# Level 2 PRA Methodology Development and Plant Applications

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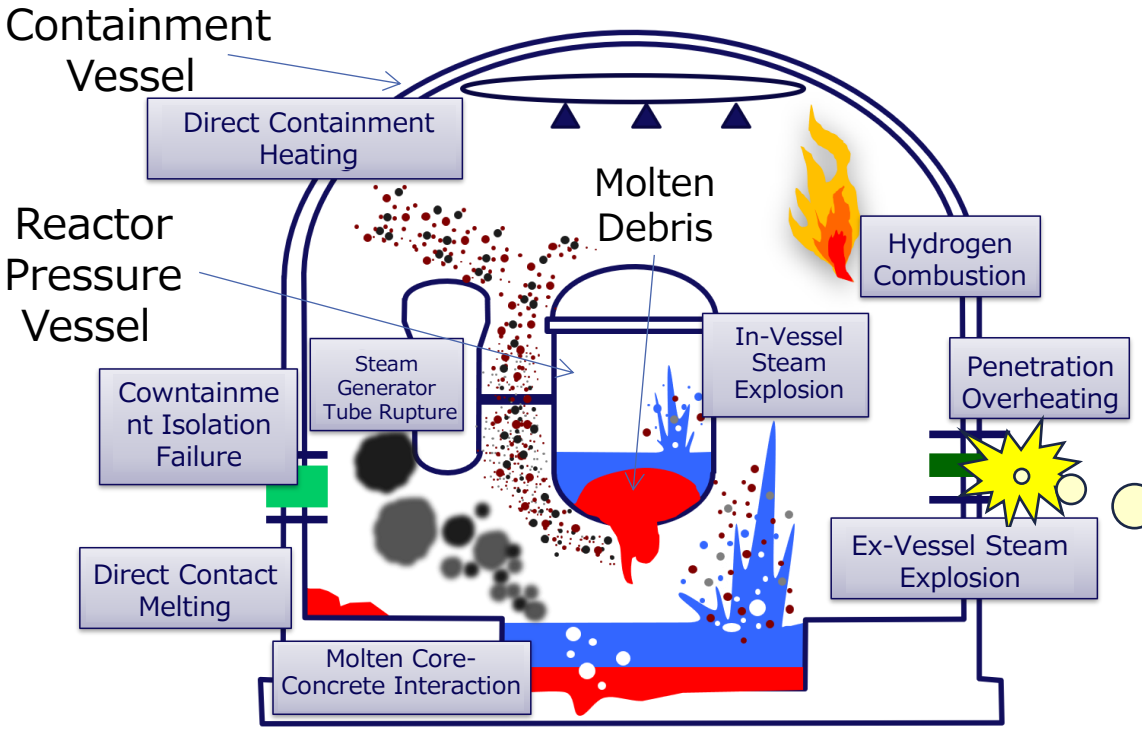
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# 1. Level 2 PRA Overview

- Level 2 PRA evaluates the frequency of accident scenarios leading to loss of function containing radioactive materials during accidents following core damage (containment failure frequency), and evaluates source term, including mass, type, and form of radioactive materials released into the environment.




Phenomena inside the containment

Containment failure due to loads such as high temperature, high pressure, and other loads.

Radioactive material release into the environment

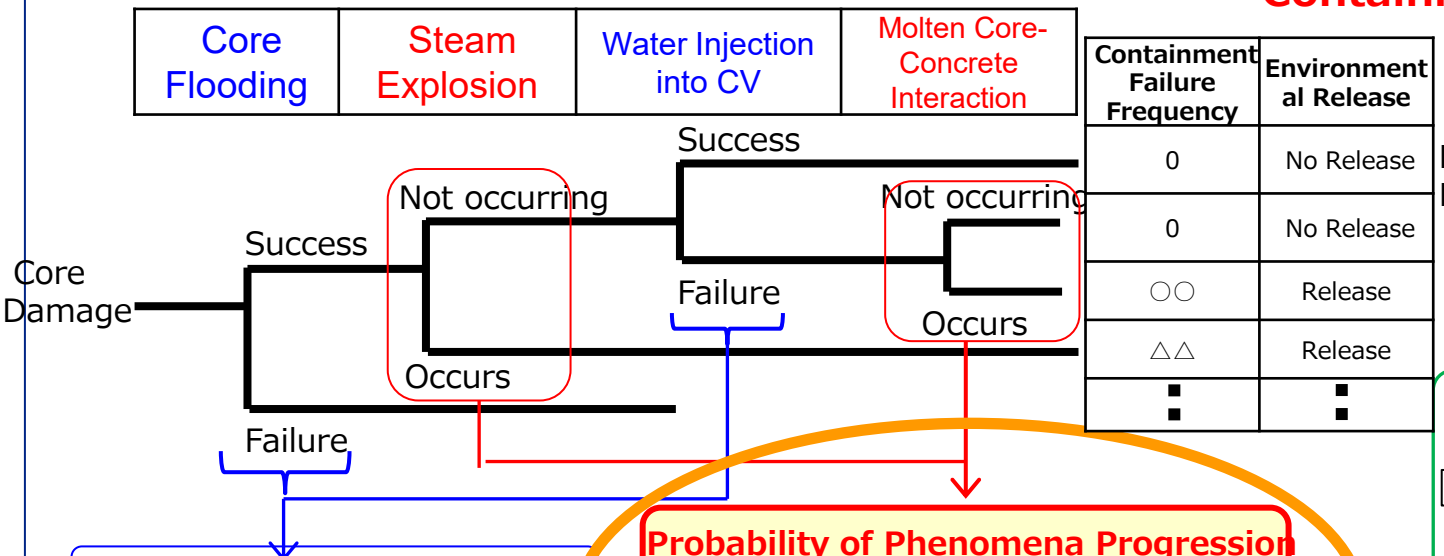
Severe accident is defined as being more severe than a design basis accident and involving significant core degradation.

