Comparison of the ENDF/B-VII.0, ENDF/B-VII.1, ENDF/B-VIII.0 and JEFF-3.3 Libraries for the Nuclear Design Calculations of the NPP Krško with the CORD-2 System

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ABSTRACT

Recently two new nuclear data evaluations have been released: ENDF/B-VIII.0 and JEFF-3.3. Since the neutron cross section data profoundly influence predictions of the nuclear systems behaviour, many researchers have been investigating new data striving for more accurate predictions. The purpose of this study is to examine the effects of the cross sections libraries on the nuclear design calculations of the NPP Krško core. ENDF/B-VII.0, ENDF/B-VII.1, ENDF/B-VIII.0 and JEFF-3.3 libraries are considered. In the first part of the paper the effect on the depletion of the typical NPP Krško fuel assembly in infinite geometry is investigated. In the second part, analysis of all 30 completed NPP Krško operating cycles is performed. Performed analysis has indicated differences of a few hundred pcm in multiplication factor for a fresh fuel due to differences in $^{235}$U cross sections. For a burned fuel assemblies differences are even larger mainly due to different rate of Pu production. Observed differences in libraries resulted in differences of several tens of ppm in critical Boron concentration on the core level. Differences in control rods worth and Boron coefficients were inside 1 %. Isothermal coefficient in the ENDF/B-VIII.0 and JEFF-3.3 cases was noticeably higher compared to the ENDF/B-VII.0 and ENDF/B-VII.1 cases.

1 INTRODUCTION

ENDF/B-VIII.0 [1] and JEFF-3.3 [2] nuclear data evaluations have been released recently. Since their release, many researchers have been investigating how the existing calculation results in a given system are influenced by the new evaluations. The purpose of this study is to examine the effects of the newly cross sections libraries on the nuclear design calculations of the NPP Krško core.

In the nuclear design process two different types of calculations are performed:

1. calculation of the neutron transport in the media, where neutron transport (or diffusion) equation is solved to obtain spatial neutron flux distribution and system reactivity,
2. fuel depletion, where Bateman equations are solved to obtain the time evolution of nuclide concentrations.
Libraries have profound impact on both aspects. In this paper CORD-2 [3] system is used for the library comparison. ENDF/B-VII.0 [4], ENDF/B-VII.1 [5], ENDF/B-VIII.0 and JEFF-3.3 libraries are considered. In the first part of the paper the effect on the depletion of the typical NPP Krško fuel assembly in infinite geometry is investigated. In the second part, analysis of all 30 completed NPP Krško operating cycles is performed. Comparison of the results gives some indications of the libraries performance, while comparison to the measurements performed on the plant gives some clues how to improve CORD-2 nuclear design capabilities.

2 BRIEF CORD-2 DESCRIPTION

The CORD-2 system [3] has been developed by the Reactor physics department of the Jožef Stefan Institute and is intended for core design calculations of PWRs. The system consists of two basic reactor physics codes: WIMS-D [6], and GNOMER [7]. WIMS-D is a well-known and widely used lattice code. Version WIMS-D5 is available from the NEA data bank in Paris. A 69-group neutron cross section library is currently used. GNOMER solves the neutron diffusion equation in three-dimensional Cartesian geometry by using Green’s function nodal method [8]. It also includes advanced features for cross section homogenization and a simple thermal-hydraulic module so that thermal feedback can be taken into account. The CORD-2 system enables determination of the core reactivity and power distribution. The package has been periodically updated and validated for the nuclear design calculations of PWR cores and has been used for the verification of the NPP Krško reload cores since 1990. The latest validation [9] [10] has been performed with a library based on the ENDF/B-VII.0 neutron data files. The library based on the ENDF/B-VII.1 data files has been included in the comparison to obtain better insight in the time evolution of ENDF/B libraries.

3 BRIEF NPP KRŠKO CORE DESCRIPTION

The NPP Krško is a 2-loop Westinghouse PWR that began electricity production in 1981. The start-up core had a rated thermal capacity of 1876 MWe, and a 626 MWe gross electric power. Currently, the thermal rating is 1994 MWe with 727 MWe gross electric power. The core consists of 121 fuel assemblies with some VANTAGE+ features. Each assembly has 235 fuel rods arranged in a 16×16 array. The remaining 21 positions contain guide tubes and are intended for control rods and in-core instrumentation. If the fuel assembly is located in non-rodded position, burnable absorber rods or neutron sources can be present in empty locations. The core features 33 Reactivity Control Cluster Assemblies (RCCA) arranged in 6 banks.

4 RESULTS AND DISCUSSION

Since the CORD-2 has been validated with the ENDF/B-VII.0 library, this library was taken as a reference library. It should be mentioned that the library has been fine tuned to match the measurements as closely as possible. Other libraries are using the same set of tuning parameters to get meaningful comparison. Therefore, presented results should not be taken in absolute sense as a measure of the particular library quality, but rather as a useful comparison that gives an insight in what should be expected, if different libraries are used in the computations.

