ABSTRACT

The radiation induced damage to the reactor pressure vessel by fast neutron fluence is a main factor when considering lifetime extension for nuclear power plants. This paper is addressing feasibility of modelling a typical pressurized water reactor and predicting dose rates in reactor pressure vessel and biological shield using the SCALE6.0 code package, developed at Oak Ridge National Laboratory. The structural identification of reactor core and internals, implementation of advanced variance reduction methodology and visualization of mesh tally results were necessary steps. The CSAS6 sequence of the SCALE6.0 code package, which includes KENO-VI Monte Carlo code, as well as MAVRIC/MONACO shielding sequence implementing CADIS and FW-CADIS methodology, was used for calculation of neutron and photon dose rates. The criticality and shielding analysis were performed based on general purpose ENDF/B-VII.0 library. In general, SCALE6.0 proved to be feasible for predicting dose rates in reactor pressure vessel and biological shield, especially in the case of FW-CADIS methodology implementation.

1 INTRODUCTION

The H.B.Robinson-2 Pressure Vessel Benchmark (HBR-2 benchmark) [1] has been chosen as a base for qualification of our methodology for reactor pressure vessel (RPV) and biological shield dose rates modelling using SCALE6.0 (Standardized Computer Analysis for Licensing Evaluation) code package [2] with FW-CADIS (Forward-Weighted Consistent Adjoint Driven Importance Sampling) methodology [3], as required by the U.S. Nuclear Regulatory Commission (U.S.NRC) Regulatory Guide 1.190 [4]. Calculational methods for determining the neutron fluence are necessary to estimate the fracture toughness of the pressure vessel materials, which is the main factor when considering lifetime extension for nuclear power plants.

The scope of the HBR-2 benchmark is to validate the capabilities of the calculational methodologies to predict the reaction rates (and specific activities) of the radiometric dosimeters irradiated in a surveillance capsule location of a downcomer (in-vessel) and in a cavity location (ex-vessel). The input data are addressing one octant of the HBR-2 core: reactor geometry, neutron source, material compositions, core power distribution and power history for the time of irradiation. The HBR-2 benchmark provides calculated reaction rates (TORT [5] results) and measured experimental data (specific activities) of the radiometric monitors at the end of irradiation during cycle 9 at the core midplane. This allows the...
assessment of the accuracy with which the calculations predict the neutron flux attenuation through the RPV. With HBR-2 benchmark as a "base case", we extended its geometry to the full-size typical PWR reactor including reactor internals, upper and lower RPV head and biological shield. Typical industrial and text-book data were used for dimensions and materials required. Next, critical neutron and photon fluxes and their dose rates in RPV and biological shield using the MAVRIC shielding sequence were studied. We also analysed the benefits of concrete boration, i.e. adding natural boron in small quantities to concrete in order to reduce its overall dose rates.

The SCALE6.0 code package, developed at Oak Ridge National Laboratory, was used for modelling of the typical PWR reactor based on HBR-2 critical core. Calculational results (reaction rates for six reactions) were compared with benchmark results obtained by TORT deterministic transport code. Satisfactory results were obtained and concrete boration showed significant reduction of neutron and gamma dose rates.

The description of the SCALE6.0 code package is given in Section 2, while HBR-2 benchmark with obtained results is given in Section 3. The analysis of the typical PWR reactor with HBR-2 core, including the results of dose rates calculations is given in Section 4. In Section 5 conclusion is presented, while the referenced literature is given at the end of the paper.

2 SCALE6.0 CODE PACKAGE

The SCALE6.0 code system was developed for the U.S.NRC to enable standardized analyses and evaluation of nuclear facilities. The criticality sequence (CSAS6) uses a 3D multi-group Monte Carlo (MC) transport code KENO-VI to provide automated, problem-dependent, cross-section processing followed by calculation of the neutron multiplication factor $k_{\text{eff}}$. KENO-VI has the ability to save fission distribution (in space and energy) of a critical system into a file (fissionSource.msm) over user-specified 3D mesh grid and energy structure of the cross section library. This methodology was implemented into SCALE6.0 code package to enable modelling criticality accident alarm systems (CAAS).

