Fast Neutron Fluence of Yonggwang Nuclear Unit 1 Reactor Pressure Vessel

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Abstract

The Code of Federal Regulations, Title 10, Part 50, Appendix H, requires that the neutron dosimetry be present to monitor the reactor vessel throughout plant life. The Ex-Vessel Neutron Dosimetry System has been installed for Yonggwang Nuclear Unit 1 after complete withdrawal of all six in-vessel surveillance capsules. This system has been installed in the reactor cavity annulus in order to measure the fast neutron spectrum coming out through the reactor pressure vessel. Cycle specific neutron transport calculations were performed to obtain the energy dependent neutron flux throughout the reactor geometry including dosimetry positions. Comparisons between calculations and measurements were performed for the reaction rates of each dosimetry sensors and results show good agreements.

KEYWORDS: vessel, dosimetry, ex-vessel, embrittlement

1. Introduction

In the assessment of the state of embrittlement of light water reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of the beltline region of the vessel is required. In Appendix G to 10 CFR 50[1], the beltline region is defined as *"the region of the reactor vessel shell material (including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the reactor core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage"*. Therefore, plant specific exposure assessments must include evaluations as a function of axial and azimuthal location over the entire beltline region.

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Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence”[2] describes state-of-the-art calculation and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence. Also included in Regulatory Guide 1.190 is a discussion of the steps required to qualify and validate the methodology used to determine the neutron exposure of the pressure vessel wall. One important step in the validation process is the comparison of plant specific neutron calculations with available measurements.

The Code of Federal Regulations, Title 10, Part 50, Appendix H[3], requires that the neutron dosimetry be present to monitor the reactor vessel throughout plant life. The Ex-Vessel Neutron Dosimetry (EVND) program for Yonggwang Nuclear Unit 1 (YGN-1) was designed for continuous monitoring of reactor vessel exposures by fast neutron (E > 1 MeV) after complete withdrawal of all six in-vessel surveillance capsules which had been located on the thermal shield between the core and the reactor vessel in the downcomer region. Six EVND capsules had been installed in the various locations of reactor cavity annulus between the reactor vessel outer wall and the biological shield at the end of cycle 14 and were withdrawn at the end of cycle 15 for the evaluation.

Evaluations of EVNDs withdrawn at the end of cycle 15 and the fast neutron exposures on the pressure vessel beltline region of YGN-1 were performed on the guidance specified in Regulatory Guide 1.190. YGN-1 EVND program employs advanced sensor sets that consist of the radiometric sensors and gradient chains.

2. Methods

In this section some of the techniques used to determine the fast neutron flux and to evaluate the EVNDs are described.