# 2. Level 2 PRA Model Structure

- Event tree representing accident progression enables the evaluation of containment failure frequency.
- Accident progression analysis code*1 enables evaluation of source term.

*Computational program for analyzing accident progression and radioactive materials release at a nuclear power plant

## Containment Failure Frequency Evaluation (Containment Event Tree)

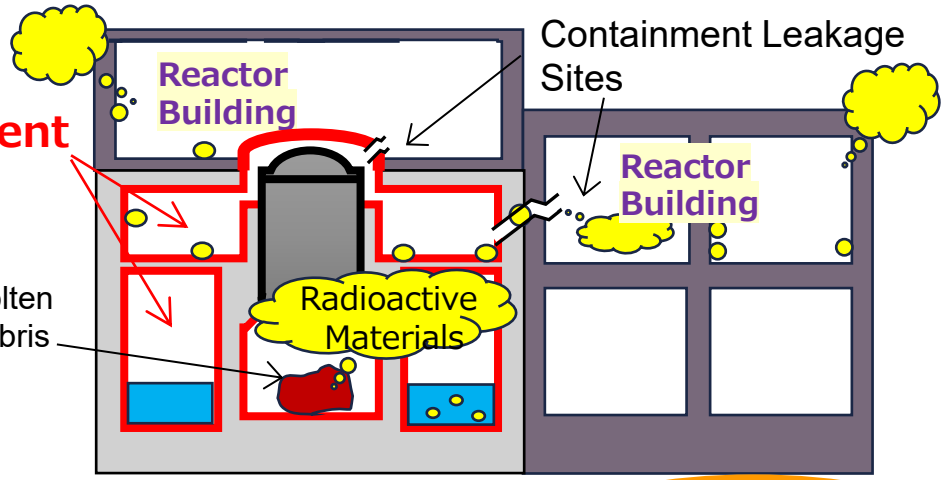



| Containment Failure Frequency | Environmental Release |
|---|---|
| 0 | No Release |
| 0 | No Release |
| ○○ | Release |
| △△ | Release |
| ⋮ | ⋮ |

**Success/Failure Probability of Mitigation Measures**

**Probability of Phenomena Progression**
- Boundary Conditions (Temperature, Pressure, Water Flow, etc.)
- Phenomenal Mechanism

## Source Term Evaluation




**Behavior of Radioactive Materials in the containment**
- Release from the molten debris
- Micro-particles behavior

**Release from Containment and Behavior in Reactor Building**
- Removal at Containment Leakage Sites
- Behavior in Reactor Building*2

*2 In some cases, radioactive materials may be released directly from the containment into the environment without passing through the building.

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# 3. Level 2 PRA Background & Gaps

## ■ Background

- TEPCO Fukushima Daiichi Nuclear Power Plant accident highlighted the importance of evaluating the risk of massive releases of radioactive cesium, which is the default for residents evacuating and zones specified as difficult-to-return. [1]
- Level 2 PRA identifies the magnitude and cause of risks related to radioactive cesium release.
- Yet, such evaluations involve many uncertainties, and simplified and conservative methods are currently applied.
- Atomic Energy Society of Japan's Level 2 PRA Standard [2] requires such uncertainties be evaluated using reasonable methods, therefore, it is necessary to develop methods capable of evaluating the best estimate values accounting for these uncertainties.

## ■ Gaps to Overcome

- Too simplistic or conservative ways to quantify the containment failure frequency and source term prevented to obtain the best estimate values with their uncertainties, required to accurately grasp where the risk was in the plant and its magnitude.

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# 4. Efforts to Overcome Gaps

■ **Simplistic or conservative ways of quantifying containment failure frequency and source term prevented obtaining the best estimate values accounting for such uncertainties so as to accurately grasp where risk were situated and their magnitude.**

- Establish a reasonable uncertainty assessment method for severe accident phenomena
- Incorporate containment fragility into containment failure frequency assessments

  ⇒ [Research Result (1)]
- Method established for evaluating behavior of particulate radioactive material in reactor building
- Method established for evaluating effects of removing particulate material at containment leak sites

  ⇒ [Research Result (2)]
- Benchmark developed methods ⇒ [Research Result (3)]

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# 5. Research Result (1) Incorporation of Containment Fragility into Containment Failure Frequency Evaluation

- Current Level 2 PRA assumes containment failure occurs when the containment temperature reaches 200°C or containment pressure reaches twice the design pressure[3] . Conservative values are applied relative to actual performance values.
- Applying a curve (containment fragility) (left figure) representing the probability of containment failure as a cumulative distribution function in relation to the magnitude of the load (such as temperature or pressure) results in a more realistic assessment compared to the conventional process, including taking into consideration the time margin for safety measures (graph at right).

