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PLANT-SPECIFIC PRESSURIZED THERMAL SHOCK INTEGRITY EVALUATION FOR KORI UNIT 1 REACTOR PRESSURE VESSEL

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ABSTRACT

During the residual life evaluation of Kori unit 1 reactor pressure vessel, the reference temperature pressurized thermal shock (RT_{PTS}) was projected to exceed screening criteria before the end of design life, or 40 years. To cope with this issue, a plant-specific pressurized thermal shock (PTS) analysis was performed following the methodology and procedures suggested in Reg. Guide 1.154. The plant-specific PTS analysis covers identification and quantification of PTS-significant sequences, thermal-hydraulic analysis, downcomer mixing analysis, and probabilistic fracture mechanics analysis to quantify the associated risk of RPV failure with various PTS transients. The step-by-step procedures adopted in the plant-specific PTS analysis and the results are described in detail. Also, some lessons learned while performing the analysis are discussed. Through the detailed analysis, it is now expected that RPV can maintain enough safety margin against pressurized thermal shock during and beyond its design life.

1. INTRODUCTION

In general, as the operation year of nuclear power plant (NPP) is accumulated, the progression of irradiation embrittlement of reactor pressure vessel (RPV) results in decrease in upper shelf energy (USE) and increase in the reference temperature-pressurized thermal shock (RT_{PTS}). The net results of irradiation embrittlement are the increased strength and reduced fracture toughness. When embrittled RPVs are subjected to high stress, existing cracks, if any, may initiate and propagate to the outer surface of the vessel. Significant thermal stress could occur from the thermal shock following cold emergency core cooling water injection at the events of various transients during operation. When system pressure remains high or slowly decreases during the thermal shock events, additional stress from the system pressure greatly increases the possibility of crack initiation and propagation.

This kind of phenomena can occur not only during loss-of-coolant accident (LOCA) type events but also non-LOCA type events. It was defined as a pressurized thermal shock (PTS) by USNRC in early 1980s [1]. USNRC subsequently initiated PTS analysis studies on H. B. Robinson (Westinghouse) [2], Oconee-1 (B&W) [3], and Calvert Cliffs-1 (CE) [4]. It was found that the failure probability of vessel was strongly dependent upon the degree of irradiation embrittlement, measured as adjusted reference PTS temperature (RT_{PTS}). Especially, some welds, containing large amount of copper and located in the core beltline region, were considered to be susceptible to radiation embrittlement and critical to the reactor vessel integrity. Based on these and other researches, PTS rule (10CFR50.61) was issued by USNRC in 1985 [5]. It was revised in 1991 and again in 1996 to reflect advanced knowledges and clarify some ambiguities.

The content of the PTS rule are;

- Define reference PTS temperature, RT_{PTS}

  RT_{PTS} = initial RT_{NDT} + shift of RT_{NDT} + Margin

- Provide methods of calculating RT_{NDT} shift adopting Reg. Guide 1.99 [6]

  a. utilizing chemistry factor tables based on copper and nickel contents

  b. utilizing surveillance specimen test data

- Define PTS screening criteria

  a. $RT_{PTS} < 270°F$ for plates, forgings, and axial welds

  b. $RT_{PTS} < 300°F$ for circumferential welds

- Require every plant to submit estimated RT_{PTS} at end-of-life (EOL) fluence
- If the estimated RT_{PTS} are to exceed the screening criteria before EOL, plant-specific PTS analysis incorporating probabilistic methods should be performed to quantify the risk of RPV failure associated with PTS phenomena for continued operation.
- Plant-specific PTS analysis should be done based on the Reg. Guide 1.154 [7].

The RPV of Kori Unit 1 is one of the typical Westinghouse 2-loop design and fabricated by B&W. Its shells were made of SA 508 Cl. 2 ring forging clad with stainless steel 308 type weld. The schematic of the RPV is shown in figure 1. As shown in the figure, there are three circumferential welds near reactor core, that is WF259, WF232/233, and WF267. Of the three welds, the one near the core midplane, or WF233 has been identified as the most controlling materials in terms of irradiation embrittlement [8]. The best estimate chemistry of WF233 was suggested as 0.29% copper and 0.68% nickel [9]. Because of extensive irradiation embrittlement, USE of the WF233 weld fell below 50 ft-lbs, which is the minimum requirement [10], after only a few years of commercial operation. A detailed fracture mechanic analysis was done to deal with the low USE issue and showed that RPV could maintain its integrity for the design life [11].

