Modeling of H.B.Robinson-2 Pressure Vessel Benchmark

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ABSTRACT

H.B.Robinson-2 Pressure Vessel Benchmark (HBR-2 benchmark) has been chosen for qualification of our methodology for pressure vessel neutron fluence calculations, as required by the U.S. Nuclear Regulatory Commission Regulatory Guide 1.190. The SCALE 6.0 code package was used for modeling of the HBR-2 benchmark. The CSAS6 sequence of the SCALE code package, which includes KENO-VI 3D Monte Carlo code, was used for criticality calculation and for generation of the neutron fission distribution file used as a neutron source-term in later shielding calculation. The ORIGEN-ARP/SCALE isotopic depletion and decay analysis sequence was used with point-depletion code ORIGEN-S to find radioactive decay neutron sources which originate from burned fuel assemblies. The MAVRIC/MONACO Monte Carlo shielding sequence with manual and automatic adjoint-based reduction variance (CADIS) techniques was used for calculation of the critical neutron fluxes in downcomer and cavity regions of H.B.Robinson-2 model. The total neutron fluxes comprise of critical and decay neutron fluxes. The comparison of preliminary calculational results and benchmark results obtained with DORT deterministic transport code showed a good agreement of the total neutron fluxes in downcomer and cavity regions.

1 INTRODUCTION

Calculational methods for determining the neutron fluence are necessary to estimate the fracture toughness of the pressure vessel materials. The H.B.Robinson-2 Pressure Vessel Benchmark (HBR-2 benchmark) [1] has been chosen for qualification of our methodology for pressure vessel neutron fluence calculations, as required by the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.190 [2]. The scope of HBR-2 benchmark is to validate the capabilities of the calculational methodologies to predict the 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 provided consist of reactor geometry, neutron source, material compositions, core power distribution and power history for the time of irradiation. The HBR-2 benchmark provides measured experimental data which are specific activities of the radiometric monitors irradiated in a surveillance capsule location of a downcomer (in-vessel) and in a cavity location (ex-vessel).

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allows the assessment of the accuracy with which the calculations predict the neutron flux attenuation inside the pressure vessel.

The SCALE 6.0 (Standardized Computer Analysis for Licensing Evaluation) [3] code package, developed at Oak Ridge National Laboratory, was used for modeling of the HBR-2 benchmark. Preliminary calculational results were then compared with benchmark results obtained using DORT deterministic transport code.

The HBR-2 benchmark is described in Section 2, while the description of the SCALE 6.0 code package is given in Section 3. The analysis of the HBR-2 benchmark, including calculational models, results of calculations, and comparison of calculated values with benchmark data, is given in Section 4. In Section 5 conclusion is presented, while the referenced literature is given at the end of the paper.

2 HBR-2 BENCHMARK DESCRIPTION

The HBR-2 facility is a 2300 MWth PWR reactor designed by Westinghouse and placed in operation in March of 1971. It is owned by Carolina Power and Light Company. The reactor core of the HBR-2 consists of 157 fuel elements and is surrounded by baffle plates, core barrel, thermal shield, reactor pressure vessel (RPV) and biological shield. Selected general data for dimensions and material composition are given in reference [1]. A quadrant of the horizontal cross section of the reactor is shown on Figure 1, with locations of the downcomer dosimeters (azimuthal angle 20°) and cavity dosimeters (azimuthal angle 0°).

![Figure 1: Horizontal cross section (midplane) of the HBR-2 benchmark facility](image)

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. A low-leakage core loading pattern was used for cycle 9 with 12 previously burned elements on a core periphery. Measured quantities are specific activities of the downcomer and cavity dosimeters irradiated on core midplane. To complete the HBR-2 benchmark analysis the
analyst must determine the multigroup and total neutron fluxes for all the locations and all the
dosimeters for which the measured values are provided.

