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ARTICLE

Analysis of the H. B. ROBINSON-2 reactor pressure vessel dosimetry benchmark using TRIPOLI-4® Monte Carlo code

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This paper presents an analysis of the H.B. Robinson-2 reactor pressure vessel dosimetry benchmark. This benchmark is based on an experiment performed at an operating PWR (Pressurized Water Reactor). It includes measurements for both in-vessel and ex-vessel dosimeters. To enlarge the TRIPOLI-4® validation database of neutron attenuation in PWR, an analysis of H.B. Robinson-2 benchmark was carried out using the TRIPOLI-4® Monte Carlo code, the DARWIN/PEPIN2 depletion code, the general-purpose point-wise library JEFF3.1.1, and the International Reactor Dosimetry File IRDF-2002. Calculated activity values are compared with measured ones provided in this benchmark. Furthermore, calculated activity values are also compared with results obtained using two calculation schemes: the deterministic code DORT and the BUGLE-96 cross section library on the one hand, and the Monte Carlo code MCNPX and the JEFF3.1.1 cross section library on the other hand.

Keywords: H. B. Robinson-2; TRIPOLI-4; Monte Carlo; DARWIN/PEPIN2; APOLLO2

1. Introduction

The H.B. Robinson-2 (HBR-2) benchmark [1] is an international benchmark dedicated to in- and ex- vessel dosimeter activation in a Westinghouse PWR. The scope of this benchmark is to validate the capabilities of computational methodologies to predict specific activities of dosimeters irradiated in specific locations (in-vessel and ex-vessel locations). High threshold dosimeters were irradiated in the H. B. Robinson unit 2 nuclear power plant during cycle #9. Dosimeters activity was measured to form the basis of experimental results of this benchmark. Data from this benchmark are available from the SINBAD database managed jointly by RSCICC and NEA[2].

This paper presents a HBR-2 benchmark analysis using the TRIPOLI-4® Monte Carlo code [3] and the DARWIN/PEPIN2 depletion code [4]. The APOLLO2 code [5] is also used to calculate neutron sources in the HBR-2 core. Different options to model the neutron source history are analyzed and compared. Finally, the C/M results are compared with C/M values calculated using two other codes: the DORT deterministic code [6], and the MCNPX Monte Carlo code [7].

This article starts with an inventory of simulation tools used to analyze the HBR-2 benchmark. Then HBR-2 Monte Carlo modeling and operating history modeling are presented in detail. Finally, results and comparisons between the calculated and measured activity values are discussed.

2. Simulation tools

Simulation tools used to carry out the HBR-2 analysis presented in this article are developed by CEA (Commissariat à l'Energie Atomique), with financial support of EDF (Electricité De France) and AREVA (French nuclear company). Nuclear data used by these codes are common to provide a uniform and consistent set of computational codes.

2.1. TRIPOLI-4®

TRIPOLI-4® is a 3D transport code using full pointwise cross section data. It is dedicated to radiation protection and shielding, nuclear criticality safety, fission and fusion reactor design, and nuclear instrumentation. It is used as a reference tool by CEA, EDF, and several other industrial or institutional partners.

The TRIPOLI-4® code is used in this study to calculate neutron spectra and reaction rate values in dosimeters.

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2.2. DARWIN/PEPIN2

DARWIN/PEPIN2 solves Bateman's generalized differential equations governing the time dependence of isotope concentrations. This code may be coupled with the TRIPOLI-4®.

In this analysis, the DARWIN/PEPIN2 code is dedicated to the activity calculations.

2.3. APOLLO2.8-3

The APOLLO2 spectral transport code is widely used by CEA, EDF, and AREVA for cross section generation and direct transport calculations, including a large range of applications in reactor physics, criticality safety studies, and fuel cycle analysis. In this analysis, the APOLLO2 code is used to calculate fission fraction values as presented in section 3.2.

3. Monte Carlo modeling

3.1. Geometry

The core of HBR-2 reactor consists of 157 fuel elements. It is surrounded by a core baffle, a core barrel, a thermal shield, a pressure vessel, and a biological shield.

Using the TRIPOLI-4® code, the geometrical modeling of the reactor, shown in the **Figure 1** and the **Figure 2**, is limited to one quarter.

