(2) Overview of the Requirements of the Level 2 PRA Standard Extended to Tsunami Events in Japan

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Abstract: This paper presents an overview of the major requirements added in the revised edition of the Level 2 PRA standard extended to tsunami events by the Atomic Energy Society of Japan.

In tsunami PRA, tsunami hazard analysis is important. The revised edition requires the use of tsunami hazards that comply with the separately established in tsunami Level 1 PRA standard and are equivalent to those used in the Level 1 PRA.

For SSCs affected in Level 2 PRA, the preparation of a tsunami equipment list is required, and the tsunami fragilities of those SSCs are required to use the fragilities evaluated by a method that meets the requirements specified in the tsunami Level 1 PRA standard. In addition, considerations of flood in buildings are requested.

Some containment failure modes specific to tsunami events such as containment direct failure due to water pressure, buoyancy force, and water flow are added.

As stated in the ASME/ANS PRA standard, each accident sequence is required to be classified into large early release or large release and then large early release frequency (LERF) and large release frequency (LRF) can be evaluated.

One of the topics carefully discussed in the revision is how to clarify the end state of containment venting: success of containment venting should be included as a containment failure event or a controlled release event. The point of discussion is how to credit filtered venting system which is recently introduced as a severe accident management feature for containment function. In the Level 2 PRA standard, whether containment venting sequences can be included in containment failure or not depends on the purpose of the evaluation.

In the analysis of accident mitigation measures, it is required to analyze the tsunami effects on access to the mitigating features, including access to the location of use the portable equipment, transport of associated materials and equipment, and field operations required for expected mitigating operations during the accident.

Keywords: Standards, Level 2 PRA, Tsunami Events

1. INTRODUCTION

The Level 2 PRA standard established by the Atomic Energy Society of Japan was based on internal events or seismic events during power operation in the previous version, and the scope of application was extended to tsunami events in the revised version.

Tsunami events have the following characteristics.

- The possible tsunami hazard at the target site is assessed on the basis of active fault data and historical earthquake data. Tsunami may also be generated by factors other than earthquakes.
- Tsunami could damage multiple SSCs in the plant at the same time, including redundant systems.
- The probability of failure of SSCs due to tsunami depends on the configuration and on the tsunami height.

- Damage to buildings and equipment caused by the tsunami may deteriorate the accessibility to the site, affecting mitigation operations and recovery.

These characteristics of tsunami events need to be properly taken into account in Level 2 PRA. In the revision extending to tsunami events, the following points were noted:

- Keep in mind that the tsunami level 2 PRA is to be implemented on the basis of the results of the tsunami level 1 PRA, and maintain consistency in the interface between the level 1 PRA and the level 2 PRA, taking into account the impact of the tsunami.
- Use the same tsunami hazard as Level 1 PRA.
- Use consistent tsunami fragility for common Level 1 PRA and Level 2 PRA installations.
- Prepare building and equipment lists for Level 2 PRA-specific equipment and evaluate tsunami fragility for them.
- Evaluation of building flooding is carried out.
- Consider tsunami-specific containment failure modes.

In addition to the above extensions to tsunami events, the following requirements were added in comparison with the U.S. NRC regulatory guides [1], the ASME/ANS PRA standard [2], and Japanese regulations.

- The large early release frequency and/or large release frequency (LERF/LRF) was reflected in the definitions and provisions.
- Required clarification of whether containment venting should be included in containment failure in light of the objectives of the Level 2 PRA.
- Required clarification of the technical validity of any credits for beneficial failures to have a beneficial effect on the mitigation function.

The main features of the revision are described in the next chapter for each of the following technical elements.

(a) Investigation of plant configurations and characteristics
(b) Classification of plant damage states and quantification of frequencies
(c) Analysis of containment loads and setting of containment failure modes
(d) Analysis of accident sequences
(e) Accident progression analysis
(f) Setting the branch probabilities of the containment event tree
(g) Quantification of containment failure frequencies
(h) Classification of release categories and quantification of frequencies
(i) Source term analysis for release categories

In this document, where tsunami is not mentioned, the applicable requirements are the same as for internal events and seismic events.

