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Abstract

The Advanced Test Reactor (ATR) is a high power (250 MW), high neutron flux research reactor operating in the United States. Powered with highly enriched uranium (HEU), the ATR has a maximum unperturbed thermal neutron flux rating of 1.0 x 10^{15} n/cm^2–s. Because of its high power and large test volumes located in high flux areas, the ATR is an ideal candidate for assessing the feasibility of converting an HEU driven reactor to a low-enriched core.

A detailed plate-by-plate MCNP ATR full core model has been developed and validated for the low-enriched uranium (LEU) fuel conversion feasibility study. Using this model, an analysis has been performed to determine the LEU density and U-235 enrichment required in the fuel meat to yield equivalent K-eff versus effective full power days (EFPDs) between the HEU and LEU cores. This model has also been used to optimize U-235 content of the LEU core, minimizing the differences in K-eff and heat flux profile between the HEU and LEU cores at 115 MW total core power for 125 EFPDs.

The LEU core conversion feasibility study evaluated foil type (U-10Mo) fuel with the LEU reference design of 19.7 wt% U-235 enrichment. The LEU reference design has a fixed fuel meat thickness of 0.330 mm and can sustain the same operating cycle length as the HEU fuel. Heat flux and fission power density are parameters that are proportional to the fraction of fission power deposited in fuel. Thus, the accurate determination of the fraction of fission power deposited in the fuel is important to ATR nuclear safety.

In this work, a new approach was developed and validated, the Tally Fuel Cells Only (TFCO) method. This method calculates and compares the fission power deposition fraction between HEU and LEU fuel plates. Due to the high density of the U-10Mo LEU fuel, the fission γ-energy deposition fraction is 37.12%, which is larger than the HEU's γ-energy deposition fraction of 19.7%. As a result, the fuel decay heat cooling will need to be improved. During the power operation, the total fission energy (200 MeV per fission) deposition fraction of LEU and HEU are 90.9% and 89.1%, respectively.

1. Introduction

The Advanced Test Reactor (ATR) is a high power (250 MW), high neutron flux research reactor operating in the United States. Powered with highly enriched uranium (HEU), the ATR has a maximum unperturbed thermal neutron flux rating of 1.0 x 10^{15} n/cm^2–s. Studies and evaluations are being performed on converting nuclear test reactors fueled with HEU to low-enriched uranium (LEU) under the reduced enrichment for research and test reactors (RERTR) program. Because of its high power and large test volumes located in high flux areas, ATR is a representative candidate for assessing the necessary modifications and evaluating the subsequent operating effects encountered when converting from HEU to LEU. The present work investigates necessary modifications and evaluates the subsequent operating effects of this conversion.

A detailed plate-by-plate MCNP (Goorley et al. 2004) ATR full core model was developed and validated for the LEU fuel conversion feasibility study. Using the current HEU U-235 enrichment of 93.0 % as a baseline, an analysis was performed which determined the LEU U-235 enrichment to be 19.7 wt% and the required fuel meat thickness to yield an equivalent K-eff versus effective full power days (EFPDs) between the HEU and LEU cores. Each ATR-HEU fuel element (FE) has 19 curved fuel plates each with a fuel meat thickness of 0.508 mm. The un-optimized (U-10Mo) fuel with a fixed fuel meat thickness of 0.330 mm (Chang et al. 2008) can sustain the same operating cycle length as HEU the fuel. The heat flux is propositional to the fraction of fission power deposited in the fuel plates. Thus, both heat flux and fission power density are important ATR safety parameters. In this work, a new approach to calculate and compare the fission power deposition fraction between HEU and LEU was developed and validated - the Tally the Fuel Cells Only (TFCO) method.

2. ATR HEU and LEU Plate-by-Plate Model

The ability to accurately predict K-eff and fission power distribution within the 19 fuel plates using the MCNP model is essential to the ATR-LEU core conversion design. The developed MCNP ATR full core plate-by-plate model is shown in Fig. 1. Its detailed validation against the ATR PDQ diffusion model with respect to ASUDAS data are reported in Chang et al. 2006.

In this work, ATR-HEU and ATR-LEU were evaluated in order to study the total fission energy deposition fraction in the fuel plate. The reference ATR-HEU FE specifications (Kim and Schnitzler 2005) are tabulated in Table 1. One of the ATR-LEU candidate (U-10Mo foil type) FE specifications is given in Table 2 (Chang et al. 2008).

