VVER-440 Local Power Peaking Experiment Benchmark

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Confidentiality: Public
Summary

Working group 3 of the IAEA Regional Project RER/9/076 was established to assess the local power peaking problem encountered in VVER-440 reactors. This problem is caused by the excessive neutron moderation in the joint region between a control element and the attached follower fuel assembly. The use of absorber plates has been studied as a means to suppress the peaked thermal flux and the increased fission power in the adjacent fuel pins. Irradiation studies using a VVER-440 control assembly model were performed by the NRI Řež plc. in Czech Republic at the LR-0 reactor between December 2004 and July 2005. A computational benchmark exercise was established in July 2006.

VTT Technical Research Centre of Finland and Fortum Nuclear Services have participated in the WG3 activities since the beginning of the project. This report presents the benchmark calculations carried out at VTT using two Monte Carlo neutron transport codes: the widely used MCNP4C developed at Los Alamos National Laboratory and PSG, which is a new reactor physics code under development at VTT.
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1 Introduction

The VVER-440 control assembly consists of an absorber element attached to a full-length follower fuel assembly, which is inserted in the reactor core as the absorber is lifted. The support structures in the joint region between the absorber and the follower contain relatively large volumes of water, which provides for excess moderation and induces a localised power peak in the adjacent fuel assemblies. This power peak may cause additional strain in the materials in certain operating conditions, which increases the probability of cladding failure. One solution to the problem is to attach absorber plates in the flow channel walls, which reduces the outward streaming of thermalised neutrons from the joint region to the surrounding fuel assemblies.

Working group 3 of the IAEA RER/9/076 Regional Project was established for assessing the localised power peaking problem. Among participants from several countries using VVER-440 technology, VTT Technical Research Centre of Finland and Fortum Nuclear Services have participated in the activity. An experimental programme was set up by the NRI Rež plc. in Czech Republic. Experimental fission rate measurements using a VVER-440 control assembly model were performed at the LR-0 reactor between December 2004 and July 2005.

The project has reached its final stage and a computational benchmark exercise was established in July 2006 [1]. This report presents the benchmark calculations carried out at VTT using two Monte Carlo neutron transport codes: the widely used MCNP4C developed at Los Alamos National Laboratory and PSG, which is a new reactor physics code under development at VTT.

2 Benchmark Calculations

The full specifications of geometry and material compositions are found in Reference [1] and the details are not repeated here. A brief description is given in the following. The MCNP and PSG geometry models are constructed using the same assumptions. The input formats are very similar, which reduces the possibility of unintentional discrepancies between the two models.

2.1 Benchmark Model

The pool-type LR-0 reactor was loaded with fuel assemblies comprised of fuel pins enriched from 2.0 to 3.3 wt-% U-235. The dimensions of the assemblies are similar to a real VVER-440 fuel, with the active height reduced to 125 cm. Another difference is that the central control assembly is inverted: the follower assembly sits on the top of the absorber element. The central position is surrounded by 18 fuel assemblies. The outermost assemblies are only partially loaded with fuel pins.

The absorber plates used for suppressing the thermal flux are made of hafnium. The plate is 76 mm wide, 150 mm high and 0.6 mm thick. The six inserts are modelled as a uniform hexagonal sleeve in the inner face of the shroud tube. The irradiation experiments were carried out with and without
Table 1. Calculation cases and critical moderator levels.

<table>
<thead>
<tr>
<th>Variant</th>
<th>Description</th>
<th>Critical water level</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>No boron in water, no hafnium inserts</td>
<td>71.750 cm</td>
</tr>
<tr>
<td>2</td>
<td>No boron in water, with hafnium inserts</td>
<td>77.000 cm</td>
</tr>
<tr>
<td>3</td>
<td>2.506 g/kg boron in water, with hafnium inserts</td>
<td>112.816 cm</td>
</tr>
<tr>
<td>4</td>
<td>2.506 g/kg boron in water, no hafnium inserts</td>
<td>111.370 cm</td>
</tr>
</tbody>
</table>

Table 1. Calculation cases and critical moderator levels.

The calculations were carried out using two transport calculation codes based on the Monte Carlo method. Although the codes have certain similarities, there are some crucial differences as well.

2.2 Calculation Tools

The calculations were carried out using two transport calculation codes based on the Monte Carlo method. Although the codes have certain similarities, there are some crucial differences as well.

