A Computational Model of the High Flux Isotope Reactor – Validation and Application to Low Enriched Uranium Fuels

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1. ABSTRACT

For most of the operating life of the High Flux reactor, direct measurements or expert opinion based on past measurements have been the sources of data for the reactivity worth and heat generation rate of specimens to be irradiated in the reactor. With the current policy of the Department of Energy being to operate the reactor to 2035 and with the goal of maximizing the reactor's capabilities and given the availability of sophisticated numerical analysis programs, a transition to computationally based estimates is believed to be the most economical and accurate practice for the future given that many of the individuals who performed the measurements are no longer employed at Oak Ridge National Laboratory. Computationally-based estimates are believed to be less prone to error, i.e. incorrect estimation of heating rates or reactivity, if the computational methodology is implemented correctly. This paper presents a three dimensional computational model of the HFIR.

The model was constructed for use with the MCNP-V code (Ref. 1) and is documented in Ref. 2. The model reflects the as-built configuration of the reactor and includes all redesign, modifications, and upgrades to HFIR that have occurred since its first approach to criticality in 1965. It uses continuous energy neutron ENDF/B-VI cross-section data libraries. The model was validated by comparison to the actual reactor performance as will be discussed subsequently.

Fig.1. The High Flux Isotope Reactor

The model has been used to perform safety basis calculations including establishing a basis for core reactivity changes due to experiment loadings and to assess power distribution responses to a variety of proposed experimental loadings (Ref. 3). Neutron flux, fission rate, neutron, gamma, and other reaction rates have all been tallied using this MCNP model. A brief review of calculations performed for target and capsule irradiations will be provided along with a brief discussion of the manner in which these calculations support HFIR operation and the safety basis.

Scoping calculations have been performed with the model to assess the possibility of converting from highly enriched uranium in the form of U_3O_8 to low enriched (20%) uranium in the form of uranium-molybdenum alloy. With U-Moly LEU fuel and maintaining the current ^{235}U core loading and fuel distribution profile (HFIR fuel thickness is variable along the plate length), the beginning of life excess core reactivity is significantly reduced but the reactor could still attain criticality.

2. HFIR DESCRIPTION

The HFIR is a multipurpose isotope production and test reactor. The HFIR site at Oak Ridge is shown in Fig. 1. With a rated power of 100 MW, and currently operating at 85 MW, HFIR produces the world's highest steady-state neutron thermal flux (measured in the central target region as 2.6×10^{15} neutrons/cm^2 • s at a power of 85 MW), making it a one-of-a-kind facility, worldwide.

The HFIR is a pressurized light-water-cooled and -moderated, flux-trap type reactor that uses highly enriched ^{235}U as the fuel. The reactor core (shown in Fig. 2) consists of a series of concentric annular regions, each approximately 61 cm high (fueled height is 51 cm). The center of the core is a 12.70-cm-diam cylindrical hole, referred to as the "flux trap," which contains 37 vertical experimental target sites.

Surrounding the flux trap are two concentric fuel elements separated by a water region. The average core life cycle for recent (2004-2005) target loadings is 22–24 d at 85 MW (depending on quantity and type of material being irradiated).

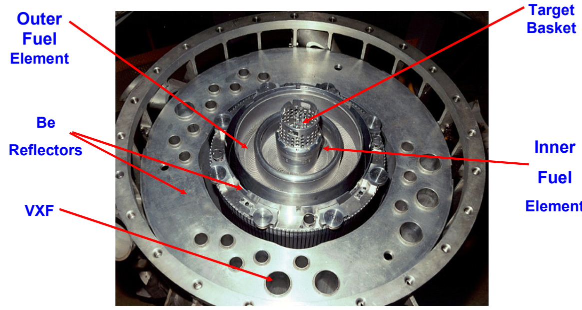

Fig.2. Top View of Reactor and Beryllium Reflector

The control plates, in the form of two thin, poison-bearing concentric cylinders, are located in an annular region between the outer fuel element and the beryllium reflector. These plates are driven in opposite directions. Reactivity is increased by downward motion of the inner cylinder, which is used only for shimming and regulation; that is, it has no fast safety function. The outer control cylinder consists of four separate quadrants, each having an independent drive and safety release mechanism. Reactivity is increased as the outer plates are raised. All control plates have three axial regions of different poison content designed to minimize the axial peak-to-average power-density ratio throughout

