSA Results: Source Terms of accident sequences



symbols	containment failure modes
TB-mu	high pressure melt ejection (HPME) at station blackout
TB-phi	drywell shell melt through at station blackout
TB-delta	overpressure failure by steam and non-condensable gas accumulations
TW-theta	overpressure failure by steam accumulation at the transient with loss of residual heat removal system
TC-theta	overpressure failure by steam accumulation at the transient without scram
S2C-theta	overpressure failure by steam accumulation at small LOCA without scram
RBR-lambda	direct failure of reactor building by earthquake

5

Evacuation models



Internal event	Evacuation model of internal event is based on information of nuclear emergency response drill. - Evacuation velocity (1st; 4km/h, on foot / 2nd; 35km/h, by bus) - No sheltering
Seismic event	Evacuation model of seismic event is based on condition of evacuation at the earthquake. - Evacuation velocity (2km/h, on foot) - Sheltering for 24 hours at the evacuation places in local area.

Evacuation models (continued)

- **Emergency Planning Zone**

  An emergency planning zone (EPZ) that is provided by the disaster prevention guideline is 8 – 10 km radius from the reactor site, and a 10km radius is assumed in the present study.

- **Warning Time**

  A declaration of "nuclear emergency" that is provided by the 15th article of the nuclear disaster special measures law is assumed to be issued at the event of the earthquake's occurrence.

- **Time Delay to Start Evacuation**

  The time delay to start evacuation is determined by the time to receive warning, the time before leaving the office, the time to return home and the time before leaving home.
  It is based on the nuclear disaster drill in Japan.

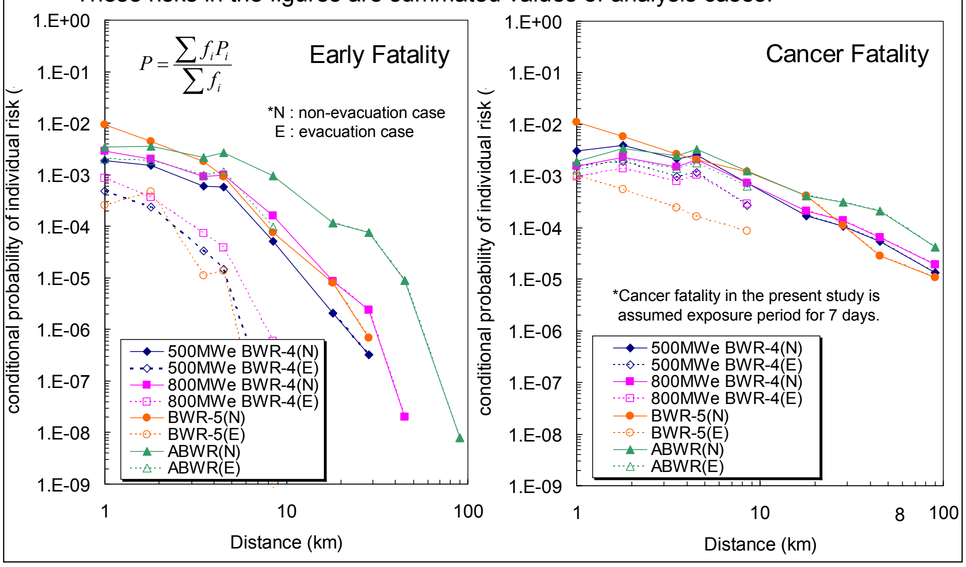
- **Release Rate and Duration of Radionuclides**

  The release rate and duration of radionuclides of Level 2 PSA results are simulated with a multiple plume model of MACCS2, but the tailing parts are neglected.

7

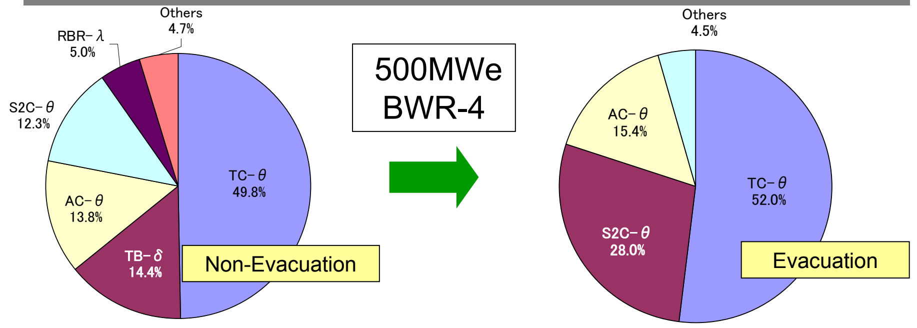
Reduction of Individual Risk by Evacuation

Individual risk in the present study is defined by conditional probability with population weighted individual risk.
These risks in the figures are summated values of analysis cases.



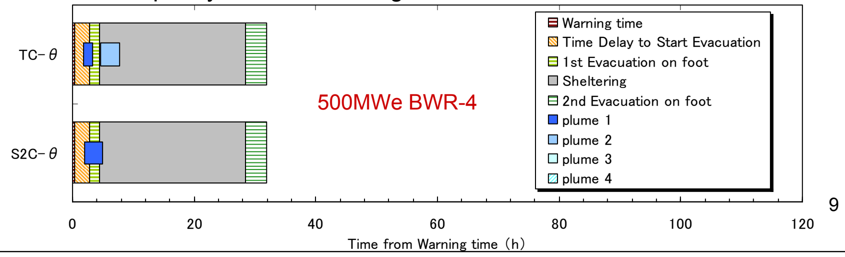
8

Risk Dominant Sequences in Individual Risks (Early Fatality)



● Evacuation and timing of radionuclides release to the environment

The risk reduction effect strongly depends on whether residents could evacuate quickly before the timing of radionuclides release or not.



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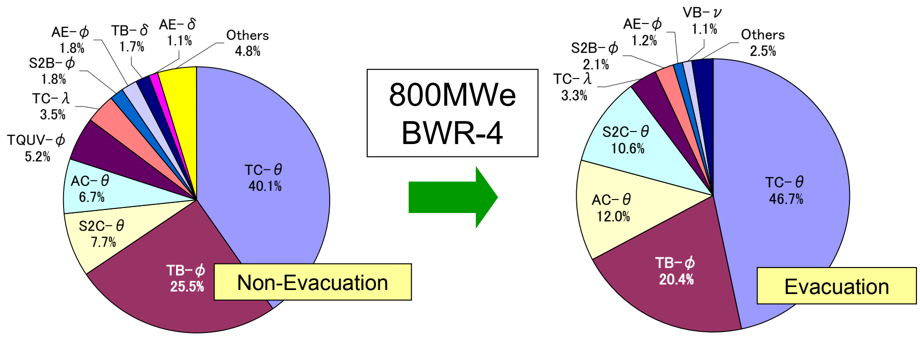
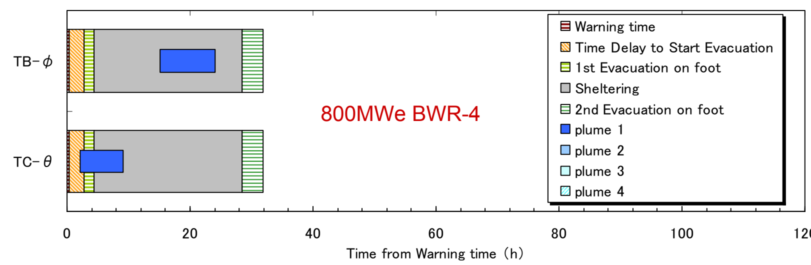


● Evacuation and timing of radionuclides release to the environment



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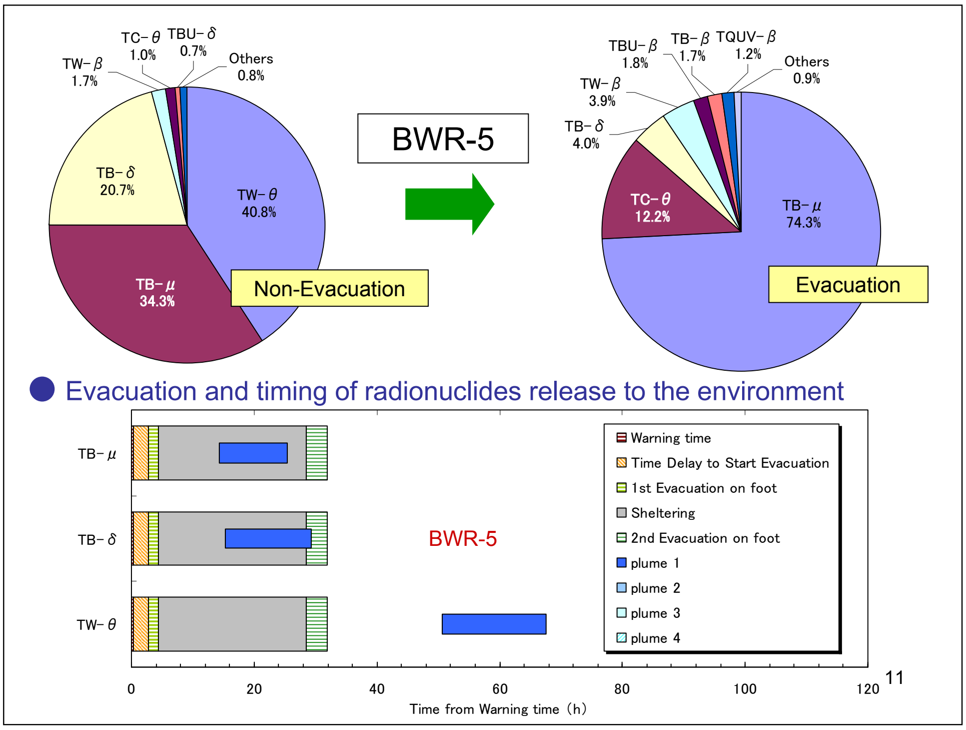
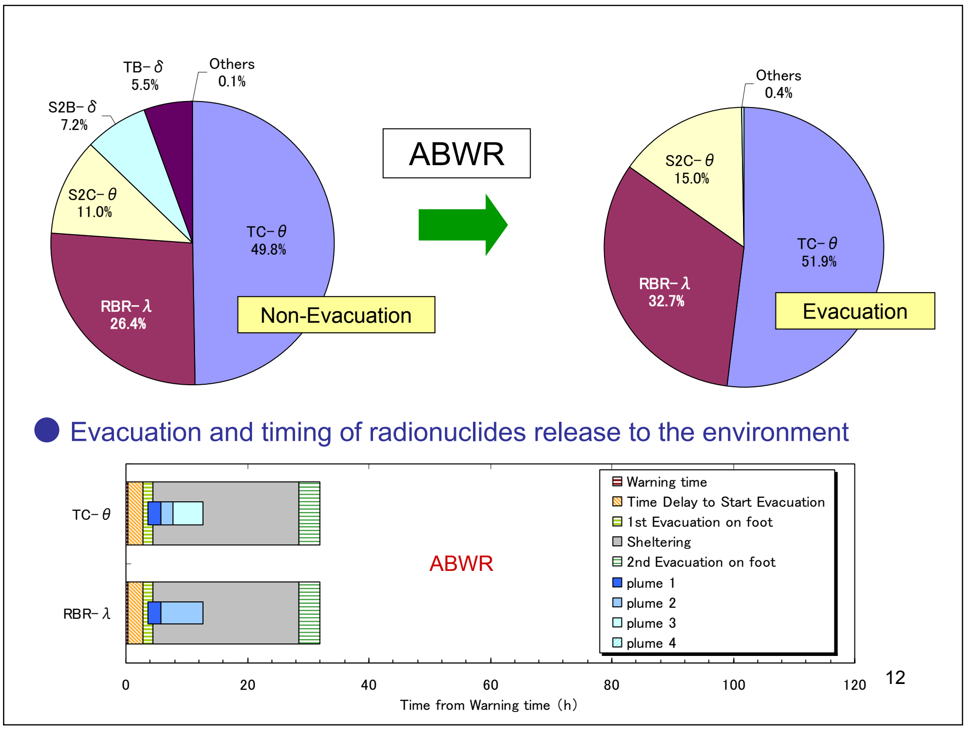


BWR-5
TC- θ 1.0%
TBU- δ 0.7%
TW- β 1.7%
Others 0.8%
TB- δ 20.7%
TW- θ 40.8%
TB- μ 34.3%
Non-Evacuation
TBU- β 1.8%
TB- β 1.7%
TQUV- β 1.2%
Others 0.9%
TW- β 3.9%
TB- δ 4.0%
TC- θ 12.2%
TB- μ 74.3%
Evacuation
Evacuation and timing of radionuclides release to the environment
TB- μ
TB- δ
TW- θ
BWR-5
Warning time
Time Delay to Start Evacuation
1st Evacuation on foot
Sheltering
2nd Evacuation on foot
plume 1
plume 2
plume 3
plume 4
0
20
40
60
80
100
120
Time from Warning time (h)
11



ABWR
TB- δ 5.5%
Others 0.1%
S2B- δ 7.2%
S2C- θ 11.0%
TC- θ 49.8%
RBR- λ 26.4%
Non-Evacuation
Others 0.4%
S2C- θ 15.0%
TC- θ 51.9%
RBR- λ 32.7%
Evacuation
Evacuation and timing of radionuclides release to the environment
TC- θ
RBR- λ
ABWR
Warning time
Time Delay to Start Evacuation
1st Evacuation on foot
Sheltering
2nd Evacuation on foot
plume 1
plume 2
plume 3
plume 4
0
20
40
60
80
100
120
Time from Warning time (h)
12

Contribution of Containment Failure modes to Individual Risk

	Frequency				Timing radionuclides release to the environment			
	500MWe BWR-4	800MWe BWR-4	BWR-5	ABWR	500MWe BWR-4	800MWe BWR-4	BWR-5	ABWR
Alpha		Small	Small			Early	Early	
Beta	Large	Large	Medium		Early	Early	Early	
Lambda	Large	Large	Small	Large	Early	Early	Early	Early
Mu		Large	Large			Early	Early	
Sigma		Medium	Medium			Early	Early	
Phi		Large				Early		
Delta	Large	Large	Large	Large	Medium	Medium	Medium	Medium
Theta-TW	Large	Large	Large		Late	Late	Late	
Theta-TC	Large	Large	Medium	Large	Early	Early	Early	Early
Nu		Large	Small			Early	Early	

: Dominant mode of Individual Risk (Early Fatality) at 1km from the site in non-evacuation case

: Dominant mode of Individual Risk (Early Fatality) at 1km from the site in evacuation case

13

Conclusions

Effect of evacuation

The conditional probability of individual risks of early fatalities and cancer fatalities became level of 10^{-3} in the case of non-evacuation in each plant. Considering evacuations, the conditional probability of individual risks in each plant reduced about 1/10 - 3/5 than those of the non-evacuation cases.