2. KEY LEVEL 2 PRA STANDARD IMPLEMENTATION REQUIREMENTS

2.1. Investigation of Plant Configurations and Characteristics

Investigate and collect the latest information necessary for the implementation of the tsunami level 2 PRA for the analyzed plants.

(a) Check the tsunami hazard data used in the tsunami level 1 PRA. Confirm that the upper limit of tsunami height has a negligible impact on the frequency of loss of containment function even if a larger tsunami height is considered when implementing tsunami level 2 PRA.
(b) Collect and analyze tsunami fragility assessment information. Investigate the equipment and buildings that would affect the progress of the accident after core damage, including the source term, if the function is lost due to the tsunami.
(c) Building and equipment lists related to tsunami level 2 PRA are prepared based on equipment modeled in internal event level 2 PRA.

(d) Site plant walkdowns or interviews with plant staff and/or design engineers will be conducted to supplement the information. The site plant walkdown checks the accessibility and operating environment of facilities that consider restoration operations when necessary, facilities that require mitigation operations, etc.
(e) Use information specific to the plant being analyzed. If similar plant information is used, ensure that it is applicable to the plant being analyzed.

2.2. Classification of Plant Damage States and Quantification of Frequencies

Based on the results of tsunami level 1 PRA, all accident sequences leading to core damage are classified into plant damage states based on the similarity of accident evolution and/or mitigation operations, and the frequency (point estimate and probability distribution) for each plant damage state is quantified. In addition, tsunami-specific accident sequences are classified. Examples of tsunami-specific accident sequences include multiple signal system damage.

It is also acceptable to integrate the tsunami level 1 PRA event tree with the tsunami level 2 PRA event tree to conduct a series of analyses from the occurrence of the initiating event to the containment failure, without using plant damage states.

2.3. Analysis of Containment Loads and Setting of Containment Failure Modes

For accident sequences resulting in containment failure due to tsunami, containment failure modes are classified (see Table 1). In addition, the loads that affect the structural integrity of the containment resulting from the development of the accident after the tsunami are analyzed and identified, and the containment failure mode is classified for the accident sequence leading to the containment failure based on the containment structural integrity evaluation for the containment load. In addition, considering the effect of tsunami, the uncertainty factor assumed in the resistance evaluation of the containment structure is analyzed, and the containment fragility evaluation for the load of the containment is carried out. Also, the location and scale of the containment failure are analyzed including uncertainties. Furthermore, the tsunami fragility of SSCs that affect level 2 PRA and are affected by tsunamis will be evaluated. If the inside of the building is flooded, a flood assessment of the building is conducted.

(a) Analysis of loads resulting in containment failure

The type of load that affects the structural integrity of the containment vessel caused by the accident progression after the tsunami is systematically extracted corresponding to the following plant state.

- Conditions from the occurrence of an accident to core damage
- Condition from core damage to reactor (pressure) vessel failure
- Condition immediately after the failure of the reactor (pressure) vessel or after the start of the molten core concrete interaction
- Burned hydrogen leaked into the building
- Condition that the inside of the building was flooded
- Condition of loss of decay heat removal function from containment vessel due to tsunami

(b) Identifying loads

For each type of extracted load, the parts of the containment structure to which the load is applied are identified for the following loads:

- Static pressure load
- Thermal load and local thermal load
- Dynamic pressure loads, local dynamic pressure loads and missiles
- Load by indirect mechanism
- Loads due to water pressure, buoyancy and water velocity caused by flooding of the containment structure

(c) Evaluation of the integrity of containment structures and setting the criteria

The resistance of the containment structure to the containment load caused by the identified accident progression is evaluated. If the resistance of containment structures is credited to decrease due to ground motions, appropriate consideration should be given. In addition, when the effect of radioactive material deposition in a building is considered, the resistance of the building is evaluated. Based on the results of the resistance evaluation, the criteria for determining the integrity of the containment structure are set.

(d) Uncertainty evaluation of containment structure resistance

Uncertainty factors assumed in the resistance evaluation are analyzed, and fragility of the containment structure against the load of the containment is evaluated. When aging such as corrosion has occurred, the effect of aging should be considered in the fragility evaluation. The uncertainty assessment uses the specific assessment results of the plant to be analyzed. In the case of multiple units in the same site, the evaluation results of individual units are used.