On the other hand, as shown in figure 2, the RT_{PTS} was projected to exceed screening criteria of 300°F before its design life [8]. To cope with this PTS issue, KEPRI initiated plant specific PTS analysis following the methodology and procedures suggested in Reg. Guide 1.154 [12]. In this paper, the specific methodology and step-by-step procedures adopted in the analysis and results are described in detail. Also, lessons learned through the analysis are described.

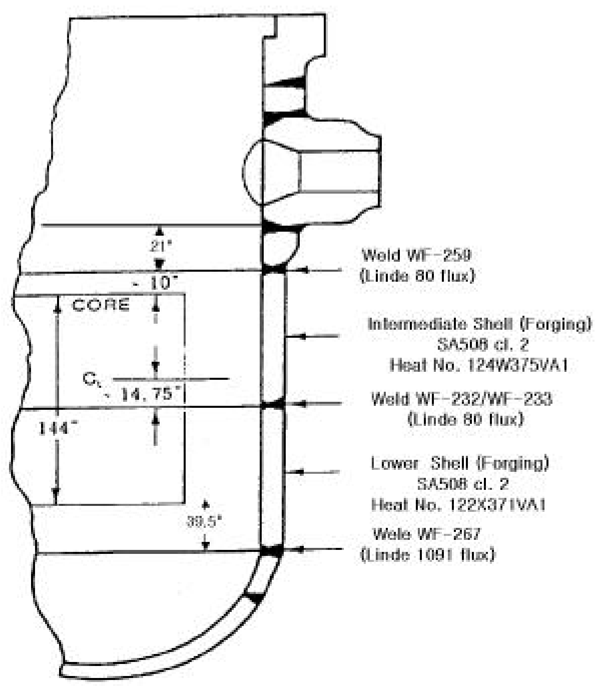

Figure 1. Schematics and materials of construction of Kori Unit 1 RPV.

2. METHODOLOGY AND PROCEDURES

According to the PTS rule, if the estimated RT_{PTS} are expected to exceed the screening criteria before end of life, a plant-specific PTS analysis should be performed to demonstrate

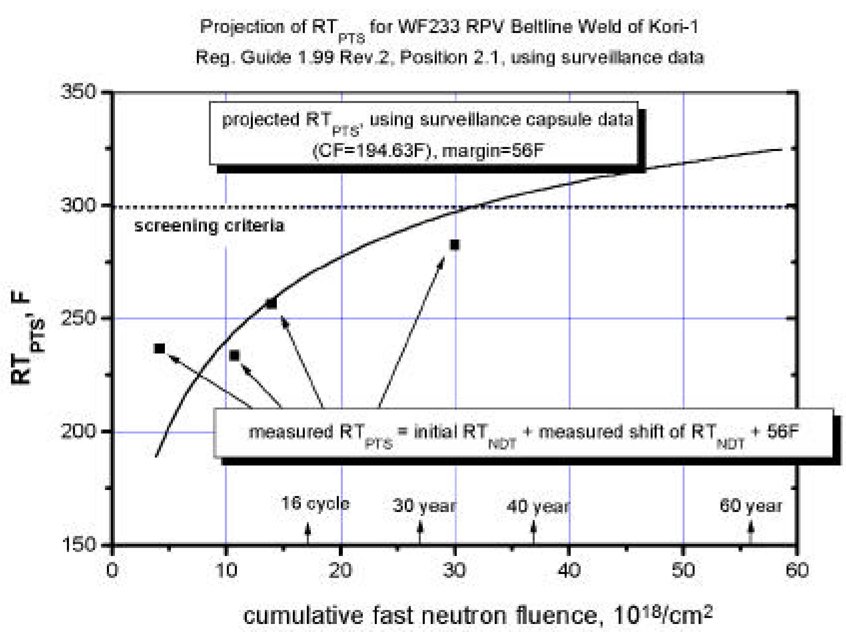

Figure 2. Projection of RT_{PTS} of a circumferential weld, WF233 of Kori Unit 1 RPV.

