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MITSUBISHI HEAVY INDUSTRIES GROUP

# ACTIVITIES OF THE COMMITTEE ON PRACTICAL APPLICATION OF PFM

## - PART 1: SENSITIVITY ANALYSIS AND ANALYSES FOR EFFECTS OF SURVEILLANCE DATA ON FAILURE FREQUENCY OF REACTOR PRESSURE VESSEL WITH ANALYSIS CONDITIONS OF A JAPANESE ACTUAL PWR PLANT -

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# Contents

# The Committee on Practical Application of PFM

[Back Ground]

The integrity of components in nuclear power plants can be quantitatively assessed in terms of probability and frequency of failure by using **probabilistic fracture mechanics (PFM)**. Although progress has been made in developing PFM analysis codes and establishing a guideline, **the practical application of PFM in regulatory decision making in Japan is limited**. Case studies using input data from **Japanese actual nuclear power plants** are desired as basic data for the application of PFM in Japan.

[Purpose]

In order to use PFM for design, operation and maintenance of actual nuclear power plants and structural components, it is confirmed **how much failure probability and how much variation** can be obtained by different PFM analysis tools and analysts under realistic conditions.

[Committee Members]

University of Tokyo
Japan Atomic Energy Agency (JAEA)
Mitsubishi Heavy Industry (MHI)
Tokyo Electric Power Company Holdings
Kansai Electric Power Company
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[Scope]

A benchmark analysis
Sensitivity analyses
Case studies for investigating various PFM applications

# PFM Analyses for Reactor Pressure Vessel of Japanese PWR Plant

PFM analyses of the reactor pressure vessel (RPV) under the accident condition (pressurized thermal shock (PTS) event) of the **Japanese representative PWR plant in Japan** were performed by PFM analysis code **PASCAL 4**[*1] and the failure frequency of the actual reactor pressure vessel was confirmed.

**1. PFM analyses (Basic analysis condition)**

PFM analysis was performed using the basic analysis conditions referring to the **Japanese PFM guideline, JEAG4640**[*2].

**2. Sensitivity Analyses**

The **accuracy** and **robustness** of the failure frequency by PFM analyses were confirmed by performing sensitivity analysis when the analysis conditions are changed.

**3. Benchmark Analyses**

The PFM analyses using the basic analysis condition were carried out in several organizations, and **the differences of the failure frequency by the different analysts** were confirmed.

**4. PFM analyses considering surveillance data**

Instead of using the embrittlement trend curve and the standard deviation of the prediction error specified in JEAC4201, the **embrittlement** and the **standard deviation** of the prediction error obtained by **Bayesian updating using the measured data** after irradiation (surveillance test data) was used, and the failure frequency was calculated.

**Item 3 will be presented in Part 2 of the series presentation.**

*1 Japan Atomic Energy Agency, “User’s manual and analysis methodology of probabilistic fracture mechanics analysis code PASCAL ver.4 for reactor pressure vessel”, JAEA-Data/Code 2017-015 (in Japanese)

*2 The Japan Electric Association, JEAG4640-2018, “The Guideline for calculation failure frequency of reactor pressure vessel based on probabilistic fracture mechanics (in Japanese) .

# Basic Analysis Condition

| Item | | Condition | Variation |
|---|---|---|---|
| Dimension of RPV | | Dimension of the Japanese PWR Plant | – |
| PTS Transients and frequency | | Refer to Beaver Valley Unit 1 (*1,*2) | |
| Weld Residual Stress | | Analysis results for Japanese plants (*3) | – |
| Flaw distribution | | Determined using the weld condition of the Japanese PWR plants and flaw distribution developed by the investigation of the Cancelled plant RPV in US (*2, *4) | |
| Irradiation Embrittlement Condition | Fluence Distribution | Fluence distribution of the Japanese PWR Plant RPV After 60 years operation | Refer to JEAG4640 equivalent to the analysis condition in US (*5) |
| | Chemical Composition | Chemical Composition of the Japanese PWR Plant RPV surveillance materials | Refer to JEAG4640 |
| Initial $RT_{NDT}$ | | Initial $RT_{NDT}$ of the Japanese PWR Plant RPV surveillance materials | Refer to JEAG4640 equivalent to the analysis condition in US (*6) |

Note: Condition in blue is refer to US analysis condition.