(e) Analysis of containment bypass events and containment isolation failure events

Based on the analysis of the scenario of containment failure due to tsunami carried out at tsunami level 1 PRA, the scale of failure and the location of failure are analyzed for containment bypass events and containment isolation failure events separately from the containment structural integrity assessment. Containment bypass includes the following events:

- Steam generator tube rupture
- Interface system LOCA
- Isolation condenser (IC) tube rupture

(f) Setting the type of containment failure mode

In addition to the results of the evaluation of the integrity of the containment structure, the analysis of the containment bypass event and the containment isolation failure event, the containment failure mode is set from the type of the containment failure mode, including the case where the integrity of the containment is maintained.

In addition to the containment failure modes in Table 1, damage to the containment body due to tsunami is considered.

If there are containment failure modes that are not included in these, they may be added. If the containment failure modes included in these are excluded, the reason for this should be clarified.

Clarify whether containment vents should or should not be included in containment failure for the purposes of level 2 PRA.

Based on the results of the integrity evaluation of the containment structure, the containment failure mode is the one that reaches the loss-of-function condition earliest for each accident sequence. Also, in order to set the source term analysis condition, the location and scale of the containment failure are analyzed including uncertainty.

In the case of LERF and/or LRF quantification, for each accident sequence, large early or large late releases should be classified from the time of release of radioactive material into the environment, and the release mode should be set.

With the accident progression, loads affecting the integrity of the containment structure occur due to various phenomena in the containment (see Figure 1). Various experimental and analytical studies have been conducted on the mechanisms and the scale of the effects of these phenomena, and are still ongoing. It is necessary to consider the latest knowledge in the evaluation of containment load due to phenomena. It is also necessary to consider the latest findings on the effects of severe accident management. Accordingly, the main findings are presented in the annex and updated periodically (see Table 2).

(g) Evaluation of tsunami fragility of important SSCs

Evaluation of the tsunami fragility of the extracted important SSCs is carried out according to the Tsunami Level 1 PRA standard [3]. In the case of flooding in the building, flooding evaluation in the building is carried out.

Table 1. Classification of Containment Failure Modes (PWR Example)

Release Category	Status		Failure Mode
Leak	Design Leakage		Intact
Venting	Containment Venting		Containment Venting
Large Early Release	Containment Bypass		SG Tube Rapture, Induced SGTR
			IS-LOCA, Induced IS-LOCA
	Containment Isolation Failure		Failure of Containment Isolation
	Containment Failure	Early	RPV vertical displacement due to blowdown forces
			Steam Explosion (In-Vessel)
			Hydrogen Detonation (before RV Failure)
			Hydrogen Detonation (after RPV Failure)
			Steam Explosion (Ex-Vessel)
			Direct Containment Heating
			Direct Melt Attack
Large Late Release		Late	Hydrogen Detonation (long after RPV Failure)
			Combustion within Reactor Buildings or Auxiliary Buildings
			Concrete Erosion
			Overtemperature
			Overpressure
			Overpressure before Core Damage

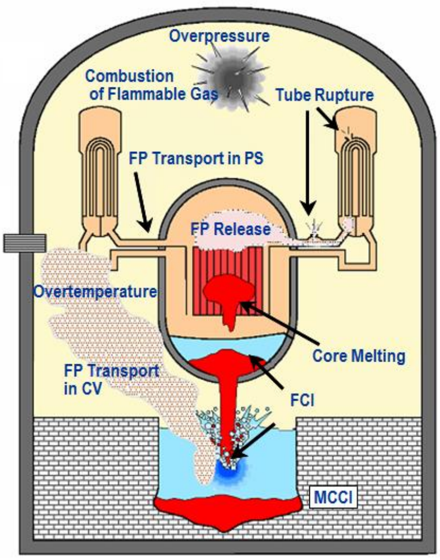

Figure 1. Conceptual Diagram Showing Impact on Level 2 PRA

Table 2. Key Findings for the Phenomenon

Phenomena	Revised and Added Information
In-Vessel Melt Behavior	Melt stratification and distribution of major species (OECD/MASCA)
Steam Explosion	Experimental information of FCI tests including KROTOS, FARO, TROI and PULiMS/SES
Molten Core Concrete Interaction	Experimental and analytical information of OECD/MCCI ‘melt spreading’ melt jet break up in water and coolability of particulate debris bed