2.2.1 MCNP4C

The main calculation tool used in this study was the Los Alamos MCNP Monte Carlo particle transport code. MCNP is a versatile general-purpose transport code featuring continuous-energy cross section data, generalised geometry description and the capability to perform neutron, photon and electron transport calculation. MCNP has been widely used for this type of calculations before, and it was likely that the code would yield good results compared to the experimental measurements, provided that the geometry model is sufficiently accurate. The most recent version of the code, MCNP5, was not available at VTT at the time of this study and the calculations were carried out using an older 4C version of the code [2].

2.2.2 PSG

PSG (Probabilistic Scattering Game) is a new Monte Carlo reactor physics code, developed at VTT since 2004. PSG can be roughly characterised as a three-dimensional continuous-energy neutron transport code, although certain special features make it particularly suitable for reactor calculations at the fuel assembly level. The main motivation for PSG development is the generation of homogenised group constants for deterministic reactor simulator codes using the Monte Carlo method. Group constant generation is also one of the topics in a related Ph.D. study, to be completed in 2007 [3]. Although specifically intended for infinite lattice calculations, PSG can basically be used for any reactor problem involving a self-sustaining fission chain reaction.

PSG uses ENDF format interaction data, read from ACE format cross section libraries. The library format is shared with MCNP and some of the interaction physics is handled in a similar manner as
well. The most significant methodological difference to MCNP is that PSG uses the delta-tracking method [4] for neutron transport. This method is essentially a rejection technique that eliminates the need to stop the random walk each time the neutron crosses the boundary between two material regions. The geometry routines are considerably simplified, as there is no need to calculate any surface distances either. This may result in a major speed-up in the tracking process in complex geometries.

The delta-tracking method also has its downsides. The track length estimate of neutron flux is not available and integral reaction rates have to be calculated using the collision estimator. The two methods are equally efficient in the typical applications of PSG, i.e. when reaction rates are integrated over the entire lattice geometry. The efficiency of the collision estimator becomes poor in regions where the total reaction rate density is low or when the integration is carried over a small volume. This is especially the case in detector-type calculations, in which the detectors may be located far or isolated from the active core region[1]. The fission rate calculations in this study involve relatively small volumes, but in a region of high collision density. It is therefore particularly interesting to see how the PSG code performs compared to MCNP, which uses the track length estimate for the fission rate tallies.

### 3 Results

Both MCNP4C and PSG were run in the $k$-eigenvalue criticality source mode with 1000 active and 100 inactive cycles of 20000 source neutrons. Two cross section libraries, one based on the ENDF/B-VI.8 evaluated nuclear data file and the other based on JEFF-3.1 data were used in the calculations. Both codes used the same ACE format library files, which significantly reduces the uncertainties between the two codes by eliminating all discrepancies originating from the fundamental interaction data.

#### 3.1 Criticality Calculations

The results of MCNP and PSG criticality calculations are summarised in Table 2. It seems that both codes tend to underestimate the multiplication factor by approximately 600 pcm when ENDF/B-VI.8 based cross section libraries are used. The values are closer to criticality with JEFF-3.1 data. A similar reactivity under-prediction is observed in the example MCNP4C and MCNP5 calculations in the benchmark specification [1][2].

The discrepancies between the two cross section libraries clearly exceed the differences between the two codes. Previous studies have shown that MCNP and PSG results generally differ by less than 100

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[1] PSG has been validated in VENUS-2 MOX-fuelled reactor dosimetry benchmark calculations [5]. It turned out that the efficiency of the collision estimator was quite poor in detector cells located far from the active core region. It was possible to obtain statistically converged results, but the required number of neutron histories was far too large for any practical application.

[2] Both ENDF/B-VI.2 and ENDF/B-VI.6 based cross section libraries were used in the example calculations. The multiplication factors range from 0.98880 to 0.99430. It is a commonly known fact that ENDF/B-VI based cross section libraries tend to under-estimate $k_{\text{eff}}$ in LWR calculations.
Table 2. Results of criticality calculations.