that total frequency of through-wall crack (TWC) due to PTS is less than 5×10^{-6}/Rx-yr for continued operation [7]. The overall flow of the plant-specific PTS analysis adopted in this study is shown in Figure 3. First, PTS initiating events were identified and event-trees were constructed by carefully analyzing plant specific data. Next, the event frequencies of the sequences are quantified by probabilistic risk analysis technique. The PTS significant transient sequences are classified and grouped, in conservative way, based on the similarity in expected thermal-hydraulic (T/H) natures and frequency of the sequences. For the representing transient sequences that would result in the most conservative results within the sequence groups, T/H analyses were performed using transient analysis codes, such as RELAP5 and RETRAN. If thermal stratification within the cold leg is suspected, mixing analyses were needed to obtain localized temperature near RPV wall in downcomer region.

The following step is the probabilistic fracture mechanics (PFM) analysis. Downcomer pressures, fluid temperatures near RPV wall, and heat transfer coefficients vs. time that were

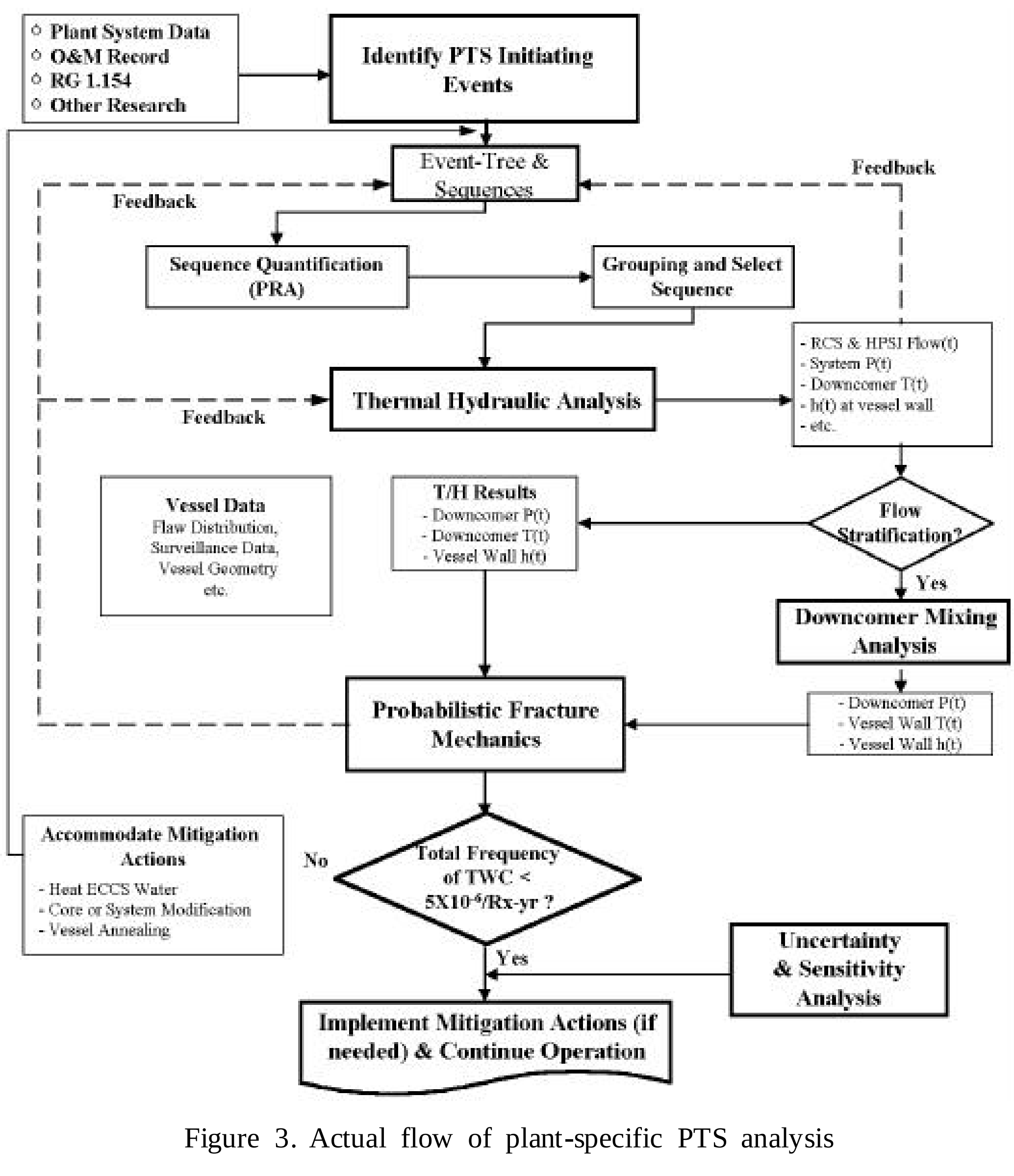

Figure 3. Actual flow of plant-specific PTS analysis

obtained from T/H and mixing analyses were provided as inputs to PFM analyses. The specific vessel data, such as, thermo-physical material properties, geometry, and surveillance capsule data et. al. were needed also. Through the PFM analysis, conditional TWC probability, P(F/E) for each representing transient sequence was calculated. TWC frequency at the event of specific PTS sequence is calculated by multiplying the overall frequency of each sequence group and P(F/E). Finally, the total TWC frequency is found by simply adding the vessel failure frequencies of all transient sequence groups analyzed.

3. KORI UNIT 1 PLANT-SPECIFIC PTS ANALYSIS

3.1. Selection of Initiators and Sequences

