

Integrity Assessment of the Kori Unit 1 Reactor Pressure Vessel

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Abstract

The reactor pressure vessel (RPV) is the most critical component of pressurized water reactor. It has to comply with various rules and regulatory guides to ensure sufficient safety and operating margins during its lifetime including extended operation. Thus, it is crucial to assure the integrity of RPV for effective Plant Lifetime Management Program (PLiM). In this paper, the status and various issues on the integrity of the RPV of Kori Unit 1 are introduced. A circumferential weld in the beltline region was projected to be unable to satisfy the minimum upper shelf energy requirement and maximum reference temperature-pressurized thermal shock requirement before 40-year lifetime. The results of the detailed integrity analyses on both issues are summarized. As integral parts of PLiM Phase II Program of KEPCO, several actions have been taken as aging management programs to assure the integrity of Kori Unit 1 reactor pressure vessel, such as, redefining initial RT_{NDT} , installing ex-vessel dosimetry, and withdrawal and testing of additional surveillance capsule. The applicability of these and other options including thermal annealing in current license period and during the extended operation are examined.

1. Instruction

The reactor pressure vessel (RPV) in a pressurized water reactor (PWR) is a key component of the pressure boundary of reactor coolant system, providing primary coolant path, housing and supporting fuel assemblies and internals. It also acts as the additional barrier to prevent uncontrollable release of radioactive materials into the containment. Therefore, the integrity of RPV is essential to the safety of nuclear power plants.

Kori Unit 1 is a typical Westinghouse 2-loop plant with gross capacity of 587 MWe. It is the first commercial nuclear power plant in Korea and has been in operation since 1978. In the life extension feasibility study, lasted for 3 years, from 1993 to 1996, its RPV was selected as the most critical component for Plant Lifetime Management (PLiM) by systematic scoping of all the components of Kori Unit 1[1]. The most significant aging effects of the RPV were identified as fatigue and radiation embrittlement. Quantitative residual life evaluation of fatigue using conservative cyclic loading conditions showed sufficient margin on fatigue usage factor for 60-year operation[2].

However, it has been found that the radiation embrittlement of the RPV materials, especially, of the circumferential weld in the beltline region has been progressed considerably. For a successful PLiM program and continued operation beyond the current license period, a comprehensive plan to attain the structural integrity of the embrittled RPV has to be established.

In this paper, the status and various issues on the integrity of the RPV of Kori Unit 1 are introduced and the results of the detailed integrity analyses are summarized. Also the various actions such as, redefining initial RT_{NDT} , installing ex-vessel dosimetry, and early withdrawal and testing of additional surveillance capsule, are described in somewhat detail. Finally the applicability of these and other options including thermal annealing in current license period and during the extended operation were examined.

2. Regulatory Requirements on RPV Integrity

Radiation embrittlement of RPV materials are characterized as the decrease in the upper-shelf energy (USE) and increase of the reference temperature-nil ductility transition temperature (RT_{NDT}) due to the accumulation of fast neutron exposure during operation. And, utilities are required to set up a surveillance program to monitor the degree of radiation embrittlement of the RPV materials. As shown in Fig. 1, USE and RT_{NDT} are measured in the surveillance test.

The USE of irradiated RPV beltline materials is required to be greater than 68 J (50 ft-lbs) to ensure sufficient resistance to brittle fracture during normal operation including heatup/cooldown processes [3]. If the USE is projected to be less than 68 J before the end of life, full volumetric inspection of the RPV beltline and fracture mechanics analysis according to regulatory guide are required to verify stability of cracks for continued operation.

When embrittled RPVs are subjected to the pressurized thermal shock (PTS), the combination of thermal stress and pressure stress can considerably increase the possibility of through-wall propagation of existing cracks. To assure the integrity of RPVs at the event of PTS, PTS rule requires that the reference temperature-pressurized thermal shock (RT_{PTS}) of RPV beltline materials including base and welds should be lower than the PTS screening criteria[4]. The rule further requires that if RT_{PTS} is expected to exceed screening criteria before the end of life, a plant-specific PTS analysis should be performed in accordance with regulatory guide[5] to quantitatively evaluate the risk of RPV failure associated with PTS.

