Improvements of Safety and Reliability by PRA

Yukihiro Kirimoto, Senior Research Scientist

Nuclear Technology Research Laboratory
Central Research Institute of Electric Power Industry

The 3rd Meeting of the Working Group on Voluntary Efforts and Continuous Improvement of Nuclear Safety
September 11, 2013
Risk in technical systems in nuclear safety

◆ Risk triplets (Kaplan, etc., 1981)
(1) What can go wrong (accident sequence)
(2) How likely it is (probability)
(3) What its consequences might be (consequence)

The combined answer to these three questions that consider (1) what can go wrong, (2) how likely it is, and (3) what its consequences might be. (Source: NRC Glossary)

◆ Risk information in PRA
For all the accident sequences that can be theoretically assumed, we investigate the occurrence frequency and the extent of impact, or the product of those, of undesirable scenarios, quantitatively (=investigate risk).
U.S. situation after the TMI accident

Occurrence of the TMI accident

- Prolonged licensing work, enhanced regulation, backfitting of the new rules
- Elevated construction costs, cancellation of construction projects, criticism against nuclear energy

1980s

- Availability factors in the 60% range, frequent occurrence of trouble (6 to 7 unscheduled outages per reactor-year)
- 100 days of refueling outage, almost no on-line maintenance performed
- NRC had not established quantitative assessment techniques.
- Perception that plants are safer during outages than during operation (no risk assessment made on the loss of external power supply)
- Release of “Troubled Plants Watch List” → Degradation of public trust, of financial assessment and of the morale of electric utilities
Efforts to regenerate U.S. reactors and the relevant background

◆ Implementation of "kaizen“ (Japanese for “improvement”) activities by industry (learning from Japan)

  ➢ Controlling not only the results but also the processes ("kanban" system, etc.)
  ➢ Quality assurance based on reality rather than on the volume of documents
  ➢ Collaboration between industry, academia and the government (trust first, penalties for offenders)
  ➢ Multiple skills (also care about relevance instead of a clear delineation between tasks)
  ➢ Long-term perspective on management with sustainability and patience in mind

◆ Reaganomics and focusing on an efficient and effective regulation

  ➢ Focusing on efficiency and effectiveness in all governmental organizations while pursuing deregulation
  ➢ Annual reporting of NRC activities and its influence on the industry; covering 90% of the budget through regulatory fees
Development of a scheme for safety improvement after the TMI accident

◆ INPO established (1979)
  ➢ Mainly comprised of members from the Navy (naval reactor engineers) at the beginning
    ■ It was organized in a military style. Because of the high social status of the staff and the small scale of electric utilities, INPO's recommendation and guidance was effective. (Members were sequentially replaced with those from utilities.)
    ■ As a principle, information is not disclosed (to facilitate the sharing of high-quality technical information).

◆ NUMARC established (1986)
  ➢ Established as an organization to contact the NRC

◆ NEI established (1994)
  ➢ Established by merging the NUMARC and other organizations to communicate with the NRC, the Congress and the media

◆ EPRI's technical supports and collaborations (INPO, NEI and EPRI shared objectives)
Factors that triggered the use of PRA in safety regulation

Reactor Safety Study (NRC, WASH-1400, RSS) of 1975

- "The frequency of severe accident occurrence is higher than the frequency of the design basis accident set by the deterministic approach."

Lessons learned from the TMI-2 accident in 1979

- A similar accident sequence had been analyzed in WASH-1400 (small-break LOCA).
- "PRA should be sophisticated for regulatory use." (Kemeny report to President Carter)

More detailed safety information can be obtained by using PRA than by using the deterministic approach.

- Safety issues unnoticed by the deterministic approach would be predicted.
- It is possible to perceive the priority of safety issues systematically.
U.S. NRC started early to develop PRA

Use of PRA in different reactor types and standardization of procedures (1979–1982)

- Reactor Safety Study Methodology Application Program
- Interim Reliability Evaluation Program


Used for Unresolved Safety Issues (early 1980s)

- ATWS rules, SBO rules
- Backfitting rules from the TMI-2 action plan
- Risk significance judgment of LERs (conditional core damage frequency, severe accident precursors)

Development of safety goals (1986)

- Personal risk (acute) + Societal risk (latent)
More emphasis on PRA utilization and regulatory change

**NUREG-1150 "Severe Accident Risks" (1986–1990)**
- Full-scope PRA of five U.S. plants featuring typical reactor types
- Internal events, **external events**, cognitive uncertainties, random uncertainties

**GL 88-20 "IPE for Severe Accident Vulnerabilities" (until 1992)**
- Identification of and measures against plant-specific vulnerabilities
- Development of plant-specific PRA (special division within the plant)

**Development of quantitative supplemental safety goals (1990)**
- $CDF = 10^{-4}/ry$, $LERF = 10^{-5}/ry$ **[including external events]**
- Benchmark values rather than regulatory requirements

**Review of the regulatory work (focusing on efficiency and effectiveness)**
- Sillin Report (1986): Emphasis on collaborations between the NRC and industry (NEI)
  -> Regulation based on actual operation and risk information (ROP) [First implementation of a civilian (NEI's) regulation plan]
NRC policy statement in 1995
— Use of PRA for nuclear safety regulation (reformed awareness of regulation) —

NRC Chairman announced the extended use of PRA (top-down style).