[Problem] *Can we evacuate under the condition of large earthquake?*
(is this unrealistic?)

It is necessary to examine the evacuation model that is considered site specific condition and preparation of human resource system.

	Early Fatality (0.8-2.0km) (dominant sequence)	Cancer Fatality (0.8-5.0km) (dominant sequence)
500MWe BWR-3	N* : 1.8E-3 (TC-theta) E* : 3.6E-4 (TC-theta, S2C-theta) (1/5)	N : 2.9E-3 (TC-theta) E : 1.4E-3 (TC-theta) (1/2)
800MWe BWR-4	N : 2.2E-3 (TC-theta, TB-phi) E : 4.5E-4 (TC-theta, TB-phi) (1/5)	N : 2.0E-3 (TC-theta) E : 1.1E-3 (TC-theta) (3/5)
BWR-5	N : 5.8E-3(TW-theta, TB-mu) E : 4.6E-4(TB-mu) (1/10)	N : 3.4E-3(TW-theta) E : 3.2E-4(TB-mu, TB-delta) (1/10)
ABWR	N : 3.9E-3 (TC-theta, RBR-lambda) E : 3.0E-3 (TC-theta, RBR-lambda) (3/5)	N : 2.2E-3 (TC-theta) E : 1.8E-3 (TC-theta, RBR-lambda) (3/5)

*N : non-evacuation case　E : evacuation case

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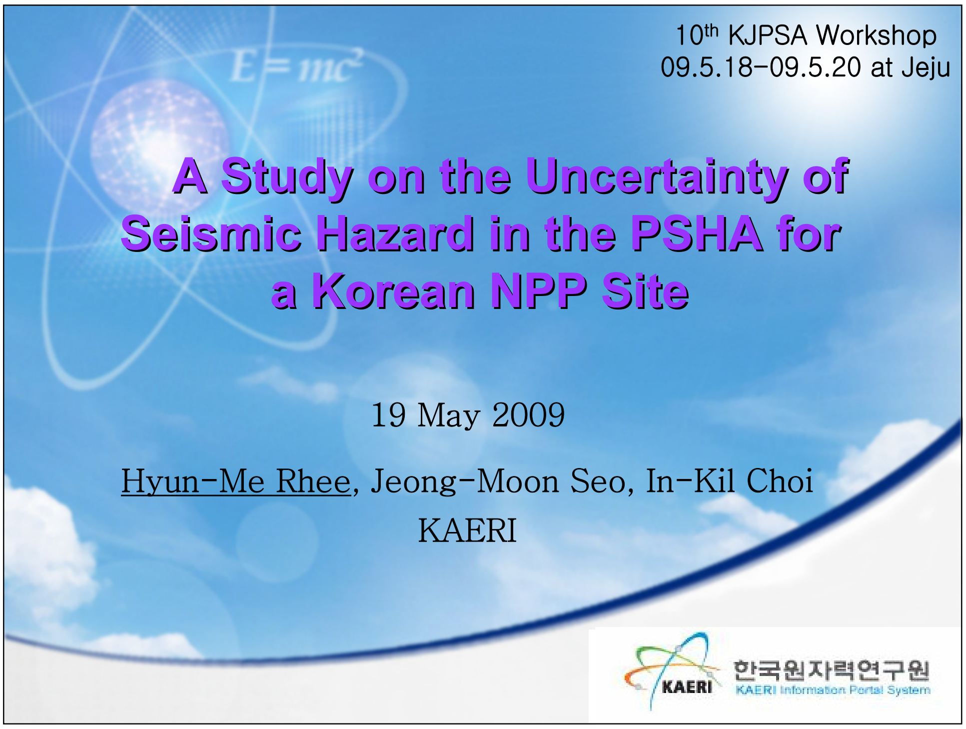
V-B-4


10th KJPSA Workshop
09.5.18-09.5.20 at Jeju
A Study on the Uncertainty of Seismic Hazard in the PSHA for a Korean NPP Site
19 May 2009
Hyun-Me Rhee, Jeong-Moon Seo, In-Kil Choi
KAERI
한국원자력연구원
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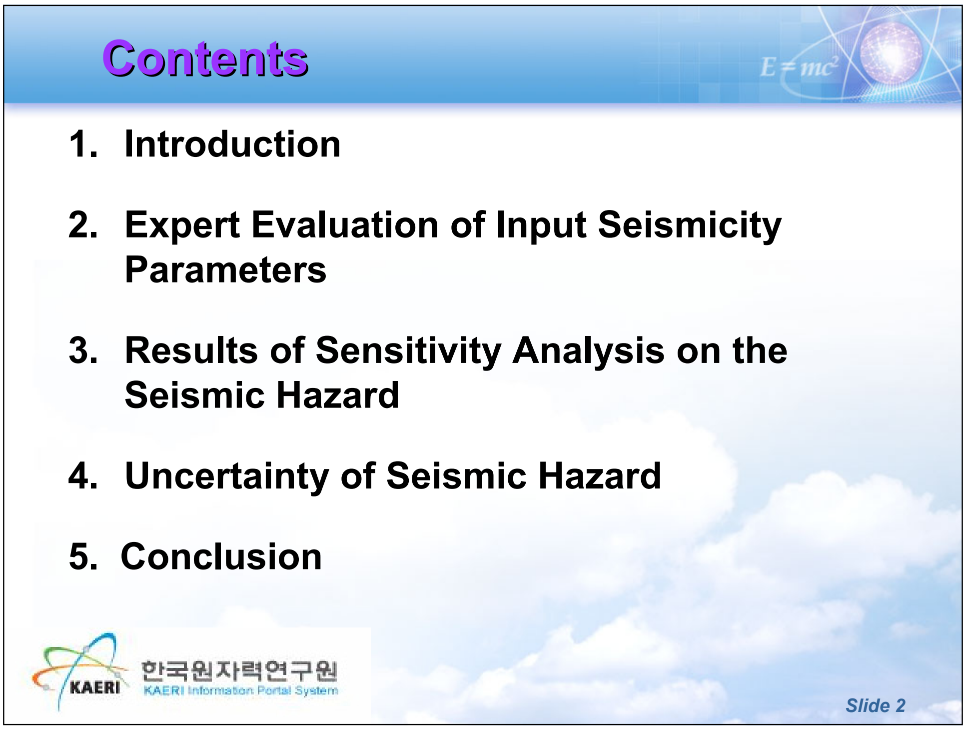
Contents
1. Introduction
2. Expert Evaluation of Input Seismicity Parameters
3. Results of Sensitivity Analysis on the Seismic Hazard
4. Uncertainty of Seismic Hazard
5. Conclusion
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Slide 2

1. Introduction

- More than ten PSHAs have been performed since 1986 for nuclear power plant sites in Korea.
- Although abundant information have been accumulated since 1990s in seismology and earthquake engineering, there still exist wide gaps in the input parameters among experts.
- The attenuation equation was known as the most uncertain parameter in the PSHA for Korean NPP sites.
- To identify the area of further improvement for reducing uncertainty, we performed this study by selecting Shinwuljin 1&2 site as a reference site.

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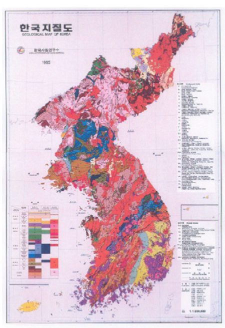
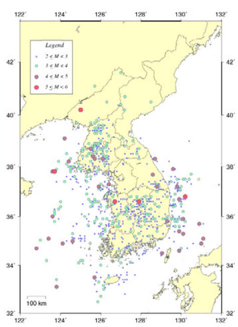
Slide 3

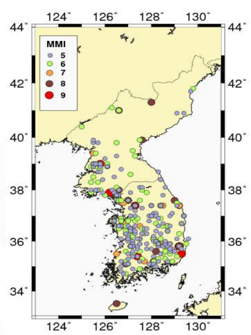
2. Expert Evaluation of Input Seismicity Parameters



Geologic map of Korea (1995, KIGAM)



Epicenter of the instrumental eq.(1978-2008, KMA)



Epicenter of the historical eq.(2-1904, Kim et al.)

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Slide 4

2. Expert Evaluation of Input Seismicity Parameters

E = mc²

❖ Expert Panels

- ❖ 4 teams for seismicity evaluation, each team consisted of 1 seismologist and 1 geologist
- ❖ 2 experts for attenuation equation evaluation

❖ Seismotectonic Structures

- ❖ Agreement on the area sources
- ❖ Significant differences among experts

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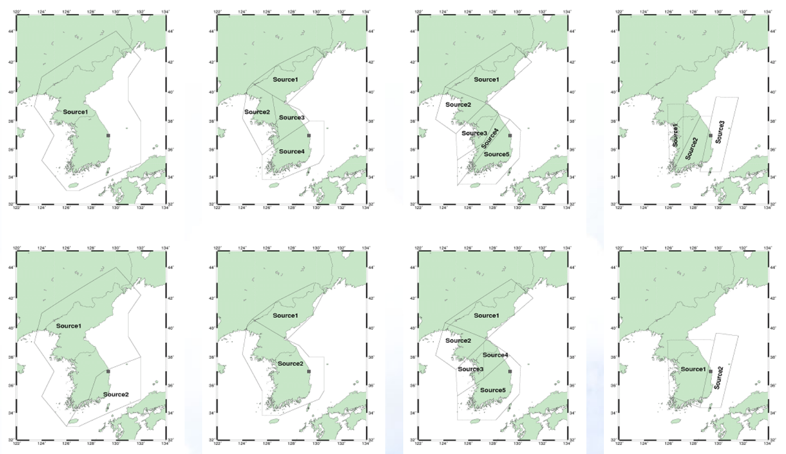
Slide 5

2. Expert Evaluation of Input Seismicity Parameters

E = mc²

❖Seismic Source Maps, best and alternative estimate



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Slide 6

2. Expert Evaluation of Input Seismicity Parameters

❖ Input by Experts

Gutenberg-Richter Parameter Values

Minimum: a=4.32, b=0.43
Maximum: a=6.25, b=0.99

Maximum Magnitude

6.2<M<7.2

Focal Depth
3-15 km

❖ Gutenberg-Richter Parameter Values of Previous Studies

$$\log_{10} N = a - bM$$

	a-value	b-value
Lee, K	6.09	0.71
Lee and Jung		0.8
Noh, M-H	5.66	1.11
KEPRI		0.61-0.64 0.98 0.89-0.92
Seo, J-M	5.44	0.84
(China)	4.77	1.01

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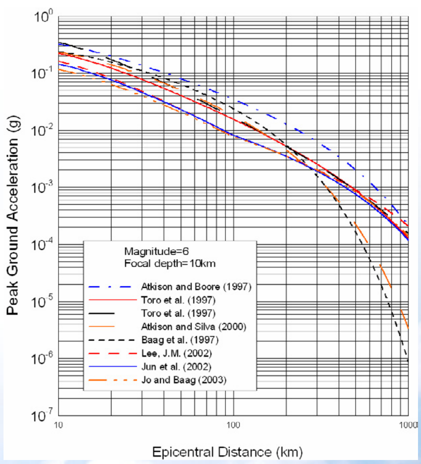
2. Expert Evaluation of Input Seismicity Parameters

❖ Attenuation Equation

Equation	expert 1	expert 2
Atkinson and Boore (1997)		10%
Midcontinent of Toro et al. (1997)		10%
Gulf Coast plain of Toro et al. (1997)		10%
Atkinson and Silva (2000)	30%	10%
Baag (1997)		15%
Lee (2002, KINS)		15%
Junn et al. (2002)		15%
Jo, N-D & Baag, C-E (2003)	70%	15%



Comparison of Att. Eqs. (M=6, D=10 km)