2.4. Analysis of Accident Sequences

In order to construct a containment event tree that classifies the accident progression in a tree figure, it is necessary to analyse the accident sequence for each plant damage state. The following process is required:

(a) Characteristic analysis of accident sequences

The important physicochemical phenomena that occur during the development of the accident, from core damage to containment failure, are analysed in relation to the plant conditions of the accident progression, and their effects on the accident progression are analysed. Possible mitigation measures related to accident mitigation and containment failure prevention shall be analysed for accessibility to equipment (including accessibility to sites of use in the case of portable equipment), transport of related materials and equipment, and workability in the field, taking into account the impact of the tsunami. Recovery, including repair of equipment and systems, may also be included in measures to prevent containment failure.

These countermeasures for mitigation and prevention of containment failure shall be analysed considering the following items. When considering mitigation measures after loss of containment function in source term analysis, the effects of combustion or detonation of combustible gas outside the containment should be considered.

- Operating procedures: Mitigation equipment related to containment failure and accident mitigation operations by operators shall be consistent with the operating procedures.
- Thermal-Hydraulic Atmospheric Conditions: The conditions under which continuous operation of equipment and systems and accident mitigation operations are possible are analysed from the thermal-hydraulic atmospheric conditions such as temperature, pressure, and water level in the containment vessel during a severe accident.
- Radioactive atmosphere conditions: From the radioactive atmosphere conditions at the time of a severe accident, the conditions under which the operator can perform accident mitigation operations and the accessibility of facilities credited as mitigation functions for field personnel are analysed.
- Monitoring conditions: To analyse the conditions for maintaining the functions of instrumentation facilities such as water level gauges and thermometers during severe accidents. In addition, the influence on the relaxation operation when the function of the instrumentation facility is not appropriate is analysed.
- Severe accident countermeasures equipment: When the equipment prepared as severe accident countermeasures and operator actions are credited as mitigation measures, the conditions under which they perform their functions are analysed.
- Alternatives: If alternatives are credited as mitigation measures, the conditions under which they perform their functions are analysed.

(b) Creating a containment event tree

Based on the results of the accident sequence feature analysis, a containment event tree is created for each plant damage state by the following process.

- Selection of heading items of the containment event tree: The heading of the containment event tree is selected from the results of the characteristic analysis of the accident sequence, such as the mitigation operation of the accident by the operator and the occurrence of physicochemical phenomena.
- Containment event tree creation: For each plant damage state, a containment event tree is created by placing the heading of the containment event tree in such a way as to preserve the causal relationship between important physicochemical phenomena, mitigation equipment activation and deactivation, and operator actions, and connecting the branches in a tree figure.

(c) Analysis of dependency

The possibility of preventing the occurrence and spread of physicochemical phenomena and the dependency of mitigation measures are analysed for all accident sequences classified in the containment event tree. If mitigation measures for core damage prevention are used again for heading mitigation measures for post-core damage events, they should be consistent with the recovery conditions of mitigation measures for core damage prevention and analysed from the operating environment such as thermal-hydraulic conditions after core damage. The following items should be appropriately reflected in the severe accident phenomenon and the dependency of mitigation measures for the event:

- Dependent heading placement: The order of causally related headings is arranged appropriately, including the correspondence between physicochemical phenomena and mitigation measures during a severe accident, and the dependencies between the headings in the containment event tree are arranged, and the headings dependent on a particular heading are placed downstream.
- Dependency when event trees are joined: When multiple event trees are combined to form a containment event tree, the dependency relationships between the headings of each event tree are aligned.

(d) Setting the containment failure mode

For all accident sequences classified in the containment event tree, the containment failure mode is set to determine the final state of the containment event tree.