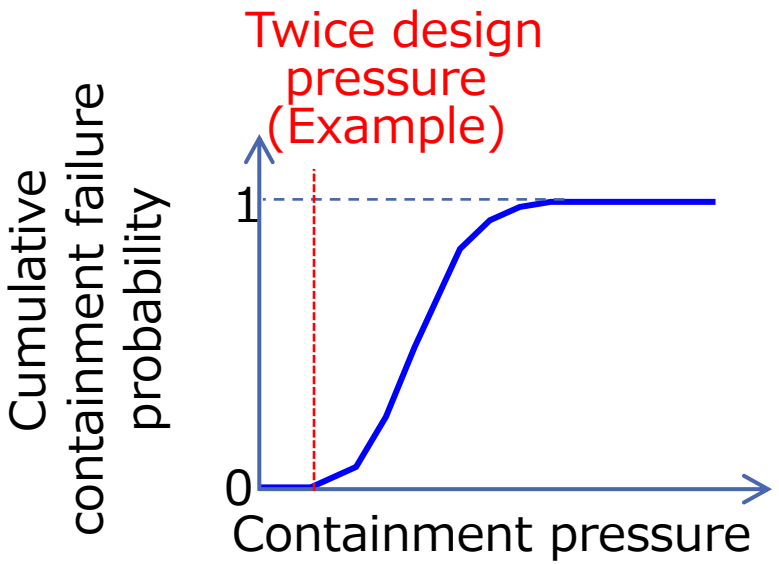
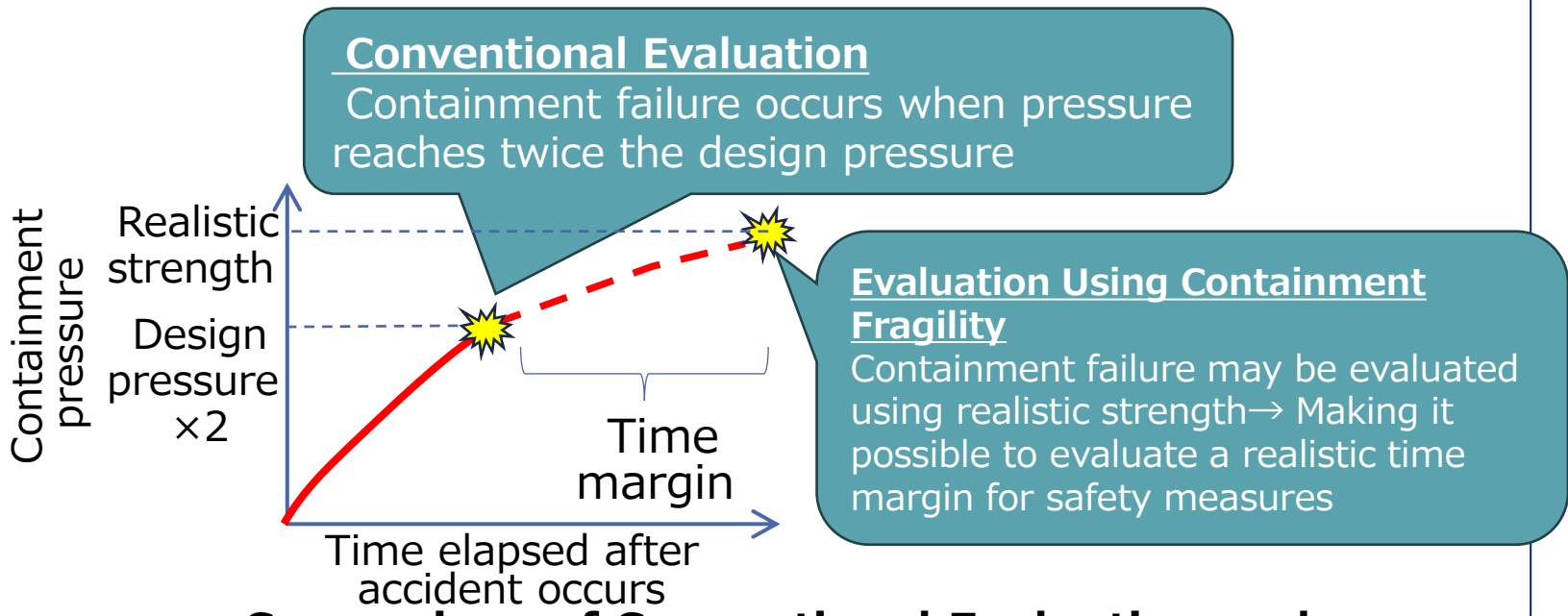


