INEEL/EXT-04-01709

Integrated Fast Neutron Flux at the End of Phase-I, Phase-II, Phase-III, and Phase-IV-1B of the MOX ZR-Cladding Tubes

G. S. Chang

March 2004

Idaho National Engineering and Environmental Laboratory
Bechtel BWXT Idaho, LLC

INEEL/EXT-04-01709

Integrated Fast Neutron Flux at the End of Phase-I, Phase-II, Phase-III, and Phase-IV-1B of the MOX ZR-Cladding Tubes

G. S. Chang

March 2004

Idaho National Engineering and Environmental Laboratory
Nuclear & Research Sciences Department
Idaho Falls, Idaho 83415

Prepared for the
U.S. Department of Energy
Office of Fissile Material Disposition Program
Under DOE Idaho Operations Office
Contract DE-AC07-99ID13727

ABSTRACT

This report using the detailed ATR quarter core model calculated neutronic tallies, the MCWO-calculated Zr-cladding fast neutron fluence (E > 0.1 MeV and E > 1.0 MeV) distributions at the end of Phase-I, -II, -III, and -IV Irradiation are tabulated in Table 1, 2, 3, and 4. At the end of the Phase-I irradiation, the MCWO-calculated Zr-cladding fast neutron fluences of the removed MOX capsules 1 and 8 are 2.68 and 2.68 x 10^{20} n/cm^2, respectively. At the end of Phase-II Irradiation are tabulated in Table 2. At the end of the Phase-II irradiation, the MCWO-calculated Zr-cladding fast neutron fluences of the removed MOX capsules 9 and 2 are 6.78 and 6.79 x 10^{20} n/cm^2, respectively. At the end of the Phase-III irradiation, the MCWO-calculated Zr-cladding fast neutron fluences of the removed MOX capsules 10 and 3 are 9.82 and 9.70 x 10^{20} n/cm^2, respectively. And, at the end of the Phase-IV part 1B irradiation, the MCWO-calculated Zr-cladding fast neutron fluences of the removed MOX capsules 4 and 13 are 1.41 and 1.39 x 10^{21} n/cm^2, respectively.

CONTENTS

INTRODUCTION

To support potential licensing of MOX fuel made from WG-plutonium and depleted uranium for use in United States reactors, an experiment containing WG-MOX fuel has been fabricated and is being irradiated in the Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL). The uninstrumented test assembly containing nine MOX fuel capsules and neutron monitor wires was inserted into the ATR for irradiation to achieve a burnup of 50 GWd/t. The Oak Ridge National Laboratory (ORNL) manages this project for the Department of Energy.

The Monte Carlo depletion tool (MCWO)[1] used in this study can provide an accurate correction factor for the reactor parameters, such as capture-to-fission ratios, isotopic concentrations and compositions, fission power, and spectrum versus burnup.

WG-MOX FUEL TESTING IN THE ATR

The initial experiment phase (Phase-I irradiation), which contained nine MOX fuel capsules, was loaded into the ATR in January 1998. After 153.5 effective full power days (EFPDs) of irradiation in Phase-I,[2] a capsule pair was withdrawn from the ATR in September 1998 after having achieved an average discharge burnup of about 8.6 GWd/t. At the end of Phase-II[3] irradiation (226.9 EFPDs), an additional capsule pair was withdrawn in September 1999 after having achieved an average discharge burnup of about 21.5 GWd/t. Also, at the end of Phase-III[4] irradiation (232.8 EFPDs), an additional capsule pair was withdrawn in September 2000, after having achieved an average discharge burnup of about 29.6 GWd/t. Post-Irradiation Examination (PIE) of these capsules has recently been completed at ORNL.