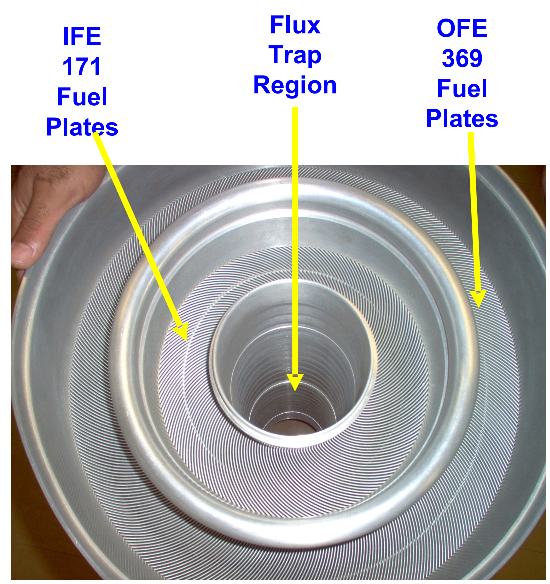

Fig. 3. HFIR fuel elements

the core lifetime. Any single rod or cylinder is capable of “shutting the reactor down”.

The fuel region is surrounded by a concentric ring of beryllium that serves as a reflector and is approximately 30 cm thick. This, in turn, is subdivided into three regions: the removable reflector, the semipermanent reflector, and the permanent reflector. The beryllium is surrounded by a water reflector of effectively infinite thickness. In the axial direction, the reactor is reflected by water.

The reactor core assembly is contained in a 244-cm-diam steel pressure vessel located in a pool of water. A few key parameters of HFIR are presented in Table 1. Significant components of the reactor are identified in Fig. 2.

Table 1. Key parameters of HFIR

Reactor power, MW	85	
Active core height, cm	50.8	
Number of fuel elements	2	
Fuel type	U_3O_8—aluminum	
Total ^{235}U loading, kg	9.43	
Enrichment, %	93.1	
Fuel element parameters	***Inner***	***Outer***
Number of fuel plates	171	369
^{235}U loading, kg	2.60	6.83
Average fuel density, gU/cm^3	0.776	1.151
^{235}U per plate, g	15.18	18.44
Burnable poison (^{10}B), g	2.8	None
Fuel plate thickness, cm	0.127	0.127
Coolant channel , cm	0.127	0.127
Minimum clad thickness, mm	0.25	0.25
Fuel plate width, cm	8.1	7.3
Fuel cycle length, d	~24	
Coolant inlet temperature, °F	120	
Coolant outlet temperature, °F	169	
Fuel plate centerline temp. °F	323	

3. FUEL ELEMENT

The HFIR consist of two concentric fuel elements separated by a water region (see Fig. 3).

The inner fuel element (IFE) consists of 171 fuel plates; each plate formed in an involute shape. The fuel plates are U_3O_8-Al fuel meat with the uranium being highly enriched (93 wt. % ^{235}U). Each plate contains 15.18 g ± 1% of ^{235}U distributed along the involute arc in a varying thickness so as to reduce power peaking in the fuel plate (shown in Fig. 4) and clad with aluminum. The plates are separated by 50 one-thousandth inch gaps (filled with water during operation in the reactor), and are held in place by two cylindrical aluminum side walls.

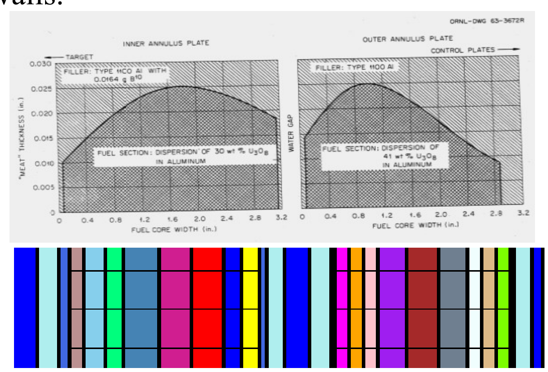

Fig. 4. Fuel “meat” distributions, MCNP Modeling compared to engineering drawing

The inner fuel element contains boron (^{10}B) as a burnable poison, primarily to help shift the power distribution from the inner element to the outer element.

The IFE fueled area is modeled by dividing it into 56 cells, 8 radial fuel regions and 7 axial fuel layers. The eight radial regions are included to approximate the varying, effective ^{235}U concentration in the radial direction of the fuel plate. The fuel is modeled by homogenizing the uranium “meat,” aluminum, and water between the plates into eight radial fueled regions.

The outer fuel element (OFE) consists of 369 fuel plates, each plate having an involute shape. Each plate contains 18.44 g ± 1% of ^{235}U. The OFE fueled area is modeled by dividing it into 63 cells, 9 radial fueled regions, and 7 axial fueled layers. The nine radial regions represent the different effective ^{235}U concentrations (atoms per barn*cm) in the radial direction of the fuel plate at beginning-of-cycle.

