Fast Neutron Fluence of Reactor Vessel Nozzles for Generation of Pressure-Temperature Limit Curve

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1. Introduction

The purpose of the Pressure – Temperature (P-T) limit curve is to prevent a failure of reactor pressure vessels during operation of the reactor coolant system. In Korea, a P-T limit curve shall meet 10CFR50 Appendix G[1] according to Nuclear Safety and Security Commission Notification 2017-20. 10CFR50, Appendix G requires a plant specific P-T limit curve for the beltline region of the reactor vessel. And the beltline region is defined as the region of the reactor vessel that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel expected to experience sufficient neutron radiation damage. Therefore, the P-T limit curves have been traditionally evaluated based on the beltline region which is the most affected by neutron irradiation embrittlement. However, due to the geometric discontinuity, the inside corner regions of the vessel nozzles are the most highly stressed regions of the reactor vessel. The most highly stressed nozzle region may result in more limiting P-T limit curves than beltline P-T limit curves. Thus, the evaluation of nozzle P-T limit curves should be evaluated and compared to the beltline P-T curves.

In 2014, the NRC issued Regulatory Issue Summary (RIS) 2014-11[2], which required the consideration of reactor vessel nozzles in generation of P-T limits curve. Reactor vessel nozzles and other discontinuities have complex geometries that can exhibit significantly higher stresses than the reactor vessel beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference nil ductility transition temperatures (RT_{NDT}) for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries. And the fast neutron fluence data is needed to calculate RT_{NDT}.

The purpose of this paper is to evaluate end of life (EOL) fast neutron fluence (E>1MeV) value of the reactor vessel nozzle for Kori Unit 3. In order to do that, 3D synthesis method of Regulatory Guide 1.190[3] was used with DORT transport code[4]. Cycle specific radial and axial power distributions and power history were also used.

2. Methods

2.1 Synthesized Neutron Flux

In performing the fast neutron fluence evaluations, plant specific forward transport calculations were carried out using the three-dimensional flux synthesis technique described in Regulatory Guide 1.190. The following synthesis approach was employed for all fuel cycles:

$$\phi(r,\theta,z,E)=\phi(r,\theta,E)\cdot\frac{\phi(r,z,E)}{\phi(r,E)}. \quad (1)$$

Where $\phi(r,\theta,z,E)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta,E)$ is the transport solution in R-θ geometry, $\phi(r,z,E)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r,E)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as used in the R-θ two-dimensional calculation.

All of the transport calculations were carried out using the DORT two-dimensional discrete ordinates code Version 3.1 [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 LWR 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.

Fig. 1 shows a plan view of the R-θ model of Kori Unit 3 reactor geometry at the core midplane. A single octant is depicted showing the arrangement of neutron pads and surveillance capsule attachments. In addition to the core, reactor internals, the pressure vessel and the primary biological shield are also included. In developing the R-θ analytical models of the reactor geometry shown in Figure 1, nominal design dimensions were employed for the various structural components. Water temperatures, coolant density in the reactor core and downcomer regions of the reactor were taken to be representative of full power operating conditions (2775 MW from cycle 1 to cycle 19, 2900 MW from cycle 20 to cycle 23). The reactor core was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

Fig. 2 shows a sectional view of the R-Z model of Kori Unit 3 reactor. The model extends radially from the centerline of the reactor core out to a location inside the primary biological shield. Axial mesh was developed from -300 cm from the core mid-plane to +265 cm, which includes lower extend of weld location between the nozzle and the vessel. By reviewing the

Kori Unit 3 reactor vessel drawings, the lower extended location of nozzle weld is estimated as +255 cm from the core mid-plane. Therefore, all fluence data were extracted at this location.

2.1 Synthesized Neutron Fluence

Using the synthesized neutron flux of section 2.1, three-dimensional fast neutron fluence can be calculated as follows:

$$f(r,\theta,z)=\sum_{i=1}^{n}[\int_{1MeV}^{\infty}\phi_i(r,\theta,z,E)dE]\Delta t_i\, , \quad (2)$$

where Δt_i is the full power operation time (sec) for the i[th] cycle and n is the last cycle of irradiation. $\phi_i(r,\theta,z,E)$ of the above equation can be obtained from equation (1) and Δt_i can be obtained from the reactor operation history database for each fuel cycle.