Slide 8

3. Results of Sensitivity Analysis

❖Range of Input Values (best estimate)

parameter	value
a-value	5.5±0.5
b-value	0.8±0.1
M_{MAX}	6.7±0.5
Focal depth	10±5

❖Attenuation Equations

Table (slide 8)

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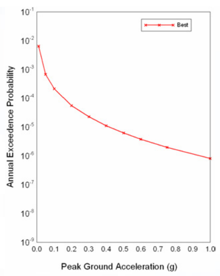
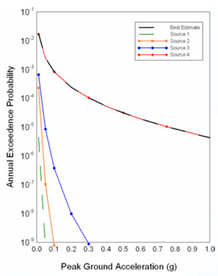
Slide 9

3. Results of Sensitivity Analysis

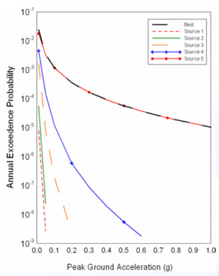
❖Sensitivity of Seismic Source Map (best estimate)



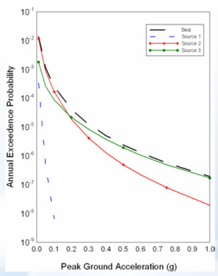
Team A



Team B



Team C



Team D

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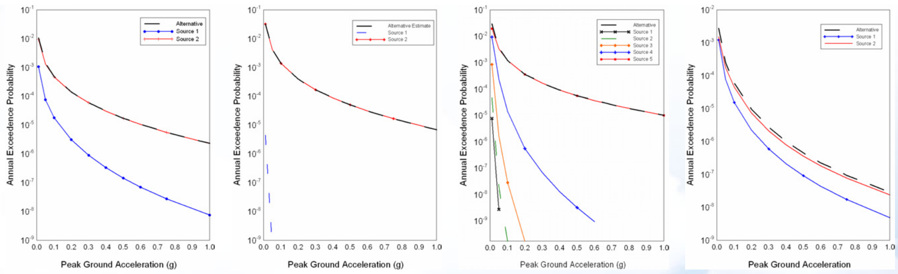
3. Results of Sensitivity Analysis

❖Sensitivity of Seismic Source Map (alternative)



Team A　　Team B　　Team C　　Team D

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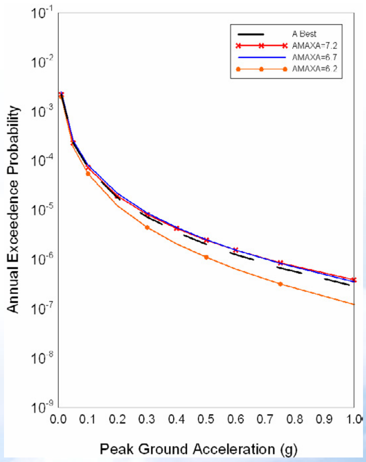
4. Results of Sensitivity Analysis

❖Sensitivity of Maximum Magnitude

- ❖ **Hazard for lager maximum magnitude (M=7.2) is much less sensitive than smaller maximum magnitude (M=6.2)**
- ❖ **Hazard for △M=1.0 shows maximum of about 1/2 order deviation at 1.0g**



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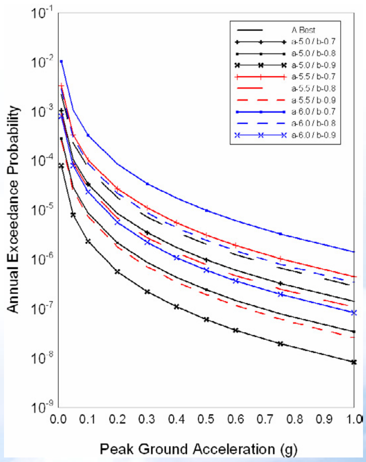
4. Results of Sensitivity Analysis

❖Sensitivity of Gutenberg-Richter Value (a-b pair)

- Hazard for △a=1.0 and △b=0.2 shows more than 2-order of deviation throughout the PGA level
- Hazard increases with bigger a-value and smaller b-value pair
- The b-value determines uncertainty range and a-value determines total hazard level.



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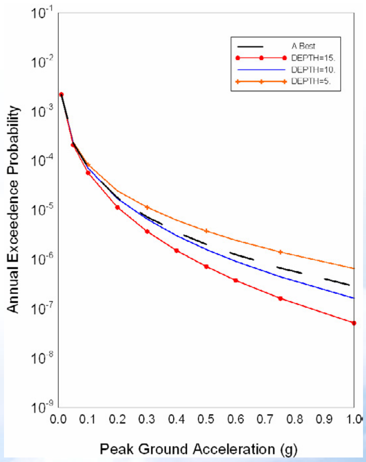
4. Results of Sensitivity Analysis

❖Sensitivity of Focal Depth

- Hazard for △d=5 km shows deviation of about 1/2-order maximum at PGA=1.0 g



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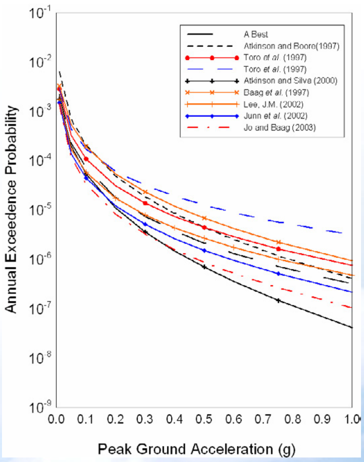
4. Results of Sensitivity Analysis

❖Sensitivity of Attenuation Equation

- Eq. of Toro et al.(1997) shows the largest hazard, and the eq. of Atkinson and Silva(2000) shows the smallest hazard.
- Maximum deviation is about 2-order at PGA level of 1.0g, and the deviation increases as PGA level increases



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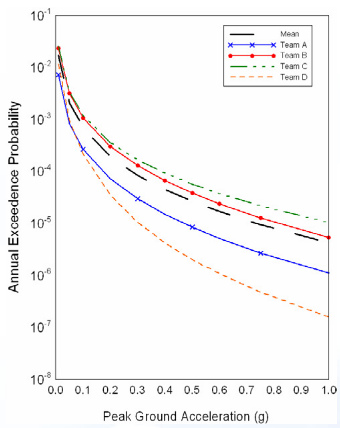
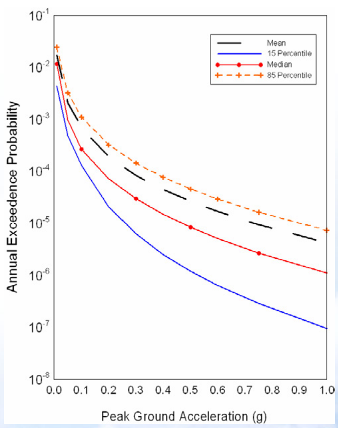
4. Uncertainty of Seismic Hazard

❖Hazard of each team



❖Uncertainty of hazard



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5. Conclusion(1)

- Sensitivity analysis showed that the uncertainty decreases in the order of recurrent frequency, attenuation equation, focal depth, maximum magnitude, seismic source.

- The Gutenberg-Richter parameter (or recurrent frequency) which was identified as the most uncertain in this study, showed more than 2-order of deviation in the hazard throughout the acceleration level considered.

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5. Conclusion(2)

- It was revealed also that the seismic parameter of the source within which a site is located primarily affected the seismic hazard.

- To reduce the excessive uncertainty which occurred in the PSHA for NPP, studies should be continuously performed with respect to the most uncertain recurrent frequency parameter.

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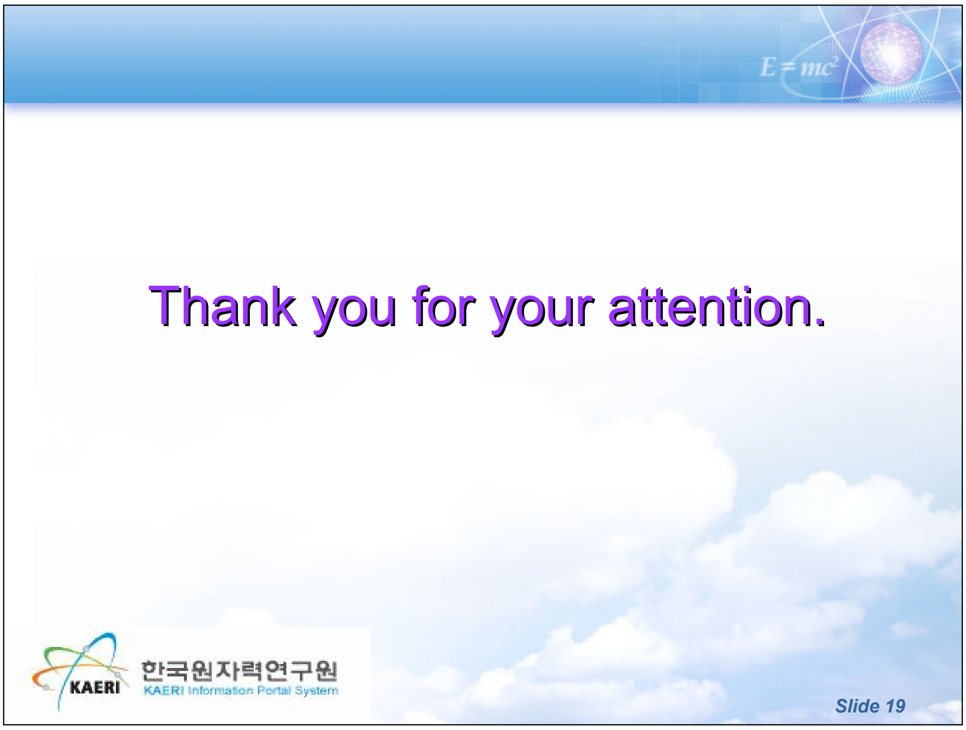
Thank you for your attention.
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Slide 19

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Appendix 1　Special Session: Status of PSA in each country or proposal for cooperation

China: Tao LIU (INET of Tsinghua University)
Professor Liu summarized recent PSA activities in China including model, technology, regulation, severe accident management, and application for advanced NPPs. She wants more and more exchange between China and other country to develop the PSA technology which will take a more important role in China.

Taiwan: Chung-Kung LO (INER)
Dr. Lo introduced background and brief history of PRA in Taiwan. The main issue of current PSA in Taiwan is the PEER review and its results of Taiwan NPPs. He provided a particular presentation on the review results and reviewer's recommendations of further enhancements. Some on-going PRA applications in Taiwan and their prospects are also explained.

Korea: Joon-Eon Yang (KAERI)
Dr. Yang presented the current PSA status & its applications in Korea. For the international cooperation, he suggested some future directions including the sharing of the experience, the expertise, and the knowledge. He emphasize that the establishment of the channel for cooperation in PSA is most important for the creation of new knowledge for us.

Japan: Mitsuhiro KAJIMOTO (JNES)
Dr. Kajimoto introduced the current status of PSA in Japan. Especially, he presented the detailed progress, activities, and database resources of PSA at JNES. For the future cooperation, he commented that the lack of human resource can be a serious problem. He insisted that the establishment of researchers list including the expert and young researchers is urgently needed.

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Overview of the PSA Related Activities in China

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

Tao ,Liu
INET of Tsinghua University, Beijing , China

Outline

- Booming Nuclear Energy Industry in China
- PSA Related Policy and Requirements
- PSA Related Organisations
- PSA Activity Categories
- HTRG PSA in INET

Booming Nuclear Energy Industry in China----Expected Goal of Nuclear Energy Development in 2020

- Installed Capacity: 40GW
- Estimate portion: 4%

- At least 2-3 units of 1000MW be constructed per year

Booming Nuclear Energy Industry in China

PSA Related Policy and Requirements ---Legal base

- “Technical policies on several significant safety issues in new NPP design", NNSA , including safety goal, severe accident requirements, Aug, 2002.
  - “Both deterministic approach and probabilistic approach should be used in the safety analysis during the design process”
  - Two quantitative probabilistic safety goal are given:
    - CDF < 10-5 / RY
    - LERF < 10-6 /RY

PSA Related Policy and Requirements ---Legal base

- “Important event sequences that may lead to a severe accident shall be identified using a combination of probabilistic methods, deterministic methods and sound engineering judgment” is issued by NNSA’s latest rules published on April 18th,2004. (Code on the Safety of NPP: Design. No. NS-R-1)

- “PSA should be performed for NPPs during safety assessment”. (Code on the Safety of NPP: Design. No. NS-R-1)

PSA Related Policy and Requirements

- For New NPPs, PSA is required and normally PSA report shall be submitted with FSAR
  - TQNPC, Tianwan, Ling dong,…
  - PSA quantitative goal should be satisfied
- For NPP in operations, PSA is positively encouraged and may be required in the period safety review