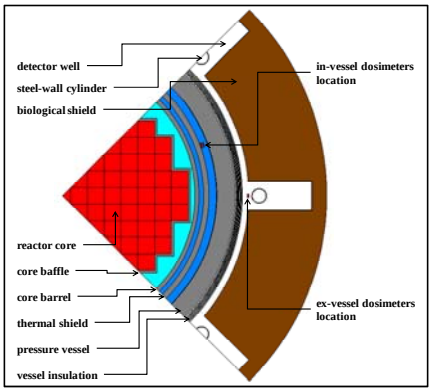

Figure 1. Reactor modeling (horizontal modeling).

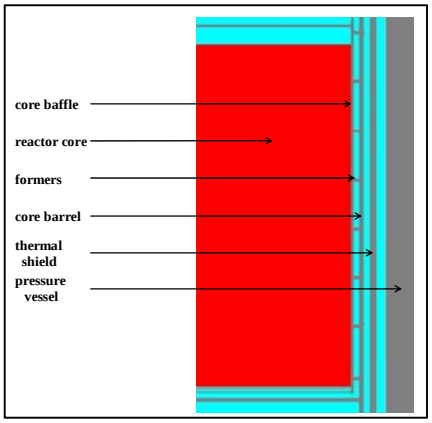

Figure 2. Reactor modeling (vertical modeling).

Individual components of the reactor and material compositions are modeled in agreement with the benchmark data. General dimensions of the H.B. Robinson unit-2 reactor are presented in **Table 1**.

Table 1. General dimensions of the HBR-2 reactor.

Elements	Inner radius	Thickness
Core baffle	-	2.858 cm
Core barrel	170.023 cm	5.161 cm
Thermal shield	181.135 cm	6.825 cm
Cladding (vessel)	197.485 cm	0.556 cm
Vessel (base metal)	198.041 cm	23.614 cm
Thermal insulation	222.964 cm	7.620 cm
Biological shield	238.760 cm	100.00 cm

The dosimeter modeling data are not provided in the benchmark. This simplification results in errors on the calculated reaction rate values, especially for fissile dosimeters: self-shielding and isotopic depletion (plutonium production in ^{238}U dosimeters for instance) are not taken into account in simulations since the fissile isotopes are not modeled. These errors are not corrected.

As well, the photofission contribution for the fissile dosimeters is not taken into account. However, this approximation is compensated for by using correcting factors [1] for measured activity values:

- 0.975 for ^{237}Np in-vessel detectors.
- 0.950 for ^{238}U in-vessel detectors.
- 0.950 for ^{237}Np ex-vessel detectors.
- 0.900 for ^{238}U ex-vessel detectors.

Concerning the copper dosimeters, the modeling simplification does not allow taking into account the ^{59}Co impurity contribution. A 0.975 correcting factor [1] must be applied to the ^{60}Co measured activity values.

3.2. Neutron sources

The HBR-2 cycle #9 is divided into eight time intervals corresponding to eight burn-up values. Eight core power distribution maps are therefore available. The use of various core power distribution maps is discussed in section 4 which outlines the operating history modeling.

For each core power distribution map, neutrons sources are modeled from the power and the burn-up distribution given for each fuel rod. The source term is limited to the fission spectra of ^{235}U and ^{239}Pu. The fission fraction values are calculated for uranium and plutonium isotopes using the APOLLO2.8-3 code.

3.3. Variance reduction

In the HRB-2 benchmark, the dosimeters are located in regions hardly reached by particles in a natural simulation. A variance reduction technique is used to reduce the simulation time and to increase its efficiency.

The variance reduction scheme in the TRIPOLI-4® code allows an automatic pre-computation of the importance map $I_{ref}(P)$.

3.4. Reactions taken into account

Dosimeters irradiated in the HBR-2 experiment, as well as their main reaction are gathered in **Table 2**. The theoretical threshold values correspond to the minimum energy values with the available cross section data in the IRDF2002 library [8]. For ^{237}Np and ^{238}U fissions, the threshold value is increased by the use of a neutron filter (the theoretical threshold is 10^{-11} MeV).

Secondary reactions are not taken into account. This is justified by the fact that dosimeters are enclosed inside gadolinium or cadmium filter (depending on their location). Thermal neutrons are absorbed in these filters, which considerably reduces secondary reactions.

Table 2. Dosimeters irradiated in the HBR-2 experiment.

