J.Korean Asso. Radiat. Prot.
Vol.26 No.3 : 237-241(2001)

EVALUATION OF THE UNCERTAINTIES IN THE MODELING AND SOURCE DISTRIBUTION FOR PRESSURE VESSEL NEUTRON FLUENCE CALCULATIONS

Yong-Il Kim and Hae-Ryong Hwang

Korea Power Engineering Company, Inc., Radiation Safety Analysis Group, 150, Duckjin-Dong, Yusong-Gu, Daejeon, Korea, 305-353

Abstract - The uncertainties associated with fluence calculation at the pressure vessel have been evaluated for the Korean Next Generation Reactor, APR1400. To obtain uncertainties, sensitivity analyses were performed for each of the parameters important to calculated fast neutron fluence. Among the important parameters to the overall uncertainties, reactor modeling and core neutron source were examined. Mechanical tolerances, composition and density variations in the reactor materials as well as application of r-θ geometry in rectilinear region contribute to uncertainty in the reactor modeling. Depletion and buildup of fissile nuclides, instrument error related to core power level, uncertainty of fuel pin burnup, and variation of long-term axial peaking factors are main contributors to the core neutron source uncertainty. The sensitivity analyses have shown that the uncertainty in the fluence calculation at the reactor pressure vessel is +12%.

INTRODUCTION

The structural integrity of the pressure vessel is one of the major factors that determine operating life of a nuclear reactor. To assure the safety margins against increased plant availability or to consider life extension of reactors, it is inevitable to require accurate estimation of neutron fluence on the pressure vessel. There are two major steps in performing fluence calculations: determination of neutron source in the reactor core and transport of neutrons throughout the reactor. To obtain accurate estimation of vessel fluence, uncertainties contained in the calculation procedure should be determined.

The uncertainty contained in the calculated exposure rate and integrated exposure can be conveniently subdivided into two broad categories. The first category involves biases or systematic uncertainties that may be present due to the calculation method or in the basic nuclear data input to the calculation. These potential biases can be obtained by comparing analytic results with measurements from plant specific surveillance or controlled benchmark experiments. The second category of uncertainty in the analysis of vessel exposure involves variations that may exist in reactor dimensions, coolant temperature, neutron source strength and source distribution, as well as in other parameters that may vary from reactor to reactor or fuel cycle to fuel cycle. This category of uncertainty can be determined via sensitivity studies performed for each of the parameters important to the fluence calculation.

In this paper, the second category of uncertainty has been examined using sensitivity analyses to determine vessel fluence uncertainty applicable for the design of APR1400 and to see how the variations in reactor geometry, material, and neutron source translate into the calculated vessel exposure.

Important input parameters that affect the analytical results at the pressure vessel were examined using 2-dimensional discrete ordinates transport code, DORT[1]. The reactor contains core, core shroud, core barrel, vessel cladding, and pressure vessel as shown in Fig. 1. Macroscopic cross sections were obtained from BUGLE-96 library[2] using GIP code[1]. The S8 angular quadrature set[3] was used for the particle directions and P3 order of scattering was applied for all nuclides.

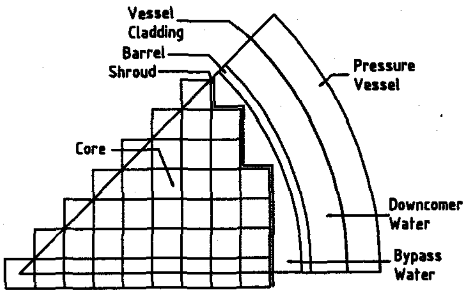

Fig. 1. APR1400 Model

UNCERTAINTY IN REACTOR MODELING

1. r-θ modeling

The reactor is composed of rectangular regions (fuel assemblies and core shroud) and cylindrical regions (including the core barrel and pressure vessel). For the vessel fluence calculation, the cylindrical geometry has been used because it adequately represents the pressure vessel, and the rectangular regions were projected onto it.

The modeling of the rectangular core regions in r-θ geometry causes a potential source of uncertainty in the geometric modeling of the reactor. The sensitivity of the solution to the modeling approach was determined by a direct comparison of the results of an r-θ computation with those of an x-y calculation in which the shroud region and core were modeled explicitly. The comparisons of interest were taken at vessel/clad interface of the pressure vessel.

To minimize errors arisen from using rectangular coordinates in the cylindrical regions and for the purpose of comparison of DORT r-θ and x-y results, homogenization of materials were carried out for those regions. The bounding uncertainties associated with this modeling approximation are determined as +6.4 %, -0.6 %.

2. Geometric Dimensions

The current design methodology generally makes use of nominal design dimensions of components to establish the reactor geometry and full power coolant temperature to determine water density in the core and downcomer regions. Sensitivity of the projected fast neutron exposure of the pressure vessel to each of these variables has been addressed via independent parametric studies.