The MC shielding analysis capabilities in SCALE6.0 are based on Consistent Adjoint Driven Importance Sampling (CADIS) methodology which is used to create an importance map, i.e. weight windows in space and energy as well as biased source distribution [6,7]. Integrated Sn code DENOVO is used for automatic generation of space-energy variance reduction (VR) parameters over Cartesian mesh for the functional module MONACO. The MONACO is a multigroup fixed-source 3D MC transport code, which is used by MAVRIC (Monaco with Automated Variance Reduction using Importance Calculations) shielding sequence of the SCALE6.0 code package. Automated VR as well as manual VR can have source description in the form of KENO-VI (CAAS) fission source or with user defined source distribution in space and energy. When computing several tallies at once or a mesh tally over a large volume of space, an extension of the CADIS method called FW-CADIS can be used to obtain uniform relative uncertainties - multiple adjoint sources are weighted inversely by the expected tally forward value from DENOVO. For criticality safety analyses the v7-238 library was used, while for the shielding calculations v7-27n19g and v7_200n47g libraries were used. Primary data for both libraries originate from the ENDF/B-VII.0 nuclear data library [8].

3 HBR-2 BENCHMARK RESULTS

The HBR-2 facility is a 2300 MWth PWR reactor designed by Westinghouse. The reactor core consists of 157 fuel elements and it is surrounded by baffle plates, core barrel,
thermal shield, RPV and biological shield (thickness 175 cm). Selected general data for dimensions and material composition are given in reference [1]. The fuel elements (matrix 15x15) have an elaborate design, but for the purpose of the out-of-the-core transport calculations, they are approximated as homogenized regions. To complete the HBR-2 benchmark analysis the analyst must determine the multigroup and total neutron fluxes together with reaction rates for all the locations and all the dosimeters for which the referenced values are provided.

The CSAS6/KENO-VI criticality calculations of the extended HBR-2 core to full-sized PWR reactor was performed using 4050 batches (i.e. neutron generations) with first 50 batches skipped in order for the fission source distribution to converge and with 5000 neutrons per batch. The obtained effective multiplication factor of the reactor was $k_{\text{eff}} = (1.00012 \pm 0.00013)$. The calculated $k_{\text{eff}}$ satisfies the Pearson's chi-square ($\chi^2$) test for normality at the 95% level. The used library for criticality calculations was v7-238. The cycle-average critical boron concentration dissolved in water was 392 ppm. Total CPU time on QuadCore Q6600 with 8 GB of RAM was 81.21 min. With CSAS6 sequence the fission source distribution (CAAS source) file was generated which contains fission distributions in space and energy for non-skipping generations over user-defined 3D Cartesian mesh. Together with total neutron source strength of $1.77 \times 10^{20}$ n/s (known from thermal power of 2300 MWth), this mesh which overlays the reactor core, is then used as the source term in MAVRIC. The KENO-VI model of the extended HBR-2 facility is shown on Figure 1.

The MAVRIC shielding sequence was used for the calculation of reaction rates in downcomer and cavity locations. The neutron source was modelled in two different ways: as a CAAS source and as a space-flat source over critical core with Watt thermal fission spectrum distribution of $^{235}\text{U}$ ($a=1.028$ MeV, $b=2.249$/MeV). To simulate spatial gradient of neutrons in Watt case, i.e. self shielding of the source, the total neutron intensity was halved, which proved to be a conservative approach [9].
The DENOVO Sn mesh for VR implementation was also used as a mesh tally, covering only the active core height and biological shield. For high-energy threshold reactions it is important to have fine energy group spacing, but heterogeneous model of full-sized PWR reactor must have sufficiently dense Sn/tally mesh for successful MC convergence. Thus certain trade-off exists between spatial (size of Sn mesh) and energy (choice of shielding library) characterization of a model. For both neutron source descriptions we conducted MAVRIC calculations with these parameters:

- v7_27n19g shielding library with 80x80x30 Sn mesh (192000 cells), 2500 batches, 3000 neutrons per batch, DENOVO tolerance $\varepsilon = 10^{-3}$ and S8/P3
- v7_200n47g shielding library with 50x50x32 Sn mesh (80000 cells), 4000 batches, 3000 neutrons per batch, DENOVO tolerance $\varepsilon = 10^{-3}$ and S6/P2

Statistical quality of the MC results, i.e. average reaction rates, can be expressed through the term relative error on 1 sigma level, which are presented in Table 1. Typical running time for calculations with v7_27n19g library was 15-22 h, while for v7_200n47g was 17-27 h. Table 2 shows difference in reaction rates between MAVRIC and reference TORT results for downcomer and cavity regions, respectively. Generally, better results are obtained for v7_200n47g library, because nuclear reactions with high-energy threshold [2] demand fine group structure to adequately describe collapsed cross sections in MeV region. RPV dosimetry calculations should not be done with Watt source and broad shielding library, since effects of artificial source attenuation by factor 2 become clear. The benefit of v7_27n19g
library is in shortening of total MAVRIC CPU time, but at expense of poor representation of benchmark reactions cross sections in fast range. Precedence should be given to KENO-VI source with v7_200n47g library, if it is manageable by computer hardware.

Table 1: HBR-2 benchmark results

<table>
<thead>
<tr>
<th>XS library/ source type</th>
<th>Dosimeters location</th>
<th>Reaction Rate (s⁻¹ atom⁻¹) for reference TORT/DORT</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>^239Np(n,f)</td>
</tr>
<tr>
<td>BUGLE-93/ SAILOR-95/ BUGLE-96 *</td>
<td>Downcomer</td>
<td>1.06·10⁻¹³</td>
</tr>
<tr>
<td>BUGLE-93/ SAILOR-95/ BUGLE-96 *</td>
<td>Cavity</td>
<td>4.00·10⁻¹⁵</td>
</tr>
<tr>
<td>v7_27n19g CAAS</td>
<td>Downcomer</td>
<td>7.38·10⁻¹⁰</td>
</tr>
<tr>
<td>v7_200n47g CAAS</td>
<td>Downcomer</td>
<td>9.88·10⁻¹⁰</td>
</tr>
<tr>
<td>v7_27n19g Watt</td>
<td>Downcomer</td>
<td>7.93·10⁻¹⁴</td>
</tr>
<tr>
<td>v7_200n47g Watt</td>
<td>Cavity</td>
<td>7.38·10⁻¹⁰</td>
</tr>
<tr>
<td>v7_200n47g Watt</td>
<td>Cavity</td>
<td>5.02·10⁻¹⁵</td>
</tr>
</tbody>
</table>

(*) TORT average values (arithmetic mean) of the three libraries.

Since spatial neutron transport was limited only to active core height and biological shield, not the full-sized reactor, calculations with fine-group library were possible. However, this was not the case when performing coupled neutron-gamma dose rates calculations of RPV and biological shield for the entire reactor, since spatial effects become more prominent regarding the model heterogeneities [10]. Broad library v7_27n19g was then used exclusively. Overall satisfactory results were obtained for extended HBR-2 benchmark which indicate correctness of the geometry and successful implementation of FW-CADIS methodology.