4. BENCHMARK RESULTS

The results of the MCNP criticality calculations were calculated using the final model (HFV4.0). The benchmarked values are shown in Table 2, and Figures 5 and 6.

Table 2. MCNP calculation results

keff (combined coll./abspt/track-length)	1.00870 ± 0.00013
Number of neutrons produced per fission	2.439
Average neutron energy causing fission	0.023304 ev.
Fission neutrons produced per neutron absorbed (capture + fission) in cells w/ fission	1.7412

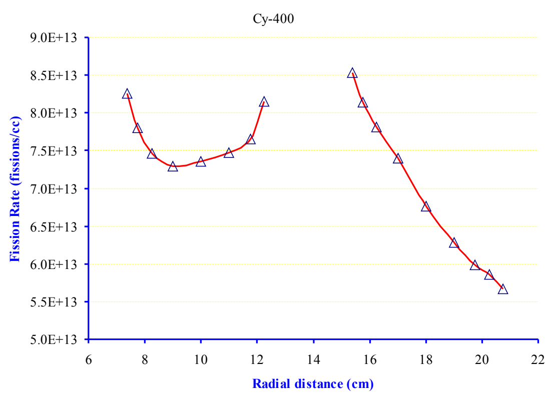

Fig. 5. Calculated fission rate [fissions/(cm^3*s)]

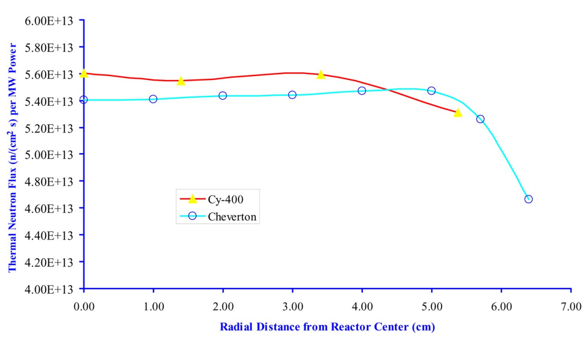

Fig. 6. Total neutron flux [n/(cm^2*s)

5. MODEL APPLICATIONS

The HFV4.0 model has been and is being used to calculate the neutronics effects of a number of upgrades, experiments and proposed studies at the HFIR.

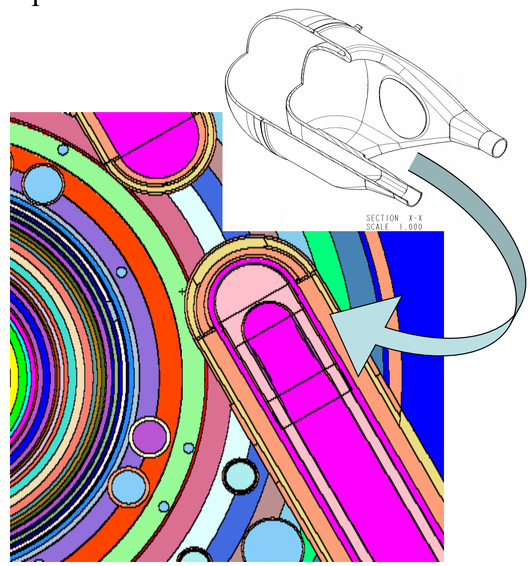

Fig. 7. HFIR Cold Source

Cold Source

Prediction of heating rates in the cold source moderator vessel currently being installed at HFIR was performed using a variant of the MCNP model (see Fig. 7 for a visualization of the MCNP model of the cold source). Liquid hydrogen will flow inside Al vessel with He coolant adjacent to portions of the vacuum tube. Al is heated from prompt fission gammas, delayed gammas, neutron absorption, and activation product decay. Total heating rates were calculated with the MCNP model and the hottest spot in the moderator vessel for nominal conditions at reactor power of 85 MW was calculated to be 2.6 W/g.

Irradiation Experiment Safety

The number of hydraulic tubes in the central target region was increased from one to three in June 2005. Hydraulic tubes allow access to the high flux region with the reactor operating, each tube can accommodate 9 separate targets. The MCNP model is to be used for estimating target worth and heat rates in the hydraulic tube and as such was validated with Cd rabbit measurements and activation wires (Ref. 4 and 5; see Fig. 8 for enlargement of one hydraulic tube in the MCNP model). Agreement between calculation and experiment was found to be within experimental uncertainty.