4.1 Fuel assembly

A typical fuel assembly with 4.95 % enrichment in infinite array was selected as a test case. A reference case scenario consists of the following reactor operational parameters:
1. fuel temperature 900 K,
2. moderator temperature 580.46 K with density 0.70871 g/cm$^3$,
3. soluble Boron concentration of 1000 ppm.

Parameters are close to the Hot Full Power (HFP) average operational parameters applied in the last NPP Krško cycles.

Differences of results in multiplication factor $k_{inf}$ from the reference ENDF/B-VII.0 library as a function of burnup are presented in Figure 1. For the fresh fuel ENDF/B-VII.1 multiplication factor is almost the same, ENDF/B-VIII.0 gives ~200 pcm lower value, while JEFF-3.3 is almost 400 pcm higher. With higher burnup $k_{inf}$ from ENDF/B-VII.1 and ENDF/B-VIII.0 libraries are higher, while JEFF-3.3 results are lower. Differences between libraries are significant and could result in differences of several tens of ppm in critical boron concentration on the core level. At the Beginning Of Cycle (BOC), where fresh fuel is mixed with burned fuel, differences compensate to some degree. However, at the End Of Cycle (EOC) interchange of libraries would require additional tuning.

![Figure 1: Differences in multiplication factor ($k_{inf}^X - k_{inf}^{ENDF/B-VII.0}$)](image)

To separate effect of neutron transport cross sections and different isotopic composition during fuel burnout, a separate calculation was performed taking into account only isotopic composition obtained in the reference ENDF/B-VII.0 case (Figure 2). Differences are much smaller compared to Figure 1. We can conclude that differences in the fresh fuel are caused by different $^{235}$U neutron cross sections, while for the burned fuel obtained nuclide composition is a primary reason for observed differences. The most obvious reason in burned fuel isotopics changes is a different rate of Pu production. To verify it, a plot, where differences in $^{239}$Pu masses from the reference ENDF/B-VII.0 case, has been prepared (Figure 3). Obtained curves nicely resemble the results in Figure 1.
4.2 Critical Boron concentrations

To evaluate the library effect on the core parameters, calculations of all completed 30 NPP Krško operational cycles have been performed. Differences from the measurements performed periodically at the plant are presented in Figures 4-7. Cases at Hot Zero Power (HZP) condition and 0 MWd/tU burnup, and HFP conditions at 150 MWd/tU, 500 MWd/tU and EOC are considered. Differences are in agreement with trends observed in single assembly case.
ENDF/B-VII.0 library gives slightly higher differences compared to ENDF/B-VII.0. ENDF/B-VII.1 results are even higher. However satisfactory grouping of BOC and EOC results can be observed. JEFF-3.3 results at BOC are similar to the ENDF/B-VII.0, while EOC values are significantly lower.

Figure 4: Differences in the critical Boron concentration, ENDF/B-VII.0

Figure 5: Differences in the critical Boron concentration, ENDF/B-VII.1
Figure 6: Differences in the critical Boron concentration, ENDF/B-VIII.0

Figure 7: Differences in the critical Boron concentration, JEFF-3.3
4.3 HZP reactivity consideration

The most important reactivity parameters at HZP, BOC conditions are presented in Table 1. Control rods worth in ENDF/B-VII.1 and ENDF/B-VIII.0 cases are up to 1 % lower compared to ENDF/B-VII.0, while in JEFF-3.3 case, rods are up to 0.4 % higher. Differences in Boron coefficient are lower than 1 %. Differences in isothermal coefficients in Cycle 29 are inside 0.13 pcm/K. In Cycle 30 differences are somewhat higher. Coefficient in the JEFF-3.3 case is higher for 1.34 pcm/K compared to the ENDF/B-VII.0 case.

Table 1: Reactivity parameters at HZP, BOC

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<th>Cycle 30</th>
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<tr>
<td>Control rods worth [pcm]</td>
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<tr>
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<td>Boron coefficient [pcm/ppm]</td>
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<td></td>
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<td>Isothermal coefficient [pcm/K]</td>
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5 CONCLUSION

Impact of the ENDF/B-VII.0, ENDF/B-VII.1, ENDF/B-VIII.0 and JEFF-3.3 neutron cross sections libraries on the nuclear core design calculations of the NPP Krško core has been assessed. Preliminary calculations performed with single fuel assembly in infinite array has indicated differences of a few hundred pcm in the fresh fuel, which are caused by different $^{235}$U neutron cross sections. For the burned fuel different rate of Pu production is a primary reason for even higher differences in multiplication factor between libraries. Comparison with measured critical Boron concentrations has shown progressively higher Boron concentrations for the ENDF/B-VII.0, ENDF/B-VIII.0 and ENDF/B-VII.1 cases with no significant bias between BOC and EOC results. JEFF-3.3 results at BOC are similar to the ENDF/B-VII.0, while EOC values are significantly lower. Differences in control rods worth and Boron coefficients are inside 1 %. Cycle 30 isothermal coefficient in the ENDF/B-VIII.0 and JEFF-3.3 case is noticeably higher compared to the ENDF/B-VII.0 and ENDF/B-VII.1 case.

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REFERENCES


