Evaluation of the Neutron Fluence at a Baffle-Former Zone in an Operating Reactor

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1. Introduction

Neutron fluence evaluation has been performed on a reactor vessel in an operating nuclear power plant in order to evaluate the radiation embrittlement which is directly related to plant safety as well as a plant operating license, based on the operating history. Because, as the operating years increase, damage may occur in the internal structures such as a baffle former bolt due to various reasons, and one of these reasons comes from the neutron fluence, so called an irradiation assisted stress corrosion cracking, thus resulting in the shutdown of a plant and the replacement of a structure which has an economic disadvantage as well as a severe effect in the integrity of a plant. Neutron flux and fluence calculations for the baffle area for one of the reactors operating in Korea have been performed for all the operating cycles from the start of the reactor using real plant operating conditions such as the operating temperature, pressure and fuel loading pattern in order to evaluate any possibility that may cause a stress corrosion cracking due to the excessive neutron irradiation.

2. Evaluation Procedure

Plant specific forward transport calculations were carried out using the three-dimensional flux synthesis technique described in USNRC Regulatory Guide 1.190[1] as below:

$$f(r,q,z) = f(r,q) \bullet \frac{f(r,z)}{f(r)} \qquad (1)$$

where $f(r,q,z)$ is the synthesized three-dimensional neutron flux distribution, $f(r,q)$ is the transport solution in the r,q geometry, $f(r,z)$ is the two-dimensional solution for a cylindrical reactor model using an actual power distribution, and $f(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,q two-dimensional calculation. All the transport calculations were carried out using the DORT 3.1 discrete ordinate code[2] and the BUGLE-96 cross-section library[3]. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for a light water reactor application. In these analyses, an anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of an angular quadrature. The specific fuel assembly enrichment and burn up 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, the neutron emission rate per fission, and the energy release rate per fission based on the burnup history of individual fuel assemblies.

Neutron dosimeters removed from the in-vessel capsules located at 17 and 20 degrees azimuthally and the ex-vessel capsules installed at 0, 15, 30, and 45 degrees azimuthally relative to the core major axes outside the reactor vessel have been analyzed to evaluate the neutron flux in the reactor vessel correctly.

The use of passive neutron sensors does not yield a direct measurement of the energy dependent neutron flux at the measurement location. Rather, the activation or fission process is a measure of the integrated effect where the time- and energy- dependent neutron flux irradiates the target materials during the corresponding reactor operation periods.

With the measurement of specific activities, the operating history of the reactor, and the physical characteristics of the neutron sensors, reaction rates referenced to a full power operation are determined. The reaction rate is a measurement value on the basis of a measured specific activity and reactor power history. On the other hand the neutron sensor reaction rates can also be derived by using the neutron spectrum from the transport calculation at the locations of dosimtry sensors and an appropriate cross-section library[4].

Three-dimensional neutron spectrum distributions representative of each fuel cycle at the location of a baffle to former area, and the flux and fluence were obtained from the above transport calculations using a synthesis technique based on the best estimate result for the reactor vessel.

3. Results and Conclusion

Flux and fluence distributions at the closest mesh to the reactor fuel within the baffle zone from 0 to 45 degrees in an azimuthal direction at a specific operating cycle are shown in Figures 1 and 2. We can see in Figure 1 that the neutron flux increases at an early stage of the operation and then decreases later on. When we look at the neutron fluence in Figure 2 which is the buildup according to the flux and the elapsed time, the accumulated neutron fluence will increase as the operating cycles accumulate. From the results shown in Figure 2, we can predict that the radiation effect will occur from around the middle of an operation when the peak fluence exceeds the threshold value, 1.0E+22 n/cm2, based on the description[5] that when the neutron fluence reaches the threshold value, the irradiation causes the generation of a stress corrosion cracking in an influenced zone of the internal structures in a commercial pressurized water reactor.

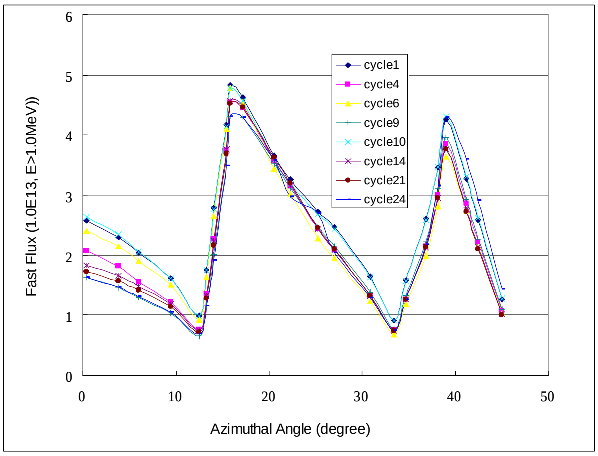

Figure1. Flux distribution at the closest mesh to the reactor fuel within the baffle zone from 0 to 45 degrees in the azimuthal direction

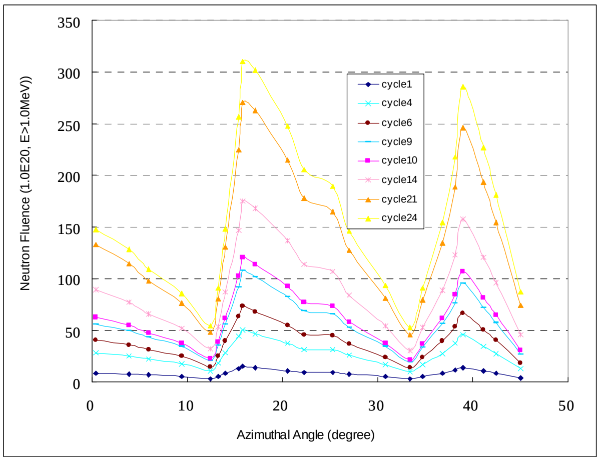

Figure2. Fluence distribution at the closest mesh to the reactor fuel within the baffle zone from 0 to 45 degrees in the azimuthal direction

Neutron flux and fluence described in Figures 1 and 2 are based on the transport calculation solution in the r,q geometry. However, $f(r,z)$ in the two-dimensional solution has to be achieved in order to complete synthesized three-dimensional neutron flux distribution $f(r,q,z)$. Figure 3 shows a single flux distribution in the axial direction. We can also see that neutron flux varies in the axial direction and also that the maximum flux does not occur in the center of the fuel height.

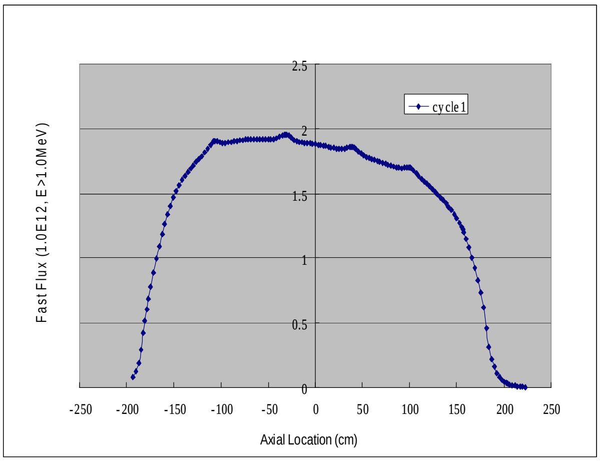

Figure3. Flux distribution for cycle 1 in the axial direction

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

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[4] RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium," July 1994.

[5] KINS/RR-269, "원자로 내부구조물 경년열화 평가," 한국원자력안전기술원 January 2005.