Relevant Organizations Working on PSA in China

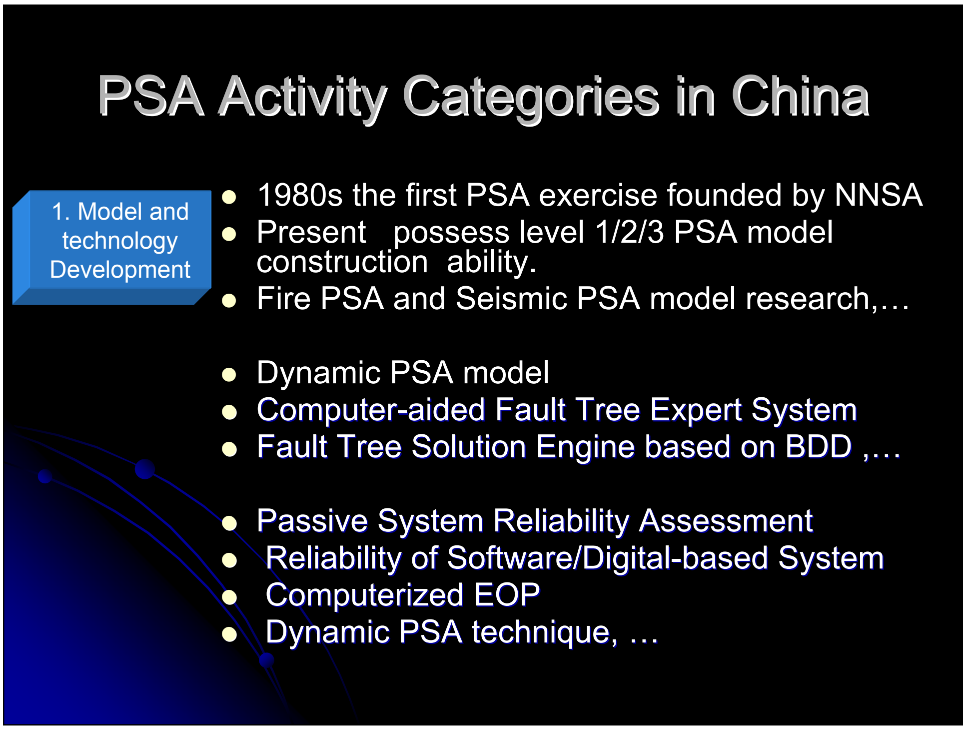
- **Nuclear Regulatory Body**
  - National Nuclear Safety Administration (NNSA), State Environment Protection Administration
- **Research Institute and University**
  - Institute of Nuclear and New Energy Technology(INET), Tsinghua University
  - Haer bin Engineering University, Shanghai Jiao Tong University, Xi'an Jiao Tong University, ….
  - China Institute of Atomic Energy(CIAE)
  - Institute of Plasma Physics, Chinese Academy of Sciences…...

Relevant Organizations Working on PSA in China

- Nuclear Energy Utilities and their research institute :
  - China National Nuclear Corporation (CNNC)
    - Nuclear Power Institute of China
    - Beijing Institute of Nuclear Engineering, …
  - State nuclear power technology Corporation (SNPTC)
    - Shanghai Nuclear Engineering Research and Design Institute
    - State Nuclear Power Technology Research Center, …
  - China Guangdong Nuclear Power Holding Co., Ltd (CGNPC)
    - China Nuclear Power Technology Research Institute
    - Suzhou Nuclear Power Institute , …

PSA Activity Categories in China

- PSA Model and technology research
- Risk-informed regulation
- Severe accident management
- Advanced NPP PSA activities
- PSA technical intercommunion


PSA Activity Categories in China
1. Model and technology Development
1980s the first PSA exercise founded by NNSA
Present possess level 1/2/3 PSA model construction ability.
Fire PSA and Seismic PSA model research,…
Dynamic PSA model
Computer-aided Fault Tree Expert System
Fault Tree Solution Engine based on BDD ,…
Passive System Reliability Assessment
Reliability of Software/Digital-based System
Computerized EOP
Dynamic PSA technique, …


PSA Activity Categories in China
2.Risk-informed regulation
From 2001, PSA-related regulations research
Risk Monitor System
Daily risk management
Period trend risk
Risk-informed …
Risk-informed TS
Risk-informed SSC categorization
Risk-informed ISI
MSPI for other systems
Technology policy of PSA application in the field of nuclear safety is to be built…

PSA Activity Categories in China

3. Severe accident management

- Establishment of the database
- Severe accident simulation based on PSA I results, to know plant-specific severe accident phenomena
- Computational Aid (CA) calculations to establish curves for quick help in SAMG
- Development of plant specific SAMG
- Review and Engineering modifications for plant-specific systems and equipment
- Revision of site emergence plan

PSA Activity Categories in China

4. Advanced NPP PSA

- PSA development and application with respect to high temperature gas cooled reactor.
- PSA development with respect to Advanced light water reactor Plant.

PSA Activity Categories in China

5.PSA Inter-communication

- Annual PSA forum from 2007
  - The 1th PSA forum organized by INET of Tsinghua University.
  - The 2th PSA forum organized by China nuclear energy association…
  - Participated by all of PSA related researchers and customers.
  - 2009 PSA forum will be held in Nov.
- PSA Training Courses

HTGR PSA in INET

- **Role of PSA in HTGR design**
  - **To support to specify the top level regulatory criteria for HTGR;**
  - **To provide inputs to identify licensing basis events including dominant beyond-design basis events which will be further evaluated within the severe accident management;**
  - **To identify the dominant accident sequences and confirm that top level design criteria are met;**

HTGR PSA in INET

Role of PSA in HTGR design (con'd)

- **To identify the dominant source terms and possible release paths to provide input to the emergency planning specifications,**
- **To provide inputs to the HTGR design, e.g. to check that the level of redundancy and diversity provided in the safety system is adequate, and to assess more detailed design issues such as the consideration of the second shutdown system, the residual heat removal system and so on.**

HTGR PSA in INET

- Present Status –preliminary PSA report.
  - Initiating event identification
  - Major accident sequences development
  - Database establishment
  - System model development

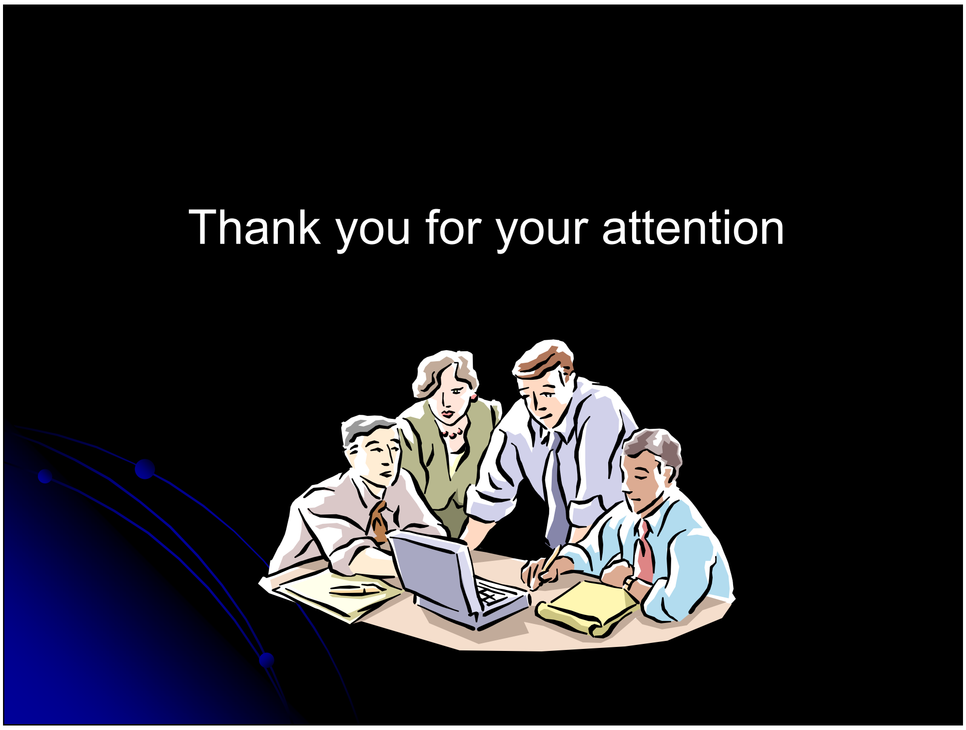
HTGR PSA in INET

Special Features for HTGR PSA Development

- No “core damage” or “large early release” pinchpoints; CDF and LERF not applicable; release categories defined for event trees
- Integrated 1&2 (&3) PSA
- Integrated passive system reliability assessment

Prospect

- PSA will play an more important role , and China urgently needs to develop PSA technology.
- More and more exchange between China and other country will benefit for PSA development .
- Welcome to our PSA forum, you will be one of us in the future…


Thank you for your attention

PRA Quality
Peer Review Results of Taiwan NPPs

Chung-Kung Lo
Associate Researcher
Nuclear Engineering Division
Institute of Nuclear Energy Research

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

Overview

- Background and Brief History
- Peer Review Results and Gap Analyses
- Comments and Regulatory Response
- PRA Applications

2

The Need of Robust PRA Quality

- **Maintenance Rule being implemented at the three operating NPPs in Taiwan**
  - A complementary measure to the license renewal program
- **Regulatory decisions of AEC on some incidents**
- **Urge for more "routine" on-line maintenances**
- **Efforts to achieve significant refueling outage shortening and lower collective exposure**
  - Risk-informed inservice inspection
  - Risk-informed inservice testing

3

A Brief History of PRA in Taiwan

Major PRA Projects (Main Sponsor)	Periods	Scope							Application	Task Force Man-year
		P	SM	TY	FR	FL	SD	L2		
Kuosheng (AEC)	1983 ◆ 1985	✓	✓	✓	✓	✓		✓	Base PRA Model	37 (4.5)*
Maanshan (AEC)	1985 ◆ 1987	✓	✓	✓	✓	✓		✓ (1992)	Base PRA Model	27.5 (2.0)
Chinshan (AEC)	1988 ◆ 1990	✓	✓	✓	✓	✓		✓	Base PRA Model	34.5 (1.0)
1st-3 (Taipower)	1994 ◆ 1997	✓	✓	✓			✓		Few cases of justification of continued operation	52
2nd-3 (Taipower)	1997 ◆ 2000	DCR & experience updates (~1999)			✓	✓	✓	CSET CPET	• TIRM (risk monitor) • PRAM (PRA Maintenance) • OLM	66
3rd-3 (Taipower)	2000 ◆ 2003	DCR & experience updates (~2002)						LERF	• TIRM-2 • FT Engine developed • NEI-00-02 peer review • Chinshan RIFA (Risk-informed Fire Wrapping Alternatives Analysis) • RI-ISI pilot	66
4th-3 (Taipower)	2004 ◆ 2007	DCR & experience updates (~2005)							• SDP tool (PRiSE) developed • Kuosheng & Maanshan RIFA • Follow-on NEI Peer Review • Maintenance Rule	66

P: Internal at-power; SM: Seismic; TY: Typhoon; FR: Internal Fire; FL:Internal Flood; SD: Shutdown; L2: Level-2
∗:from US Consultant

4

Scope

- **The first peer review of PRA models in Taiwan was conducted in 2002 per NEI-00-02**
- **The Follow-on peer reviews**
  - 3 reviews on Chinshan (BWR 4), Kuosheng (BWR 6), Maahshan (PWR) PRA models respectively in August, November of 2006, and January of 2007
  - Review team organized by ABS Consulting Inc. authorized via bid process
  - Team members from ABS and FP&L
- **Scope includes Level 1 + (ie. Level 1 and Containment system analyses), internal and external at-power and shutdown events PRA**

5

Review Results after F&O Addressed

- **F&O levels distribution (Chinshan NPP at-power PRA)**

Number of 'Fact and Observation's / Technical Element	Importance Level of F&Os									F&O SUM	
	A/B		C			D					
	2002	Follow-on (2006)	2002	Follow-on (2006)		2002	Follow-on (2006)			2002	Follow-on (2006)
Accident Sequence	8	0	10	6	7	3	2	1	0	21	16
Data Analysis	4	0	9	3	2	7	7	2	1	20	15
Dependence	1	0	8	7	0	0	0	1	1	9	9
Fire	0	0	8	6	0	2	2		0	10	8
Human Reliability Analysis	5	0	11	5	3	4	3	5	1	20	17
Initiating Event Analysis	8	0	18	10	6	3	1	5	0	29	22
L2 (Containment System)	2	0	3	2	0	4	1	1	0	9	4
Quantification	3	0	4	4	1	4	4		1	11	10
Seismic Analysis	2	0	10	6	1	1	1	0	0	13	8
Structure Analysis	2	0	2	2	2	0	0	0	0	4	4
System Analysis	4	0	16	11	2	1	1	2	2	21	18
Thermohydraulic Analysis	1	0	9	4	1	1	1	2		11	8
Typhoon Analysis	1	0	3	3	0	0	0	0	0	4	3
SUM	41	0	111	69	25	30	23	19	6	182	142

6

Gap Analysis after F&O Addressed

- ASME Standard Gap Analysis (Chinshan at-power PRA)

High Level Requirements	# of Supporting Requirements	# of CC I	# of CC II	# of CC III	# Not met	NA	Not reviewed
HLR-AS	21	0	3	14	3	1	0
HLR-DA	28	2	11	10	5	0	0
HLR-HR	34	3	8	18	5	0	0
HLR-IE	29	3	6	15	5	0	0
HLR-IF	28	0	2	11	6	0	9
HLR-LE	36	4	12	12	6	2	0
HLR-QU	31	3	3	13	9	1	2
HLR-SC	15	0	2	7	5	0	1
HLR-SY	41	0	7	32	2	0	0
Sum (Internal Events)	263	15	54	132	46	4	12
HLR-FR	25	0	1	1	2	1	20
HLR-HA	25	2	1	0	2	0	20
HLR-SA	24	6	4	5	9	0	0
Sum (Seismic Event)	74	8	6	6	13	1	40
Typhoon Event	14	0	3	1	5	1	4
Total*	351	23	63	139	64	6	56