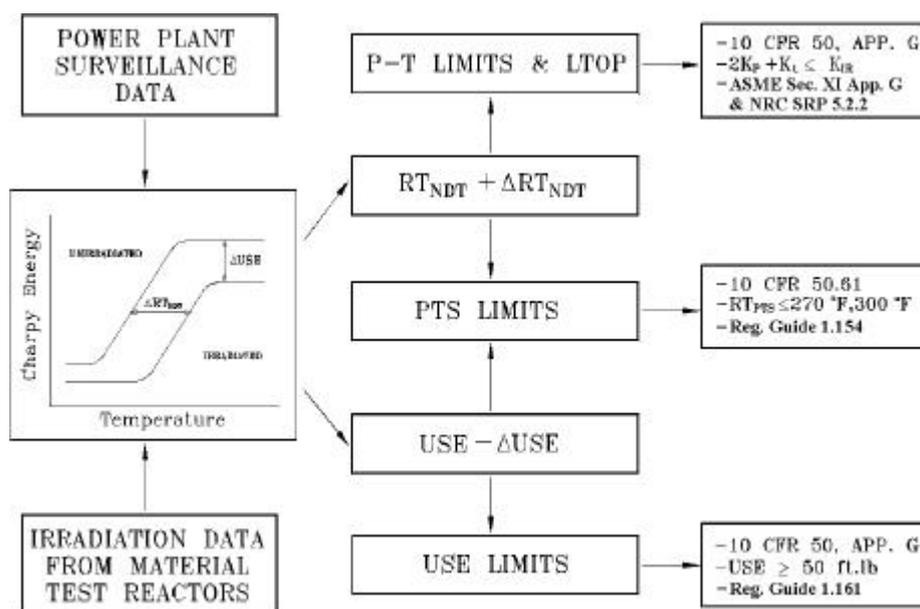


Figure 1. Rules, regulations, and codes relevant to RPV integrity

Additionally, radiation embrittlement affects the operability of the plant by shifting the P-T limit curves. The P-T limit curves are determined such that, for given heatup/cool-down rate, pressures are lower than the pre-defined values at certain temperature to assure sufficient margin against brittle fracture. Lower-bound fracture toughness curve, K_{R} curve determined as a function of RT_{NDT} is used for the calculation. Additional limitation on pressure is imposed by low-temperature overpressure protection (LTOP) system at lower temperature part of P-T limit curve to prevent accidental violation of pressure limit during transient conditions. Therefore, available operating windows during heatup/cool-down processes are squeezed down to the narrow region between the P-T limit curves with LTOP system and pump cavitation curve. As the radiation embrittlement progresses, the operation window becomes narrower and narrower, causing difficulties for operators to heatup/cool-down the reactor.

In summary, radiation embrittlement of the RPV materials has significant impacts on the safety and operability of the reactor. As the radiation embrittlement of the RPV materials will certainly progress during the continued operation period beyond the current license period, a comprehensive program is needed to attain the integrity of the RPV throughout the life of the nuclear power plants.

3. Status of Kori Unit 1 RPV

3.1 Design and Fabrication

The RPV of Kori Unit 1 is one of the typical Westinghouse 2-loop design and fabricated by B&W, with inner diameter of 132 inches and thickness of 6.5 inches. The schematic of the RPV is shown in Fig. 2. Its cylindrical shells were made of SA 508 Cl. 2 ring forging internally clad with stainless steel 308 type weld. There are three circumferential welds near the reactor core, those are WF259, WF232/233, and WF267. The chemical compositions of the welds are summarized in table 1.

3.2 Characteristic of Welds

Like many of the early RPVs fabricated by B&W, linde 80 flux was used in the beltline region welds of Kori Unit 1 RPV. The WF232/233 weld close to the midplane of the core is consisted of two weld materials. The inner portion of the weld is WF232 which contains less copper and nickel, and the outer portion of the weld is WF233. In surveillance program, more susceptible WF233 is included as the limiting material.

The concern over the high copper content in the welds made using linde 80 flux and its effects on radiation embrittlement prompted the extensive reanalysis of the weld chemistry[6]. B&W published a comprehensive chemistry analysis report on 177-FA type RPV with linde 80 flux welds and recommended to use the reanalyzed chemistry instead of fabrication data. The best-estimate chemistry of 0.29% copper and 0.68% nickel for WF233 weld, suggested in the report, has been used in RT_{PTS} calculation and subsequent plant specific PTS evaluation of Kori Unit 1 RPV[7].

