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Proceedings of the 10th Korea-Japan Joint Workshop on PSA

–For Asian PSA Network–

May 18-20, 2009, Haevichi Hotel & Resort, Jeju, Korea

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The tenth Korea-Japan Joint Workshop on Probabilistic Safety Assessment (PSA) was held in the Jeju island of Korea, on May 18-20, 2009 organized by Korea Atomic Energy Research Institute (KAERI). The purpose of the workshop was to provide a forum for presentation and discussions on experiences and technical achievements related to PSA, risk-informed and performance-based approach, and other relevant issues in both countries.

Since the first Korea-Japan Joint Workshop on PSA started in 1992, the workshops have provided an important and timely opportunity for exchange and discussion of the relevant information to all PSA practitioners and users of risk information in the industry, research, academia and regulatory arena. This was the tenth anniversary of the Joint Workshop with the main theme of “For Asian PSA Network” and participants included those from China, Taiwan and the United States of America besides Korea and Japan.

Two keynote speeches were presented by the former chairmen of this workshop, Prof. Chang-Sun Kang of Seoul National University and Prof. emeritus Shunsuke Kondo of Tokyo University. We had two special lectures, 70 papers presented by experts at 10 technical sessions related PSA, the special session on the status of PSA in Korea, Japan, China and Taiwan and panel discussion on their cooperation in PSA. This report provides the summary of each session, and all the presentation materials presented in the 10th Korea-Japan Joint Workshop on PSA.

Keywords: PSA, Risk Informed Regulation, Severe Accident, Reactor Safety

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* Korea Atomic Energy Research Institute (KAERI)

第 10 回日韓 PSA ワークショップ報文集
－アジアに向けた PSA ネットワーク－
2009 年 5 月 18 日～20 日、ヘビチホテル&リゾート、済州島、韓国

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第 10 回日韓確率論的安全評価（PSA）ワークショップは、韓国原子力研究所の主催で 2009 年 5 月 18 日から 20 日に韓国の済州島で開催された。ワークショップの目的は、PSA、リスク情報を考慮し性能に基づくアプローチ、及び他の関連課題に関する両国の経験や技術的到達点についての発表と議論のフォーラムを提供することにあった。

1992 年に第 1 回日韓 PSA ワークショップが開始されて以後、本ワークショップは産業界、研究機関、大学及び規制機関の PSA 実務者やリスク情報の利用者に、その時々に PSA 関連の情報交換と議論の重要な機会を提供してきた。今回は、 “アジアに向けた PSA ネットワーク”をテーマに第 10 回の記念すべき共同ワークショップとなり、韓国、日本両国の他、中国、台湾、米国からも参加者があった。

本ワークショップの前共同議長である Chang-Sun Kang ソウル大学教授及び近藤駿介東京大学名誉教授による基調講演が企画された。2 つの特別講演の他、PSA に関連した 10 の技術セッションに 70 件の発表があり、両国の他、中国及び台湾における PSA の現状に関する特別セッションと PSA における協力についてパネルディスカッションが設けられた。本報告書には、各セッションのまとめと共に、基調講演及び特別講演を含む全発表の資料を掲載した。

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Summary of Opening Session

Opening remarks (Un-Chul LEE)
Prof. Lee introduces participants from China and Taiwan. Prof. Lee emphasized the role of Korea-Japan Joint Workshop on PSA (KJPSA) in the cooperation between Korea and Japan in the PSA area. As the slogan "For Asian PSA Network" says, Prof. Lee suggests close collaboration among Asian countries to improve the PSA technology, especially in the newly expanded area of PSA such as risk-based technology and risk-informed regulation, application, and design. Prof. Lee wishes every participant has good time during the workshop.

Welcome Address (Toshimitsu HOMMA)
On behalf of all Japanese participants, Dr. Homma expresses gratitude to the organizers in Korean side (KAERI, KINS, KHNP, KEPRI, and KOPEC). Dr. Homma mentioned that the organizing committee has done wonderful work in the preparation of the workshop. Dr. Homma expressed special thanks to general secretary Dr. Yang and Mr. Han and Dr. Choi. Dr. Homma introduces Dr. Kajimoto of JNES for giving the message from Prof. Hirano, one of the co-chair of the workshop.

Welcome Address (Message from Prof. Mitsumasa HIRANO, read by Dr. Mitsuhiro KAJIMOTO)
Prof. Hirano says he is sorry for not being able to attend the workshop due to the new flu. Prof. Hirano introduces the PSA-related activities of Japanese Nuclear Safety Commission. Prof. Hirano also introduces the development of a standard for the PSA technology by Atomic Society of Japan. Prof. Hirano mentions the importance of the PSA technology for the introduction of risk-informed regulation. Prof. Hirano emphasizes the importance of continued effort in the advancement of the PSA technology and the sharing of information, and the contribution of the workshop for those purposes.

Congratulatory Address (Dr. Jong-In LEE)
Dr. Lee expresses his gratitude to the honorary chairs and the chairs of the workshop. Dr. Lee says that he is a member of the workshop and he also participated previous workshops several times. Dr. Lee mentions the importance the PSA technology in the practical application of risk-informed regulation. Dr. Lee says that he expects the workshop to provide important and timely discussions to all PSA and RIR-related people. Dr. Lee also says that he hope all the participants have constructive discussions on the establishment of the PSA network in Asia.

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Keynote Speech

Chang-Sun KANG
Shunsuke KONDO

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The 10th Korea-Japan Joint Workshop on PSA

Current Issues and Challenges on Nuclear Safety

May 18, 2009
C-S. Kang
Professor Emeritus, Seoul National University
Member, INSAG-IAEA
Special Advisor to the President, KAERI

0

Contents of Presentation

- ❖ INSAG Activities
- ❖ Challenges to Current and Emerging Key Safety Issues
  - ✓ Global Nuclear Safety Regime
  - ✓ Operational Nuclear Safety
  - ✓ Relationship between Nuclear Safety and Security
  - ✓ PSA Update

1

What is INSAG?

➡1985-2003: International Nuclear Safety Advisory Group (INSAG)

❖ Advising DG of IAEA on Nuclear Safety

➡2003-date: International Nuclear Safety Group (INSAG)

❖Independent from IAEA-DG

- INSAG will provide authoritative advice and guidance on nuclear safety approaches, policies and principles.
- INSAG will provide recommendations and opinions on *current and emerging nuclear safety issues* to the IAEA, the nuclear community and the public.

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17 Members of INSAG-VII (2007-2009)

❖ **Asmolov**,	Director, Kurchatov Institute,	Russian Federation
❖ **Alonso**,	Chair of Nuclear Technology, Politechnical University,	Spain
❖ **Echávarri**,	Director-General,	OECD/NEA
❖ **Kang**,	Nuclear Engineering Department, Seoul National University,	Korea
❖ **Laaksonen**,	Director General, Radiation and Nuclear Safety Authority (STUK),	Finland
❖ **Meserve**,	President, Carnegie Institute of Washington,	U.S.A.
❖ **Sharma**,	Chairman, Atomic Energy Regulatory Board,	India
❖ **Torgerson**,	President, AECL,	Canada
❖ **Birkhofer**,	ISaR Institute for Safety and Reliability GmbH,	Germany
❖ **Lauvergeon**,	Chairperson, AREVA,	France
❖ **Chang**,	General Manager, China Power Investment Corporation,	China
❖ **Drabova**,	President, State Office for Nuclear Safety (SUJB),	Czech Republic
❖ **Rising**,	Vice-President, Vattenfall Ab,	Sweden
❖ **Weightman**,	HM Chief Inspector, Nuclear Safety Directorate, HSE,	UK
❖ **Suzuki**,	Chairperson, Nuclear Safety Commission,	Japan
❖ **Couto**,	Head of Nuclear Regulation and Licensing,	Argentine
❖ **Slabber**,	PBMR Technical Director,	South Africa

3

Documents Produced by INSAG

75-INSAG-1	Summary report on the post-accident review meeting on the Chernobyl accident	1986
75-INSAG-2	Radionuclide source terms from severe accidents to nuclear power plants with light water reactors	1987
75-INSAG-3	Basic safety principles for nuclear power plants	1988
75-INSAG-4	Safety culture	1991
75-INSAG-5	The safety of nuclear power	1992
75-INSAG-6	Probabilistic safety assessment	1992
75-INSAG-7	The Chernobyl accident: Updating of INSAG-1	1993
INSAG-8	A common basis for judging the safety of nuclear power plants built to earlier standards	1995
INSAG-9	Potential exposure in nuclear safety	1995
INSAG-10	Defense in depth in nuclear safety	1996
INSAG-11	The safe management of sources of radiation: Principles and strategies	1999
INSAG-12	Basic safety principles for nuclear power plants 75-INSAG-3 Rev. 1	1999
INSAG-13	Management of operational safety in nuclear power plants	1999
INSAG-14	Safe management of the operating lifetimes of nuclear power plants	1999
INSAG-15	Key practical issues in strengthening safety culture	2002
INSAG-16	Maintaining knowledge, training and infrastructure for research and development in nuclear safety	2003
INSAG-17	Independence in regulatory decision making	2003
INSAG-18	Managing change in the nuclear industry: The effects on safety	2003
INSAG-19	Maintaining the design integrity of nuclear installations throughout their operating life	2003
INSAG-20	Stakeholder involvement on nuclear issues	2006
INSAG-21	Strengthening the global nuclear safety regime	2006
INSAG-22	Nuclear safety infrastructure for a national nuclear power programme supported by the IAEA Fundamental Safety Principles	2008
INSAG-23	Improving the international system for operating experience feedback	2008
INSAG-24	Relationship between safety and security (in the final draft form)	2009
INSAG-25(?) probably on PSA update		

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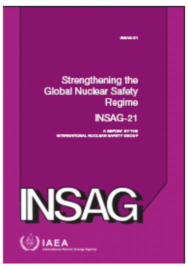
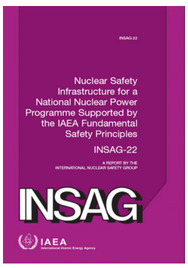
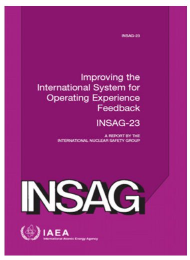
Key Safety Issues under Discussion

- Global Nuclear Safety Regime: INSAG-21 and 22
- Operational Nuclear Safety: INSAG-23
- Relationship between Nuclear Safety and Security: INSAG-24 (Draft)
- PSA Update: Potential INSAG-25 (?)







5

Challenges to Key Safety Issue: "Strengthening Global Safety Regime"

- Global Harmonization of Nuclear Safety
- Establishment of Nuclear Safety Infrastructure for Newcomers
- Trade Practice between Exporter and Importer of Nuclear Systems
- More Legally Binding Mechanism for Global Safety
- International or Multinational Safety Review (ex. MDEP)

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Challenges to Key Safety Issue: "Enhancing Operational Nuclear Safety"

- **Operating Experience Feedback**
  - **Comprehensive analysis of events**
  - **Proper dissemination of results**
- **Share of knowledge**
  - **Up-to-date safety related R&D results and operating experiences**
- **Complacency**
  - **Continuing investments in staff, systems and equipment**
- **Life Extension and Power Ascension**
  - **Ageing and Safety Margins**
- **Reliance on Contractors**
  - **Operator's responsibility of controlling contractors**
- **Safety Culture**
  - **Leadership and Management**
  - **Operators**
  - **Independent External Review**

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Challenges to Key Safety Issue: "Developing the Relationship Between Nuclear Safety and Security"

Security: ***Terrorists' Attack (9/11/01)***

relates to the prevention, detection and response to malevolent acts, theft and sabotage which could ***lead to accidents or threats of causing accidents***.

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What is Nuclear Safety?
What is Nuclear Security?

- ***Nuclear Safety*** - "the achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards".

- ***Nuclear Security*** - "the prevention and detection of and response to theft, sabotage, unauthorized access, illegal transfer or other malicious acts involving nuclear material, other radioactive substances or associated facilities".

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Nuclear Security

- Nuclear security has been focused on:
  - nuclear weapons,
  - NPT regime and
  - disarmament.
- New dimensions
  - Post cold war
  - Post 9/11
- ***No clear distinction among:***
  - *safeguards,*
  - *safety, and*
  - *security.*

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Security: Changes and Challenges - Post Cold War, Post 9/11 -

Main Players
- Nation-states
- Bi-polar superpowers
- Gov-industry-military complex

- **Sub-national, Non-states, small states**
- **Global network**
- **Mass media and public**

Threats
- High density, high intensity, centralized
- Lower Probability
- Physical overkill

- **Low density, low intensity, decentralized**
- **Higher probability**
- **Socio-psychological terror**

Motives
- Geopolitical
- Predictable - calculable

- **Malevolent**
- **Unpredictable and Incalculable**

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Security: World Inventories of Nuclear Facilities and Materials

- Pu >1,670 tons civil, 155 tons military
- HEU >175 tons civil, >1,720 tons military
- 442 operating nuclear power plants in 31 States
  - 270,000 tons of spent fuel produced (1,800 tons Pu)
- 248 operating research reactors (>100 with HEU)
- 240 shut down research reactors (several 10 with HEU)
- 18 conversion plants
- 40 fuel fabrication plants
- 7 reprocessing plants
- 13 enrichment plants
- 89 storage facilities
- > 100,000 Category I and II radioactive sources
- > 1,000,000 Category III radioactive sources

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INSAG-24: "Relationship Between Nuclear Safety and Security"

Backgrounds:

- Recent ***terrorist events*** catalyst for the development of international nuclear security legal instruments to address this increased threat.
- Nuclear safety and nuclear security have a common purpose "***the protection of people, society and the environment***" and many common principles, although their implementation may differ.
- Potential synergies, but also need to manage the impact of one discipline on the other to avoid ***potential conflicts***.
- The purpose of the report is to provide a better understanding of nuclear safety and nuclear security ***interfaces*** and the ways to take them into proper account.

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INSAG-24: *Arrangements for Safety & Security*

- ***Legislative and regulatory framework*** set up by the State
- ***Responsibility of the State***:
  - designate competent authorities
  - define rules for confidentiality and information protection and carries out checks on individuals
  - continuously assess the threat
  - define the design basis threat
- ***Responsibility of operators***
  - take prime responsibility for the installation and this responsibility cannot be delegated.
  - be in the best position to identify the risks and to ensure compliance with regulatory requirements.
- In case of emergencies, ***shared responsibilities*** between the operating organization (on-site) and the state/competent authorities (off-site).
- Management of a crisis linked with malicious acts demands a greater ***number of State bodies*** than managing a crisis purely dependent on safety.

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INSAG-24: Common Basic Principles & Potential Divergences

- **Leadership and management**
  - Similar principles for safety culture and security culture
  - Higher involvement of the State to define security measures
  - Differences in handling of information: transparency vs. confidentiality
  - Individuals of diverse backgrounds and experience for security
- **Optimization of protection**
  - Assessment of the risk, using a graded approach
  - Permanent safety and security systems
  - Improved dispositions of feedback (techniques, experience, threats)
- **Prevention of event occurrences**
  - Defense in depth for safety described in INSAG 10,
  - Defense in depth for security in the Amended CPPNM and INFCIRC/225
  - Close cooperation between security & safety specialists
- **Emergency preparedness and response**
  - Complementarity between contingency and emergency plans
  - Safety measures under the operator's responsibility, while security measures under the State responsibility
  - Performance of joint exercises

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Challenges to Key Safety Issue: "PSA Update"

Backgrounds:

- **INSAG-6** ("Probabilistic Safety Assessment")
  - provides general views on the PSA performance and applications
  - discusses benefits and cautions for the potential users of the PSA
  - highlights areas of PSA that require deeper elaboration
- **INSAG-12** ("Basic Safety Principles for Nuclear Power Plants")
  - probabilistic safety criteria were proposed, and suggests the core damage frequency and frequency of large off-site release for operating plants and plants under design
  - Both INSAG-6 and INSAG-12 encourage consistent performance and application of PSA as one of the safety assessment tools.
- **New Safety Guides on PSA**:
  - DS394 (Performance and Application of Level 1 PSA)
  - DS393 (Performance and Application of Level 2 PSA)
    - Recommendations are based on current good practices.
    - Current trends, challenges and problematic areas of PSA are not included. (ex. Risk-informed decision making, probabilistic safety goals, new methodological challenges, etc.)

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PSA: Current Status

- **Scope:** full scope Level-2 PSA.
- **Quality:** extremely increased.
- **Data:** combination of plant-specific and generic data
- **Role:** wide applications in safety analysis, design, operation, licensing, etc.
- **Review:** independent review as an integrated part of the PSA process
- **Safety Goals:** numerical goals defined in many countries, but no international consensus.
- **Methodology:** reasonable procedures, powerful codes for PSA, but problems in assessing low probable events and ageing phenomena

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PSA: New Areas for Further Development

- Consideration of Ageing Effects
- Reliability of Passive Systems, Software and Digital Systems
- Internal and External Hazards PSA
- Consideration of Extreme Events and Global Warming
- Consideration of the Uncertainties and RIDM
- Probabilistic Safety Goals and Acceptance Criteria
- PSA Knowledge Management and Succession Planning
- International Cooperation and Communication on PSA

*Reference: New challenges and emerging issues in risk assessment for nuclear power plants, IAEA, NSNI/SAS/2009/1/March 18

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PSA: New Areas for Further Development (1/3)

- ***Consideration of Ageing Effects***
  Danger of underestimation
- ***Reliability of Passive Systems, Software and Digital Systems***
  Need for comprehensive reliability PSA models including the realistic assessment of the risk
- ***Internal and External Hazards PSA***
  Further maturity of PSA methodology with the emphasis on external hazards PSA
  - Hazards frequency assessment
  - Hazards impact to passive systems and computer-based digital systems
- ***Consideration of Extreme Events and Global Warming***
  - Natural phenomena: earthquake, tsunamis and seiches, hurricanes, cyclones and typhoons, floods, tornadoes
- ***Consideration of the Uncertainties and RIDM***
  - Advantages of quantitative estimates for the uncertainties
  - Comprehensive uncertainties estimates in the decision making process

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PSA: New Areas for Further Development (2/3)

- **Probabilistic Safety Goals and Acceptance Criteria**
  No consensus in Member States on the use of Probabilistic Safety Goals (PSG)
  - *Probabilistic safety criteria and objectives:*
    - ***Core damage frequency, large early release frequency, large release frequency, health effects, etc.?***
  - *Other important related questions require consensus on:*
    - ***What should be compared with the criteria/ target? Mean, median, 95% bound of risk estimates, etc.?***
    - ***Are numerical risk limits: Formal requirements or orientation values/targets?***
    - ***Is the scope of the PSA required to meet the criteria/objectives: Internal initiating events, internal and external hazards, operation modes, etc.?***
    - ***Is the generic PSG in terms of frequency (f) of doses (D) to an individual resident at a nuclear site boundary?***
    - ***Is the trend to develop criteria for risks to people: consensus on the specific notions of the risk "tolerability" and "de minimis" limits?***
  - *Comprehensive PSGs need to be defined.*
    - ***What are the objectives of the Safety Goals?***
    - ***Should not only specify numbers?***
    - ***Should be formulated in a way encouraging further safety enhancement even when they are met?***

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PSA: New Areas for Further Development (3/3)

- **PSA Knowledge Management and Succession Planning**
  - Training on PSA and succession
  - Experienced PSA analysts approaching the retirement age
  - Young nuclear engineers to get on-job training.
  - Development of recommendations for knowledge management and succession planning.
  - PSA documentation for appropriate knowledge management and smooth succession
- **International Cooperation and Communication on PSA**
  - Need for improved communication of PSA results and risk insights
  - Not limited to nuclear engineers and scientists: **Scientists in other fields, and the general public**
  - Wider international co-operation and co-ordination

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Summary

Current Issues and Challenges on Nuclear Safety

- Strengthen Global Nuclear Safety Regime
- Operational Nuclear Safety
- Relationship between Nuclear Safety and Security
- PSA Update

PSA: New Areas for Further Development

- Consideration of Ageing Effects
- Reliability of Passive Systems, Software and Digital Systems
- Internal and External Hazards PSA
- Consideration of Extreme Events and Global Warming
- Consideration of the Uncertainties and RIDM
- Probabilistic Safety Goals and Acceptance Criteria
- PSA Knowledge Management and Succession Planning
- International Cooperation and Communication on PSA

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Actions Necessary to Promote Nuclear Energy Utilization for Solving Global Problems We Face

Shunsuke Kondo
Chairman
Japan Atomic Energy Commission

Japan-Korea PSA Workshop
May 18, 2009

Global Energy Problems

- Global demand for energy will continue to grow at considerable pace as there are 1.6 billion people who have not access to electricity and the 2.4 billion who have no access to modern energy systems.
- It is necessary to solve the problems of climate change, energy security and the volatility of the price of oil and gas.
- Japan and other major developed countries are committing to reduce greenhouse-gas (GHG) emissions to 50 % of the current level in the first half of this century.
- This means that the global GHG emissions should be 13 GtC/yr below that of business-as-usual case in 2050.

Global Energy Solutions

- Achieving this target while increasing the supply of energy requires the global community to make utmost effort to deploy not only energy conservation and high-efficiency energy technologies but also non-GHG emitting energy production technologies such as renewable, nuclear, and carbon-sequestration technologies, on a gigantic scale.
- Example: to avoid even 1/10 of the target or 1.3 GtC/yr, it is necessary to replace 900 GWe coal fired plants with nuclear power plants.
- However, there is a large uncertainty even in the future of nuclear energy: the OECD Nuclear Energy Agency has projected that global nuclear power capacity in 2050 will be between 580 and 1400 GWe.

Nuclear Energy Vision

Nuclear energy will contribute as one of the mainstay technologies for electricity and heat generation to the fostering of economic growth/poverty eradication, energy security and low-carbon economy in many parts of the world.

Objectives for Global Nuclear Community

I. Sustain safe and efficient operation of nuclear power plants, installing new plants that are necessary to satisfy the need for electricity/GHG emission reduction and managing used fuel in appropriate ways:

II. Shape environment for facilitating the peaceful uses of nuclear energy in every part of the world:

III. Realize competitive and more sustainable nuclear energy technology through unremitting R&D activities.

To sustain stable operation of nuclear power plants and install new capacity;

- Maintain the public trust in both plant operator's safety management and government's regulatory activities for nuclear safety, security and nonproliferation through the promotion of
  - Open and transparent risk communication with the public unremittingly.
  - Steady business risk management activities, carefully considering lessons learned from operating experiences worldwide, new developments in science and technology, and changes in organizational culture and business environment that can have negative influences on the safe operation of the plant.
- Make it possible to deliver safe disposal of radioactive wastes.
- Prepare and execute plant ageing management activities to ensure their high capacity factor and superior safety and economic performance throughout their life of 60 years at least;
- Assure market force to continue to drive the construction of nuclear power plants that are necessary for satisfying the anticipated need for electricity/GHG emission reduction.

The 16 July 2007 Earthquake at Kashiwazaki-Kariwa NPP of TEPCO

- The seismic input to the plant significantly exceeded the level of design-basis seismic input of the plant. Nevertheless, the operating units were automatically shutdown and all plants behaved in a safe manner, during and after the earthquake.
- No significant damage of safety-related structures, systems and components (SSCs) of the plant has been reported, whereas non-safety related SSCs were affected mainly due to significant soil deformation as they were not connected to the bedrock.
- The public confidence in both nuclear safety regulation and operators' safety management was shaken by the intense media attention to the fire of a non-safety-related transformer and the inadvertent release of radioactivity, though the amount was extremely minor.

Lessons Learned from The Seismic Events at the KK NPPs

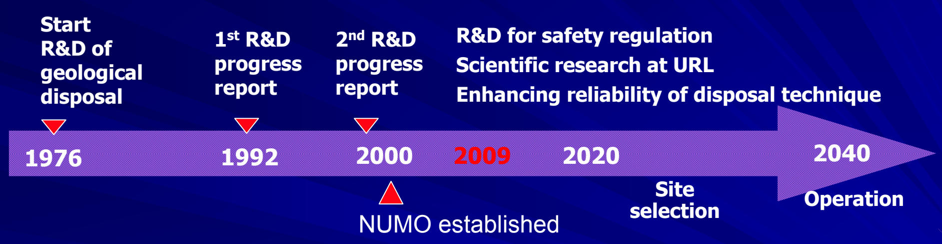
- Need for considering the inclination of a nearby fault toward the plant and the geological structure of the underground of the plant in the evaluation of seismic input from the fault and the flexibility of floor in the evaluation of the response of plant structures to the input.
- Need for reviewing the appropriateness of seismic design of seismic class C structures and components from the viewpoint of business continuity planning.
- Need for paying close attention to the appropriateness of emergency planning and fire-fighting capability in extreme seismic conditions.

Lessons Learned for Risk Management

- Due attention should be paid not only to the sources of knowledge risk but also to the sources of relationship risk and process-engagement risk.
  – Knowledge risk materializes when knowledge base is deficient due to neglect of lessons learned from experiences anywhere and new knowledge in science and technology that have impact upon the risk.
  – Relationship risk appears when ineffective collaboration and communication among functions and/or organizations exist and insufficient knowledge is applied to risk assessment.
  – Process engagement risk arises when faulty operational procedures exist and distorted knowledge is applied in risk assessment.
- It is the task of leadership in enterprise and government administration to name these risks and seek to rise to them.

NUMO: Nuclear Waste Management Organization

Activities of JAEA



Activities of NUMO:

- Encourage communities to apply for suitability review as an area for siting a geological disposal facility of high-level radioactive waste (HLW).
  ~ 2012 Selection of areas for detailed investigation
  ~ 2027 Selection of a site for repository
- Promote technology development for the improvement of safety, economy and efficiency of the geological disposal of HLW.