Based on the careful review of systems, operating procedures, Reg. Guide 1.154 [7], and previous studies [2,3,4], the main steam line break (MSLB), steam generator tube rupture (SGTR), loss of main feedwater (LOMFW), small break loss of coolant accident (SBLOCA), and loss of heat sink (LOHS) were selected as the potential PTS initiating events. Component actuation and operator actions affecting overcooling of RPV or repressurization were selected as headings and branches for event-tree construction. It should be noted that some of the operator actions, for example, restart of tripped reactor coolant pump, that could have been beneficial in alleviating the consequences of the transients were intentionally omitted. It could simplify the event trees construction and T/H analysis and, eventually, give somewhat conservative analysis results.

The probabilities of success or failure of branches associated with the key components were calculated using the EPRI database or those of similar plants. Probabilities associated with operator actions are determined by interview and questionnaire with plant operators. A total of 134 potential overcooling sequences from 5 initiating events were identified. The sequences were classified and grouped based on the similarity in T/H nature. Also, the overcooling sequences with frequency less than 1×10^{-10}/Rx-yr were categorized as a set of residual groups. A total of 24 representative sequences were selected for further analysis in the next steps.

3.2. Thermal-Hydraulic Analysis

The purpose of T/H analysis is to obtain pressures, fluid temperatures, and heat transfer coefficients at RPV inner surface in the beltline region to be used as PFM inputs. The representative sequences belong to the initiating events of MSLB, SGTR, LOMFW, and LOHS were analyzed using RETRAN-3D [13]. On the other hand, those of SBLOCA were calculated using RELAP5/MOD3.2 code [14], because of better capability in predicting two-phase flow behavior expected in SBLOCA. The RETRAN-3D and RELAP5/MOD3.2 models were benchmarked against the full power trip test data and the normal operating data, and the results was in good agreement within ±5%.

For each sequence selected above, T/H behavior was analyzed for 2 hours as recommended in Reg. Guide 1.154 [7]. Figure 4 shows T/H analysis result for one of the overcooling sequences, that is an SGTR type transient at hot zero power, showing downcomer temperatures, pressures, and heat transfer coefficients vs. time. Of the overcooling sequences analyzed, LOHS was identified as the event producing the lowest downcomer temperature

