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SENSITIVITY STUDY OF MODELING UNCERTAINTY ON RISK-INFORMED DECISION MAKING

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ABSTRACT

In the field of nuclear regulation, risk-informed decision making regarding licensing basis (LB) changes is being employed. Within risk-informed decision making a consideration of uncertainty propagation caused by parameters, models and such is a key issue. This paper describes a methodology for the risk-informed decision making considering both parameter uncertainty and modeling uncertainty that may have a significant impact on the decision. For this purpose, we classify modeling uncertainties of PRAs into three types in terms of mathematical form. Then a sensitivity analysis is performed using a simplified Probabilistic Risk Assessment (PRA) model developed by RISKMAN computer software and a hypothetical License Amendment Request to revise the Technical Specification Allowed Outage Time (AOT) for the High Pressure Coolant Injection (HPCI) system at a Boiling Water Reactor (BWR). This sensitivity analysis uses two risk metrics, which are core damage frequency (CDF) and the change in core damage frequency (ΔCDF). To both of them, we evaluate parameter uncertainty propagation by using Monte Carlo method and take confidential intervals for the metrics into account instead of point estimation. The modeling uncertainty has a huge impact on ΔCDF, and therefore, the decision is sensitive to the uncertainty related to structural change of the model.

1. INTRODUCTION

In the field of nuclear regulation, risk-informed decision making regarding licensing basis (LB) changes is being employed. Risk-informed decision making means using results of Probabilistic Risk Assessments (PRAs) for regulation of safety related activities in Nuclear Power Plants (NPPs). Instead of existing deterministic regulation, the risk-informed decision making will contribute to enhance safety and economic efficiency of NPPs, and to improve rationality and accountability of safety related activities in plants. Within the risk-informed decision making, a consideration of uncertainty is a key issue since the uncertainty may have a significant impact on the decision. The uncertainty can be categorized as either aleatory or epistemic uncertainty. Aleatory uncertainty reflects our inability to predict random observable events. On the other hand, epistemic uncertainty is associated with incompleteness in the analysts' state of knowledge about accidents at NPPs. The epistemic uncertainty is further divided into parameter, modeling and completeness uncertainty.

This paper describes a methodology for the risk-informed decision making considering both parameter uncertainty and modeling uncertainty which may have a significant impact on the decision. In this paper, modeling uncertainties of PRAs are categorized into three types in terms of mathematical form. Then a sensitivity analysis of modeling uncertainty related to logic structure of PRA models on risk-informed decision making regarding a LB change is performed to provide an example of the implementation of the proposed methodology. Previous study (Reinert et al., 2006) has performed a sensitivity analysis of modeling uncertainty which is related to a single basic event with concentrating on application of Level I, at-power, internal events PRAs. In this paper, however, we focus on Level I, at-power, both internal and external event PRAs and both parameter and modeling uncertainty that is related to logic structure. The decision making process which we use for the analysis is related to a hypothetical License Amendment Request to revise the Technical Specification allowed outage time (AOT) from 7 days to 14 days for the high pressure coolant injection (HPCI) system at a representative Boiling Water Reactor (BWR), Mark I plant. For the decision making, acceptance guidelines provided by AESJ with respect to a plant's core damage frequency (CDF) and the change in core damage frequency (ΔCDF) are used. For both risk metrics, we evaluate parameter uncertainty propagation by using Monte Carlo method and take confidential intervals for the metrics into account instead of point estimation.

2. RISK INFORMED DECISION MAKING

Since issuing PRA Policy Statement (USNRC, 1995), the US Nuclear Regulatory Commission (USNRC) has encouraged NRC staff to use PRA in nuclear regulatory activities. This is a paradigm shift in nuclear regulation and the new nuclear regulation has been called Risk-Informed Regulation (RIR). The RIR is not intended to be an alternative to traditional deterministic safety assessments but is used in concert with the traditional deterministic way. The Regulatory Guide 1.174 (USNRC, 2002) describes this issue providing a framework for risk-informed integrated decision making. In this guide, five principles for RIR regarding LB changes are provided. The principles are as follows:

(1) The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.
(2) The proposed change is consistent with the defense-in-depth philosophy.
(3) The proposed change maintains sufficient safety margins.
(4) When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (USNRC, 1986)
(5) The impact of the proposed change should be monitored using performance measurement strategies.