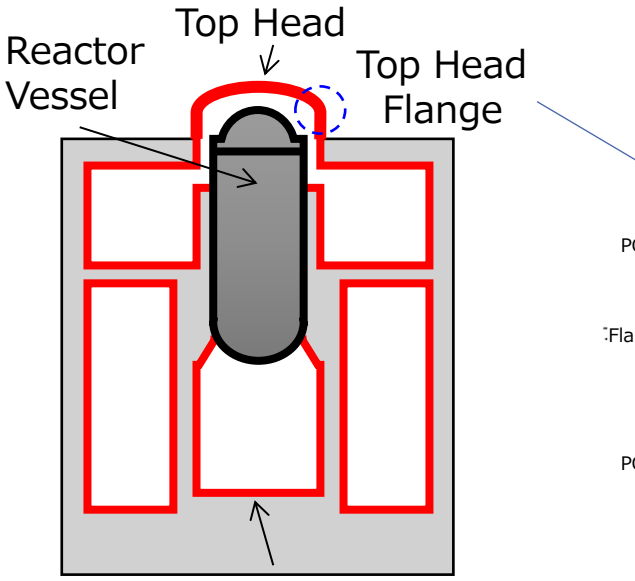

**Illustration of Containment Fragility**



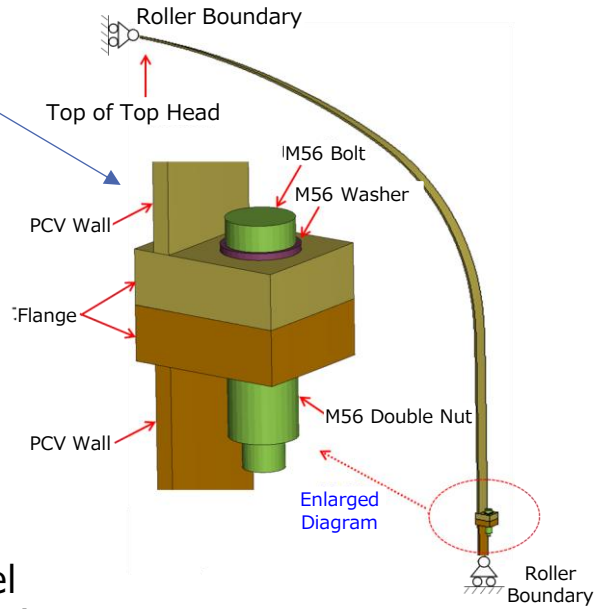

**Comparison of Conventional Evaluation and One Applying Containment Fragility**

# 5. Research Result (1) Incorporation of Containment Fragility into Containment Failure Frequency Evaluation

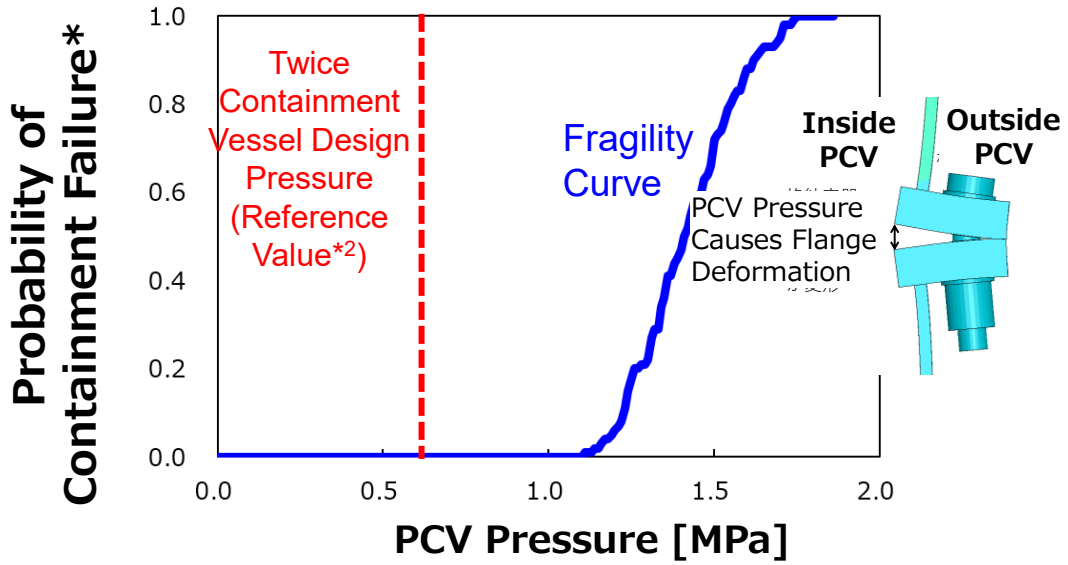
- Focusing on the top head of the containment (diagram at left) that is the site typically used when evaluating overpressure failure of BWR pressurized containment vessel (PCV), structural analyses were performed under typical scenario conditions to create a containment fragility curve (graph at right)[4] .
- **A fragility curve applicable to top heads of the same type has been constructed, and case studies that may be applied to creation of fragility curves for different leak sites have been also developed.**