Thickness tolerances on the stainless steel (core shroud, core barrel) and corresponding tolerances placed on the inner radius of these steel components and pressure vessel were examined. To determine the potential impact of the reactor internals manufacturing and assembly tolerances on the analytical prediction of the fast neutron fluence of the pressure vessel, calculations were performed for cases representing minimum and maximum shielding between the reactor core and the pressure vessel. These extreme conditions were then compared to the nominal calculation to establish an upper bound uncertainty in the use of nominal vs. as-built internals dimensions. The uncertainties associated with each dimensional tolerance in the calculated exposure of the pressure vessel are +7.2 % and -5.5 %.

3. Material Compositions

Radiation shielding materials between reactor core and pressure vessel are stainless steel in internals and water in the bypass and downcomer regions. Among elements in reactor internals, Fe, Ni and Cr are major nuclides from a viewpoint of neutron shielding and their concentrations amount to approximately 97 %.

Two factors were considered in the sensitivity analyses associated with material composition of reactor internals: possible variations of composition in steel constituents within the chemical requirements and the internal density changes due to temperature rise caused by radiation heating.

To see the sensitivity of neutron fluence to the individual concentrations of each nuclide, transport calculations were performed for different material combinations with limiting concentrations of Ni and Cr. The concentration of Fe in the composite such as stainless steel depends on the concentrations of all other elements in the composite. The results of this sensitivity analysis show that the pressure vessel fluence uncertainty is +0.27 % when reactor internals have minimum Ni and maximum Cr compositions, and -0.26 % when maximum Ni and minimum Cr compositions.

Conventional calculation methodology assumes that the temperatures of reactor internals are the same with reactor coolant cold-leg temperature. However, according to the calculations[4] for the temperature distribution of reactor internals, the average temperature in the active region of the core barrel reaches 680.5 °F maximum. The calculation of heating rates for the temperature distribution was based on the core power distribution of LOCA limiting condition, so the calculated temperatures in reactor internals can be regarded as somewhat conservative. Sensitivity calculation shows +0.57 % fluence uncertainty when the average temperatures in the core shroud and barrel was assumed to be 650 °F.

The sensitivity of the calculated vessel fluence to fluctuations in water temperature was likewise determined via a parametric study in which water temperature and, hence, coolant density was varied over a range of several degrees F relative to nominal conditions.

Temperature variations of ±4 °F was chosen from the acceptable operation area bounds[5] of the reactor coolant cold-leg temperature at normal full power operation. Sensitivity calculation shows that coolant temperature variations of ±4 °F translate into +3.47 % and -3.30 % fluence uncertainties.

UNCERTAINTY IN CORE NEUTRON SOURCE

1. Source Parameters

Uncertainty calculations involving source parameters such as fission spectrum, neutron yield per fission and energy release per fission were performed via an evaluation of the sensitivity of calculated fluence at the pressure vessel to the varying core average burnup of the reactor. These burnup sensitivity studies encompass significant perturbations in those source parameters due to the buildup of plutonium isotopes as the core burnup increases. The multi-group source distribution can be determined by combining the power distribution with a power-to-source conversion factor, ν/E_R, and a source spectrum χ. The neutron source density for energy group g is expressed as

$$S_{ig} = c_0 \times \chi_g \times \nu / E_R \times P_i, \qquad (1)$$

i = fuel pin index

WHERE P_i = pin power

c_0 = energy generation rate in [MeV/cm^3,sec].

To see the effects of buildup of plutonium isotopes to the source parameters, calculations using nuclides number densities at the beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) of equilibrium core were compared. The power-to-source conversion factors and source spectrums were averaged over the fissile nuclide number densities for each burnup stage. In this case, the same pin power distribution was applied to the fluence calculations for the purpose of sensitivity analysis of source parameters. The results of this evaluation indicate that the maximum relative difference in vessel fluence between BOC and MOC is -2.4 %, between EOC and MOC is +2.8 %.

2. Absolute Power Distribution

The thermal output of reactor core is used for source magnitude, c0, as shown in Eq. (1). Therefore, the uncertainty contained in absolute power level is directly reflected in uncertainty of pressure vessel fluence. US NRC, from Regulatory Guide 1.49, regulates the maximum core thermal power and approves 2% margin of licensed thermal power level by considering permissible error of instrument which determine the power level. Hence, (2% of uncertainty are assigned to the absolute power level or reactor core thermal output.

3. Relative Pin Power Distribution

The relative power distribution (RPD) for the vessel fluence calculation is calculated from fuel pin burnup as

$$P_i = \frac{Bu_i^E - Bu_i^B}{\overline{\Delta Bu}} = \frac{\Delta Bu_i}{\overline{\Delta Bu}}, \tag{2}$$

where Bu_i^B, Bu_i^E = *burnup of fuel pin i at BOC and EOC*

$\overline{\Delta Bu} = core\ average\ cycle\ burnup\ (= \frac{1}{N}\sum_{i=1}^{N} \Delta Bu_i)$.

