Amendment of Standard for Procedures of Level 1 Probabilistic Risk Assessment of Nuclear Power Plants during Power Operation

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ABSTRACT

A standard which is intended to provide requirements and methodologies for conducting level 1 probabilistic risk assessment (PRA) was amended. In this standard, PRA is carried out to evaluate the risk of nuclear power plants by evaluating the core damage frequencies due to internal initiating events during power operation. This standard was approved and published by the Atomic Energy Society of Japan (AESJ) on the deliberations at the Subcommittee on Level 1 PRA under the Risk Technical Committee of the AESJ Standards Committee. AESJ formulated and issued this document after deliberation by the Risk Technical Committee and the Standards Committee.

PRA of a nuclear power plant is a probabilistic approach for comprehensively and quantitatively assessing plant risk. This involves quantitative analysis of accident scenarios up to core damage and of the further evolution of events after core damage, with careful attention to events that are related to core or fuel damage, finally to arrive at an estimation of event frequencies and consequences.

In Japan, PRA has been implemented to evaluate the validity of Accident Management Strategies and the quantitative safety of nuclear power plants in the Periodic Safety Review. Furthermore, in the regulatory area, the Nuclear Regulation Authority, which is the new regulatory authority founded in 2012, intends to utilize PRA in their new safety regulation positively.

As the validity of PRA has come to be accepted, preservation of the quality and transparency of PRA has become important issues. Recognizing that preparation of a standard for PRA procedure is effective in addressing these issues, the AESJ standards committee is preparing a procedures guide for nuclear facilities.

As described above, this standard replaces the 2008 issue - A Standard for Procedures of Probabilistic Safety Assessment of Nuclear Power Plants during Power Operation (Level 1 PSA):2008 - of the same standard. We decided to make this amendment because five years have passed since the 2008 issue was published. For the present amendment, we updated various requirements in view of advancements in PRA techniques and to improve the quality and transparency of PRA.

KEYWORDS

PRA, Level 1, Standard

1. INTRODUCTION

As the validity of Probabilistic Risk Assessment (PRA) has been recognized and it is utilized in a wide range of applications in Japan, the preservation of the quality and transparency of PRA has become important issues. Recognizing that a standardized procedure for the implementation of PRA is effective in addressing these issues, the Atomic Energy Society of Japan (AESJ) Standards Committee has been working on the development and improvement of the standards on the implementation of PRA for nuclear facilities.

Among the PRAs implemented to comprehensively evaluate risks associated with internal events during power operations of a nuclear power plant, the standard on level 1 PRA focuses on the process up to the evaluation of core damage frequencies and describes the standardized method of implementing PRA including the requirements to be met and concrete evaluation methods.

This Standard replaces the 2008 issue "A Standard for Procedures of Probabilistic Safety Assessment of Nuclear Power Plants during Power Operation (Level 1 PSA): 2008[1]" of the same standard. We have decided to add some amendments to the existing standard as five years have passed since the 2008 issue was published. For the present amendment, we have updated various requirements in view of advancements in PRA techniques and to improve the quality and transparency of PRA.

This paper summarizes the policy for the establishment of the level 1 PRA standard, which will be formally published soon, and the contents of the standard.

2. POLICY OF REDRAFTING LEVEL 1 PRA STANDARD AND APPLICABLE SCOPE OF THE REVISED LEVEL 1 PRA STANDARD

2.1. Policy of redrafting Level 1 PRA Standard

In redrafting the level 1 PRA standard, it was intended to establish industry own safety assurance measures with the quality required to meet the regulatory requirements by incorporating the latest scientific and technical findings as far as possible and considering evaluation results and experience with PRA peer reviews conducted in Japan. To be more specific, the contents of the level 1 PRA standard were enhanced by:

- ✓ comparing with the internationally recognized ASME/ANS PRA standard[2] and IAEA safety standards[3] to identify the items which can be incorporated in the level 1 PRA standard taking into account the updated technical knowledge,

- ✓ coordinating with the contents of the "standard for PRA parameters[4]" and "standard for shutdown PRA[5]", which were published after the 2008 issue of level 1 PRA standard, to achieve the consistency, and

- ✓ incorporating the latest knowledge which has been identified following the Fukushima Daiichi NPP accident.

2.2. Applicable scope of Level 1 PRA Standard

This standard describes the specific method of performing level 1 PRA which includes the processes up to the evaluation of frequencies of accident sequences, which initiate from an internal event and result in core damage, among the PRAs targeted for nuclear power plants during power operations. The applicable scope of the 2008 issue remains unchanged in this standard.

3. CONTENTS OF LEVEL 1 PRA STANDARD

3.1. Structure of Level 1 PRA Standard

This standard consists of 14 sections. Each section includes the main text and appendix (requirements) which describe the requirements. In addition, reference information is shown in the appendix (reference) and explanation, which help understand the requirements of the standard.