**Schematic Diagram of ABWR Pressurized Containment Vessel**




**Flange section of top head for structural analysis[4]**



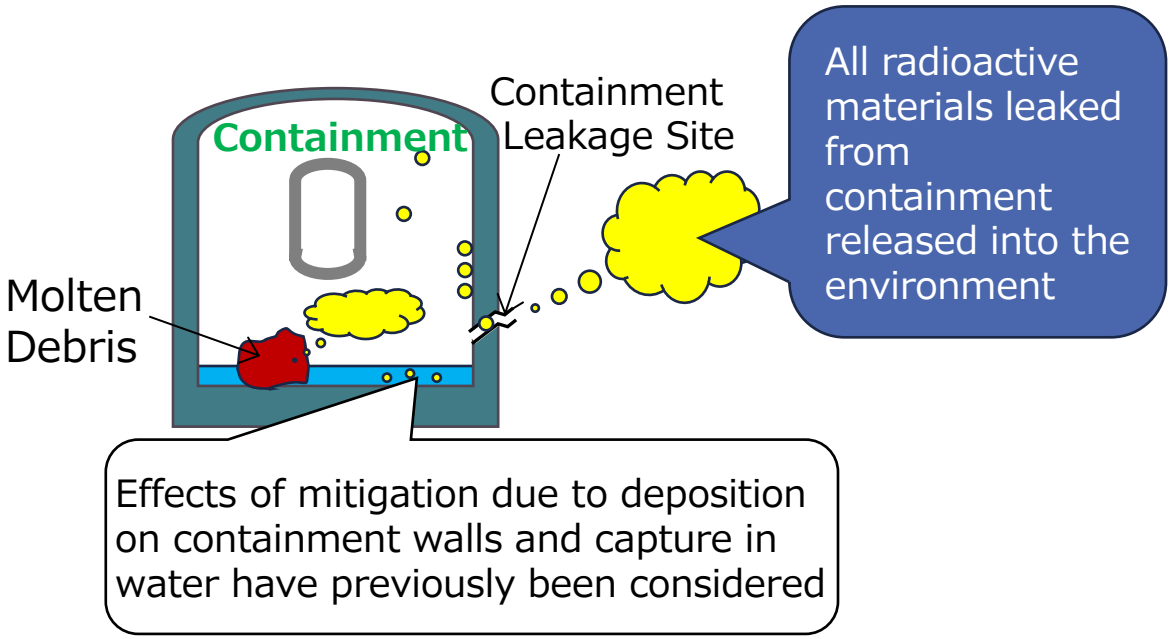

**Probability of Containment Failure vis-à-vis PCV pressure[4]**

*1 Containment failure assumed to occur when flange deformation reaches 5 mm pressure (as shown in the illustration in the figure) due to an increase in PCV pressure.

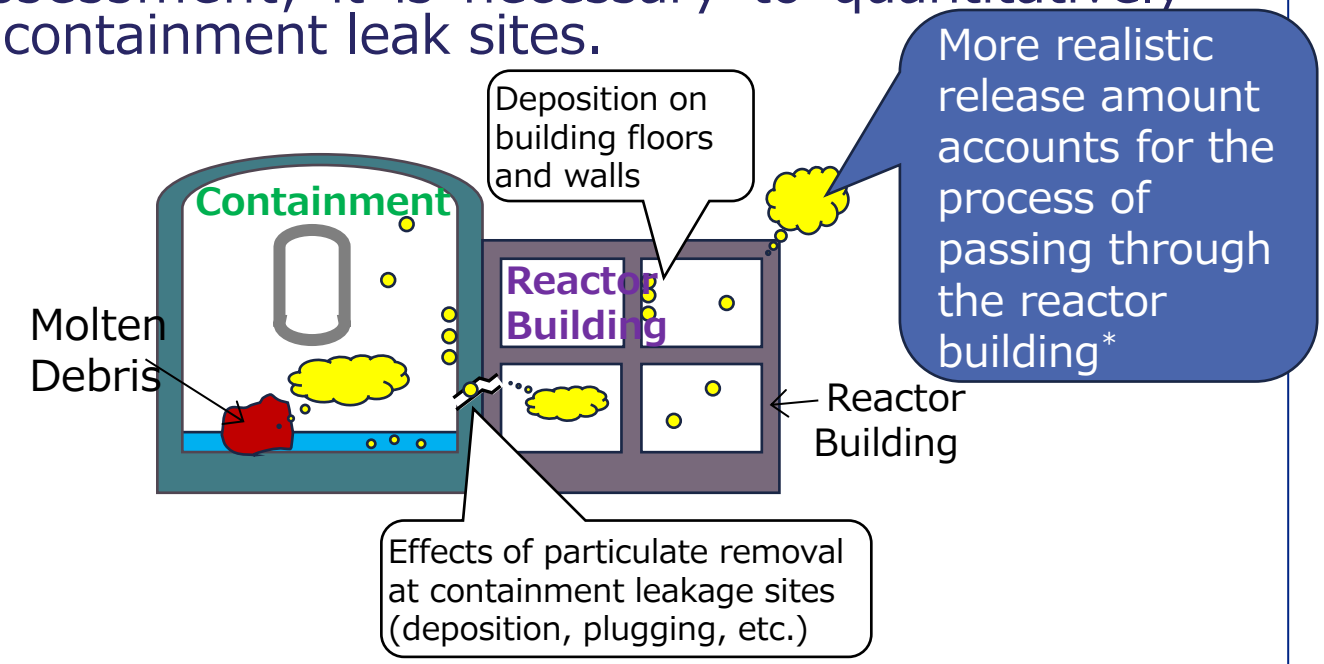
*2 0.62 MPa gauge: Reference value of ABWR

# 5. Research Result (2) Establishment of Evaluation Method Removing Particle Materials in Containment Leak

- Conventional source term analyses have been used for evaluations assuming all radioactive material released from the containment is released into the environment.
- However, considering how radioactive cesium behaves as particulate material, it has been reported that significant deposition and plugging of narrow leakage pathways in the containment have reduced the amount released into the environment[3] .
- To achieve a realistic source term assessment, it is necessary to quantitatively evaluate particulate materials removal at containment leak sites.