Since the events to be analysed depend on the progress of the accident, it is effective to examine the progress of the accident by dividing it by each phase. For example, at the time of reactor (pressure) vessel failure, there is a possibility of a high-pressure melt ejection event, and after reactor (pressure) vessel failure, there is a possibility that the molten core is released onto the containment floor and molten core concrete interaction occurs. For this reason, it is effective to set the accident progress phase at the boundary of reactor (pressure) vessel failure as follows.

-T1: Early in the accident
-T2: Immediately after reactor (pressure) vessel failure
-T3: Late in the accident

A containment event tree can also be constructed for each accident development phase. In this case, the containment event trees for each accident progression phase are causally related to each other, and the dependency should be thoroughly examined. Table 3 shows an example of the heading of the containment event tree for each accident progression phase.

2.5. Accident Progression Analysis

It is necessary to carry out accident progression analysis for data necessary for setting the branching probability of headings of containment event tree and for verification of headings. The following process is required:

Table 3. Containment Event Tree Headings (PWR Example)

Accident Progression Phase	Headings on Severe Accident Phenomena, States of Facilities, Human Actions
T_1: Early up to RPV breach	・ Core cooling ・ Containment isolation failure ・ RCS depressurization ・ Containment bypass (including SGTR, Induced-SGTR, IS-LOCA) ・ Recovery of reactor water level ・ In-vessel steam explosion ・ Hydrogen control ・ Hydrogen combustion ・ Containment failure (due to hydrogen combustion) ・ RV failure
T_2: Just after RPV breach	・ Dispersion of core melt ・ Water presence in the cavity ・ DCH ・ Hydrogen combustion ・ Containment failure (due to hydrogen combustion, direct melt attack, DCH, ex-vessel steam explosion)
T_3: Late	・ Air space cooling in containment ・ Recovery of safety facilities ・ Water heat sink in containment ・ Hydrogen combustion ・ Containment failure (due to hydrogen combustion, Over-pressure) ・ MCCI ・ Containment failure (due to Over-temperature)

(a) Selection of accident sequences for analysis

As the accident sequence to be subjected to the accident progression analysis, an accident sequence representative of the containment failure mode is selected for each plant damage state according to the following conditions from the accident sequences classified in the containment event tree.

- Operation allowable time: The accident sequence is selected from the combination of mitigation operations in which the allowable time of operation is the most severe among the mitigation operations of the accident.
- Frequency: Select an accident sequence with a high frequency of core damage.

(b) Setting accident sequence analysis conditions

The contents of heading items such as physicochemical phenomena, operation of equipment and systems, and mitigation operations during a severe accident included in the accident sequence classified in the containment event tree are considered in the analysis. If the damage to SSCs resulting from the accident progression is credited to have a beneficial effect on the mitigation function, its technical validity should be clarified.

The heading combinations in the containment event tree for each accident sequence are simulated. Furthermore, the following analysis conditions are set.

- Plant facility conditions: Based on the results of the investigation of plant configurations and characteristics, the analysis conditions of plant facilities are set from the core, the shape of the reactor (pressure) vessel, the configuration of the mitigation facilities, the capacity of the facilities, and the operation logic, etc.
- Accident sequence analysis conditions: Set the analysis conditions for the accident sequence to be analyzed according to the success or failure of the mitigative operation in the operating procedure at the time of the accident.

(c) Analysis of accident progression

Analytical codes that have been applied to actual plant scale analysis using analytical models validated under severe accident conditions are used. For each accident sequence, the progress of the accident specific to the accident sequence is clarified by analyzing the thermal hydraulic behavior of the plant. The analysis should include:

- Thermal-hydraulic behavior in reactor cooling system: Thermal-hydraulic behavior including water level, temperature and pressure in core and reactor cooling system, core heat up, core melting behavior, heat generation due to metal-water reaction, and generation behavior of combustible gas, etc. during severe accident are analyzed.
- Thermal-hydraulic behavior inside and outside the containment vessel: Thermal-hydraulic behavior including water level, temperature and pressure inside the containment vessel during severe accident, generation behavior of non-condensable gas due to molten core concrete interaction, and generation behavior of combustible gas, etc. during severe accident are analyzed. In addition, flammable gases may remain in reactor buildings due to leakage from containment vessels. When a building is damaged, the deposition effect of radioactive materials in the building and the availability of mitigation measures (for example, water cannons) for reducing the amount of radioactive materials released are affected. When these are credited, the migration of combustible gases outside the containment vessel is analyzed.
- Timing of event: The timing of core damage, reactor (pressure) vessel failure and containment failure shall be analyzed. In addition, the effect of deposition of radioactive materials in the building and the time of failure of the building are analyzed when mitigation measures are credited to reduce the radioactive materials released.
- Atmosphere at the time of the event: The water level, temperature and pressure of the atmosphere, gas components, debris temperature and composition at the time of the event are analyzed.