The Siting of a HLW Disposal Facility

- In 2000, the AEC decided that the activity to decide the site for a HLW disposal facility should be promoted in an open and transparent way and the site should be determined after detailed suitability review of the area of municipalities that apply for invitation.
- In 2004, the NUMO, an organization authorized to promote the disposal activity, started to invite mayors of municipalities to apply for site suitability review.
- However, no mayor has successfully applied so far : even the announcement of a mayor to study the merit and demerit of the application has paralyzed the administrative affairs of the municipal office due to the intense media attention and rallies and demonstrations to protest the announcement.
- The Government as well as the NUMO have started to strengthen public information activities on the possible public support for the sustainable development of the municipalities that locate the site from the view point of equity of benefit, as well as safety and the importance of the disposal facility.

To shape the environment for facilitating the peaceful uses of nuclear energy in every part of the world;

- Build a global consensus that nuclear energy is an essential measure against global warming/poverty eradication;
- Support countries considering the introduction of nuclear power internationally;
- Strengthen the international system for ensuring nuclear safety, security and nonproliferation;

To shape environment for facilitating the peaceful uses of nuclear energy in every part of the world;

- Build a global consensus that nuclear energy is an essential measure against global warming/poverty eradication;
  - Induce to consider the construction of NPP as a clean development mechanism (CDM) project activity of post-Kyoto Protocol framework to be determined;
  - Induce the World Bank to set up innovative financing schemes for NPP construction and that for sea-water desalination in particular.
- Support countries considering the introduction of nuclear power internationally; and
- Strengthen the international system for ensuring nuclear safety, security and nonproliferation;

To shape environment for facilitating the peaceful uses of nuclear energy in every part of the world;

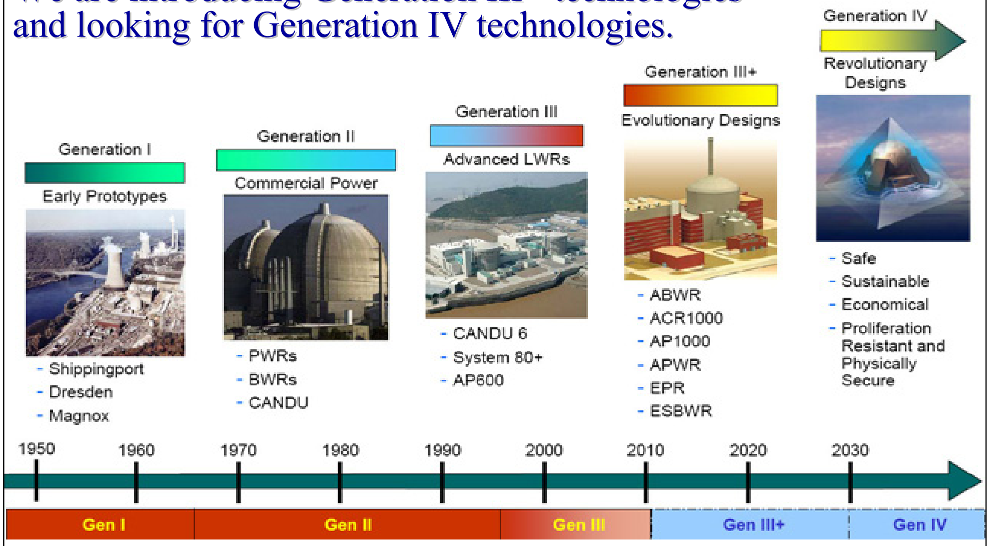
- Build a global consensus that nuclear energy is an essential measure against global warming;
- Support countries considering the introduction of nuclear power internationally;
  - Support the IAEA and strengthen its human and financial resources as it is developing international references and standards, providing guidance, organizing workshops and offering a service to review the progress in infrastructure development.
  - Promote dialogues, provide consultation and services through bilateral and multilateral frameworks such as GNEP and FNCA, recognizing that human resource development and stakeholder engagement are central issues that need urgent attention.
- Strengthen the international system for ensuring nuclear safety, security and nonproliferation;

To shape environment for facilitating the peaceful uses of nuclear energy in every part of the world;

- Build a global consensus that nuclear energy is an essential measure against global warming;
- Support countries considering the introduction of nuclear power internationally;
- Strengthen the international system for ensuring nuclear safety, security and nonproliferation;
  - Make sure to adhere to international conventions related to nuclear safety, nonproliferation and security.
  - Reinforce the IAEA's legal authority in nuclear verification, safety and security, universalizing the Additional Protocol and accepting various IAEA review missions for mutual learning.
  - Actively promote the establishment of credible multilateral nuclear fuel supply assurances, as a complement to the market, with a view to reducing incentive to have a national nuclear fuel cycle facility.

Evolution of Nuclear Energy Technology

We are introducing Generation III+ technologies and looking for Generation IV technologies.



http://www.gen-4.org/Technology/evolution.htm

Pursue Technology Innovation

- Aiming for innovation of LWR technology;
  - Pursue high performance of reactor plant materials, instrumentation and controls, and fuel, and better management of used fuel and waste.
  - Develop innovative LWRs incorporating advances in science and technology; high burn-up fuel, seismic isolation technology, advanced construction technology, advanced information technology and so on.
- Aiming for realization of sustainable nuclear energy technology from the long term perspective, promote R&D of Generation IV nuclear energy systems that have potential to make significant contributions in the future to sustain low-carbon society.
  - Fast reactor and its fuel cycle technology that satisfies the request for enhanced safety, reliability and utilization of fuel, increased proliferation resistance, friendliness to the neighbor, low heat generation rate of radioactive waste;
  - Promising nuclear energy technologies, such as high temperature water-splitting technology and grid appropriate reactors, that contribute to new missions and markets such as sea-water desalination, hydrogen production, district heating etc.

Conclusion

- Nuclear energy is one of the key energy supply sources of the future. It can make a major contribution to the fostering of economic growth/poverty eradication, energy security and low-carbon economy in many parts of the world.
- Global nuclear community should rise to contribute to sustain safe and efficient operation of nuclear power plants, install new plants that are necessary to satisfy the need for electricity/GHG emission reduction and manage used fuel in appropriate ways.
- Safety, security and nuclear safeguards should be ensured at any plant in any country. The community should support states willing to develop a nuclear power program in their efforts to establish required infrastructure, in close cooperation with the IAEA.
- It is also vitally important for the community to pursue to realize sustainable nuclear energy technologies.
- International collaboration is essential to the success in rising to these challenges, not only because the collaboration could reduce the duplication of efforts but also because it could produce better and brighter solutions in our pathway to the goals.

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Special Lectures

Katsumi EBISAWA
Joseph BRAVERMAN

Japan Nuclear Energy Safety Organization (JNES)

The 10th Korea-Japan Joint Workshop on PSA

Reflection of Lesson learned from Niigata-ken Chuetsu-oki (NCO) Earthquake to Seismic PSA

May 18-20, 2009
Jeju Island, Korea

Incorporated Administrative Agency
Japan Nuclear Energy Safety Organization
Katsumi EBISAWA
ebisawa-katsumi@jnes.go.jp

Japan Nuclear Energy Safety Organization (JNES)

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II. Lesson learned from NCO Earthquake
II-1. Amplification of Earthquake Motion
II-2. Flexibility of Building Floor
II-3. Integrity of Component
III. Reflection to Methodology of Seismic PSA
III-1. Seismic Hazard Evaluation
III-2. Fragility Evaluation
III-3. Accident Sequence Evaluation
IV. Conclusions

1

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I. Background

■ The lessons learned from Niigata-ken Chuetsu-oki (NCO) earthquake occurred July, 2007 extremely affected to the nuclear seismic safety.
The main lessons are as follows.

① Why did the observed seismic ground motions far exceed those designed ?
② Why were the response spectra derived from the observed seismic motions different from those of the vibration model of the conventional seismic design ?
③ Why did the safety related functions of shutdown, cooling, and containment maintained effectively ?

The overview of cause investigation will be introduced.

■ The Nuclear Safety Commission of Japan revised the seismic design review guide on 2006. In consideration of residual risk in new guide, Atomic Energy Society of Japan established a seismic PSA implementation standard in 2007.
The above lessons reflected to each evaluation methods of seismic hazard, fragility and accident sequence in seismic PSA.
The overview of reflection will be also introduced.

2

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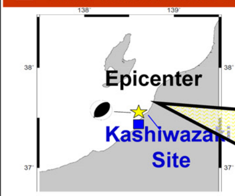
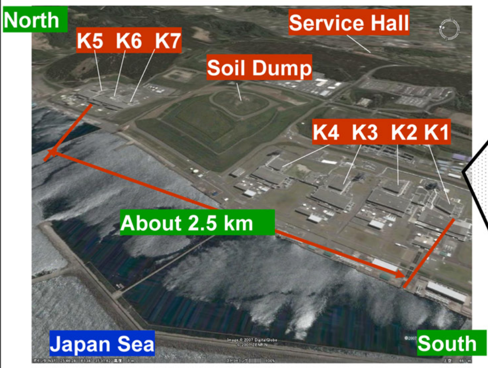
II. Cause Investigation on NCO EQ

■ Characteristics of Earthquake

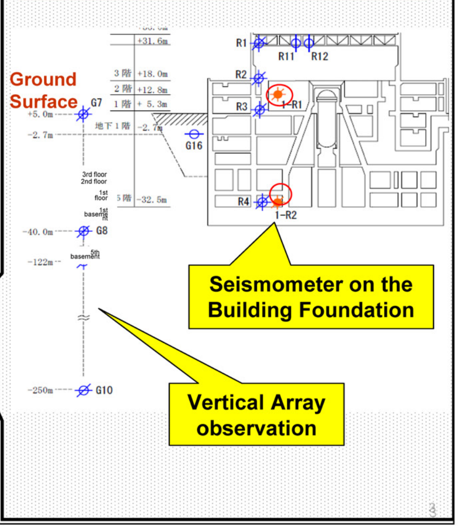


Main shock:
・July 16, 2007
・Mj : 6.8
・Depth : 10 km
・Epicenter Distance: 14 km

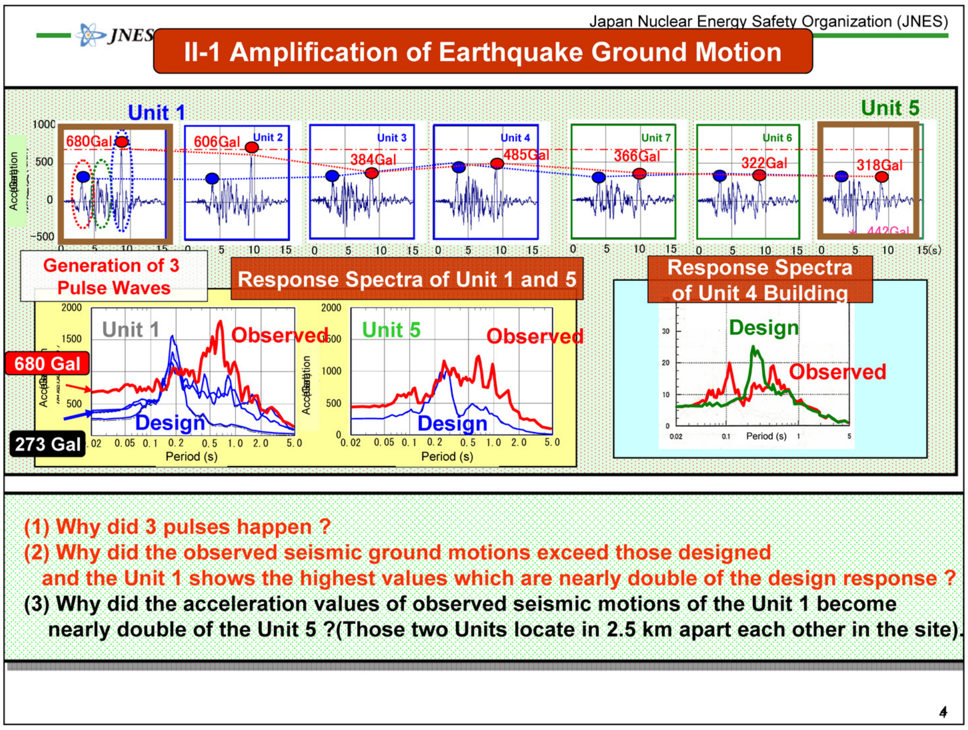
■ Location of Units K1-7

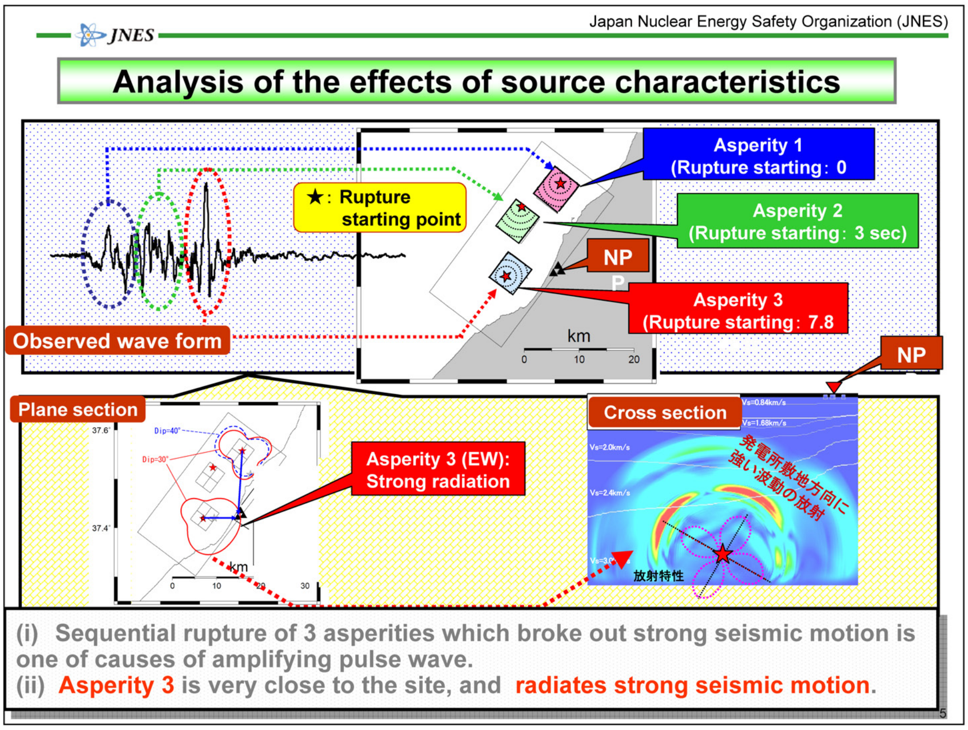


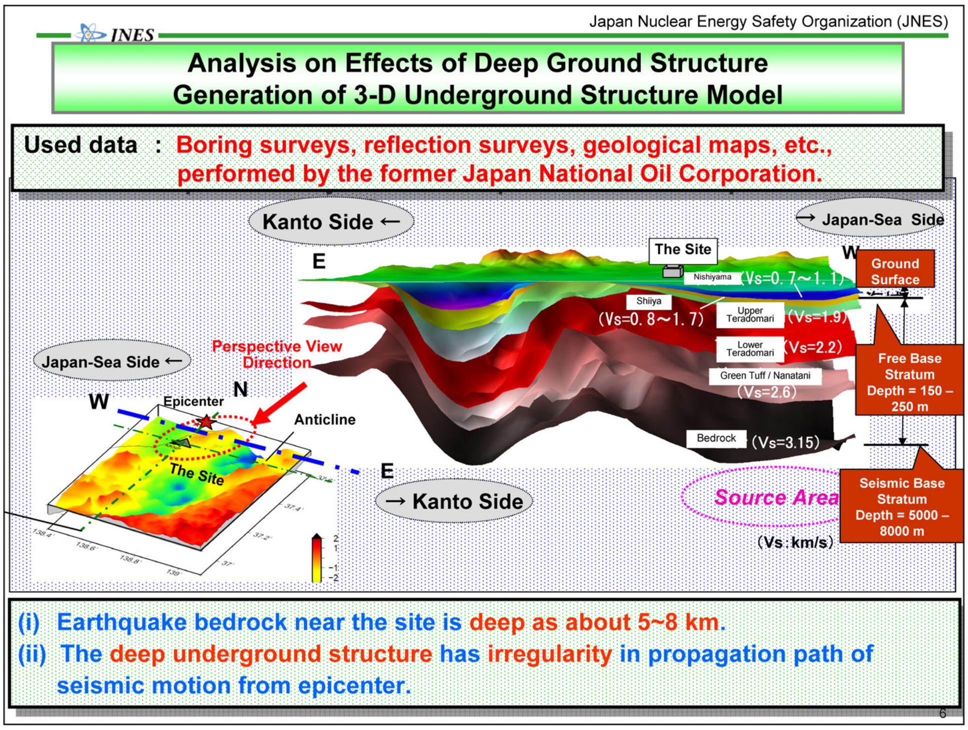
■ Earthquake Motion observation at Reactor Building

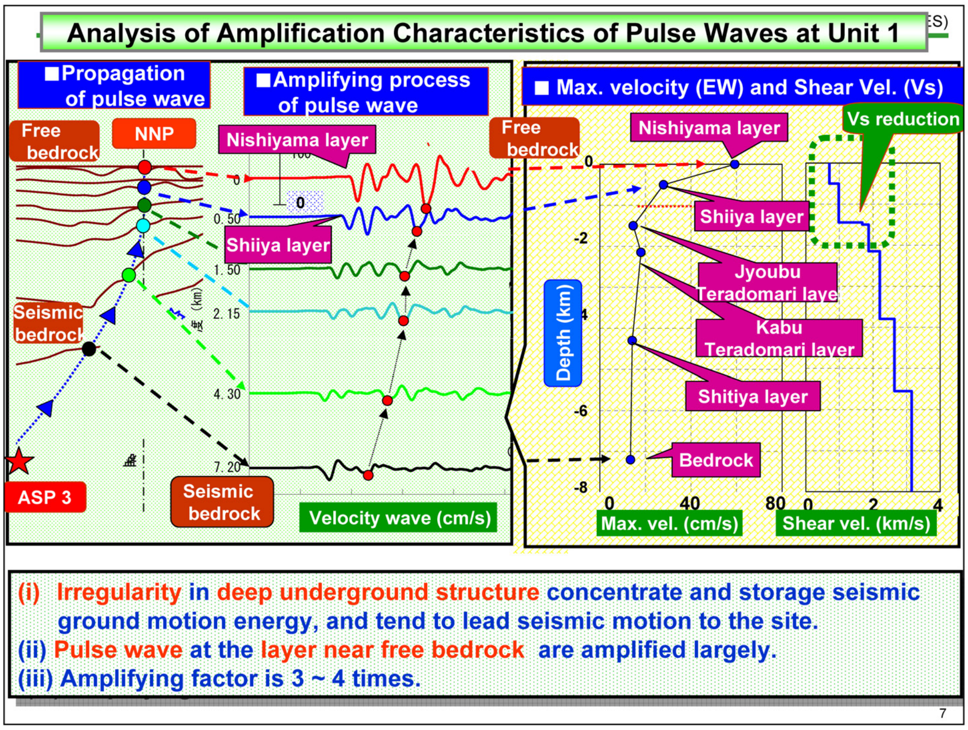


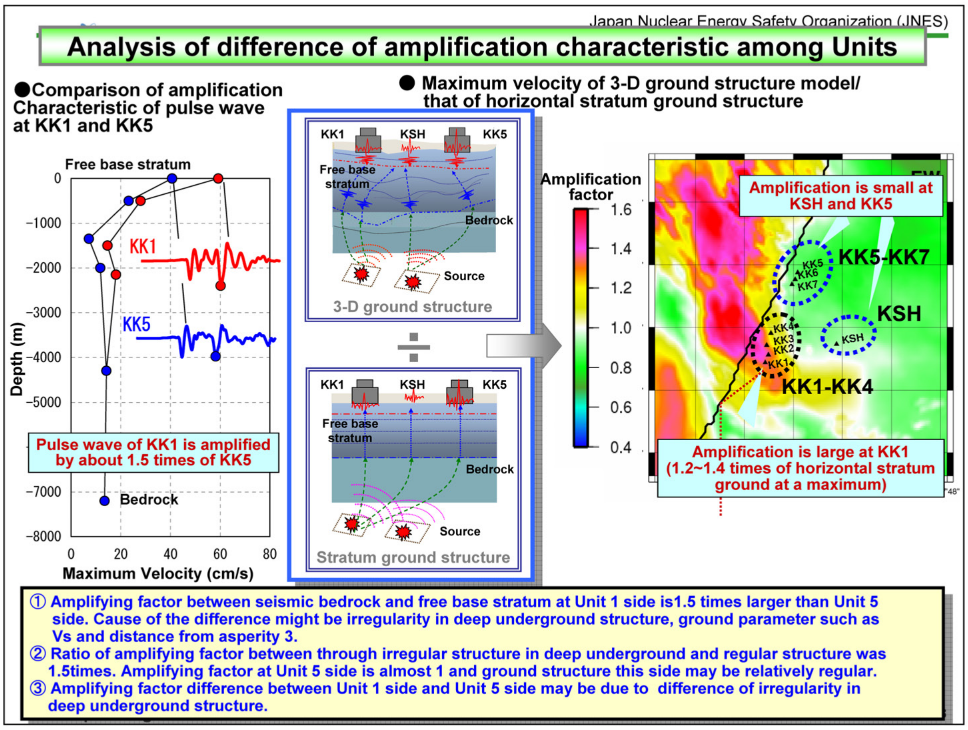
3

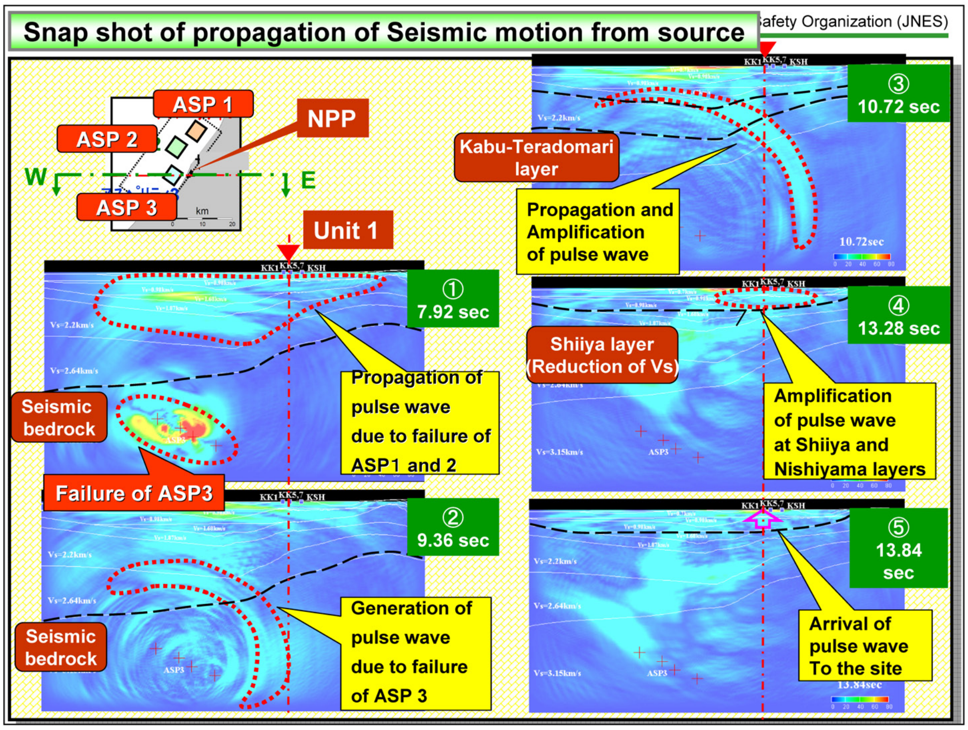

Japan Nuclear Energy Safety Organization (JNES)
JNES
II-1 Amplification of Earthquake Ground Motion
Unit 1
Unit 5
680Gal
606Gal
Unit 2
Unit 3
384Gal
Unit 4
485Gal
Unit 7
366Gal
Unit 6
322Gal
318Gal
442Gal
Acceleration
Generation of 3 Pulse Waves
Response Spectra of Unit 1 and 5
Response Spectra of Unit 4 Building
Unit 1
Observed
680 Gal
Design
273 Gal
Unit 5
Observed
Design
Period (s)
Design
Observed
Period (s)
(1) Why did 3 pulses happen ?
(2) Why did the observed seismic ground motions exceed those designed and the Unit 1 shows the highest values which are nearly double of the design response ?
(3) Why did the acceleration values of observed seismic motions of the Unit 1 become nearly double of the Unit 5 ?(Those two Units locate in 2.5 km apart each other in the site).
4