*1 Arcieri, W. C., et al., "RELAP5 Thermal Hydraulic Analysis to Support PTS Evaluations for the Oconee-1, Beaver Valley-1, and Palisades Nuclear Power Plants," NUREG/CR-6858, U.S. Nuclear Regulatory Commission, (2004).
*2 EricksonKirk, M., et al., "Technical Basis for Revision of the Pressurized Thermal Shock （PTS） Screening Limit in the PTS rule （10CFR50.61） ," NUREG-1806, U.S. Nuclear Regulatory Commission, (2006).
*3 Hirota, T., et al., "Proposal for Update on Evaluation Procedure for Reactor Pressure Vessels against Pressurized Thermal Shock Events in Japan," ASME Pressure Vessels and Piping Conference, PVP2014-28392, (2014).
*4 Kirk, M. E., et al., "Sensitivity Studies of the Probabilistic Fracture Mechanics Model Used in FAVOR," NUREG-1808, (2006).
*5 Kirk, M. E., et al., "Probabilistic Fracture Mechanics - Models, Parameters, and Uncertainty Treatment Used in FAVOR Version 04.1," NUREG-1807, (2007).
*6 U.S. Nuclear Regulatory Commission, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," 10CFR50.61.

# Basic Analysis Condition

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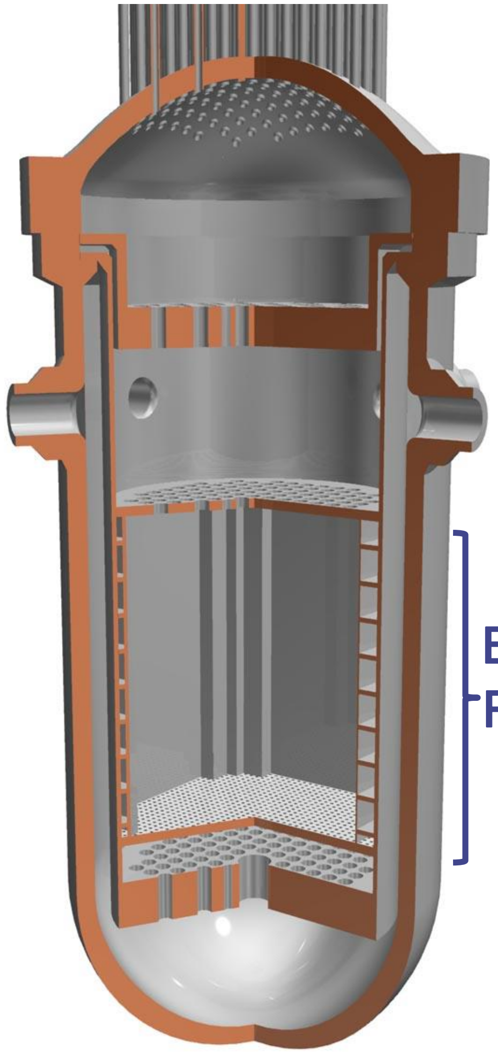
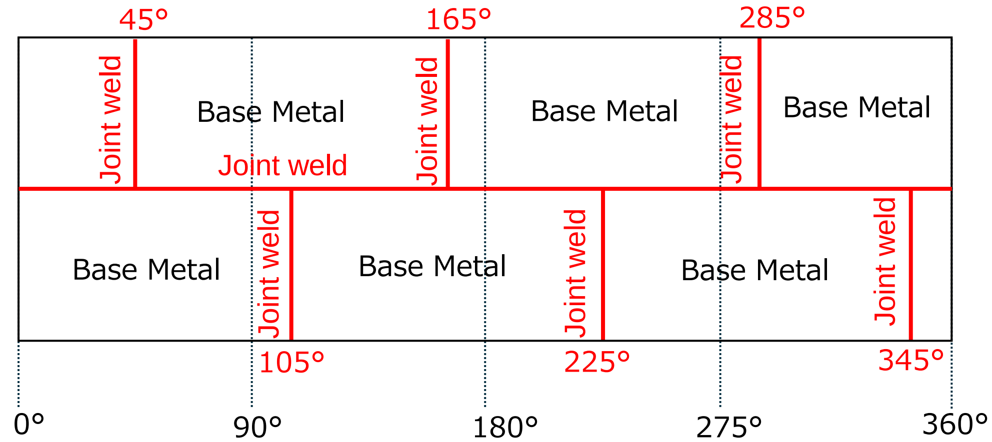