3 SCALE 6.0 CODE PACKAGE [3]

The SCALE 6.0 code system was developed for the U.S. Nuclear Regulatory
Commission to enable standardized analyses and evaluation of nuclear facilities. The system
has the capability to perform criticality, shielding, radiation source term, spent fuel
depletion/decay, reactor physics, and sensitivity analyses using well established functional
modules.

The Criticality Safety Analysis Sequence No. 6 (CSAS6) uses a 3D multi-group Monte
Carlo transport code KENO-VI to provide automated, problem-dependent, cross-section
processing followed by calculation of the neutron multiplication factor. Also, KENO-VI has
the ability to save fission distribution (in space and energy) of a critical system into a file
(fissionSource.msm) using user-specified 3D mesh grid and energy structure from the cross
section library. The methodology is implemented into SCALE 6.0 code package to enable
modeling criticality accident alarm systems (CAAS).

ORIGEN-ARP is a sequence of the SCALE 6.0 code system used to perform rapid and
accurate point-depletion and decay calculations. The ARP module interpolates pregenerated
cross-sections libraries to create problem-dependent cross sections for use with ORIGEN-S
code.

The Monte Carlo shielding analysis capabilities in SCALE 6.0 are based on Consistent
Adjoint Driven Importance Sampling (CADIS) methodology which is used to create an
importance map for weight windows in space and energy as well as biased source
distribution. The functional module Monaco is a multigroup fixed-source 3D Monte Carlo
transport code, which is used by MAVRIC (Monaco with Automated Variance Reduction
using Importance Calculations) shielding sequence of the SCALE 6.0 code package.
MAVRIC also enables manual variance reduction (RV) based on Russian roulette and
splitting methods. Automated RV (CADIS methodology), as well as manual RV can have
source description in the form of fission source distribution mesh originating from 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 more uniform relative
uncertainties - multiple adjoint sources weighted inversely by the expected tally forward
value.

There are several multigroup cross section libraries distributed within SCALE 6.0 for
criticality safety analyses and shielding calculations. For criticality safety analyses the V7-238
library was used, while for the shielding calculations V7-27n19g library was used. Primary
data for both libraries originate from the latest ENDF/B-VII nuclear data library [4].

4 ANALYSIS OF THE HBR-2 BENCHMARK

The analysis of the HBR-2 benchmark has been performed using the SCALE 6.0 code
package. The calculational models of the HBR-2 benchmark within CSAS6, ORIGEN-ARP
and MAVRIC sequences of SCALE 6.0 code package have been determined. The preliminary
results of the calculations, i.e. total neutron fluxes in downcomer and cavity regions, using the
established models, have been compared with the HBR-2 calculated benchmark data (DORT).
4.1 Calculational model for CSAS6 sequence

The CSAS6/KENO-VI criticality calculations of the HBR-2 benchmark facility was performed using 4000 batches (i.e. neutron generations) with first 300 batches skipped in order for the fission source distribution to converge and with 5000 neutrons per batch. The obtained effective multiplication factor of 1/4 of the core with white boundary conditions (isotropic return) was $k_{\text{eff}} = (0.99971\pm0.00014)$. The calculated $k_{\text{eff}}$ satisfies the Pearson's chi-square ($\chi^2$) test for normality at the 95% level. Total CPU time on QuadCore Q6600 PC equipped with 4 CPUs (2.4 GHz) and 8 GB of RAM was 85.11 min. With CSAS6 sequence the fission source distribution 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 $4.42\cdot10^{19}$ n/s (known from thermal power), this mesh which overlays the reactor core, is then used as the source term in MAVRIC. The KENO-VI model of the HBR-2 facility is shown on Figure 2 and fission source distribution from criticality calculation is depicted on Figure 3.