To assure the integrity of the MOX fuel pellets, when the burnup reaches 40 GWd/t at the end of Phase-IV part 1A and 1B, the capsules 4 and 13 were withdraw and transfer to ORNL for the Post Irradiation Examination (PIE). This PIE indicated that the MOX fuel had performed as expected with no abnormality; therefore, the MOX fuel will continue the irradiation to 50 GWd/t. The Phase-IV part 1A MOX capsule arrangement was achieved by placing Capsules 6 and 12 in the two front top positions, Capsules 4 and 13 in the two front middle positions, and Capsule 5 in the back middle position as shown in Table 4A Capsule ID column. The other four assembly positions are filled with dummy SST capsules. The new refined MCNP model of the MOX fuel test assembly with the Al thermal shield located in the small NW I-24 hole for Cycle 124C was used to calculate the fission (f7:n) heating tallies in the fuel capsules. This report utilizes the MCNP-calculated results to provide the neutron/fission heat rate and burnup distribution data during Phase-IV part 1A irradiation Cycle 124C in the weapons-grade mixed oxide (WG-MOX) fuel test assembly located in the small I-24 hole.

To increase the linear heat generation rate (LHGR) during, the Phase-IV part 1B, starting at Cycle 126B, the MOX fuel test assembly was moved to SW I-23 position. The new refined MCNP model of the MOX fuel test assembly with the Al thermal shield located in the SW small I-23 hole for Cycles 126B and 127A was used to calculate the fission (f7:n) heating tallies in the fuel capsules. Based on the beginning of cycle 126B, the neutronics tallies were generated by the quarter core MCNP model. MCNP-calculated fission heat tallies were normalized for a SW quadrant power of 29.1 MW (SW lobe power 23 MW), which is the SW quadrant power for Cycle 126B and 127A from the CSAP.

MCWO METHOD

In general, reactor physics analysis consists of multistep analysis methods. The multigroup diffusion equation with node-wise constant cross sections requires the fuel assembly to be appropriately homogenized. However, the complex spectral transitions in the WG-MOX fuel pellet present a serious challenge. The major source of uncertainty in the fuel burnup calculation comes from burnup-dependent cross-section (XS) and resonance treatment of neutron fluxes in the MOX fuel pellet. To avoid these problems, a validated depletion tool is used, which applies the Monte-Carlo code MCNP,[6] coupled with an isotope depletion code, ORIGEN-2[7]; this is the MCWO[1,8] methodology. MCWO was used to analyze the fission power density ratio and cumulative burnup of MOX fuel pellets versus irradiation effective full power days (EFPDs). MCWO can provide an accurate correction factor for the reactor parameters, such as capture-to-fission ratios, isotopic concentrations and compositions, fission power and spectrum versus burnup.

MCWO was used to track fuel burnup and heat rates as functions of irradiation time. The fission power distribution and linear heat generation rate (LHGR) of the MOX fuel capsules were provided for the thermal analysis. Temperature distributions were needed to make sure that the MOX pins met the ATR safety requirements and to analyze the behavior of the fission gas release. The MCWO-calculated results were also provided to ORNL for the experiment-specific Capsule Assembly Response Thermal and Swelling (CARTS) code fuel performance analysis.

In the ATR test environment, the total heating rate in the MOX fuel pellet is the sum of the neutron, prompt gamma, and fission products delayed gamma heating (in the fuel pellet and coming from the ATR core), which is a rather complicated process. However, the fission heating tally (F7) in MCNP assumes all the prompt gamma heat generated from the fission in the fuel pellet is deposited locally without any leakage, which can compensate for the incoming gamma heating from ATR core. Furthermore, the total heating rate is dominated by fission heat (about 96%) in the fuel pellet. So, to simplify the as-run physics analysis, only the fission heat tally F7 was chosen for the total heat rate (LHGR) and burnup (LHGR-estimated burnup) conversion.

WG-MOX FUEL TESTING ASSEMBLY MODEL

MCNP can model extremely complex three-dimensional geometries. MCWO is quite accurate over a given region because MCNP-generated reaction rates are integrated over the continuous-energy nuclear data and the space within the region. Thus, any oddly or regularly shaped region in MCNP can usually be depleted. Applying this capability allows calculation of detailed nuclide concentration and power distributions within the MOX capsule as functions of burnup. Its disadvantage is a longer computational time to achieve the required tally precision and minimize statistical fluctuations in the results.