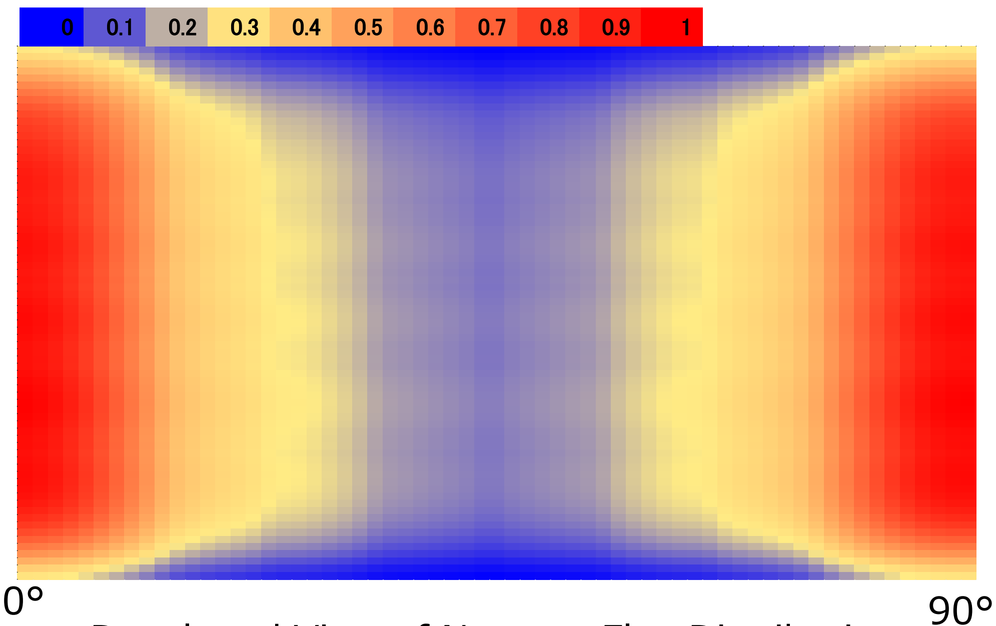



PWR Reactor Pressure Vessel




Developed View of Base metal and Joint Weld region




Developed View of Neutron Flux Distribution

# Analysis Procedure

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| Item | | Procedure | Variation |
|---|---|---|---|
| Stress Intensity Factor Solution | Semi-elliptical Surface flaw | Refer to *1 | – |
| | Embedded Elliptical flaw | Refer to *2 | – |
| | Infinite Length Surface Flaw (circumferential and axial) | Refer to *3 | – |
| Irradiation Embrittle Trend Curve | | JEAC 4201-2007 including 2013 addenda (*4) | Standard deviation based on the Japanese surveillance data |
| Fracture Toughness | $K_{Ic}$ | Refer to *5 | Weibull distribution refer to *5 |
| | $K_{Ia}$ | Refer to *5 | Log-normal distribution refer to *5 |
| Warm Prestressing | | ACE model refer to *6 | – |
| Through Wall Cracking Condition | | $a/t$ > 0.8 or Plastic collapse | – |

*1 Marie, S. and Chapuliot, S., "Improvement of the Calculation of the Stress Intensity Factors for Underclad and Through-clad Defects in a Reactor Pressure Vessel Subjected to a Pressurized Thermal Shock," International Journal of Pressure Vessels and Piping, 85, p517-531, (2008).