**Source Term Evaluation Using Conventional Methods**



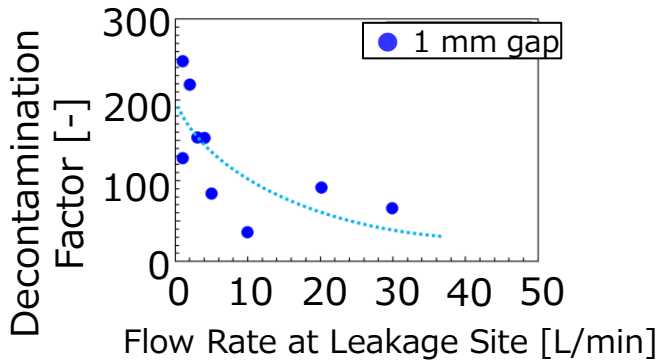

**Realistic Source Term Evaluation Sought**

* In some cases, radioactive materials may be released from the containment directly into the environment without passing through the building.

# 5. Research Result (2) Establishment of Evaluation Method Removing Particle Materials in Containment Leak

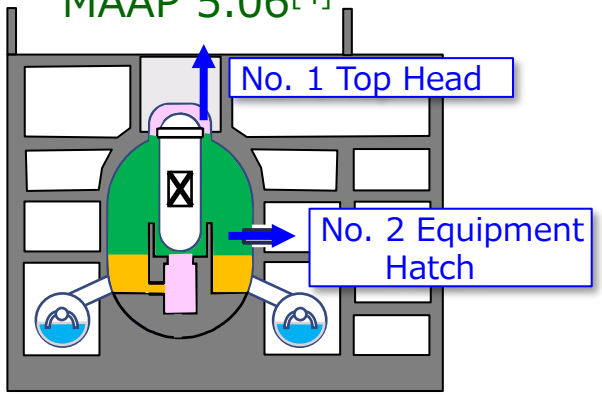
- Experiments were conducted to study the effect of cesium removal at potential containment leak sites, such as equipment hatches. A correlation for evaluating the decontamination factor*1, which indicates cesium removal performance, was developed[4][5] .
- **Analyses were developed incorporating the decontamination factor into accident progression analysis code MAAP. A source term analysis method was developed in accordance with release pathways during an accident.**

Experiment Measuring Decontamination Factor Correlation (Target: flange-type leakage sites)



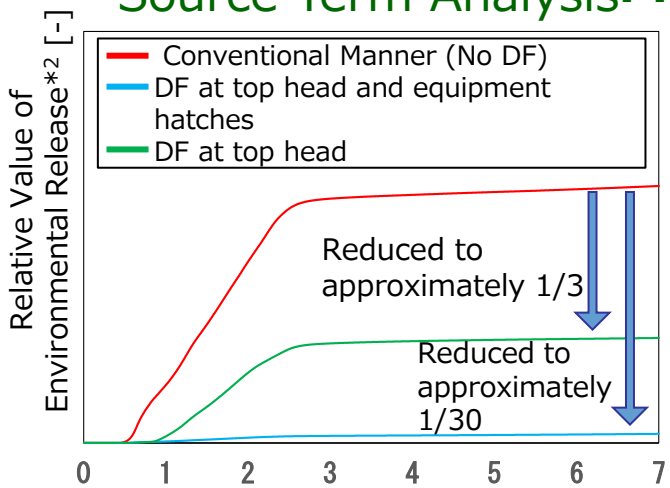

Decontamination factor varies with flow velocity

Experimental Results Incorporated into MAAP 5.06[4]




Sites Leaking from PCV

Source Term Analysis[4]



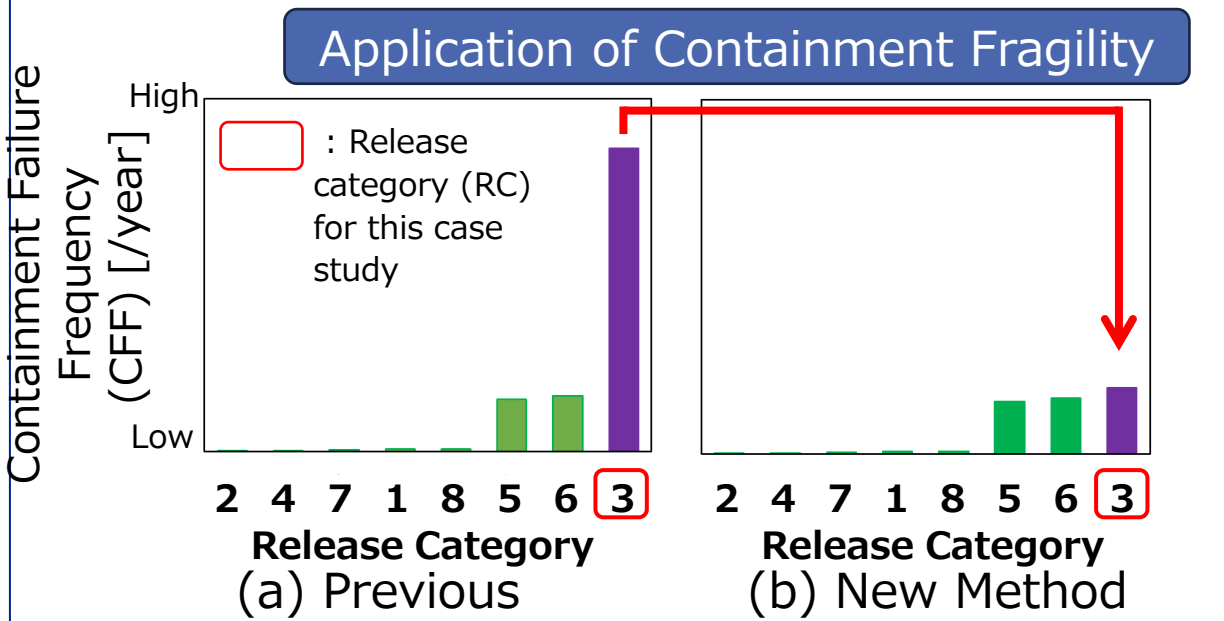

*2 Ratio of amount of Cesium released into the environment in relation to amount of Cesium contained in reactor core just after reactor shutdown.