Dosimeters' reaction	Threshold	Reaction product half-life
^{237}Np(n,f)^{137}Cs	10^{-5} MeV	30 years
^{238}U(n,f)^{137}Cs	10^{-5} MeV	30 years
^{58}Ni(n,p)^{58}Co	0.38 MeV	71 days
^{54}Fe(n,p)^{54}Mn	0.67 MeV	312 days
^{46}Ti(n,p)^{46}Sc	3.20 MeV	84 days
^{63}Cu(n,α)^{60}Co	2.23 MeV	5.3 years

4. Operating history modeling

The operating history remains at an almost constant level during 524 days. The dosimeters' activity values are calculated at the end of the irradiation to be compared with the measured activity values provided in the HBR-2 benchmark. To model the operating history, three different methodologies described hereafter are considered.

4.1. Methodology A

For a dosimeter *i*, the reaction rate $R_{i,5}$ is calculated using the core power distribution #5 with the total core power P_5. This core power distribution has been proposed in the HBR-2 benchmark to be the reference core power distribution since it corresponds to the mid-cycle [1]. The daily core power P_d is taken into account by multiplying the reaction rate by P_d/P_5: the reaction rate value is taken proportional to the core power. Basically, the daily reaction rate value $R_{i,d}$ for the dosimeter *i* at *d* day is given by the Eq (1).

$$R_{i,d} = R_{i,5} \times \frac{P_d}{P_5} \tag{1}$$

4.2. Methodology B

The methodology B follows the methodology A. But here, the core power redistribution is not neglected. To take the changes in core power distribution into account, a F_j/F_5 ratio is applied. F stands for the average relative power of the fuel elements whose contribution to the dosimeters' irradiation is large [1]. The F_j values were obtained from DORT calculations for each time interval (the HBR-2 cycle #9 is divided into eight time intervals). These F_j values are assumed constant during the corresponding interval *j*. The daily reaction rate value $R_{i,j,d}$ for the dosimeter *i* at *d* day is given by the Eq (2).

$$R_{i,j,d} = R_{i,5} \times \frac{P_d}{P_5} \times \frac{F_j}{F_5} \tag{2}$$

4.3. Methodology C

The weakness of the methodology B lies in the fact that the daily reaction rates are strongly linked to a calculated ratio F_j/F_5 (in this benchmark, these ratio values were calculated using the DORT code). To make the daily reaction rate values independent from any calculated ratio, and to calculate these values with the sole use of the TRIPOLI-4® code, the reaction rate value $R_{i,j,d}$ for a dosimeter *i* is calculated independently for the eight available core power distribution *j* in the methodology C. The daily reaction rate value $R_{i,j,d}$ for the dosimeter *i* at *d* day is given by the Eq (3).

$$R_{i,j,d} = R_{i,j} \times \frac{P_d}{P_j} \tag{3}$$

The total computation time is eight times longer using this methodology. However, this method is more realistic. The computation time is not a critical issue when using the Green's functions [3] with the TRIPOLI-4® code. A single Monte Carlo simulation is required with a core power distribution (#5 for instance). Then, for each dosimeter, the reaction rate values are calculated for all other core power distributions by reading the Green's function information.

5. Analysis of neutron spectra and dosimetry results

5.1. In-vessel neutron spectra results

Neutron spectra are calculated at the in-vessel dosimetry location, using the 47 neutron-group structure provided in the HBR-2 benchmark. The calculated neutron spectra using TRIPOLI-4®/JEFF3.1.1 and DORT/BUGLE-96 are compared. A significant difference is observed between these spectra for energy lower than 1 eV. There are no details in the HBR-2 benchmark concerning the DORT calculation to explain this difference. A finer energetic grid for energy lower than 1 eV would have helped to get an explanation.

The ratio between TRIPOLI-4® and DORT calculated spectra is plotted on the **Figure 3** from 1 eV up to 17.33 MeV. The ratio values stay within the ±10% range (grey area on the Figure 3). This result points out a good agreement between the TRIPOLI-4®/JEFF3.1.1 and the DORT/BUGLE-96 neutron spectrum calculations.