*2 ASME, "Boiler and pressure vessel code, Section XI, rules for inservice inspection of nuclear power plant components", BPVC-XI-2021, American Society of Mechanical Engineers, (2021).

*3 The Japan Society of Mechanical Engineers, Codes for Nuclear Power Generation Facilities, rules on Fitness-for Service for nuclear Power Plants, JSME S NA1-2012 (2012) (in Japanese).

*4 The Japan Electric Association Code, Method of Surveillance Tests for Structural Materials of Nuclear Reactors, JEAC4201-2007 including 2013 addenda (2013) (in Japanese).

*5 Takamizawa H. et al. "User's Manual and Analysis Methodology of Probabilistic Fracture Mechanics Analysis Code PASCAL Ver. 5 for Reactor Pressure Vessels", JAEA-Data/Code 2022-006 (2023) (in Japanese).

*6 Moinereau, D., et al., "Validation of ACE Analytical Criterion for Warm Pre-Stress Evaluation in RPV Integrity Assessment," ASME Pressure Vessels and Piping Conference, PVP2015-45103, (2015).

# PFM Results (Basic Analysis Condition)

- 95th percentile of TWCF is **about 2 orders of magnitude lower than the acceptance criteria** for failure frequency (1E-6 /reactor/year) in the USA.
- Circumferential surface flaw in base metal and circumferential embedded flaw are the main contributor to TWCF.

Analysis Results (Basic Analysis Condition)

| | 50%ile | 95%ile | 99%ile | mean |
|---|---|---|---|---|
| FCI(/ry) | 3.79E-09 | 5.07E-06 | 3.98E-05 | 1.67E-06 |
| TWCF(/ry) | 1.06E-11 | 1.42E-08 | 4.41E-07 | 2.33E-08 |

Contribution to Mean FCI and TWCF

| Flaw Category | | | FCI (/ry) | TWCF (/ry) |
|---|---|---|---|---|
| Base Metal | Embedded | Axial | 4.40E-12 | 4.38E-12 |
| | | Circumferential | 4.99E-12 | 5.52E-14 |
| | Surface | Axial | – | – |
| | | Circumferential | 5.68E-07 | 1.25E-08 |
| Weld | Embedded | Axial | 9.25E-10 | 9.09E-10 |
| | | Circumferential | 1.10E-06 | 9.37E-09 |
| | Surface | Axial | – | – |
| | | Circumferential | 1.22E-08 | 5.00E-10 |
| Total | | | 1.67E-06 | 2.33E-08 |

FCI: Frequency of Crack Initiation
TWCF: Through Wall Crack Frequency

# Sensitivity Analysis

The **accuracy** and **robustness** of the calculated failure frequency are confirmed by performing sensitivity analysis.

- The sensitivity analyses by **changing some uncertainties** (standard deviations) affecting the failure, which **do not have sufficient domestic data**, were conducted and the effects on the failure frequency are confirmed.
- In addition, the uncertainties which can refine the PFM analyses results are confirmed.
- Among the 61 transient conditions, **10 dominant transient conditions** (about 96% of TWCF) are selected and used for the analyses.
- The following standard deviation (SD) are changed in the sensitivity analyses.

| | | |
|---|---|---|
| 1. SD of fluence | : 10%, 20% | (basic analysis: 13.1%) |
| 2. SD of Cu | : 0.005%, 0.02% | (basic analysis: 0.01%) |
| 3. SD of initial $RT_{NDT}$ | : $0^oC$, $15^oC$ | ( basic analysis: $9.4^oC$) |
| 4. SD of ETC | : $0^oC$ | ( basic analysis: $8.9^oC$) |

