Evaluation of Kori Unit 1 Reference Temperature for PTS

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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 beltline region of a reactor vessel where a reduced fracture resistance exists due to neutron irradiation.

The Nuclear Regulatory Commission (NRC) amended its regulations for light water cooled nuclear power plants to clarify several items related to the future toughness requirement for reactor pressure vessels, including pressurized thermal shock requirements. The revised PTS rule[1], 10CFR Part 50.61, was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

The PTS requirements 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 assessment of RT_{PTS} must use the calculation procedure given in the PTS Rule, and must specify the bases for the projected value of RT_{PTS} for each vessel beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.
- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewal term, if applicable for the plant.
- The RT_{PTS} screening criterion values for the beltline regions are:

  270˚F for plate, forgings, and axial weld materials, and
  300˚F for circumferential weld materials.

In this paper, the fast neutron fluence was determined by the neutron transport calculations for Kori Unit 1. And the RT_{PTS} value for the end of life of the plant was projected using the fast neutron fluence obtained by neutron transport calculations according to the various core loading pattern possible for the future cycles.

2. Methods

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{PTS} for each weld and plate or forging in the reactor vessel beltline [1,2].

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

where

$RT_{NDT(U)}$ = reference temperature for a reactor vessel material in the unirradiated condition,

M = margin to be added to account for uncertainties in the value of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures.

ΔRT_{PTS} is the mean value of the transition temperature shift, or change in RT_{PTS}, due to irradiation, and must be calculated using Equation 2.

$$\Delta RT_{PTS} = (CF) \cdot f^{(0.28-0.10\log f)} \qquad (2)$$

CF (˚F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Table 1 and 2 of the PTS Rule (10CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 3.

$$CF = \frac{\Sigma[A_i \cdot f_i^{(0.28-0.10\log f_i)}]}{\Sigma[f_i^{0.56-0.20\log f_i}]} \qquad (3)$$

In Equation 3, A_i is the measured value of ΔRT_{PTS} and f_i is the fluence for each surveillance data point. CF was evaluated as 190.76˚F[3] for the vessel weld which is the leading material of the Kori Unit 1 and thus the ΔRT_{PTS} of Equation 2 is determined by the EOL fast (E>1.0MeV) neutron fluence, f, which is the best estimate value projected up to the end of plant life. The fast neutron fluence on the inside surface of the vessel is determined by the full power operation time and fast neutron flux at that position. Basically the fast neutron flux of this position is highly depends on the relative powers of the core periphery assemblies located on the main axis of the reactor core. Relative assembly power distribution of the reactor core is determined by the fuel management strategy and physics design, for example, if the fresh fuel assemblies are loaded at the core periphery location, the neutron leakage will be increased and thus the fast neutron flux will be also increased on the pressure vessel. If the twice burned fuel assemblies are loaded at the same position, the fast neuron flux will be severely decreased and thus the RT_{PTS} projection will be increasing slowly. If a strong neutron absorbing material such as Hafnium is loaded at the assembly guide tubes for the core periphery assemblies, the relative power of these assemblies are more suppressed and thus the fast neutron flux on the pressure vessel will be highly reduced. Note that the neutron absorbing material is only located at the mid region of the active core height because the relative powers of the core top and bottom regions are always small. Figure 1 shows the comparisons of the axial power distributions for the core periphery assembly with and without neutron absorbing material (Hafnium) loaded in the guide tubes. As shown in this figure, the axial power distribution of the location where the neutron absorbing material is loaded is highly suppressed.

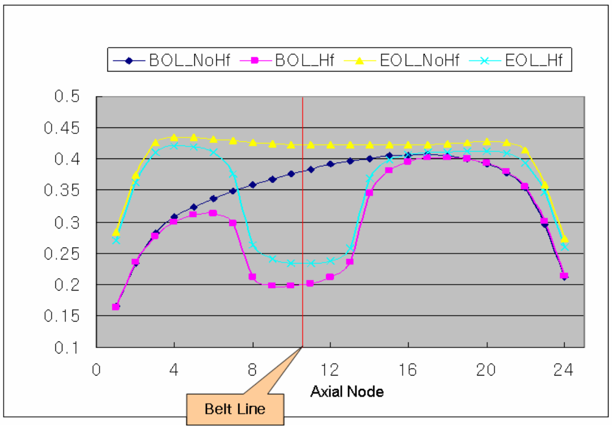

Figure 1. Axial Power Distribution Comparison with and without Hafnium Rods

3. Results and Conclusion

RT_{PTS} projections for the Korea Nuclear Unit 1 reactor vessel weld metal were estimated using the fast (E > 1.0 MeV) neutron fluences calculated for the various fuel management strategies. The first strategy is "high leakage loading pattern (LP)" where the fresh fuel is loaded at the core periphery of cardinal axes. The second one is "medium leakage LP" where once burned fuel is loaded there and the third one is "low leakage LP" where twice burned fuel is loaded there. The last one is "neutron reduction program" where the strong neutron absorbing material (Hafnium) is loaded in the guide tubes of the interested assemblies which is also twice burned fuels. Figure 2 shows the RT_{PTS} projections for the various fuel management strategies.

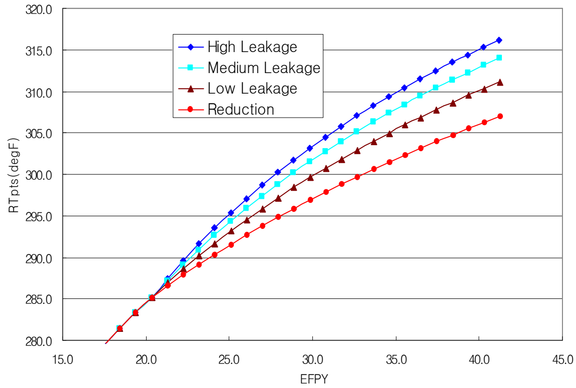

Figure 2. RT_{PTS} Projections for the Various Fuel Management Strategies

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] Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," USNRC, May 1988.

[3] 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," KAERI, July 2004.