(d) Analysis of occurrence and load of physicochemical phenomena during severe accidents

To analyze the occurrence of physicochemical phenomena during a severe accident affecting a radioactive material confinement function and/or the load of a containment vessel associated with the occurrence.

The criteria for determining the integrity of the containment structure are compared with the analytical results, and the presence of containment failure and the subsequent effect on the accident development are analyzed.

(e) Validating the headings of the containment event tree

For the headings of the containment event tree, check whether the following causality is appropriate from the analysis results of the accident progression.

- Causal relationship between the occurrence of physicochemical phenomena and the conditions for their occurrence during the severe accident
- Causal relationship between the occurrence of physicochemical phenomena during a severe accident and mitigation operations for the accident

2.6. Setting the Branch Probabilities of the Containment Event Tree

The conditions for obtaining the branch probabilities of the containment event tree are set by reflecting the optimal prediction based on the latest severe accident technical knowledge and the accident progression analysis results. Based on the criteria for determining the integrity of the containment structure by identifying the load of the containment vessel and evaluating the integrity of the containment structure, analyzing the characteristics of the accident sequence and arranging the dependencies, and the results of accident progression analysis for each accident sequence and the load of the containment vessel due to physicochemical phenomena, the conditions for evaluating the average value and uncertainty distribution of each branch probability of the containment event tree including equipment and systems, human error, and physicochemical phenomena are set. When a point estimation analysis is performed, conditions for evaluating the average value of the branch probabilities are set.

(a) Setting the probability distribution for a branch of the containment event tree

The probability distributions of failure probabilities and human error probabilities of equipment and systems related to mitigation measures, and bifurcation probabilities of physicochemical phenomena are set as follows.

- Equipment and systems: The average value and uncertainty distribution of parameters such as equipment failure rate and the average value and uncertainty distribution of damage probability due to tsunami are evaluated from the average value and uncertainty distribution of system reliability analysis methods such as fault tree, and the conditions of failure probability are set.
- Operation: Set the conditions for the human error probability from the mean value and uncertainty distribution of the human error probability by the human reliability analysis. Set the conditions for operation failure probability, including equipment and system recovery operations, from the following analysis:

    - Time allowance for equipment that is credited to operate based on the timing of the event according to the accident progression analysis and the time required for equipment operation
    - Stressors for operators due to atmospheric conditions at the time of the accident and operability after the tsunami

    Conditions for significant human error probabilities shall be established based on one of the following methods or a combination thereof, taking into account the accident progression after the occurrence of the tsunami and the effects of the tsunami.

    - Field studies or interviews conducted jointly with plant staff or training instructors
    - Simulator check
    - Plant specific thermal-hydraulic analysis results

- Physicochemical phenomena: Physicochemical constraints such as thermal-hydraulic conditions, conditions of occurrence of phenomena, and physical properties obtained from accident evolution analysis results for each accident sequence, or domestic and foreign experimental results and their analysis results are used to compare the load of the containment vessel caused by the phenomena with the criteria for determining the integrity of the containment structure, and to analyze the load of the containment vessel and the uncertainty of the containment fragility corresponding to the failure mode. The probability distribution is established by examining the average value and the uncertainty distribution of the probability that the containment vessel will lose its function due to the phenomena. For phenomena with large uncertainties, the governing factors of the phenomena, their uncertainty distribution, and the causal relationship between the governing factors of the phenomena and the structural integrity of the containment vessel are clarified, and the probability distribution is set using the decomposition event tree method or the ROAAM method.
- Engineering judgement: When analyzing the probability of occurrence of an event, if the relevant information is scarce and it is difficult to set conditions for the branch probabilities from analytical methods, the branch probabilities of the containment event tree may be set by engineering judgement. When engineering judgment is used, the basis for setting is clarified from the results of accident progression analysis.