＊: Fire events not included

7

Review Results after F&O Addressed

- F&O levels distribution (Kuosheng NPP at-power PRA)

Number of 'Fact and Observation's / Technical Element	Importance Level of F&Os									F&O SUM	
	A/B		C			D					
	2002	Follow-on (2006)	2002	Follow-on (2006)		2002	Follow-on (2006)			2002	Follow-on (2006)
Accident Sequence	7	0	10	5	3	4	4	2	1	21	15
Data Analysis	5	0	7	1	3	6	5	2	1	18	12
Dependence	1	0	7	7	0	0	0	0	1	8	8
Fire	0	0	7	6	0	1	1	0	0	8	7
Human Reliability Analysis	5	0	9	3	1	0	0	5	1	14	10
Initiating Event Analysis	6	0	13	2	6	0	0	8	0	19	16
L2 (Containment System)	3	0	2	0	0	3	2	1	0	8	3
Quantification	1	0	2	1	0	2	2	0	1	5	4
Seismic Analysis	4	0	7	5	1	1	1	1	0	12	8
Structure Analysis	2	0	2	2	2	0	0	0	0	4	4
System Analysis	7	0	14	10	5	0	0	1	2	21	18
Thermohydraulic Analysis	1	0	5	3	1	0	0	1	0	6	5
Typhoon Analysis	1	0	2	1	0	0	0	1	0	3	2
SUM	43	0	87	46	22	17	15	22	7	147	112

8

Gap Analysis after F&O Addressed

- ASME Standard Gap Analysis (Kuosheng at-power PRA)

High Level Requirements	# of Supporting Requirements	# of CC I	# of CC II	# of CC III	# Not met	NA	Not reviewed
HLR-AS	21	0	3	14	3	1	0
HLR-DA	28	2	11	10	5	0	0
HLR-HR	34	3	8	17	6	0	0
HLR-IE	29	4	5	15	5	0	0
HLR-IF	28	0	2	11	6	0	9
HLR-LE	36	4	12	12	6	2	0
HLR-QU	31	3	3	13	9	1	2
HLR-SC	15	0	2	7	5	0	1
HLR-SY	41	0	7	32	2	0	0
Sum (Internal Events)	263	16	53	131	47	4	12
HLR-FR	25	0	1	1	2	1	20
HLR-HA	25	2	1	0	2	0	20
HLR-SA	24	5	4	5	10	0	0
Sum (Seismic Event)	74	7	6	6	14	1	40
Typhoon Event	14	0	3	1	5	1	4
Total*	351	23	62	138	66	6	56

＊: Fire events not included

9

Review Results after F&O Addressed

- F&O levels distribution (Maashshan NPP at-power PRA)

Number of 'Fact and Observation's / Technical Element	Importance Level of F&Os									F&O SUM	
	A/B		C			D					
	2002	Follow-on (2006)	2002	Follow-on (2006)		2002	Follow-on (2006)			2002	Follow-on (2006)
Accident Sequence	1	0	9	1	1	8	2	3	0	18	7
Data Analysis	3	0	9	2	2	9	3	6	0	21	13
Dependence	0	0	11	7	0	1	1	2	0	12	10
Fire	0	0	8	6	0	2	2	1	0	10	9
Human Reliability Analysis	1	0	8	2	0	5	4	3	1	14	10
Initiating Event Analysis	8	0	19	2	4	4	2	6	3	31	17
L2 (Containment System)	1	0	5	0	0	4	1	3	0	10	4
Quantification	2	0	2	1	1	5	5	1	1	9	9
Seismic Analysis	1	0	5	5	1	1	1	0	0	7	7
Structure Analysis	2	0	1	0	2	1	1	1	0	4	4
System Analysis	3	0	16	3	0	6	4	6	1	25	14
Thermohydraulic Analysis	1	0	8	3	1	4	3	1	0	13	8
Typhoon Analysis	0	0	3	2	0	1	1	1	0	4	4
SUM	23	0	104	34	12	51	30	34	6	178	116

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Gap Analysis after F&O Addressed

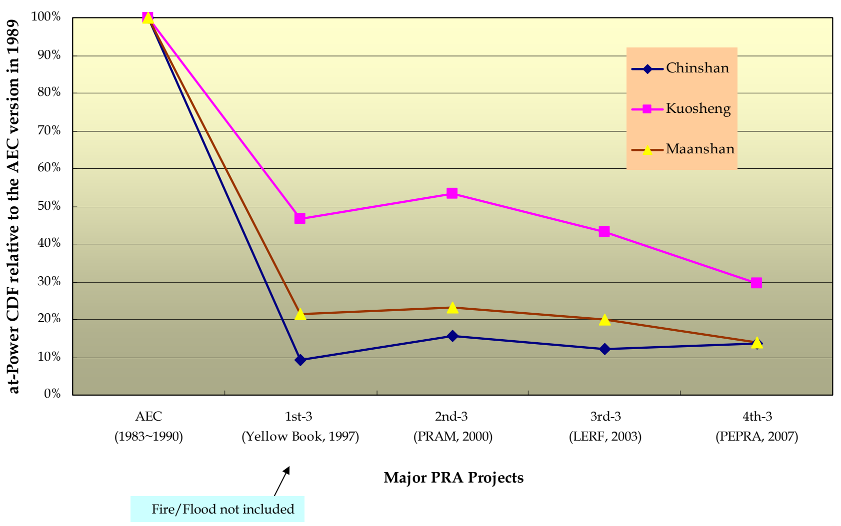
- ASME Standard Gap Analysis (Maanshan at-power PRA)

High Level Requirements	# of Supporting Requirements	# of CC I	# of CC II	# of CC III	# Not met	NA	Not reviewed
HLR-AS	21	0	3	17	0	1	0
HLR-DA	28	2	11	9	6	0	0
HLR-HR	34	3	8	18	5	0	0
HLR-IE	29	4	5	14	6	0	0
HLR-IF	28	0	2	11	6	0	9
HLR-LE	36	4	14	12	6	0	0
HLR-QU	31	3	3	13	10	1	1
HLR-SC	15	0	2	7	5	0	1
HLR-SY	41	0	7	32	2	0	0
Sum (Internal Events)	263	16	55	133	46	2	11
HLR-FR	25	0	1	1	2	1	20
HLR-HA	25	2	1	0	2	0	20
HLR-SA	24	6	4	5	9	0	0
Sum (Seismic Event)	74	8	6	6	13	1	40
Typhoon Event	14	0	3	1	5	1	4
Total*	351	24	64	140	64	4	55

*: Fire events not included

11

Calculated Risk Variation



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Reviewers' Recommendations of Further Enhancements

- **Can be more realistically reflect the risk characteristics of the plants**
  - Incorporate into PRA models the more recent generic data and new failure modes in conjunction with collected plant-specific operation experience
- **Some initiating events need more rigorous presentations**
  - The impact of HE events on CDFs under LOOP
  - Consideration of accident mitigation of LOIA events
  - The impact of HE events on LERF under SGTR
  - The recovery operation of feedwater under general transients
- **Consider the updates of Seismic and Fire PRAs**

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Response of the AEC

- **The PRA follow-on peer review results were submitted for record by Taipower and approved docketed by the AEC in February, 2009**
- **Concerns of the AEC**
  - PRA documentation quality control
  - The upgrade and update of seismic PRA models to incorporate advanced methods and more recent earthquake experiences
  - Alignment of the PRA self-assessment and peer review process with RG 1.200
  - Classification and configuration control of PRA models

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On-going PRA Applications

- Supporting Maintenance Rule implementation
- Base PRA models enhancement, including the Lungmen NPP, to meet ASME internal events standard
  - Objective: Capability Category II in general
- RI-ISI program plan development for the 4-th 10-year operating interval of Kuosheng NPP
- SDP tools (ie. PRiSE) kernel update and enhancement to include shutdown events
- ASME PRA standard introduction
- A fire SDP tool being developed for Chinshan

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PRA Applications in View

- Supporting self management of On-line Maintenance
- Fire PRA upgrade in support of NFPA-805 transition
- SDP tools enhancement to include external events (ie. Seismic, Typhoon)
- Tech Spec AOT Extensions
- Risk-informed IST
- Risk-informed Tech Spec

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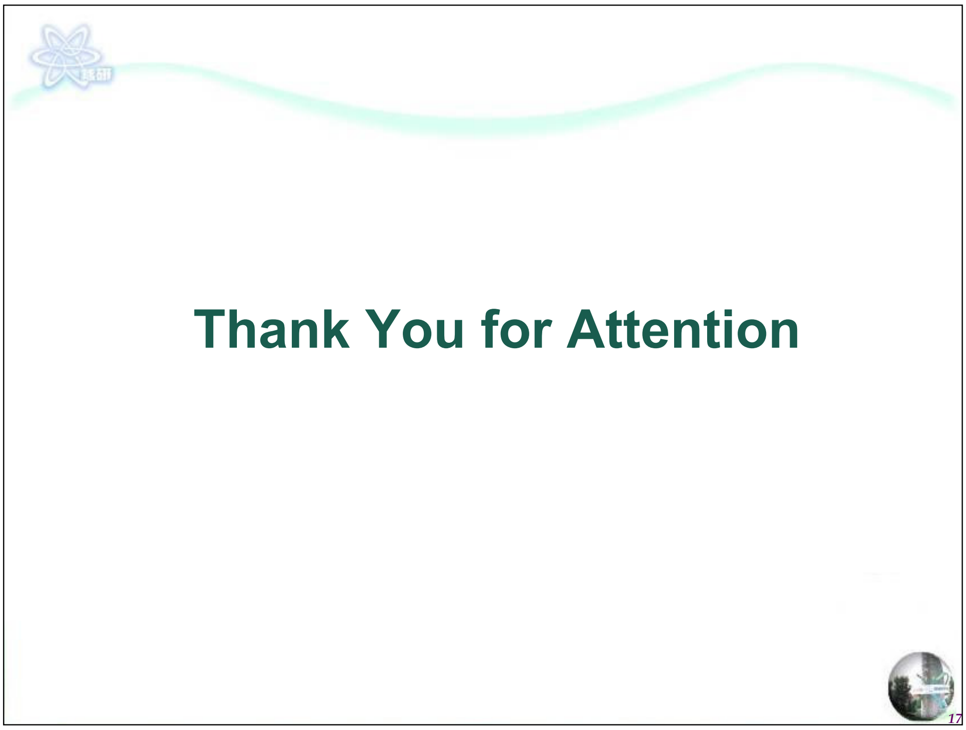

Thank You for Attention
17

Current Status of PSA in Korea & International Cooperation

19 May 2009

Joon-Eon YANG

Integrated Safety Assessment Division
Korea Atomic Energy Research Institute

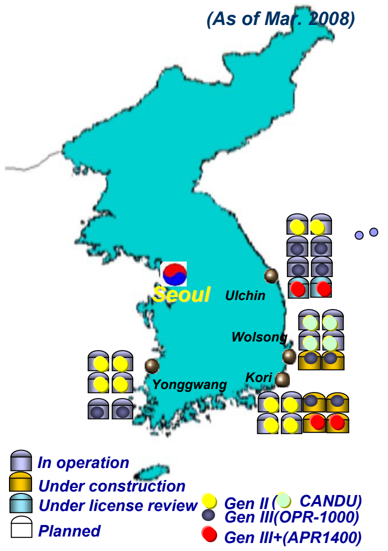
Contents

- Current Korean Status
- PSA & Its Applications in Korea
- Current Issues in Korea
- Future Directions for International Cooperation

2

Current Korean Status

- Building New Plants
  - 20 operating NPPs
  - 6 under construction, 2 under licensing review process
- From PSA to RIPBA
  - Two Statements
    - Nuclear Safety Policy Statement
    - Severe Accident Policy Statement
  - From Generic to Specific



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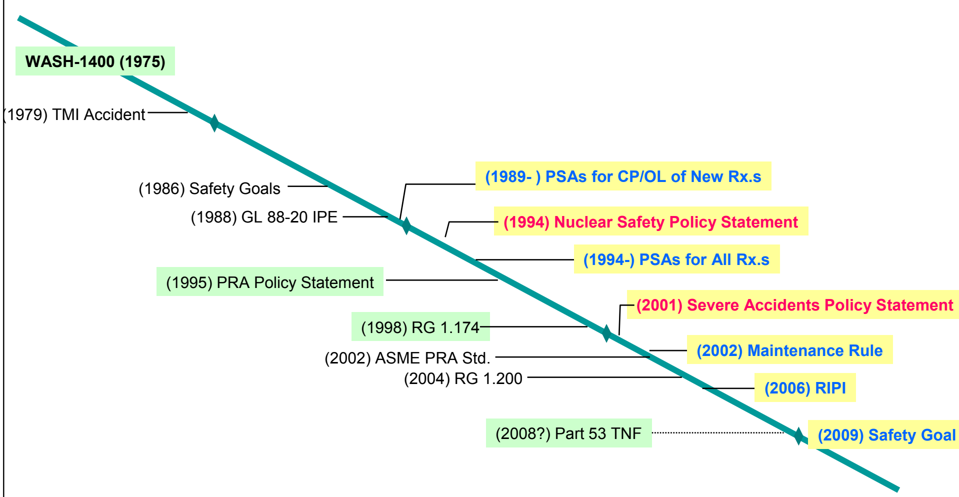
Major Milestones in Korean PSA/RIPBA