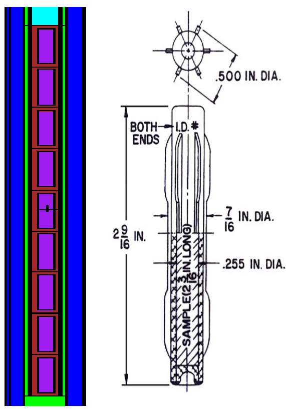

Fig. 8. HT Modeling

Be Internal Reflector

The HFIR staff are considering the installation of a beryllium internal reflector by the use of beryllium rods in the target region in order to increase the reactivity of the HFIR. Five beryllium loading arrangements were investigated and are shown in Fig. 9. Results showed an improved neutron economy, and consequently an increase the fuel cycle length. Calculated values are shown in Table 3. Calculations also confirmed that perturbation in local power density (power profile) were acceptable.

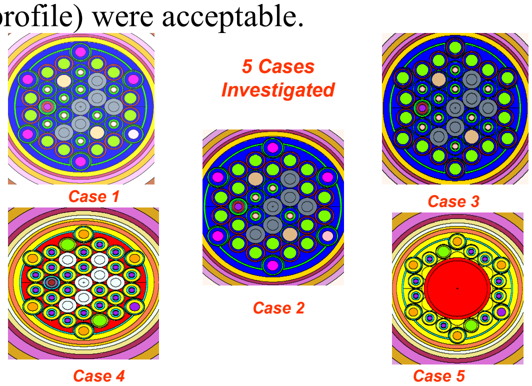

Fig. 9. Beryllium Loading Arrangements

Fuel cycle and core depletion

The model can be automatically linked to the ORIGEN code to perform core depletion studies, via linkage codes such as MONTEBURNS (Ref. 6), ALEPH, and others. This linkage gives the model the capability of calculating K-eff, fuel isotopic composition, fluxes, fission rate, and other neutronics parameters at any point in the cycle.

The design of a new advanced fuel element of a 12 kg ^{235}U core is currently being investigated. The model is being used to calculate the new core neutronics parameters, and to estimate the increase in fuel cycle length. Preliminary results show a substantial increase in the length of the fuel cycle (see Fig. 10).

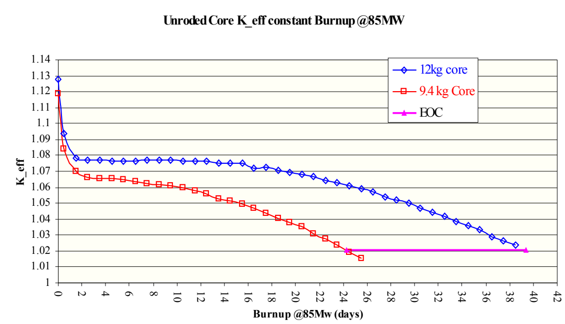

Fig. 10. MCNP K_eff calculation results of current and proposed 12kg core

Table 3. MCNP calculation results for Be reflector effect on BOC core reactivity

Case number or reference	Final *keff* (col/abs/trk len)	Increase in reactivity	
		(absolute)	(cents)
Cycle-400	1.00863 ± 0.00012	—	—
Case 1—12 beryllium rods	1.01258 ± 0.00013	0.00428	56.32
Case 2—18	1.01468 ±	0.00605	79.61

beryllium rods	0.00012		
Case 3—18 Be rods PTP	1.01418 ± 0.00012	0.00555	73.03
Case 4—central solid Be reflector	1.02090 ± 0.00023	0.0126	165.79
Case 5—Be reflector over target region	1.02132 ± 0.00013	0.01302	171.32

Low Enriched Uranium (LEU)

During FY06, low enriched uranium (LEU) fuels will be studied with the MCNP model. These will be used to verify results of deterministic HFIR models (VENTURE diffusion theory; ATTILA finite element)

Scoping studies have shown that when the existing HEU loading is "changed" in MCNP model to LEU, k-eff at the beginning-of-cycle decreases from 1.008 to 0.930 ($10 loss in reactivity due to ^{238}U; the model included 20% enriched uranium, the same ^{235}U spatial distribution as the current fuel; the same control element position at beginning-of-cycle as the current fuel; the average U density in "meat" region of plate between clad increased from ~1 g U/cc to ~5 g U/cc). Criticality can be achieved by removing control elements from the core but comparison with VENTURE (Ref. 7) shows that the cycle length would be reduced from ~24 to ~4 days.

6. CONCLUSIONS

The MCNP model is a 3-D detailed and accurate representation of the HFIR cycle 400. Benchmark calculations of eigenvalues, neutron fluxes, and reaction rates were performed using the model and compared with other published and or measured values.

The model can accurately calculate reactor parameters with reasonable confidence. Model input in any region can easily be modified, in order to incorporate design changes, or experiments loading. Benchmark results are used as a reference to study the effect of new designs, modifications, and experiments.

7. REFERENCES

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