The first principle is about the abidance of rules. Second and third principles show that traditional deterministic approaches remain as important elements of nuclear safety regulation. The fourth principal is the one that we treat in this paper. Fifth principle requires that proposed licensing basis changes are performed properly and if necessary, it should be corrected.

Small risk increases which are required by the forth principle are defined using the acceptance guideline with respect to CDF and ΔCDF of Fig. 1. This acceptance guideline provided in RG 1.174 requires that the value of risk metrics (i.e., CDF and ΔCDF) are calculated by a full-scope (including internal events, external events, full power, low power, and shutdown) PRA and then compared with this acceptance guideline. In this guideline, there are three regions. In region I, no risk changes are allowed. Therefore, if the CDF value or ΔCDF value exceeds the threshold line between region I and region II, it means that the proposed LB change is prohibited. In region II, small risk changes are allowed, but it is required to track cumulative impacts of the LB changes. In region III, the risk increases are very small and the proposed LB change is justified. Fig. 1 also shows gradual shading, darkening when one moves to upward and to the right. This gradation corresponds to the level of review that the proposed LB changes will be given.

The Atomic Energy Society of Japan (AESJ) issued the standard of implementation on use of risk information in changing the safety related activities in NPPs in 2010 (AESJ, 2010). This standard defines the same principle as RG 1.174, but the acceptance guidelines are different. The standard categorizes the acceptance guidelines into two types. One is for internal events PRAs and the other is for full-scope PRAs. These acceptance guidelines are shown in Fig. 2 and Fig. 3. Fig. 2 is the acceptance guideline with respect to CDF and ΔCDF of internal events PRAs and Fig. 3 is the acceptance guideline with respect to CDF and ΔCDF of full-scope PRAs. This categorization of acceptance guidelines reflects the fact that full-scope PRAs have not been established although the internal events PRAs are established. Therefore the standard requires qualitative or both qualitative and quantitative assessment that shows the impacts of the proposed LB changes on external hazards risks are limited when assessing risks of no external hazard or a part of external hazards.

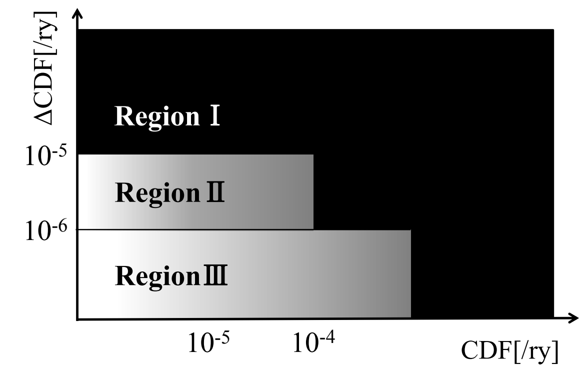

Fig. 1 Acceptance guideline for CDF (USNRC)

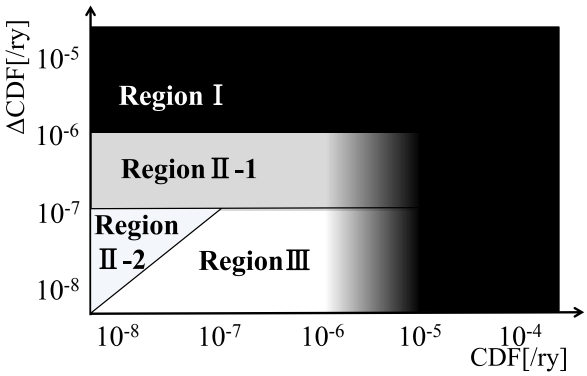

Fig. 2 Acceptance guideline for CDF of internal events PRAs (AESJ)