Extension and supplement of the deterministic approach (defense-in-depth)

- Advantages of the probabilistic approach:
  Potential safety issues and measures can be studied systematically and widely.
  Logical prioritization based on risk significance is possible.

Eliminate unnecessary regulatory burden and concentrate resources on key safety issues.

- What is the real situation, and what should be done to improve safety and efficiency?

Publication of a guidance on acceptable risk criteria and on PRA utilization

Regulatory guide on changing regulatory requirements
RG 1.174
1. Enhance the grounds for safety decision-making
2. Effectively use NRC and industry resources
3. Comply with the philosophy of defense in depth
4. Monitor the impacts using performance measurement strategies
Decision-making related to risk-informed regulatory changes in the U.S.

The probabilistic approach is an extension of the deterministic approach

1. Meet the current regulatory requirements*
2. Comply with the philosophy of defense in depth
3. Maintain sufficient safety margins
4. An increase in CDF or risk is small and is consistent with the safety goal policy statement.
5. Monitor the impacts using performance measurement strategies

Traditional analysis

PRA

[Element 1] Define change
[Element 2] Perform engineering analysis
[Element 3] Define implementation/monitoring program
[Element 4] Submit proposed changes

※ Appended reference 1: An application for approval of the extension of allowable outage time (AOT) for the emergency diesel generators of Browns Ferry units 2 and 3, as well as the NRC review results, are described.

* Excluding requirements related to 10 CFR 50.12 "Specific exemptions" or 10 CFR 2.802 "Petition for rulemaking"
<table>
<thead>
<tr>
<th></th>
<th>Deterministic safety assessment</th>
<th>PRA (probabilistic risk assessment)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Events to be studied</strong></td>
<td>A few representative events that are considered most severe among all the postulated events</td>
<td>All the events are considered to be significant</td>
</tr>
<tr>
<td><strong>Frequency of accident occurrence</strong></td>
<td>Assumed to occur unambiguously (without discussion of occurrence frequency)</td>
<td>Because occurrence frequency has a probability distribution, the evaluation is made in terms of median or mean and range of uncertainty.</td>
</tr>
<tr>
<td><strong>Accident analysis method</strong></td>
<td>Analyzed under conservative assumptions using scenarios specified in safety assessment guidelines and the like (e.g., assuming a single failure in the most effective accident mitigating system).</td>
<td>In considering various conceivable accident sequences, all the significant accident sequences are analyzed based on realistic assumptions (assuming multiple failures in mitigating systems).</td>
</tr>
<tr>
<td><strong>Risk assessment</strong></td>
<td>No assessment or qualitative assessment</td>
<td>Quantitative assessment</td>
</tr>
<tr>
<td><strong>Uncertainty handling</strong></td>
<td>Uncertainties are avoided by following the &quot;procedure of conservatively arranged accident analysis.&quot;</td>
<td>Quantitative assessment including propagation of uncertainties (Because realistic assessment is attempted, uncertainties are significant when dealing with an area where only limited knowledge is available.)</td>
</tr>
<tr>
<td><strong>Interpretation of assessment results</strong></td>
<td>An interpretation is made for each accident.</td>
<td>A comprehensive interpretation is made based on all the accident sequences.</td>
</tr>
<tr>
<td><strong>Application examples</strong></td>
<td>Application for reactor establishment (Annex 10)</td>
<td>Reactor Safety Study (WASH-1400) Severe Accident Risks (NUREG-1150) Identification of severe accident management (AM) measures and effectiveness assessment PSA in periodic safety reviews (PSR)</td>
</tr>
</tbody>
</table>
Examples of use of PRA in the U.S.

### NRC

**Identification of safety issues and prioritization**

**NRC's internal processes**
- Generic safety issues
- Backfitting rules
- Reactor oversight process

**Regulatory requirements**
- Maintenance rule (10 CFR 50.65)
  - Monitoring of maintenance effectiveness / Risk management (on-line maintenance)

### Electric utilities

**Simultaneous pursuit of safety and cost effectiveness by eliminating unnecessary regulatory burdens and by proposing alternatives**

**Operators' voluntary efforts**
- RI-ISI (10 CFR 50.55a)
- Changes to technical specifications (10 CFR 50.90)
  - AOT extension / STI extension
- Changes to special treatment requirements (10 CFR 50.69)
- Performance-based fire protection program (10 CFR 50 Appendix R)
- Mitigation of ECCS design requirements