5.2. Ex-vessel neutron spectra results

Neutron spectra are calculated at the ex-vessel dosimetry location, using the 47 neutron-group structure

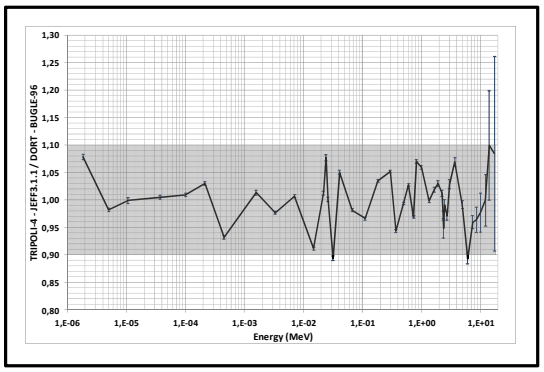

Figure 3. In-vessel neutron spectrum: comparison between TRIPOLI-4 – JEFF3.1.1 and DORT – BUGLE-96.

provided in the HBR-2 benchmark. The calculated neutron spectra using TRIPOLI-4®/JEFF3.1.1 and DORT/BUGLE-96 are compared.

A significant difference is observed between these two spectra at low energies. But in contrast to the in-vessel calculations, this difference remains important up to approximately 10 keV as highlighted by the **Figure 4**. Then between 10 keV and 4 MeV, the ratio reaches the 10% area (grey), but shows strong variations. The main explanation is probably to be found in large discrepancies between JEFF3.1.1 and BUGLE-96 cross section libraries (point-wise versus multigroup libraries). These differences have significant impact on cross section data, especially for elastic scattering for iron. The attenuation of neutron in metal structures (vessel) may be miscalculated using the multigroupe library.

The ex-vessel dosimeter analysis presented in section 5.3.2 shows that the ex-vessel neutron spectrum calculated by the DORT code may be underestimated at energies below than 10 keV.

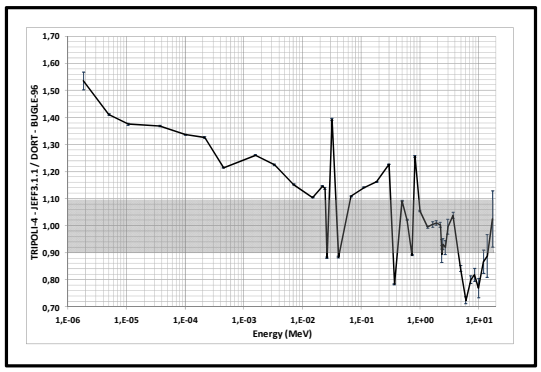

Figure 4. Ex-vessel neutron spectrum: comparison between TRIPOLI-4 – JEFF3.1.1 and DORT – BUGLE-96.

5.3. Dosimetry results

Activity values are calculated using the TRIPOLI-4® Monte Carlo code, and the DARWIN/PEPIN2 depletion code. In the following sections, this calculation scheme will be referred to as "T4/DP2".

5.3.1 In-vessel dosimetry results

Using the three methodologies presented in section 4, the C/M values for in-vessel dosimeters are presented in the **Table 3**. The average C/M and standard deviation (s. d.) values are also presented in this table.

Table 3. In-vessel dosimetry results (T4/DP2).

Dosimeters	Methodology A	Methodology B	Methodology C
^{237}**Np**	0.94	0.94	0.94
^{238}**U**	0.85	0.85	0.85
^{58}**Ni**	0.80	0.89	0.87
^{54}**Fe**	0.84	0.87	0.86
^{46}**Ti**	0.79	0.87	0.84
^{63}**Cu**	0.87	0.87	0.86
Average	0.85	0.88	0.87
s. d.	0.05	0.03	0.03

The C/M values are close regardless of the chosen methodology. However, the methodologies B and C give better C/M values, especially for dosimeters with a short reaction product half-life (nickel and titanium dosimeters). This result is explained by a better inclusion of the core power redistribution effect (see section 4).

The T4/DP2 C/M results are compared with the C/M values obtained from two different calculation schemes: the DORT/BUGLE-96 [1] and the MCNPX/JEFF311 [9] calculation schemes. We only consider methodology C results for the T4/DP2 calculation scheme since the power history is realistically described with this methodology (methodologies A and B have simplifications as presented in section 4).

The C/M average values are presented on **Table 4** for each calculation scheme.

Table 4. Average C/M values for three calculation schemes (in-vessel dosimeters).

