Neutron Fluence Calculation Methodology for Reactor Vessel

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ABSTRACT

This technical report provides the methodology for calculating the neutron fluence for the Advanced Power Reactor 1400 (APR1400) reactor vessel and an evaluation of the bias and uncertainties associated with neutron transport calculations.

The common approach for the estimation of the reactor vessel fluence is to perform fixed-source transport calculations using a particle transport code. To obtain fast neutron flux at the APR1400 reactor vessel, the DORT transport code[1)] and BUGLE cross section library[2)] are used for the transport calculations.

Bias is normally ascribed to the uncertainties in the nuclear cross-section data or to the transport calculation methodology itself. The bias for the fluence evaluation can be determined by comparing the transport calculations and the measured data from controlled benchmark experiments. For the APR1400 neutron fluence evaluation, measured data from the VENUS-1 experimental reactor are used[3)].

Uncertainties due to the variations in the input data for the fluence calculations are obtained by performing sensitivity analyses of the parameters that are important to fast neutron fluences, which include reactor modeling and core neutron source.

The results of the neutron fluence calculations for the APR1400 at various points of the reactor vessel are presented in this technical report. The results include the bias and uncertainties that are evaluated.

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1) CCC-650/DOORS3.2, “One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System,” Radiation Safety Information Computational Center, Oak Ridge National Laboratory, 1998.

2) D. T. Ingersoll, J. E. White, R. Q. Wright, H. T. Hunter, C. O. Slater, N. M. Greene, R. E. MacFarlane, R. W. Roussin, “Production and Testing of the VITAMIN-B6 Fine-Group and the BUGLE-93 Broad Group Neutron/Photon Cross Section Libraries Derived from ENDF/B-VI Nuclear Data, ” DLC-175, ORNL-6795, Oak Ridge National Laboratory, April 1994.

3) Rulko, R. P., “VENUS-1 Benchmark on Dosimetry Computations,” OECD NEA, NDB/97/0941/rr, Organisation for Economic Co-operation and Development Nuclear Energy Agency, June 1997.

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ACRONYMS AND ABBREVIATIONS

APR1400	Advanced Power Reactor 1400
BOC	Beginning Of Cycle
DORT	Two-Dimensional Discrete Ordinates Transport Code
EFPY	Effective Full-Power Year
EOC	End Of Cycle
FA	Fuel Assembly
GIP	Group-Organized Cross Section Input Program
MeV	Mega-electron Volt
MOC	Middle Of Cycle
MWD/T	Mega-Watt Day per Tonne
MW_{th}	Mega-Watt thermal
RG	Regulatory Guide
RPD	Relative Power Distribution
S_n	Symbolization of discrete ordinates approximation indicating N discrete values of the direction cosine
TS	Trade Secret
VENUS	Vulcain Experimental Nuclear Study

1 INTRODUCTION

This technical report presents the methodology for calculating the neutron fluence for the Advanced Power Reactor 1400 (APR1400) reactor vessel and an evaluation of the bias and uncertainties associated with neutron transport calculations.

The fission neutrons starting from the reactor core are attenuated by the core shroud, core support barrel, and reactor coolant, all of which are between the core and RV. The methodology for calculating neutron flux presented in this report is in accordance with Regulatory Position 1, "Neutron Fluence Calculational Methods," in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 1). For the evaluation of neutron fluence in the reactor vessel, the DORT code (Reference 2) with BUGLE cross-section library (Reference 3) is used to calculate neutron flux distributions in the reactor system.

The uncertainty associated with the calculated exposure rate and integrated exposure is divided into two broad categories. The first category involves biases or systematic uncertainties that may be the result of the methodology itself or that may be in the basic nuclear data input to the calculation. The potential biases can be obtained by comparing analytic results with measurements from controlled benchmark experiments. The first category includes the following:

- Cross Sections
- S_n Method

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 from fuel cycle to fuel cycle. This category of uncertainty can be examined via sensitivity studies performed for each of the parameters important to the overall evaluation. The important input parameters that are addressed in this report are as follows:

- Reactor Modeling
  - r-θ Modeling
  - Geometric Dimensions
  - Material Compositions
- Core Neutron Source
  - Source Parameters
  - Relative Power Distribution
  - Axial Power Distribution

2 NEUTRON TRANSPORT CALCULATION METHODS

The DORT code (Reference 2) is used to evaluate neutron flux distributions. DORT is widely used in the nuclear industry for flux-distribution evaluations of reactor vessels. DORT is a discrete ordinates S_n code, and can perform calculations in (x, y), (r, θ), and (r, z) geometries.