3.2 Radiation Embrittlement of Beltline Weld

The fracture toughness of RPV materials shows strong temperature dependency. From low temperature to high temperature, the lower-shelf energy region, transition region, and the upper-shelf energy region are defined. Reference temperature-nil ductility transition temperature (RT_{NDT}) is the conceptual threshold temperature below which the material shows full brittle fracture. It is determined according to ASME NB-2331 in which the initial RT_{NDT} be the higher of the nil-ductility transition

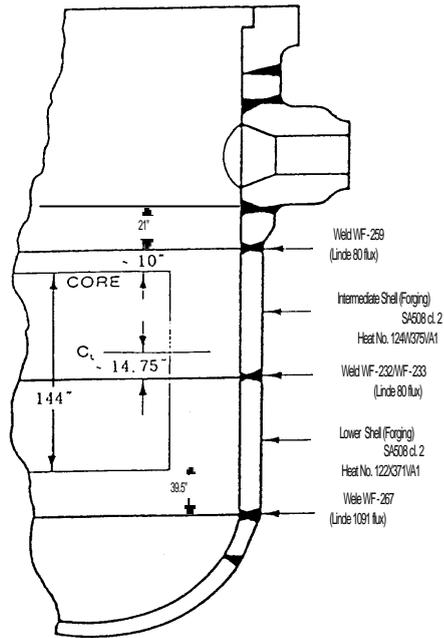


Figure 2. Schematics of Kori Unit 1 RPV and materials of construction

temperature (NDTT) from drop weight test or $33.3 \text{ } (60 \text{ })$ below the index temperature for 68 J (50 ft-lb) of absorbed energy in Charpy impact test. The intent of the NB-2331 is to define the conservative reference temperature in assessing fracture toughness of RPV materials. For WF233 weld of Kori Unit 1 RPV, NDTT was measured as $-28.9 \text{ } ,$ and the index temperature for 68 J of absorbed energy was measure as $10 \text{ } .$ Then the initial RT_{NDT} was defined as -23.3 which is the higher of -28.9 and $10 \text{ } - 33.3 = -23.3 \text{ } [8].$

In surveillance capsules, forging materials, weld, heat affected zone materials are included. Four out of six capsules have been withdrawn and tested already. The results of the surveillance test are summarized in table 2. As shown in table 2, the reduction of USE and the increase in RT_{NDT} have been progressed considerably. Especially, USE of WF233 weld has been below the minimum requirement of 68 J since the first surveillance test. RT_{PTS} was also projected to exceed the screening criteria of $148.9 \text{ } (300 \text{ })$ at about 27EFPY[1].

Table 1. Composition and initial RT_{NDT} s of welds near the beltline region.

Weld ID	WF-259	WF-232	WF-233	WF-267
Location	Nozzle Shell/ Inter. Shell	Inter. Shell/ Low. Shell(ID)	Inter. Shell/ Low. Shell(OD)	Lower Shell/ Lower Head
Filler wire heat no.	T29744	8T3914	T29744	T49544
Flux type & lot no.	Linde80, lot 8806	Linde80, lot 8790	Linde80, lot 8790	Linde 0091, lot3490
Cu in Weld Qual. Test	0.21%	0.14%	0.23%	0.24%
Ni in Weld Qual. Test	0.66%	0.69%	0.61%	0.52%
Cu in BAW-1799 [6]	-	0.18% (retest)	0.29%	-
Ni in BAW-1799 [6]	-	0.64% (retest)	0.68%	-
initial RT_{NDT}	-20.6 (generic)	-20.6 (generic)	-23.3 (measured)	-48.9 (generic)

Table 2. Summary of surveillance tests of circumferential weld in beltline region

Capsule	Fluence (10^{19} n/cm ²)	Shift in RT _{NDT} ,	Adjusted RT _{NDT} ,	Upper shelf energy, J
Unirradiated	-	-	-23.3*	90.2
V	0.417	106.1	113.9	65.0
T	1.072	103.9	111.7	56.9
S	1.399	115.6	123.4	63.3
R	3.002	131.1	152.2	54.6

4. Analysis Performed

4.1 Low Upper Shelf Toughness Evaluation

Since the very first surveillance test, USE of the beltline weld, or WF233 had been below the App. G minimum requirement of 68 J. App. G requires to perform equivalent margin analysis as well as full volumetric inspection of beltline region of the RPV and fracture toughness testing. Fig. 3 shows the fracture toughness test results of the specimens whose fluence levels are equivalent to 42 operating years. As shown in the figure, crack-resistance curve of Kori Unit 1 surveillance specimen is greater than the characteristic J-R curves of linde 80 welds suggested in regulatory guide[9]. Then the characteristic J-R curves of linde 80 welds, which is more conservative, are compared with the applied J for level A/B service loading conditions. From Fig. 4, it was concluded that there is sufficient margin against brittle fracture even though the USE was about 54.6 J.