Table 2: MAVRIC to TORT difference in reaction rates (%)

<table>
<thead>
<tr>
<th>XS library/ source type</th>
<th>Dosimeters location</th>
<th>^239Np(n,f)</th>
<th>^238U(n,f)</th>
<th>^58Ni(n,p)</th>
<th>^54Fe(n,p)</th>
<th>^46Ti(n,p)</th>
<th>^63Cu(n,α)</th>
</tr>
</thead>
<tbody>
<tr>
<td>v7_27n19g CAAS</td>
<td>Downcomer</td>
<td>-30.20</td>
<td>-35.74</td>
<td>-40.09</td>
<td>-36.28</td>
<td>-17.64</td>
<td>0.51</td>
</tr>
<tr>
<td>v7_200n47g CAAS</td>
<td>Downcomer</td>
<td>-6.46</td>
<td>-13.01</td>
<td>-12.83</td>
<td>-14.18</td>
<td>6.02</td>
<td>-10.32</td>
</tr>
<tr>
<td>v7_27n19g Watt</td>
<td>Downcomer</td>
<td>-12.22</td>
<td>-15.29</td>
<td>-18.29</td>
<td>-11.93</td>
<td>12.53</td>
<td>20.49</td>
</tr>
<tr>
<td>v7_200n47g Watt</td>
<td>Downcomer</td>
<td>-13.46</td>
<td>-16.12</td>
<td>-23.99</td>
<td>-15.34</td>
<td>5.81</td>
<td>-14.95</td>
</tr>
<tr>
<td>v7_200n47g Watt</td>
<td>Cavity</td>
<td>25.51</td>
<td>0.04</td>
<td>-7.25</td>
<td>1.20</td>
<td>29.71</td>
<td>-3.29</td>
</tr>
</tbody>
</table>
4 PWR REACTOR DOSE RATES CALCULATIONS

Our next step was to develop a shielding model of a typical PWR including biological shield and upper/lower core structures. For a convenience we have upgraded already developed HBR-2 benchmark model for the reactor core. The neutron and gamma dose rates were calculated in RPV and biological shield of typical PWR reactor with critical HBR-2 core. Neutron mesh-source was calculated prior by KENO-VI and the fission photons were internally added to the neutron source to account fissions of $^{235}$U. Fission photon multiplicity in ENDF/B-VII.0 for $^{235}$U is 7.04. The built-in response functions that converts flux-to-dose-rates were used for neutron (ID 9029) and gamma (ID 9504) calculations. These responses also serve as a forward weighting functions in FW-CADIS methodology. All these calculations are done only with v7_27n19g library. Gamma MC transport is actually coupled neutron-gamma transport, since generation of secondary gammas is included by radiative capture.

Initial results of critical neutron flux and MC relative errors (Figure 2) are for global cylindrical unit ($r=413.765$ cm, $Z_{\text{top}}=1340$ cm, $Z_{\text{bot}}=-800$ cm) with vacuum boundary and they demonstrate the MAVRIC’s FW-CADIS versatile ability to calculate heavy shielding problem in reasonable time. Adjoint source was defined as biological shield which was inversely weighted with expected DENOVO dose rates. Sn mesh over global unit included $60\times60\times90$ cells. Quadrature and Legendre order was $S_6/P_2$. Nine million histories were used and CPU time was 15 h. One can observe that areas with smallest relative uncertainty correspond to biological shield, which is region of highest importance. White regions represent space which is excluded by MAVRIC due to its negligible contributions to the tallies in the area of interest.