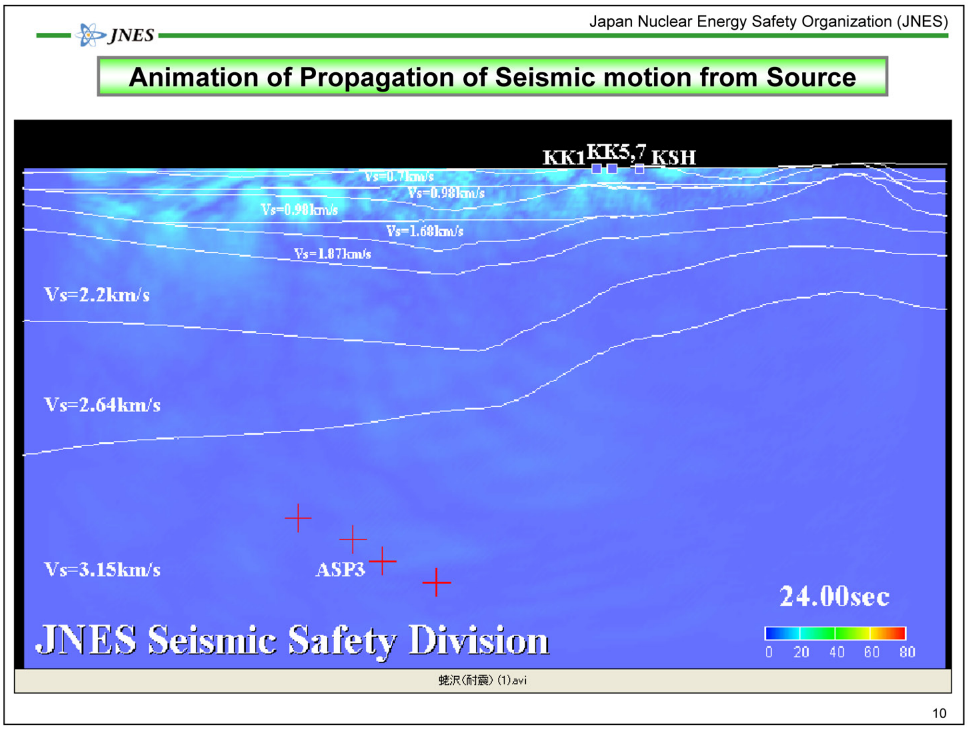

Japan Nuclear Energy Safety Organization (JNES)
JNES
Analysis of the effects of source characteristics
Asperity 1 (Rupture starting: 0
★: Rupture starting point
Asperity 2 (Rupture starting: 3 sec)
NP
Asperity 3 (Rupture starting: 7.8
Observed wave form
km
NP
Plane section
Asperity 3 (EW): Strong radiation
Cross section
発電所敷地方向に強い波動の放射
放射特性
(i) Sequential rupture of 3 asperities which broke out strong seismic motion is one of causes of amplifying pulse wave.
(ii) Asperity 3 is very close to the site, and radiates strong seismic motion.
5

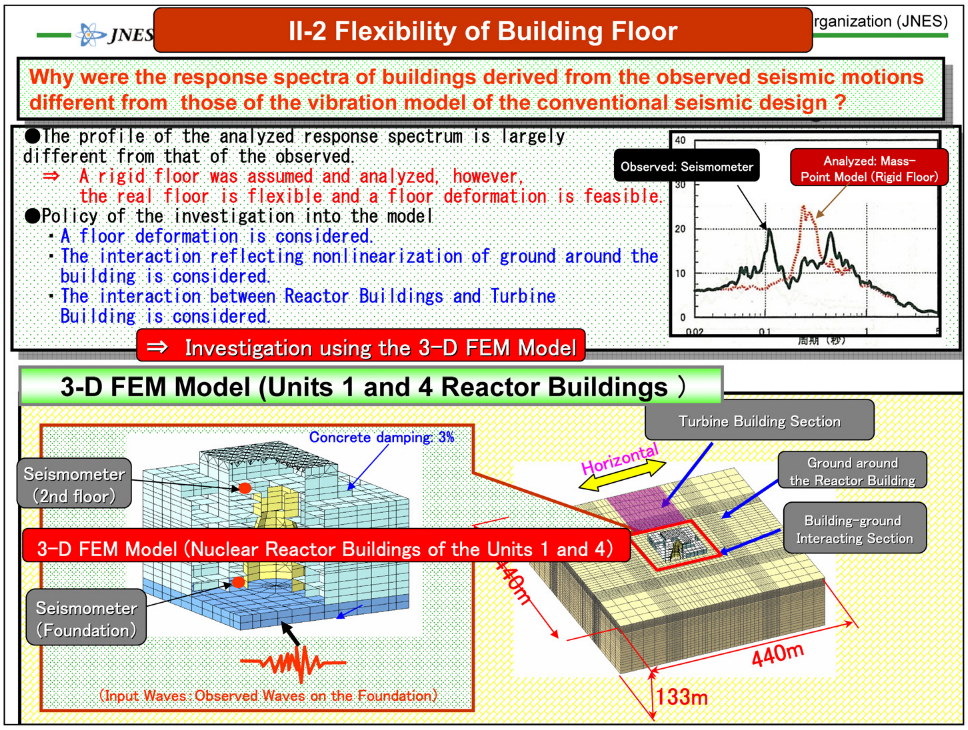

Japan Nuclear Energy Safety Organization (JNES)
JNES
Analysis on Effects of Deep Ground Structure
Generation of 3-D Underground Structure Model
Used data : Boring surveys, reflection surveys, geological maps, etc., performed by the former Japan National Oil Corporation.
Kanto Side ←
→ Japan-Sea Side
The Site
E
W
Ground Surface
Nishiyama
(Vs=0.7~1.1)
Shiiya
(Vs=0.8~1.7)
Upper Teradomari
(Vs=1.9)
Lower Teradomari
(Vs=2.2)
Green Tuff / Nanatani
(Vs=2.6)
Bedrock
(Vs=3.15)
Free Base Stratum Depth = 150 – 250 m
Seismic Base Stratum Depth = 5000 – 8000 m
Japan-Sea Side ←
Perspective View Direction
N
W
Epicenter
Anticline
The Site
E
→ Kanto Side
Source Area
(Vs: km/s)
(i) Earthquake bedrock near the site is deep as about 5~8 km.
(ii) The deep underground structure has irregularity in propagation path of seismic motion from epicenter.
6

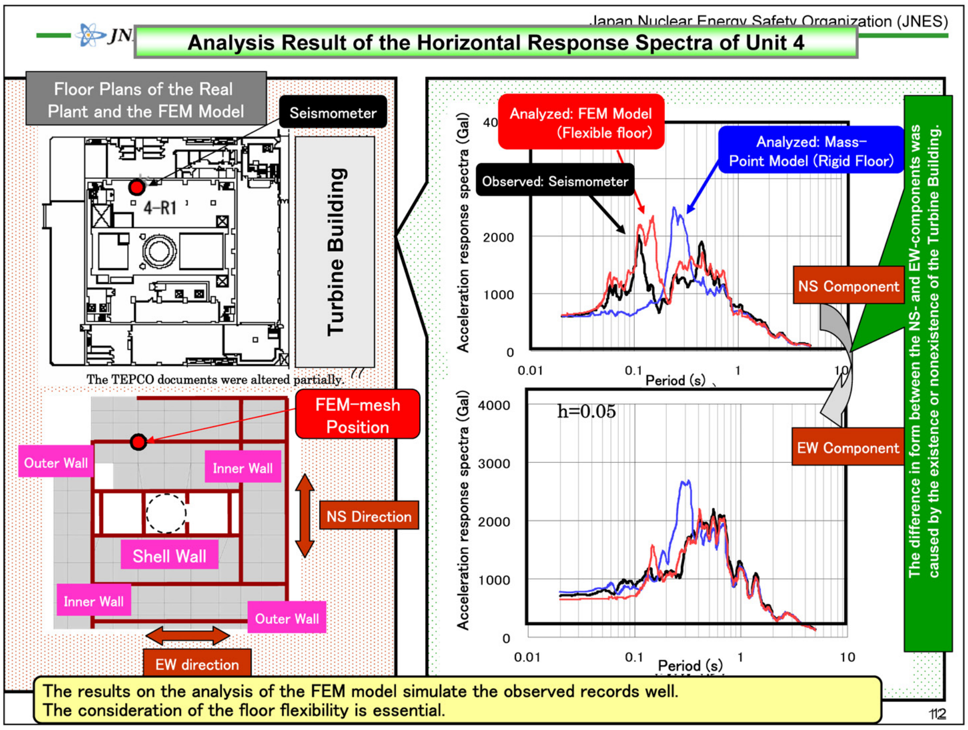

Analysis of Amplification Characteristics of Pulse Waves at Unit 1
■Propagation of pulse wave
■Amplifying process of pulse wave
■ Max. velocity (EW) and Shear Vel. (Vs)
Free bedrock
NNP
Nishiyama layer
Free bedrock
Nishiyama layer
Vs reduction
Shiiya layer
Shiiya layer
Jyoubu Teradomari layer
Kabu Teradomari layer
Shitiya layer
Bedrock
Seismic bedrock
Seismic bedrock
ASP 3
0
0.50
1.50
2.15
4.30
7.20
Velocity wave (cm/s)
Depth (km)
0
-2
-4
-6
-8
0
40
80
Max. vel. (cm/s)
0
2
4
Shear vel. (km/s)
(i) Irregularity in deep underground structure concentrate and storage seismic ground motion energy, and tend to lead seismic motion to the site.
(ii) Pulse wave at the layer near free bedrock are amplified largely.
(iii) Amplifying factor is 3 ~ 4 times.
7

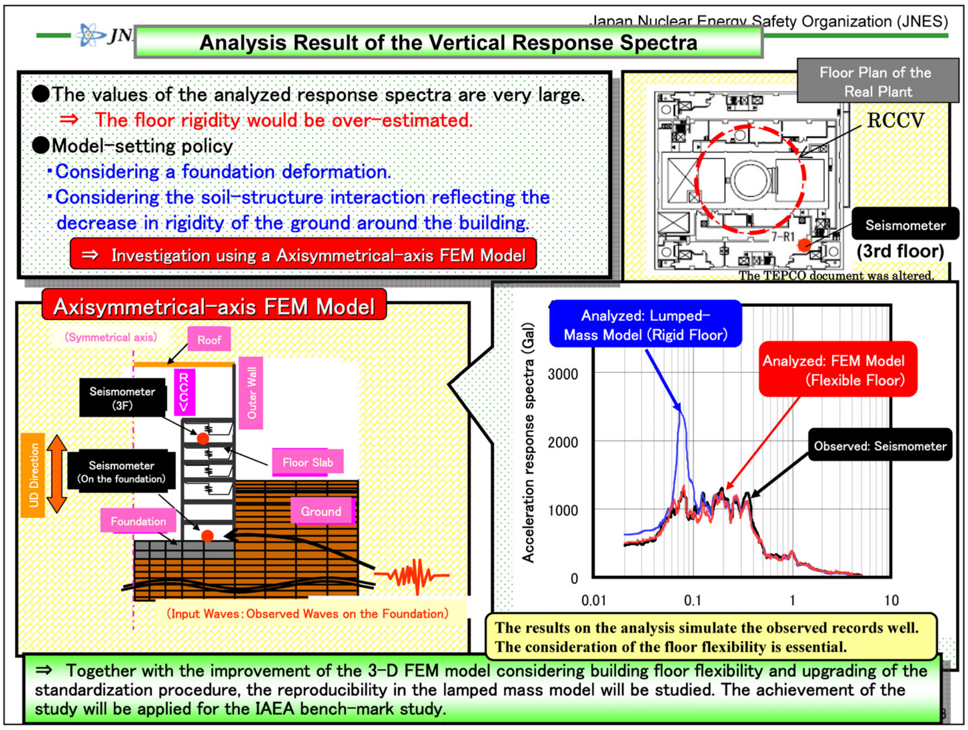

Japan Nuclear Energy Safety Organization (JNES)
Analysis of difference of amplification characteristic among Units
●Comparison of amplification Characteristic of pulse wave at KK1 and KK5
● Maximum velocity of 3-D ground structure model/ that of horizontal stratum ground structure
Free base stratum
KK1
KK5
Depth (m)
Maximum Velocity (cm/s)
Bedrock
Pulse wave of KK1 is amplified by about 1.5 times of KK5
KK1 KSH KK5
Free base stratum
Bedrock
Source
3-D ground structure
Stratum ground structure
Amplification factor
Amplification is small at KSH and KK5
KK5-KK7
KSH
KK1-KK4
Amplification is large at KK1 (1.2~1.4 times of horizontal stratum ground at a maximum)
① Amplifying factor between seismic bedrock and free base stratum at Unit 1 side is1.5 times larger than Unit 5 side. Cause of the difference might be irregularity in deep underground structure, ground parameter such as Vs and distance from asperity 3.
② Ratio of amplifying factor between through irregular structure in deep underground and regular structure was 1.5times. Amplifying factor at Unit 5 side is almost 1 and ground structure this side may be relatively regular.
③ Amplifying factor difference between Unit 1 side and Unit 5 side may be due to difference of irregularity in deep underground structure.


Safety Organization (JNES)
Snap shot of propagation of Seismic motion from source
ASP 1
ASP 2
ASP 3
NPP
W
E
Unit 1
Kabu-Teradomari layer
Propagation and Amplification of pulse wave
③ 10.72 sec
① 7.92 sec
Propagation of pulse wave due to failure of ASP1 and 2
Seismic bedrock
Failure of ASP3
Shiiya layer (Reduction of Vs)
④ 13.28 sec
Amplification of pulse wave at Shiiya and Nishiyama layers
② 9.36 sec
Generation of pulse wave due to failure of ASP 3
Seismic bedrock
⑤ 13.84 sec
Arrival of pulse wave To the site


JNES
Japan Nuclear Energy Safety Organization (JNES)
Animation of Propagation of Seismic motion from Source
KK1 KK5,7 KSH
Vs=0.7km/s
Vs=0.98km/s
Vs=0.98km/s
Vs=1.68km/s
Vs=1.87km/s
Vs=2.2km/s
Vs=2.64km/s
Vs=3.15km/s
ASP3
24.00sec
JNES Seismic Safety Division
0 20 40 60 80
10


JNES
II-2 Flexibility of Building Floor
rganization (JNES)
Why were the response spectra of buildings derived from the observed seismic motions different from those of the vibration model of the conventional seismic design ?
●The profile of the analyzed response spectrum is largely different from that of the observed.
⇒ A rigid floor was assumed and analyzed, however, the real floor is flexible and a floor deformation is feasible.
●Policy of the investigation into the model
・A floor deformation is considered.
・The interaction reflecting nonlinearization of ground around the building is considered.
・The interaction between Reactor Buildings and Turbine Building is considered.
⇒ Investigation using the 3-D FEM Model
Observed: Seismometer
Analyzed: Mass-Point Model (Rigid Floor)
3-D FEM Model (Units 1 and 4 Reactor Buildings)
Concrete damping: 3%
Seismometer (2nd floor)
Seismometer (Foundation)
3-D FEM Model (Nuclear Reactor Buildings of the Units 1 and 4)
(Input Waves: Observed Waves on the Foundation)
Turbine Building Section
Horizontal
Ground around the Reactor Building
Building-ground Interacting Section
440m
440m
133m


Japan Nuclear Energy Safety Organization (JNES)
Analysis Result of the Horizontal Response Spectra of Unit 4
Floor Plans of the Real Plant and the FEM Model
Seismometer
Turbine Building
4-R1
The TEPCO documents were altered partially.
FEM-mesh Position
Outer Wall
Inner Wall
NS Direction
Shell Wall
Inner Wall
Outer Wall
EW direction
Analyzed: FEM Model (Flexible floor)
Analyzed: Mass-Point Model (Rigid Floor)
Observed: Seismometer
Acceleration response spectra (Gal)
NS Component
EW Component
Period (s)
h=0.05
The difference in form between the NS- and EW-components was caused by the existence or nonexistence of the Turbine Building.
The results on the analysis of the FEM model simulate the observed records well.
The consideration of the floor flexibility is essential.
112

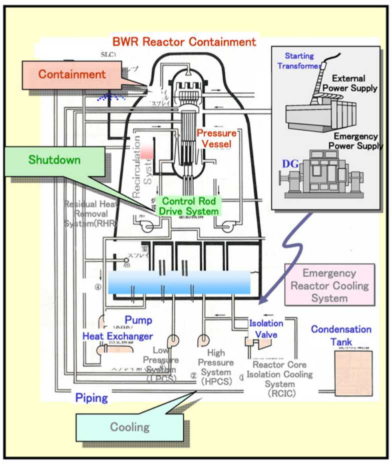

Japan Nuclear Energy Safety Organization (JNES)
Analysis Result of the Vertical Response Spectra
●The values of the analyzed response spectra are very large.
⇒ The floor rigidity would be over-estimated.
●Model-setting policy
・Considering a foundation deformation.
・Considering the soil-structure interaction reflecting the decrease in rigidity of the ground around the building.
⇒ Investigation using a Axisymmetrical-axis FEM Model
Floor Plan of the Real Plant
RCCV
Seismometer
(3rd floor)
7-R1
The TEPCO document was altered.
Axisymmetrical-axis FEM Model
(Symmetrical axis)
Roof
RCCV
Outer Wall
Seismometer (3F)
Seismometer (On the foundation)
UD Direction
Floor Slab
Foundation
Ground
(Input Waves: Observed Waves on the Foundation)
Analyzed: Lumped-Mass Model (Rigid Floor)
Analyzed: FEM Model (Flexible Floor)
Observed: Seismometer
Acceleration response spectra (Gal)
The results on the analysis simulate the observed records well.
The consideration of the floor flexibility is essential.
⇒ Together with the improvement of the 3-D FEM model considering building floor flexibility and upgrading of the standardization procedure, the reproducibility in the lamped mass model will be studied. The achievement of the study will be applied for the IAEA bench-mark study.

Japan Nuclear Energy Safety Organization (JNES)

JNES

II-3 Integrity of safety related SSCs and about 3500 troubles

Observed ground motion far exceeded the design of KK NPP. However Safety related SCCs (Shout down, Cooling and Containment) function maintained.
TEPCO reported the damages or nonconformities of about 3500 that were not important accidents.
They are the precious information to identify accident scenarios on seismic PSA



① Damage on wheel driving shaft connection of over head crane(k-6)
② Crack on connection valve junction between main condenser water boxes and sea water leak (k-4)
Degradation of water tightness of watertight doors for RCIC and RHR (k-7)
Dislocation of blowout panel of reactor building (k-3)
③ None
④ Dislocation of air duct connected to Stack
⑤ Slope failure of a part of east side slope of the switch yard
⑥ None
⑦ None. In-leak of water due to fire fighting pipe failure and flooding on lowest basement floor
⑧ Detection of iodine at main stack (K-7)
⑨ Fire of in-house transformer 3B (K-3)
Oil leak of start-up transformers 3B (K-3,4,6)
⑩ Subsidence of yard
⑪ Falling down of the secondary structures of ceiling
⑫ None

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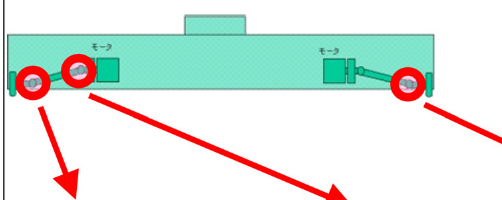
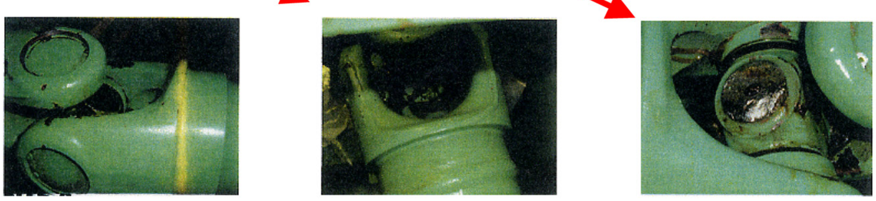
JNES

Functional failure of ceiling crane at unit 6

- Part : joint part
- Failure mode : shearing and bending by putdown earthquake motion





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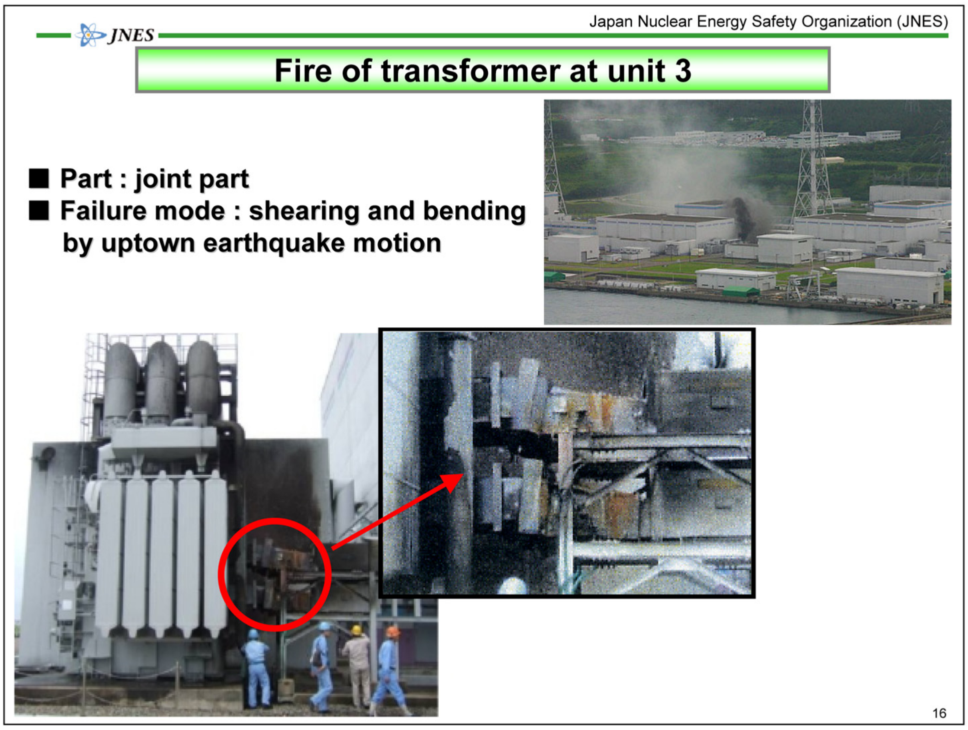
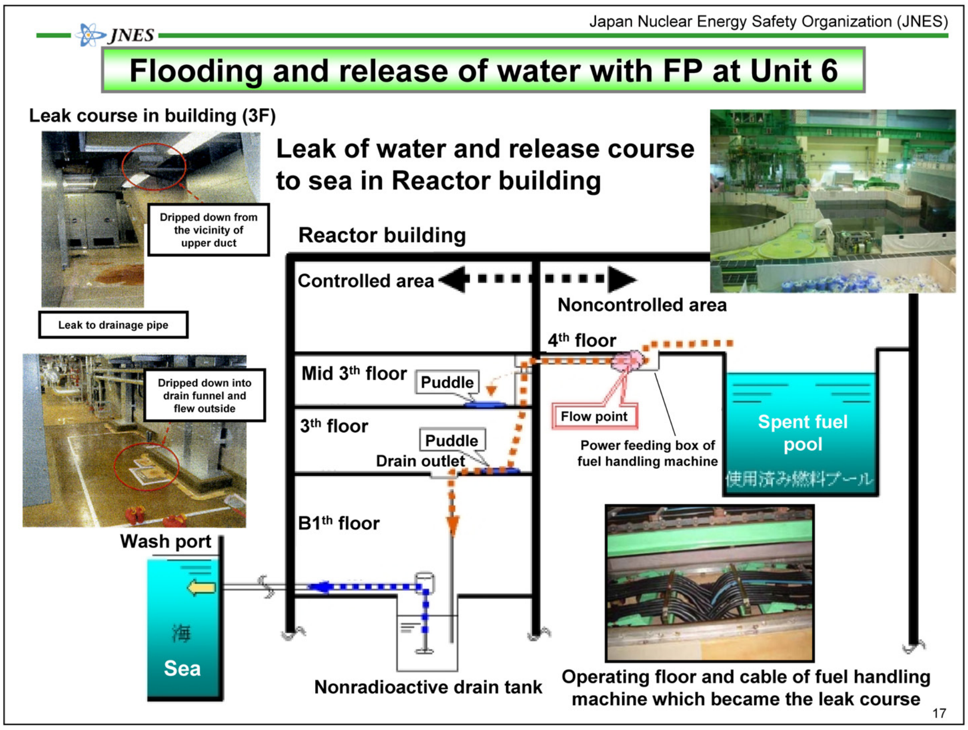

Japan Nuclear Energy Safety Organization (JNES)
JNES
Fire of transformer at unit 3
■ Part : joint part
■ Failure mode : shearing and bending by uptown earthquake motion
16


Japan Nuclear Energy Safety Organization (JNES)
JNES
Flooding and release of water with FP at Unit 6
Leak course in building (3F)
Dripped down from the vicinity of upper duct
Leak to drainage pipe
Dripped down into drain funnel and flew outside
Wash port
海
Sea
Leak of water and release course to sea in Reactor building
Reactor building
Controlled area
Noncontrolled area
4th floor
Mid 3th floor
Puddle
3th floor
Puddle
Drain outlet
B1th floor
Flow point
Power feeding box of fuel handling machine
Spent fuel pool
使用済み燃料プール
Nonradioactive drain tank
Operating floor and cable of fuel handling machine which became the leak course
17

Japan Nuclear Energy Safety Organization (JNES)

JNES

III. Reflection to Seismic PSA Methodology

III-1. Seismic Hazard Evaluation

① Treatment of Seismic Source Model at near Site:
Utilization of Fault Model

② Treatment of Uncertainty for Earthquake Motion:
Management of Logic Tree

③ Treatment of Stress concentration Zone

III-2. Fragility Evaluation

① Floor Flexibility of Building

② Integrity of Component

③ Functional Failure of Ceiling Crain under Vertical motion

④ Utilization to Seismic Margin Evaluation

III-3.Accident Sequence Evaluation

① Reimprovement of Accident Scenarios

② Improvement of Frequency of Fire and Flooding caused by Earthquake

③ Improvement of Evaluation Method of Core Damage Frequency for Multi-NPPs

18

JNES

rganization (JNES)

III-1 (1) Handling of near site seismic source - Effectiveness of source model -

■Lesson learned from NCO earthquake:
cause analysis on amplification factor of NCO earthquake is achieved by source model (fault model) method and 3-D underground structure model.
In other word, cause analysis is difficult if source model method is not available

■Situation of seismic hazard evaluation :
In many case, seismic hazard of nuclear site is dominated by near sources inside 60km radius area, never the less by specified source or by average hazard of the region. So, detail evaluation of near sources is essential and setting of upper limit is very important. Seismic hazard evaluation of JAEA seismic PSA standard procedure describe both prediction by attenuation relationship and by source model.