1. Applicable scope
2. Cited codes and standards
3. Terms and definitions
4. Implementation procedure of level 1 PRA
5. Investigation of plant information
6. Selection of initiating events and estimation of frequencies
7. Identification of success criteria
8. Analysis of accident sequences
9. System reliability analysis
10. Human reliability analysis
11. Preparation of parameters
12. Quantification of accident sequences
13. Uncertainty analysis and sensitivity analysis
14. Documentation

The practical procedures of level 1 PRA, which are described from section 5 through 14, are shown in Fig.1 in a flow chart. Results from some processes are fed back to the following process and other processes may be performed in parallel.

As the validity of PRA has been recognized and it is utilized in a wide range of applications, assurance of the quality of PRA becomes more important. In order to assure quality PRA, this standard also includes the requirements regarding the utilization of expert judgments, implementation of peer review and quality assurance methods.

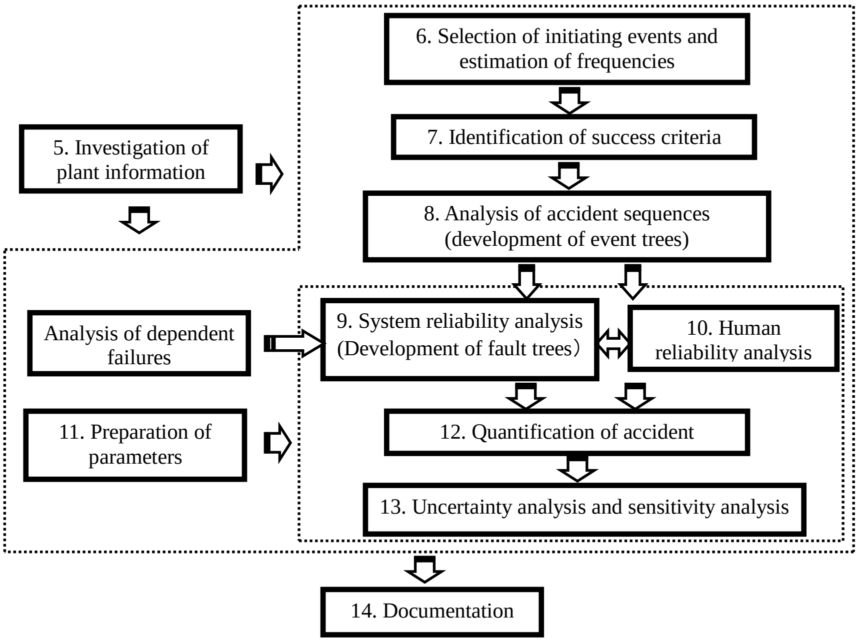

Fig.1 Flow of level 1 PRA procedure

The next section will explain the items to be implemented in each process of level 1 PRA and the issues raised in redrafting the standard.

3.2. Items to be implemented in each process of Level 1 PRA

3.2.1. Investigation of plant information

It is necessary to collect information required to implement level 1 PRA. If collected information is not sufficient for intended activities, either one of or combination of following methods will be used to supplement the information:

- ✓ Plant walkdowns
- ✓ Interviews with plant personnel and/or design engineers

The necessity of plant walkdowns was discussed during the process of redrafting the standard. Plant walkdowns are regarded essential in performing PRA for external events. For this standard targeted for internal events, it has been decided to conduct plant walkdowns as necessary after going through the collected design information.

3.2.2. Selection of initiating events and estimation of their frequencies

Initiating events which may lead to core damage shall be selected and the mean value of each initiating event frequencies and the distribution of probabilities shall be estimated.

In redrafting the standard, following issues were discussed:

- ✓ Initiating events shall be selected through the analysis of past incidents and the identification of initiating events in a systematic manner so that no necessary initiating event will be omitted. Then, analysis of precursor events shall be performed (the events which may not lead to plant trip shall be investigated and analyzed) to examine if any essential initiating event has been omitted from the analysis and identification or not.

- ✓ The screening criteria for initiating events was set the frequency of the event was less than 1E-7/reactor year. However, interfacing systems LOCA and C/V bypass events shall not be excluded.

- ✓ Calculation of the frequency of initiating events using the Bayesian approach shall be described according to the description in the PRA parameter standard. Other methods that can be used to calculate the frequency of initiating events when sufficient data are not available and it makes accurate estimation difficult are also described. If significant uncertainties are expected to exist in estimating the frequency of initiating events, the factors causing uncertainties, which may largely affect the estimation of the frequency of initiating events, are identified in order to determine the factors causing significant effects on the PRA result.

3.2.3. Identification of success criteria

Success criteria shall be identified according to the safety function and system by identifying the safety functions required to prevent core damage and analyzing the result of thermal hydraulic analysis and/or structural analysis which have been implemented.

In redrafting the standard, following issues were discussed:

- ✓ The success criteria by the system shall be identified for the support system as well as for the frontline system.

- ✓ The mission time for each mitigating system shall be set as one of the success criteria by the system. The concrete concept of setting the mission time shall be clearly stated.