Figure 2: KENO-VI model for a HBR-2 facility

Figure 3: Fission source distribution
4.2 Calculational model for ORIGEN-ARP sequence

Previously burned fuel elements together with fresh ones comprise radioactive decay neutron sources of core periphery, which has to be accounted for in the calculation of the total neutron fluxes on dosimeters. ORIGEN-ARP burnup/depletion code was used to find neutron spectrum and total neutron sources for fuel elements in the proximity of the dosimeters. That data was used as a source term in MAVRIC calculations, i.e. radioactive neutron fluxes from selected burned fuel elements with dominant influence on dosimeters (19, 30, 43, 56, 70, 71, 85, 86) have been determined (Figure 1). The selection of used elements has been based on engineering judgment and was calculationally verified.

4.3 Calculational model for MAVRIC sequence

4.3.1. Radioactive neutron fluxes from burned fuel

Selected fuel elements on the reactor core periphery were identified as neutron sources (ORIGEN-ARP), and for each fuel element the MAVRIC sequence was used for determination of neutron fluxes in cavity and downcomer region (Table 1). MAVRIC calculation with FW-CADIS was used to construct importance map which is used by Monaco Monte Carlo. Calculation is based on Sn mesh 56x60x32 cm with total 107520 voxels (volumetric pixel i.e. volumetric picture element). Neutron histories with 500 batches and 8000 neutrons per batch were set for all fuel elements, except for boundary elements 85 and 86, which required 3 times and 6 times more histories, respectively. Examples of MAVRIC results for fuel element 71 are depicted on Figures 4 and 5. Adjoint and forward fluxes for fuel element 71 are shown on Figure 4, while neutron flux and its relative uncertainty are shown on Figure 5.

Table 1: Neutron fluxes from burned fuel

<table>
<thead>
<tr>
<th>Fuel element</th>
<th>CPU time (hr)</th>
<th>Cavity dosimeter</th>
<th>Downcomer dosimeter</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>$\bar{\phi}$ (n/cm$^2$s)</td>
<td>$\sigma$ (n/cm$^2$s)</td>
</tr>
<tr>
<td>86</td>
<td>5.10</td>
<td>$5.28 \times 10^5$</td>
<td>$9.56 \times 10^4$</td>
</tr>
<tr>
<td>71</td>
<td>5.07</td>
<td>$3.26 \times 10^6$</td>
<td>$2.14 \times 10^5$</td>
</tr>
<tr>
<td>56</td>
<td>7.63</td>
<td>$5.97 \times 10^4$</td>
<td>$4.36 \times 10^3$</td>
</tr>
<tr>
<td>43</td>
<td>6.63</td>
<td>$1.58 \times 10^6$</td>
<td>$5.24 \times 10^5$</td>
</tr>
<tr>
<td>30</td>
<td>4.64</td>
<td>$1.29 \times 10^6$</td>
<td>$9.32 \times 10^4$</td>
</tr>
<tr>
<td>19</td>
<td>5.04</td>
<td>$3.78 \times 10^5$</td>
<td>$3.43 \times 10^4$</td>
</tr>
<tr>
<td>85</td>
<td>10.65</td>
<td>$3.18 \times 10^5$</td>
<td>$1.38 \times 10^4$</td>
</tr>
<tr>
<td>70</td>
<td>5.24</td>
<td>$9.69 \times 10^4$</td>
<td>$1.85 \times 10^3$</td>
</tr>
<tr>
<td>sum</td>
<td>50</td>
<td>$4.03 \times 10^6$</td>
<td></td>
</tr>
</tbody>
</table>
4.3.2. Critical neutron fluxes from reactor core

Modeling of the HBR-2 facility with MAVRIC sequence has been performed using 3000 batches and 8000 neutrons per batch for 1/4 core with white boundary (isotropic return). All the calculations have been performed using V7-27n19g cross section library. Four different approaches have been analyzed:

1. manual RV with CAAS fission source description (manual RV),
2. automated RV using FW-CADIS methodology with user defined neutron source distribution in space and energy (adjoint RV Watt),
3. automated RV using FW-CADIS methodology with CAAS fission source description (adjoint RV FW-CADIS),
4. automated RV using CADIS methodology with CAAS fission source description (adjoint RV CADIS).

