Benchmark Calculations for VENUS-2 MOX-Fueled Reactor Dosimetry

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1. Introduction

As a part of a Nuclear Energy Agency (NEA) Project, it was pursued the benchmark for dosimetry calculation of the VENUS-2 MOX-fueled reactor [1]. In this benchmark, the goal is to test the current state-of-the-art computational methods of calculating neutron flux to reactor components against the measured data of the VENUS-2 MOX-fuelled critical experiments. The measured data to be used for this benchmark are the equivalent fission fluxes which are the reaction rates divided by the U^{235} fission spectrum averaged cross-section of the corresponding dosimeter. The present benchmark is, therefore, defined to calculate reaction rates and corresponding equivalent fission fluxes measured on the core-mid plane at specific positions outside the core of the VENUS-2 MOX-fuelled reactor. This is a follow-up exercise to the previously completed UO_2-fuelled VENUS-1 two-dimensional and VENUS-3 three-dimensional exercises [2]. The use of MOX fuel in LWRs presents different neutron characteristics and this is the main interest of the current benchmark compared to the previous ones.

2. Methods and Results

In this benchmark, a full set of the source term is not provided. However, the fission rate distribution of 121 fuel pins measured on the core mid-plane and the axial fission rate distribution of 6 fuel pins are given to be able to obtain the source term as exact as possible. The reference core average fission rate and corresponding power should be used to define the fission source for neutron transport calculations.

In this work, using each of the TORT, MCNP4C2, and MCNPX codes, source terms were calculated, and then, dosimetry calculations were pursued using the source terms.

2.1 Source Term Calculation

In MCNP4C2 and MCNPX calculations, the SSW card of MCNP code was used to generate a KCODE (criticality calculation) fission source file. ENDF/B-VI.8 cross-section library was used for the transport calculation. In order to revise the MCNP results for the total fission source, two multiplication constants were calculated to be 1.15909E+13 sources/sec and 1.15976E+13 sources/sec for the MCNP4C2 and MCNPX results, respectively. All the MCNP results were multiplied by the values.

In TORT calculation, the VENUS-2 core has been modeled with a $119\times126\times111$ mesh in Cartesian coordinate using a S8 order symmetric quadrature set. 3-D discrete model was also developed using BOT3P code. 35-group, P_3 Legendre polynomial, cross section was generated for 20 material mixtures using NJOY and TRANSX code with the ENDF/B-VI.8 library. The relative power distribution in the VENUS-2 core was generated using TORT code. A multiplication constant of 1.13736×10^{13} sources/sec was also calculated for the total fission source and then all the TORT results were multiplied by the value.

2.2 Dosimetry Calculation

The equivalent fission fluxes at the several important positions on the core mid-plane were measured using ^{58}Ni(n,p), ^{115}In(n,n'), ^{103}Rh(n,n'), ^{64}Zn(n,p), ^{237}Np(n,f), and ^{27}Al(n,α) detectors. ^{64}Zn(n,p) detectors, which have the same activation threshold of 2.8 MeV as ^{58}Ni(n,p), have been used for measurement at that energy beyond the core barrel. ^{27}Al(n,α) detectors, which have a threshold of 7.6 MeV, have been used to observe the performance of the calculation tools at that high energy level. The detectors were placed along the core mid-plane at the 34 locations in the outer core region, core baffle, water reflector, core barrel, and neutron pad.

In the MCNP4C2 and MCNPX calculations, the SSR card was used in the subsequent dosimetry calculation using SSW card generated in above section 2.1.2. IRDF-90 (version 2) cross-section library was used for dosimeter calculation.

In the TORT calculation, the neutron source generated in the above section 2.1 was used in the subsequent dosimetry calculation. IRDF-90 (version 2) cross-section library was also used for the dosimetry calculation.

2.3 Results

The dosimeter cross-sections averaged over the U^{235} fission spectrum to convert the calculated reaction rates into equivalent fission fluxes were calculated and are summarized in Table 1. For Al detector, the result from TORT gives an agreement of 24.4% error in comparison with those from MCNP. But, for the remaining detectors, the results give a good agreement of less than 3.1% error.

The equivalent fission fluxes were calculated and are shown at the measured positions, which are specified in the benchmark [1], for the detectors in Table 2. For all of the detectors excepting Al detector, the results give a good agreement of less than about 15% error in comparison with the relative errors between each codes. However, for Al detector, the results from TORT give an agreement of more than 20% error at the many points in comparison with those from MCNP.

3. Conclusion

In this work, the dosimeter cross-sections averaged over the U^{235} spectrum were calculated and then the equivalent fission fluxes divided by the dosimeter cross-sections were calculated using TORT, MCNP4C2 and MCNPX codes.