*1 Ratio of cesium concentration flowing into containment leakage sites from the containment. Higher decontamination factor indicates that more cesium is removed at the containment leakage site.

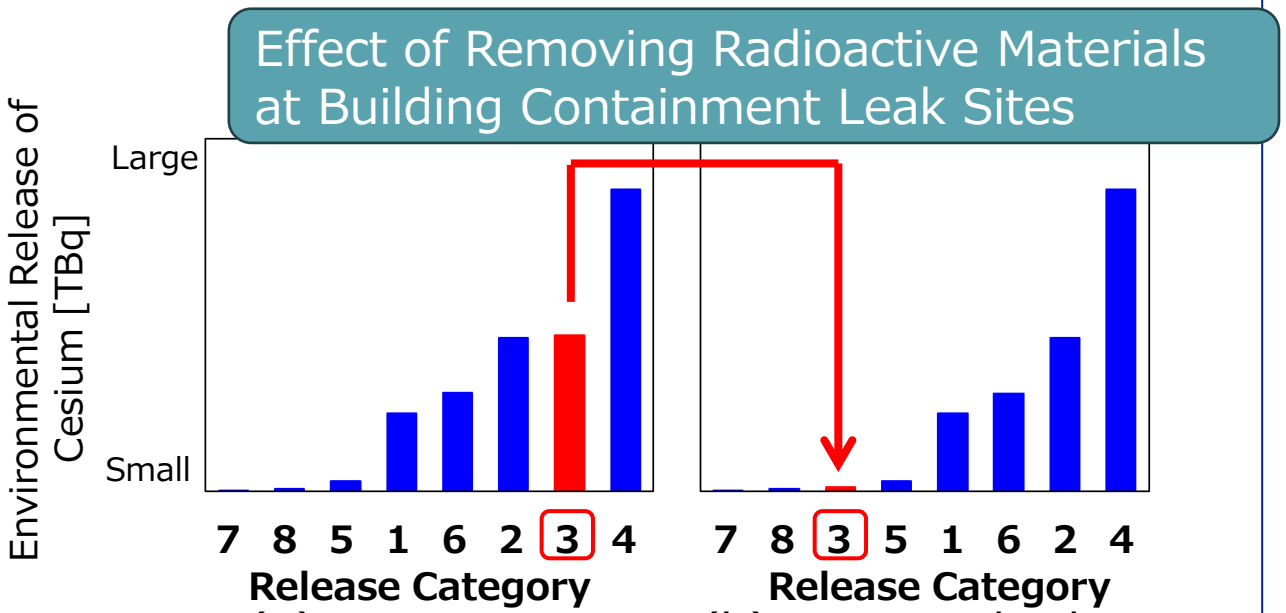
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# 5. Research Result (3) Benchmarking Developed Methods

- New methods were applied to Level 2 PRA of a model plant[6] and the process re-evaluated.
- Re-evaluation of the release categories* where the containment failure frequency and amount of cesium released into the environment are relatively high showed risk values varied significantly from the previous process.
- Results indicated the previous methods were very conservative and applying the research methods resulted in a reasonable assessment.




**Results for Containment Failure Frequency[4]**




**Results for Cesium Released into the Environment[4]**

*Groups of accident scenarios evaluated in Level 2 PRA classified according to release characteristics of radioactive materials.

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# 6. Current Achievements & Future Actions

## ■ Current Achievements

- Evaluation methods developed for containment fragility and for effect of removing particulate materials at containment leakage sites, which have significant impact on results, to achieve Level 2 PRA with high plausibility for RIDM.
- New methods applied to the Level 2 PRA model plant evaluation.
- Results showed the new methods largely improve conservativeness in the conventional outcome and enable evaluation of best estimate Level 2 PRA.