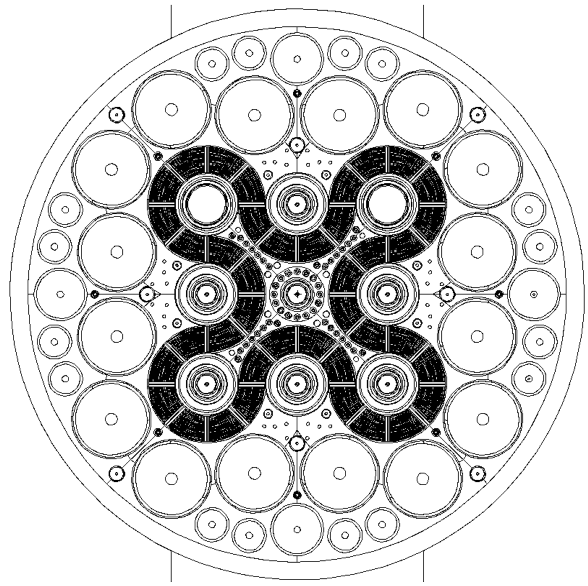

Fig. 1. ATR MCNP full core model with 19 fuel plates per FE

Table 1
Specifications for a standard ATR HEU FE with B-10 in the 4 inner/outer fuel plates

HEU Plate No.	Fuel Meat Thickness (mil)	Fuel Meat Volume (cc)	U-235 Mass (g)	B-10 Mass (g)	U-235 Density (g/cc)
1	20	23.69	24.3	0.063	1.026
2	20	29.54	29.1	0.078	0.985
3	20	31.12	38.7	0.044	1.243
4	20	32.70	40.4	0.045	1.235
5	20	34.29	52.1	--	1.520
6	20	35.87	54.6	--	1.522
7	20	37.45	57.0	--	1.522
8	20	39.03	59.4	--	1.522
9	20	40.61	61.8	--	1.522
10	20	42.19	64.2	--	1.522
11	20	43.78	66.6	--	1.521
12	20	45.36	69.0	--	1.521
13	20	46.94	71.4	--	1.521
14	20	48.52	73.8	--	1.521
15	20	50.10	76.3	--	1.523
16	20	51.69	64.0	0.071	1.238
17	20	53.27	65.9	0.073	1.237
18	20	54.22	53.8	0.143	0.992
19	20	52.64	52.6	0.143	0.999
Total	--	792.99	1075	0.660	--

Table 2
Specifications for an un-optimized ATR LEU FE with B-10 in the 4 inner/outer fuel plates

LEU Plate No.	Fuel Meat Thickness (mil)	Fuel Meat Volume (cc)	U-235 Mass (g)	B-10 Mass (g	U-235 Density (g/cc)
1	13.00	15.28	46.12	0.063	3.02
2	13.00	19.10	57.64	0.178	3.02
3	13.00	20.12	60.73	0.044	3.02
4	13.00	21.14	63.80	0.005	3.02
5	13.00	22.15	66.86	--	3.02
6	13.00	23.63	71.32	--	3.02
7	13.00	24.68	74.49	--	3.02
8	13.00	25.72	77.62	--	3.02
9	13.00	26.77	80.79	--	3.02
10	13.00	27.79	83.89	--	3.02
11	13.00	28.86	87.11	--	3.02
12	13.00	29.89	90.22	--	3.02
13	13.00	30.94	93.40	--	3.02
14	13.00	31.98	96.52	--	3.02
15	13.00	33.04	99.73	--	3.02
16	13.00	33.42	100.88	0.001	3.02
17	13.00	34.45	103.96	0.033	3.02
18	13.00	35.07	105.85	0.133	3.02
19	13.00	34.74	104.85	0.343	3.02
Total	--	518.77	1565.77	0.8	--

3. Fission Energy Deposition Fraction in Fuel Plates

The recoverable fission released energy of U-235 and Pu-239 are 200 MeV and 207.2 MeV, respectively. Once the fraction of the fission energy deposited in the fuel is known, multiplying it with the corresponding nuclides fission released energy (200 or 207.2 MeV), the fission energy deposited in the fuel cells can be determined. For simplicity, the U-235 fission energy of 200 MeV was chosen. The approximate energy release is as follows:

Kinetic energy of fission fragments	167 MeV
Fission products delayed β-energy	7 MeV
Kinetic energy of fission neutrons	5 MeV
Energy of prompt gamma rays	6 MeV
Fission products delayed γ-energy	7 MeV
Excess neutrons radioactive capture	8 MeV
Total	200 MeV