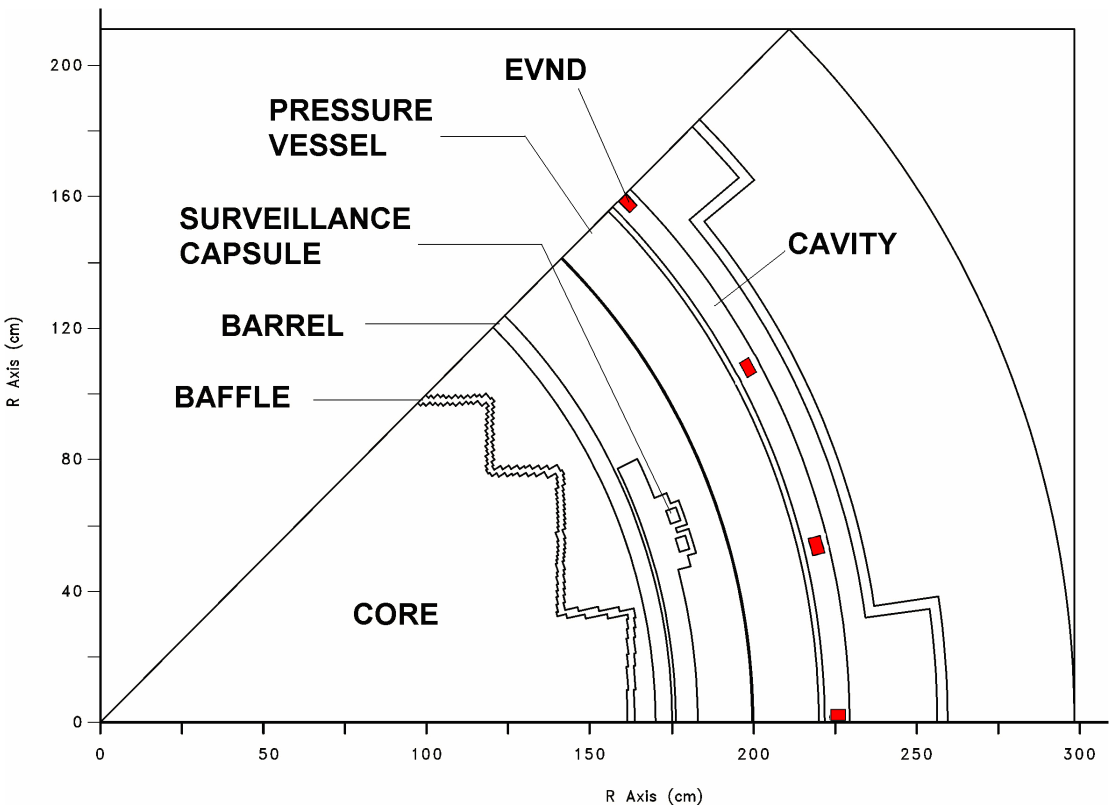
2.1 Neutron Transport Calculations

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

$$\phi(r,\theta,z) = \phi(r,\theta) \bullet \frac{\phi(r,z)}{\phi(r)} \tag{1}$$

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 YGN-1 analysis all transport calculations were carried out by 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.

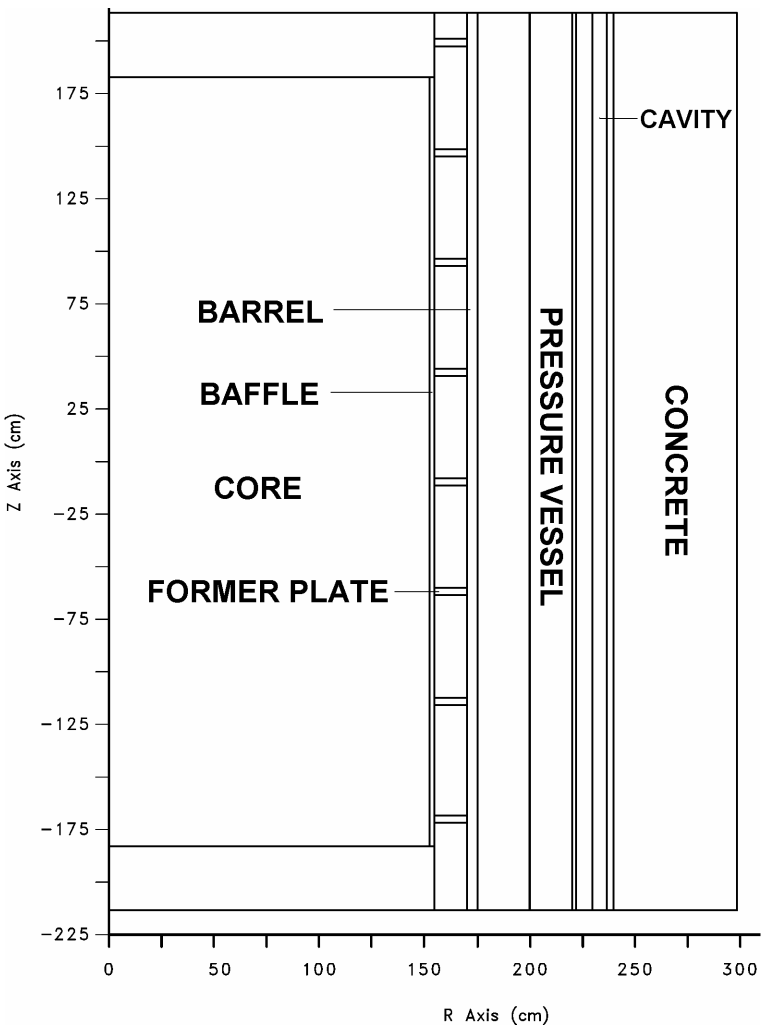
Figure 1: R-θ Geometry for Neutron Transport Calculations