There are three MOX fuel test sections axially, with the center section at the core midplane, and three fuel capsules in each section, for a total of nine fuel capsules in the test assembly, which were all included in the ATR MCNP Core Model (ATRM) as shown in Figure 1. The WG-MOX test fuel pellet comprises five percent PuO_2 and 95% depleted UO_2. Each fuel capsule is 0.415 cm in radius and 15.24 cm in length and contains 15 MOX fuel pellets. Channel 1 capsules are located away from the ATR core center, behind the capsules in channels 2 and 3. The validated MCWO method was used to perform the neutronics analysis of WG-MOX fuel in the ATR. The prediction of nuclide profiles and burnup distributions in irradiated MOX fuel pellets via this new methodology can provide valuable data for MOX fuel performance evaluation.

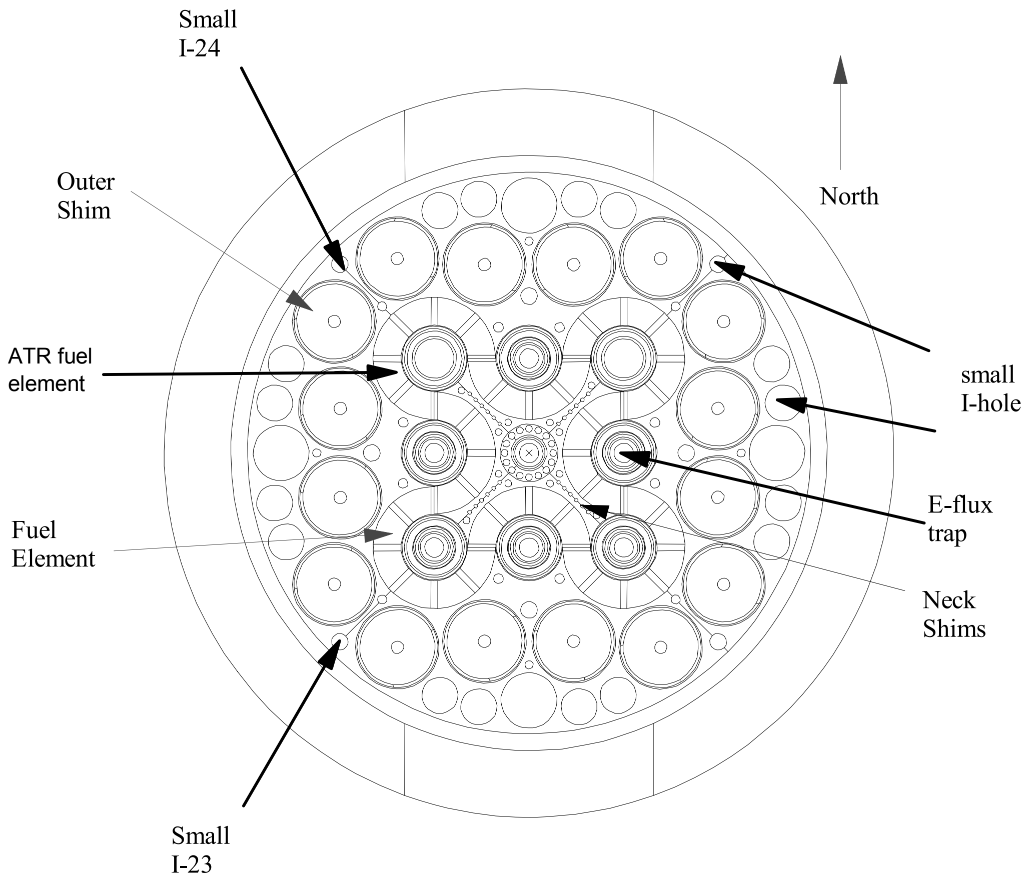

Figure 1. ATR MCNP core model cross-section view.