To determine the reactor vessel neutron flux distribution, (r, θ) geometry is selected to effectively model the circular shape of the core support barrel and reactor vessel. Rectangular geometry of the fuel assemblies and the core shroud are finely approximated by giving enough number of (r, θ) meshes. The approximation in the geometrical modeling is considered in the uncertainty evaluation. Axial variations of neutron fluxes are considered by applying long-term axial power distributions with a bounding axial peaking factor of 1.15 to 2-dimensional (r, θ) calculation results.

For the DORT calculation, the BUGLE-93 (Reference 3) cross section library is used. BUGLE-93 is a multigroup cross-section library based on ENDF/B-VI, which provides a coupled 47 neutron 20 gamma ray group cross-section data set produced for light water reactor shielding and reactor pressure vessel dosimetry applications.

An S_8 fully symmetric angular quadrature set and P_3 Legendre expansion of the scattering cross-sections are used for the DORT transport calculation. A point-wise inner iteration flux convergence criterion of 0.001 is used for the vessel fluence calculation as recommended by RG 1.190.

Total uncertainty of neutron flux estimation methodology is within 20 percent, which is evaluated by sensitivity analysis covering input parameters for neutron transport calculations and bias evaluation from a comparison of calculation results with the experimental data from the VENUS-1 test reactor benchmark experiment (Reference 4). A detailed description of the bias and uncertainty evaluation is provided in Section 3.

2.1 Geometry Modeling

The geometric model of the APR1400 includes the reactor core, reactor internals, surveillance capsules, pressure vessel with stainless steel cladding, and cavity wall. Nominal design dimensions and material composition data are used in the development of the input parameters of the geometric model. The coolant number densities in the reactor core, bypass, and downcomer regions are derived based on the full-power operating temperatures and pressures of the APR1400.

A quarter model of the horizontal cross-section of the APR1400 reactor is shown in Figure 2-1, and the number density of each structure component is shown in Table 2-1. The reactor is modeled using r-θ coordinates with 197 radial and 244 azimuthal meshes. In the geometry model as shown in the Figure 2-1, interior assemblies are not included in the model since the fast neutron flux on the reactor vessel originates primarily from the peripheral assemblies.

Considering the r-θ quarter core symmetry model, reflective boundary conditions are applied to the bottom, top, and left boundaries of the calculation model. The vacuum boundary condition is applied to the right boundary of the model.

2.2 Core Neutron Source and Fission Spectra

The conservative pin power distribution for the APR1400 vessel fluence calculation is used for the spatial source distribution in the DORT transport calculation. The conservative power distribution is obtained by composing peripheral fuel assemblies (FAs) as hottest FAs in their locations among those from many fuel cycles, which makes bounding peripheral power distribution. Figure 2-2 represents the octant core assembly relative power distribution for vessel fluence calculation.

The core thermal power of 3,983 MW_{th} is used for the calculation of volumetric fission neutron source strength. The neutron source spectrum is determined by combining the volumetric fission density with the number of neutrons per fission (ν_i), relative fission rates (f_i), and fission spectra (χ_i) for each fissionable nuclide. The calculated core neutron source and fission spectra for each fissionable nuclides are shown in Table 2-2.

Table 2-1 Nuclide Number Densities of Reactor Materials (1 of 2)

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Table 2-1 Nuclide Number Densities of Reactor Materials (2 of 2)

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Table 2-2 Fission Spectra and Neutron Source Spectrum (1 of 2)

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Table 2-2 Fission Spectra and Neutron Source Spectrum (2 of 2)

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Figure 2-1 Reactor Geometry Model for Neutron Fluence Calculation

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Figure 2-2 Octant Core Assembly Power for Vessel Fluence Calculation

3 BIAS AND UNCERTAINTY

3.1 Bias Analysis

As described in Section 1, the first category of uncertainty involves biases or systematic uncertainties. For the bias analysis, the experimental results of the VENUS-1 test reactor are used. Figure 3-1 shows the configuration of the VENUS-1 test reactor and distributed specimens (detector points) at various regions. The experiment performed at the VENUS-1 reactor provides activity data at the water gap II and neutron pad, which correspond to the downcomer surveillance capsule and pressure vessel locations of a power reactor, respectively.