■Direction of improvement:
(1) Recognize effectiveness of source model prediction in viewpoint of ;
・detail evaluation of seismic source and ground motion
・ resolution on mechanism of seismic source and ground motion
and outgrow from evaluation by attenuation equation only

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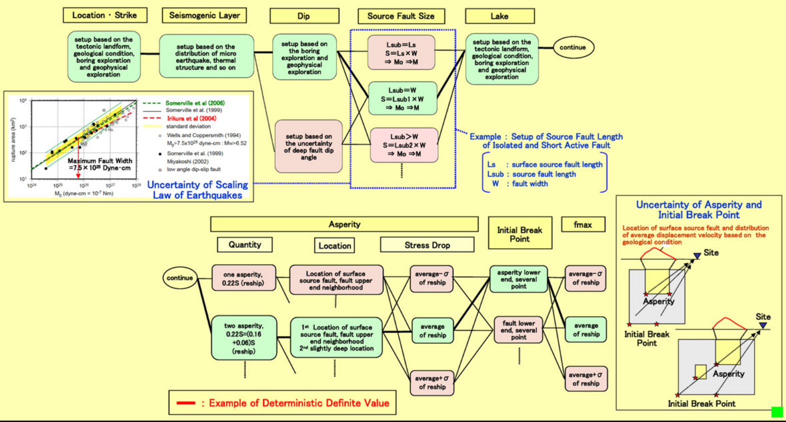
JNES Japan Nuclear Energy Safety Organization (JNES)

(2) Detail evaluation is achieved by source model method, setting 16 parameters. However, at the site where less information on these parameters, caution should be paid on large uncertainty. Hereafter, to improve source model method, quantitative evaluation on uncertainty of each parameters and interaction between parameters.

【Note】: Consultancy meeting of IAEA Seismic Hazard Evaluation guide DS422 was held in Tokyo at February 2009. Based on opinion of Japanese specialist of this area(Takada, Ebisawa, Irikura, Okumura, Kameda) , source model method is discussed and adopted to the guide.



20

ES)

III-1 (2) Treatment of Uncertainty in Earthquake and Ground Motion - Utilization of Logic Tree -

■ New Seismic Design Review Guide

New Guide requires consideration of uncertainty concerned with the evaluation process of the Design Basis Ground Motion (DBGM) Ss and referring of its exceedance probability.

■Lessons learned from NCO Earthquake

Latest findings from the NCO Earthquake also show the necessity of consideration of uncertainty and referring of exceedance probability in the determination of DBGM Ss.

In the open committees of regulatory body, deliberation on Ss of utilities and evaluation of exceedance probability in the probabilistic seismic hazard has been carried out but it is not proceeding efficiently. The reasons are 1) there are few committees knowing probabilistic seismic hazard evaluation fully enough, 2) there is no rule of its deliberation, and 3) utilities' evaluation does not necessarily meet to the AESJ Seismic PSA Implementation Standards.

■Direction of Improvement

There are many opinions from regulatory body and also utilities that seismic hazard evaluation for each site should be performed along the open deliberation rule by the public organization such as JNES.

(2) JNES proposed a draft framework of the open deliberation rule, which was made so as to be able to utilize logic tree practically, referring to the implementation procedure of logic tree in AESJ Seismic PSA Implementation Standards.

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① Uncertainty factors in determination of ground motion Ss are roughly classified into those in earthquake parameters and those in ground motion, and both of them are studied.
It is a common understanding that the final target is determination of the design basis ground motion necessary to structure design, not setting earthquake parameters.
② It is confirmed that there are two kinds of uncertainty factors; one is those which can be treated by probability, and the other is those which can not be expressed by probability such as difference of expert opinions.
It is re-confirmed that it is especially quite important to form consensus between experts regarding the latter factors.
③ It is re-confirmed whether there are any effective methods other than logic tree method in order to quantify uncertainty or not. If there are none, then it should not be cloud but make clear whether to apply logic tree method.
④ It is first priority to secure explanation-ability and transparency through whole deliberation process. It is prohibited to make discussions only for the sake of discussion such as staying and hesitating on the way of the course from setting earthquake parameters to determination of basic ground motions.

⑤ Sensitivity analysis on the factors proposed as expert opinions is performed timely on the way of logic tree formation and their contributions to basic ground motions are shown quantitatively. Factors with less contributions are left late without persistence and progress of deliberation should be promoted. Technical issues are clarified concerning these factors with less contributions with leaving evidence, and commended to academic society and/or association.

⑥ It is a common understanding that recent scientific knowledge and findings are utilized usefully, but that there is a possibility of facing a situation in which there is no way other than engineering judgment finally in the determination of basic ground motions.
⑦ In a case that there occurs discussion how ground motion Ss influences the function of structures, reference information will be given. But emphasis is put persistently on the discussion of determination of ground motion.

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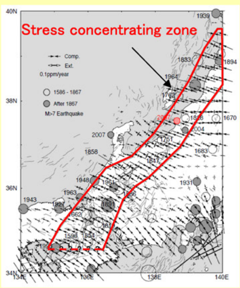
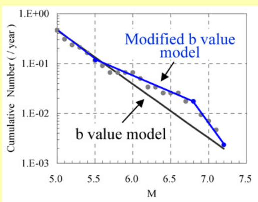
III-1 (3) Handling of stress concentrating

■Lessons learned from NCO earthquake
Seismic activity around NCO epicenter area is much high and so called "Stress concentrating zone". And there also exists " Earthquake blank region"

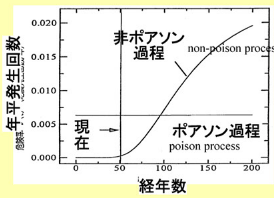
■ Direction of improvement
Handling on seismic hazard evaluation:
(1) "Stress concentrating zone": Revise b value in Guten Rihiter equation of seismic source in that zone
(2) : " Earthquake blank region": Adopt Non-Poisson Process (Renewal Process) in the earthquake frequency evaluation at that region



"Stress concentrating zone"
Revise B value



" Earthquake blank region"
Non-Poison process



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III-2 (1) Consideration of Building Floor Flexibility

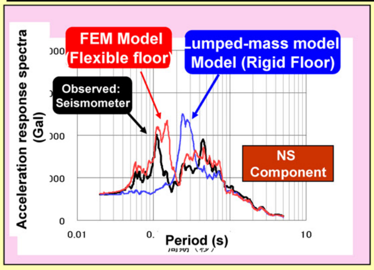
■Lesson learned of building floor flexibility

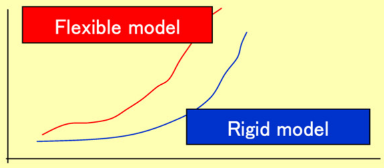
1) Building model considering floor flexibility, instead of current rigid floor model, should be adopted for fragility evaluation of building.

2) Floor response spectrum or time history acquired using building model considering floor flexibility, instead of current rigid floor model, should be adopted for fragility evaluation of facilities.

3) In case detail evaluation of fragility is not required, flexible floor model is not essential requirement

	Building floor model	Floor response spectrum	Fragility
Current handling	rigid	Floor response value by rigid floor model is smaller than that of by flexible floor model at 0.1～0.3 Sec. region. So, facility at the region like piping is non-safety side.	In case of piping, fragility by rigid floor is smaller than that by flexible floor
From now	flexible		





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III-2 (2) Consideration of Component Integrity

■**Lessons learned from the Integrity of S-class Components**

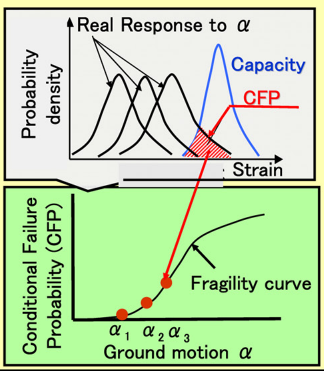
(1)These lessons are used to verify the reliability of fragility evaluation by the following manner:

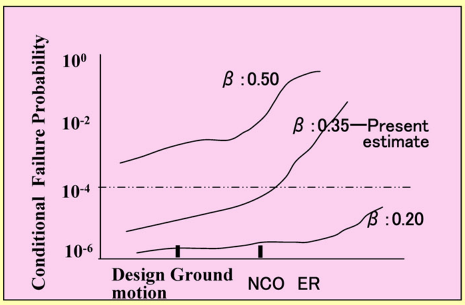
① Confirmation of maintenance of component function at the observed ground motion α means that CFP(α)〔Conditional Failure Probability〕 is extremely close to 0.

② CFP(α)is caluculated using median mean value and logarithmic standard deviation β of realistic response and capacity. Important parameter β is to be confirmed whether the value is appropriate or not.

③ A probability extremely close to 0 is regarded around10^{-4}

(2) Analysis examples confirm that β used in Japan is appropriate.





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III-2 (3) Consideration of Effectiveness of Vertical motion

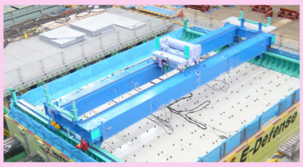
■ **Lessons learned from the damage of R/B over head crane**

1) The cause of the damage of the over head crane was identified as the effect of vertical response
2) Implementation of review for fragility evaluation method considering vertical response
3) Review policy for fragility evaluation is:
Firstly, to review identification of facilities to end to be affected by vertical response,
Then, for each facility identified, to review failure mode and mechanism,
Adding the above, to propose countermeasures to prevent failure and to verify their effectiveness
4) Implementation of shaking table test to verify the policy of 3)
⇒ Over head crane

Typical facilities tend to be affected by up-down motion

- Vertical shaft pump
- Piping
- Over head crane
- Refueling machine

Over Head Crane



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In the revision of the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities, vertical motions are considered for design basis ground motion and improvement of the assessment method of the vertical nonlinear response for structure and equipment is required. At the Chuetsu-oki Earthquake a overhead crane in the Kashiwazaki-Kariwa power plant was damaged and the study on the influence of the vertical motions is thought to be important.

■ Target Equipment
・Overhead crane (consisting of a garters, a trolley, a hanging load, lugs, etc.)

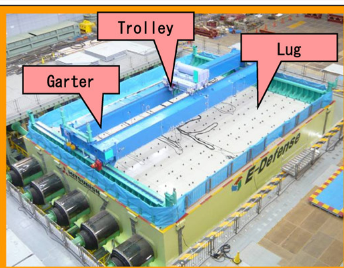
■ Contents of the Test

● Component Tests (performed in FY 2007)
① Factor analyses of the functional limit
② The mutual uplift and the collision assessment of garter/trolley/hanging load
③ The assessment of the restitution coefficient of wheels

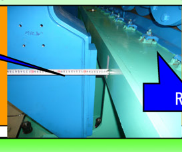
● Reduced Scale Model Test (performed in Oct. 2008)
(1) Additional investigation point from NCE
⇒ Effect confirmation of the fall-prevention work (lug), etc.
(2) Analysis in detail is ongoing. Preliminary reports are available on the JNES homepage.
・Effectiveness of lugs was confirmed.
・The uplift behavior was understood.

Video of Whole View
Video of Wheel Part

http://www.jnes.go.jp/katsudou/topics2008.html#081128



Fall-prevention Work (Lug)



⇒ The uplift mechanism for the vertical motions will be clarified and the nonlinear analysis method will be improved. The effectiveness of the fall-prevention work will be confirmed and the results will be applied to the integrity criteria in the seismic back-check.

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III-2 (4) Utilization to Plant Seismic Margin Estimation

■Lessons learned from Plant Seismic Margin

- "Shutdown", "Cooling" and "Confinement" function were kept and plant safety was maintained. The reason is to be said due to adequate plant safety margin but not yet explained quantitatively.
- JNES is planning to estimate the margin quantitatively by utilizing seismic PSA.

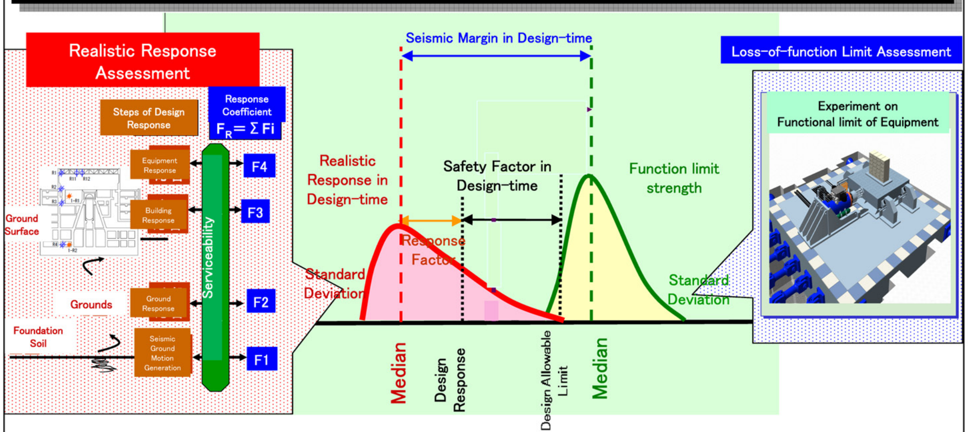
- Fragility evaluation method of AESJ Seismic PSA Implementation Standard include JNES method(detailed direct method), JAERI method (less detailed, partially factor method) and Zion method (simplified, factor method).
  - JEARI method: Useful for seismic margin estimation because;
    (1)Realistic response and capacity of components are treated separately. So deviation between them can be indicated clearly.
    (2)Conservativity factors of design response in realistic response evaluation are separately treated in ground motion setting, ground response, building response and component response. So, each contribution is indicated individually. Zion method is hard to explain the seismic margin rationally.

- For fragility evaluation, at least JAERI method or JNES method is preferable, considering application of the fragility evaluation method to seismic margin estimation hereafter.

- Although the development of capacity data base is an task, an idea is that the difference in seismic design of Korea and Japan is expressed by coefficient and Japanese capacity data are corrected as to meet in Korea

- The average seismic margin can be evaluated by the comparison with the median of the "functional limit strength" and the "realistic response".
- The seismic margin including variation can be understood quantitatively by the consideration of each "logarithmic standard deviations".

Application of the Seismic PSA Method



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III-3 (1) Reconstruction of Accident Scenario

Examples of extensive accident scenarios described in the S-PSA Implementation Standards

A. Influence of main shock its core damage

- A1. Direct effect — ①Building/Structure/component with safety function (Damage of As, A equipment)
- A2 Indirect effect
  - A2-1 Those except for building / structure/component with safety function (Secondary effect affeting As/A equipment and leading to core damage)
    - ･Indoor equipment
      - ① Effect of fall of overhead crane to containment vessel / pressure vessel
      - ② Effect of damage of B/C equipment to As/A component
      - ③ Effect of turbine missile to adjoining building
    - ･Outdoor equipment
      - ④ Effect of collapse of stack to building
      - ⑤ Effect of land slope collapse to building / surrounding facility
      - ⑥ Loss of off-site power due to damage of electric grid tower, etc.
      - ⑦ Loss of cooling function due to stop of water supply
  - A2-2 Effect due to human error
    - ⑧ Operation error of plant operators/ workers due to high stress during and after earthquake
    - ⑨ Failure of back-up operation by blackout due to damage of insulator in transformer, etc.
    - ⑩ Obstacle of passage in the power due to soil liquefaction/ damage of landslide-preventing wall
    - ⑪ Effect of damage of secondary material on ceiling, etc. to operators, etc.
    - ⑫ Mistake or error through planning / design of plant, selection of materials, production and assembling

B. Influence of after shock to core damage — ⑬Evaluation of increment of core damage frequency due to after shock.

Status of damage of NPP due to the NCO Earthquake

① none

① (Unit 6) Damage of Universal joint of the axle of overhead crane
② (Unit 4)Crack of joint part of connection value of condensation tank, leakage of seawater
(Unit 7) Deterioration of seal of RCIC and RHR systems
(Unit 3) Break of blowout panel of reactor building
③ None
④ Stack Slant, duct displacement
⑤ Land Slide in the east of switch yard
⑥ None
⑦ Flooding in the basement of building due to pipe break of fire extinguishing system

⑧ (Unit 7) Detection of iodine in main stack
⑨ (Unit 3) House transformer fire
(Units 3,4,6) Oil leakage in start-up transformer
⑩ Ground subsidence in the power station
⑪ Falling of secondary material of ceiling
⑫ None

⑬ None

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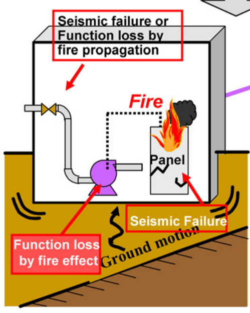
III-3 (2) Evaluation of Seismically caused fire and flooding

Current situation : A fire of C class transformer caused by the Chuetsu-oki earthquake called attention to possibility of fire of A class equipment

Earthquake
- No fire →Accident scenario only for earthquake
- Fire →Accident scenario for seismically caused fire

Earthquake
- No flooding →Accident scenario only for earthquake
- Flooding →Accident scenario for seismically caused Flooding



Development of seismic cause fire probability evaluation method
- Grouping of fire sources (including transformer)
- Study of generating mechanism
- Development of prob. evaluation method

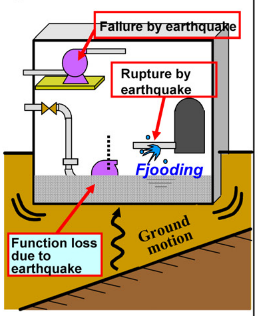
$P_{SF}(\alpha)=P_S(\alpha)*P_F(\alpha)$

$P_{SF}(\alpha)=$Fire probability
$P_S(\alpha)=$ Failure probability of fire source
$P_F(\alpha)=$Ignition prob.<1.0
$\alpha=$ Intensity of seismic motion

Development of seismic cause flooding probability evaluation method
- Grouping of flooding sources (tanks, piping etc.)
- Study of generating mechanism
- Development of prob. evaluation method

$P_{SW}(\alpha)=P_S(\alpha)*P_W(\alpha)$

$P_{SW}(\alpha)=$Flooding probability
$P_S(\alpha)=$ Failure probability of fire source
$P_W(\alpha)=$Flooding prob. $\fallingdotseq 1.0$(*)
$\alpha=$ Intensity of seismic motion



Screening by probability

(*) Flooding probability depend on the existence and capacity of drain equipment

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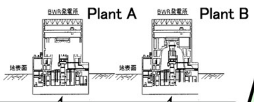
III-3 (3) Evaluation of Multi-Unit Plants

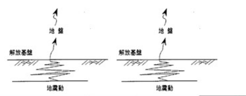
Recognition : multi-unit on same site / possibility of simultaneous loss of functions / essential in regard to risk-based assessment

Task : Enhancement of CDF assessment methodology by considering response and fragility correlations and seismic motion correlation in terms of intensity and time

CDF : multi units under correlation conditions

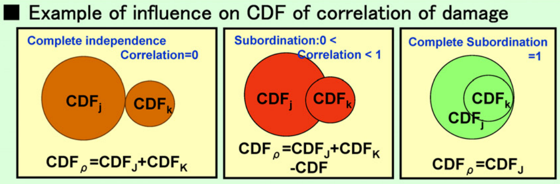


equipments : correlation in response and resistance



Input E.Q. motions : correlation of input motiions

■ Example of influence on CDF of correlation of damage



CDF$_i$: CDFof Plant i, CDF : Overlap area of CDF$_J$ and CDF$_k$

■ Evaluation method of correlation of damage between plants

Correlation coefficient of damage between facilities in plant
=F(Correlation coefficient of response, Correlation of capacity)

- Correlation of response :
  Vibration characteristics of equipment
  Frequency characteristic of input motion
- Correlation of capacity

		setting floor	
		same	different
period	same (0.1秒)	1.0	0.7~0.8
	different (0.02~0.1秒) (0.1~0.5秒)	0.6~0.7	0.5~0.6

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IV. Conclusions

- ■ The lessons learned from NCO EQ extremely affected to the seismic safety. The overview of cause investigation was introduced.
- ■ The above lessons reflected to each evaluation methods of seismic hazard, fragility and accident sequence in seismic PSA. The overview of reflection was also introduced.
- ■ JNES is ready to share the lessons learned from NCO EQ with international nuclear community e.g. through various chances like IAEA, OECD/NEA, NRC and individual countries.
- ■ In order to actively contribute to further improvement of seismic safety, JNES will be contribute to the IAEA's International Seismic Safety Center.

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Aging Related Degradation Assessment of Structures and Passive Components for use in Performing PSAs

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Brookhaven National Laboratory

Presented to
The 10th Korea-Japan Joint Workshop on PSA
May 18-20, 2009

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Office of Science U.S. DEPARTMENT OF ENERGY

Presentation Outline

- Need for Consideration of Aging Degradation in Nuclear Power Plants (NPPs)
- NRC Regulations and Regulatory Guidance Related to Aging
- Component Aging Degradation Assessment Process for Use in PSA
- Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSAs
- BNL/KAERI Collaboration Program on Aging

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Need for Consideration of Aging Degradation in Nuclear Power Plants (NPPs)

- U.S. Nuclear Power Plants (NPPs) are aging. Many of the plants are approaching their 40-year design life
- Management of age-related degradation is important for the current safe operation of nuclear power plants and for licensing renewal
- Past studies and inspections have identified aging degradation of structures and passive components
- Little is known about how degradation could affect the response and resistance of structures and passive components under various design loads
- Lack of reliable inspection techniques for inaccessible areas

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NRC Regulations and Regulatory Guidance Related to Aging

Maintenance Rule

- 10 CFR 50.65 - Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Maintenance Rule)
- NRC Regulatory Guide 1.160, Rev. 2, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants

License Renewal Rule

- 10 CFR Part 54 – Requirements for Renewal of Operating Licenses for Nuclear Power Plants (License Renewal Rule)
- NUREG-1800, Rev. 1 Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants
- NUREG-1801, Rev. 1 Generic Aging Lessons Learned (GALL) September 2005
- Regulatory Guide 1.188 “Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses”

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NRC Regulations and Regulatory Guidance Related to Aging (Cont'd)

Other

- 10 CFR 50.55a Codes and Standards - Imposes the inservice inspection (ISI) requirements of the ASME Boiler and Pressure Vessel (B&PV) Code
- 10 CFR 50, Appendix J - Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
- RG 1.127, Revision 1 - Inspection of Water-Control Structures Associated with Nuclear Power Plants
- RG 1.35 Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments Rev. 3
- NRC Regulatory Guide 1.35.1, Determining Prestressing Forces for Inspection of Prestressed Concrete Containments, July 1990
- RG 1.147, Rev. 15 – Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1
- RG 1.54, Rev. 1 – Service Level I, II, and III Protective Coatings applied to Nuclear Power Plants
- RG 1.192, Operation and Maintenance Code Case Acceptability

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Component Aging Degradation Assessment Process for Use in PSA

- **Selection of Critical Components**
- **Identify/Improve Analytical or Test Methods**
- **Perform Analyses/Tests**
- **Develop Fragility Curves for Use in PSA**

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Selection of Critical Components

- Higher Risk Significant Components
- Structures and Passive Components
- Those Most Affected by Aging
  - Past Experience
- Adequacy of Existing Programs

Structures and Passive Components to Consider

Anchorages
Cable Tray Systems
Concrete
Conduit Systems
Containment
Cooling Tower
Electrical Conductors
Exchangers
Filters
HVAC Duct
Insulation/seal
Piping System
RPV
Structural Seismic Gap
Structural Steel
Tanks
Vessels
Water-Control Structures

Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Past US Studies Related to Age-Related Degradation of Structures and Passive Components:

1. Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants; NUREG/CR-6679 by BNL
2. Probability-Based Evaluation of Degraded Reinforced Concrete Components in Nuclear Power Plants; NUREG/CR-6715 by BNL
3. Risk-Informed Assessment of Degraded Buried Piping Systems in Nuclear Power Plants; NUREG/CR-6876 by BNL

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Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSAs

Reinforced Concrete Shear Walls

Development of Seismic Fragility Curves:

- Identify analytical method
- Benchmark analytical method
- Design representative member
- Define limit state/capacity
- Develop structural statistics for member
- Perform fragility analysis
  - Undegraded
  - Degraded – various levels

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Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls – Identify Analytical Method:

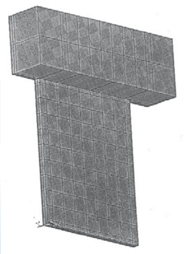
- Computerized Solution
  - Finite Element Method (FEM)
- Empirical Equations
  - Barda et al. Methodology
- Testing

Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls - Benchmark Analytical Method:

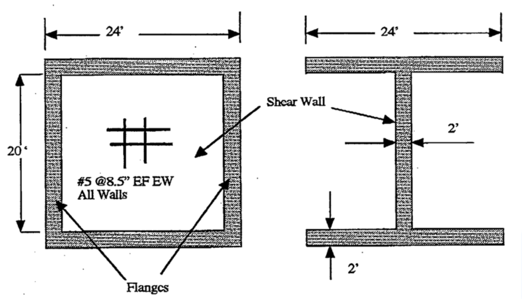
- Testing – Done in Japan

- Computerized Solution
  - Finite Element Method (FEM)

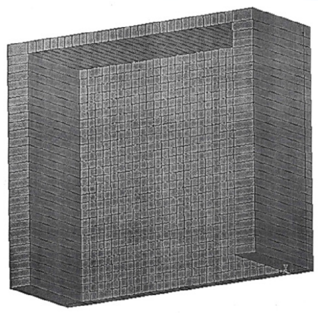


Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls – Design representative member



Representative Shear Wall Design



Finite Element Model of Representative Shear Wall Design

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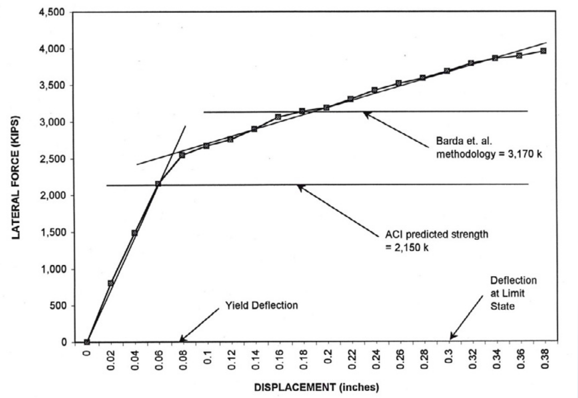
Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls - Define limit state/capacity



Load Deflection Curve

Limit State Equals

4 x Elastic Limit

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Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls - Develop structural statistics for member:

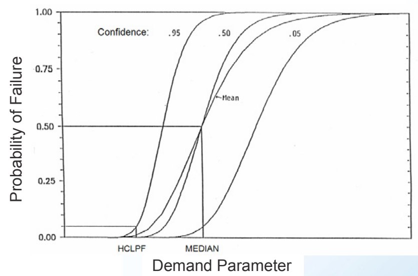
Property	Mean	V_C	CDF
Concrete (4,000 psi)			
Comp. Strength	4,400 psi	0.16	N
Splitting strength	475 psi	0.18	N
Initial tangent modulus	3,834 ksi	0.18	N
Max comp. strain	0.004	0.20	N
Grade 60 reinforcement			
Yield strength	71 ksi	0.10	LN
Modulus of Elasticity	29,000 ksi	NA	NA
Placement of reinforcement			
Effective depth, d	d (in)	0.5/d	N
Analysis Shear (B_{sh})	1.00	0.14	N

Note: 1 in. = 25.4 mm; 1 psi = 6.895 kPa; 1 ksi = 6.895 MPa
Vc = Coefficient of Variation
CDF = cumulative distribution function
N = normal distribution; LN = lognormal distribution; NA = not applicable

Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls – Fragility Analysis:

- Fragility Curve is the conditional probability of failure for a given value of demand (e.g., pga in g's)



$$F_R(x) = \Phi\left[\frac{\ln(x/m_R)}{\beta_R}\right]$$

$\Phi[\ ]$ = standard normal probability integral

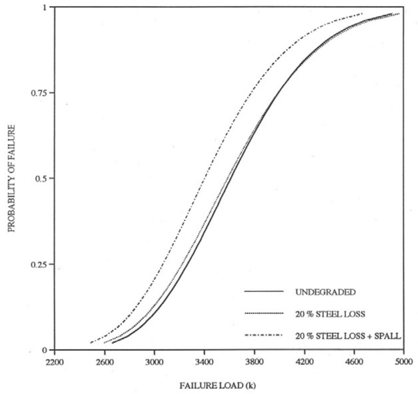
m_R = median capacity

x = demand parameter

β_R = logarithmic standard deviation

Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls – Perform Fragility Analysis:



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Degradation Fragility Curves

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BNL/KAERI Collaboration Program on Aging

- **Objective**: Development of Seismic Capability Evaluation Technology for Degraded Structures and Components
- **Scope:** Specific tasks over a 5 year period to develop seismic fragility methodology of structures and passive components considering aging degradation effects
- **Results:** To be used in support of periodic safety reviews, license renewal applications, and for upgrade of the seismic safety of NPPs in Korea

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BNL/KAERI Collaboration Program on Aging

Scope of Research

- **Year 1: Study of Degradation Occurrences – Completed**
  - Collect and review degradation occurrences in US nuclear power plants
  - Identify important aging characteristics needed for seismic capability evaluations
  - Results documented in BNL Report-81741-2008, KAERI/RR-2931/2008
- **Year 2: Review Time-Dependent Material Degradation Models – Completed**
  - Identify modeling methodologies for the long-term behavior of material degradation in NPPs
  - Focus on the most common time-dependent changes in material properties (e.g., loss of material and cracking)
  - Results documented in BNL Report-82249-2009, KAERI/TR-3757/2009
- **Year 3: Seismic Fragility Analysis – Just Initiated**
  - Intent to demonstrate seismic fragility calculation methodology
  - Select representative structure/passive component and perform fragility analysis – undegraded & degraded conditions
  - Computerized finite element analysis method or closed form solution

BNL/KAERI Collaboration Program on Aging (Cont'd)

Scope of Research

- **Year 4: Technical Assistance to KAERI for Fragility Analysis of Other Structures/Components – Future**
  - Identify important aging characteristics for other structures/components
  - Identify suitable analytical/test methods for determining seismic fragility
  - Perform seismic fragility analyses / process available test data
- **Year 5: Technical Assistance to KAERI for Degradation Acceptance Criteria - Future**
  - Similar to BNL's recent approach for NRC aging research project (see BNL NUREG/CRs)
  - Assist in defining/developing acceptance criteria for seismic risk – one possible source: NRC Regulatory Guide 1.174, Rev. 2, entitled "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"
  - Should consider effects on core damage frequency (CDF) and large early release frequency (LERF)

Issues to Consider for Further Research

- Develop an Operating Experience database for each operating NPP and for each class of operating NPPs
- Initiate and maintain an Operating Experience database for each new-generation NPP and for each class of new-generation NPPs
- Further improve analytical methods to assess fragility and impact on safety that take into account effects of aging
- Incorporate test data on fragility capacity into PSA assessments
- Develop improved and more specific acceptance criteria for degradation on both a deterministic and probabilistic basis
- Improve condition assessment methods and inspection tools to assess potential degradation of structures and passive components - especially in inaccessible areas

Brookhaven Science Associates 21 BROOKHAVEN NATIONAL LABORATORY

Session I-A

Risk Informed Regulation

Session I-A Summary

Chair: Mitsuhiro KAJIMOTO (JNES), Chang-Ju LEE (KINS)

I-A-1. Akihide HIDAKA(JAEA):Recent Revision of Regulatory Guide on Classification of Safety Importance using Risk Information
Mr. Hidaka pointed out Japan's current guideline on the maintenance program using PSA information. Relating with Japan's new inspection program, he explained recent revision of regulatory guide in terms of operating consideration for SSCs with safety functions, as well as the policy for utilization of risk information.

I-A-2. Do Sam KIM (KINS-Korea): Development of the Performance Goals for the Korean Nuclear Power Plants
Since "the policy on the severe accident" in 2001, Korea had to set performance goals considering results of each levels of PSA in NPPs. Mr. Kim presented about the outline & flowchart of the performance goal which has been recently developed for Korean nuclear reactors. Some application principles for utilizing performance goals are suggested.

I-A-3. Dae-Wook CHUNG (KINS-Korea): A Risk-Informed and Performance-Based Approach for Improving Regulatory Inspection Program and System in Korea
Mr. Chung presented about the R&D status for developing integrated safety performance assessment program, as well as suggested overall framework for graded periodic inspection program.

I-A-4. Huichang YANG (ENESYS Co., Ltd.-Korea): Development of Risk Evaluation Program SElF for Inspection Findings
Mr. Yang presented about current status and methodology for developing KlNS SElF (significance evaluation inspection finding) program, as a supporting tool of integrated safety performance assessment program, as previously explained by Mr. Chung.

I-A-5. Tsuyoshi UCIDDA (JNES-Japan): The Improvement of Regulatory Inspection System utilizing "Risk Information" in Japan
Mr. Uchida presented about the direction of current improvement of regulatory inspection system in Japan, on the viewpoint of the utilization of risk information. He explained many applying areas in terms of the utilization of risk information, such as maintenance program, performance criteria, regulatory inspection, etc.

I-A-6. Yong Suk LEE (Future and Challenge-Korea): A Study on the Risk-Informed Performance Indicators and Thresholds for Graded Regulation
Mr. Lee presented about the current status and methodology for developing risk-informed performance indicators, as a supporting tool of integrated safety performance assessment program, as explained by Mr. Chung.

I -A-1

Presented at 10th KJPSA workshop,
May 18 - 20, 2009, Jeju, Korea

Recent Revision of Regulatory Guide on Classification of Safety Importance using Risk Information

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1

Introduction

- "Regulatory guide for reviewing classification of Importance of Safety Functions in Light Water Nuclear Power Reactor Facilities [http://www.nsc.go.jp/NSCenglish/guides/lwr/L-DS-I_01.pdf]", ***Safety Importance Classification Guide*** was deterministically prepared in 1990 by NSC.
  - To provide fundamental criteria to the relative importance of safety functions in applying various requirements for safety design in the process of licensing review.
- The guide, prepared originally for design phase, ***had been also referred to construction and operation phases because the reliability of safety function should be maintained at all phases according to their importance***.
- At operation phase, type or interval of maintenance for SSC in NPPs had been defined based on the guide.

2

Use of risks for inspection

- **New inspection program** was initiated by NISA in January 2009.
  - Linkage with plant life management (PLM)
  - Elongation of time interval of periodical inspection
  - Enhancement of effectiveness in inspection
    - Endorsement of ***the maintenance program*** prepared by utilities based on ***JEAC-4209***
- In the maintenance program, type or interval of the maintenance is defined considering the PSA results and the safety importance classification guide.
- In present framework, if there are deviations between the PSA results and the deterministic based safety importance classification guide then the most conservative safety classification (higher safety class) is applied.
- Possible reason for this conservatism was that the former guide prepared originally for design phase does not describe clearly what concept is kept during operation phase and the policy for utilization of risk information as well as insufficient experiences in this field.

3

Safety Importance Classification Guide (1/2)

Table 1. Classification of Safety Function Importance

Categorization by function / Classification by safety importance		Safety functions	
		Abnormality prevention functions	Abnormality mitigation functions
SSC related to safety	Class 1 Class 2 Class 3	PS-1 PS-2 PS-3	MS-1 MS-2 MS-3
SSC not related to safety	No class	No safety functions	

Class 1: Maintain the highest reliability reasonably achievable
Class 2: Maintain high reliability
Class 3: Reliability equivalent to or higher than ordinary industrial facilities
PS: SSCs that loss of their functions may cause abnormal conditions, leading to undue radiation exposure of the public.
MS: SSCs that have the functions to prevent escalation of abnormal conditions or to mitigate undue radiation exposure of the public.

4

Safety Importance Classification Guide (2/2)

Table Attached to Commentary : Examples of Importance Classification of Safety Function in PWR and BWR

Class	Function	SSC (PWR)	SSC (BWR)
PS-1	1) Reactor Coolant Pressure Boundary	Components and pipelines comprising the reactor coolant pressure boundary	Same as on the left
MS-1	3) Prevention of overpressure in reactor coolant pressure boundary	Pressurizer safety valve (Opening function)	Safety relief valve (Opening function as safety valve)
PS-2	1) Reseating of safety valves and relief valves	Pressurizer safety valve (Related with reseating function)	Safety relief valve (Related with reseating function)

The rest is omitted.

5

Outlines of Revision of Safety Importance Classification Guide

- On March 9, 2009, the guide was partly revised to show the policy for use of risks when the guide is referred to operation phase.
- According to NSC's policy on RIR introduction, present revision should treat use of risks not for design phase but for operation phase.
- Method of revision
  - Main text should not be changed and instead, some statements are ***added to the commentary of the guide***.
  - ***Preparation of relevant document*** which complements the added statements to show detailed policy for use of risks during operation phase.
- Points of issue
  - What requirements are demanded by the guide for operation phase ?
    - No change of safety function class from design to operation phase
    - Table of SSC classification attached to commentary is example.
  - When the maintenance level is determined separately from the SSC classification attached to commentary under above conditions, are there any contradictions to the guide ?

6

Statements added to Commentary

- Contents of commentary of the guide
  - I. Objective
  - II. – IV. (An omission)
  - V. Design Consideration for SSCs with Safety Functions
    - (An omission) Specific measures to be taken for ensuring the required level of reliability in the operation depend on the characteristics, etc. of individual SSCs. Therefore, specific measures to meet the individual reliability requirements shall be adequately determined in the light of the fundamental objectives of this guide.
- Added the following in present revision
  - For example, when the concrete measures or requirements for maintenance of SSC are determined for operation phase, it is adequate to refer to risks such as operational experience and/or PSA results maintaining the safety function specified in this guide. This reflects recent progress in PSA technology as well as the viewpoint of enhancement of scientific rationality, consistency and transparency in nuclear safety and appropriate allocation of limited resources.

7

Highlight of Relevant Document

- It is expected that activities of regulatory body and utilities be more detailed and effective by utilization of risks in operation and higher reliability be maintained by appropriate allocation of limited resources.
- The guide prescribes that classification of safety function should be kept from design to operation phase while table of SSC classification attached to the commentary is an example (beyond the guide).
- Although the level of maintenance for SSC with safety function has been conventionally determined based on examples in the guide, it is more appropriate to be realized considering also risk significance.
- By doing so, even though the SSC is assigned to high safety importance in the examples, the level of maintenance can be changed to appropriate method in case of low risk significance as far as the safety function is kept.
- It is preferable for utilities to use PSA that reflects actual design and operation management considering voluntarily performed AM.
- Adequacy of PSA results and level of maintenance determined by reference to risks should be sufficiently confirmed by both utilities and regulatory body.

8

Example of Use of risks for inspection

System	Components	Risk significance	Safety importance classification guide	Maintenance significance
High press. core spray	Motor valve Strainer, etc.	High Low	MS-1	**High**
Low press. core spray	Motor valve Strainer, etc.	Low	MS-1	**High (present)** **Low (future)**

Criteria for risk significance

High: F-V ≧ 0.005 or RAW ≧ 2, Low : F-V ＜ 0.005 and RAW ＜ 2

Maintenance Significance for SSC → High → Preventive maintenance - Time or condition based maintenance

Maintenance Significance for SSC → Low → Breakdown maintenance partially including preventive maintenance

Utilization of risk information in the maintenance program would enable further effective and individually-targeted maintenance / inspection for SSC.

9

Summary

- The present revision of the safety importance classification guide clarified that what the guide requires for operation phase is the preservation of safety functions determined at design phase.
- As long as the safety functions are maintained, the maintenance level for SSCs can be determined (including downgrade) using risks separately from the classification examples in the guide and it does not contradict the fundamental policy of the guide.
- Present revision showed the way to utilization of risk information which allows that the less conservative safety classification between the guide and PSA results can be applied to the maintenance significance for SSCs if the trouble information data and experiences are accumulated in future.

10

I -A-2

Development of the Performance Goals (Draft) for Korean Nuclear Power Plants

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

Table of Contents

I. Background
II. Selection of the performance measures
III. Development of the performance goal criteria
IV. Application principles
V. Summary

I. Background

I.1 Policy on the Severe Accident (2001.8)

■ **Safety Goals** (Quantitative Health Objectives: QHOs)

- ✓ The risk of prompt fatalities to an average individual in the vicinity of the nuclear power plant (NPP) should not exceed 0.1% of the risks resulting from other accident.
- ✓ The risk of cancer fatalities to the population in the area of nuclear power plant should not exceed 0.1% of the sum of cancer fatality risks from all other causes.

■ **To facilitate the achievement,**

- ✓ **The performance goals** will be set considering prevention of core damage and reduction of radioactive material release

I.2 Roles of the performance goals

■ **Needs to decide whether a NPP satisfy the safety goals**

- ✓ **Risk surrogates that could be directly compared to the QHOs**
- ✓ **Satisfaction of the performance goals means**
  - → achievement of the safety goals.
  - → meeting the engineering objectives for the operation and design of NPPs

■ **Criteria for RIR/RIA**

■ **Requirements for the performance goals**

- ✓ **"Prevention of core damage and reduction of radioactive material release" (The policy on severe accident)**
- ✓ **Results at each levels of PSA can be the performance measures.**

※ **PSA of NPPs**

- Level 1 PSA : Core damage frequency, etc.
- Level 2 PSA : Determination of release frequency, etc.
- Level 3 PSA : Risks to the public, etc.

II. Selection of the performance measures

II.1 Selected Measures

- **Must conform to the policy on severe accident**
  - ✓ **Prevention of core damage and reduction of radioactive material release**
- **Considerations**
  - ✓ **Representative parameters of facilities related to the core integrity (Level 1 PSA) and containment isolation function (Level 2 PSA)**
  - ✓ **Should be clearly defined to easily quantify.**
  - ✓ **Sufficient international application experience and following domestic practice in the similar application.**

⇨ **Selected Measures for the performance goals**

- First measure: **Core Damage Frequency (CDF)**
- Second Measure : **Large Early Release Frequency (LERF)**

II.2 Core Damage Frequency (CDF)

- **Definition**

  : the frequency of an accident which can cause the fuel in the reactor to be damaged

  ✓ **PWR**
  - **Core damage corresponds to the case where a peak clad temperature goes above a threshold criteria**
  - **In Korea, the threshold is 1204 ℃ (Notice of MEST 2008-16: Performance of ECC system)**

  ✓ **CANDU**
  - **Severe core damage: failure of two or more fuel channels**

II.3 Large Early Release Frequency (LERF)

■ **Definition**

✓ the frequency of those accidents leading to **significant, unmitigated releases** from containment in a time frame **prior to effective evacuation** of the close-in population such that there is a potential for early health effects (NUREG/CR-6595)

■ **Definitions in the Application**

Method 1:

⇨ **NUREG/CR-6595, Appendix A.2 First definition**

- ✓ Early containment failure
- ✓ Containment bypass
- ✓ Containment isolation failure

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

Method 2: LERF definition 2

: Accidents with short (<6 hours) delay time from core uncovery to atmospheric release of radioactive materials based on PSA results.

= Case 1 + Containment failure before reactor vessel break

❖ **This case obtained by investigating the available evacuation time from the data reported by the licensee's Level 2 PSA. They are as follows.**

1) Early containment failure
: core uncovery ~ containment failure : 1.75 ~ 6.6h
2) Containment bypass
: core uncovery ~ atmospheric release : 0.1 ~ 6.4h
3) Containment isolation failure
: core uncovery ~ atmospheric release : 0.3 ~ 0.77h
4) Containment failure before reactor vessel break
: core uncovery ~ atmospheric release : 0.5 ~ 1.0 h

Method 3: LERF definition 3

⇨ **NUREG/CR-6595, Appendix A.2 Second definition**

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

Method 4: LERF definition 4

⇨ **NUREG/CR-6595, Appendix A.2 third definition**

: LERF is the frequency of early failure and bypass containment failure modes that have a release fraction of iodine equal to or greater than about 10%.

III. Development of the Performance goal criteria

III.1 Procedures

1. Evaluation of the risks corresponding to safety goals

- ✓ **The risk of accident and the risk of cancer fatality from other causes (obtained based on the statistical data: 1983~2006)**

2. Evaluation of the conditional probability of prompt and cancer fatality

3. Determination of the Performance goals

- ✓ **LERF criteria to satisfy the goal of early fatality**
- ✓ **CDF criteria to satisfy the goal of cancer fatality**

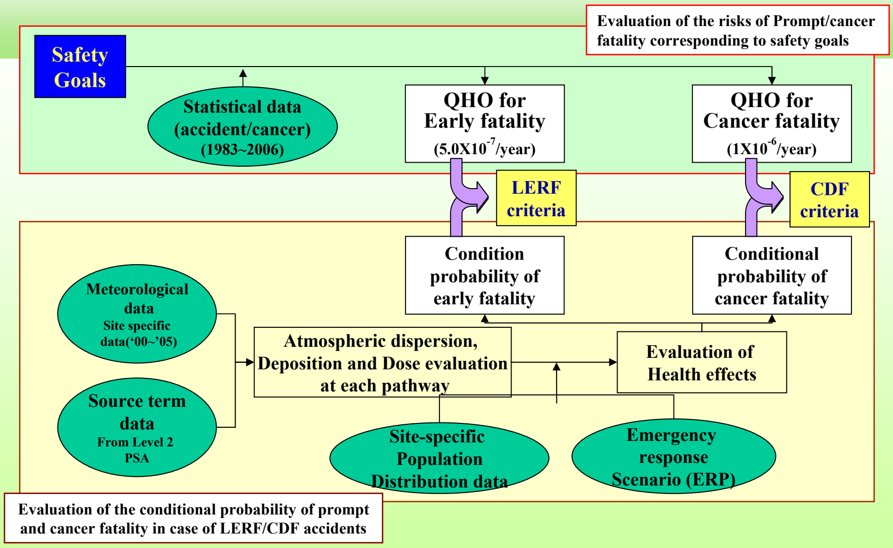

Fig. 1. Flowcharts for the development of performance goal criteria

III.2 Evaluation of the risks corresponding to safety goals

■ Risk of early and cancer fatalities in Korea

(From KNSO, 1983~2006)

✓ Accident fatality

- **69.35 per 100,000 annually (Fig. 2)**
  : Average risk = 6.935×10^{-4}/y, 6.0×10^{-4}/y after the year 2000
- **Risk of early fatality corresponding to the safety goal**
  : 6.0×10^{-4}/y$\times$0.1% = 6×10^{-7}/y
  : Conservatively, 5×10^{-7}/y selected as a goal

✓ Cancer fatality

- **111.49 per 100,000 annually (Fig. 2)**
  : Average risk = 1.115×10^{-3}/y
  : It is continuously increasing: 7.2×10^{-4}/y$\rightarrow$ 1.37×10^{-3}/y
- **Risk of cancer fatality corresponding to the safety goal**
  : 1.37×10^{-3}/y$\times$0.1% = 1.37×10^{-6}/y
  : Conservatively, 1×10^{-6}/y selected as a goal

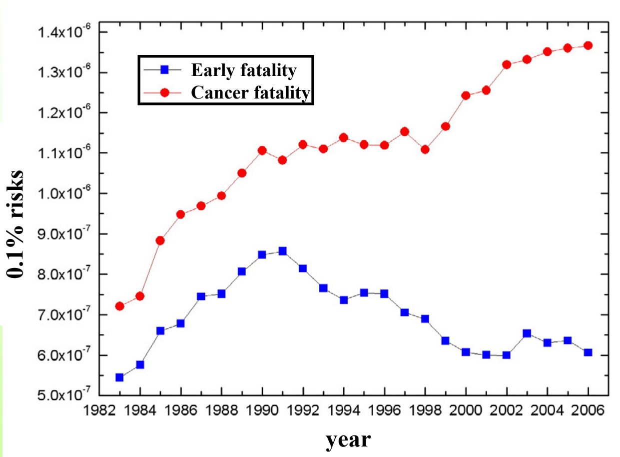

Fig. 2. 0.1% risks of the accident and cancer fatality

III.3 LERF criteria to satisfy the goal of early fatality

- **Individual Early Risk (IER)**

$$IER = \sum_{1}^{N} LERF_n \times CPEF_n = LERF \times CPEF_{AVG} \qquad (1)$$

where,

$LERF_n$ **: frequency of the release capable of causing early fatalities for LERF accident sequence "*n*"**

$CPEF_n$ **: conditional probability of early fatality**

$$LERF = \sum_{1}^{n} LERF_n$$

$$CPEF_{AVG} = \sum_{1}^{N} \frac{LERF_n}{LERF} \times CPEF_n$$ **: average of CPEFs**

■ Conditional probability of early fatality ($CPEF_n$)

✓ Conditional probability of an individual becoming a prompt (or early) fatality for an accident sequence "*n*"

$$CPEF_n = \frac{EF_n}{TP(\mathbf{1.6}km)} \quad \textbf{(2)}$$

where,

EF_n : number of early fatalities within 1.6km conditional on the occurrence of accident sequence "*n*"

$TP(\mathbf{1.6}km)$: total population to 1.6 *km*

✓ **Evaluation of $CPEF_n$ using MACCS2 code based on**

- **Source term information derived from Level 2 PSA, and**
- **Site-specific meteorological data, population distribution with emergency response scenario**

■ Derivation of LERF criteria

✓ **CPEF for the determination of performance goal :**

- **For conservatism, we selected the CPEF of the case only with dose dependent relocation without evacuation for internal events**
- **Various definitions of LERF does not make significant differences in CPEF, thus, in this study, we used the LERF definition 2 (Available time for evacuation < 6h).**

✓ **Performance goal for the early fatality**

$$IER = LERF \times CPEF < 5.0 \times 10^{-7} / y$$

- **CPEF : 1.63×10^{-3}(Wolsong) ~ 7.0×10^{-2}(Ulchin3,4)**

⇨ **LERF : 7.0×10^{-6}/y~ 3.0×10^{-4}/y**

III.4 CDF criteria to satisfy the goal of cancer fatality

■ Individual Latent Risk (ILR)

$$ILR = \sum_{1}^{N} F_n \times CPLF_n = CFF \times CPLF_{AVG} \quad \textbf{(3)}$$

where,

F_n **: frequency of occurrence of accident sequence "*n*"**

$CPLF_n$ **: Conditional probability of cancer fatality**

$CFF = \sum_{1}^{n} F_n$ **: Containment failure frequency**

$CPLF_{AVG} = \sum_{1}^{N} \frac{F_n}{CFF} \times CPLF_n$ **: Average value of CPLFn**

■ Conditional probability of cancer fatality ($CPLF_n$)

✓ Conditional probability of cancer fatality for an accident sequence "*n*"

$$CPLF_n = \frac{LF_n}{TP(8.0km)} \quad \textbf{(4)}$$

where,

LF_n **: number of cancer fatality within 8.0 km**

$TP(8.0km)$ **: total population to 8.0 *km***

✓ **Evaluation of $CPLF_n$ using MACCS2 code based on**

- **Source term information derived from Level 2 PSA, and**
- **Site-specific meteorological data, population distribution with emergency response scenario**

■ **Derivation of CDF criteria**

✓ **Selection of CPLF for the performance goal**

• **The selected criteria must be used for all Korean NPPs**

: For conservatism, the case with no protective action was selected.