RESULTS AND DISCUSSION

The experimental results of the Average Power Test (APT) include observations from the fuel fabrication process, PIE findings, U and Pu isotopic composition, and MOX fuel burnup.[2,3,4,5] All of the capsules were visually examined in the transfer canal at the ATR during the shuffling and transfer to ORNL for post-irradiation examination (PIE). All of the irradiated capsules appeared as fresh as they did at the original insertion. No changes in the external dimensions were noted. No appreciable scratches or wear spots were observed as might occur from fretting. MCWO was used to track fuel burnup and heat rates as functions of irradiation time. In summary, no anomalous indications were seen.

The weapons-grade mixed oxide (WG-MOX) fuel capsules were located in the small I-24 hole PHASE-I to PHASE-IV-1A. This report describes the results of the detailed MCNP Advanced Test Reactor (ATR) quarter core model physics analysis performed to provide the integrated fast neutron flux (E > 0.1 MeV, and > 1.0 MeV) at the end of PHASE-I, PHASE-II, PHASE-III, and PHASE-IV-1B of the MOX cladding Zr tubes.

The following flux normalization factor was used with the corresponding quadrant power (MW) to convert the MCNP tallies to the integrated fast neutron flux. Flux normalization factor = (fission neutron / fission) x (fission /MeV) x (MeV / quadrant core power MW-sec)

$$= (2.43) \times (0.005) \times (6.25E+18 \text{ MeV/MW-sec})$$
$$= 7.5938 \times 10^{16} \text{ n/sec per quadrant MW.}$$

MCWO-Calculated, Phase-I

Zr-cladding Fast Neutron Fluence distributions at the end of Phase-I Irradiation

Using the detailed ATR quarter core model calculated neutronic tallies, the MCWO-calculated Zr-cladding fast neutron fluence (E > 0.1 MeV and E > 1.0 MeV) distributions at the end of Phase-I Irradiation are tabulated in Table 1. At the end of the Phase-I irradiation, the MCWO-calculated Zr-cladding fast neutron fluences of the removed MOX capsules 1 and 8 are 2.68 and 2.68 x 10^{20} n/cm^2, respectively.

Table 1. The integrated fast neutron flux (E > 0.1 MeV and E > 1.0 MeV) of MOX fuel pin's Zr-cladding at the end of Phase-I irradiation.

Target location		Capsule ID[a]	Fast neutron fluence (E > 0.1 MeV)	Fast neutron fluence (E >1.0 MeV)
			N/cm^2	
Top Zr-cladding	Back 1	4	3.79E+020	2.10E+020
	Left 2	10	4.47E+020	2.49E+020
	Right 3	3	4.40E+020	2.49E+020
Middle	Back 4	5	4.08E+020	2.26E+020
	Left 5	1	4.77E+020	2.68E+020
	Right 6	8	4.75E+020	2.68E+020
Bottom	Back 7	13	3.83E+020	2.13E+020
	Left 8	9	4.52E+020	2.52E+020
	Right 9	2	4.52E+020	2.52E+020

MCWO-Calculated, Phase-II

Zr-cladding Fast Neutron Fluence distributions at the end of Phase-II Irradiation

Using the detailed ATR quarter core model calculated neutronic tallies, the MCWO-calculated Zr-cladding fast neutron fluence (E > 0.1 MeV and E > 1.0 MeV) distributions at the end of Phase-II Irradiation are tabulated in Table 2. At the end of the Phase-II irradiation, the MCWO-calculated Zr-cladding fast neutron fluences of the removed MOX capsules 9 and 2 are 6.78 and 6.79 x 10^{20} n/cm^2, respectively.