4.2 Pressurized Thermal Shock Evaluation

As shown in Fig. 5, the RT_{PTS} of WF233 weld has been projected to exceed the screening criteria around 27 EFPY, or 34 operating years. USNRC PTS rule specifies that if the estimated RT_{PTS} are expected to exceed the screening criteria before the end of life, a plant-specific PTS analysis should be performed to demonstrate that total frequency of through-wall cracking (TWC) due to PTS is less

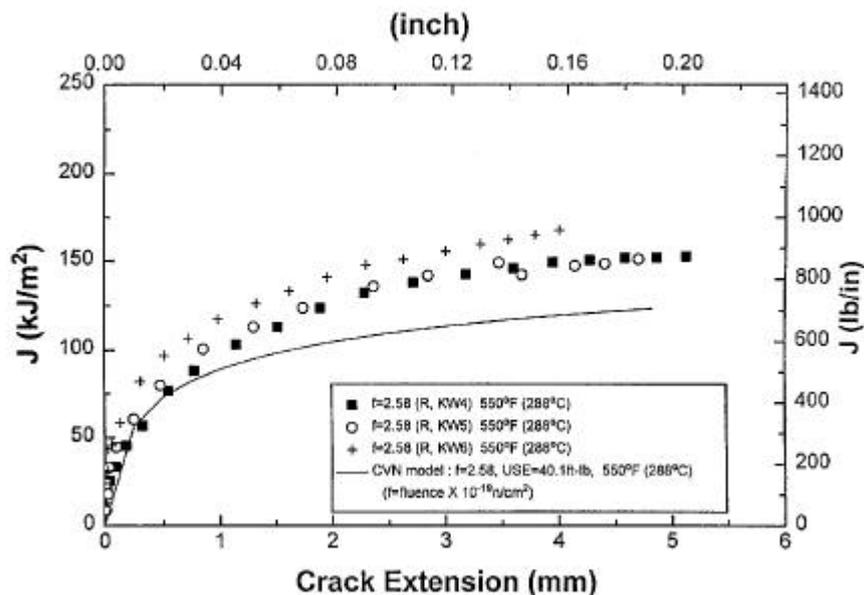


Figure 3. J-R curves of WF233 welds and CVN model in RG 1.161.

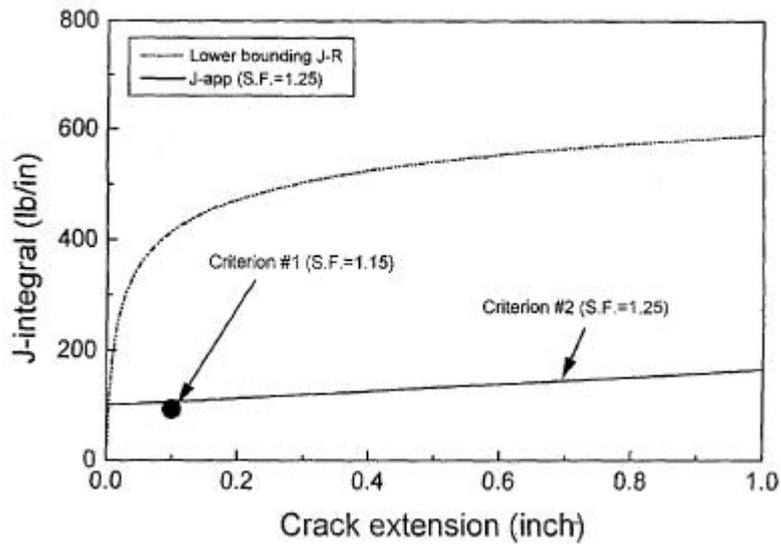


Figure 4. Crack initiation and instability analysis for level A/B service loading conditions.

than 5×10^{-6} per reactor-year for continued operation[5].