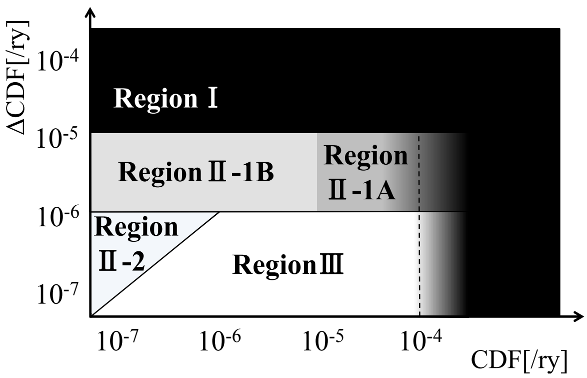

Fig. 3 Acceptance guideline for CDF of full-scope PRAs (AESJ)

The regions defined in AESJ acceptance guidelines are also different from NRC's one. This paper focuses on not only internal events PRAs but also external events PRAs, so we will deal with the acceptance guideline for full-scope PRAs. The region I is same as NRC's definition. In this region, no risk increases are allowed. Region II is further categorized into three regions: Region II-1A, Region II-1B and Region II-2. The region II-1A requires check of CDF and compensatory measures to control the risk increases with respect to the proposed LB changes. The region II-1B requires the compensatory measures same as region II-1A. The region II-2 requires examining the necessity of the compensatory measures same as region II-1A. This region is defined by relative value of risks metrics. In this acceptance guideline, when the value estimated by dividing ΔCDF by CDF is not greater than 1, the ratio of the risk increase is interpreted as insignificant. In the region III, the risk is very small and so the proposed LB changes are allowed.

The acceptance guidelines of RG 1.174 and AESJ are not intended to be interpreted as overly prescriptive. They are intended to provide an indication from the quantitative point of view. What is important is to follow the five principles described above. And also, RG 1.174 and AESJ standard says that when performing comparison of risk metrics with the acceptance guidelines, it is required to assess impacts of uncertainties of PRAs.

The uncertainties of PRAs are categorized as either aleatory or epistemic uncertainty. The former is uncertainty results from randomness associated with the events of the model and is inevitable. The latter is uncertainty caused by incompleteness of the analysts' state of knowledge or lack of data used for PRAs. Therefore getting knowledge about PRAs can reduce the latter uncertainty.

USNRC NUREG-1855 (USNRC, 2009) provides categorization of the epistemic uncertainty. According to NUREG-1855, the epistemic uncertainty is further classified into three uncertainties: parameter uncertainty, modeling uncertainty and completeness uncertainty. The definitions of these three epistemic uncertainties provided in NUREG-1855 are as follows:

- Parameter Uncertainty
  Parameter uncertainty is the uncertainty in the values of the parameters of a model given that the mathematical model has been agreed to be appropriate.

- Modeling Uncertainty
  Model uncertainty is related to an issue for which no consensus approach or model exists and where choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, and introduction of a new initiating event).

- Completeness Uncertainty
  Completeness uncertainty relates to risk contributors that are not in the PRA model. These types of uncertainties either are ones that are known but not included in the PRA model or ones that are not known and therefore not in the PRA model.

We call the model uncertainty defined in NUREG-1855 "Modeling Uncertainty". According to NUREG-1855, "*Although the analysis of parameter uncertainty is fairly mature and is addressed adequately through the use of probability distributions on the values of the parameters, the analysis of the model and completeness uncertainties cannot be handled in such a formal manner.* " Hence this paper focuses on treatment of modeling uncertainty.

3. CATEGORIZATION OF MODELING UNCERTAINTY

In this section, we provide categorization of modeling uncertainty. According to NUREG-1855 and EPRI technical report (EPRI, 2008), the sources of modeling uncertainty are linked to:

- A single basic event
- Multiple basic events
- The logic structure of the PRA
- Logical combinations

Basic events of PRAs are categorized into three types as follows:

- Occurrence of initiating events.
- States if unavailability or failure of structure, systems, and components (SSCs).
- Human failures that contribute to the failure of the system designed to protect against the undesirable consequences should an initiating event occur.

