Reference Temperature for Pressurized Thermal Shock of Kori Unit 1

C. S. Yoo,a B. C. Km,a K. O. Chang,a S. L. Lee,a W. S. Gong,a K. J. Choi,a J. H. Park b
a Korea Atomic Energy Research Institute, 150 Dukjin-dong, Yuseung-gu, Daejeon 305-353, Korea, csyoo@kaeri.re.kr
b Chungnam National University, 220 Gung-dong, Yuseung-gu, Daejeon 305-764, Korea

1. Introduction

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts in the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation.

The requirements of PTS Rule[1], 10 CFR Part 50.61, consist of the following:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of Reference Temperature for PTS (RT_{PTS}), accepted by the NRC, for each reactor vessel beltline material for the end of life (EOL) fluence of the material.
- The RT_{PTS} screening criterion values for the beltline region are 270ºF for plates, forgings, and axial weld materials, and 300ºF for circumferential weld materials.

In this paper evaluations of RT_{PTS} in the weld metal which is the most limiting material for Kori Nuclear Unit 1 (Kori-1) pressure vessel were performed with respect to the various core loading patterns to be considered in the future cycles.

2. Methods

2.1 Reference Temperature for PTS (RT_{PTS})

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Eq.1 must be used to calculate values of RT_{PTS} for each weld and plate or forging in the reactor vessel beltline[1].

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \tag{1}$$

where $RT_{NDT(U)}$ is the reference temperature for a reactor vessel material in the pre-service or unirradiated condition, and M is margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M is evaluated from Eq.2.

$$\mathrm{M} = \sqrt{\sigma_U^2 + \sigma_\Delta^2} \tag{2}$$

In the above equation σ_U and σ_Δ are the standard deviations for $RT_{NDT(U)}$ and ΔRT_{PTS}, respectively. The $RT_{NDT(U)}$ and M in Eq.1 have been estimated as -10ºF and 56ºF, respectively for weld metal of Kori-1 pressure vessel from the five surveillance capsule evaluations[2].

ΔRT_{PTS} in Eq.1 is the mean value of the transition temperature shift, or change in RT_{PTS}, due to irradiation, and must be calculated using Eq.3.

$$\Delta RT_{PTS} = (CF) * f^{(0.28-0.10\log f)} \tag{3}$$

f is the best estimate neutron fluence, in unit of 10^{19} n/cm^2 (E>1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The fluence values at the various operating time can be calculated by neutron transport calculations described in the next section. CF(ºF) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF as Eq.4.

$$CF = \frac{\sum_{i=1}^{n}[A_i \times f_i^{(0.28-0.10\log f_i)}]}{\sum_{i=1}^{n}[f_i^{(0.56-0.20\log f_i)}]} \tag{4}$$

where A_i is the measured value of ΔRT_{PTS} and f_i is the fast (E>1.0 MeV) neutron fluence for each surveillance data point. It is noted that the CF of weld metal for Kori-1 pressure vessel have been evaluated as 190.76ºF in Reference 2.

2.2 Neutron Transport Calculations

Plant specific forward transport calculations were carried out using the three-dimensional flux synthesis technique described in Regulatory Guide 1.190[3] as below:

$$\phi(r,\theta,z) = \phi(r,\theta) \bullet \frac{\phi(r,z)}{\phi(r)} \tag{5}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation. For Kori-1 analysis all transport calculations were carried out using the DORT 3.1 discrete ordinate code[4] and the BUGLE-96 cross-section library[5]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. The fuel assembly specific enrichment and burnup data were used to generate the spatially dependent neutron source throughout the reactor core. This source description included the spatial variation of isotope dependent (^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) fission spectra, neutron emission rate per fission, and energy release per fission based on the burnup history of individual fuel assemblies.

Nuclear Design Reports for cycle 1 through cycle 22 were used to obtain the assembly-wise enrichments, burnup, and core average axial power distributions for each burnup point. Accumulated operating times for each cycle corresponding to 100% reactor power were obtained from site power history files.

3. Results and Conclusion

RT_{PTS} for Kori-1 reactor vessel weld metal were estimated using the fast (E > 1.0 MeV) neutron fluences calculated using operation history and neutron flux from transport calculations illustrated in section 2.2 of this paper for cycle 1 through cycle 22. Note that cycle 23 is being operated at this point. RT_{PTS} beyond cycle 22 were also projected on the basis of three different kinds of core loading patterns (LPs) to be considered for Kori-1 nuclear design. The first one is "high leakage LP" where the fresh fuel is loaded at the core periphery of cardinal axes, which is like cycle 13. The second one is "medium leakage LP" where once burned fuel is loaded there, which is like cycle 19. The last one is "low leakage LP" where twice burned fuel is loaded there, which is like cycle 22. Note that the fast neutron irradiation on the reactor vessel is highly dependent on the core loading pattern.

RT_{PTS} projections evaluated with respect to Effective Full Power Year (EFPY) for the three different LPs are shown in Figure 1. This figure shows that the reactor can be operated 2.4 EFPY more (about 3 cycles) by using low leakage LP compared to high leakage LP if the RT_{PTS} screening criterion is 300 ºF. By the same way, the operating life time can be extended about 6 cycles more when we maintain low leakage LP for the future cycles if the RT_{PTS} screening criterion is 310 ºF.

The fluences used in this paper are calculated values obtained only from the transport calculations, which are in general more conservative than the best estimate values. Best estimate values of the reactor vessel fluence are obtained from the In-vessel and/or Ex-vessel neutron dosimetry evaluations. For Kori-1 reactor the 1st Ex-vessel neutron dosimetry evaluation is being performed by KAERI. More accurate RT_{PTS} projections for the future cycles up to EOL can be possible using the results of Ex-vessel neutron dosimetry evaluation.

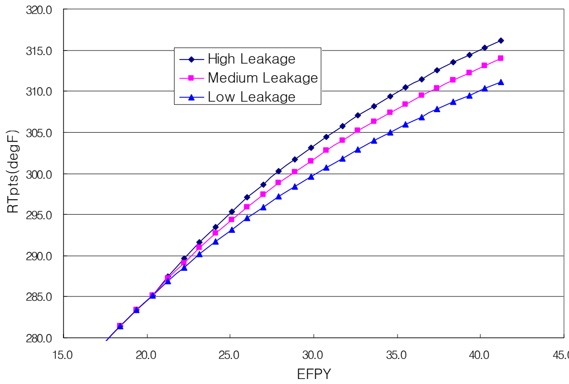

Figure 1. RT_{PTS} projections for Kori-1 pressure vessel weld metal with various core loading patterns

REFERENCES

[1]10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume60, No.243, December 19, 1995, effective January 18, 1996.

[2]KAERI-ST-K1-PT-008-1/02, "Report for the Pressure-Temperature Limit Curves of Kori Nuclear Power Plant Unit 1 Using Code Case N-588," July 2004.

[3]USNRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001

[4]RSIC Computer Code Collection CCC-650, "DOORS 3.1, One-Two and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

[5]RSIC Data Library Collection DLC-185, "BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.