First case is a standard Monte Carlo calculation with Russian roulette and splitting, without importance map calculation and using manually defined transport weight windows. Fission distribution in space and energy is determined by KENO-VI calculation and used as a source term for MAVRIC.

Second case is adjoint and forward driven importance map calculation for MAVRIC shielding sequence with user-provided description of the source in space and energy. The source distribution is defined using the thermal strength of the system being analyzed and Watt spectrum for thermal fission of $^{235}$U with standard parameters ($a = 1.028$ MeV, $b = 2.249$ /MeV). Source is sampled from cylindrical body containing the core, with restriction to mixture 1 (homogenized core). MAVRIC calculation is based on Sn mesh 56x60x32 cm with total 107520 voxels.

Third and fourth cases use estimated adjoint (and forward) fluxes for an automated preparation of transport weight windows for MAVRIC. Source term is a fission distribution from KENO-VI calculations. MAVRIC calculation is based on Sn mesh 56x60x32 cm with total 107520 voxels.

Statistical quality (convergence) of the Monte Carlo calculations can be expressed through the term relative error on 1 sigma level. Table 2 shows critical fluxes and their uncertainty. Relative error of MAVRIC calculations using manual variance reduction is much higher than adjoint ones.

Figures 6, 7 and 8 are presenting the results of FW-CADIS (adjoint) methodology for case 3. On Figure 6 adjoint flux and biased source distribution are given. Figure 7 presents...
forward flux and mesh importance map for group 1 (20 MeV - 6.37 MeV), while on Figure 8 neutron flux and its relative uncertainty are shown. Similar figures are obtained for cases 2 and 4, which also employ adjoint RV. Case 1 which employs manual RV has different trend in neutron transport (Figure 9): there is no directional behavior of neutrons towards important regions, i.e. cavity and downcomer. In this case Monte Carlo method can not properly establish the “importance” of space resulting in inadequate usage of CPU time on unimportant regions of space. This is evident in Table 2.

Table 2: Neutron fluxes from critical core

<table>
<thead>
<tr>
<th>Case</th>
<th>CPU time (hr)</th>
<th>Cavity dosimeter</th>
<th>Downcomer dosimeter</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>φ (n/cm²s)</td>
<td>σ (n/cm²s)</td>
</tr>
<tr>
<td>1.</td>
<td>84.08</td>
<td>3.77·10⁹</td>
<td>1.79·10⁴</td>
</tr>
<tr>
<td>2.</td>
<td>11.27</td>
<td>1.53·10¹⁰</td>
<td>8.14·10⁸</td>
</tr>
<tr>
<td>3.</td>
<td>11.18</td>
<td>4.23·10⁹</td>
<td>5.39·10⁸</td>
</tr>
<tr>
<td>4.</td>
<td>14.92</td>
<td>4.02·10⁹</td>
<td>4.85·10⁸</td>
</tr>
</tbody>
</table>

Figure 6: Adjoint flux and biased source distribution for case 3

Figure 7: Forward flux and mesh importance map for case 3 for group 1 (20 MeV - 6.37 MeV)
4.4 Total neutron fluxes

The total neutron fluxes comprise of critical and decay neutron fluxes. The results of the calculations (Table 1 and Table 2) show marginal contribution of radioactive decay fluxes from burned fuel elements on core periphery relative to critical flux. Total neutron fluxes obtained by SCALE 6.0 are given in Table 3. Although, the difference percentage for cavity ranges from -19% to -80%, and for downcomer ranges from -17% to 92%, it has to be noted...
that the details of reference DOR T calculations (DORT fluxes are $1.88 \cdot 10^{10}$ n/cm$^2$s for cavity and $2.92 \cdot 10^{11}$ n/cm$^2$s for downcomer region) are either missing or are vaguely described in reference [1]. Therefore we conclude that a performed SCALE 6.0 analysis is justified. However, SCALE 6.0 results should be treated as preliminary ones since their comparison to measured HBR-2 data was not possible due to lack of information regarding dosimeter specifications in reference [1].

