Impact of the cross section library on ^{93m}Nb activity in VVER-1000 reactor dosimetry

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Abstract—One of the objectives of reactor dosimetry is determination of activity of irradiated dosimeters, which are placed on reactor pressure vessel surface, and calculation of neutron flux in their position. The uncertainty of calculation depends mainly on the choice of nuclear data library, especially cross section used for neutron transport and cross section used as the response function for neutron activation. Nowadays, number of libraries already exists and can be still used in some applications. In addition, new nuclear data library was recently released. In this paper, we have investigated the impact of the cross section libraries on activity of niobium, one of the popular materials used as neutron fluence monitor. For this purpose, a MCNP6 model of VVER-1000 was made and we have compared the results between 14 commonly used cross section libraries. A possibility of using IRDFF library in activation calculations was also considered. The results show good agreement between the new libraries, with the exception of the most recent ENDF/B-VIII.0, which should be further validated.

Keywords—Cross-section libraries, MCNP, ^{93m}Nb, Reactor dosimetry, VVER-1000

I. INTRODUCTION

THE radiation effects on the steel material are the main concern in reactor pressure vessel lifetime. These

effects can be quantified by neutron fluence for energies above 0.5 MeV for VVER reactor type. Number of activation materials is used for VVER reactor dosimetry. Niobium is the ideal material – due to reasonably low threshold energy and large half-life, almost all neutrons emitted from the reactor and transported to the ex-vessel monitor contribute to the measured niobium activity. Other dosimeters have higher threshold energy and half-life shorter than reactor cycle irradiation time, resulting in the loss of around 90 % of neutron interactions in the ex-vessel area (see Fig. 1).

However, the measurement of niobium activity in practice has several difficulties. Usually, characteristic X-ray peaks of ^{93m}Nb with energies between 16 keV and 19 keV are measured by HPGe detectors. Low energies of emitted photons present the main disadvantage, as they require different approach than typical 1-2 MeV gamma lines emitted by other activation materials commonly used in reactor dosimetry. Moreover, impurities in irradiated niobium (e.g. tantalum) can cause additional issues in the measurement. It is therefore difficult to obtain reliable results with low uncertainties.

Measuring of niobium activity in ex-vessel monitors can be verified (or complemented) by transport calculations. The main uncertainty in transport calculations represents nuclear data, especially the cross sections used in all steps of the calculation. In this paper, impact of the cross section library choice on the ex-vessel niobium monitors activity was investigated.

II. CALCULATION MODEL

Transport calculations can be done by either deterministic or statistical calculation codes. Both methods have their advantages and disadvantages. For calculations described in this paper, MCNP6 Monte Carlo code was used. This code allows detailed description of geometry and usage of continuous-energy cross sections. However, the calculation time is influenced by number of output quantities and number of points or cells, for which the desired results are tallied.



Fig. 1. Neutron spectrum of LR-0 reactor (1 - behind barrel, 2 - before RPV, 3 - behind RPV) and response functions of fluence detectors according to IRDFF-1.05 library. The radiation effects on the steel material can be quantified by neutron fluence for energies above 0.5 MeV (dashed line).

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In order to compare the impact of library choice and maintain calculation time under reasonable limits, 2-D model of VVER-1000 reactor was used. The model (see Fig. 2) describes all parts of the reactor. The reactor core consists of 163 fuel assemblies of TVSA-T type, which are homogenized at all levels. Inner parts of reactor between the core and reactor pressure vessel (RPV), such as core basket, barrel, etc., correspond with exact geometrical model. Behind RPV is air cavity, homogenized thermal insulation, serpentinite concrete and building concrete (all parts have its own steel coating). The serpentinite concrete contains 27 ionizing chambers and counterweight channels positioned with a step of 12°.

With respect to the typical temperature gradient in the reactor building, mentioned parts of the model were divided into several regions according to temperature, from 578.5 K in reactor core to 563 K in RPV. Between the outer surface of RPV and the air cavity, the temperature decreases linearly down to 303 K. For further statistical accuracy, variance reduction with cell-based weight windows was used in the model to maintain constant neutron population above 0.5 MeV in spatial cells.

III. TRANSPORT AND ACTIVATION CALCULATIONS

The neutron source was prepared by a Fortran script using effective source method [2]. Input data are files from MOBY-DICK macrocode [3], which contains information about the number of neutrons generated in the reactor core during the fuel campaign on the selected actinides. Each input file contains about 100 time-points in selected campaign for all axial layers, fuel assemblies and pins (in the case of VVER-1000, it is 48 axial layers, 163 fuel assemblies and 331 pins).

To calculate the neutron fluence, the neutron sources are summed and normalized. The number of neutrons in macrocode is determined for the stationary state by integrating the neutron flux and macroscopic cross-sections for fission. In the activity calculations, an effective source is prepared similarly to the neutron fluence. However, a gradual decrease of source due to radioactive decay of products is also taken into account, and effective neutron source has to be therefore prepared for each activation detector.

To ensure statistical accuracy of the calculation, the simulation was run with 1 billion neutrons for each computational variant. By removing neutrons with energies under 0.4 MeV, the speed of calculation was increased approximately seven times. The limit of 0.4 MeV was chosen sufficiently low under 0.5 MeV due to possible upscattering on coolant's hydrogen nuclei.