The overall flow of the plant-specific PTS analysis of the specific NPP is shown in Fig. 6[10]. First, PTS initiating events are identified and event-trees are 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 based on the similarity in thermal-hydraulic (T/H) nature and frequency of the sequence. For the selected transient sequences, T/H analyses are performed using transient analysis codes, such as RELAP5 and RETRAN. If thermal stratification within the cold leg is suspected, detailed mixing analyses are needed to obtain localized temperature near RPV wall in downcomer region.

As shown in Fig. 6, downcomer pressure, fluid temperature near RPV wall, and heat transfer coefficient obtained from T/H and mixing analyses are provided as inputs to probabilistic fracture mechanics(PFM) analyses. The specific vessel data, such as, physical material properties, geometry, and surveillance capsule data et. al. are needed also. Through the PFM analysis, conditional TWC probability, $P(F/E)$ for each transient sequence is calculated. TWC frequency at the event of specific PTS sequence is calculated by multiplying the sequence frequency and $P(F/E)$. Finally, the total TWC frequency is found by simply adding the vessel failure frequencies of all transient sequences analyzed. The integrated PTS risk calculated by the above procedure are compared with the limit specified in Reg. Guide 1.154, that is, 5.0×10^{-6} per reactor-year to determine the integrity of the RPV at the events of potential PTS transients. As shown in Fig. 7, the through wall cracking probability of Kori Unit 1 RPV due to PTS events are estimated as less than 5.0×10^{-6} per reactor-year even after 46.4 EFPY, equivalent to 60 operating years. Through the detailed analysis, it is now expected that RPV can maintain enough safety margin against pressurized thermal shock during its design life and extended operation period.

4.3 Redefine Initial RT_{NDT}

Projection of RT_{PTS} for WF233 RPV Beltline Weld of Kori-1
 Reg. Guide 1.99 Rev.2, Position 2.1, using surveillance data

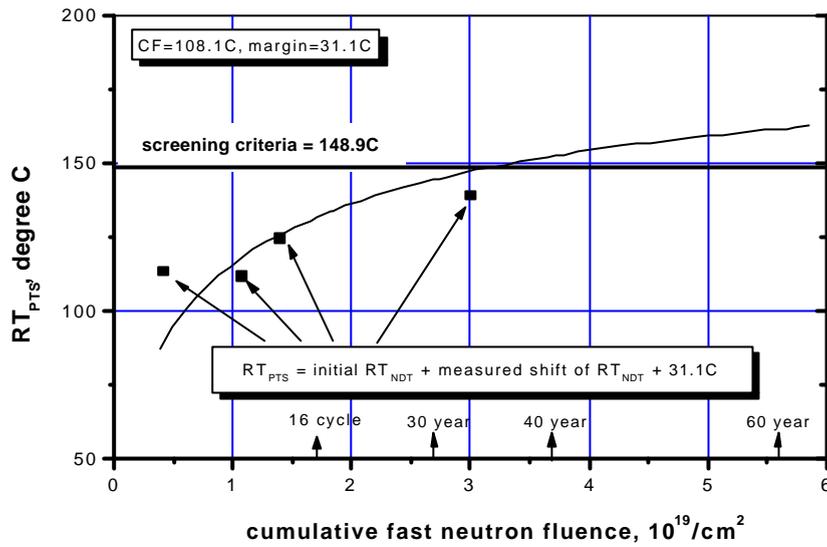


Figure 5. The projected RT_{PTS} of the beltline weld material.