Both the fission fragment kinetic energy and the fission products delayed β-energy are assumed to be deposited 100% locally within the fuel plate (167 MeV + 7 MeV = 174 MeV). The fission neutron kinetic energy is assumed to completely escape from the fuel cells without any significant impact on determining the fission energy deposition fraction (relatively small energy release, 5 MeV). The remaining fission energy due to total γ-energy (6 MeV + 7 MeV + 8 MeV = 21 MeV) will be used to calculate the deposition fraction

To determine the fraction of fission energy deposited in the fuel plate, traditionally, one needs to calculate not only the fission energy deposited in the fuel, but also calculate the escaped γ-heat and neutron source deposited in the surrounding coolant, structural material, and peripheral reflector, which is not an easy task within the ATR model. Moreover, MCNP cannot compute the delayed γ-ray directly, a separate model must be setup for the delayed photon transport simulation.

Instead, for case A, the ATR full core model is modified to have reflecting boundaries at the side, top and bottom of the test loop, which forces the photon source generated in the fuel plates to be deposited as well as all the generated photon energy in the fuel plates and the surrounding materials in the test loop to be deposited without any source leakage.

Then, using the same model for case B, void the surrounding coolant, structural material, and peripheral reflector, which now forces all the photon energy to be completely deposited in the fuel plates. Note that in this approach we need to Tally the photon energy deposited in the Fuel Cells Only (TFCO).

The ratio of the tally results in the fuel cells between case A and B is the fission energy deposition fraction in the fuel. In the case of radioactive capture, the incident neutron is absorbed into the nucleus, thus raising the energy level of the nucleus. The compound nucleus then emits a photon to rid itself of the excess energy in the form of photon. Note that the excess neutrons radioactive captured γ-energy (8 MeV) could be produced outside of the fuel cell.

The typical unit lattice model in the next section, shows that the fraction of the (n,γ) captured in the fuel cell is 82.5%, which represents 98.6% of the neutrons radioactive captured γ-energy. As a result, TFCO can calculate an accurate ratio of the neutron induced fission and γ-heat source deposited in the fuel cells as demonstrated in the next section.

4. TFCO Method Validation

A simple and well defined typical PWR unit cell has been chosen as the basic unit cell configuration for the fuel neutronics analysis validation. UO_2 with 95% of theoretical density was used for the fuel composition. The fuel pins have diameter of 0.94-cm and are clad with 0.0864 cm of Zr. A gap of 0.008 cm separates the UO_2 pin from the clad tube. The study case for the UO_2 fuel compositions (12 month cycle) has an average of 1.85 wt% of U-235 and a lattice pitch of 1.5 cm. To account for the fuel and core boundaries, the lattice was extended from 1.5 to 1.8 cm and the upper to lower boundaries were extended from 366 cm to 424 cm with mixed water and structural materials (Chang et al. 2008) as shown in Fig. 2.

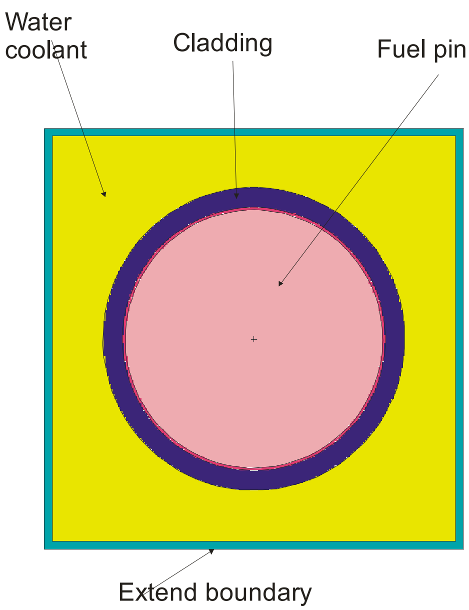

Fig. 2. PWR unit lattice model with extended boundary

First, the PWR unit lattice model was run with the MCNP KCODE option to generate the needed '*equilibrium*' photon spectrum in the fuel pin.

Then, the generated photon spectrum in the fuel pin was used in the lattice model with the MCNP fixed source option and SDEF card. The MCNP-calculated results generated with this fixed source option are tabulated in Table 3. Note, the F6 tallies have units of MeV per gram per photon and the results were normalized to 5 kW per unit lattice. The photon energy deposited in the fuel pin is 4.08 kW per unit lattice, which represents a photon energy deposition fraction in fuel pin (PF_{SDEF}) of 81.62%. Next, using the same SDEF model with reflecting boundaries, void all cell materials except the fuel pin. This forces all fixed photon source to be absorbed only in the fuel pin.