Since the locations of fuel pins are changed for each cycle and EOC burnup strongly depends on their location in the core rather than their initial burnup, three variables in Eq. (2) can be considered as independent to each other. If we assume that BuiB, BuiE and are independent random variables, the uncertainty of pin power Pi defined in Eq. (2) can be derived by using variance property with the aid of Taylor series expansion as

$$\sigma(P_i) = \sigma\left(\frac{Bu_i^E - Bu_i^B}{\overline{\Delta Bu}}\right) \cong \sqrt{\frac{(\sigma_i^E)^2 + (\sigma_i^B)^2}{(\overline{\Delta Bu})^2} + \frac{(\Delta Bu_i)^2}{(\overline{\Delta Bu})^4}\sigma^2(\overline{\Delta Bu})}, \tag{3}$$

where σ_i^B and σ_i^E are burnup uncertainties of fuel pin *i* at BOC and EOC.

Since the second term in the square root of Eq. (3) can be neglected, the uncertainty of pin power Pi is rewritten as

$$\sigma(P_i) = \frac{1}{\overline{\Delta Bu}}\sqrt{(\sigma_i^E)^2 + (\sigma_i^B)^2}. \tag{4}$$

APR1400 employs longer cycle fuel management such as three batch refueling scheme, which means a third of fuel assemblies (FAs) are replaced for each cycle. Fresh, once-burned, twice-burned FAs exist at BOC and once-, twice-, thrice-burned FAs exist at EOC. Based on cycle average burnup $\overline{\Delta Bu}$, the expected average burnup of once-, twice- and thrice-burned FAs were assumed to be $\overline{\Delta Bu}$, 2$\overline{\Delta Bu}$, and 3$\overline{\Delta Bu}$ respectively. Therefore, representative average burnups at BOC and EOC for arbitrary cycles were calculated as and $\overline{\Delta Bu}$ and 2$\overline{\Delta Bu}$, which were obtained simply by averaging expected average FA burnups. According to the report[6] for physics biases and uncertainties, maximum uncertainty of calculated 1-pin burnup is ±2.5 % in proportion to fuel burnup under 30000 MWD/T and fixed with ±750 MWD/T over 30000 MWD/T. Hence, if the maximum uncertainty of calculated 1-pin burnup, (=±0.025), is applied to Eq. (4), the maximum uncertainty of pin power Pi is expressed as

$$\sigma_{max}^{RPD} = \frac{1}{\overline{\Delta Bu}}\sqrt{(2\alpha\overline{\Delta Bu})^2 + (\alpha\overline{\Delta Bu})^2} = \sqrt{5}\alpha = \pm 0.056\ . \tag{5}$$

To obtain RPDs that would be used for the fluence uncertainty calculations, bounding uncertainties of (0.056 were added to each reference pin power and then normalized to get core average power density of unity. The sensitivity analysis to the RPD was done by comparing fluence results obtained from new RPDs with those from reference RPD. The results of this evaluation indicate that the core design uncertainty (fuel pin burnup uncertainty) translates into the vessel fluence uncertainties of +3.2 % and -3.5 %.

4. Axial Power Distribution

The uncertainty in the axial power distribution averaged over the irradiated period translates directly to an uncertainty in the calculated neutron flux external to the core. The long-term axial peaking factor is calculated from burnup difference between EOC and BOC for a given cycle per each plane of core height, representing cycle average power peak, not an instantaneous power peak in axial direction. The basic idea for the calculation of long-term axial peaking factor corresponds to that of pin power distribution used for vessel fluence calculation. Hence, the long-term axial peaking factors were chosen to obtain uncertainty in the axial power distribution. The axial peaking factors vary from 1.11 to 1.06 for APR1400 various cycles, yielding an average value of 1.085. The upper and lower limits around average axial peaking factor, a variation of ±2.5 %, are taken to be applicable for the uncertainty in the axial power distribution. This uncertainty value is liberal enough to encompass the entire change in axial shape over the course of the fuel cycles.

RESULT AND CONCLUSION

The evaluation of uncertainties included in vessel fluence calculation has been performed through sensitivity analyses. The sensitivity of calculated fast neutron exposures to the input parameters on the pressure vessel were examined for limiting values. The vessel fluence uncertainties associated with reactor modeling and neutron source for the APR1400 are presented in Table 1 with those calculated from other groups. The sources of uncertainties were classified according to the calculations performed in this paper, so the details or scopes of the uncertainty calculations in the classified items were somewhat different.

TABLE 1. Vessel Fluence Uncertainties of Pressurized Water Reactors

Source of Uncertainty		Uncertainty [%]		
		ORNL[7]	W[8]	APR1400
Reactor Modeling	Geometric Modeling	9.9[a]	5.9	9.6
	Steel Density	3.6	-	0.6
	Coolant Density	3.9	4	3.5
Subtotal		11.3	7.1	10.3
Neutron source	Source Parameters	6.4	2	2.8
	Absolute Power Level		5	2
	Relative Power Distribution		4	3.2
	Axial Power Distribution		5	2.5
Subtotal		6.4	8.4	5.3
Total		13	11	12

[a] Method and modeling uncertainty was included.

All combined uncertainties, such as subtotal or total uncertainties, were obtained from root-sum-square of individual uncertainties. The total uncertainty of APR1400 represents upper tolerance limit of +12%, while that of ORNL or W represents bounding tolerance limits expressed as ±13% or ±11%.

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