* Appended reference 2: Examples of RI-ISI and maintenance rules (monitoring of safety-significant equipment and on-line maintenance)
* Appended reference 3: Trends in the use of risk information in U.S. safety regulations
Configuration of NRC regulatory guidelines

Application by operators

- **Licensing**
  - Risk-informed licensing changes

- **In-service testing**
  - Risk-informed IST changes

- **Quality assurance**
  - Risk-informed quality assurance changes

- **Technical specifications**
  - Risk-informed technical specifications changes

- **In-service inspection**
  - Risk-informed ISI changes

NRC review

Regulatory guides / Standard Review Plan (SRP) specific to each application type

- Regulatory Guide 1.174
  - Overview
  - SRP Chap. 19
    - General guidance

- Regulatory Guide 1.175
  - In-service testing
  - SRP Sec. 3.9.7
    - In-service testing

- Regulatory Guide 1.176
  - Graded quality assurance
  - SRP Sec. 16.1
    - Technical specifications

- Regulatory Guide 1.177
  - Technical specifications
  - SRP Sec. 3.9.8
    - In-service inspection

- Regulatory Guide 1.200
  - Quality of PRA
  - SRP Sec. 19.1

ASME PRA standards and industry's peer review programs

ANS PRA standards

© CRIEPI
Trends in the use of PRA/PSA by international organizations

**OECD**
- Mainly in the field of regulatory application of PSA, WGRisk prepares reports on quantitative safety goals, the status of PSA application and the development of procedures; holds workshops; and produces State-of-the Art Reports (SOAR) and Technical Opinion Papers (TOP).

**IAEA**
- IAEA develops international standards on risk-informed regulations and publishes technical documents (TECDOC) on decision-making processes and other issues. It also participates in WGRisk to study international trends and prepares technical reports and guidelines. Furthermore, it is steadily studying the use of PSA in nuclear fuel cycle facilities.
Trends in the use of PRA/PSA in Europe

- France: The current safety regulation is mainly based on a deterministic approach. While PSA is being used in an auxiliary manner, there is a shift toward increased use of PSA.
- Sweden: Safety regulation based on PSA has been pursued. RI-ISI was accepted in 1994 when enhancing regulations to reduce the risk of pipe failure caused by stress corrosion cracking. Currently PSA is being used together with the U.S. approach. There are no quantitative safety goals, and operators are setting up their own targets.
- UK: The nuclear safety assessment principles (SAPs) set forth quantitative safety goals and require PSA. There are industry activities such as the use of a “living PSA” at Sizewell B and participation in ASME RI-ISI procedures.
- Spain and the Republic of Korea: Because these countries adopt the U.S. regulatory system as a whole, they use the risk-informed approach in line with the U.S. Development of databases based on domestic performance data would be one of the future challenges.
## Development of domestic parameters for PRA

It is also important to collect data on intact components for parameter calculation (data on operations and numbers)

<table>
<thead>
<tr>
<th>Parameter type</th>
<th>Development of domestic estimation procedures (Development based on the U.S. approach)</th>
<th>Collection and organization of domestic data</th>
</tr>
</thead>
</table>
| Equipment failure rate / Failure probability* | **The Bayesian approach**※ has mostly been established. From general equipment failure rate to individual evaluation                                                                                             | **Current estimations are made on the basis of the NUCIA PRA database.**  
• Japan Nuclear Safety Institute is shifting to a failure database using a **maintenance activity management index (function failure events)**.                                                                                                         |
| Unavailable time / Unavailability*      | **The Bayesian approach** has mostly been established.                                                                                                                                       | **Conventionally, values obtained from overseas literature have been used.**  
• Japan Nuclear Safety Institute is shifting to a failure database system using a **maintenance activity management index**.                                                                                                                     |
| Frequency of initiating event occurrence* | **The Bayesian approach** has mostly been established.                                                                                                                                       | **Japan Nuclear Safety Institute is developing a database system.**  
• Extending to fire and flooding                                                                                                                                                                                                                                                               |
| Common cause failure parameters*        | The procedure has mostly been established.  
Preparation of a guide for making common-cause **engineering decisions**                                                                                                                       | **CRIEPI and Japan Nuclear Technology Institute have analyzed BWR and PWR components, respectively.**  
• Japan Nuclear Safety Institute is developing a database system.                                                                                                                   |
| Human error rate                        | Application of the THERP approach  
Studying second generation techniques                                                                                                                                                    | **Studying data to be collected** by current-generation and next-generation techniques (not easy to derive from performance data alone).  
**International joint researches and collaborations involving the Self-Defense Forces and other industries such as the airline industry are necessary.**                                                                                      |

* Procedures are described in the Atomic Energy Society of Japan’s Standards for PSA Parameters. Application examples are to be supplied in the future.
* Appended reference 4: Example of application to the assessment of failure rate parameters using domestic data and uncertainties based on Bayesian statistics.
## Reliability improvements in assessment procedures and data of PRA

