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Evaluation of Ex-Vessel Neutron Dosimetry for Yonggwang Unit 1

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1. Introduction

The Code of Federal Regulations, Title 10, Part 50, Appendix H[1], 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reactor cavity annulus between the reactor vessel outer wall and the biological shield at the end of cycle 14 and were withdrawn at the end of cycle 15 for the evaluation.

Evaluations of EVNDs withdrawn at the end of cycle 15 and the fast neutron exposures on the pressure vessel beltline region of YGN-1 were performed on the guidance specified in Regulatory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[2]. YGN-1 EVND program employs advanced sensor sets that consist of the radiometric sensors illustrated in Table 1 and gradient chains.

Table 1. Summary of EVND sensors for YGN-1

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. Methods

In this section some of the techniques used to evaluate the EVNDs are described.

2.1 Neutron Transport Calculations

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

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

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 using the DORT 3.1 discrete ordinate code[3] and the BUGLE-96 cross-section library[4]. 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.

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 0, 15, 30, and 45 degrees relative to the core major axes. For the azimuthal 0 degree, 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

F = weight fraction of the target isotope

Y = number of product atoms produced per reaction

P_j = average core power during period j (MW)

P_{ref} = reference core power level of the reactor (MW)

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 (1.0 for single cycle irradiation)

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.

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[5] 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.

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					

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.

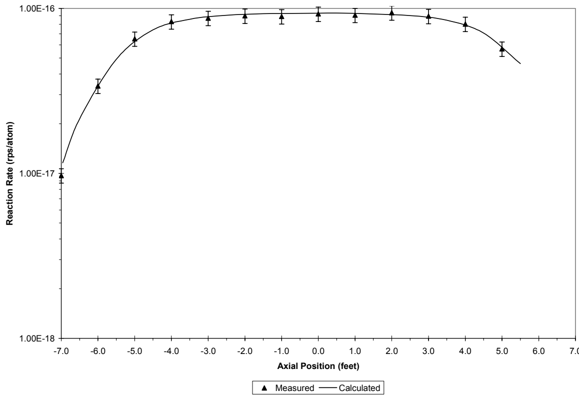

Figure1. comparisons of $^{54}Fe(n,p)^{54}Mn$ reaction for bead chain installed at azimuthal angle 0 degree for YGN-1

Table 1 shows that the differences between measurements and calculations are less than 20% for each capsule, which means these analyses are satisfying the acceptable criterion required by Regulatory Guide 1.190[2]. 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.

REFERENCES

[1] CFR Part50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," U.S. NRC, January 1, 1998.

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

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

[4] 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.

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