The transport calculations are performed using the DORT code. The calculations are done for (x, y) geometry to obtain radial flux distribution and (r, z) with (r) geometries for axial factors. Figure 3-2 shows the DORT x-y calculation model for VENUS-1 configuration (Reference 4). Considering the x-y quarter core symmetry model, reflective boundary conditions are applied to the left and bottom boundaries, and the vacuum boundary condition is applied to the right and top boundaries of the model. For cylindrical geometry DORT calculations to obtain the axial factor, reflective boundary condition is applied to left boundary, and vacuum boundary conditions are applied to the other boundaries.

The BUGLE-96 (Reference 5) cross section library based on ENDF/B-VI (Release 3) is used in the benchmark calculations. S_8 symmetric quadrature set with P_3 order of scattering cross-sections are used, and a point-wise flux convergence criterion of 0.001 is used in the DORT models for the VENUS-1 benchmark calculation.

The source distribution is based on the experimental power distribution (Reference 4) as shown in Figure 3-3, which represents normalized pin power distribution at core mid-plane as core average fission density becomes 1.0 (fission/sec/pin). Fission spectrum (χ_i) from U^{235} fission provided by the BUGLE-96 cross-section library is used for the source energy distribution.

The equivalent fission fluxes calculated for each detector point are given in Table 3-1 for five types of dosimeters. Table 3-2 shows the ratio of the calculated to measured (C/M) value for each dosimeter. Since all of the calculated-to-measured ratios are within 10 percent, an average value of C/M of each region is determined as a bias for the region.

Table 3-3 summarizes the calculated biases for each region in the VENUS-1 test reactor. The bias for the neutron pad is used as a bias (6 percent) for the vessel fluence calculation using the 2-dimensional DORT code.

3.2 Uncertainty Analysis

The second category of uncertainty is examined using sensitivity analyses to determine the vessel fluence uncertainty applicable to the APR1400 design.

Important input parameters that affect the analytical results at the pressure vessel are examined using the 2-dimensional discrete ordinates code (DORT) and the typical vessel fluence calculation model, which contains the reactor core, core shroud, core barrel, vessel cladding, and pressure vessel. Figure 3-4 shows the DORT r-θ octant core model for sensitivity analysis calculation.

Macroscopic cross sections are obtained from the BUGLE-96 library using GIP code (Reference 2). The S_8 angular quadrature set is used for the particle directions, and the P_3 order of scattering is applied to all nuclides in the uncertainty calculations.

A conservative pin power distribution is used for the space-dependent source term, and the core thermal power of 3,983 MW_{th} is used for the calculation of volumetric fission neutron source. The core inlet, outlet, and average coolant temperatures are taken from the normal full-power operating condition.

3.2.1 r-θ Modeling

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 this modeling approach is 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 are modeled explicitly. The comparisons of interest are taken at vessel/clad interface of the pressure vessel.

The bounding uncertainties associated with this modeling approximation are determined as +6.4 percent and -0.6 percent.

3.2.2 Geometric Dimensions

Thickness tolerances on the stainless steel (core shroud, core support barrel) and corresponding tolerances placed on the inner radius of these steel components and the pressure vessel are 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 are performed for cases representing minimum and maximum shielding between the reactor core and the pressure vessel. These extreme conditions are then compared to the nominal calculation to establish an upper bound uncertainty in the use of nominal versus as-built internals dimensions.

The uncertainties associated with each dimensional tolerance in the calculated exposure of the pressure vessel are +7.2 percent and -5.5 percent.

3.2.3 Material Compositions

Radiation shielding materials between the reactor core and pressure vessel are stainless steel in internals and water in the bypass and downcomer regions.