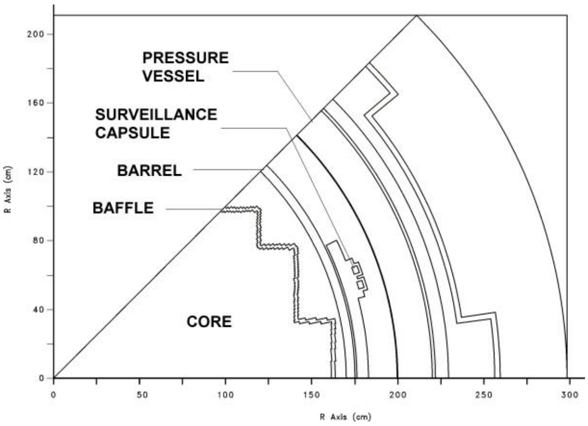

Fig. 1. Kori Unit 3 R-θ Geometry for Neutron Transport Calculations

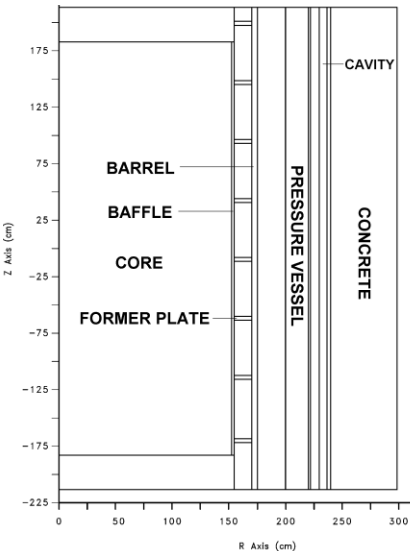

Fig. 2. Kori Unit 3 R-Z Geometry for Neutron Transport Calculations

3. Results and Conclusions

The axial profile of fast neutron fluence at the end of cycle 23 for the azimuthal angle 0 degree and vessel inner radius (IR) is shown in Fig. 3. As mentioned above the lower extended nozzle weld is +255 cm from core mid-plane. Therefore, the fast neutron fluence of this location is about 1.1E+17n/cm^2.

Fig. 4 shows the fast neutron fluence projection for the nozzle location from the end of cycle 23 (25.77 EFPY) up to 40 EFPY. In this calculation, the average fast neutron flux from cycle 20 to cycle 23 was assumed for future operation. As shown in Fig. 4, the fast neutron fluence for nozzle location at 32 EFPY is about 1.25E+17n/cm^2.

According to 10CFR50 Appendix H[6], the embrittlement effect due to neutron irradiation should be considered if the fast neutron fluence is greater than 1.0E+17n/cm^2 at the end of life of the plant. Therefore, the nozzle material embrittlement effect should be considered for P-T limit curve generation for Kori Unit 3.

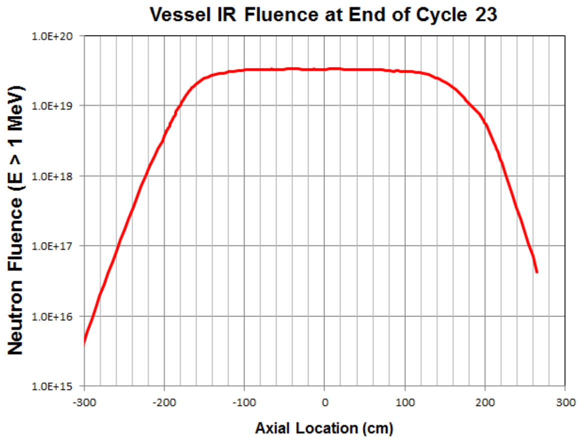

Fig. 3. Kori Unit 3 Fast Neutron Fluence Profile at the End of Cycle 23

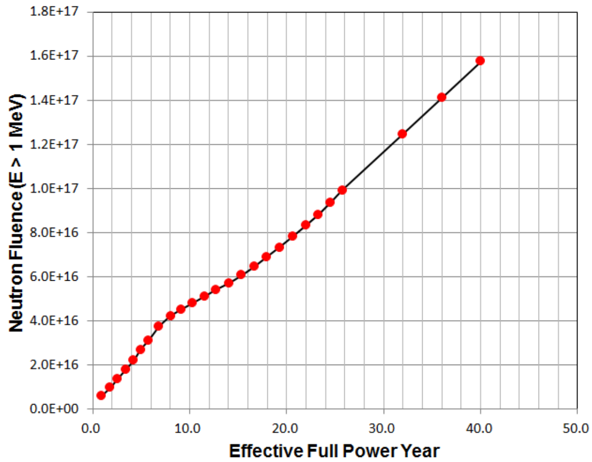

Fig. 4. Kori Unit 3 Fast Neutron Fluence Projection for Nozzle Weld Location

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