## Results of Sensitivity Analysis

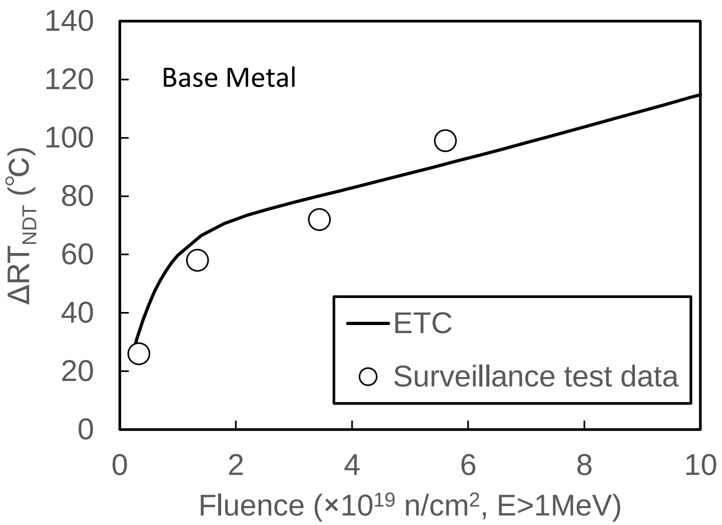
- The effects of SD of fluence, Cu and ETC on TWCF were not large.
- On the other hand, **the effect of SD of the $RT_{NDT}$ initial value was large**.
- The SD of the $RT_{NDT}$ initial value for the basic analysis condition was set based on the 10CFR50.61 in the USA. Therefore it is desirable to refine the calculated failure frequency by setting the **more suitable SD of the $RT_{NDT}$ initial value for Japanese plant RPV materials**.
- However, it was confirmed that **the failure frequency was still lower than the acceptance criteria in the USA**, even if the SD of $RT_{NDT}$ initial value was set excessively large.

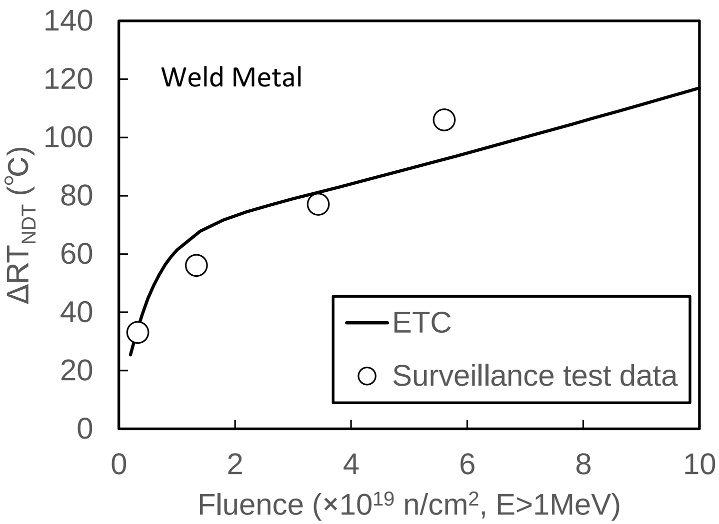
Comparison of mean TWCF (/ry) by sensitivity analyses

| | Item | Case 1 | Basic Analysis | Case 2 |
|---|---|---|---|---|
| SD of fluence | SD | 0.1 | 0.131 | 0.2 |
| | Mean TWCF | 2.00e-8 | 2.24e-8 | 3.16e-8 |
| SD of Cu | SD | 0.005% | 0.01% | 0.02% |
| | Mean TWCF | 2.30e-8 | 2.24e-8 | 2.31e-8 |
| SD of $RT_{NDT}$ initial value | SD | 0°C | 9.4°C | 15°C |
| | Mean TWCF | 9.00e-9 | 2.24e-8 | 1.17e-7 |
| SD of ETC | SD | 0°C | 8.9°C | – |
| | Mean TWCF | 1.71e-8 | 2.24e-8 | – |

# PFM analyses considering surveillance data

- As one of case studies for investigating various PFM application, calculation of failure frequency of RPV using surveillance test data for the RPV material was conducted.
- In the calculation of PFM, instead of using the embrittlement trend curve and the standard deviation of the prediction error specified in JEAC4201, **the embrittlement and the standard deviation of the prediction error obtained by Bayesian updating using the measured data after irradiation (surveillance test data)** was used, and the failure frequency was calculated.
- Among the 61 transient conditions, **10 dominant transient conditions** (about 96% of TWCF) are selected and used for the analyses.