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4

Present Status of Korean PSA

Unit	Scope & Purpose of PSA	Completed Date
KR-1	PSA for TMI action Plan	'02.11
	PSA update for Risk monitor	'07.05
KR-2	PSA for TMI action Plan	'03.12
	PSA update for Risk monitor	'07.06
KR-3,4	PSA for TMI action Plan	'92.08
	PSA update for Risk monitor	'03.06
YG-1,2	PSA (LEVEL I) for TMI action Plan	'92.08
	PSA update (LEVEL I, II) for Severe Accident Policy statement	'03.12
	PSA update for Risk monitor	'07.12
YG-3,4	PSA (LEVEL I) for the requirement for construction & operation permits	'94.02
	PSA update (LEVEL I, II) for Severe Accident Policy statement	'04.12
	PSA update for Risk monitor	'05.06
YG-5,6	PSA (LPSD PSA) for the requirement for construction & operation permits	'00.12
	PSA update for Risk monitor	'05.12
UC-1,2	PSA for Severe accident policy	'05.12
	PSA update for Risk monitor	'06.12
UC-3,4	PSA (LEVEL I) for the requirement for construction & operation permits	'97.1
	PSA update (LEVEL I, II) for Severe Accident Policy Statement	'04.12
	PSA update for Risk monitor	'05.06
UC-5,6	PSA (LOSD PSA) for the requirement for construction & operation permits	'02.06
	PSA update for Risk monitor	'06.06
WS-1	PSA for Severe Accident Policy statement	'03.12
	PSA update for Risk monitor	'07.02
WS-2,3,4	PSA for the requirement for construction & operation permits	'97.1
	PSA update for Risk monitor	'07.02

5

On-going Programs in Korea

- PSA
  - Level 2 PSAs for all units have been finished & being updated (with Risk Monitor)
  - Plant Specific Reliability DB
- Risk-informed Applications
  - RI-ISI: Under Review of Regulatory Body
  - RI-TS: Extension of STI of ESFAS/RPS
    - Permitted for some NPPs
    - New Projects are on-going for other NPPs (AOT Extension of EDG)
  - RI-ILRT
- Performance-monitoring
  - MR: Standardized MR Approach is being tested by KHNP
- Regulatory Activity
  - Guidelines for RIR & PSA Quality
  - RIPI (Risk-informed Periodic Inspection) is being tested by KINS
  - Regulatory PSA Model

6

Current Issues in Korea

- RIPBA in Korea
  - Korean Specific RIPBA Framework
    - Safety Goal
  - PSA Quality
- Licensing/RI-Design of New NPPs
  - Digital I&C PSA & HRA
  - PSA for the D.C. of SMART
  - RI-Design Framework
- Another issues
  - Human Performance
  - Seismic PSA
  - PSA Tools
    - AIMS & FTREX
    - SAREX & FORTE

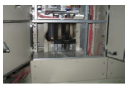
7

Some Research Results of KAERI

- Digital I&C PSA
  - DI&C Induced IE DB
  - Rel. Analysis Methods for HW/SW
  - HRA for Digitalized MCR
    - Based on the trainings at simulator of advanced reactor

- Enhancing Human Performance
  - OPERA DB
    - video taped >100 cases of operator trainings at simulator
  - Complexity Measure: TACOM

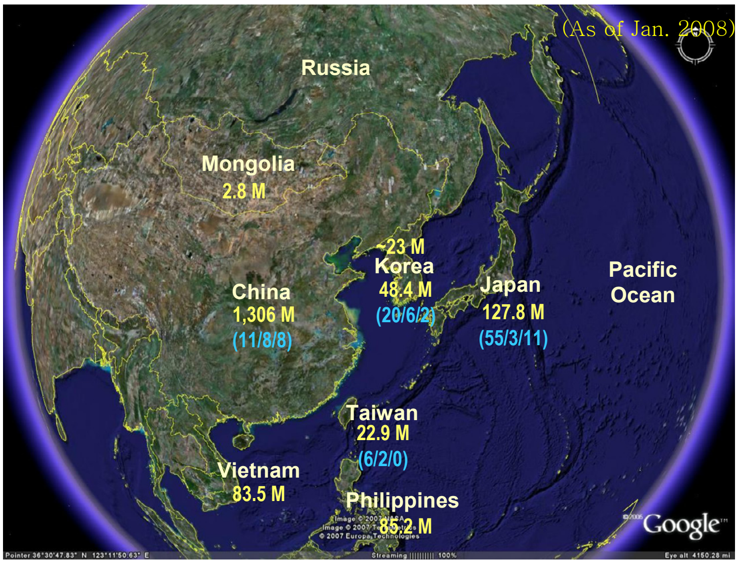
- Seismic PSA
  - Korean Specific Hazard Curve
  - Seismic PSA Code
  - Shaking Table Exp. for Digital MCC



- Integration of Internal & External PSA Model
  - A S/W for the Automatic Generation of External PSA Model (AIMS)
    - One Top Internal PSA Model + Fire DB → One Top Fire PSA Model
  - New Quantification Algorithm for External PSA Model (JSTAR)
- Integrated Assessment of Risk & Performance
  - Detailed FT Model for BOP
- FREX
  - ZBDD based
  - EPRI R&R: CAFTA + FTREX

8

Nuclear Power in East Asia: 92/19/22=133



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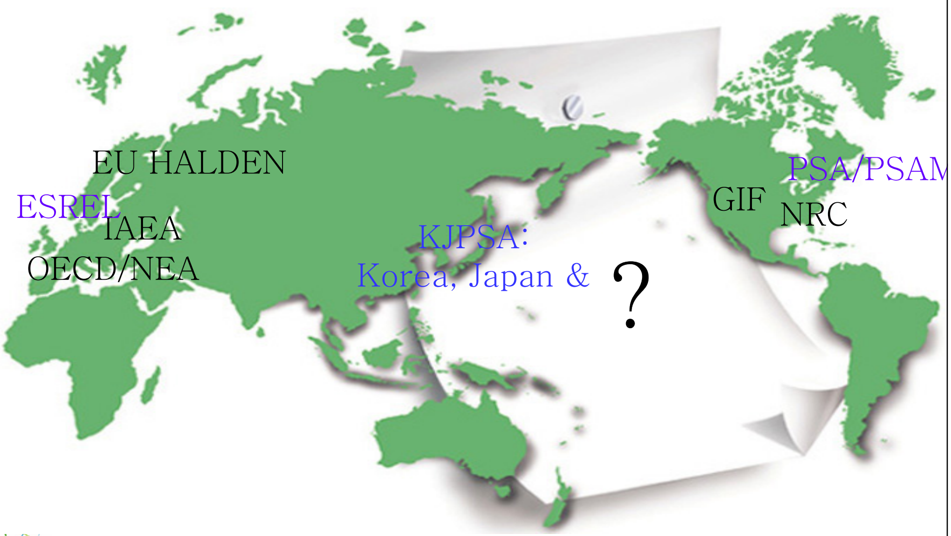
International Cooperation in the World



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International Cooperation in East Asia

- Sharing the Experience
  - Ex.) CCF Data
- Sharing the Expertise
  - Ex.) Expert Pool for Peer Review
  - Ex.) Training & Education of New Generation
- Sharing the Knowledge
  - Ex.) Digital I&C, Seismic
    → Joint Development or V&V

→ Establishing the Channel for Cooperation in PSA: Creating New Knowledge for Us

11

Status of PSA in Japan & Future Cooperation

May 19, 2009

Mitsuhiro KAJIMOTO

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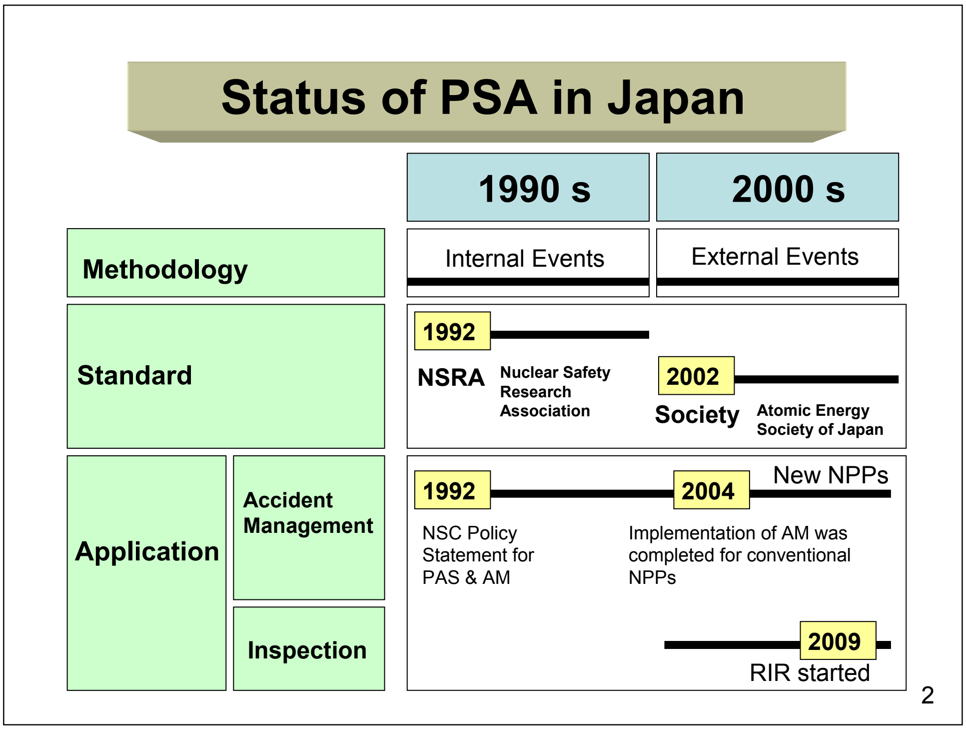
3-17-1, Toranomon, Minato-ku, Tokyo,
105-0001 Japan

The 10th Korea-Japan Joint Workshop on PSA (KJPSA), at Haevichi Hotel & Resort, Jeju, Korea (May 18-20, 2009)

1



Methodology Development

Organization	Field	Tools
JAEA	Level 1 PSA	NUPRA
	Level 2 PSA	THALES-2
	Level 3 PSA	OSCAAR
JNES	Level 1 PSA	NUPRA
	Level 2 PSA	MELCOR CFD (FLUENT)
	Level 3 PSA	MACCS-2
Industries	Level 1 PSA	RSIKMAN, etc.
	Level 2 PSA	MAAP
	Level 3 PSA	MACCS-2

3

Progress of PSA at JNES

Category	Initial Events	Level 1 PSA	Level 2 PSA	Level 3 PSA
Internal Events	Random (Rated/ Shutdown)			
	Fire			
	Flooding			
External Events	Seismic			
	Tsunami			

Well established   in progress   not yet

4

PSA Database at JNES

PSA	Resource	comment
Level 1 PSA	Experience	event data for 55 plants (NuCIA)
	Generic	CRIEPI, NSRA, PSA Data in USA
Level 2 PSA	Analysis	ROAAM Approach
	Experiments	OECD/NEA Projects International Cooperation Projects
Level 3 PSA	measurements	Metrological data, population distribution around site

5

Development of PSA Standard

Standards Committee in *Atomic Energy Society of Japan* has been developing various standards for PSA.

	Standard	Status
1	Shutdown PSA (Level 1 PSA)	Released in 1992, Rev.1 in progress
2	Seismic PSA (Level 1.5 PSA)	Released in 2007
3	Level 1 PSA	Released in 2009
4	Level 2 PSA	Released in 2009
5	Level 3 PSA	Released in 2009
6	Parameter for PSA	in progress
7	RIR	in progress
8	Fire PSA	under discussion
9	Flooding PSA	under discussion

http://www.aesj.or.jp/sc/index.html

6

PSA Applications

Nuclear Safety Commission (NSC)

PSA & Severe Accident

(1) NSC, "Accident Management as a Countermeasure for Severe Accidents at Light Water Nuclear Power Reactor Facilities," NSC decision statement (1992).

Safety Goal & Performance Objectives

(1) NSC, “Interim Report on Safety Goals,” NSC Special Committee on Safety Goals (2003).

(2) NSC, “Performance Objectives for NPP,” NSC Special Committee on Safety Goals (2006).

Risk Informed Regulation (RIR)

(1) NSC, “Basic Policies for RIR Introduction,” NSC decision statement (2003).

(2) NSC, “Interim Report on RIR Implementation,” NSC Task Force on RIR Implementation (2007).

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PSA Applications (Continued)

Nuclear & Industrial Safety Agency (NISA)

PSA & Severe Accident

(1) NISA, “Preparation of Accident Management for Light Water Type Nuclear Power Stations” NISA Report (former NREA) (1996).

(2) NISA & JNES, "PSA Quality Guidelines for NPP Applications," NISA/JNES Sub-Committee on RIR (2006).

Risk Informed Regulation (RIR)

(1) NISA, “Basic Concept to Apply ‘Risk Information’ to Nuclear Safety Regulation,” NISA Committee on Nuclear Safety (2005)

(2) NISA, "High-level Guidelines for Utilization of 'Risk Information' in Safety Regulations for NPPs," NISA/JNES Sub-Committee on RIR (2006).