It is found that the reaction rates calculated from TORT code were overestimated in the inner regions (central hole and inner baffle) and underestimated in the outer regions (remaining regions) in comparison with those from MCNP codes. For Al detector, the results from TORT code were overestimated in both the inner and outer regions. It is also found that the equivalent fission fluxes calculated from TORT code were also overestimated in the inner regions and underestimated in the outer regions for all of the detectors including Al detector. It comes from the fact that the dosimeter cross-section calculated from TORT code is overestimated as high as 24.4% in comparison with those from MCNP codes for Al detector, presented in Section 2.3.

It is expected that this study can be used as the basic data to analyze MOX-fueled reactor with the previously completed two and three dimensional MOX-fueled benchmarks.

Acknowledgment

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Reference

[1] Chi Young Han et al., "VENUS-2 MOX-Fuelled Reactor Dosimetry Calculations - Benchmark Specification", NEA/NSC/DOC(2004)6, Nuclear Energy Agency (2004).

[2] Hehn, G. and B-C. Na, "Prediction of Neutron Embrittlement in the Reactor Pressure Vessel: VENUS-1 and VENUS-3 Benchmarks," NEA/NSC/DOC(2000)5, Nuclear Energy Agency (2000).

Table 1. Dosimeter Cross-section Averaged over the U^{235} Fission Spectrum

[Unit: mbarn]

Reaction	^{58}Ni (n,p)	^{115}In (n,n')	^{103}Rh (n,n')	^{64}Zn (n,p)	^{237}Np (n,f)	^{27}Al (n,α)
TORT	104.754	181.203	695.670	38.121	1318.832	0.780
MCNP4C2	101.853	181.983	694.545	36.997	1315.735	0.627
MCNPX	101.853	181.983	694.545	36.997	1315.735	0.627

Table 2. Equivalent Fission Flux at a Position of Each Region

[Unit: neutrons/cm^2/sec]

Measurement Region	Code	^{58}Ni (n,p)	^{115}In (n,n')	^{103}Rh (n,n')	^{64}Zn (n,p)	^{237}Np (n,α)	^{27}Al (n,α)
Inner Baffle	TORT	1.5296E+09	1.9280E+09	2.3013E+09	1.4728E+09	2.6400E+09	1.4340E+09
	MCNP4C2	1.4917E+09	1.8746E+09	2.2645E+09	1.4367E+09	2.6113E+09	1.2394E+09
	MCNPX	1.4927E+09	1.8559E+09	2.2491E+09	1.4378E+09	2.5926E+09	1.2750E+09
Outer Baffle	TORT	5.0238E+08	6.2196E+08	7.3228E+08	4.8469E+08	8.3155E+08	4.7045E+08
	MCNP4C2	6.1992E+08	7.3311E+08	8.5976E+08	5.9997E+08	9.7100E+08	5.4494E+08
	MCNPX	5.9691E+08	7.1097E+08	8.4769E+08	5.7929E+08	9.6490E+08	7.3352E+08
Barrel	TORT	6.8040E+07	7.9698E+07	9.2456E+07	6.5541E+07	1.0565E+08	8.3994E+07
	MCNP4C2	8.2787E+07	9.2179E+07	1.0654E+08	7.9697E+07	1.2200E+08	1.2529E+08
	MCNPX	7.8660E+07	9.0520E+07	1.0555E+08	7.6422E+07	1.2132E+08	9.9276E+07
Neutron Pad	TORT	5.9303E+06	7.3538E+06	9.1853E+06	5.6309E+06	1.0762E+07	9.4044E+06
	MCNP4C2	6.3139E+06	7.7862E+06	9.8748E+06	5.9666E+06	1.1511E+07	8.3686E+06
	MCNPX	6.0864E+06	8.0328E+06	1.0121E+07	5.6414E+06	1.1783E+07	7.7331E+06
Central Hole	TORT	1.1390E+09	1.2806E+09	1.4403E+09	1.1122E+09	1.6779E+09	1.2366E+09
	MCNP4C2	2.8859E+08	3.1108E+08	3.5134E+08	2.8293E+08	4.2820E+08	3.6930E+08
	MCNPX	2.9841E+08	3.1712E+08	3.5685E+08	2.9153E+08	4.3414E+08	3.5001E+08
Water Gap	TORT	2.1803E+07	2.3661E+07	2.6999E+07	2.1178E+07	3.1335E+07	3.2185E+07
	MCNP4C2	2.3915E+07	2.5648E+07	2.9622E+07	2.3271E+07	3.6253E+07	3.5608E+07
	MCNPX	2.5623E+07	2.5760E+07	2.9673E+07	2.5014E+07	3.6324E+07	4.7725E+07
Reflector	TORT	6.4396E+08	7.5604E+08	8.7103E+08	6.2614E+08	1.0075E+09	6.4848E+08
	MCNP4C2	6.6239E+08	7.6038E+08	8.7752E+08	6.4596E+08	1.0360E+09	6.7472E+08
	MCNPX	6.5130E+08	7.3938E+08	8.6245E+08	6.3395E+08	1.0230E+09	7.8989E+08