Plan view of the R-θ model of YGN-1 reactor geometry at the core midplane is shown in Figure 1. A single octant is depicted showing the arrangement of baffle, core barrel, and surveillance capsule attachments. In addition to the core, reactor internals, pressure vessel and primary biological shield, the models developed for these octant geometries also included explicit representations of the surveillance capsules, the pressure vessel cladding, the pressure vessel reflective insulation, and the reactor cavity liner plate. As shown in this figure EVND sets are installed at the location of azimuthal angles of 0º, 15º, 30º, and 45º in the reactor cavity.

A section view of the R-Z model of YGN-1 reactor is shown in Figure 2. The model extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over a fourteen foot axial span from an elevation one foot below to one foot above the active fuel region. The axial extent of the model was chosen to permit the determination of the maximum exposure of vessel materials in the beltline region opposite the reactor core.

Figure 2: R-Z Geometry for Neutron Transport Calculations



Three-dimensional neutron spectrum distributions representative of cycle 15 of YGN-1 at the location of pressure vessel and EVND were obtained from the above transport calculations.

2.2 Neutron Dosimetry Evaluations

Total six EVND capsules containing seven neutron sensors listed in Table 1 had been installed at the various locations of reactor cavity. These locations correspond to azimuthal locations of 0º, 15º, 30º, and 45º relative to the core major axes. For the azimuthal 0º, three EVND capsules had been positioned at top, middle, and bottom of the actual core height and the others are located at the middle of core for each azimuthal angle. Stainless steel bead chains containing iron, nickel, and cobalt as radiometric monitors had been also positioned from top to bottom of the reactor core to monitor the axial gradients of flux distributions at each azimuthal angle of cavity.

The use of passive neutron sensors does not yield a direct measure 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 on the target materials during the corresponding reactor operation periods.

Having the measurement of specific activities, the operating history of the reactor, and the physical characteristics of the neutron sensors, reaction rates referenced to full power operation are determined from the following equation:

$$R = \frac{A}{N_0 F\, Y \sum_{j=1}^{n} \frac{P_j}{P_{ref}} C_j\, [1-e^{-\lambda t_j}]\, [e^{-\lambda t_d}]} \tag{2}$$

where,

R	=	reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus),
A	=	measured specific activity (dps/gm),
N_0	=	number of target element atoms per gram of sensor,
F	=	weight fraction of the target isotope in the sensor material,
Y	=	number of product atoms produced per reaction,
P_j	=	average core power level during irradiation period j (MW),
P_{ref}	=	maximum or reference core power level of the reactor (2775MW for Kori Unit 3),
C_j	=	calculated ratio of ϕ(E>1.0Mev) during irradiation period j to the time weighted average ϕ(E>1.0Mev) over the entire irradiation period,
λ	=	decay constant of the product isotope (1/sec),
t_j	=	length of irradiation period j (sec),
t_d	=	decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

Table 1: Sensors Used in Ex-vessel Neutron Dosimetry Sets

Material	Reaction	Product Half-Life
Copper	$^{63}Cu(n,\alpha)^{60}Co$	5.271 yr
Titanium	$^{46}Ti(n,p)^{46}Sc$	83.79 dy
Iron	$^{54}Fe(n,p)^{54}Mn$	312.3 dy
Nickel	$^{58}Ni(n,p)^{58}Co$	70.82 dy
^{238}U	$^{238}U(n,f)^{137}Cs$	30.07 yr
^{237}Np	$^{237}Np(n,f)^{137}Cs$	30.07 yr
Cobalt-Al	$^{59}Co(n,\gamma)^{60}Co$	5.271 yr

2.3 Reaction Rates from Transport Calculations

The reaction rate derived from Eq.2 is the measurement value on the basis of the measured specific activity and reactor power history. On the other hand the neutron sensor reaction rates can be also derived using the neutron spectrum from the transport calculation at the location of EVND and appropriate cross-section library as bellow:

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g}) \tag{3}$$

where R_i is the calculated reaction rate of sensor i, σ_{ig} is multi-group dosimeter reaction cross-section, and ϕ_g is neutron spectrum, and δ is uncertainty. SNLRML dosimetry cross-section library[6] was used in this analysis.