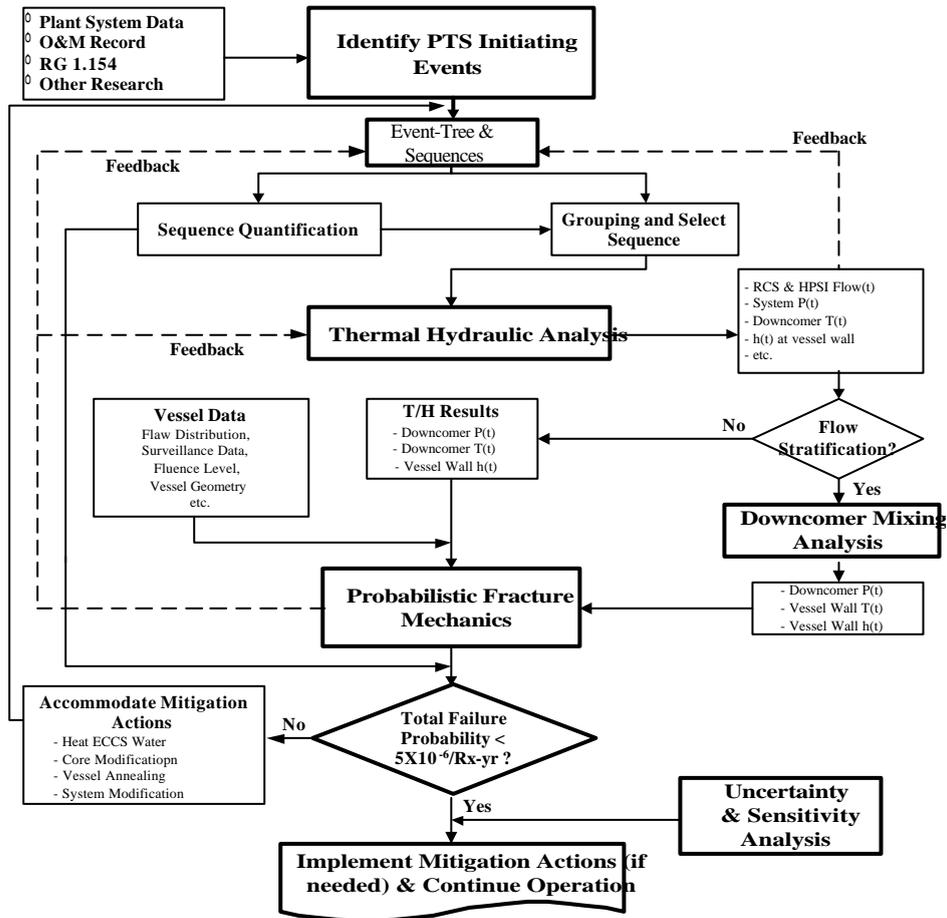


Figure 6. Overall flow of the plant-specific PTS analysis

As previously mentioned, initial RT_{NDT} of WF233 weld, -23.3 was determined by drop weight test and Charpy impact test according to NB-2331. However, the initial RT_{NDT} defined by NB-2331 methods showed considerable variation within the same group of materials. Recent progress in fracture mechanics testing, especially the development of Master-Curve method[11], opened the way that by using a small number of Charpy-like specimens, taken from the archive materials, a less conservative initial RT_{NDT} can be redefined. After the formal issuance of standard test methods by ASTM[12] and ASME Code Case-629[13], a testing program is underway to redefine the initial RT_{NDT} of WF233 weld of Kori Unit 1 RPV.

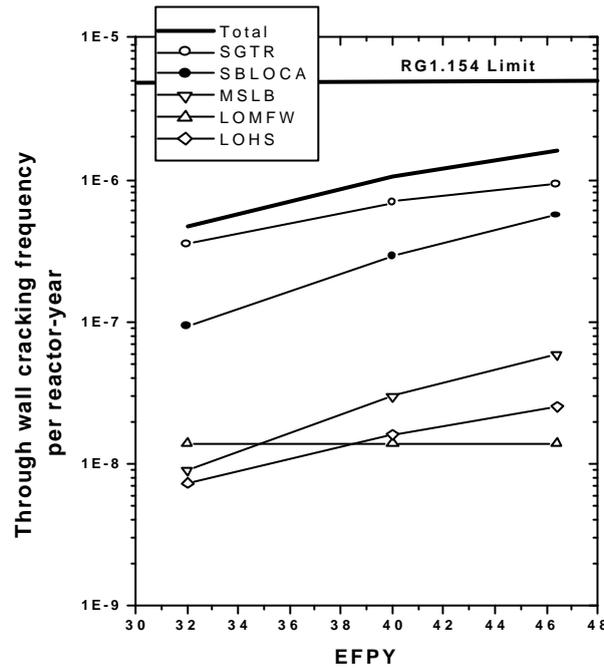


Figure 7. Calculated through-wall-cracking frequency

The fracture toughness test results using pre-cracked Charpy specimens are shown in Fig. 8. As shown in the figure, mean T_0 of WF233 weld is -83.3 , and statistically conservative (at 99% confidence level) lower bound T_0 is estimated as -65 . From this, RT_{T_0} is determined as $T_0 + 19.4 = -45.6$. However, based on the Charpy test results, chemistry analysis results, and microstructural analysis, the fracture characteristics of WF233 is considered to be worse than the average line 80 welds but better than WF70 (which is thought to have the worst fracture characteristics among line 80 welds). Because of the limitation of fracture toughness test database, it is currently under consideration that using -32.2 (which is the lower bound initial RT_{NDT} of line 80 welds and approved by USNRC for Zion Unit 1) as the initial RT_{NDT} of WF233 weld rather than the lower bound RT_{T_0} of -45.6 for the sake of conservatism.