The MCNP-calculated F6 tallies (F6-p) with all materials defined and the F6 tallies (F6-p,void) with all materials voided except for the fuel pin are:

(F6-p) = 2.0171E-3 MeV/gram/photon,
(F6-p,void) = 2.4526E-3 MeV/gram/photon,
respectively.

The ratio of (F6-p) to (F6-p,void) was calculated to be 82.24%, which is the fraction of photon energy deposited in a fuel pin (PF_{VOID}).

These results demonstrate excellent agreement between the PF_{SDEF} and $PF_{VOID.}$ As a result, it can be concluded that the TFCO method can accurately predict the photon energy deposition fraction in the fuel pin.

Table 3
PWR unit lattice model with MCNP *SDEF* option calculated γ-heat deposition rate and fraction

Cell/Region	Fraction of fission heat deposition	Fission heat deposited
Fuel pin	0.81620	4.08
Gap	0.00000	0.00
Cladding	0.09742	0.49
Top extended region	0.00149	0.01
Bottom extended region	0.00149	0.01
Primary coolant	0.05823	0.29
Sides extended region	0.02519	0.13
KW per unit lattice	--	5.00

5. TFCO-Calculated Results and Discussion

As discussed in the TFCO method validation section, first, the ATR full core model was run with the MCNP KCODE option to generate the needed '*equilibrium'* photon spectrum in the fuel plates for both the ATR-HEU and ATR-LEU model. Then, the generated photon spectrum in the fuel plate was input to the ATR-HEU and ATR-LEU full core model with the MCNP fixed source option using the SDEF card. MCNP can be used to calculate the photon energy deposition tallies (F6-p) in the fuel plates only. To calculate all the γ-heat source deposited (F6-p,void) in the fuel plates locally, void all of the surrounding materials in the SDEF model except fuel plates. This requires all photon sources generated in the fuel plates to be deposited in the fuel plates without any leakage. The MCNP-calculated values for the ATR-HEUs (F6-p) and (F6-p,void), without normalization are:

(F6-p) = 1.404E-06 MeV/gram/photon,

(F6-p,void) = 6.702E-06 MeV/gram/photon,

respectively. Because of different U-235 enrichment, density, and spectrum between the ATR-HEU and ATR-LEU fuels, the MCNP-calculated total photon energy deposited in the ATR-LEU fuel plates (F6-p) and (F6-p,void), without normalization are:

(F6-p) = 1.437E-06 MeV/gram/photon,

(F6-p,void) = 3.872E-06 MeV/gram/photon,

respectively. The ratio of (F6-p) to (F6-p,void) is the fraction of photon energy deposited locally. The MCNP-calculated results indicate that the ratio (PF_{VOID}) of the photon source deposited in the ATR-HEU and ATR-LEU fuel plates (locally) are 19.66% and 37.12%, respectively. Since the ATR-LEU has a much higher density (16.88 g/cc) than the ATR-HEU (4.4 g/cc), the ATR-LEU was expected to have the higher photon energy deposition fraction of 37.12%. Although, the current HEU cooling system can handle 100% of the fission product decay heat deposition, we need to carefully evaluate the safety margin of the LEU fuel decay heat removal.

The fraction of the total fission energy deposited in the ATR-HEU fuel plates is (174 MeV + 21 MeV × 19.66%) / 200 = 89.06%; as compared to the fraction of the total fission energy deposited in the ATR-LEU fuel plates which is (174 MeV + 21 MeV × 37.12%) / 200 = 90.90%. As a result, during the power operation, the ratio of the total fission energy deposited in the ATR-HEU relative to the ATR-LEU fuel is 97.98%, which is not a significant difference.

6. Conclusions

The heat flux and fission power density are important ATR parameters as well as being proportional to the fraction of fission power deposited in the fuel plates, thus the fraction of fission power deposited in the fuel plates from neutron fissions is important. The developed TFCO method can calculate the ratio of (F6-p) to (F6-p,void) efficiently, which can provide a best-estimate fraction of fission energy deposited locally. The photon energy deposited in the ATR-HEU and ATR-LEU fuel plates (locally) are 19.66% and 37.12%, respectively. However, the fraction of the total fission energy deposited in the HEU and LEU fuel plates are 89.06% and 90.90%, respectively. The same modeling approach to accurately calculate the fission energy deposition ratio can also be used in the Advanced Fuel Cycle Initiative (AFCI), Reduced Enrichment for Research and Test Reactors (RERTR), and Advanced Gas-cooled Reactor (AGR) fuel research and development programs.

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