<table>
<thead>
<tr>
<th>Type of assessment procedure</th>
<th>Development of domestic assessment procedures</th>
<th>Reliability improvements in assessment procedures</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hazard assessment (earthquake, tsunami, etc.)*</td>
<td>Assessment procedures for earthquakes and tsunamis have mostly been established. Those for fires and other hazards are being developed.</td>
<td>・ Standardization through collaborations between multiple academic societies, and assessments at each plant are being conducted. ・ Demonstration testing for improving data reliability through international joint research and other collaborations are required.</td>
</tr>
<tr>
<td>Fragility assessment (earthquake, tsunami, etc.)*</td>
<td>Same as above. As for tsunamis, standardization of procedures that collaborate with hazard assessment is being studied.</td>
<td>・ Standardization through collaborations between multiple academic societies, and studies on implementation cases are being conducted. ・ Demonstration testing for improving data reliability through international joint research and other collaborations, is required.</td>
</tr>
<tr>
<td>Combined hazard assessment</td>
<td>Procedures to assess earthquake-associated external events are being developed.</td>
<td>・ Standardization and studies on implementation cases are being conducted through collaborations between multiple academic societies; international joint research and other collaborations are also necessary.</td>
</tr>
<tr>
<td>Impact assessment of multiple-unit sites</td>
<td>Shared facilities, interdependence, etc. are incorporated in single-plant PRA models.</td>
<td>・ A large-scope assessment of the risk impact needs to be studied together with the development of PRA for external events. ・ International joint research and other collaborations are required.</td>
</tr>
<tr>
<td>Digital equipment reliability</td>
<td>Procedures to assess software reliability are studied.</td>
<td>・ In the stage of failure assessment similar to that of conventional equipment. ・ Procedures to assess software reliability are studied.</td>
</tr>
</tbody>
</table>

* Standard procedures have been developed by the Atomic Energy Society of Japan regarding PRA for earthquakes and tsunamis. Accumulation of implementation cases is necessary.
Use of risk information — For extended domestic application

- Achieving a scheme where plants themselves maintain and improve assessment and data collection practices
  - Plants themselves construct and maintain the mechanism of risk management processes, conduct plant-specific assessments reflecting plant characteristics and on-site perceptions, grasp the specific sources of uncertainties and other information, and use such information for comprehensive judgment as an extension of the deterministic approach.
  - Put emphasis on grasping the real situation; collected data may not necessarily be disclosed in this regard. A well-established risk management system at each plant site is essential for plant-specific risk assessments. This process is monitored under the regulation, and information can be disclosed as necessary.

- Use of mutual assessments between operators, and development of a voluntary regulation system by a private organization
  - Conduct performance assessments based on clear assessment procedures, and develop a stringent voluntary regulation system. The private organization for this purpose must have enhanced authority over operators, expertise and technical capabilities, provide technical support, and lead the industry.
  - Share expertise and technical information with regulators, improve safety by the effective use of resources and pursue high explanatory capabilities by using risk information.

- Development of PRA-related assessment techniques
  - A national-scale effort is required to conduct a demonstration testing of new assessment techniques, such as those for external events, to improve data reliability.
Reference 1:

An example of changes to technical specifications by applying Regulatory Guide (RG) 1.177

— Extension of allowable outage time (AOT) for the emergency diesel generators of Browns Ferry units 2 and 3 —
## An example of changes to technical specifications by applying RG 1.177