<table>
<thead>
<tr>
<th>Variant</th>
<th>ENDF/B-VI.8</th>
<th>JEFF-3.1</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>MCNP</td>
<td>PSG</td>
</tr>
<tr>
<td>1</td>
<td>0.99368 (0.00016)</td>
<td>0.99304 (0.00020)</td>
</tr>
<tr>
<td>2</td>
<td>0.99235 (0.00017)</td>
<td>0.99336 (0.00021)</td>
</tr>
<tr>
<td>3</td>
<td>0.99291 (0.00016)</td>
<td>0.99344 (0.00019)</td>
</tr>
<tr>
<td>4</td>
<td>0.99453 (0.00016)</td>
<td>0.99451 (0.00019)</td>
</tr>
</tbody>
</table>

pcm in LWR calculations, which is also the case for most of the values in Table 2. The systematic under-prediction of the multiplication factor may result from a flaw in the geometry model, but it may also originate from the evaluated nuclear data.

3.2 Reaction Rate Distributions

The benchmark specification [1] provides measured fission rates at five pin positions near the central control assembly (see Figure 1). The results are given as axial distributions relative to the distribution at position M4.

The MCNP model uses the standard F4 reaction rate tally in the nested assembly-pin lattice for calculating the fission rate distributions. The selected fuel pins are divided into 50 equally-spaced axial segments between $z = 8$ cm and $z = 104$ cm, where the origin is set to the active fuel bottom, as in the benchmark specification. Due to the symmetry of the core, a 30° sector of the full geometry was used in the calculations to attain better tally statistics.

The PSG geometry description is very similar to MCNP, and it is unlikely that there are any significant differences between the models. The fission rates are calculated inside cylindrical cells, superimposed over the selected fuel pins. The cells are divided into 50 axial segments, as in the MCNP model. PSG was run using a 30° symmetry option for improving the statistics.

Results calculated by the two codes using ENDF/B-VI.8 data libraries are plotted in Figures 2–5 and results calculated using JEFF-3.1 data in Figures 6–9. The error bars show the $1\sigma$ (68%) confidence intervals. The axial power peak is clearly visible at pin position M1, especially without boronisation. The hafnium inserts reduce the peaking factor from about 1.6 to less than 1.4.

All calculated distributions seem to be quite consistent with the measured results and there are no major differences between the two cross section libraries. The axial power peak at position M1 is slightly over-estimated compared to the measured results when the hafnium plates are not used. Other systematic discrepancies are observed as well, but the results are generally in a good agreement.

The standard deviations in the PSG results are about 50% larger than the MCNP error estimates. This is an anticipated result, owing to the use of the less efficient collision estimate compared to the track length estimate used by MCNP. The differences, however, are not as large as was initially expected,
Figure 1. Positions of the measured fuel pins relative to the central control assembly (CA). Figure taken from Reference. [1].

Based on previous detector calculations [5]. The efficiency problems of the collision estimator become apparent when the reaction rates are calculated above the critical water level, where the collision rates are low. The error bars show a significant increase in the dry lattice values of variants 1 and 2.

4 Summary and Conclusions

The VVER-440 local power peaking experiments carried out at the NRI Řež plc. in Czech Republic were modelled using Monte Carlo codes MCNP4C and PSG. The measured results consist of axial fission rates at selected pin positions near the joint region of a central control assembly. The exper-
ments were carried out using fresh and boronised water and with and without hafnium inserts used for suppressing the localised thermal flux peak.

All calculated results are in a good agreement with the measured distributions, although there are some small systematic discrepancies as well. One of the most interesting results of this study is that the PSG code is able to calculate the pin-wise fission rate distributions with reasonable statistical accuracy. This was not completely clear at the beginning, since the code uses the collision estimate of neutron flux for calculating integral reaction rates. The method is known to pose some efficiency problems in detector-type calculations, in which reaction rates are integrated over small volumes in regions of low neutron activity. The detector regions in this case were located within the active core and the high collision rate compensates for the small volume.

References


Figure 2. Variant 1 (no boron, no hafnium inserts), ENDF/B-VI.8 data.

Figure 3. Variant 2 (no boron, with hafnium inserts), ENDF/B-VI.8 data.
Figure 4. Variant 3 (2.506 g/kg boron, with hafnium inserts), ENDF/B-VI.8 data.

Figure 5. Variant 4 (2.506 g/kg boron, no hafnium inserts), ENDF/B-VI.8 data.
**Figure 6.** Variant 1 (no boron, no hafnium inserts), JEFF-3.1 data.

**Figure 7.** Variant 2 (no boron, with hafnium inserts), JEFF-3.1 data.
**Figure 8.** Variant 3 (2.506 g/kg boron, with hafnium inserts), JEFF-3.1 data.

**Figure 9.** Variant 4 (2.506 g/kg boron, no hafnium inserts), JEFF-3.1 data.