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Future Cooperation

Fields of Cooperation

Field	Item	Contents
Methodology	Computer Code Database	Shearing Code Development with Cooperation.
Standard	-	-
Application	Experience	Shearing Good practice Analysis

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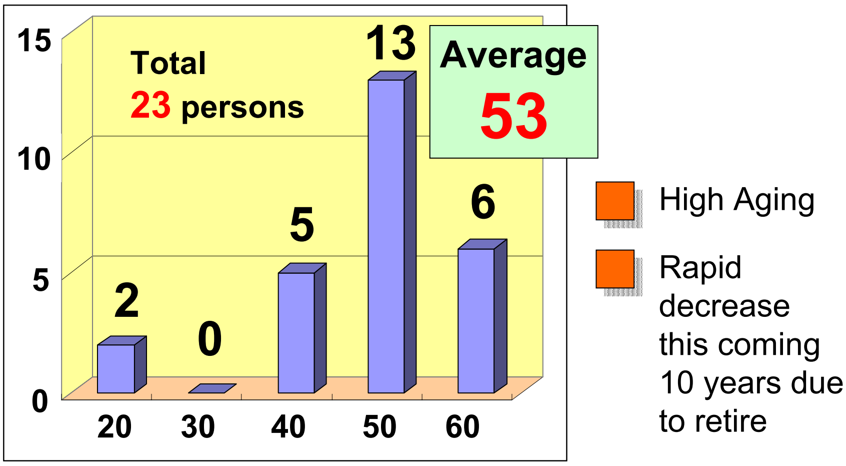
Future Cooperation (Continued)

Serious Problem : Lack of Human Resource



Researchers for PSA & Severe Accident at JNES

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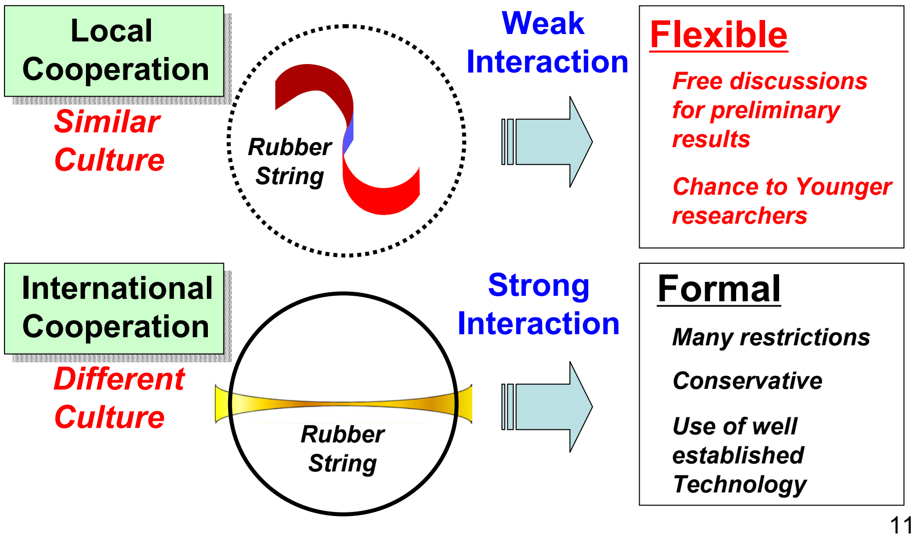
Network with Rubber Strings
= How to establish Cooperation =



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Network with Rubber String: major objective:
technical transfer for young generation, establish PSA families in Asian area

- **Establish List of Researchers**

  Lists of Expert and young researchers in various PSA field (Research Institute, Industry, University)

- **Families in PSA Field**

  Establish families and heads in each PSA Field based on the Lists

  Presentation of their activities at the special session in KJPSA Workshop

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Appendix 2 Panel Discussion

Theme : How to Improve the Cooperation in PSA
Moderator : Prof. Un-Chul LEE (Seoul National Univ.)
Panelists : Akihide HIDAKA, Akira YAMAGUCHI, Mitsuhiro KAJIMOTO (Japan), Byung Sik LEE, Chang-Ju LEE, Joon-Eon YANG (Korea), Chung-Kung LO (Taiwan), Tao LIU (China)

■ Theme: Cooperation in PSA

Topic: 1. Exchange Program, 2. Information Exchange, 3. Making Young Generation be involved in PSA

■ Dr. Akihide HIDAKA (JAEA)

- ✓ Risk Informed (RI) is introduced in Japan. Transparency is important in PSA, and DB. PSA quality of Japanese NPP could not be evaluated through peer review by Japanese experts since Japanese hesitate to criticize each other. Thus, other country could perform peer review of Japanese NPP as the 3rd party.
- ✓ NUSIA is open to public. Like Finland T-book, Japanese DB could be available to other country.

■ Mr. Byung Sik LEE (KHNP)

- ✓ KHNP(Utility) performed many RI applications such as RI-ILRT, RI-ISI, MRule, Risk Monitor, etc. and tried to improve PSA Quality. Equipment Reliability is developed. Finally, on-line maintenance will be performed.
- ✓ Let’s share the other international cooperation.

■ Prof. Akira YAMAGUCHI (Univ. of Osaka)

- ✓ Let’s share the idea, and exchange the information.
- ✓ Quality, robustness, and validation of PSA is important. Let’s do a common work to reduce an uncertainty and to improve a PSA quality.
- ✓ Infrastructure such as human resource should be enhanced. Human resource related to PSA is limited. (No PSA curriculum in University). New international nuclear research institute is going to be built around Kansai. PSA field could be trained or studied.

■ Dr. Chang-Ju LEE (SNU)

- ✓ It is not easy to have a cooperation between countries. For example, for COOPRA case, there is little cooperation except information exchange.
- ✓ Suggestion:
    - ・ Let’s assign a living program coordinator each country.
    - ・ e.g.) Hidaka for DB in Japan…
    - ・ Technical support for special topics

  - Present the results to K-J PSA Workshop

- Dr. Mitsuhiro KAJIMOTO (JNES)
  - ✓ Let's have small working group meeting within KJPSA
- Dr. Joon-Eon YANG (KAERI)
  - ✓ Informally several people were gathered and discussed about cooperation in PSA as a small working group meeting. It is necessary to issue news letters during 2 years.
- Dr. Mitsuhiro KAJIMOTO (JNES)
  - ✓ PSA peer review is industry side problem, not regulatory side issue.
  - ✓ Question: Why the component failure data is good in japan ?
  - ✓ Answer: The component failure data is not much better than others. However. the frequency of such event is very small.
- Dr. Dae-Wook CHUNG (KINS)
  - ✓ In a working group or session chairman meeting, let's decide topics in which two countries are commonly interesting.
- Dr. Joon-Eon YANG (KAERI)
  - ✓ If a topic (i.e., young generation issue in PSA) is selected among Asian countries, then it is no problem in making a working group in OECD/NEA.
- Dr. Key Yong SUNG (KINS)
  - ✓ Severe Accident Group is active. Let's include severe accident group in Asian networks.
- Prof. Un-Chul LEE (SNU)
  - ✓ The workshop title change from K-J PSA workshop to Asian PSA workshop is a little bit too early. Maybe, 2 years later, if many Chinese and Taiwanese participate, then let's discuss again.
- Dr. Chung-Kung LO (INER)
  - ✓ Information Exchange: I promise I could be a contact point. If you need information, please ask me.
- Dr. Joon-Eon YANG (KAERI)
  - ✓ For an exchange of information, this k-j psa workshop internet site will be kept. The address is; asian.psa.re.kr
  - ✓ This workshop paper and presentation material will be posted in this site.

- Summary by Prof. Un-Chul LEE (SNU)
  - ✓ A small working group which handles the following topics should be operated as soon as possible.
    - Collaborative work in PSA
    - Information exchange program
    - Training each other
  - ✓ Japanese participants showed very good job. Especially, JAES coped well the earthquake accident, and well explained it.
  - ✓ For the Korean side, those who were involved in PSA is getting older and older. Fresh young generation is necessary. Please help us how to make the young generation be interested in PSA.

- Summary by Dr. Toshimitsu HOMMA (JAEA)
  - ✓ Participants are 80 persons.
  - ✓ Participant countries are Korea, Japan, USA, China, Taiwan
  - ✓ It is the 3rd times that K-J PSA workshop was held in Jeju island.
  - ✓ At the beginning of K-J PSA workshop, I was the only person who presented a paper about level 3. Now, many persons presented papers about level 3.
  - ✓ I thanks Dr. Yang, Dr. Han, Dr. Choi for preparing this nice K-J PSA workshop.
  - ✓ Let's publish a report about this K-J PSA workshop including this panel discussion.

Appendix 3 List of Participants

Family Name	Given Name(s)	Organization	Country
AHN	Kwang-Il	Korea Atomic Energy Research Institute	Korea
AHN	Sang-Kyu	Korea Institute of Nuclear Safety	Korea
BRAVERMAN	Joseph	Brookhaven National Laboratory	USA
CHOI	In-Kil	Korea Atomic Energy Research Institute	Korea
CHOI	Jong-Soo	Korea Institute of Nuclear Safety	Korea
CHOI	Kwang-Hee	Korea Electric Power Research Institute	Korea
CHOI	Seong-Nam	Korea Institute of Nuclear Safety	Korea
CHOI	Sun Yeong	Korea Atomic Energy Research Institute	Korea
CHOI	Young	Korea Atomic Energy Research Institute	Korea
CHUNG	Bagsoon	Korea Electric Power Research Institute	Korea
CHUNG	Dae-Wook	Korea Institute of Nuclear Safety	Korea
CHUNG	Ku Young	Korea Institute of Nuclear Safety	Korea
FUNAYAMA	Kyoko	Japan Nuclear Energy Safety Organization	Japan
HA	Jaejoo	Korea Atomic Energy Research Institute	Korea
HAHM	Daegi	Korea Atomic Energy Research Institute	Korea
HAN	Sang Hoon	Korea Atomic Energy Research Institute	Korea
HIDAKA	Akihide	Japan Atomic Energy Agency	Japan
HOMMA	Toshimitsu	Japan Atomic Energy Agency	Japan
HSU	Pi-Lin	Institute of Nuclear Energy Research	Taiwan
HWANG	Seok Won	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
IMAI	Hidetaka	Tokyo Electric Power Company	Japan
IN	Young H.	ERIN Engineering & Research, Inc.	USA
ISHIKAWA	Jun	Japan Atomic Energy Agency	Japan
JANG	Dong Ju	Korea Institute of Nuclear Safety	Korea
JEE	Moon-Hak	Korea Electric Power Research Institute	Korea
JEONG	Jongtae	Korea Atomic Energy Research Institute	Korea
JERNG	Dong Wook	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
JIN	KwangMan	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
JIN	Youngho	Korea Atomic Energy Research Institute	Korea
JUNG	Tae Sang	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
KAJIMOTO	Mitsuhiro	Japan Nuclear Energy Safety Organization	Japan
KANG	Chang-Sun	Seoul National Univ.	Korea
KANG	Kyungmin	Korea Institute of Nuclear Safety	Korea
KANG	Sun-Koo	Korea Power Engineering Company	Korea

Family Name	Given Name(s)	Organization	Country
KIM	Do Sam	Korea Institute of Nuclear Safety	Korea
KIM	Dong-Ha	Korea Atomic Energy Research Institute	Korea
KIM	Dong-kyu	Korea Power Engineering Company	Korea
KIM	Euna	Korea Atomic Energy Research Institute	Korea
KIM	Han-Chul	Korea Institute of Nuclear Safety	Korea
KIM	Kilyoo	Korea Atomic Energy Research Institute	Korea
KIM	Man Cheol	Korea Atomic Energy Research Institute	Korea
KIM	Min Kyu	Korea Atomic Energy Research Institute	Korea
KIM	Myung-Ki	Korea Electric Power Research Institute	Korea
KIM	Myungro	Korea Power Engineering Company	Korea
KIM	Myungsu	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
KIM	See Darl	Korea Atomic Energy Research Institute	Korea
KIM	Se-Won	Korea Institute of Nuclear Safety	Korea
KIM	Tae Woon	Korea Atomic Energy Research Institute	Korea
KIM	Tae-hyeong	Korea Institute of Nuclear Safety	Korea
KIM	Tae-jin	Seoul National Univ.	Korea
KONDO	Shunsuke	Tokyo Univ.	Japan
LEE	Bang Jin	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
LEE	Byung Sik	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
LEE	Chang-Ju	Korea Institute of Nuclear Safety	Korea
LEE	Jong-In	Korea Institute of Nuclear Safety	Korea
LEE	Jung-Jae	Korea Institute of Nuclear Safety	Korea
LEE	Kwang-Nam	Korea Power Engineering Company	Korea
LEE	Seung Jun	Korea Atomic Energy Research Institute	Korea
LEE	Un-Chul	Seoul National Univ.	Korea
LEE	Yong Suk	Future&Challenge Technology Co., Ltd.	Korea
LIM	Hak-Kyu	Korea Power Engineering Company	Korea
LIM	Ho-Gon	Korea Atomic Energy Research Institute	Korea
LIU	Tao	INET, Tsinghua University	China
LO	Chung-Kung	Institute of Nuclear Energy Research	Taiwan
OGURA	Katsunori	Japan Nuclear Energy Safety Organization	Japan
OH	Seong Jong	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
OH	Hae-Cheol	Korea Electric Power Research Institute	Korea
PANG	Whan-Ki	Korea Hydro & Nuclear Power Co.,Ltd.	Korea
PARK	Goon-Cherl	Seoul National Univ.	Korea
PARK	Jin-Hee	Korea Atomic Energy Research Institute	Korea