---

**Extension of allowable outage time (AOT) for the emergency diesel generators of Browns Ferry units 2 and 3**

<table>
<thead>
<tr>
<th>Item</th>
<th>Applicant’s assessment (submitted on March 12, 1997)</th>
<th>NRC’s review (safety assessment on August 2, 1999)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. System proposed for AOT change</td>
<td>Emergency diesel generators (EDGs)</td>
<td>Adequacy of the proposed change described on the left is reviewed.</td>
</tr>
<tr>
<td>2. AOT before and after the change</td>
<td>From 7 days to 14 days (temporal change to facilitate preventive maintenance conducted every 12 years)</td>
<td></td>
</tr>
<tr>
<td>3. Grounds for the proposed AOT change [Element 1]</td>
<td>To perform, as recommended by the EDG manufacturer, the 12-year periodic preventive maintenance (including an overhaul) of the EDGs during power operation, at least 13 days of AOT is required.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Substantial flexibility is given to the preventive maintenance of the EDGs by extending the AOT.</td>
<td></td>
</tr>
<tr>
<td>4. Adequacy from engineering assessment [Element 2]</td>
<td>Off-site power systems are redundant, diversified and reliable.</td>
<td>The change of AOT is acceptable for the following reasons.</td>
</tr>
<tr>
<td></td>
<td>Browns Ferry units 1, 2 and 3 have a total of eight EDGs: four for units 1 and 2, and four for unit 3. Because unit 1 is currently shut down, four EDGs are available to each of units 2 and 3.</td>
<td>• The AOT of the EDGs remove the need for limiting conditions for operation, and reduces the number of times the EDGs start up.</td>
</tr>
<tr>
<td></td>
<td>The distribution systems of on-site and off-site power have been improved to enhance reliability.</td>
<td>• Risk-based maintenance scheduling is being implemented.</td>
</tr>
<tr>
<td></td>
<td>Procedures have been established for minimizing the number of times equipment enters AOT and for properly managing other maintenance and testing operations when the EDGs are not in service.</td>
<td>• The on-site distribution system has been improved.</td>
</tr>
<tr>
<td></td>
<td>In conducting the 12-year periodic preventive maintenance of the EDGs during power operation, the unavailability of the EDGs can be suppressed by entering 14-day AOT just once compared to entering 7-day AOT multiple times.</td>
<td>• There have been no occurrence of complete loss of off-site power in the past.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• The off-site power system has been improved to reduce the probability of station blackout.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• The other power systems are not tested or maintained during the extended AOT.</td>
</tr>
<tr>
<td>Item</td>
<td>Applicant's assessment (submitted on March 12, 1997)</td>
<td>NRC's review (safety assessment on August 2, 1999)</td>
</tr>
<tr>
<td>------</td>
<td>---------------------------------------------------</td>
<td>-----------------------------------</td>
</tr>
</tbody>
</table>
| 5. Compensatory measures for the change of AOT [Element 2] | The following compensatory measures will be taken during the EDG AOT.  
- Two or more off-site power supplies will be kept available.  
- Switchyard operations will be restricted, and maintenance work at switchyards at high risk will be suspended. |  |
- PRA used | PSA for multiple units based on the revised version of PSA for the IPE of unit 2 (units 2 and 3 are in operation while unit 1 is shut down)  
- PSA of levels 1 and 2 for internal events. Plant-specific data are used for equipment failure rates and unavailability due to testing and maintenance.  
- This PSA model was reviewed by the PSA quality review team of BWROG and was found to be of sufficient quality. | No significant defects have been found in the risk analysis meant to justify the AOT change.  
- The business operator's risk analysis procedure is of sufficient quality to be used for the AOT change. |
|  | Assessment method |  |
|  | The increase in the unavailability of the EDGs by the AOT extension is calculated according to RG 1.177:  
(Normal unavailability) + (Unscheduled increase in unavailability) + (Increase in unavailability by maintenance).  
- Conditional CDF is calculated by multiplying baseline CDF by risk achievement worth (RAW).  
- ICCDP = [RAW × (Baseline CDF) - (Baseline CDF)]  
× (Proposed AOT period)  
- ICLERP = (LERF/CDF) × ICCDP |  |
<p>|  | Consideration of uncertainties | Taken into account in peer reviews. |</p>
<table>
<thead>
<tr>
<th>Item</th>
<th>Applicant’s assessment (submitted on March 12, 1997)</th>
<th>NRC’s review (safety assessment on August 2, 1999)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>7. Results of incremental risk assessment</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>[Element 2] [First stage]</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Unit 2</td>
<td>Unit 3</td>
<td></td>
</tr>
<tr>
<td>$\Delta$CDF</td>
<td>$\Delta$LERF</td>
<td></td>
</tr>
<tr>
<td>$4.5 \times 10^{-8}$ per reactor-year</td>
<td>$2.0 \times 10^{-8}$ per reactor-year</td>
<td></td>
</tr>
<tr>
<td>$9.1 \times 10^{-8}$ per reactor-year</td>
<td>$2.0 \times 10^{-8}$ per reactor-year</td>
<td></td>
</tr>
<tr>
<td>ICCDP</td>
<td>ICLERP</td>
<td></td>
</tr>
<tr>
<td>$3.7 \times 10^{-8}$</td>
<td>$1.2 \times 10^{-8}$</td>
<td></td>
</tr>
<tr>
<td>$6.2 \times 10^{-8}$</td>
<td>$1.9 \times 10^{-8}$</td>
<td></td>
</tr>
<tr>
<td>The values ($\Delta$CDF, $\Delta$LERF, ICCDP and ICLERP) assessed by the business operator are small enough to meet the guidelines of RG 1.174 and RG 1.177.</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>8. Risk judgment criteria and assessment results</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>[Element 2] [First stage]</td>
<td></td>
<td></td>
</tr>
<tr>
<td>The incremental risk caused by the AOT extension is negligible. The baseline CDF and LERF meet the RG 1.174 criteria, and ICCDP and ICLERP sufficiently meet the RG1.177 allowable limits ($&lt;5 \times 10^{-7}$ and $&lt;5 \times 10^{-8}$, respectively).</td>
<td>RG 1.174 and RG 1.177 are used as risk judgment criteria, and it is concluded that the effect of the change on the risk is minimal, and that the AOT extension can be supported.</td>
<td></td>
</tr>
<tr>
<td><strong>9. Avoiding plant system configurations that are risk-significant</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>[Element 2] [Element 3] [Second stage]</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Whether system configurations are risk-significant is judged in reference to a maintenance risk matrix, and the staff in charge of scheduling will be warned of risk-significant system configurations.</td>
<td>The proposed revision in the procedures is adequate for avoiding risk-significant system configurations</td>
<td></td>
</tr>
<tr>
<td>When a severe weather is expected, risk-significant maintenance operations will not be planned or conducted during the expected period. This will be officially reflected in the plant procedures.</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>10. Configuration risk management program (CRMP)</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>[Element 2] [Element 3] [Third stage]</td>
<td></td>
<td></td>
</tr>
<tr>
<td>The development of CRMP described in RG 1.177 is not necessary at present. It is more appropriate to postpone the development until the maintenance rules are revised.</td>
<td>Because the business operator later proposed a CRMP concordant with RG 1.177, the business operator has satisfied the intentions of the guide.</td>
<td></td>
</tr>
<tr>
<td>(A model program described in RG 1.177 was presented in a letter dated June 18, 1999, and a commitment was made as to incorporating it into the technical specifications requirement manual of units 2 and 3.)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>The program is the same as the business operator’s other approved CRMP for the Sequoyah plant.</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>11. Comprehensive assessment</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>The incremental risk caused by the AOT extension is negligible and does not have a significant impact.</td>
<td>The results and findings of PRA as well as the engineering assessment support the proposed AOT change.</td>
<td></td>
</tr>
</tbody>
</table>
Reference 2:
Example of PRA use in the U.S.