Therefore modeling uncertainty related to these issues is defined as modeling uncertainty related to basic events. Modeling uncertainty that is related to only a single basic event is separated from those related to multiple basic events since the latter modeling uncertainty may have the combined impact on results of PRAs. The modeling uncertainty related to the logic structure of the PRA is the uncertainty related to alternative methods or assumptions that could possibly introduce new cutsets in existing sequences by changing the structure of event trees, or even entirely new classes of accident. The logical combination stands for a combination of basic events and logic structure. According to NUREG-1855, "*This combination may impose a synergetic impact on the uncertainty of the PRA results. The resulting uncertainty from their total impact may be greater than the sum of their individual impacts.*" Therefore consideration of the logical combinations is important.

This categorization is made since not every application involves every aspect of the PRA. But there is no difference between modeling uncertainty related to a single basic event and the one related multiple basic events in terms of mathematical form. Therefore we provide new categorization of modeling uncertainty in terms of mathematical form as follows:

(1) Modeling uncertainty related to basic events
This is modeling uncertainty related to either a single basic event or multiple basic events. This modeling uncertainty is described as F(a) → F(a') mathematically. The arrow represents a change from a base PRA model to a modified PRA model considering the modeling uncertainty. The "a" stands for input variable and the "F" stands for mathematical form of logical structure of PRAs. And the "a'" is modified input variable to address the modeling uncertainty. Therefore we can say that this modeling uncertainty is related to only input variables of PRA models.

(2) Modeling uncertainty related to logic structures
This is modeling uncertainty related to logic structures of PRAs. This modeling uncertainty is described as F(a) → F'(a) mathematically. The "F'" stands for mathematical form of logical structure of modified PRAs.

(3) Modeling uncertainty related to logical combinations
This is modeling uncertainty related to both basic events and logic structures of PRA models. We call this uncertainty modeling uncertainty related to logical combinations. This modeling uncertainty is described as F(a) → F'(a') mathematically.

The EPRI technical report (EPRI, 2008) provides examples of sources of modeling uncertainty.

4. SENSITIVITY ANALYSIS OF MODELING UNCERTAINTY

As noted in section 2, it is important to assess the impacts of uncertainty of PRAs in risk-informed decision making regarding LB changes. NUREG-1855 provides guidance on the treatment of uncertainties associated with PRAs in risk-informed decision making. In this section, we introduce a methodology for considering modeling uncertainties in the risk-informed decision making regarding LB changes.

Fig. 4 shows the overall process provided in NUREG-1855 in order to assess modeling uncertainties in the risk-informed decision making. As described in Fig. 4, only the relevant sources of uncertainties that have the potential to impact the decision are considered key. Therefore, after listing relevant sources of model uncertainties, sensitivity analyses are performed to determine importance of the source of model uncertainty to the acceptance criteria. This sensitivity analyses are categorized into two types: a conservative sensitivity analysis and a realistic sensitivity analysis. In this paper we focus on the conservative sensitivity analysis. The following passage describes the methodology for the sensitivity analysis for each modeling uncertainty defined in section 3.

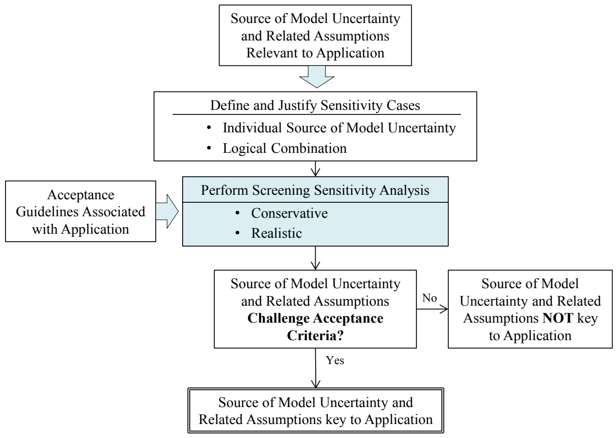

Fig. 4 Process to identify key sources of modeling uncertainty and related assumptions

4.1 Modeling Uncertainty Related to Basic Events

For each identified source of modeling uncertainty related to basic events, a conservative sensitivity analysis is performed. In this analysis, the following metrics are used for risk-informed decision making regarding LB changes:

CDF_{base} the value of the CDF estimate in the base PRA (i.e., the frequency at which core damage is expected to occur at the point if no plant changes are made)

CDF_{new} the value of the CDF estimate in the modified PRA to account for changes proposed to the licensing basis