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

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

Sensitivity calculations show that uncertainties contained in the material compositions translate into +3.5 percent and -3.3 percent fluence uncertainties.

3.2.4 Source Parameters

Uncertainty calculations involving source parameters such as fission spectrum, neutron yield per fission, and energy release per fission are 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 the 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 (υ/E_R) and a source spectrum (χ_g). The neutron source density for energy group g is expressed as

$$S_{ig} = c_0 \times \chi_g \times \upsilon / E_R \times P_i \qquad (3.1)$$

Where

i = fuel pin index

P_i = pin power

c_0 = fission 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 are compared. The power-to-source conversion factors and source spectra are averaged over the fissile nuclide number densities for each burnup stage. In this case, the same pin power distribution is 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 percent and between MOC and EOC is +2.8 percent.

3.2.5 Absolute Power Distribution

The thermal output of the reactor core is used for source magnitude, c_0, as shown in Eq. (3.1). Therefore, the uncertainty contained in the absolute power level is directly reflected in uncertainty of pressure vessel fluence. The U. S. Nuclear Regulatory Commission, as documented in RG 1.49 (Reference 6), regulates the maximum core thermal power and approves a 2 percent margin of licensed thermal power level by considering the permissible error of instrument, which determines the power level. Hence, ±2 percent of uncertainty is assigned to the absolute power level or reactor core thermal output.

3.2.6 Relative Power Distribution

The relative pin power of each fuel pin for the vessel fluence calculation is calculated from fuel pin burnup as

$$P_i = \frac{Bu_i^E - Bu_i^B}{\overline{\Delta Bu}} = \frac{\Delta Bu_i}{\overline{\Delta Bu}}, \qquad (3.2)$$

Where

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

$\overline{\Delta Bu} =$ core average cycle burnup.

According to Combustion Engineering's "Methodology Manual for Physics Biases and Uncertainties" (Reference 7), the maximum uncertainty of calculated 1-pin burnup is ±2.5 percent in proportion to fuel burnup less than 30,000 MWD/T and fixed with ±750 MWD/T greater than 30,000 MWD/T. The locations of FAs are changed for each cycle, and EOC burnups of fuel pins are significantly affected by their locations in the core. Hence, each BOC and EOC burnup in Eq. (3.2) can be considered an independent variable. When the maximum burnup uncertainty (σ_{max}) of ±2.5 percent is applied to the uncertainty of each BOC and EOC burnup, the uncertainty of pin power (P_i) is calculated as:

$$\sigma(P_i) = \sqrt{2} \times \sigma_{max} = \pm 3.5\% \qquad (3.3)$$

The vessel fluence is directly proportional to the relative power distribution (RPD) in the core region, and the uncertainty of vessel fluence calculation ascribed to RPD can there be taken as ±3.5 percent.

3.2.7 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 the 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 calculation basis of long-term axial peaking factor (long-term axial power distribution) corresponds to that of pin power distribution used for vessel fluence calculation. Hence, the long-term axial peaking factors are chosen to obtain uncertainty in the axial power distribution. The axial peaking factors vary from 1.11 to 1.06 for the APR1400 various core fuel cycles, yielding an average value of 1.085. The upper and lower limits around average axial peaking factor, a variation of ±2.5 percent, 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.

3.3 Summary of Bias and Uncertainty Evaluation

The evaluated bias for vessel fluence calculation due to the systematic errors in the methodology itself and the basic nuclear data are determined as +6 percent as shown in Table 3-3 for the neutron pad region.

The evaluation of uncertainties included in the vessel fluence calculation is performed through sensitivity analyses. The sensitivities of calculated fast neutron exposures to the input parameters on the pressure vessel are examined for limiting values. The vessel fluence uncertainties associated with reactor modeling and neutron source for the the APR1400 are presented in Table 3-4.

A total of 20 percent uncertainty, which contains bias (6 percent) and uncertainties from input parameters for analysis (11.6 percent) with additional margin, are considered in the vessel fluence estimation.