(b) Branch dependency

In the branch with dependency from the analysis of the characteristics of the accident sequence and the arrangement of the dependency, the average value of the branch probability and the conditions of the uncertainty distribution are set from the following process.

- Dependencies in the containment event tree: If there is a dependency between branches of the containment event tree, set the branch probability condition to match the dependency condition.

- Dependence with core damage prevention measures: If there is a dependency between core damage prevention measures and mitigation measures for post-core damage accidents, set conditions consistent with the success or failure conditions assumed in core damage prevention measures.

2.7. Quantification of the Containment Failure Frequencies

Point estimation evaluation and uncertainty analysis of the frequency of the containment failure mode and/or the containment failure frequency are carried out based on the created containment event tree from the frequency of the plant damage state and the branch probabilities for each branch of the containment event tree. Importance and sensitivity analyses should identify factors that influence and/or dominate the evaluation results, and validation should confirm the validity of the analysis results.

(a) Point estimation evaluation

Using the average values of the frequency of the plant damage state and the branch probabilities of the containment event tree, the frequency of the following containment failure modes and/or the containment failure frequency are evaluated.

- Frequencies of containment failure modes
- Frequencies of containment failure in plant damage states
- Total containment failure frequency

The frequency CFF_i of the accident sequence i leading to the containment failure is determined from the conditional probability Q_i (h, a) of the accident sequence i leading to the containment failure with respect to the frequency h (h) at the tsunami height h and the ground motion intensity a at the tsunami height h, which are determined from the tsunami hazard curve, by the following equation:

$$CFF_i = \int_{h_{min}}^{h_{max}} \int_{a_{min}}^{a_{max}} h(h) \cdot P(h,a) \cdot Q_i(h,a) da dh \quad (1)$$

where P (h, a) is the conditional probability density function (joint probability) of ground motion intensity a at tsunami height h. The frequency h (h) at the tsunami height h is obtained from the following equation based on the tsunami hazard curve H (h) (annual excess frequency of tsunami exceeding the tsunami height h).

$$h(h) = -\frac{dH(h)}{dh} \quad (2)$$

The upper limit of integration in Equation (1), that is, the upper limit h_{max} of the considered tsunami height, sets the tsunami height that does not significantly affect the containment failure frequency. The lower bound h_{min} of the integral in Equation (1) is the tsunami height at which the effect of the tsunami on the reactor facility is negligible.

The containment failure frequency CFF_{total} is determined by the sum of the frequencies CFF_i of all accident sequences i leading to the containment failure, i.e., the following equation:

$$CFF_{total} = \sum_i CFF_i \quad (3)$$

(b) Uncertainty analysis

Taking the probability distribution of the frequency of the plant damage state and the probability distribution of the branch in the containment event tree as input, random sampling by Monte Carlo method or equivalent uncertainty propagation analysis technique is used to evaluate the frequency of the containment failure mode and the average value and uncertainty distribution of the containment failure frequency. Since the number of samples in the Monte Carlo method depends on the number of parameters to be varied and the probability distribution thereof, a number that allows the sampled values to appropriately represent the uncertainty distribution given as input is selected.

The propagation of uncertainty is analyzed for uncertainty factors included in tsunami hazard, realistic bearing capacity of SSCs, realistic response, etc., and the distribution of containment failure frequency and parameters representing the distribution (mean, median, 5% confidence, 95% confidence, etc.) are obtained.

(c) Importance analysis

When identifying factors that dominate the frequency of containment failure modes or the containment failure frequency, an important analysis should be performed using the following methods:

- Study of the factors to be analyzed: The target of the importance analysis is selected from the accident sequence that results in the preceding failure of the containment vessel, the equipment failure and the mitigation operation of the accident in the containment event tree, and the bifurcation of physicochemical phenomena.
- Importance analysis: The Fussell-Vesely importance and the risk achievement worth are analyzed from the branch probability or the conditions for obtaining the branch probability of the selected branch.