Family Name	Given Name(s)	Organization	Country
PARK	Jong-Hyuck	Korea Electric Power Research Institute	Korea
PARK	Rae-Joon	Korea Atomic Energy Research Institute	Korea
RHEE	Hyun Me	Korea Atomic Energy Research Institute	Korea
SATO	Eisuke	TEPCO SYSTEMS Corp.	Japan
SEO	Mi-Ro	Korea Electric Power Research Institute	Korea
SONG	JinHo	Korea Atomic Energy Research Institute	Korea
SUH	Namduk	Korea Institute of Nuclear Safety	Korea
Sung	Key Yong	Korea Institute of Nuclear Safety	korea
TAKAHARA	Shogo	Japan Atomic Energy Agency	Japan
TAMAKI	Hitoshi	Japan Atomic Energy Agency	Japan
UCHIDA	Tsuyoshi	Japan Nuclear Energy Safety Organization	Japan
UEDA	Yoshinori	Japan Nuclear Energy Safety Organization	Japan
WOO	Tae-Ho	Seoul National Univ.	Korea
YAMAGUCHI	Akira	Osaka Univ.	Japan
YANG	Huichang	ENESYS Co., Ltd.	Korea
YANG	Joon-Eon	Korea Atomic Energy Research Institute	Korea
YOON	Won Hyo	Korea Institute of Nuclear Safety	Korea
YOU	young-woo	en2t Co	Korea
YU	Yu	INET, Tsinghua University	China

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国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位	
	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表２．基本単位を用いて表されるSI組立単位の例

組立量	SI 基本単位	
	名称	記号
面積	平方メートル	m^2
体積	立法メートル	m^3
速さ，速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s^2
波数	毎メートル	m^{-1}
密度，質量密度	キログラム毎立方メートル	kg/m^3
面積密度	キログラム毎平方メートル	kg/m^2
比体積	立方メートル毎キログラム	m^3/kg
電流密度	アンペア毎平方メートル	A/m^2
磁界の強さ	アンペア毎メートル	A/m
量濃度(a)，濃度	モル毎立方メートル	mol/m^3
質量濃度	キログラム毎立法メートル	kg/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率(b)	（数字の） 1	1
比透磁率(b)	（数字の） 1	1

(a) 量濃度（amount concentration）は臨床化学の分野では物質濃度（substance concentration）ともよばれる。
(b) これらは無次元量あるいは次元１をもつ量であるが、そのことを表す単位記号である数字の１は通常は表記しない。

表３．固有の名称と記号で表されるSI組立単位

組立量	SI 組立単位			
	名称	記号	他のSI単位による表し方	SI基本単位による表し方
平面角	ラジアン(b)	rad	1 (b)	m/m
立体角	ステラジアン(b)	sr(c)	1 (b)	m^2/m^2
周波数	ヘルツ(d)	Hz		s^{-1}
力	ニュートン	N		$m\ kg\ s^{-2}$
圧力，応力	パスカル	Pa	N/m^2	$m^{-1}\ kg\ s^{-2}$
エネルギー，仕事，熱量	ジュール	J	N m	$m^2\ kg\ s^{-2}$
仕事率，工率，放射束	ワット	W	J/s	$m^2\ kg\ s^{-3}$
電荷，電気量	クーロン	C		s A
電位差（電圧），起電力	ボルト	V	W/A	$m^2\ kg\ s^{-3}\ A^{-1}$
静電容量	ファラド	F	C/V	$m^{-2}\ kg^{-1}\ s^4\ A^2$
電気抵抗	オーム	Ω	V/A	$m^2\ kg\ s^{-3}\ A^{-2}$
コンダクタンス	ジーメンス	S	A/V	$m^{-2}\ kg^{-1}\ s^3\ A^2$
磁束	ウエーバ	Wb	Vs	$m^2\ kg\ s^{-2}\ A^{-1}$
磁束密度	テスラ	T	Wb/m^2	$kg\ s^{-2}\ A^{-1}$
インダクタンス	ヘンリー	H	Wb/A	$m^2\ kg\ s^{-2}\ A^{-2}$
セルシウス温度	セルシウス度(e)	℃		K
光束	ルーメン	lm	cd sr(c)	cd
照度	ルクス	lx	lm/m^2	$m^{-2}\ cd$
放射性核種の放射能(f)	ベクレル(d)	Bq		s^{-1}
吸収線量, 比エネルギー分与, カーマ	グレイ	Gy	J/kg	$m^2\ s^{-2}$
線量当量, 周辺線量当量, 方向性線量当量, 個人線量当量	シーベルト(g)	Sv	J/kg	$m^2\ s^{-2}$
酸素活性	カタール	kat		$s^{-1}\ mol$

(a)SI接頭語は固有の名称と記号を持つ組立単位と組み合わせても使用できる。しかし接頭語を付した単位はもはやコヒーレントではない。
(b)ラジアンとステラジアンは数字の１に対する単位の特別な名称で、量についての情報をつたえるために使われる。実際には、使用する時には記号rad及びsrが用いられるが、習慣として組立単位としての記号である数字の１は明示されない。
(c)測光学ではステラジアンという名称と記号srを単位の表し方の中に、そのまま維持している。
(d)ヘルツは周期現象についてのみ、ベクレルは放射性核種の統計的過程についてのみ使用される。
(e)セルシウス度はケルビンの特別な名称で、セルシウス温度を表すために使用される。セルシウス度とケルビンの単位の大きさは同一である。したがって、温度差や温度間隔を表す数値はどちらの単位で表しても同じである。
(f)放射性核種の放射能（activity referred to a radionuclide）は、しばしば誤った用語で"radioactivity"と記される。
(g)単位シーベルト（PV,2002,70,205）についてはCIPM勧告2（CI-2002）を参照。

表４．単位の中に固有の名称と記号を含むSI組立単位の例

組立量	SI 組立単位		
	名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa s	$m^{-1}\ kg\ s^{-1}$
力のモーメント	ニュートンメートル	N m	$m^2\ kg\ s^{-2}$
表面張力	ニュートン毎メートル	N/m	$kg\ s^{-2}$
角速度	ラジアン毎秒	rad/s	$m\ m^{-1}\ s^{-1}=s^{-1}$
角加速度	ラジアン毎秒毎秒	rad/s^2	$m\ m^{-1}\ s^{-2}=s^{-2}$
熱流密度，放射照度	ワット毎平方メートル	W/m^2	$kg\ s^{-3}$
熱容量，エントロピー	ジュール毎ケルビン	J/K	$m^2\ kg\ s^{-2}\ K^{-1}$
比熱容量，比エントロピー	ジュール毎キログラム毎ケルビン	J/(kg K)	$m^2\ s^{-2}\ K^{-1}$
比エネルギー	ジュール毎キログラム	J/kg	$m^2\ s^{-2}$
熱伝導率	ワット毎メートル毎ケルビン	W/(m K)	$m\ kg\ s^{-3}\ K^{-1}$
体積エネルギー	ジュール毎立方メートル	J/m^3	$m^{-1}\ kg\ s^{-2}$
電界の強さ	ボルト毎メートル	V/m	$m\ kg\ s^{-3}\ A^{-1}$
電荷密度	クーロン毎立方メートル	C/m^3	$m^{-3}\ sA$
表面電荷	クーロン毎平方メートル	C/m^2	$m^{-2}\ sA$
電束密度，電気変位	クーロン毎平方メートル	C/m^2	$m^{-2}\ sA$
誘電率	ファラド毎メートル	F/m	$m^{-3}\ kg^{-1}\ s^4\ A^2$
透磁率	ヘンリー毎メートル	H/m	$m\ kg\ s^{-2}\ A^{-2}$
モルエネルギー	ジュール毎モル	J/mol	$m^2\ kg\ s^{-2}\ mol^{-1}$
モルエントロピー, モル熱容量	ジュール毎モル毎ケルビン	J/(mol K)	$m^2\ kg\ s^{-2}\ K^{-1}\ mol^{-1}$
照射線量（Ｘ線及びγ線）	クーロン毎キログラム	C/kg	$kg^{-1}\ sA$
吸収線量率	グレイ毎秒	Gy/s	$m^2\ s^{-3}$
放射強度	ワット毎ステラジアン	W/sr	$m^4\ m^{-2}\ kg\ s^{-3}=m^2\ kg\ s^{-3}$
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2\ sr)$	$m^2\ m^{-2}\ kg\ s^{-3}=kg\ s^{-3}$
酵素活性濃度	カタール毎立方メートル	kat/m^3	$m^{-3}\ s^{-1}\ mol$

表５．SI 接頭語

乗数	接頭語	記号	乗数	接頭語	記号
10^{24}	ヨタ	Y	10^{-1}	デシ	d
10^{21}	ゼタ	Z	10^{-2}	センチ	c
10^{18}	エクサ	E	10^{-3}	ミリ	m
10^{15}	ペタ	P	10^{-6}	マイクロ	µ
10^{12}	テラ	T	10^{-9}	ナノ	n
10^{9}	ギガ	G	10^{-12}	ピコ	p
10^{6}	メガ	M	10^{-15}	フェムト	f
10^{3}	キロ	k	10^{-18}	アト	a
10^{2}	ヘクト	h	10^{-21}	ゼプト	z
10^{1}	デカ	da	10^{-24}	ヨクト	y

表６．SIに属さないが、SIと併用される単位

名称	記号	SI 単位による値
分	min	1 min=60s
時	h	1h =60 min=3600 s
日	d	1 d=24 h=86 400 s
度	°	1°=(π/180) rad
分	’	1’=(1/60)°=(π/10800) rad
秒	”	1”=(1/60)’=(π/648000) rad
ヘクタール	ha	$1ha=1hm^2=10^4m^2$
リットル	L，l	$1L=1l=1dm^3=10^3cm^3=10^{-3}m^3$
トン	t	$1t=10^3$ kg

表７．SIに属さないが、SIと併用される単位で、SI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位で表される数値
電子ボルト	eV	$1eV=1.602\ 176\ 53(14)\times10^{-19}J$
ダルトン	Da	$1Da=1.660\ 538\ 86(28)\times10^{-27}kg$
統一原子質量単位	u	1u=1 Da
天文単位	ua	$1ua=1.495\ 978\ 706\ 91(6)\times10^{11}m$

表８．SIに属さないが、SIと併用されるその他の単位

名称	記号	SI 単位で表される数値
バール	bar	$1\ bar=0.1MPa=100kPa=10^5Pa$
水銀柱ミリメートル	mmHg	1mmHg=133.322Pa
オングストローム	Å	$1\ Å=0.1nm=100pm=10^{-10}m$
海里	M	1 M=1852m
バーン	b	$1\ b=100fm^2=(10^{-12}cm)2=10^{-28}m^2$
ノット	kn	1 kn=(1852/3600)m/s
ネーパ	Np	SI単位との数値的な関係は、対数量の定義に依存。
ベル	B	
デジベル	dB	

表９．固有の名称をもつCGS組立単位

名称	記号	SI 単位で表される数値
エルグ	erg	$1\ erg=10^{-7}\ J$
ダイン	dyn	$1\ dyn=10^{-5}N$
ポアズ	P	$1\ P=1\ dyn\ s\ cm^{-2}=0.1Pa\ s$
ストークス	St	$1\ St\ =1cm^2\ s^{-1}=10^{-4}m^2\ s^{-1}$
スチルブ	sb	$1\ sb\ =1cd\ cm^{-2}=10^4cd\ m^{-2}$
フォト	ph	$1\ ph=1cd\ sr\ cm^{-2}\ 10^4lx$
ガル	Gal	$1\ Gal\ =1cm\ s^{-2}=10^{-2}ms^{-2}$
マクスウェル	Mx	$1\ Mx = 1G\ cm^2=10^{-8}Wb$
ガウス	G	$1\ G\ =1Mx\ cm^{-2}\ =10^{-4}T$
エルステッド(c)	Oe	$1\ Oe \triangleq (10^3/4\pi)A\ m^{-1}$

(c) ３元系のCGS単位系とSIでは直接比較できないため、等号「 ≙ 」は対応関係を示すものである。

表10．SIに属さないその他の単位の例

名称	記号	SI 単位で表される数値
キュリー	Ci	$1\ Ci=3.7\times10^{10}Bq$
レントゲン	R	$1\ R = 2.58\times10^{-4}C/kg$
ラド	rad	$1\ rad=1cGy=10^{-2}Gy$
レム	rem	$1\ rem=1\ cSv=10^{-2}Sv$
ガンマ	γ	1γ=1 nT=10-9T
フェルミ		1フェルミ=1 fm=10-15m
メートル系カラット		1メートル系カラット = 200 mg = 2×10-4kg
トル	Torr	1 Torr = (101 325/760) Pa
標準大気圧	atm	1 atm = 101 325 Pa
カロリー	cal	1cal=4.1858J（「15℃」カロリー），4.1868J（「IT」カロリー）4.184J（「熱化学」カロリー）
ミクロン	µ	$1\ µ\ =1µm=10^{-6}m$

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