- RI-ISI

- Maintenance Rule: The monitoring of critical components

- Maintenance Rule: On-line maintenance
Risk-informed in-service inspection (ISI) of piping

ASME Sec. XI ISI

- Scope of application
  Determined by design stress (Classes 1 to 3, non-code)

- Ineffective for environmentally induced failures (e.g., FAC and IGSCC)

Optimization based on operational experience is necessary.

Risk-Informed ISI

<table>
<thead>
<tr>
<th>POTENTIAL FOR PIPE RUPTURE</th>
<th>CONSEQUENCES OF PIPE RUPTURE</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY</td>
</tr>
<tr>
<td></td>
<td>NONE</td>
</tr>
<tr>
<td>HIGH FLOW ACCELERATED CORROSION</td>
<td>LOW Category 7</td>
</tr>
<tr>
<td>MEDIUM OTHER DEGRADATION MECHANISMS</td>
<td>LOW Category 7</td>
</tr>
<tr>
<td>LOW NO DEGRADATION MECHANISMS</td>
<td>LOW Category 7</td>
</tr>
</tbody>
</table>

- Prioritization of inspection areas according to the risk significance
  The number of inspection areas can be reduced without increasing the risk.

(Reference: EPRI TR-112657-REVB-A)
Maintenance Rule: The monitoring of critical components

- Monitor the effectiveness of the maintenance
- Monitor the performance of risk-significant SSCs.
- Determine risk-significant SSCs by risk assessment using PRA.
- SSCs that do not satisfy performance criteria shall be strictly controlled until they satisfy the criteria.

All SSCs in the plant

<table>
<thead>
<tr>
<th>RISC-1</th>
<th>RISC-2</th>
</tr>
</thead>
<tbody>
<tr>
<td>High risk significance</td>
<td>STR intensive area</td>
</tr>
<tr>
<td>RISC-3</td>
<td>RISC-4</td>
</tr>
<tr>
<td>Low risk significance</td>
<td>STR exempted</td>
</tr>
</tbody>
</table>

- Safety related
- Non-safety related

Risk-based classification

Design-based classification

STR: special treatment requirement
RISC: risk-informed safety class

SSC: structures, systems and components
Maintenance Rule: On-line maintenance

- For high risk-significant SSCs, risk assessment is required in that regards before preventive maintenance and during maintenance.

- Conduct a configuration risk management program (CRMP) in the plant.
Reference 3:

Trends in the use of risk information in U.S. safety regulation

Risk-informed and performance-based plan (RPP)
Trends in the use of risk information
Risk-informed and performance-based plan (RPP) (2007 and onward)

- Further development of past plans

- Rebuilding plans in three fields: reactor safety, materials safety and waste management.

- Among various initiatives, those to be continued, terminated or newly introduced are discussed.

- Evaluate the effectiveness of implemented initiatives.