5. Additional Activities

5.1 Ex-Vessel Dosimetry

Within the surveillance capsules, several dosimeter materials are included to measure the fast

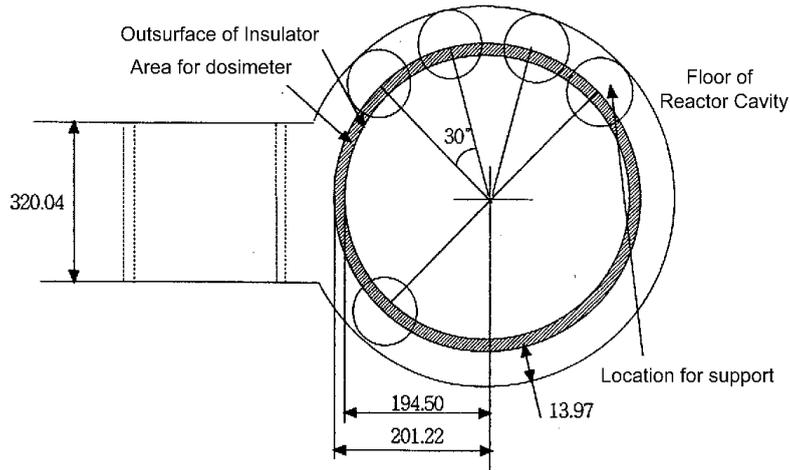


Figure 9. Schematics of the locations for ex-vessel dosimeter installation

be discussed in here. First, reactor core can be modified to reduce the fast neutron flux at vessel inner wall. This can be achieved either by installing shielding materials around the area of high fast neutron flux, or by modifying fuel loading pattern with which less fast neutron can escape from core into the vessel cavity. Of the two, low leakage loading pattern known as L³P had been already adopted in Kori Unit 1 since the 4-th fuel cycle. Even lower leakage loading pattern, such as low-low leakage loading pattern (L⁴P) seems possible. Though, applicability of such loading pattern should be carefully reviewed by comparing the benefit of slower radiation embrittlement and the cost of fuel economics and the effects of increased peaking factor.

The second option is vessel annealing to recover the fracture toughness of RPV materials[15]. Kori Unit 1 seems to have a special advantage over other plants in that there are only one circumferential weld in the core beltline region, located more than a foot below the core mid-plane. This would minimize the potential heating zone and, consequently, thermal stress around nozzle area located far away from the WF233 weld. However, the application of thermal annealing of the RPV of Kori Unit 1 seems not an imminent issue at this time considering the verification of structural integrity at the event of PTS.

Only realistic concern on the RPV during the period of continued operation is the shrinkage of operating window during heatup/cooldown process. However, recent trend of reevaluating the excessive conservatism in P-T Limit curves would provide widened operating window. Nonetheless, vessel annealing seems to be the promising option to solve the problems of low upper shelf energy, PTS, and shrinking P-T limit, all at once.

6. Summary

The various aspects of RPV integrity during the continued operation and their inter-relationship are reviewed. The results of detailed integrity analyses, such as low upper shelf toughness evaluation and pressurized thermal shock evaluation of Kori Unit 1 RPV are briefly summarized. Despite of the radiation embrittlement, it has been concluded that there would be enough margin of safety to ensure the integrity, even after 40 years.

As integral parts of PLIM Phase II Program, several activities to attain the integrity of the

embrittled reactor pressure vessel, such as, redefining initial RT_{NDT} , ex-vessel dosimetry, and early withdrawal and testing of surveillance capsule are explained. It is expected that the aging effects of radiation embrittlement are manageable during the period of continued operation.

The applicability of low-low leakage loading pattern to reduce the neutron flux at the RPV and thermal annealing to recover most of the radiation embrittlement are also reviewed. Though, the possibility of application of thermal annealing in the near future is low, the effectiveness of it is so evident that due consideration should be given in the long-term aging management option. A comprehensive aging management program covering all the issues mentioned in this paper is being developed to attain the integrity of the RPV during the lifetime.

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