- ✓ The identified success criteria, in particular uncertainties associated with the success criteria, may cause significant effects on the evaluation result. Therefore, factors causing the uncertainties, which are closely related to the identification of success criteria, shall be determined in order to clarify the factors significantly affecting the PRA result.

3.2.4. Analysis of accident sequences

Accident sequences which may lead to core damage are developed in a comprehensive manner for the selected initiating events based on the identified success criteria.

In this section, there is no item that has been significantly changed from the 2008 issue. Instead, this section describes the clarification of the relationships with other sections (for example, the dependency of initiating events is identified in section 3.2.2, and modeled in this section or the section of system reliability analysis), and explanation of terminology.

In addition, to enhance the contents of the appendix, the concept of setting the order of headings and several examples are introduced and the dependency between mitigating systems is shown.

3.2.5. System reliability analysis

A system reliability model, which corresponds to the headings of an event tree developed in the analysis of an accident sequence, shall be prepared, and the unavailability of the system shall be determined using the model.

In redrafting the standard, following issues were discussed:

✓ The contents shall be improved to include more details and more instances by referring to the ASME/ANS PRA standard.

✓ To clearly set the criteria for screening the basic events, the basic events whose failure rates are lower by 2 orders of magnitude may be excluded according to the ASME/ANS PRA standard.

✓ To estimate the parameters used for common cause failures, Article 7 of the PRA parameter standard shall be applied.

3.2.6. Human reliability analysis

Human errors which may occur in the course of performing activities prior to the occurrence of an initiating event and mitigating operations after the occurrence of an event are identified and the probabilities of the human errors are determined. For this purpose, human error probabilities are evaluated by using the human reliability analysis method.

In redrafting the standard, the contents are improved to include more details by referring to the ASME/ANS PRA standard. In addition, the human reliability analysis methods are discussed.

The THERP method can be used in a wide range of applications in a unified manner since the analysis method and values used in THERP have been fully developed. In addition, the THERP method has been used in PRAs performed by the industry, and thus the validity of this method in evaluating the risk improvement measures has been verified.

On the other hand, research activities on the next generation human reliability evaluation methods, such as ATHEANA, are under progress. However, with less experience in using those advanced methods in PRA, they are not acceptable as the PRA method as equivalent to THERP in this revised standard. Therefore, like the 2008 issue, this standard recommends the THERP method as the most reliable option.

3.2.7. Preparation of parameters

Equipment failure rates, which are necessary to perform the system reliability analysis, quantification of accident sequences, uncertainty analysis and sensitivity analysis, as well as other parameters required to evaluate the equipment unavailability due to tests or maintenance activities, shall be developed.

In redrafting the standard, it was determined to estimate the parameters according to the Bayesian approach based on plant specific data in principle. The structure of this section was changed according to the PRA parameter standard. The requirements described in this section are those used in the PRA parameter standard.

3.2.8. Quantification of accident sequences

Frequencies of accident sequences leading to core damage are calculated, core damage frequencies are determined and major results are analyzed. In addition, the importance indicator is identified considering the purpose and quantitatively evaluated to identify the dominant factors contributing to the core damage frequency.

In redrafting the standard, detailed requirements are described in the appendix (requirements) and the order of the requirements in the main text was changed in accordance with the practical procedure of PRA. As a result, the order of the requirements in the main text becomes the same as those in the shutdown PRA standard.

To clarify the distinction between the requirements and instances, which can be used in the examination and review of the evaluation results, the instances are described in the appendix (reference).

3.2.9. Uncertainty analysis and sensitivity analysis

The uncertainties associated with core damage frequencies and the sensitivity of factors affecting the PRA result shall be identified.

In redrafting the standard, considerations about SOKC in the uncertainty analysis shall be described in the appendix (reference), and the procedure of sensitivity analysis shall be additionally described by referring to the shutdown PRA standard. In addition, the mean value of the results of uncertainty analyses shall be regarded as the result of quantification according to the purpose of PRA.

3.2.10. Documentation

The purposes, applicable scope, methods used, conditions, models, parameters, and evaluation results of level 1 PRA shall be described as detailed as possible.

In redrafting the standard, examples of items which need to be described in the individual sections shall be listed in the appendix (reference).

4. CONCLUSIONS

Among the PRAs implemented to comprehensively evaluate risks associated with internal events during power operations of a nuclear power plant, the standard on level 1 PRA focuses on the process up to the evaluation of core damage frequencies and describes the requirements to be met and concrete evaluation methods. The Implementation standard for level 1 PRA has been revised as five years have passed since the 2008 issue was published. For the present amendment, we have updated various requirements in view of advancements in PRA techniques and to improve the quality and transparency of PRA.

The revised level 1 PRA standard is expected to contribute to the advancement of risk analysis for nuclear power plants in Japan, which will clarify the core damage frequencies and their effects to a more detailed extent, and thus these improvements will support the decision making process in various fields, such as the safety design, operation and maintenance and safety regulations.

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