$CDF^{+}_{j,base}$ the CDF value estimate in the base PRA with the basic event j set to 1

$CDF^{+}_{j,new}$ the CDF value estimate in the modified PRA with the basic event j set to 1

Using these values, the metrics ΔCDF and ΔCDF^{+} are defined as follows:

$$\Delta CDF = CDF_{new} - CDF_{base} \tag{1}$$

$$\Delta CDF^{+} = CDF^{+}_{j,new} - CDF^{+}_{j,base} \tag{2}$$

In NUREG-1855, mean values are used for the risk metrics CDF and ΔCDF. However this paper use not only mean value but also 5 percentile and 95 percentile of CDF and ΔCDF. This is because the purpose of this study is performing detail assessment of impacts of uncertainties on risk-informed decision making. As for the issue, more details of the reasoning behind the approach to use mean value can be found in SECY-97-221 (USNRC, 1997).

After calculating the set of risk metrics ($CDF^{+}_{j,base}$, ΔCDF^{+}), we compare these metrics with the acceptance guideline shown in Fig. 3 and make decision whether or not to approve the change. If the pair of the metrics were to lie in a region I, the modeling uncertainty would be potentially key and more detailed analysis would be required. In the case that the modeling uncertainty is related to multiple basic events, the concept of setting all relevant basic events to 1 simultaneously is used for assessing the impacts of the modeling uncertainty on the risk-informed decision making. However, the combined impacts of the modeling uncertainty should be considered.

4.2 Modeling Uncertainty Related to Logic Structures

For each identified source of modeling uncertainty related to logic structures, a conservative sensitivity analysis is performed. In this analysis, the following metrics are used for risk-informed decision making regarding the LB change:

$CDF^{+}_{j,base}$ the base PRA CDF value estimate where the base PRA has been modified to address the j[th] source of model uncertainty that is linked to the logic structure of the PRA.

$CDF^{+}_{j,new}$ the base PRA CDF value estimate where the PRA as modified for the proposed LB change, has been further modified to address the j[th] source of model uncertainty that is linked to the logic structure of the PRA.

Using these two values and Eq. (2), the metrics CDF and ΔCDF are calculated where the PRA as modified for the proposed LB change, has been further modified to address the j[th] source of modeling uncertainty that is linked to the logic structure of the PRA. After calculating the set of risk metrics ($CDF^{+}_{j,base}$, ΔCDF^{+}), we compare these metrics with the acceptance guideline shown in Fig. 3 and make decision whether or not to approve the change.

4.3 Modeling Uncertainty Related to Logical Combinations

This modeling uncertainty is uncertainty related to both basic events and logic structures. For this modeling uncertainty, consideration of the synergetic impacts of the combination upon the uncertainty of the PRA results is required.

The sensitivity analysis for this type of modeling uncertainty is performed in two steps. Firstly the same approach as the case of modeling uncertainty related to logic structures is performed. The modified PRA structure then can be used for assess the impacts of modeling uncertainty related to basic events.

5. CASE STUDY

We present a case study to provide an example of the implementation of the methodology mentioned in this paper. The example risk-informed regulatory application is a hypothetical License Amendment Request to revise the Technical Specification Allowed Outage Time (AOT) from 7 days to 14 days for the High Pressure Coolant Injection (HPCI) system at a representative Boiling Water Reactor (BWR), Mark I plant. The purpose of the extension of AOT is to provide additional time to perform testing, maintenance, or make repairs without significantly affecting plant safety. This increased flexibility in work scheduling may benefit system reliability.

In order to perform the case study, a simplified external event and an internal event are provided using RISKMAN computer software. As an external event, seismically induced Loss of Offsite Power (LOOP) initiator event tree (ET) is provided and as an internal event, Medium Loss of Coolant Accident (MLOCA) is provided. These event tree structures are shown below in Fig. 5 and Fig. 6.