: CPLF for Kori 2 (6.25×10^{-3}) were selected

✓ **Performance for the cancer fatality**

$$ILR = CFF \times 6.25 \times 10^{-3} < 1.0 \times 10^{-6} / y$$

: if we set the conditional containment failure probability (CCFP) equal 1, then

⇒ CFF = CCFP X CDF = CDF

⇨ Core Damage Frequency (CDF) < 1.0×10^{-4}/년

IV. Application Principles

IV.1 Application level

Safety targets or objectives rather than strict limits

❖ **Due to**

✓ **the Uncertainty and insufficient analysis range of PSA**

✓ **Consideration of the plant designed before PSA application**

✓ **Not sufficient experience of the risk application**

■ **Application area of the performance goals**

✓ **Risk criteria for the NPP design and operation**

✓ **Criteria for RIR/RIA**

✓ **Parameters representing the plant safety level**

IV.2 Application principles

- **CANDU plants**
  - ✓ **Although the design concepts are different,**
  - ✓ **Same performance goals to the PWR should be used because the defense in-depth concept (prevention of core damage and mitigation of the atmospheric release) applies also.**

- **Definition of core damage for the performance goal**
  - **PWR : Peak clad temperature >1204℃**
  - **CANDU : Failure of two or more fuel channels**

IV.2 Application principles (continued)

- **New plants**
  - ✓ **The increase of risk due to the addition of new nuclear power plants should be low as much as possible.**
  - ✓ **Goals : One tenth of that of operating nuclear power plant**
- **Initiating events**
  - ✓ **All initiating (internal, external) events must be considered. (except security and physical protection)**

V. Summary

- **Development of the performance goals**
  - ✓ Draft :
    - CDF : 1.0 x 10^{-4}/y (same as the IAEA criteria)
    - LERF : 7.0x10^{-6}/y~ 3.0x10^{-4}/y (Need further research)
- **Future works**
  - ✓ **Need to clarify the following topics**
    - **CDF definition in CANDU reactor**
    - **Uncertainties (especially related to the Level 2 and 3 PSA)**
    - **Consideration on the application of initiating events**
    - **Application strategy of performance goals to actual plants.**

I -A-3

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

A Risk-Informed and Performance-Based Approach for Improving Regulatory Inspection Program in Korea

2009. 5. 18

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CONTENTS

1. **Introduction**
   - Current Status and Perspective of Nuclear Power in Korea
   - Summary of PSA and Risk-Informed Activities
2. **Approach to Improving Regulatory Inspection Program**
   - Overall Framework for Graded Regulation
   - Developments of Risk-Informed Periodic Inspection (RIPI) Program
   - Integrated Safety Performance Assessment (ISPA) Program
3. **Graded Periodic Inspection (GPI) Program**
4. **Development of Individual ISPA Sub-Programs**
   - KINS-SEIF Program
   - Risk-Informing KINS Safety Performance Indicator (SPI) Program
5. **Future Works**

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2

1. Introduction

Current Status and Perspectives of Nuclear Power

- 20 NPPs are in operation
  - Nuclear power accounts for around 40% of national electricity supply
- 4 more NPPs are under construction (APR-1400, advanced type)
- Furthermore, national energy strategy is announced recently
  - Number of NPPs will be almost doubled by 2030
  - Nuclear power will account for more than 60% of national electricity supply
- As regulator, the KINS needs more effective and efficient way in nuclear safety regulation through;
  - Managing and improving safety more effective way
  - Distributing regulatory resources more efficient way
  - Encouraging the licensees to voluntarily improve safety performance

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1. Introduction

Summary of PSA and Risk-Informed Activities

- Since 1989, PSA has been one of licensing submittals for new NPP
- By "Severe Accident Policy Statement", at least level 2 PSA for all operating nuclear units had been completed in 2006, and subject to periodic update
- In parallel, extensive R&D programs have been underway by both KINS and industry for more than 10 years
- Since 2006, the KINS has been working on the development and implementation of risk-informed regulation as appropriate, mostly for
  - Review of licensee application to RI-ISI and RI-STI/AOT changes and
  - Improving regulatory inspection program

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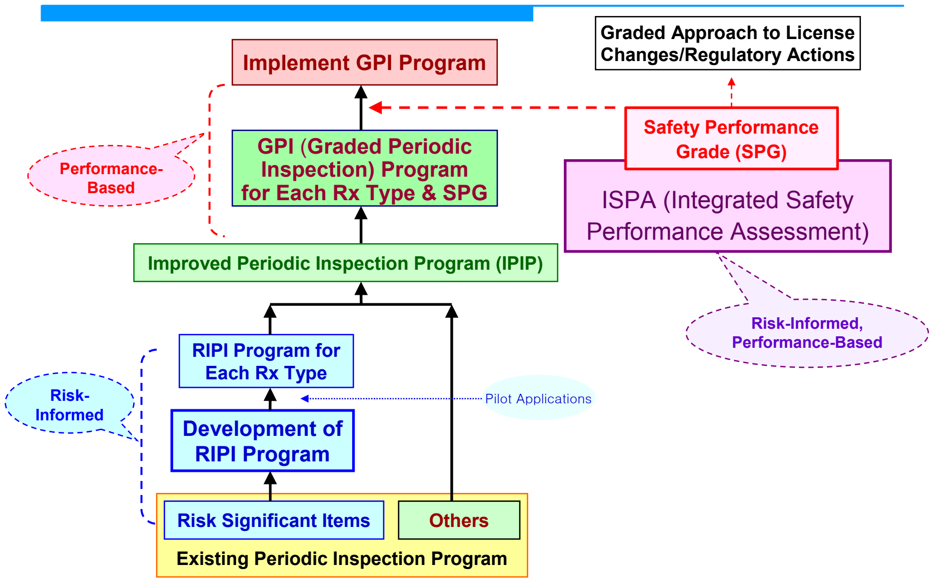
2. Approach to Improving Regulatory Inspection Program



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2. Approach to Improving Regulatory Inspection Program

❑ Developments of Risk-Informed Periodic Inspection (RIPI) Program

- Basic Ideas
  - incorporating risk-significant failure events into related inspection item, and
  - Adjusting inspection resources commensurate with risk significance and performance records
- Key improvements are focused on those inspection items related to the prevention (or minimization) of highly risk significant
  - common cause failure events,
  - post-accident operator errors (i.e., errors during EOP performance) events, and
  - root causes of independent failure events
- RIPI program has been developed and incorporated into the regulatory inspection program for all 20 operating NPPs since 2006

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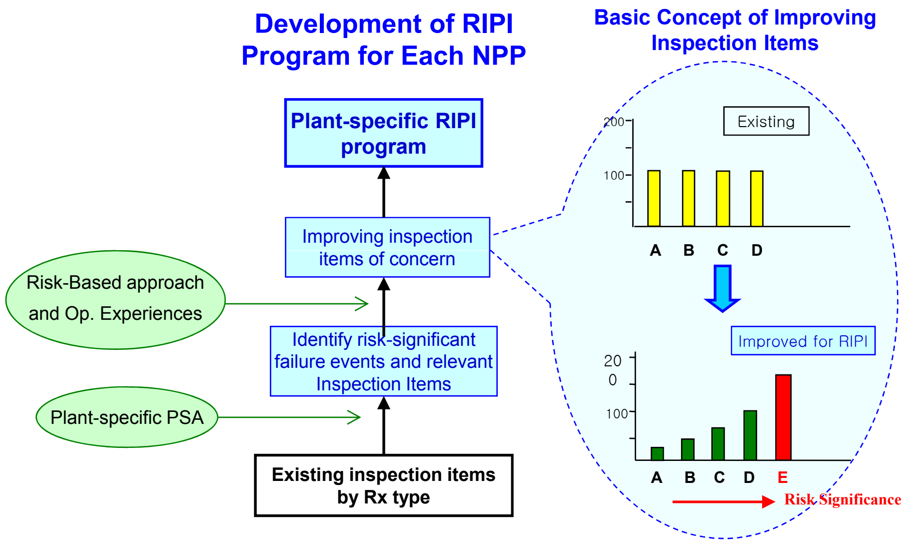
2. Approach to Improving Regulatory Inspection Program



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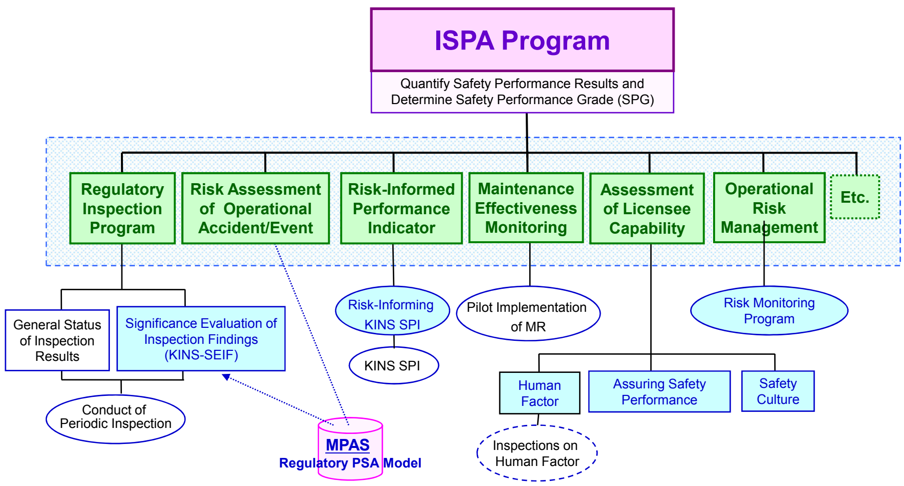
7

2. Approach to Improving Regulatory Inspection Program

❑ Integrated Safety Performance Assessment (ISPA) Program



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8

3. Graded Periodic Inspection (GPI) Program

❑ Basic Ideas

- Determine the "safety performance grade (SPG)" of each NPP based on the result of ISPA program

  3 safety performance grades : **Excellent** <20% **Average** 70-100% **Poor** <10%

- The inspection program is differentiated by the SPG of each NPP

  Excellent Grade : Conduct relaxed IPIP* (benefit)

  Average Grade : Conduct IPIP

  Poor Grade : Conduct Enhanced IPIP**

  * Inspection items are relaxed and Inspection resources are decreased (~20%)

  ** Inspection Items remain the same and inspection resources are increased (~30%)

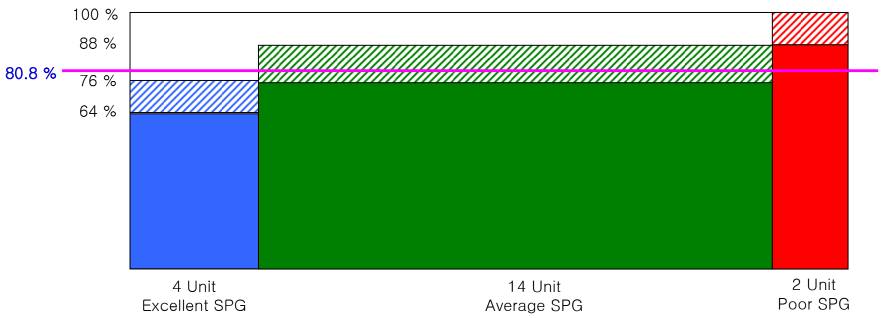
- With GPI program, it is expected the KINS is able to
  - Managing and improving safety more effective way
  - Distributing regulatory resources more efficient way
  - Encouraging the licensees to voluntarily improve safety performance

3. Graded Periodic Inspection (GPI) Program

❑ Estimated Savings in Inspection Man-days (Example)

SPG (# of Units)	Minimum Man-days	Average Man-days	Maximum Man-days
Excellent (4)	256	280	304
Average (14)	1064	1148	1232
Poor (2)	176	188	200
Total (20)	1496 (74.8%)	1616 (80.8%)	1736 (86.8%)

– 100 man-days are assumed to be needed per each unit with Existing Inspection Program)



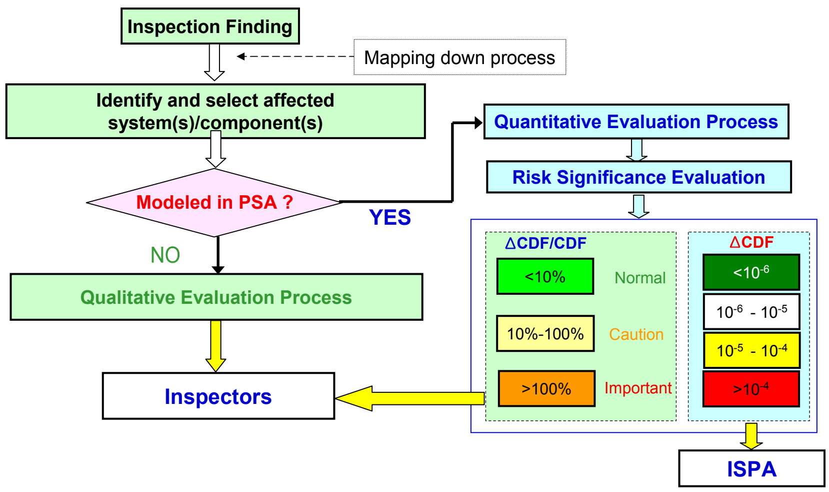
4. Development of Individual ISPA Sub-Programs

❑ KINS-SEIF Program

- KINS SEIF stands for "KINS Significance Evaluation of Inspection Findings"
- PC-based fast-running, easy-to-handle computer program for the inspectors to evaluate risk significance of inspection findings
  - The inspector himself/herself can perform the evaluation at the site
  - For every inspection finding, both qualitative and quantitative evaluations are available in the program provided the affected component is modeled in PSA
  - Multiple components/systems affected by the inspection finding can be selected for calculating ΔCDF of inspection finding
  - The inspector can be aware of the risk insights of inspection finding he identifies
  - All evaluations are reported and managed in KINS headquarter via internet
- Two types of risk information are produced
  - Absolute ΔCDF (and color) of inspection finding regardless of base CDF for providing input to ISPA process (subject to detailed evaluation by risk analyst)
  - Relative portion of ΔCDF of inspection finding to base CDF for providing insight to inspectors
- The development of KINS-SEIF program is completed and beta version will be released to selected inspectors in May 2009 for final comments

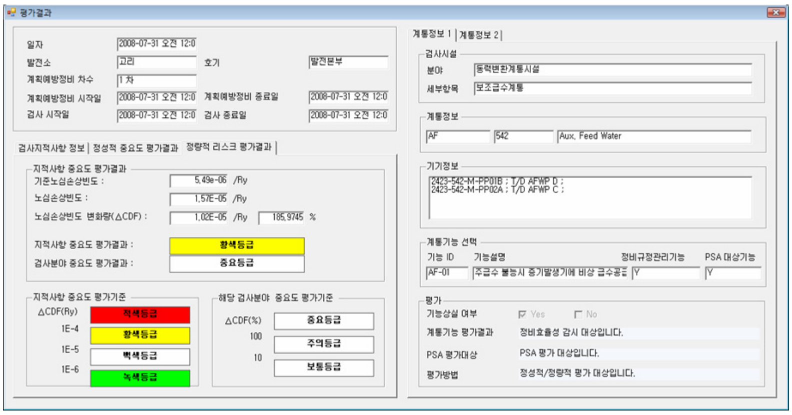
4. Development of Individual ISPA Sub-Programs

Evaluation Process of KINS-SEIF Program



4. Development of Individual ISPA Sub-Programs

Snapshot of KINS-SEIF Program



4. Development of Individual ISPA Sub-Programs

❑ Risk-Informing KINS Safety Performance Indicators (SPI) Program

- KINS already operates safety performance indicators (SPI) program to measure the safety performance status of each licensee and it is open to the public
- The SPI program is being risk-informed using MSPI and USwC approaches for the use in the ISPA program as follows;

Risk-Informed SPI		KINS SPI (Existing)
Category	**Performance Indicator**	
Initiating Events	Simple Reactor Trip	Reactor Trip
	Unplanned Scram with Complication (USwC)	
	Power Change	Power Change
Mitigating System	MSPI – EDG System	EDG System
	MSPI – HPSIS	HPSIS
	MSPI - AFWS	AFWS
	MSPI – RHRS	
	MSPI – CWS	

5. Future Works

- Future works needed
  - Quantification process for each individual ISPA sub-program
  - Determination of weighting factors among sub-programs
  - Finalization of Rx type specific RIPI program
  - Finalization of categorization criteria for safety performance grade (SPG)
  - Others, if necessary

I -A-4

10th Korea-Japan PSA Workshop, Jeju, Korea

DEVELOPMENT OF RISK EVALUATION PROGRAM SEIF FOR INSPECTION FINDINGS

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2009. 5. 18

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http://www.enesys.co.kr

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CONTENTS

- BACKGROUND AND OBJECTIVES
- METHODOLOGY
- FUNCTION AND STRUCTURE OF SEIF
- EXAMPLE APPLICATION
- FUTURE WORKS

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BACKGROUND AND OBJECTIVES

❖ **Background**

- USNRC has been implemented Significance Determination Process (SDP) under Reactor Oversight Process (ROP) for years.
- In Korea, Integrated Safety and Performance Assessment (ISPA) Program, in which the processes to evaluate and assess the safety and performance from quantitative and qualitative view points, are included, is being developed as a part of graded regulation framework.

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BACKGROUND AND OBJECTIVES

❖ **Objectives**

- Significance Evaluation of Inspection Findings (SEIF) has been developed to :
  - ✓ Provide risk significance of inspection findings,
  - ✓ provide information which can be used in grading regulatory activity, and
  - ✓ Provide supporting tool for ISPA as an implementation plan of risk and performance-based graded regulation.

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METHODOLOGY

- **Significance Evaluation Methodology**
  - Identification of Information in Inspection Findings
    - ✓ Contents of inspection findings/SSCs
    - ✓ Major safety functions of identified SSCs
    - ✓ PSA-scoped SSCs
  - Qualitative and/or Quantitative Evaluation
    - ✓ for PSA-scoped SSCs: quantitative evaluation using MPAS/AIMS/FTREX
    - ✓ for non PSA-scoped SSCs: qualitative evaluation
    - ✓ significance is categorized as GREEN/WHITE/YELLOW/RED
    - ✓ significance of inspection fields is categorized as NORMAL/CAUTION/SIGNIFICANT

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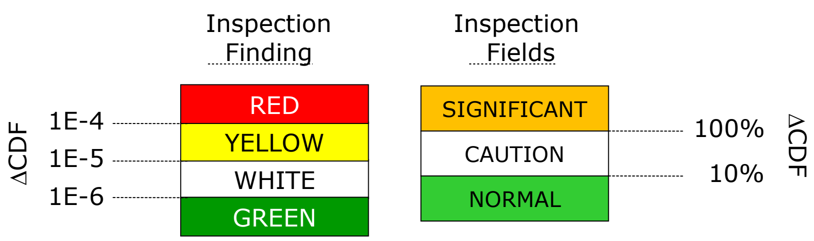
METHODOLOGY (continued)

- **Significance Evaluation Methodology (continued)**
  - Qualitative and/or Quantitative Evaluation (continued)
    - ✓ Significance Determination Criteria



- ✓ Different regulatory action will be imposed depending on the significance category
- ✓ Framework of graded regulation is being developed

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METHODOLOGY

- **Significance Evaluation Methodology**
  - Qualitative and/or Quantitative Evaluation
    - ✓ Qualitative evaluation
      - Evaluate the significance in terms of
        - Initiating events
        - Mitigating systems
        - Integrity of defense-in-depth barriers
      - Phase 1 evaluation of SDP was referred

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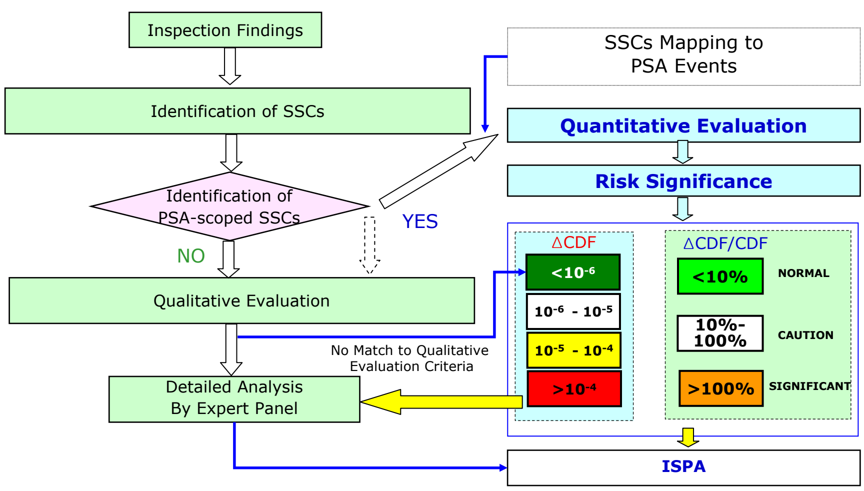
METHODOLOGY

- **SEIF Evaluation Process**



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FUNCTION AND STRUCTURE OF SEIF

❖ **Identification of Inspection Finding**

- Plant/unit
- O/H Info., inspection fields and relevant SSCs
- SSCs status(unavailability) and other assumptions
- Evaluation method selection

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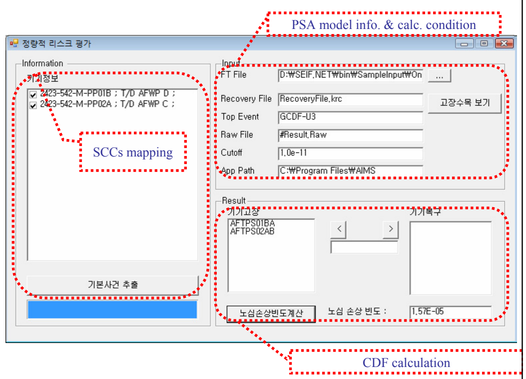
FUNCTION AND STRUCTURE OF SEIF

❖ **Quantitative Evaluation**

- SSCs mapping to events in PSA model
- CDF calculation using regulatory PSA model and AIMS*/FTREX*

* Developed by KAERI



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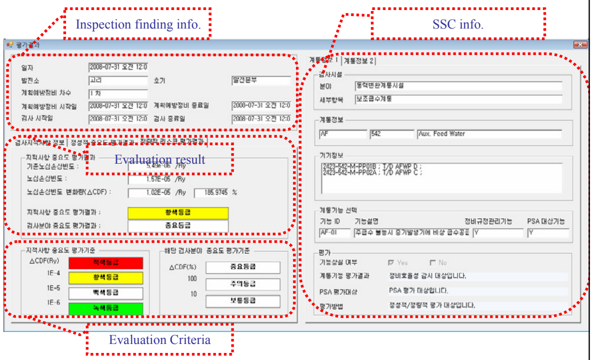
FUNCTION AND STRUCTURE OF SEIF

❖ Quantitative Evaluation Results

- CDF, ΔCDF, evaluated category
- Risk significance of relevant inspection fields



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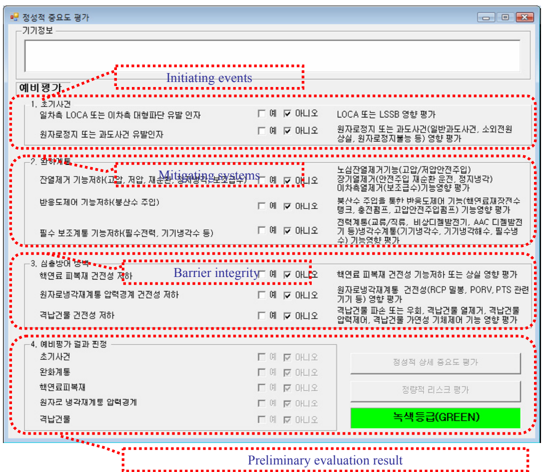
FUNCTION AND STRUCTURE OF SEIF

❖ Qualitative Evaluation

- Initiating events
- Mitigating systems
- DID barrier integrity
- For inspection findings regarded to be significant, those will be transferred to detailed analysis process and/or expert panel



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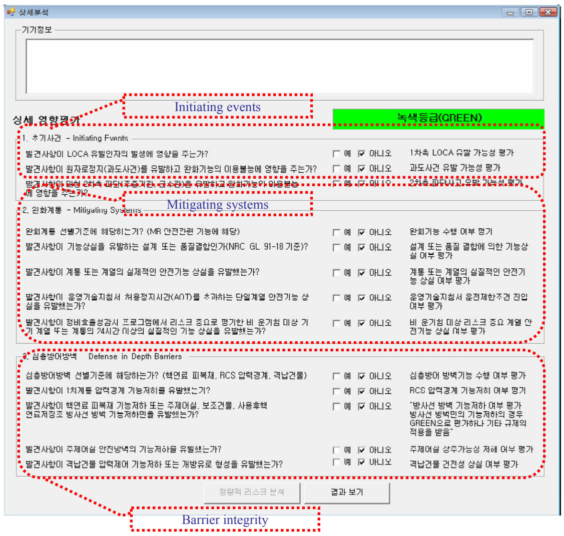
FUNCTION AND STRUCTURE OF SEIF

Qualitative Evaluation

- Detailed inquiries for
  - Initiating events
  - Mitigating systems
  - DID Barrier integrity
- USNRC SDP phase 1 process was referred



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EXAMPLE APPLICATION

Application of SEIF to Example Inspection Findings

No.	Inspection Findings	Year	CDF	EVALUATION RESULT	ΔCDF
1	Inappropriate in-service test for safety class valves	2004	7.63E-03	RED	7.62E-03
2	Inappropriate test of minimum flow differential pressure for CS pumps	2005	6.75E-05	YELLOW	6.20E-05
3	Inappropriate temperature control for primary/secondary component cooling water	2006	2.70E-03	RED	2.69E-03
4	Shutdown cooling system sampling valve leakage	2008	-	GREEN	-
5	Lack of procedures for EDG and related equip. surveillance	2007	1.35E-04	RED	1.30E-04
6	Inappropriate activity on the exceed of performance parameter of EDG	2006	1.25E-05	WHITE	7.01E-06
7	Lack of periodic test procedures and lack of periodic tests for condensate vacuum system	2005	1.62E-05	YELLOW	1.07E-05

Note : Reference CDF is 5.49e-06/Ry.