Surveillance Test Data and ETC

# Bayesian update of ETC error

- **Bayesian update** was performed according to *1.
  The prediction error by ETC for the surveillance test data was assumed to be normal distribution for the mean and inverse gamma distribution for the variance, and treated as normal inverse gamma distribution which can be calculated simultaneously.
- Prior distribution: ETC prediction errors (measured - predicted) of $\Delta RT_{NDT}$ **for all Japanese surveillance test data**

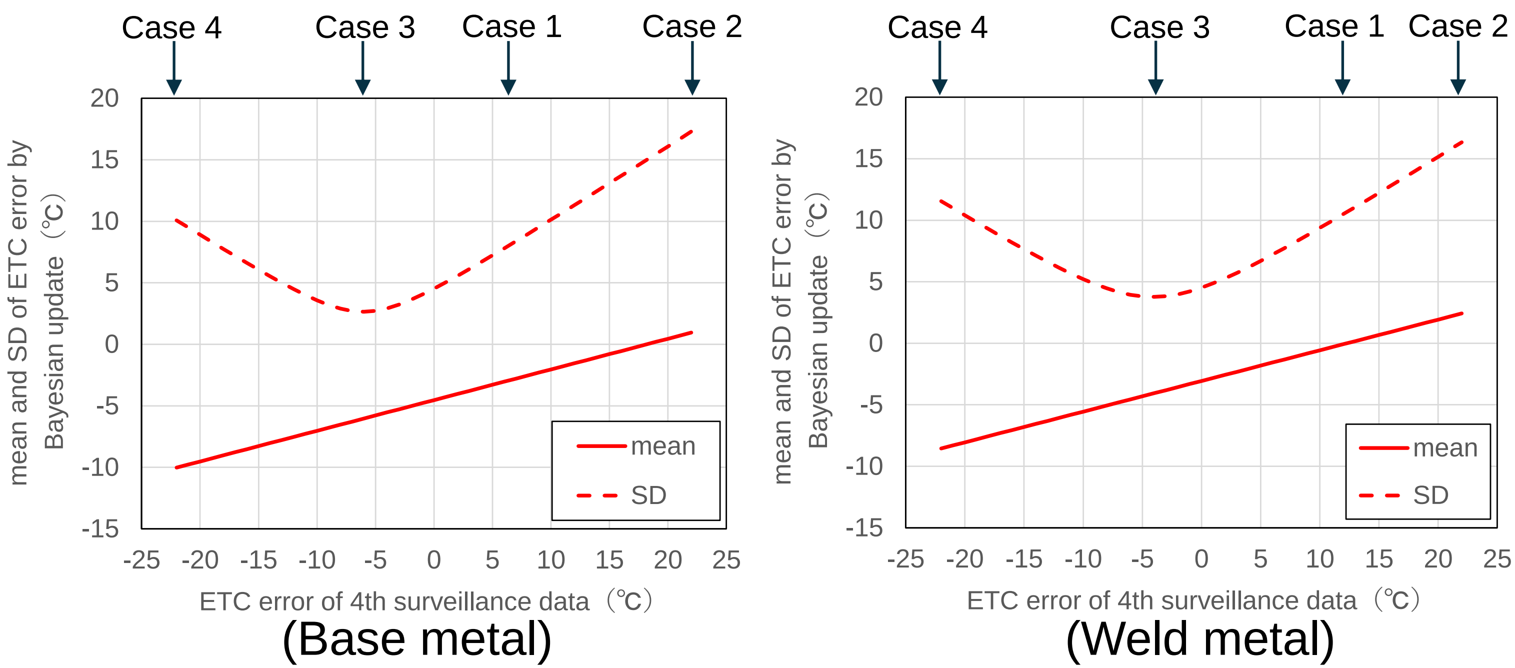
| | |
|---|---|
| Mean | 1.1°C |
| Standard deviation (SD) | 9.5°C |

- Bayesian updating for base metal and weld metal was performed using **each surveillance test data**.
- In Bayesian updating, **4th surveillance data** was changed **parametrically** in the range of margin for ETC (22°C), which was conservatively determined considering mean and 2 x SD for ETC prediction error.