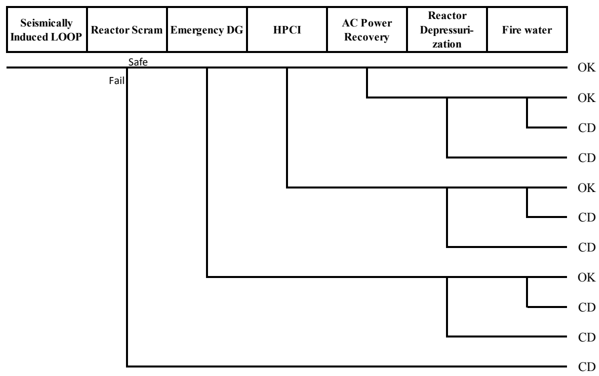

Fig. 5 Seismically Induced LOOP ET

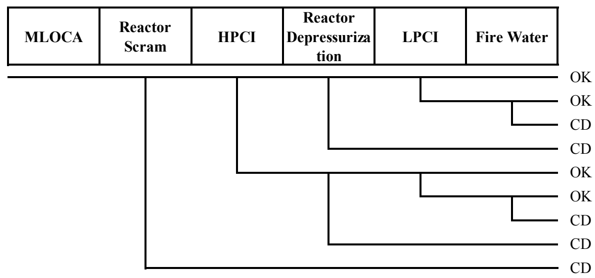

Fig. 6 MLOCA ET

The new annual average CDF due to the change in the AOT (i.e., CDF_{new}) is given by the following equation:

$$CDF_{new} = \left(\frac{T_A}{T_{cycle}}\right) CDF_A + \left(1 - \frac{T_A}{T_{cycle}}\right) CDF_{base} \tag{3}$$

where CDF_A is CDF evaluated from the PRA model with the HPCI system out of service and CDF_{base} is CDF evaluated from the base PRA model.

$$\Delta CDF = CDF_{new} - CDF_{base} \tag{4}$$

The relevant input data to Eq. (3) and Eq. (4) and the results are shown in Table 1. In this case study, we consider both parameter uncertainty and modeling uncertainty. Hence we propagate parameter uncertainty by using Monte Carlo method and take confidential intervals into account. The results are shown as mean, 5 percentile and 95 percentile. These values represent a point in the CDF acceptance guideline as shown in Fig. 7.

Table 1 PRA Input Values and base CDF and ΔCDF

Parameter and Metrics	Total		
	mean	5%	95%
CDF_{base}	1.35E-06	8.21E-09	4.42E-06
CDF_{A}	1.01E-05	8.26E-08	3.82E-05
CDF_{new}	1.68E-06	1.11E-08	5.72E-06
ΔCDF	3.34E-07	2.85E-09	1.29E-06
T_{A}		14 Days	
T_{cycle}		365 Days	

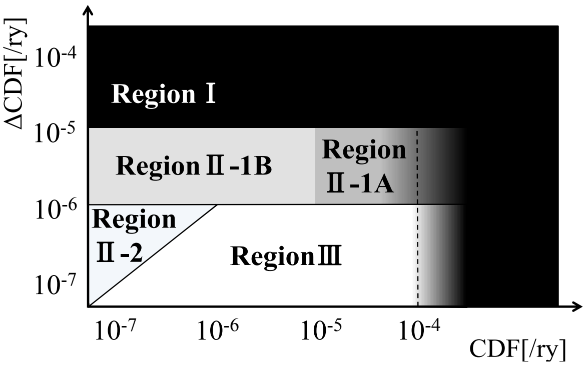

Fig. 7 Acceptance guideline for CDF of full-scope PRA (AESJ)

5.1 Assessment of Modeling Uncertainty

In this case study, we focus on a modeling uncertainty related to logic structure. The modeling uncertainty that we assess is associated with modeling of High Pressure Coolant Injection (HPCI) in a PRA for a BWR. In core damage sequences of seismically induced LOOP event tree and MLOCA event tree, failure of HPCI can be coupled with failure of reactor depressurization and failure of low-pressure injection (e.g., fire water injection) due to the following three reasons: Firstly, human error probability of reactor depressurization following HPCI failure increases due to the deterioration of environment where operators work, increase of stress and shortening of available time for performing reactor depressurization successfully. Secondary, in the case of HPCI fail following a certain time working of HPCI, the failure probability of reactor depressurization increases due to deterioration of the steam condensing performance. The HPCI uses suppression pool for water source and heat sink. Therefore after a certain time working of HPCI, the pressure and temperature in the PCV (Power Containment Vessel) may rise, and therefore, failure probability of reactor depressurization could increase. Thirdly, the low-pressure injection system doesn't work after the failure of reactor depressurization. Because of these three reasons, credibility for the reactor depressurization and low-pressure injection is key issue. In general, modeling uncertainties associated with the issue could have synergetic impact on PRA.