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FUTURE WORKS

- **Methodology Enhancement and Modification**
  - Evaluation process and methodology modification through pilot application
  - Qualitative Significance Evaluation Method Enhancement and Refinement
- **Development of Interface with KINS Information System and Web-based SEIF**
- **Development of Accident/Event Significance Evaluation (ASP) Methodology and Evaluation Module**

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JNES

The Improvement of Regulatory Inspection System Utilizing "Risk Information" in Japan

18th – 20th May, 2009

The 10th Korea-Japan Joint Workshop on PSA

T. Uchida[1], T. Miyazaki[2], Y. Kasagawa[2], M. Sugawara[1], S. Miura[3] and M. Yamashita[1]

1:Probabilistic Safety Assessment Group
Nuclear Safety Analysis and Evaluation Office
Nuclear Energy System Safety Division
2:Inspection Engineering Group
Inspection Affair Division
3:Policy Planning and Coordination Division

Japan Nuclear Energy Safety Organization (JNES)

JNES

Contents

1. Direction of the improvement of the regulatory inspection system
2. Utilization of "Risk Information" to review Maintenance Program
   2.1 Utilization of "Risk Information" to identify "Importance of Systems / Functions for Maintenance"
   2.2 Utilization of "Risk Information" to Performance Criteria (PC)
3. Utilization of "Risk Information" to identify the safety significant activities for safety preservation regulatory inspection
4. Significance Determination Process (SDP)

1

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1. Direction of the improvement of the regulatory inspection system in Japan

Background

- NISA, JNES and utilities have extensively discussed the way of regulatory inspection system for NPPs.

  - Based on the discussion, NISA issued the interim report showing the direction of the improvement of regulatory inspection system for NPPs including the utilization of "Risk Information" to the system. (See the next page)

  - NISA, JNES and utilities have been extensively preparing the "New Inspection System for NPPs", which started since January 2009.
    - ◇ NISA improved the ordinances of METI in order to introduce the new regulatory inspection system. These ordinances were issued on January 2009.
    - ◇ NISA and JNES have developed the requirements on the maintenance program, regulatory review procedures, methodologies for the comprehensive plant evaluation of the safety performance of NPPs and so on.
    - ◇ Utilities have improved industry association level codes / guides and safety preservation rules.

2

JNES

Direction of the improvement of regulatory inspection system of NPPs on the viewpoint of the utilization of "Risk Information"

1. Enhancement of the inspection system for maintenance activities based on the maintenance program

Application of "Risk Information" to review maintenance program

⇒ NISA/JNES uses the above "Risk Information" to……

- The review of the importance of SSCs for maintenance, which utilities established
- The review of performance criteria

2. Introduction of intensive inspection activities focusing upon safety significant preservation activities

Application of "Risk Information" to identify the safety significant activities of utilities, which safety preservation regulatory inspection should be focused on.

- Identification of utilities' activities, which significantly affect risk of NPPs
- Assessment of risk impacts of system configuration controls (e.g. management of outage schedules for maintenance of safety related systems)

3. Ensuring the plant safety intensively through comprehensive evaluation of individual plant features

Application of "Risk Information" to the comprehensive regulatory assessment on safety performance of NPPs

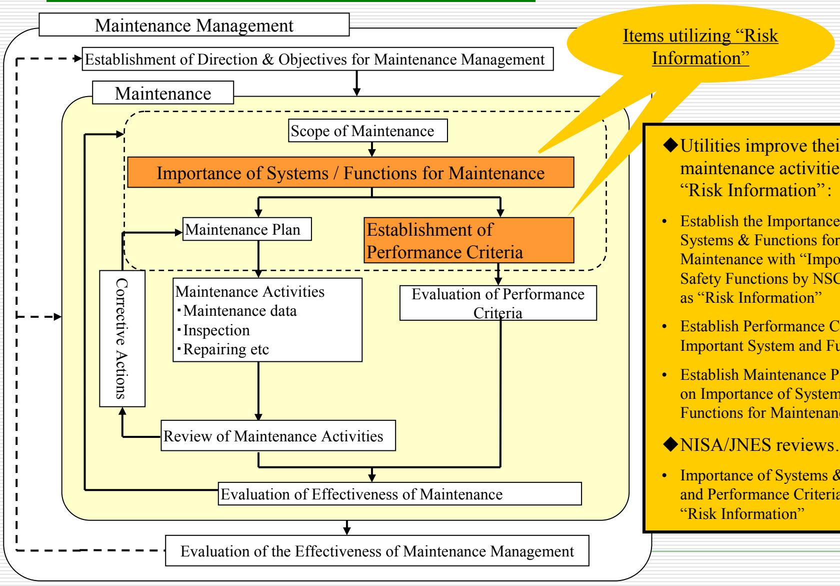
⇒ Development of methods and criteria for the safety performance assessment with "Risk Information"

- Significance Determination Process (SDP)
- Performance Indicators (PIs)

3

JNES

2. Utilization of "Risk Information" to review Maintenance Program



◆Utilities improve their maintenance activities with "Risk Information":

- Establish the Importance of Systems & Functions for Maintenance with "Importance of Safety Functions by NSC" as well as "Risk Information"
- Establish Performance Criteria for Important System and Functions
- Establish Maintenance Plan based on Importance of Systems & Functions for Maintenance

◆NISA/JNES reviews…

- Importance of Systems & Functions and Performance Criteria with "Risk Information"

4

NSC: Nuclear Safety Commission

JNES

2.1 Utilization of "Risk Information" to identify "Importance of Systems / Functions for Maintenance"

- JEAC4209-2007* requires to establish the importance of Systems / Functions for maintenance according to "Importance of safety functions issued by NSC " as well as "Risk Information"

NISA/JNES review on the following viewpoint:

① "Importance of Systems / Functions for Maintenance" should be established based on "Risk Information" as well as "Importance of Safety Functions" issued by NSC

② The following systems / functions should be identified as important for maintenance:

- Systems / Functions, which are classified to Class-1 & 2 of "Importance of Safety Functions"
- Systems / Functions of with "High" risk importance

*JEAC4209-2007 : "Maintenance Code of NPPs" ,Japan Electric Association , endorsed by NISA in 2008

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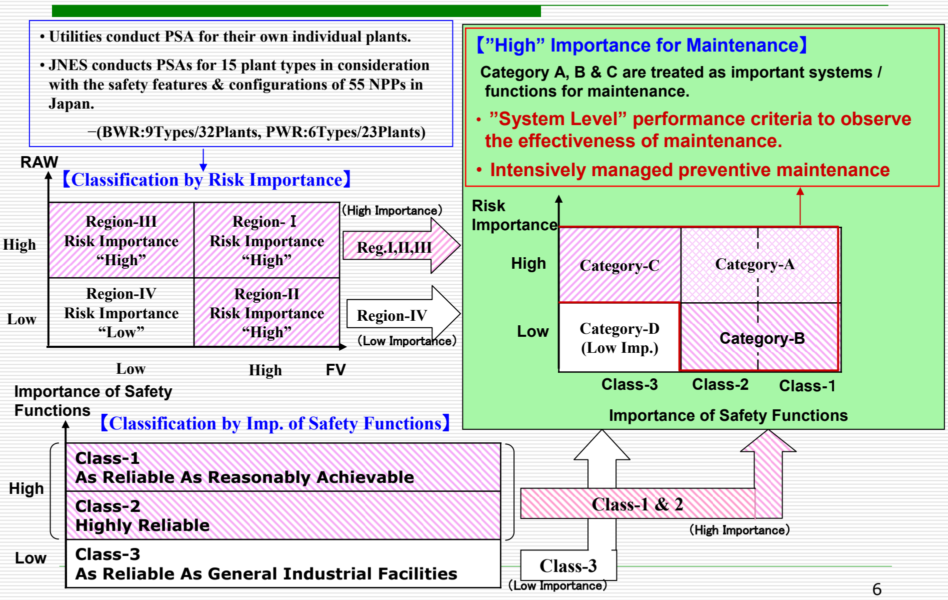
"Importance of Systems / Functions for Maintenance"



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2.2 Utilization of "Risk Information" to Performance Criteria (PC)

☐JEAC4209-2007 requires to establish the Performance Criteria (PC), which shows the performance of plant level and system level, based on "Importance of Systems / Functions for Maintenance".

☐JEAC4209-2007 requires to establish the quantitative objectives for PC considering operation experience (e.g. system unavailability), the importance for maintenance and Allowable Outage Time (AOT).

NISA/JNES reviews on the following viewpoint:

① PCs should consist of "Plant Level" & "System Level" criteria

② Followings should be included in "Plant Level" criteria
- ✓ Number of "Unplanned SCRAMs"
- ✓ Number of "Unplanned Power Changes (>5% of rated power)
- ✓ Number of "Unplanned ESF Actuations"

③ "System Level" criteria should consist of MPFF & UA-H
- ✓ Criteria should be assigned to the systems / functions of "High Importance for Maintenance"
- ✓ Objectives of criteria should be assigned in consideration with "Importance of Maintenance", AOT & ICCDP corresponding to objectives UA-H

MPFF: Maintenance Preventive Functional Failure UA-H: Unavailability hours

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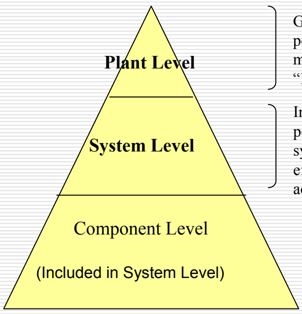
An example of Performance Criteria



Generic criteria of plant safety performance provided by maintenance activities regardless of "Importance for Maintenance".

[Examples]
Number of unplanned SCRAMs/7000hrs
Number of unplanned power changes/7000hrs
Number of unplanned actuation of ESF

Individual indicators of the performance of safety important systems / functions, showing the effectiveness of maintenance activities for safety.

【Example】
Unavailability hours (UA-H)

Outage Time of Mitigation Systems (stand-by-system only)

Maintenance Preventive Functional Failures (MPFF)

Number of functional failure of components which can cause system functional failures, which could be caused by such as poor maintenance.

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*: JEAG4210 "Maintenance Guide of NPPs", JEA issued this guide in 2007.

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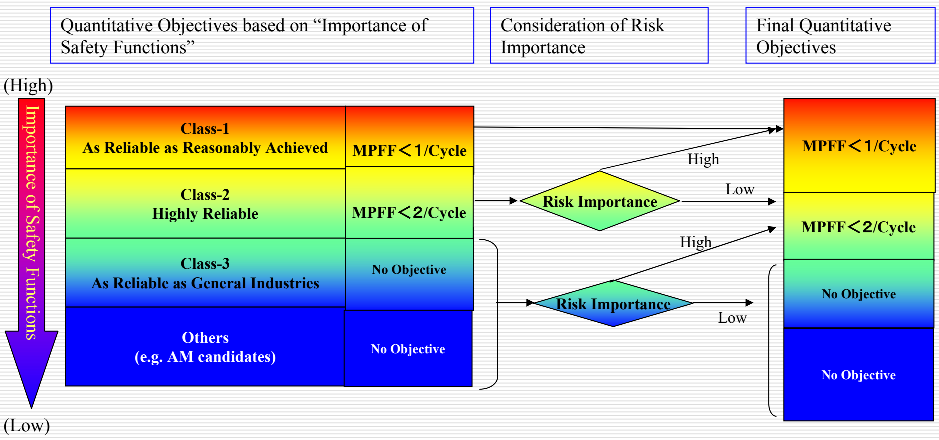
An Example of quantitative objectives of PC (MPFF)



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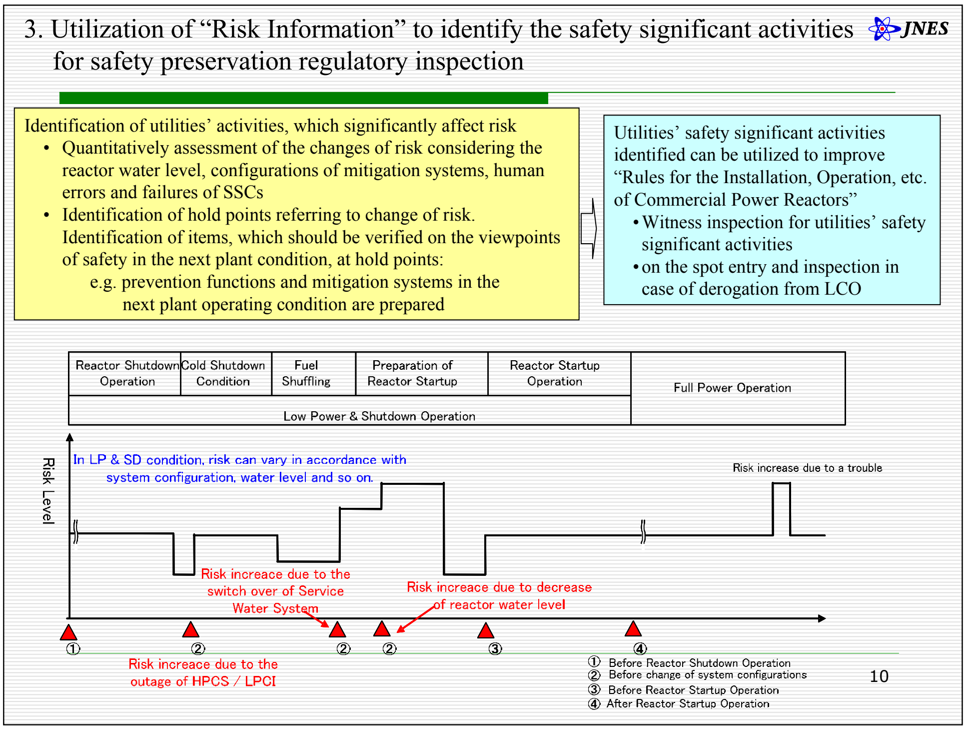
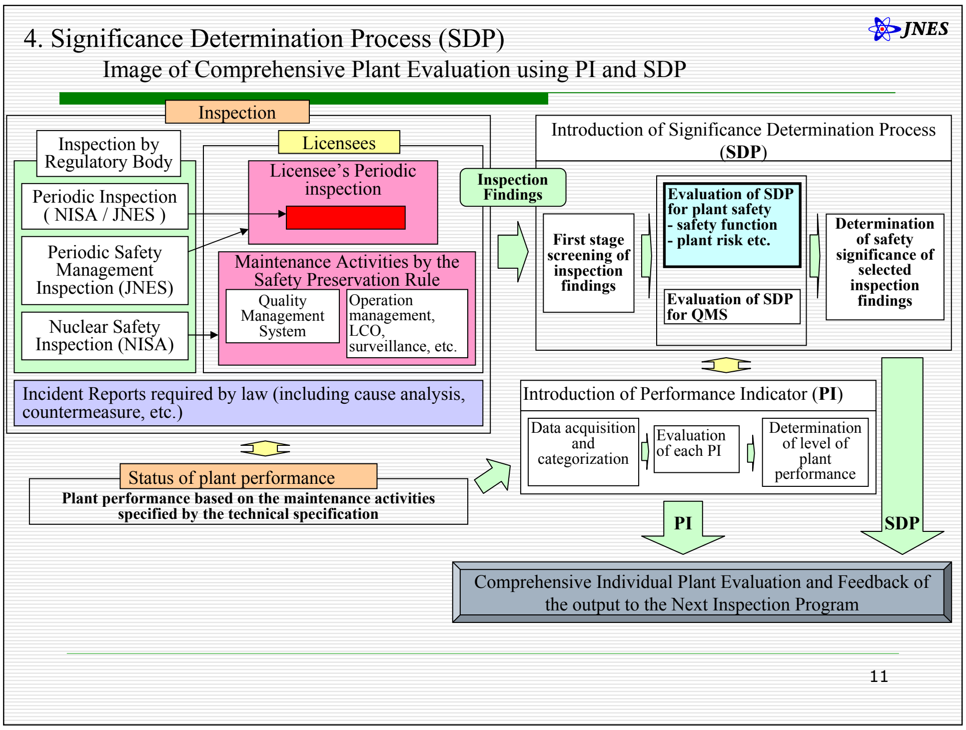
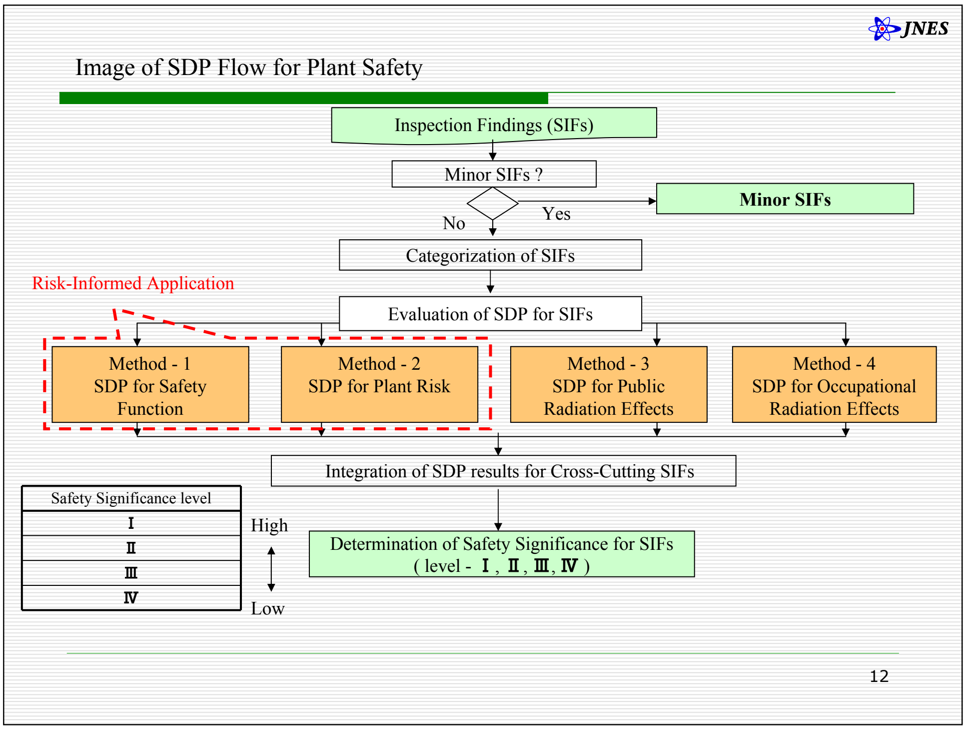
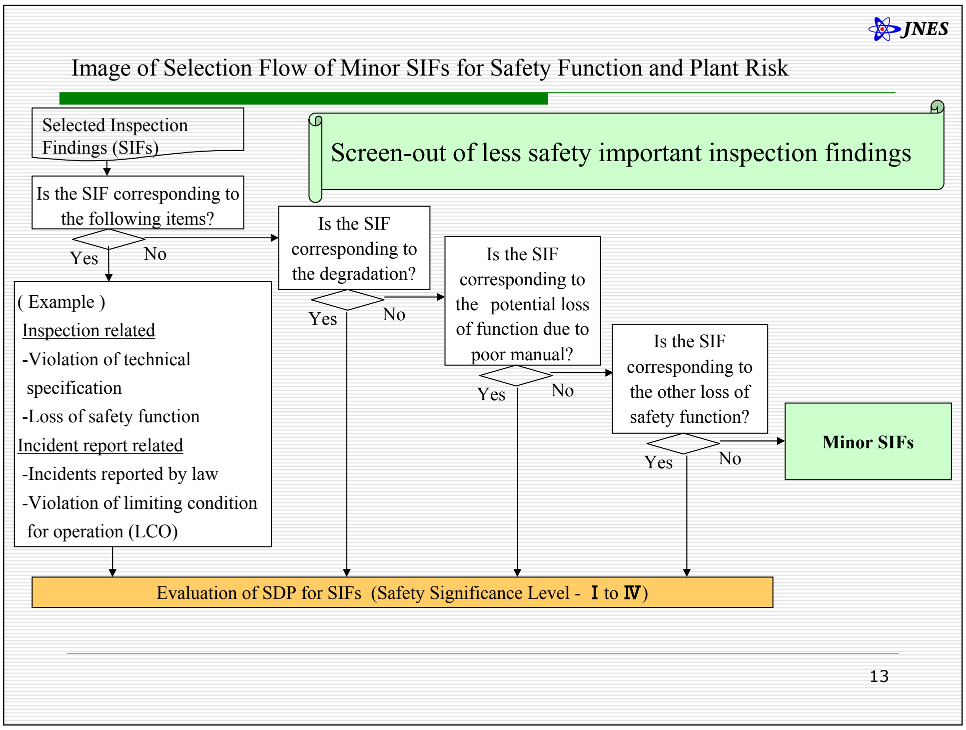
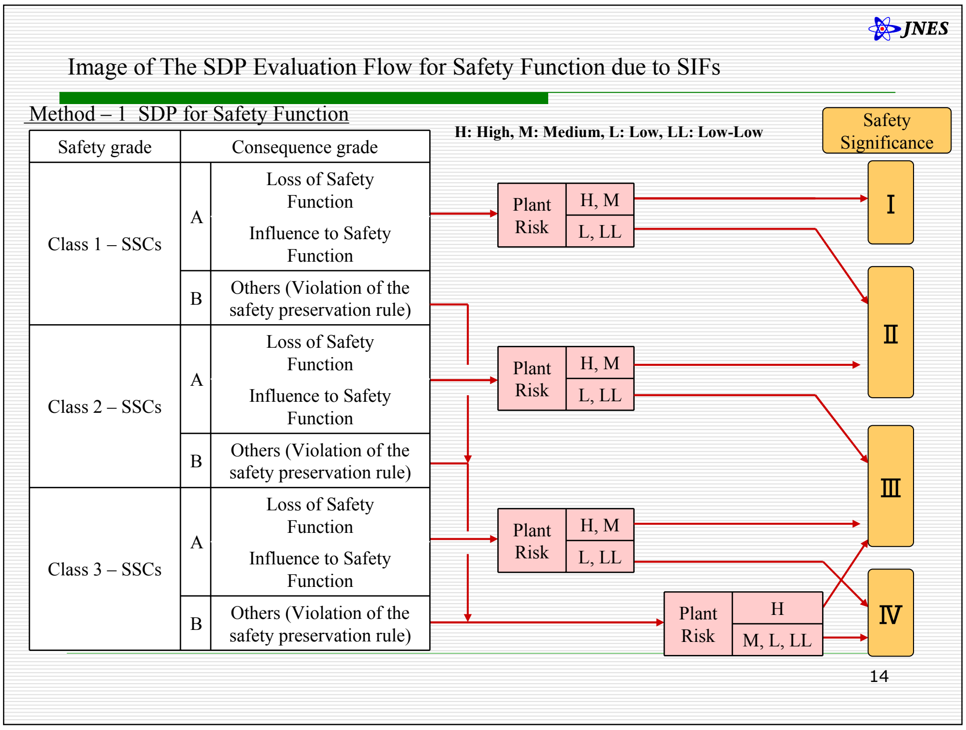
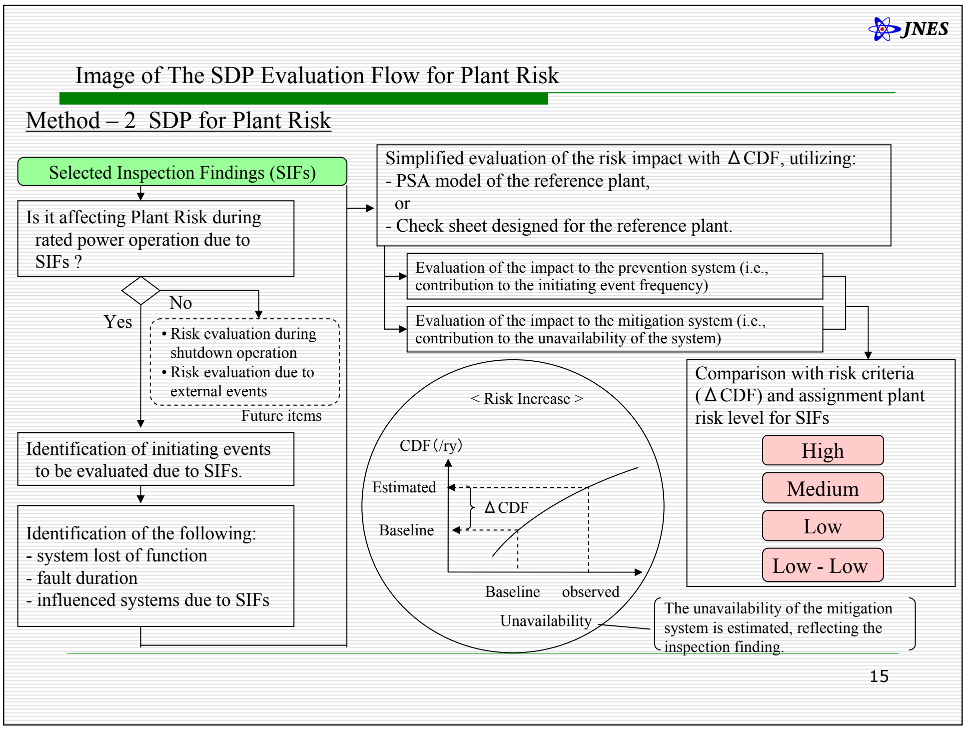