3. Results and Conclusion

Average ratio of measured to calculated (M/C) reaction rates of seven sensors for six EVND capsules are summarized in Table 2. The comparisons of reaction rates of $^{54}Fe(n,p)^{54}Mn$ for stainless steel bead chain installed at azimuthal angle 0 degree are shown in Figure 1.

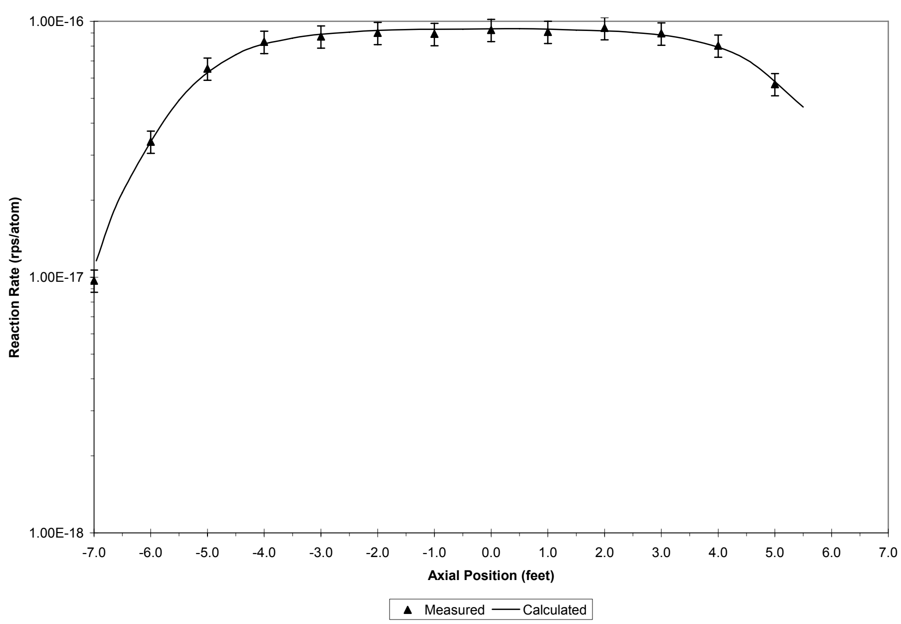
Table 2 shows that the calculated reaction rates are in good agreement with the measured reaction rates. The M/C ratio of capsule C shows the opposite trend compared to other capsule results. Capsule C has been positioned at the bottom of the core where the axial flux gradient was so steep. Therefore the calculated flux evaluated by the synthesis method might have a lot of uncertainty compared to the others. Figure 1 shows that the stainless steel bead chains can be used for monitoring the axial gradients of flux distributions at the reactor cavity locations.

The neutron spectrum calculated by the neutron transport calculations and synthesis method can be validated by the comparisons of the reaction rates between calculation and measurement. And the fast neutron flux (E > 1 MeV) can be obtained by integrating the resultant spectrum above 1 MeV energy range. The fast neutron fluence which is the most important input data in the assessment of the state of embrittlement of the pressure vessel can be obtained by multiplying the resultant fast neutron flux obtained from the neutron transport calculations for the reactor vessel by the full power operation time. And the EVND sets can be used in the validation of the evaluation of fast neutron fluence used in the assessment of vessel embrittlement.

Table 2: Comparisons of measured and calculated reaction rate for EVND capsules of YGN-1

Capsule	A	B	C	D	E	F
M/C	0.94	0.91	1.16	0.89	0.91	0.89
Average	0.95					

Figure 1: Comparisons of $^{54}Fe(n,p)^{54}Mn$ reaction for bead chain installed at azimuthal angle 0 degree



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