Considering this modeling uncertainty, we modify the seismically induced LOOP initiator ET and the MLOCA initiator ET. This modification reflects no credit for reactor depressurization and fire water injection after failure of HPCI. Fig. 8 shows the modified ET of seismically induced LOOP.

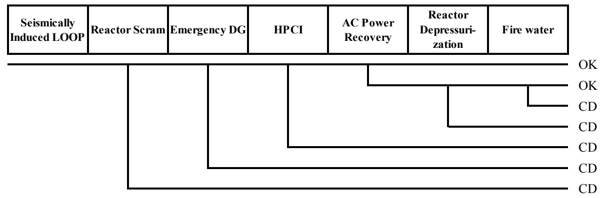

Fig. 8 Seismically induced LOOP ET considering modeling uncertainty

In this modified PRA, we calculate CDF^{+}_{base}, CDF^{+}_{A}, CDF^{+}_{new} and ΔCDF^{+}, where these terms are now defined as follows:

CDF^{+}_{base} the annual average PRA CDF where the base PRA has been modified to address the modeling uncertainty related to the logic structure.

CDF^{+}_{A} the annual average PRA CDF evaluated from the modified PRA model with the HPCI system out of service.

CDF^{+}_{new} the annual average PRA CDF where the PRA, as modified for the LB change application, has been further modified to address the modeling uncertainty related to the logic structure. This metric is calculated by the Eq. (5).

$$CDF^{+}_{new} = \left(\frac{T_A}{T_{cycle}}\right) CDF^{+}_{A} + \left(1-\frac{T_A}{T_{cycle}}\right) CDF^{+}_{base} \tag{5}$$

ΔCDF^{+} Increase of CDF regarding the LB change application. This metric is obtained as the result of Eq. (6).

$$\Delta CDF^{+} = CDF^{+}_{new} - CDF^{+}_{base} \tag{6}$$

The CDF^{+}_{base} and ΔCDF^{+} are obtained as shown in Table 2.

Table 2 Modified PRA Output Values

Metrics	Total		
	mean	5%	95%
CDF^{+}_{base}	6.49E-06	7.19E-08	2.38E-05
CDF^{+}_{A}	2.63E-04	2.07E-05	1.53E-03
CDF^{+}_{new}	1.63E-05	8.62E-07	8.17E-05
ΔCDF^{+}	9.83E-06	7.90E-07	5.79E-05

Table 3 and Fig. 9 show a comparison between the results and acceptance guideline provided in Fig. 7. In this case, only the 95 percentile of the modified PRA

ΔCDF (i.e., ΔCDF^{+}) exceeds the threshold line between Region I and Region II and is included in Region I. Therefore it is prohibited to perform the LB change (i.e., extension of AOT for HPCI system) with considering the modeling uncertainty when we use 95 percentile for risk metrics. As for the mean value of the risk metrics, the set value ($CDF^{+}_{base,}$ ΔCDF^{+}) is almost on the line between region I and region II-1B. Thus compensatory measures are required in order to control the risk increases with respect to the proposed LB changes. These results show that the modeling uncertainty related to the modeling of HPCI has impact on the decision regarding the LB change. In other words, it is important to take measures in order to assure the success of reactor depressurization and fire water injection. For example, introducing a filtered containment venting system with rupture disk could prevent PCV from overpressure caused by HPCI and could lead to reduce the modeling uncertainty and mitigate the impact of the modeling uncertainty on the decision.

The modification of PRA to address the modeling uncertainty was quite severe since the PRA model used in this case study was very simple and had only one Emergency Core Cooling System (ECCS). In addition, the way of considering the modeling uncertainty associated with modeling of HPCI was very severe. Hence more detail PRA model and more realistic sensitivity analysis would be required in order to assess the impact of the modeling uncertainty on the decision making. Furthermore, it is important to consider that the result obtained by the case study cannot be applied to every BWR since the PRA model used in this paper consisted of just two initiating events of one selected design of a BWR. Therefore it would be required to perform site-specific PRA in order to assess the impact of the modeling uncertainty on the decision making.