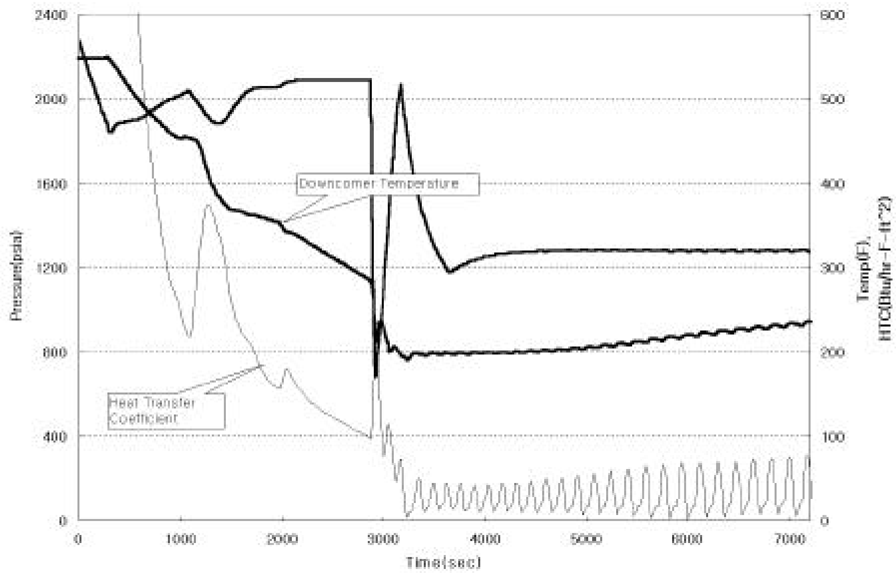

Figure 4. Temperature, pressure, and heat transfer coefficient vs. time during an SGTR type transient at hot zero power.

owing to high safety injection and charging flow rate during the feed and bleed operation.

3.3. Mixing Analysis

Since the above mentioned T/H codes do not model fluid behavior with sufficient detail to predict thermal stratification phenomenon in cold leg and downcomer region, additional mixing analysis is needed to find local temperature near RPV wall. In order to determine which overcooling sequences are affected by thermal stratification, thermal stratification criteria of T. G. Theofanous [15] was applied. For the mixing analysis, 3-dimensional computational fluid dynamics (CFD) codes, such as, PHEONICS and CFX-4 codes were used. The PHEONICS [16] model was benchmarked to the experimental results of CREARE-1/2 scale test [17] as shown in figure 5, and the CFX-4 modeling was validated by the PHEONICS code. At first, REMIX [18] was also considered as a potential mixing code for the PTS analysis. However, due to the difference in geometry modeled in REMIX and the plant specific features, REMIX results deviated as much as 100°F from PHEONICS results. Consequently, REMIX results were used for comparison purpose only, and not used as PFM inputs.

Calculated fluid temperature near RPV wall from RETRAN, PHEONICS, and REMIX for SGTR at hot zero power are shown in figure 6. When the stratification was considered, the local temperatures near RPV inner wall were 40~120°F lower than the temperature calculated by RETRAN which is the volume averaged fluid temperature in downcomer region. Overall, 6 overcooling sequences were analyzed using PHEONICS or CFX-4. Mixing analysis results are summarized in table 1. The potential significance of stratification, that can be measured as the temperature difference between the T/H and mixing analyses, were greater for SGTR type transients.