*1 Kevin P. Murphy et al.. Conjugate Bayesian analysis of the Gaussian distribution. 2017.

# Results of Bayesian update of ETC error

- Mean and standard deviation (SD) of the prediction error of $\Delta RT_{NDT}$ obtained by Bayesian updating using the surveillance test data are shown below, when 4th surveillance data was changed parametrically.
- **SD increased** as the predicted error of the 4th surveillance test **deviates from plausible error by the past 3 surveillance data** (around -5ºC for both material), and the uncertainty of the prediction error increased.
- In PFM analyses considering surveillance data, 4 sets of mean and SD of prediction error were additionally used as Case 1 - 4.




Case 1: Test results
Case 2: Large $\Delta RT_{NDT}$
Case 3: Small SD
Case 4: Small $\Delta RT_{NDT}$

Results of Bayesian updating of mean and standard deviation for ETC prediction error

# Results of PFM analyses considering surveillance data

- As expected, the larger the mean and standard deviation of the predicted errors, the higher the failure frequency.
- In case of **assuming the excessively large embrittlement** in the surveillance test, the **TWCF increased by one order of magnitude**, however, **still below the acceptance criteria in the USA**.
- It is considered that the **importance of surveillance tests** (interval and number of the surveillance capsule withdrawal) can be examined by evaluating the effect of surveillance test data on the failure frequency.

Results of PFM analyses considering surveillance data

| | Base metal | | Weld Metal | | FCI (/ry) | TWCF (ry) |
|---|---|---|---|---|---|---|
| | Mean of ETC error | SD of ETC error | Mean of ETC error | SD of ETC error | | |
| **0. Basic Analysis** | 1.1°C | 8.9°C | 1.1°C | 8.9°C | 5.07E-06 | 1.42E-08 |
| **1. Test results** | -2.9°C | 8.2°C | -0.1°C | 10.4°C | 2.28E-07 | 4.74E-09 |
| **2. Large $\Delta RT_{NDT}$** | 1.0°C | 17.3°C | 2.4°C | 16.3°C | 1.25E-06 | 9.92E-08 |
| **3. Small SD** | -6.0°C | 2.6°C | -4.1°C | 3.8°C | 6.56E-08 | 1.17E-09 |
| **4. Small $\Delta RT_{NDT}$** | -10.0°C | 10.1°C | -8.5°C | 11.6°C | 7.31E-08 | 1.34E-09 |

# Conclusion

- The committee on practical application of PFM was established to calculate failure frequency of reactor pressure vessel (RPV) using analysis conditions of **Japanese actual nuclear power plants**. Part 1 of this series presentation describes the results of **sensitivity analyses** and **PFM considering surveillance data** as a case study.
- As the results of the sensitivity analyses, the effect of the **standard deviation of the $RT_{NDT}$ initial value on the failure frequency was larger** than that of the standard deviations of other parameters. However, it was confirmed that the failure frequency was **still lower than the acceptance criteria in the USA**.
- In the PFM analyses, instead of using the ETC and the standard deviation of the prediction error specified in JEAC4201, **the mean and the standard deviation of the prediction error obtained by Bayesian updating using the surveillance test data** was used. In case of assuming the **excessively large embrittlement** in the surveillance test, the failure frequency was **still below the acceptance criteria in the USA**. It is considered that the **importance of surveillance tests** (interval and number of the surveillance capsule withdrawal) can be examined by evaluating the effect of surveillance test data on the failure frequency.

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