- Identify the objectives, bases and goals of the PRA use, and communicate with the public and stakeholders.
## Field of operating reactors: Rulemaking

<table>
<thead>
<tr>
<th>Initiative</th>
<th>Program</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>ECCS requirements</td>
<td>Redefinition of LOCAs</td>
<td>Make risk-informed changes to LOCA-related technical requirements (10 CFR 50.46a)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>LOCA-LOOP</td>
<td>Remove the requirement to consider a LOOP in conjunction with a large LOCA.</td>
<td></td>
</tr>
<tr>
<td>PTS requirements</td>
<td>10 CFR 50.61a</td>
<td>Voluntary risk-informed alternative PTS limits are set.</td>
<td>Completed</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Unnecessary conservatism in the PTS rules is reduced.</td>
<td>Completed</td>
</tr>
</tbody>
</table>
## Field of operating reactors: Licensing 1/3

<table>
<thead>
<tr>
<th>Initiative</th>
<th>Program</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>Risk-related Regulatory Guides</td>
<td>Revision of RG 1.200</td>
<td>Methods to assess the technical adequacy of PRA</td>
<td>Completed</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Completed as RG 1.200 R2, issued in March 2009</td>
</tr>
<tr>
<td></td>
<td>Revision of RG 1.201</td>
<td>Guidelines for the classification of SSCs by safety significance</td>
<td>Waiting for the revision of PRA standards</td>
</tr>
<tr>
<td>Risk-informed technical specifications</td>
<td>Risk-informed sophistication of standard technical specifications</td>
<td>Initiative 1</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Initiative 4b (AOT)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Initiative 5 (ST)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Initiative 6</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Initiative 8</td>
<td></td>
</tr>
</tbody>
</table>
### Field of operating reactors: Licensing 2/3

<table>
<thead>
<tr>
<th>Initiative</th>
<th>Program</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fire protection</td>
<td>NFPA Standard 805 National Fire Protection Association</td>
<td>Pilot application</td>
<td>Shearon Harris Completed in June 2010</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Oconee Completed in December 2010</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Standardization</td>
<td>Completed</td>
</tr>
<tr>
<td></td>
<td></td>
<td>RI and PB fire protection activities</td>
<td></td>
</tr>
<tr>
<td>Digital systems PRA</td>
<td>Risk-informed review of digital systems</td>
<td>Problem analysis (long term)</td>
<td>Waiting for additional studies</td>
</tr>
</tbody>
</table>
## Field of operating reactors: Licensing 3/3

<table>
<thead>
<tr>
<th>Initiative</th>
<th>Program</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>PRA quality</td>
<td>PRA standards and peer review guidance</td>
<td>Developed by standard development organizations (SDO) and industry</td>
<td></td>
</tr>
<tr>
<td></td>
<td>RG 1.200</td>
<td>1. A technically acceptable PRA guide</td>
<td></td>
</tr>
<tr>
<td></td>
<td>A guide for judging the technical adequacy of PRA</td>
<td>2. SDO consensus standards and peer review guidance</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>3. Proving that the technical adequacy of PRA is sufficient</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>4. Documentation</td>
<td></td>
</tr>
<tr>
<td></td>
<td>NUREG-1855: Guidance on the treatment of uncertainties</td>
<td>• Specific methods to express uncertainties in PRA</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Uncertainty analysis is conducted to study how results are affected.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• The results of the uncertainty analysis are taken into account in the decision-making.</td>
<td></td>
</tr>
</tbody>
</table>
# Field of operating reactors: Oversight 1/4

<table>
<thead>
<tr>
<th>Initiative</th>
<th>Program</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor performance data Program</td>
<td>A data collection and analysis program</td>
<td>Collect and analyze performance data for use in regulatory processes to identify potential safety issues.</td>
<td></td>
</tr>
<tr>
<td>The Accident Sequence Precursor (ASP) program</td>
<td>Operating experience is systematically studied to identify potential core damage sequences. Precursor events are classified into plant specific and generic events, and ASP is used as a trend index for the core damage risk.</td>
<td>Completed. The results and findings of the ASP program are presented in annual reports to the NRC and the Congress.</td>
<td></td>
</tr>
<tr>
<td>Risk-informed emergency action levels (EALs)</td>
<td>The core damage probability is calculated from the SPAR model for the emergency action levels (EALs) of each emergency classification level (ECL).</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
# Field of operating reactors: Oversight 2/4

<table>
<thead>
<tr>
<th>Initiative</th>
<th>Program</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>The reactor performance data program</td>
<td>The Industry Trends Program (ITP)</td>
<td>Industry performance trends are monitored to confirm the maintenance of safety.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>The Mitigating Systems Performance Index (MSPI)</td>
<td>Risks associated with changes in mitigating systems performance are monitored.</td>
<td>Office of Nuclear Regulatory Research (RES) MSPI project completed on March 31, 2010, report in preparation</td>
</tr>
<tr>
<td>Consequential Steam Generator Tube Rupture (C-SGTR)</td>
<td>Development of a detailed risk assessment method</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Risk-informed decision-making</td>
<td>Improvement of decision-making processes at the Office of Nuclear Reactor Regulation (NRR)</td>
<td>Development of instructions on emergency decision-making</td>
<td>Completed</td>
</tr>
</tbody>
</table>
## Field of operating reactors: Oversight 3/4