The calculations were done for 20th axial layer of fuel height in hot state, which is slightly below half the height of the core. In the first approach, we have used ENDF/B-VII.1 library, which is therefore referred as "reference library" further in this paper. The calculated neutron fluence and Nb detector activity is shown in Fig. 3 and 4. The statistical error is between 2.7 % and 6.4 % for neutron fluence and around 4 % for Nb activity calculation, which is not good enough for RPV-lifetime prediction, but it is sufficient for our intent of nuclear data libraries comparison.



Fig. 2. Calculation model of VVER-1000 in MCNP6 with cell boundaries.



Fig. 3. Neutron fluence on RPV surface (MCNP6, library ENDF/B-VII.1).



Fig. 4. Activity of niobium detector (MCNP6, library ENDF/B-VII.1).

IV. NUCLEAR DATA

In reactor dosimetry, nuclear data are used in two steps:

1) Nuclear data for transport

Nuclear data for transport include materials cross sections, angle distributions, etc. These data are used for transport of neutrons between the source (reactor core) and the place of detection, i.e. inner surface of RPV for neutron fluence, or outer surface of RPV for activities of ex-vessel monitors.

2) Nuclear data for activation

Nuclear data for activation include materials cross sections. These data describe nuclear reactions on exvessel monitors.

Different nuclear data libraries can be used in both of the steps. This approach is sometimes used in combination with dosimetric library IRDFF, which can be used only for an activation part of calculation. In this study, we have also considered this option along with the variant of using the same library for both transport and calculation (i.e. combining the library with itself).

The analysis was focused on comparison of 14 continousenergy libraries from different countries, released between years 1979-2018. This includes various versions of the following libraries:

- Russian Evaluated Neutron Data Library (BROND),
- Russian File of Evaluated Neutron Data (RUSFOND),
- Chinese Evaluated Nuclear Data Library (CENDL),
- US Evaluated Nuclear Data Library (ENDF/B),
- Joint Evaluated Fission and Fusion File (JEF, JEFF),
- Japanese Evaluated Neutron Data Library (JENDL).

The source of all ENDF/B libraries is the official MCNP6 distribution, other libraries were processed by NJOY calculation code.

V. RESULTS

The calculations of neutron fluence and niobium activity were done for all 14 libraries described above. As the libraries ENDF/B-V.0 and CENDL-3.1 does not contain ^{93m}Nb data, we have used cross section values of ^{93m}Nb from reference library. The results were azimuthally centered and compared to ENDF/B-VII.1 (see Fig. 5 and Fig. 6).

With a few exceptions, the influence of nuclear data on



Fig. 5. Relative comparison of selected nuclear data libraries with reference library (ENDF/B-VII.1), showing impact of nuclear data library on calculated neutron fluence on RPV surface.

calculation results is mainly due to the library used for neutron transport - in the first approximation, the conclusions made for neutron fluence can be generalized to

activation detectors. In most of the cases, combining the libraries with IRDFF as activation library provides better results than using the same library. The only exception is JENDL-4.0.

The results confirmed some of our expectations about older libraries, e.g. in the case of ENDF/B-V.0, where a clear discrepancy is apparent due to incorrect description of inelastic scattering on iron. As for the more recent libraries, the results show good agreement between most of them. In this case, the impact of library choice on neutron fluence and niobium activity is small, in order of few percent. The exception is the most recent ENDF/B-VIII.0 library, where the results are significantly different.

To explain this difference, the calculations with this library should be further validated. The disagreement could be caused by incorrect format conversion from ENDF-6 to ACE, which is tested by author organization LANL mainly for criticality tasks. Other possibility is error in calculation model or in actual cross-section data, similar to problem presented by [5] with JENDL-3.3 and angle distribution of oxygen-16. Eventually, it may also turn out that the new data are correct. A detailed study on ENDF/B-VIII.0 performance for reactor dosimetry is currently under way.

For the future studies of cross section libraries, there is a possibility of performing nuclear data uncertainty propagation analysis, which uses uncertainties of nuclear data as part of the most recent libraries.

I ABLE I Comparison of cross section libraries		
Transport library	Same activation library	IRDFF v.1.05 activation library
BROND-3.1	1.042 (9)	1.013 (14)
CENDL-3.1	0.929 (14)	0.931 (14)
ENDF/B-V.0	0.830 (14)	0.827 (14)
ENDF/B-VI.0	0.999 (30)	1.000 (45)
ENDF/B-VI.8	0.975 (32)	0.974 (46)
ENDF/B-VII.0	0.973 (10)	0.972 (14)
ENDF/B-VII.1	1.000 (10)	1.000 (14)
ENDF/B-VIII.0	0.778 (10)	0.775 (14)
JEF-2.2	0.825 (32)	0.925 (45)
JEFF-3.3	0.910 (32)	1.019 (46)
JENDL-3.3	0.975 (12)	1.014 (31)
JENDL-4.0	1.050(11)	1.091 (35)
RUSFOND-2008	1.036 (12)	1.010 (32)
RUSFOND-2010	1.097 (11)	1.064 (33)



Fig. 6. Relative comparison of selected nuclear data libraries with reference library (ENDF/B-VII.1), showing impact of nuclear data library on calculated activity of Nb detectors using combination of selected library with itself, or IRDFF v.1.05, as activation library.

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