3. Utilization of "Risk Information" to identify the safety significant activities for safety preservation regulatory inspection
JNES
Identification of utilities' activities, which significantly affect risk
• Quantitatively assessment of the changes of risk considering the reactor water level, configurations of mitigation systems, human errors and failures of SSCs
• Identification of hold points referring to change of risk. Identification of items, which should be verified on the viewpoints of safety in the next plant condition, at hold points:
e.g. prevention functions and mitigation systems in the next plant operating condition are prepared
Utilities' safety significant activities identified can be utilized to improve "Rules for the Installation, Operation, etc. of Commercial Power Reactors"
• Witness inspection for utilities' safety significant activities
• on the spot entry and inspection in case of derogation from LCO
Reactor Shutdown Operation
Cold Shutdown Condition
Fuel Shuffling
Preparation of Reactor Startup
Reactor Startup Operation
Full Power Operation
Low Power & Shutdown Operation
Risk Level
In LP & SD condition, risk can vary in accordance with system configuration, water level and so on.
Risk increase due to a trouble
Risk increase due to the switch over of Service Water System
Risk increase due to decrease of reactor water level
Risk increase due to the outage of HPCS / LPCI
① ② ② ② ③ ④
① Before Reactor Shutdown Operation
② Before change of system configurations
③ Before Reactor Startup Operation
④ After Reactor Startup Operation
10


4. Significance Determination Process (SDP)
Image of Comprehensive Plant Evaluation using PI and SDP
JNES
Inspection
Inspection by Regulatory Body
Licensees
Periodic Inspection (NISA / JNES)
Licensee's Periodic inspection
Periodic Safety Management Inspection (JNES)
Maintenance Activities by the Safety Preservation Rule
Quality Management System
Operation management, LCO, surveillance, etc.
Nuclear Safety Inspection (NISA)
Incident Reports required by law (including cause analysis, countermeasure, etc.)
Status of plant performance
Plant performance based on the maintenance activities specified by the technical specification
Inspection Findings
Introduction of Significance Determination Process (SDP)
First stage screening of inspection findings
Evaluation of SDP for plant safety
- safety function
- plant risk etc.
Evaluation of SDP for QMS
Determination of safety significance of selected inspection findings
Introduction of Performance Indicator (PI)
Data acquisition and categorization
Evaluation of each PI
Determination of level of plant performance
PI
SDP
Comprehensive Individual Plant Evaluation and Feedback of the output to the Next Inspection Program
11


JNES
Image of SDP Flow for Plant Safety
Inspection Findings (SIFs)
Minor SIFs ?
Yes
No
Minor SIFs
Categorization of SIFs
Risk-Informed Application
Evaluation of SDP for SIFs
Method - 1
SDP for Safety Function
Method - 2
SDP for Plant Risk
Method - 3
SDP for Public Radiation Effects
Method - 4
SDP for Occupational Radiation Effects
Integration of SDP results for Cross-Cutting SIFs
Safety Significance level
Ⅰ
Ⅱ
Ⅲ
Ⅳ
High
Low
Determination of Safety Significance for SIFs
(level - Ⅰ, Ⅱ, Ⅲ, Ⅳ)
12


JNES
Image of Selection Flow of Minor SIFs for Safety Function and Plant Risk
Selected Inspection Findings (SIFs)
Screen-out of less safety important inspection findings
Is the SIF corresponding to the following items?
Yes
No
(Example)
Inspection related
-Violation of technical specification
-Loss of safety function
Incident report related
-Incidents reported by law
-Violation of limiting condition for operation (LCO)
Is the SIF corresponding to the degradation?
Yes
No
Is the SIF corresponding to the potential loss of function due to poor manual?
Yes
No
Is the SIF corresponding to the other loss of safety function?
Yes
No
Minor SIFs
Evaluation of SDP for SIFs (Safety Significance Level - Ⅰ to Ⅳ)
13


JNES
Image of The SDP Evaluation Flow for Safety Function due to SIFs
Method – 1 SDP for Safety Function
H: High, M: Medium, L: Low, LL: Low-Low
Safety Significance
Safety grade
Consequence grade
Class 1 – SSCs
Class 2 – SSCs
Class 3 – SSCs
A
B
Loss of Safety Function
Influence to Safety Function
Others (Violation of the safety preservation rule)
Plant Risk
H, M
L, LL
H
M, L, LL
Ⅰ
Ⅱ
Ⅲ
Ⅳ
14


JNES
Image of The SDP Evaluation Flow for Plant Risk
Method – 2 SDP for Plant Risk
Selected Inspection Findings (SIFs)
Is it affecting Plant Risk during rated power operation due to SIFs ?
No
Yes
• Risk evaluation during shutdown operation
• Risk evaluation due to external events
Future items
Identification of initiating events to be evaluated due to SIFs.
Identification of the following:
- system lost of function
- fault duration
- influenced systems due to SIFs
Simplified evaluation of the risk impact with ΔCDF, utilizing:
- PSA model of the reference plant,
or
- Check sheet designed for the reference plant.
Evaluation of the impact to the prevention system (i.e., contribution to the initiating event frequency)
Evaluation of the impact to the mitigation system (i.e., contribution to the unavailability of the system)
< Risk Increase >
CDF (/ry)
Estimated
Baseline
ΔCDF
Baseline
observed
Unavailability
Comparison with risk criteria (ΔCDF) and assignment plant risk level for SIFs
High
Medium
Low
Low - Low
The unavailability of the mitigation system is estimated, reflecting the inspection finding.
15

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Determination of Safety Significance for the inspection findings

(1) If the inspection finding affected cross-cutting areas (for example, the safety-function and the radiation effects) , the safety significance should be determined using a higher level of each safety significance.

(2) For the inspection finding affecting both the safety-function and plant risk, the safety significance should be represented by that of safety-function which included the result of risk evaluation.

(3) If the SIFs affected only the plant risk, the plant risk level should be translated into the safety significance level as follows:

Plant Risk Level	Safety Significance Level
High	I
Medium	II
Low	III
Low - Low	IV

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JNES

4. Summary

- The new regulatory inspection system for maintenance program started since January 2009. JNES has been reviewing the maintenance programs for individual NPPs under the request of NISA.
  - The request of NISA includes the review of “Importance of Systems / Functions” & Performance Criteria, those are established by utilities utilizing of “Risk Importance”.
  - In order to review these items in maintenance program, JNES has developed
    - ✓ “Risk Importance” for 15 types / 55 plants
    - ✓ Detail requirements for utilities
    - ✓ Review points & procedures
- The new regulatory inspection system for safety significant activities in safety preservation inspection started since January 2009. In order to support NISA , JNES
  - Identified the candidates of safety significant activities using “Risk Information”
  - Developed the information on the characteristics of system configurations and the timing of safety significant activities for safety preservation inspection manual.
- JNES has developed the methods and “Risk Information” for Comprehensive Plant Evaluation (CPE) including PI & SDP.
  - JNES has prepared the manuals for these evaluations via trial evaluation experiences
  - Brushing up the methods and manuals via trial use of CPE including PI & SDP in FY2009
  - After the trial use, NISA/JNES intend essential use of CPE including PI &SDP

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10th Korea-Japan PSA Workshop, Jeju, Korea

A Study on the Risk-Informed Performance Indicators and Thresholds for Graded Regulation

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FNC Technology Co., Ltd.
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I. Introduction

- The research for the development of risk-informed and performance-based regulatory oversight (Integrated Safety and Performance Assessment (ISPA) Program) is ongoing in KINS.
- The risk-informed PI(Performance Indicator) is the one of the main components for risk-informed and performance-based regulatory oversight.
- In this study, feasibility of some risk-informed PIs to be used in ISPA program has been evaluated and threshold re-evaluation has been performed for currently used KINS PIs.
  - New indicator feasibility study
    - For BRIIE, USwC, MSPI
  - Threshold re-evaluation
    - For currently used KINS PIs in Reactor Safety Area
  - The results in this study requires further refinement, and have not been formally approved by the KINS

I. Introduction

- Overview – PI system currently used in KINS
  - 2 safety areas, 5 categories, and 11 indicators
  - Color coded by 4 levels – Not used in formal regulatory process (e.g. graded regulation)
- PI of "Reactor Safety Area" is evaluated in this study

Area	Category	Specific Performance Indicator	Grade & Color Coding (quarterly)			
			Excellent	Good	Normal	Warning
			Green	Cyan	Yellow	Orange
Reactor Safety	Operational Safety	Unplanned Reactor Scram	<0.75 (3/yr)	≥0.75 (3/yr)	≥1.5 (6/yr)	≥5 (20/yr)
		Unplanned Power Reduction	<1.5 (6/yr)	≥1.5 (6/yr)	≥3 (12/yr)	≥5 (20/yr)
	Multiple Barrier	Fuel Reliability	<50% TS limit	≥50% TS limit	≥70% TS limit	≥100% TS limit
		Reactor Coolant Leakage	<50% TS limit	≥50% TS limit	≥70% TS limit	≥100% TS limit
		Containment Reliability	<90%	≥90%	≥80%	≥60%
		Emergency Preparedness	<90%	≥90%	≥80%	≥60%
	Safety System	SI System Unavailability	<0.015	≥0.015	≥0.05	≥0.1
		EDG System Unavailability	<0.025	≥0.025	≥0.05	≥0.1
		AFW System Unavailability	<0.015	≥0.015	≥0.05	≥0.1
Radiation Safety	On-site Rad. Safety	Radiation Collective Dose	<1manSv	≥1manSv	≥3manSv	≥5manSv
	Off-site Rad. Safety	Public Dose/Environmental Radiation	<0.0625mSv	≥0.0625mSv	≥0.25mSv	≥0.6mSv

TS : Technical Specification

I. Introduction

❑ Comparison of NRC and KINS PI (Reactor Safety)

NRC Performance Indicator			KINS Performance Indicator	Items Performed in this Study
Program	Category	Name of PI		
ITP (Industry Trend Program)	Initiating Event	BRIIE	(Not Included)	New Indicator Feasibility Study
ROP (Reactor Oversight Process)	Initiating Event	Unplanned Scrams	Unplanned Reactor Scram	Threshold Re-evaluation
		Unplanned Power Changes	Unplanned Power Reduction	Threshold Re-evaluation
		Unplanned Scrams with Complications (USwC)	(Not Included)	New Indicator Feasibility Study
	Mitigating System	Safety System Functional Failures	(Not Included)	
		MSPI – Emergency AC Power Systems	EDG System Unavailability	New Indicator Feasibility Study
		MSPI – High Pressure Injection Systems	SI System Unavailability	New Indicator Feasibility Study
		MSPI – Heat Removal Systems	AFW System Unavailability	New Indicator Feasibility Study
		MSPI – Residual Heat Removal Systems	(Not Included)	New Indicator Feasibility Study
		MSPI – Cooling Water Systems	(Not Included)	New Indicator Feasibility Study
	Barrier	RCS Activity	Fuel Reliability	Threshold Re-evaluation
		Reactor Coolant Leakage	Reactor Coolant Leakage	Threshold Re-evaluation
		(Deleted)	Containment Reliability	
		(In a Separate Cornerstone)	Emergency Preparedness	

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II. New Indicator Feasibility Study

❑ Initiating Event – BRIIE(Baseline Index for Initiating Events)

- ❖ "Industry level" initiating event PI in USA
- ❖ Scope (PWR)
  - ➢ TRAN, LOCHS, LOMFW, LOOP, LOAC, LODC, SORV, LOIA, VSLOCA, SGTR
  - ➢ Based on NUREG-5750 initiating event data (1987~1995), and risk-significance (CDF contribution > 1%)
- ❖ BRIIE expanded CDF risk coverage of initiating event in NRC from less than 20% (ROP) to approximately 60% by covering risk-significant events.
- ❖ BRIIE feasibility for Korean NPPs (scope)
  - ➢ BRIIE also covered approximately 60% of CDF for majority of Korean NPPs

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II. New Indicator Feasibility Study

- Initiating Event – BRIIE(Baseline Index for Initiating Events)
  - BRIIE monitors individual initiating events at the industry level against performance-based prediction limit.
  - BRIIE feasibility for Korean NPPs (performance threshold)
    - Almost 1/5~1/2 of the prediction limits compared to US PWRs

Initiating Event	Mean Frequency (/yr)	Baseline Year	Critical Year	Expected Occurrence Rate(/yr)	95% Prediction Limit	95% Prediction Limit (US PWRs)
GTRN	9.10E-01	229.82	195.51	16.38	21	59
LOCV	2.04E-01	229.82	195.51	3.67	6	10
LOFW	5.36E-02	229.82	195.51	0.96	3	15
LOOP	3.93E-02	229.82	195.51	0.71	2	8
LOAC	1.43E-02	229.82	195.51	0.26	1	3
LODC	1.06E-02	229.82	195.51	0.19	1	2
LOIA	1.78E-02	229.82	195.51	0.32	1	3
SLOCA	1.55E-03	459.64	391.02	0.03	0	2
SGTR	1.06E-02	229.82	195.51	0.19	1	2

II. New Indicator Feasibility Study

- Initiating Event – USwC(Unplanned Scrams with Complications)
  - "Plant level" initiating event PI in USA
    - Complements "Unplanned Scrams" PI
    - Monitors potentially risk-significant scrams than normal scram

Category	USwC Criteria
Reactivity Control	Did two or more control rods fail to fully insert?
Turbine Trip	Did the turbine fail to trip?
Power available to Emergency Busses	Was power lost to any ESF bus?
Need to Actuate Injection Sources	Was a Safety Injection signal received?
Availability of Main Feedwater	Was MF unavailable or not recoverable using approved procedures following the scram?
Utilization of Scram Recovery EOPs	Was scram response procedure unable to be completed without re-entering another EOP?

  - USwC feasibility for Korean NPPs (performance threshold)
    - Analyzed the scrams of Korean NPPs during the recent 5 years (2002.10.1 ~ 2007.9.30) to determine USwC threshold.
    - It will be able to use the USNRC threshold (>1 per year) if USwC is selected as initiating PI for Korean NPPs. (5% NPPs exceeded threshold)

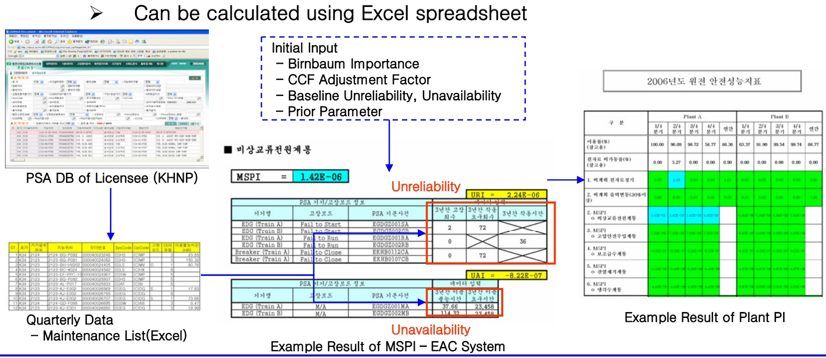
II. New Indicator Feasibility Study

- Mitigating System – MSPI(Mitigating System Performance Index)
  - "Plant level" mitigating system PI in USA
    - Replacement of SSU(Safety System Unavailability) Indicator in NRC
    - SSU is still used in KINS
  - MSPI Definition ⇒ MSPI = URI + UAI (Δ CDF Linear Approximation)

    (URI : Unreliability Index, UAI : Unavailability Index)
    - Reflects plant-specific design and operation
    - Allows trade-offs between Unreliability and Unavailability to optimize system performance
    - No penalty for on-line preventive maintenance hours up to pre-planned baseline
    - Performance Threshold : Δ CDF (RG 1.174)
      - Green/White 1.0E-6, White/Yellow 1.0E-5, Yellow/Red 1.0E-4

II. New Indicator Feasibility Study

- Mitigating System – MSPI(Mitigating System Performance Index)
  - MSPI feasibility for Korean NPPs
    - A main precondition for implementation of MSPI
      - Level 1 PSA for internal events for all domestic NPPs – Available
      - PSA DB system for all domestic NPPs – Will be Available (2010)
    - Can be calculated using Excel spreadsheet



III. Threshold Re-Evaluation

- Basic Philosophy for Setting Performance Thresholds (SECY 99-007)
  - Must be clearly defined
  - Risk-informed to the extent practical
  - Accommodate defense in depth and indications based on existing regulatory requirements and safety analyses
  - Consistent with other NRC risk applications (e.g. Regulatory Guide 1.174) and existing regulatory requirements and safety analyses
  - Thresholds should provide sufficient differential to allow meaningful differentiation in performance and limit false positives
  - Sufficient margin should exist between nominal performance bands to allow for licensee initiatives to correct performance problems before reaching escalated regulatory involvement thresholds
  - Where appropriate plant-specific design differences should be accommodated
  - ....

III. Threshold Re-Evaluation

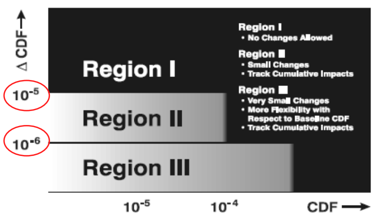
- NRC Conceptual Framework for Setting Performance Threshold

Probabilistic Basis

Historical Data

Safety Subsidiary Goal : CDF < 1E-4/yr

Acceptance Guideline for CDF (RG 1.174)



Deterministic Basis

Technical Specification

Safety Margin

Expert Judgement, etc

- Performance Threshold (SECY 99-007) -

- GREEN (Acceptable) -
• Cornerstone objectives fully met
• Nominal Risk/Nominal Deviation From Expected Performance
• Δ CDF < 1E-6

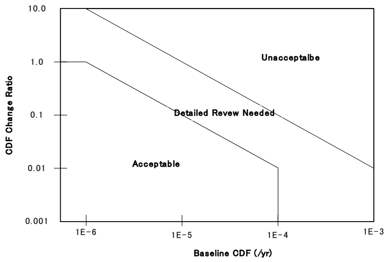
- WHITE (Acceptable) -
• Cornerstone objectives fully met with minimal reduction in safety margin
• Outside bounds of nominal performance
• Within Technical Specification limits
• Δ CDF < 1E-5

- YELLOW (Acceptable) -
• Cornerstone objectives met with significant reduction in safety margin
• Technical Specification limits reached or exceeded
• Δ CDF < 1E-4

- RED (Unacceptable) -
• Plant performance significantly outside design basis
• Loss of confidence in ability of plant to provide assurance of public health and safety with continued operation
• Unacceptable margin to safety

III. Threshold Re-Evaluation

- Korean Safety Subsidiary Goal and Acceptance Guideline for CDF
  - Safety Subsidiary Goal : CDF < 1E-4/yr
    - Same as NRC
  - Acceptance Guideline for CDF (KINS/GT-N24)
    - Same as NRC, except for very small baseline CDF (below 1.0E-7/yr) NPP
    - All operating reactors in Korea have CDF higher than 1.0E-7/yr



  - NRC performance threshold values of can be applied in Korea. <u>However, color coding scheme need to be modified.</u>

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III. Threshold Re-Evaluation

- Color coding scheme of NRC and KINS PI
  - KINS used similar PI threshold values with NRC
  - However, KINS color coding scheme which represents degree of safety level was different.

(NRC Performance Color Coding Scheme)

Green	Acceptable (Objectives full met)
White	Acceptable (Minimal reduction in safety margin)
Yellow	Acceptable (Significant reduction in safety margin)
Red	Unacceptable

(KINS Performance Color Coding Scheme)

Green	Excellent
Cyan	Good
Yellow	Normal
Orange	Warning

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III. Threshold Re-Evaluation

❑ Initiating Event – Threshold of Unplanned Reactor Scram

(Current PI)

Performance Indicator	Green	Cyan	Yellow	Orange
Unplanned Reactor Scram	<0.75 (3/yr)	≥0.75 (3/yr)	≥1.5 (6/yr)	≥5 (20/yr)

(Proposed PI)

Performance Indicator	Green	White	Yellow	Red
Unplanned Reactor Scram	<0.75 (3/yr)	≥0.75 (3/yr)	≥1.5 (6/yr)	≥5 (20/yr)

❖ Green/Cyan threshold (3/yr)

➢ Unplanned Scrams (1996~2006) ranged from 0.56/yr~2.25/yr

❖ Cyan/Yellow threshold (6/yr), Yellow/Orange threshold (20/yr)

PSA Model	Cyan/Yellow (ΔCDF 1E–5/yr)	Yellow/Orange (ΔCDF 1E–4/yr)	Initiating Event Frequency in PSA
CE	12.18	107.34	1.74
Westinghouse	6.15	52.55	1.21
Framatome	7.30	61.43	0.97
AECL	25.23	237.16	1.83

❖ Threshold values can remain unchanged. However, color coding need to be modified.

➢ ΔCDF 1.0E–4 must be interpreted as RED(unacceptable), not ORANGE(warning)

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III. Threshold Re-Evaluation

❑ Initiating Event – Threshold of Unplanned Power Reduction

(Current PI)

Performance Indicator	Green	Cyan	Yellow	Orange
Unplanned Reactor Reduction	<1.5 (6/yr)	≥1.5 (6/yr)	≥3 (12/yr)	≥5 (20/yr)

(Proposed PI)

Performance Indicator	Green	White	Yellow	Red
Unplanned Reactor Reduction	<1.5 (6/yr)	≥1.5 (6/yr)	N/A	N/A

❖ No threshold for White/Yellow, Yellow/Red because the indicators could not be directly tied to risk data

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III. Threshold Re-Evaluation

- Barrier Integrity – Threshold of Fuel Reliability, Reactor Coolant Leakage, Containment Leakage, Emergency Preparedness

(Current PI)

Performance Indicator	Green	Cyan	Yellow	Orange
Fuel Reliability	<50% TS limit	≥50% TS limit	≥70% TS limit	≥100% TS limit
Reactor Coolant Leakage	<50% TS limit	≥50% TS limit	≥70% TS limit	≥100% TS limit
Containment Reliability	<90%	≥90%	≥80%	≥60%
Emergency Preparedness	<90%	≥90%	≥80%	≥60%

(Proposed PI)

Performance Indicator	Green	White	Yellow	Red
Fuel Reliability	<50% TS limit	≥50% TS limit	≥100% TS limit	N/A
Reactor Coolant Leakage	<50% TS limit	≥50% TS limit	≥100% TS limit	N/A
Containment Reliability	<90%	≥90%	≥60%	N/A
Emergency Preparedness	<90%	≥90%	≥60%	N/A

- Green – Expected performance
- White – Within Technical Specification limits
- Yellow – Technical Specification limits exceeded

IV. Summary and Future Work

- "Reactor Safety Area" PIs (Performance Indicators) to be used in Korean ISPA Program have been proposed in this study (draft version)

Area	Category	Specific Performance Indicator	Grade & Color Coding (yearly)			
			Green	White	Yellow	Red
React or Safety	Operational Safety	Unplanned Reactor Scram	<3	≥3	≥6	≥20
		Unplanned Power Reduction	<6	≥6	N/A	N/A
		USwC	<2	≥2	N/A	N/A
	Multiple Barrier	Fuel Reliability	<50% TS limit	≥50% TS limit	≥100% TS limit	N/A
		Reactor Coolant Leakage	<50% TS limit	≥50% TS limit	≥100% TS limit	N/A
		Containment Reliability	<90%	≥90%	≥60%	N/A
		Emergency Preparedness	<90%	≥90%	≥60%	N/A
	Safety System	MSPI – Emergency AC Power Systems	<1E-6 and <PL	≥1E-6 or ≥ PL	≥1E-5	≥1E-4
		MSPI – High Pressure Injection Systems	<1E-6 and <PL	≥1E-6 or ≥ PL	≥1E-5	≥1E-4
		MSPI – Heat Removal Systems	<1E-6 and <PL	≥1E-6 or ≥ PL	≥1E-5	≥1E-4
		MSPI – Residual Heat Removal Systems	<1E-6 and <PL	≥1E-6 or ≥ PL	≥1E-5	≥1E-4
		MSPI – Cooling Water Systems	<1E-6 and <PL	≥1E-6 or ≥ PL	≥1E-5	≥1E-4

⋆ PL = Performance Limit ⋆ MSPI requires 3 years data

- Review of "Radiation Safety Area" PI is needed in the future
- Consultation with KINS Operational Safety Analysis Department (which is practical PI management team) is needed to finalize proposed Pis.

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