Table 3-1 DORT Calculation Results for VENUS-1 Experiment

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Table 3-2 Calculation to Measurement Ratios for VENUS-1 Experiment

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Table 3-3 Bias of Each Region for Equivalent Fission Flux

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Table 3-4 Uncertainty of Vessel Fluence Calculation for the APR1400

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Figure 3-1 Configuration of VENUS-1 Test Reactor

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Figure 3-2 DORT x-y Calculation Model for VENUS-1 Test Reactor

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Figure 3-3 Fuel Pin Power Distribution of VENUS-1 Test Reactor

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Figure 3-4 Octant Core DORT r-θ Calculation Model for Uncertainty Evaluation

4 EFFECT OF CROSS-SECTION LIBRARIES ON THE FAST NEUTRON FLUX

Figure 4-1 shows a comparison of the BUGLE-93 and BUGLE-96 cross-section libraries in the vessel fluence calculation, fast neutron fluxes (E > 1 MeV), at the inner surface of the reactor vessel. The relative differences range from 2.8E-04 percent to 0.16 percent and the relative difference of maximum flux point is 0.09 percent.

It can be concluded that there is no meaningful difference between the two cross-section libraries on the vessel fluence calculation. Therefore, the uncertainty and bias obtained from the analysis using BUGLE-96 cross-section library can be applied to the vessel fluence evaluation using BUGULE-93 cross-section library.

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Figure 4-1 Comparison of Fast Neutron Fluxes Using BUGLE-93 and BUGLE-96

5 RESULTS OF NEUTRON TRANSPORT CALCULATIONS

Table 5-1 shows typical neutron fluxes at various points in the reactor vessel and reactor core region. The maximum fast neutron fluences (E > 1 MeV) inside the reactor vessel at 55.8 effective full-power years (based on a plant capacity factor of 93 percent and a life of 60 years) is shown in Table 5-2.

Figure 5-1 shows the azimuthal distribution of fast neutron fluences at inner surface, 1/4 thickness, and 3/4 thickness in the reactor vessel. Figure 5-2 shows the radial distribution of fast neutron fluence at 40 degree azimuth, the intermediate azimuthal direction between the two surveillance capsules.

All calculated results are obtained on the basis of the methodology and calculation conditions described in the previous sections of this report, in which bias and uncertainty are taken into account. Calculated results also include a long-term axial peaking factor of 1.15 to account for the effect of axial flux variation since the results are obtained from a 2-dimensional (r, θ) geometry transport calculation.

Table 5-1 Typical Neutron Fluxes inside the Reactor Vessel

(Unit: n/cm^2-sec)

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Uncertainty of 20 % and axial peaking factor of 1.15 are included

Table 5-2 Fast Neutron Fluences at the Reactor Vessel

Reactor Vessel Location	Radial Position (cm)	Fast Neutron Fluence (n/cm^2) (E > 1 MeV, 55.8 EFPY)
At the inside surface (peak)	231.775	9.5×10^{19}
At the 1/4 thickness location (peak)	237.528	5.1×10^{19}
At the 3/4 thickness location (peak)	249.034	1.0×10^{19}

Uncertainty of 20 % and axial peaking factor of 1.15 are included

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Figure 5-1 Azimuthal Distributions of Fast Neutron Fluence in the Reactor Vessel

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Figure 5-2 Radial Distribution of Fast Neutron Fluence

6 CONCLUSION

Neutron fluence for the APR1400 reactor vessel has been calculated by using DORT code and BUGLE cross-section library. The calculation methodology is in accordance with Regulatory Position 1, "Neutron Fluence Calculational Methods," in Regulatory Guide (RG) 1.190. The calculated results include bias and uncertainty due to the transport calculation methodology itself and to the variations in the input data. The bias for the neutron fluence calculation has been determined by comparing the transport calculations and the measured data from benchmark experiments of the VENUS-1 experimental reactor. The uncertainty due to the variations in the input data for the fluence calculation has been obtained by performing sensitivity analyses of the important parameters for the reactor modeling and core neutron source description.

The peak fast neutron fluence (E > 1 MeV) over 60 years of operation with a plant capacity factor of 93 percent has been calculated as 9.5×10^{19} (neutrons/cm^2) at vessel/clad interface of the APR1400 reactor vessel. This fluence result includes a total uncertainty of 20 percent and a long-term axial peaking factor of 1.15.

7 REFERENCES

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