Table 3: Total neutron fluxes for dosimeters

<table>
<thead>
<tr>
<th>Case</th>
<th>Total neutron flux (n/cm$^2$s)</th>
<th>Difference to DORT (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>cavity</td>
<td>downcomer</td>
</tr>
<tr>
<td>1</td>
<td>$3.774 \cdot 10^9$</td>
<td>$2.421 \cdot 10^{11}$</td>
</tr>
<tr>
<td>2</td>
<td>$1.530 \cdot 10^{10}$</td>
<td>$5.601 \cdot 10^{11}$</td>
</tr>
<tr>
<td>3</td>
<td>$4.234 \cdot 10^9$</td>
<td>$1.661 \cdot 10^{11}$</td>
</tr>
<tr>
<td>4</td>
<td>$4.024 \cdot 10^9$</td>
<td>$1.701 \cdot 10^{11}$</td>
</tr>
</tbody>
</table>

An important issue that has to be addressed when discussing Monte Carlo method is a burden the applied calculational model will place on CPU time and memory resources. All calculations have been performed on QuadCore Q6600 PC equipped with 4 CPUs (2.4 GHz), and 8 GB of RAM.

For MAVRIC sequence with different approaches the Figure-of-Merit (FOM) factor [5] is defined to account for the time it takes to achieve a given level of uncertainty: $FOM = 1/\text{CPU time}/(\text{relative error})^2$. The FOM factor can easily be used to introduce another quality parameter, so-called Speed-up [6], addressing benefits of using adjoint-based RV compared to manual RV techniques. Speed-up is defined as a ratio of adjoint FOM factor and manual FOM factor. FOM factor and speed-up for different MAVRIC approaches are given in Table 4. Application of automated RV technique based on adjoint fluxes significantly improves the quality of MAVRIC Monte Carlo calculations, evident from speed-up ranging from ~100 to more than ~100000.

Table 4: FOM factors and speed-ups

<table>
<thead>
<tr>
<th>Case</th>
<th>Sn mesh x<em>y</em>z</th>
<th>Batches/Neutrons</th>
<th>CPU time (hr)</th>
<th>FOM factor</th>
<th>Speed-up</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>cavity</td>
<td>downcomer</td>
</tr>
<tr>
<td>1</td>
<td>n/a</td>
<td>3000/8000</td>
<td>84.08</td>
<td>0.0527</td>
<td>0.0368</td>
</tr>
<tr>
<td>2</td>
<td>56<em>60</em>32</td>
<td>3000/8000</td>
<td>11.27</td>
<td>31.234</td>
<td>130.255</td>
</tr>
<tr>
<td>3</td>
<td>56<em>60</em>32</td>
<td>3000/8000</td>
<td>11.18</td>
<td>5.519</td>
<td>31.485</td>
</tr>
<tr>
<td>4</td>
<td>56<em>60</em>32</td>
<td>3000/8000</td>
<td>14.92</td>
<td>4.608</td>
<td>4189.008</td>
</tr>
</tbody>
</table>

5 CONCLUSIONS

The analysis of the HBR-2 benchmark has been performed using the SCALE 6.0 code package. The calculational models of the HBR-2 benchmark within CSAS6, ORIGEN-ARP and MAVRIC sequences of SCALE 6.0 have been determined. The results of calculations using the established models have been compared with HBR-2 benchmark data.

The comparison of preliminary calculational results with MAVRIC/MONACO Monte Carlo shielding code and benchmark results with DORT deterministic transport code showed that the performed SCALE 6.0 analysis is justified and applicable for deep penetration problems.

REFERENCES