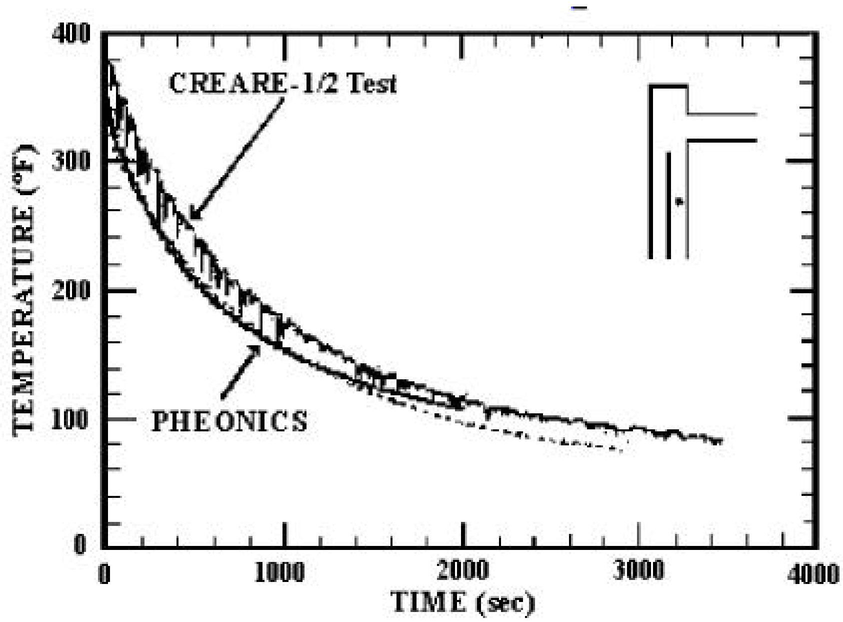

Figure 5. Comparison of CREARE-1/2 test and PHEONICS result

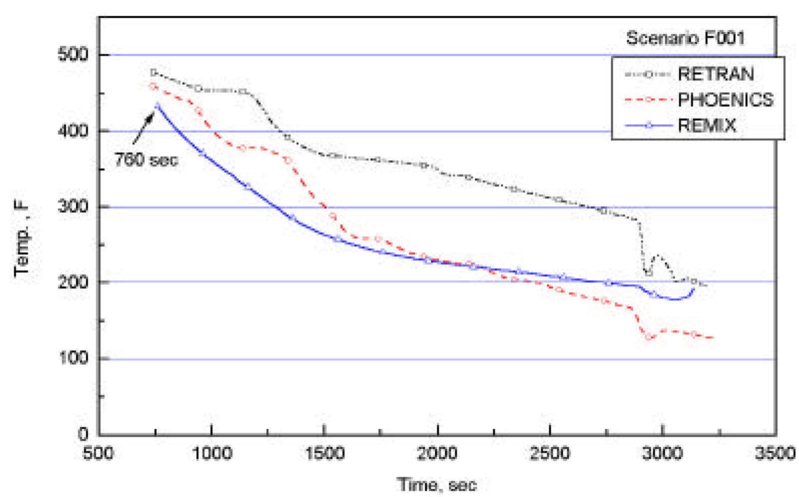

Figure 6. Comparison of fluid temperature near RPV wall from RETRAN, PHEONICS, and REMIX during SGTR at hot zero power

3.4. Probabilistic Fracture Mechanic Analysis

To calculate conditional TWC probability, P(F/E) of the RPV during the overcooling transients, probabilistic fracture mechanics (PFM) analysis were performed. Of the several analytical codes developed, VISA-II [19] and FAVOR [20] were selected for the PFM analysis of this study. However, recent benchmarking study revealed that the calculated

Table 1. Summary of downcomer mixing analysis results

Initiating Event		Sequence ID	Mixing Code	Minimum Temp. (T/H) (°F)	Minimum Temp. (Mixing) (°F)	Maximum Pressure (psia)
SGTR	HZP	F001	PHOENICS	191	115.4	2,090
		F016	"	151	100.0	2,080
LOHS	FP	H001	"	82	84.0	1110
SBLOCA	HZP	I001	CFX-4	100	87.0	520
		I003	"	170	158.0	500
		I006	"	160	159.0	550

failure frequency may vary considerably depending on the way to treat thermal hydraulic boundary conditions, to calculate stress intensity factors, etc [21]. For better interpretation of conditional failure probability calculated by each code, analysis results of the two codes for hypothetical thermal-hydraulic conditions and RPVs were compared [22]. As shown in the figures 7 and 8, it was found that (1) the TWC frequency of VISA-II was 1.5~2 times higher than that of FAVOR for simple but severe PTS transients for circumferential cracks and (2) as the T/H conditions are complicated the results of VISA-II become more conservative in part by the tendency of choosing inputs conservatively. After careful consideration of benchmarking analysis results, more conservative VISA-II results were decided to be submitted to regulatory body. The results using FAVOR was used as comparison purpose and potential backup materials to emphasize the conservatism associated with the analysis.