(d) Sensitivity analysis

Based on the assumptions and conditions that may have a significant impact on the results of the accident development analysis, the following cases should be included in the sensitivity analysis. For the extracted assumptions and analysis conditions, the parameters and parameter values to be included in the sensitivity analysis are set from the analysis results of the accident progression, and the effects on the results are grasped by performing the sensitivity analysis.

- Occurrence of physicochemical phenomena: If the occurrence or non-occurrence of a specific physicochemical phenomenon is likely to change in the results of the accident progression analysis due to the assumptions and conditions of the analysis, the phenomenon shall be subjected to a sensitivity analysis, and the occurrence or non-occurrence of the phenomenon shall be examined by conducting an analysis against the limit values of physicochemical values of the parameters involved in the occurrence of the phenomenon. In addition, cases where the load of the analyzed containment vessel is close to the criteria for determining the integrity of the containment vessel structure are also considered.

  - Example 1: In the case where the analysis result of the accident progression is the atmospheric condition where the occurrence of steam explosion, direct containment heating, hydrogen combustion, etc. may change.
  - Example 2: If the pressure or temperature in the analyzed containment is close to the criteria for determining the integrity of the containment structure.

- Accident sequence: If the accident progression changes or transitions to a different accident sequence under the assumptions and conditions of the analysis, such as the mode and timing of the loss of function of the equipment or system, the subject of the sensitivity analysis shall be examined to determine the effect of the change in the assumptions or conditions of the analysis on the timing of events in the process of the accident progression or the possibility that the target accident sequence transitions to another accident sequence.

  - Example: In the case where the condition of loss of function of the equipment or system such as opening and fixing of the relief safety valve, or the time of occurrence of such loss of function is treated as an assumption or condition of the analysis.

- Tsunami PRA: To examine the sensitivity of different assumptions, model selection, data selection, etc. to how different conditions affect assessment results such as containment failure frequency.

(e) Assumptions and validation of containment event tree quantification

Review the following items to confirm the validity of the containment event tree quantification assumptions and results:

- Analysis results: The total frequency of plant damage states and the total frequency of all containment failure modes are compared to confirm that the analysis results of the frequency of containment failure modes are valid.
- Important model assumptions: If the bifurcation probability set by the engineering judgment dominates the results, the validity of the engineering judgment is confirmed from the accident progression analysis results.

2.8. Classification of Release Categories and Quantification of frequencies

All accident sequences are classified into release categories with similar release behavior of radioactive materials released into the environment, using the source term release path, failure location, failure size, and evaluation conditions related to mitigation of source term effects.

In the case of a tsunami event, an accident sequence corresponding to a containment failure mode in which the containment body is damaged by a tsunami or a containment failure mode in which the containment body is damaged dependently with damage to a reactor building etc. caused by a tsunami is classified into an independent release category.

In order to obtain the frequency of release categories, point estimation evaluation and uncertainty analysis are carried out using the conditions for obtaining the frequency of plant damage states and the branch probabilities of the containment event tree.

Sensitivity analyses are used to identify the impact of important assumptions and conditions on the results of the evaluation.

2.9. Source Term Analysis for Release Categories

For each classified release category, the accident sequence to be analyzed is selected, the source term analysis is performed, and the source term is evaluated.

Uncertainty analysis of the source term is carried out for each release category with large frequency of occurrence and large source term, and the average value and uncertainty distribution of the source term are evaluated.

Perform a sensitivity analysis of the source term with regard to assumptions and analysis conditions that may significantly affect the analysis results of the source term, and grasp their effects on the source term.

3. SUMMARY AND FUTURE PLANS

This paper introduces the main revisions of the Level 2 PRA standard established by the Atomic Energy Society of Japan, which has been extended from internal events and seismic events to tsunami events. In addition to the extension to tsunami events, LERF/LRF was reflected in the definitions and regulations by comparison with the U.S. NRC regulatory guides, ASME/ANS PRA standards, and Japanese regulations. It also required clarification as to whether containment venting should be included in the containment failure in light of the objectives of the level 2 PRA. In addition, the technical validity of beneficial failures, which have a beneficial effect on the mitigation function, should be clarified.

In the future, this Level 2 PRA standard will be revised to extend it to a shutdown state and to reflect the structure hierarchy.

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