<table>
<thead>
<tr>
<th>Initiative</th>
<th>Program</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maintaining the technical bases of PRA</td>
<td>Development of standardized plant analysis risk (SPAR) models</td>
<td>Development of plant-specific models and support of user needs</td>
<td></td>
</tr>
<tr>
<td>Maintenance and updating of the SAPHIRE code</td>
<td>Maintenance and updating of software and user manuals</td>
<td></td>
<td></td>
</tr>
<tr>
<td>The technical guidance</td>
<td>Guidelines for analysis methods, and support for improvements and user needs</td>
<td></td>
<td></td>
</tr>
<tr>
<td>The technical support</td>
<td>Support for user needs, and on-call support for senior analysts and NRR</td>
<td>In practice</td>
<td></td>
</tr>
</tbody>
</table>
# Field of operating reactors: Oversight 4/4

<table>
<thead>
<tr>
<th>Initiative</th>
<th>Program</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>PRA quality</td>
<td>The PRA standards and peer review guidance</td>
<td>(Same as in &quot;Licensing&quot;)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>RG 1.200: A guide for judging the technical adequacy of PRA</td>
<td>(Same as in &quot;Licensing&quot;)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>NUREG-1855: A guidance on the treatment of uncertainties</td>
<td>(Same as in &quot;Licensing&quot;)</td>
<td></td>
</tr>
<tr>
<td>Sophistication of human reliability</td>
<td>Applicability to regulatory use; coordination at the working level; no</td>
<td>procedures and practices</td>
<td></td>
</tr>
<tr>
<td>analysis (HRA)</td>
<td>guidelines on stringency; shortage of empirical data</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Supervision of risk-informed emergency</td>
<td>Identification of boundary scenarios</td>
<td>Review of the applicability of regulatory systems</td>
<td></td>
</tr>
<tr>
<td>planning</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Risk-informed security regulation</td>
<td></td>
<td>Study of possibilities</td>
<td></td>
</tr>
</tbody>
</table>
Reference 4:

An example of application to the assessment of failure rate parameters using domestic data and uncertainties based on Bayesian statistics
Updating in terms of uncertainties based on Bayesian statistics

- Limitation in terms of knowledge and perception ⇒ The degree of confidence is expressed in terms of probability distribution for the unknown parameter (failure rate).

Bayesian processing

- An assessment is possible if prior information is obtained properly even when records are limited.
- The distribution is updated in response to the accumulation of information.

\[ \pi(\theta|D) = \frac{f(D|\theta) \cdot \pi(\theta)}{P(D)} \]

- Estimation of posterior distribution of \( \lambda \) Obtained as a probability density

The data is generated from the probability model of a likelihood function for the given failure rate.
A simple example of Bayesian statistics

The event occurrence rates \( (x/t) \) are the same \( (1E^{-4}/h) \) for all data sets.

Source: NUREG/CR-6823
A failure rate estimation model taking into account uncertainties in the number of failure events

To cope with data uncertainties, uncertainties in failure data collection from NUCIA is incorporated as a data collection probability model and an assessment is made using the hierarchical Bayesian method. Introduced as a prior distribution in the case of plant-specific assessments.

$\leftarrow$ represents probabilistic dependence, and $\leftarrow$ logical dependence.

When estimating a parameter, inductive inference is made from data in the reverse direction of these dependencies.

$P_i$: data collection probability at plant $i$
$X_i$: number of failure events at plant $i$
$Y_i$: number of observed events at plant $i$
$\alpha, \beta$: parameters of data collection probability distribution

$X_i \sim Bin(p_i, Y_i) = \sum_{y=0}^{Y_i} C_{Y_i} p_i^y (1 - p_i)^{Y_i - y}$

Data collection probability model

$\leftarrow$ represents probabilistic dependence, and $\leftarrow$ logical dependence.

$\lambda_i$: plant-specific failure rate at plant $i$
$X_i$: number of failure events at plant $i$
$T_i$: exposure time at plant $i$
$\mu, \sigma$: parameters of the distribution of the population
$a, b$: Hyperparameters

$X_i \sim Poisson(\lambda_i, T_i) = \exp(-\lambda_i T_i) \frac{(\lambda_i T_i)^{X_i}}{X_i!} = \exp(-\mu_i) \frac{\mu_i^{X_i}}{X_i!}$

Time failure rate model
An example of data reliability improvement in Japan: Data elaboration on failure rate

- Collection of function failure data (improved efficiency and data elaboration)
  In the process of judging maintenance preventable function failure (MPFF), PRA failure events are collected from events identified as function failure.

- Estimation of equipment failure rate
  Development according to the following steps.
  - Development of a general failure rate data spanning 21 years (reports and data will be released in NUCIA's Reliability Information)
  - Updating to 26 years the general failure rate data (reports and data will be released in NUCIA's Reliability Information)
  - Study on one-step Bayesian estimations using function failure data, with general failure rate data as a prior distribution
  - Updating to 29 years the general failure rate data (fiscal 1982–2010 data of NUCIA's trouble information)
  - Since FY 2011, the collection of failure data has shifted from NUCIA to function failure information
  - Estimation of plant-specific failure rate by the hierarchical Bayesian method using only function failure data