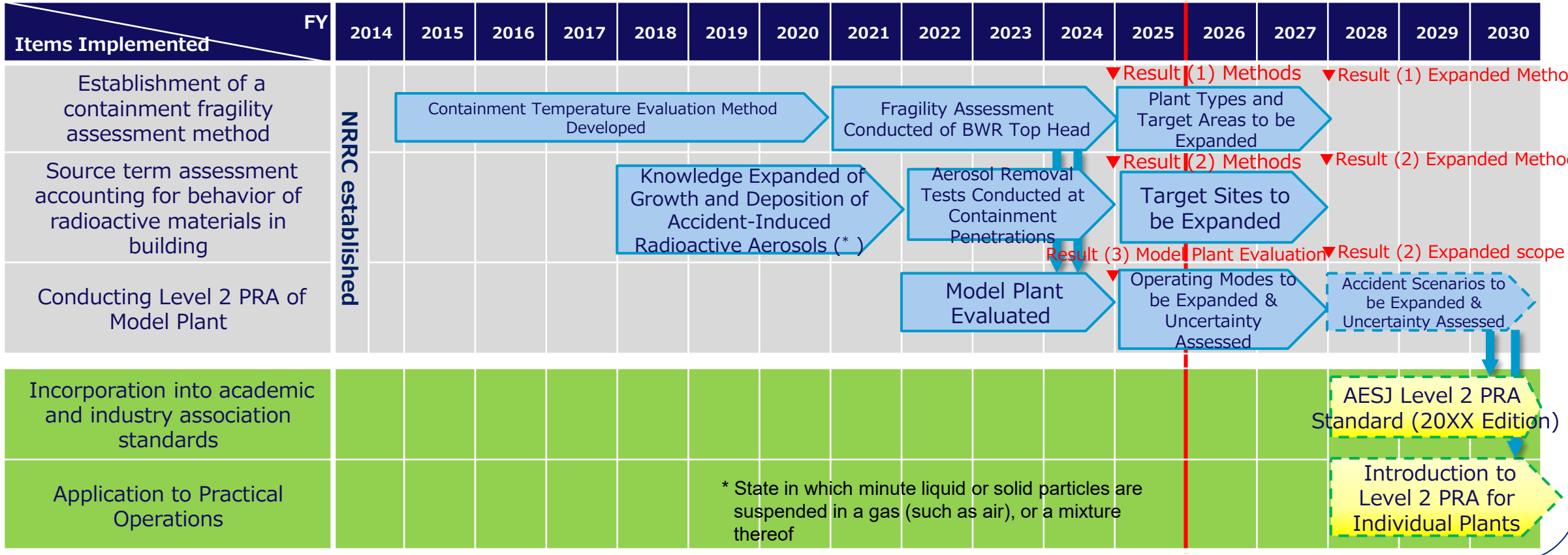
## ■ Future Actions

- Expand scope of application (operating modes, accident scenarios) to enable Japanese utilities to utilize the methods developed in this study.
- Furthermore, work on the following remaining tasks below to realize rational Level 2 PRA,
  - Establish more plausible methods to quantify distribution range of source terms.
  - Expand knowledge of severe accident phenomena, such as molten core-concrete interaction and hydrogen detonations, and the deposition of radioactive materials in piping while developing methods of analyzing uncertainty.

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# [Appendix] Progress of the Updates for Overcoming Gaps

- In the rational assessment of containment failure frequency and source terms, we established methods for containment fragility assessment and the evaluation of radioactive material behavior in the building, both of which have a significant impact on the results, and have identified a path toward achieving a rational Level 2 PRA.



# [Appendix] Status of Initiatives

| Gaps to Overcome | Description | Progress |
|---|---|---|
| Too simplistic or conservative methods to quantify containment failure frequency and source term prevented best estimate values from being obtained with their uncertainties as required to accurately grasp where risk present in the plant and such magnitude. | Establish methods for evaluating probability of severe accident phenomena | Implemented by 2027 |
| | Establish methods for evaluating containment fragility | Implemented by 2027 |
| | Upgrade methods of analyzing accident progression utilizing AI and machine learning | Launched in 2025 |
| | Model plant evaluation for Level 2 PRA at-power | Implemented by 2030 |
| | Model plant evaluation for Level 2 PRA at shutdown | Implemented by 2030 |
| | Establish methods for evaluating behavior of aerosolized radioactive materials in buildings | Implemented by 2025 |
| | Establish methods for evaluating pool scrubbing removing radioactive materials | Implemented by 2027 |
| | Establish methods for evaluating the aerosol removal effect at the containment leak parts | Implemented by 2027 |
| | Establish methods for evaluating effect of removing aerosol in piping | Launched in 2025 |
| | Model plant evaluation for Level 2 PRA at-power | Implemented by 2030 |
| | Model plant evaluation for Level 2 PRA at shutdown | Implemented by 2030 |

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# [Appendix] References & Sources

[1] Accident Investigation and Verification Committee for the Tokyo Electric Power Company Fukushima Nuclear Power Plant, “Report of the Diet Accident Investigation Committee,” September 12, Heisei 24

[2] Atomic Energy Society of Japan Standard: Implementation Guidelines for Probabilistic Risk Assessment of at-power Nuclear Power Plants (Level 2 PRA Edition): 2022 (AESJ-SC-RK012:2022)

[3] Nuclear Power Engineering Organization, “Summary Report on Safety Assessment of Critical Structures (Reactor Containment Reliability Demonstration Project),” March Heisei 15

[4] Nakamura, Watanabe, Satake, Nishi, et al., “Advancements in Level 2 PRA during Power Operation (Series of 4 Presentations),” 1I09–1I12, The Atomic Energy Society of Japan 2025 Fall Conference, September 10–12, 2025, Kitakyushu International Conference Center

[5] K. Nakamura, “The Development of Analytical Models for Realistic Evaluation of Fission Product Release and Establishment of the Foundation to Implement Level 2 PRA,” ICONE29, Shenzhen, China, and Virtual, August 8–12, 2022

[6] CRIEPI Report NR25010 “Development of Evaluation Techniques for a Consistent Level 1 and Level 2 Tsunami PRA”

[7] CRIEPI Report NR24012 “Development of a Process for Evaluating the Effectiveness of Water Injection Scrubbing in Reducing Fission Products Following Steam Generator Heat Transfer Tube Rupture—Implementation of an SG Scrubbing Model Using API in MAAP”

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