Low USE of the weld was reflected by lowering maximum fracture toughness of the weld. Assumed flaw distribution was Marshall flaw distribution considering pre-service inspection [23]. At least ten million trials of simulation was applied for PFM analysis. If

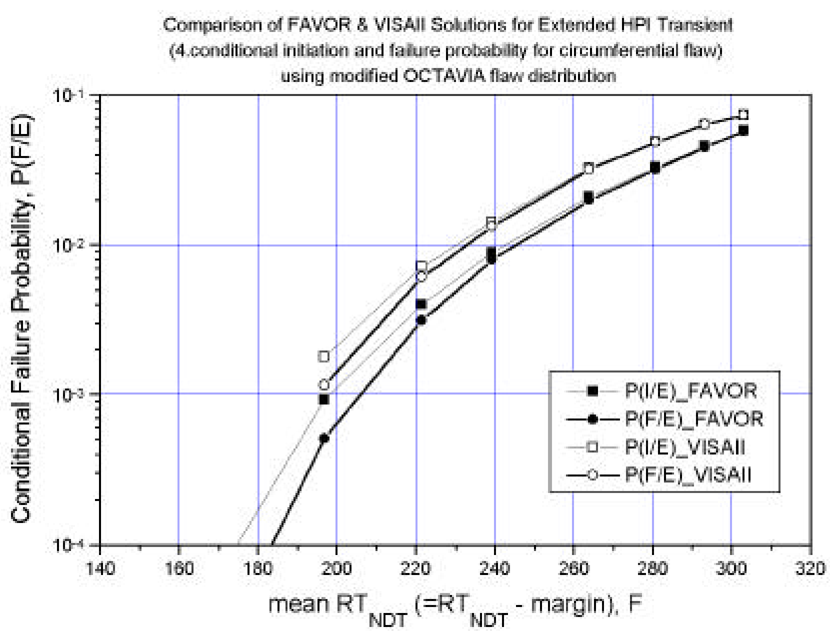

Figure 7. Comparison of circumferential flaw initiation and failure probabilities at the event of Extented HPI transient [22].

no TWC was observed during the simulation, number of simulation was increased by 10 times. If P(F/E) is zero, despite of the increased number of simulation, it was conservatively assumed as 3×10^{-8}.

Example of the PFM analysis result for some of the PTS transients are summarized in table 2. Of the sequences analyzed, P(F/E)s of the transients associated with SGTR were the greatest. These high P(F/E)s are thought to be attributed to the repressurization, as shown in figure 4. The high repressurization pressure, in turn, was thought to be the results of the high shut-off head of safety injection. LOHS transient with the lowest final temperature and modest pressure was identified as the second most severe PTS transient. The I001 transient (SBLOCA) that has a very low final temperature showed smaller P(F/E) because of the relatively low pressure during the transient.

Table 2. Examples of PFM analysis results using VISA-II Code

Initiating Event		Sequence ID	Temperature (°F)			Cooling rate, β (min^{-1})	HTC (BTU/hr .ft^2F)	P(F/E)		
			Initial	Min.	Final			32 EFPY (40 op. yrs)	40 EFPY (50 op. yrs)	46.4EFPY (60 op. yrs)
SGTR	FP	E001	541.5	363.0	487.0	P[1]	400	3.00×10^{-8}	3.00×10^{-8}	3.00×10^{-8}
		E016	541.5	214.0	347.4	P[1]	300	4.77×10^{-6}	2.63×10^{-5}	4.66×10^{-5}
	HZP	F001	547.5	115.4	115.4	0.1	150	5.27×10^{-4}	2.42×10^{-3}	3.42×10^{-3}
		F016	547.5	100.0	125	P[1]	200	7.72×10^{-3}	1.13×10^{-2}	1.41×10^{-2}
LOMFW	FP	G001	541.5	540.0	547.3	P[1]	5,137	3.00×10^{-8}	3.00×10^{-8}	3.00×10^{-8}
		G004	542	488.0	548.0	P[1]	5,140	3.00×10^{-8}	3.00×10^{-8}	3.00×10^{-8}
LOHS	FP	H001	541.5	84.0	84.17	P[1]	200	7.26×10^{-4}	1.60×10^{-3}	2.53×10^{-3}
SB LOCA	FP	I001	545	85.0	85.0	P[1]	166	3.60×10^{-5}	1.12×10^{-4}	2.19×10^{-4}
		I003	545	141.8	165.1	P[1]	205	3.00×10^{-8}	5.29×10^{-8}	1.94×10^{-7}
		I006	545	145.2	153.3	P[1]	188	3.70×10^{-8}	3.19×10^{-7}	1.40×10^{-6}