If the forward-weighting scheme with approximated DENOVO tally value is deactivated, uniform convergence of MONACO mesh tally to the same relative uncertainty is not possible. In this CADIS case, a uniform adjoint source strength is defined over the entire mesh tally volume (shield), so areas of low flux (dose rates) would have less adjoint source strength than areas with high flux (dose rates). Figure 3 shows repeated initial results of critical neutron flux and MC relative errors but with CADIS methodology. Clearly, there is no uniform flux convergence, neither obtaining global results with small MC uncertainties.
The neutron and gamma dose rate calculations are addressing the ability to reduce the neutron and gamma fluxes in biological shield by adding small amount of natural boron [11]
or boron-carbide (B₄C) [12] to concrete. The fast flux reaching the concrete walls in the RPV cavity will undergo neutron thermalization resulting in radiative neutron capture by impurities. Boration of shielding materials is an attractive way to capture neutrons without creating secondary gamma radiation by means of \(^{10}\text{B}(n,\alpha)^{7}\text{Li}\) reaction, since alpha particles do not present radiation hazard. The proposed quantities of natural boron to be added in concrete were 0 w/o, 0.3 w/o, 0.7 w/o, 1 w/o and 2 w/o. Also B₄C was investigated with concentrations of 5 w/o and 20 w/o. The PWR reactor geometry is now axially-reduced only to regions where biological shield exist, below RPV upper head (Z_{top}=650 cm). Adjoint source is still concrete in global unit, the Sn mesh was 55x55x80 cells with S₆/P₂, but 42 million histories were used and CPU time was about 37 h. It is evident from initial shielding results that the maximum neutron flux on biological shield is obtained in core midplane (z=0), so it was of interest to find the radial profile of neutron and gamma dose rates for all borating cases, as well as their axial profiles on inner surface of the biological shield (i.e. the first 15 cm depth).

Figures 4 and 5 are showing radial neutron and gamma dose rate profiles at core midplane and for all borating cases (cavity concrete wall is starting at 318.13 cm radial from core). Figures 6 and 7 are showing axial neutron and gamma dose rate profiles in the 15 cm concrete wall depth and for all borating cases (active core height is ± 182.88 cm axial). Since boration with 0 w/o natural boron and 20 w/o B₄C forms an envelope of dose rates in all cases, we see that increasing concentration of boron dose not imply lower dose rates, so some kind of saturation is observed. Results with 0.7 w/o of natural boron would be satisfactory. It is in accordance with proposed values by other authors [13].

![Figure 4: Radial neutron dose rate profiles in rem/h for core midplane (z=0)](image-url)
Figure 5: Radial gamma dose rate profiles for core midplane ($z=0$)

Figure 6: Axial neutron dose rate profiles in first 15 cm of concrete
Comparison of the results with and without boron shows the expected decrease in neutron and gamma dose rates by factor-of-few to 10 times, with boron in the biological shield. The average values with 1 sigma bars are depicted for every 15 cm radial and 18 cm axial. Results for the external surface of biological shield (175 cm thickness) have much higher errors and are shown mainly for the trends. The dose rates on inner surface of the non-borated shield are 50 times higher for neutrons than for gammas.

5 CONCLUSIONS

A detailed shielding analysis of the typical PWR reactor with critical HBR-2 core was performed. Validation and verification of extended geometry was done by repeating HBR-2 benchmark reaction rates for six nuclear reactions in downcomer and cavity regions. Satisfactory results were obtained which indicate correctness of the geometry and successful implementation of FW-CADIS methodology of SCALE6.0 code package. A carefully constructed model of the typical PWR reactor was developed to analyse attenuation of neutron and gamma fluxes through structural components and concrete using MAVRIC shielding sequence. The dose rates were obtained by using built-in response functions for neutrons (ID 9029) and photons (ID 9054). The analysis examined the impact of borating the cavity concrete wall surrounding the RPV. A small amount of natural boron or boron-carbide (B₄C) was added to the concrete in a homogenous fashion to observe net effects of ¹⁰B capturing thermal neutrons by ¹⁰B(n,α)⁷Li reaction. The results showed that boration makes a considerable reduction in the neutron and gamma dose rates level of concrete, i.e. accelerating the attenuation of neutron and gamma fluxes through shield by several to 10 times. Since the location of the maximum thermal flux is on the thin layer of the cavity shield surface, only a small amount of concrete mass would need to be borated for the practical purpose. Borating of the concrete used in the biological shield would result in a reduction of the total dose during power plant lifetime and subsequent lower decommission costs.
REFERENCES