MCWO-Calculated, Phase-III

Zr-cladding Fast Neutron Fluence distributions at the end of Phase-III Irradiation

Using the detailed ATR quarter core model calculated neutronic tallies, the MCWO-calculated Zr-cladding fast neutron fluence (E > 0.1 MeV and E > 1.0 MeV) distributions at the end of Phase-III Irradiation are tabulated in Table 3. At the end of the Phase-III irradiation, the MCWO-calculated Zr-cladding fast neutron fluences of the removed MOX capsules 10 and 3 are 9.82 and 9.70 x 10^{20} n/cm^2, respectively.

The loading pattern for the Extension of Phase III-2 for Equalization of Burnup was suggested by the project managers. At completion of Phase-III irradiation (Cycle 122C, August 6), the five capsules kept at INEEL for participation in Phase IV. Capsules 5, 6, and 12 alone are in the extension of Phase III-2. The purpose would be to equalize capsule burnup before proceeding beyond 30 GWd/MT, and could be best accomplished by placing Capsules 6 and 12 in the two front middle positions with Capsule 5 in the back middle position. The MCWO-calculated Zr-cladding fast neutron fluence (E > 0.1 MeV and E > 1.0 MeV) distributions at the end of Phase-III-2 extension Irradiation are tabulated in Table 3-A.

Table 2. The integrated fast neutron flux (E > 0.1 MeV and E > 1.0 MeV) of MOX fuel pin's Zr-cladding at the end of Phase-II irradiation.

Target location		Capsule ID[a]	Fast neutron fluence (E > 0.1 MeV)	Fast neutron fluence (E > 1.0 MeV)
			N/cm^2	
Top Zr-cladding	Back 1	6	6.89E+020	3.94E+020
	Left 2	10	1.19E+021	6.70E+020
	Right 3	3	1.17E+021	6.61E+020
Middle	Back 4	5	9.10E+020	6.37E+020
	Left 5	4	1.02E+021	7.06E+020
	Right 6	13	1.01E+021	6.96E+020
Bottom	Back 7	12	6.86E+020	4.00E+020
	Left 8	9	1.19E+021	6.78E+020
	Right 9	2	1.19E+021	6.79E+020

Table 3. The integrated fast neutron flux (E > 0.1 MeV and E > 1.0 MeV) of MOX fuel pin's Zr-cladding at the end of Phase-III irradiation.

Target location		Capsule ID[a]	Fast neutron fluence (E > 0.1 MeV)	Fast neutron fluence (E > 1.0 MeV)
			N/cm^2	
Top Zr-cladding	Back 1	6	1.24E+021	7.00E+020
	Left 2	10	1.76E+021	9.82E+020
	Right 3	3	1.74E+021	9.70E+020
Middle	Back 4	5	1.33E+021	8.65E+020
	Left 5	sst	2.59E+020	1.03E+020
	Right 6	sst	2.59E+020	1.05E+020
Bottom	Back 7	12	1.23E+021	7.13E+020
	Left 8	4	1.60E+021	1.03E+021
	Right 9	13	1.59E+021	1.02E+021

Table 3-A. The integrated fast neutron flux (E > 0.1 MeV and E > 1.0 MeV) of MOX fuel pin's Zr-cladding at the end of Phase-III extension irradiation.

Target location		Capsule ID[a]	Fast neutron fluence (E > 0.1 MeV)	Fast neutron fluence (E > 1.0 MeV)
			N/cm^2	
Top Zr-cladding	Back 1	*SST		
	Left 2	*SST		
	Right 3	*SST		
Middle	Back 4	5	1.57E+021	9.90E+020
	Left 5	6	1.54E+021	8.60E+020
	Right 6	12	1.53E+021	8.75E+020
Bottom	Back 7	*SST		
	Left 8	*SST		
	Right 9	*SST		

MCWO-Calculated, Phase-IV

Zr-cladding Fast Neutron Fluence distributions at the end of Phase-IV part 1 Irradiation

Using the detailed ATR quarter core model calculated neutronic tallies, the MCWO-calculated Zr-cladding fast neutron fluence (E > 0.1 MeV and E > 1.0 MeV) distributions at the end of Phase-IV part 1A and 1B Irradiation are tabulated in Table 4. At the end of the Phase-IV part 1A and 1B irradiation, the MCWO-calculated Zr-cladding fast neutron fluences (E > 1.0 MeV) of the removed MOX capsules 4 and 13 are 1.41 and 1.39 x 10^{21} n/cm^2, respectively.