[Note] (1) Polynomial fitting

3.5. Integrated PTS Risk and Sensitivity

The total TWC frequency (or, integrated PTS risk) was calculated as follows;

$$\text{Integrated PTS risk} = \sum (P(E) \times P(F/E))$$

where P(E): event frequency of specific overcooling sequence
P(F/E): conditional TWC probability at the event of specific sequence

The integrated PTS risk calculated by above equation was shown in figure 8 and compared with the limit specified in Reg. Guide 1.154 (that is, 5.0×10^{-6}/Rx-yr) to determine the integrity of the RPV at the events of potential PTS transients. As shown in the figure, among the PTS initiating events, SGTR was the most dominant contributors to the PTS risk. Despite of high P(F/E), LOHS contributed little to the integrated PTS risk due to the small sequence frequency. The calculated PTS risk associated with SGTR and SBLOCA represented more than 90% of the integrated PTS risk.

As shown in figure 8, additional increase in the PTS risk during the extended operation period is not significant. This was not an unusual behavior considering that irradiation embrittlement correlation used in the study tend to predict near-saturation behavior at high fluence, and Marshall flaw distribution assumes less flaws in smaller size whose stability strongly depend on the amount of neutron fluence.

The results of the sensitivity study are summarized in table 3. As shown in the table, flaw density is the most sensitive parameter in calculating the PTS risk of RPV. The downcomer temperature and the SGTR frequency were ranked second and third, respectively.

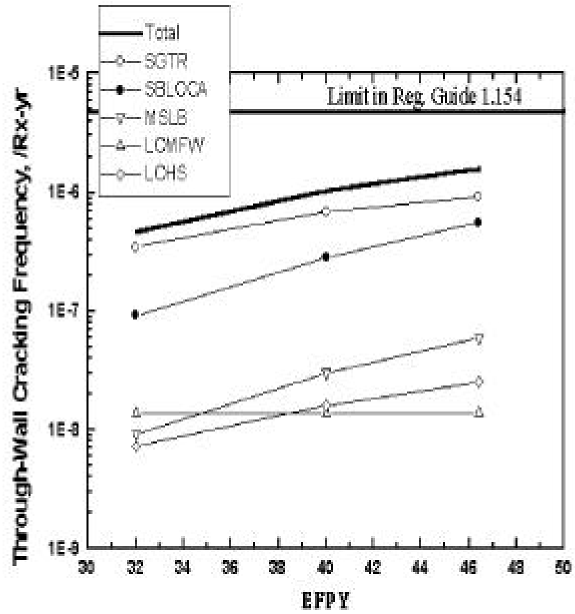

Figure 8. Calculated through-wall-cracking frequency (using VISA-II results)

4. CONCLUSIONS

A plant-specific PTS analysis has been conducted following the methodology and procedures suggested in Reg. Guide 1.154. The plant-specific PTS analysis covers identification of PTS-significant sequences, T/H analysis, mixing analysis, and PFM analysis to quantify the associated risk of RPV failure with various PTS transients. Throughout this PTS analysis, at least two analysis codes were used in T/H analysis, mixing analysis, and PFM analysis. By taking this approach, the advantages and disadvantages of each codes are studied in detail.

Through the plant specific PTS integrity evaluation, following conclusions were drawn:

- Commercial CFD codes, such as PHEONICS and CFX-4 would be appropriate choice for mixing analysis rather than REMIX code.
- VISA-II code showed a quite conservative conditional failure probabilities due to the limitation in treating thermal hydraulic input data and calculation module of stress intensity factors.
- Additional increase in the PTS risk during the extended operation period is not

Table 3. Results of sensitivity analysis of parameters

Parameter	**Sensitivity (TWCI / TWCo)**	**Rank**
Flaw density	74.37	1
Downcomer fluid temperature	13.15	2
Event frequency of SGTR at hot zero power	3.06	3
Copper contents	2.29	4
Initial RT_{NDT}	2.25	5
Convective heat transfer coefficient	1.89	6
Event frequency of SBLOCA at full power	1.54	7
Fast neutron fluence	1.51	8

significant.

- Through the detailed analysis, it is now expected that Kori Unit 1 RPV can maintain enough safety margin against pressurized thermal shock during and beyond its design life.

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