Dosimeters	T4/DP2 (C)	DORT	MCNPX
Average	0.87	0.91	0.84
s. d.	0.03	0.03	0.03

DORT/BUGLE-96 calculation scheme gives C/M value close to 1. However, all C/M results stay close regarding the standard deviation values. They are underestimated by about 10% compared to measured values. Neutron sources or cross section data may explain this result since the three independent calculation schemes give the same trend.

5.3.2 Ex-vessel dosimetry results

Using the three methodologies presented in section 4, the C/M values for ex-vessel dosimeters are presented in the **Table 5**. The average C/M and standard deviation (s. d.) values are also presented in this table.

For the ex-vessel dosimeters, C/M values are very close to each over regardless of the chosen methodology. Calculated activity values are moderately sensible to the core power redistribution since the distance between the core and dosimeters is relatively large.

As with the in-vessel dosimeters, the C/M values obtained by the T4/DP2 calculation scheme and the

Table 5. Ex-vessel dosimetry results (T4/DP2).

Dosimeters	Methodology A	Methodology B	Methodology C
^{237}Np	0.86	0.86	0.85
^{238}U	0.97	0.97	0.96
^{58}Ni	0.88	0.94	0.98
^{54}Fe	0.99	1.01	1.02
^{46}Ti	0.90	0.95	0.98
^{63}Cu	0.99	0.99	0.98
Average	0.93	0.95	0.96
s. d.	0.05	0.05	0.05

methodology C are compared with the calculated C/M values using the DORT/BUGLE-96, and the MCNPX/JEFF311 calculation schemes. The C/M average and the standard deviation values are presented in the **Table 6**.

Table 6. Average C/M values for the three calculation schemes (ex-vessel dosimeters).

Dosimeters	T4/DP2 (C)	DORT	MCNPX
Average	0.96	0.90	0.92
s. d.	0.05	0.09	0.08

The best C/M values are obtained using the T4/DP2 calculation scheme. In addition, it provides the lowest standard deviation value. This satisfactory result is partly due to the good agreement concerning fissile dosimeters, especially neptunium dosimeters (DORT/BUGLE-96: C/M = 0.61, MCNPX/JEFF3.1.1: C/M = 0.76, T4(JEFF3.1.1)/DP2: C/M = 0.85). Since the contribution for energy lower than 10 keV is about 10% for neptunium dosimeters (and less than 1% for uranium dosimeters), this agreement shows that the calculated neutron spectrum is better using the TRIPOLI-4®/JEFF3.1.1 calculation scheme instead of the DORT/BUGLE-96 or even the MCNPX/JEFF3.1.1 calculation schemes. As a result, the neutron spectrum calculated using the DORT code is underestimated for energy lower than 10 keV. This could have been confirmed if some 1 keV threshold dosimeters had been irradiated in the H. B. Robinson-2 benchmark.

An inverse trend in the result of TRIPOLI-4® and DORT between in-vessel and ex-vessel is observed. This reflects a different neutron attenuation calculation, probably due to the cross section description of these two codes (point-wise versus multigroup libraries).

6. Conclusion

H. B. Robinson-2 benchmark was analyzed using TRIPOLI-4®, DARWIN/PEPIN2, and APOLLO2 codes. Using the most realistic operating history modeling, a good agreement between calculated and measured activity values is shown. The average C/M values are 0.87 ± 0.03 for the in-vessel dosimeters, and 0.96 ± 0.05 for the ex-vessel dosimeters.

These results were compared with the C/M values obtained from other calculation scheme: DORT/BUGLE-96 and MCNPX/JEFF311. For in-vessel dosimeters, the C/M values are slightly better using the DORT/BUGLE-96 calculation scheme (0.91 ± 0.03). On the other hand for ex-vessel dosimeters, the best C/M result (0.96 ± 0.05) is obtained using the TRIPOLI-4®/JEFF3.1.1 – DARWIN/PEPIN2 calculation scheme. This analysis points out that the calculated neutron spectrum for the ex-vessel dosimeters is underestimated using the DORT/BUGLE-96 calculation scheme. The calculated neutron spectrum for ex-vessel dosimeters seems to be more accurate when using the TRIPOLI-4®/JEFF3.1.1 calculation scheme. This result is based on fissile dosimeter analysis the energy threshold of which is lower than 10 keV.

Acknowledgements

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