Table 4. The integrated fast neutron flux (E > 0.1 MeV and E > 1.0 MeV) of MOX fuel pin's Zr-cladding at the end of Phase-IV part 1A irradiation.

Target location		Capsule ID[a]	Fast neutron fluence (E > 0.1 MeV)	Fast neutron fluence (E > 1.0 MeV)
			N/cm^2	
Top Zr-cladding	Back 1	sst		
	Left 2	6	1.95E+021	1.08E+021
	Right 3	12	1.94E+021	1.09E+021
Middle	Back 4	5	1.97E+021	1.20E+021
	Left 5	4	2.04E+021	1.27E+021
	Right 6	13	2.03E+021	1.25E+021
Bottom	Back 7	sst		
	Left 8	sst		
	Right 9	sst		

Table 4-A. The integrated fast neutron flux (E > 0.1 MeV and E > 1.0 MeV) of MOX fuel pin's Zr-cladding at the end of Phase-IV part 1B irradiation.

Target location		Capsule ID[a]	Fast neutron fluence (E > 0.1 MeV)	Fast neutron fluence (E > 1.0 MeV)
			N/cm^2	
Top Zr-cladding	Back 1	sst		
	Left 2	6	2.21E+021	1.21E+021
	Right 3	12	2.19E+021	1.22E+021
Middle	Back 4	5	2.21E+021	1.32E+021
	Left 5	4	2.31E+021	1.41E+021
	Right 6	13	2.30E+021	1.39E+021
Bottom	Back 7	sst		
	Left 8	sst		
	Right 9	sst		

REFERENCES

1. G. S. CHANG and J. M. RYSKAMP, "Depletion Analysis of Mixed Oxide Fuel Pins in Light Water Reactors and the Advanced Test Reactor," Nucl. Technol., Vol. 129, No. 3, p. 326-337 (2000).

2. R. N. MORRIS, C. A. BALDWIN, B. S. COWELL, S. A. HODGE, et. al. “MOX Average Power Early PIE: 8 GWd/MT Final Report,” Oak Ridge National Laboratory, ORNL/MD/LTR-172, November 1999.

3. R. N. MORRIS, C. A. BALDWIN, S. A. HODGE, L. J. OTT, C. M. MALONE, N. H. PACKAN, “MOX Average Power Intermediate PIE: 21 GWd/MT Final Report,” Oak Ridge National Laboratory, ORNL/MD/LTR-199, December 2000.

4. R. N. MORRIS, C. A. BALDWIN, S. A. HODGE, N. H. PACKAN, “MOX Average Power 30 GWd/MT PIE: Final Report,” Oak Ridge National Laboratory, ORNL/MD/LTR-212, November 2001.

5. R. N. MORRIS, C. A. BALDWIN, S. A. HODGE, N. H. PACKAN, “MOX Average Power 40 GWd/MT PIE: Final Report,” Oak Ridge National Laboratory, ORNL/MD/LTR-241, Volume 1, August 2003.

6. J. BRIESMEISTER (Editor), “MCNP—A General Monte Carlo N-Particle Transport Code, Version 4C,” LA-13709-M, Los Alamos National Laboratory (2000).

7. A. G. CROFF, “ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials,” Nuclear Technology, Vol. 62, pp. 335-352, 1983.

8. G. S. CHANG, R. C. PEDERSEN, “Weapons-Grade MOX Fuel Burnup Validation in ATR,” Trans. Am. Nucl. Soc., Vol. 84, p. 239, 2001.