Table 3 Comparisons of Results to Acceptance Guideline

Metrics	Total Value			Acceptance Guideline	Below Acceptance Guideline		
	mean	5%	95%		mean	5%	95%
CDF_{base}	1.35E-06	8.21E-09	4.42E-06	1.00E-04	Yes	Yes	Yes
ΔCDF	3.34E-07	2.85E-09	1.29E-06	1.00E-05	Yes	Yes	Yes
CDF^{+}_{base}	6.49E-06	7.19E-08	2.38E-05	1.00E-04	Yes	Yes	Yes
ΔCDF^{+}	9.83E-06	7.90E-07	5.79E-05	1.00E-05	Yes	Yes	No

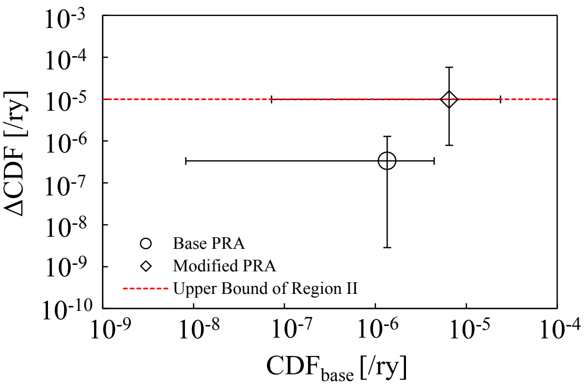

Fig. 9 Sensitivity Analysis of CDF and ΔCDF

6. CONCLUSIONS

In this paper, modeling uncertainties of PRAs have been categorized into three types in terms of mathematical form. Then a sensitivity analysis of modeling uncertainty related to a logic structure of a PRA model on risk-informed decision making regarding a LB change has been performed. We focused on Level I, at- power, both internal and external event PRAs and the decision making process related to a hypothetical License Amendment Request to revise the Technical Specification AOT from 7 days to 14 days for the HPCI system. For the decision making, the acceptance guideline provided by AESJ with respect to a plant's CDF and ΔCDF have been used.

In our case study, we considered both parameter uncertainty and modeling uncertainty. The modeling uncertainty was associated with modeling of HPCI in a PRA for a BWR. As a result, only 95 percentile of the modified PRA ΔCDF (i.e., ΔCDF^{+}) exceeded the threshold line between Region I and Region II and was included in Region I. Thus the decision was sensitive to the modeling uncertainty related to the logic structure and it was prohibited to perform the LB change (i.e., extension of AOT for HPCI system) with considering the modeling uncertainty when using 95 percentile for risk metrics. As for the mean value of the risk metrics, the set value ($CDF^{+}_{base,}$ ΔCDF^{+}) is almost on the line between region I and region II-1B. Thus compensatory measures are required in order to control the risk increases with respect to the proposed LB change.

These results show that the modeling uncertainty related to the modeling of HPCI has impact on the decision regarding the LB change. In other words, it is important to take measures in order to assure the success of reactor depressurization and fire water injection. For example, introducing a filtered containment venting system with rupture disk could prevent PCV from overpressure caused by HPCI and could lead to reduce the modeling uncertainty and mitigate the impact of the modeling uncertainty on the decision.

The modification of PRA to address the modeling uncertainty was quite severe since the PRA model used in this case study was very simple and had only one Emergency Core Cooling System (ECCS). In addition, the way of considering the modeling uncertainty associated with modeling of HPCI was very severe. Hence more detail PRA model and more realistic sensitivity analysis would be required in order to assess the impact of the modeling uncertainty on the decision making. Furthermore, it is important to consider that the result obtained by the case study cannot be applied to every BWR since the PRA model used in this paper consisted of just two initiating events of one selected design of a BWR. Therefore it would be required to perform site-specific PRA in order to assess the impact of the modeling uncertainty on the decision making.

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