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STATUS OF DESIGN EFFORT FOR LEU FUEL FOR THE HIGH FLUX ISOTOPE REACTOR

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ABSTRACT

The feasibility of low-enriched uranium (LEU) fuel as a replacement for the current, high enriched uranium (HEU) fuel for the High Flux Isotope Reactor (HFIR) has been under study since 2006. The current design for inner and outer element fuel plates includes a fuel region with spatially varying thickness along two dimensions. The fuel region is "beveled axially", having reduced thickness regions of the U/Mo foils along the lower edges of the plates but not the upper edges. Reactor performance studies have been completed for conceptual plate designs and show that the reactor performance is unchanged or, for some parameters, slightly enhanced by converting to LEU fuel from HEU fuel and returning the reactor power to 100 MW from 85 MW. Two limiting transients have been analyzed and the response of the reactor with LEU fuel has been shown to be bounded by the current, HEU fuel.

1. Introduction

A description of the High Flux Isotope Reactor (HFIR) has been published in the Proceedings of the 2007 International Meeting on Reduced Enrichment for Research and Test Reactors [1]. During the past fiscal year, work continued on a design for a new, low enriched uranium (LEU) fuel for HFIR based on a uranium/molybdenum alloy (10 wt % molybdenum alloyed to uranium, termed U-10Mo). This design is documented in [2] and [3] with the work reported here being that performed since these references were issued.

The geometry of the LEU HFIR fuel plates and elements would be unchanged from the current, high enriched uranium (HEU) design but the fuel region inside the plate would be changed from

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the U_3O_8-dispersed-in-aluminum-particles to a metal sheet of U-10Mo – hereinafter termed a foil. A design goal is for the LEU fuel to maintain the HFIR neutron source performance at the level obtained from the current HEU fuel. Samples of the U-10Mo fuel are currently being irradiated at the Advanced Test Reactor (ATR) at the Idaho National Laboratory (INL). Manufacture of the fuel will not be performed by the contractor for Oak Ridge National Laboratory but specifications for the configuration and quality of the fuel are the responsibilities of the reactor operator.

2. Fuel plate design

In HFIR fuel plates, the thickness of the fuel material inside the plate is variable. A cross-section of the current, HEU fuel plate design is shown in Fig. 1 [width (direction along the plate) and thickness (direction through the plate) are shown; height (coolant flow direction) is into the page]. The LEU fuel will also be graded in a similar fashion as that shown in Fig. 1, i.e. the fuel thickness varies along the width of the plate. Unlike HEU fuel, the LEU plates will be beveled at the bottom edge (coolant exit location) for reasons that will be described subsequently. The foil will be approximately 8 cm (3 in.) wide, have a height of 50 cm (20 in.), and a variable thickness averaging approximately 300 μm (12 mils).

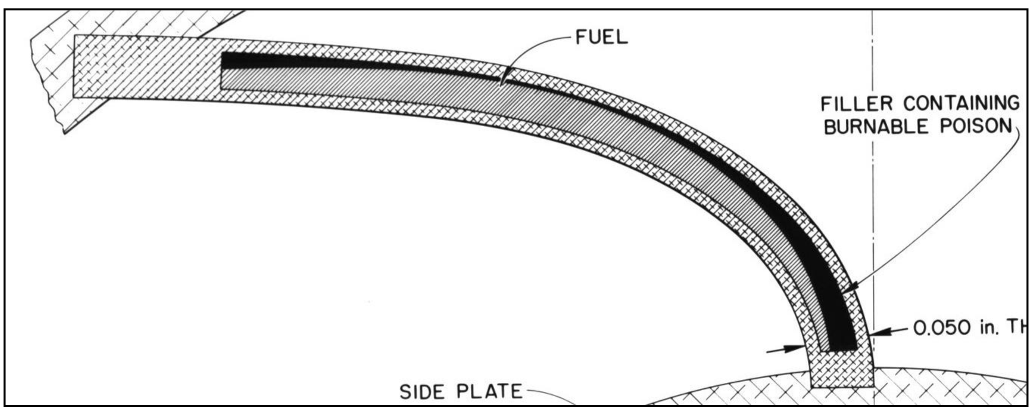

Fig. 1. Fuel profile inside current HEU inner element fuel plate.

Recent studies have shown that the ^{235}U content of the reactor core will increase from 9.4 kg with HEU to ~25 kg for LEU. Thus the total uranium content in a HFIR core will increase from 10.1 kg to ~125 kg. This increase is considerably greater than the expected HFIR LEU core loadings that were reported in [2] and is cause for careful review of neutronics methods and data.

A Monte Carlo-based depletion method – the ALEPH computational tool [4] – was used to determine the LEU core loading needed to maintain the same "calendar-day" cycle length as exists for the HEU fuel cycle. The MCNP-V [5] program was used as the "computational engine" called by ALEPH. The MCNP model was a slightly modified version of that documented in [6].

Two benchmarking activities for ALEPH/MCNP-V were conducted in conjunction with the LEU studies. A recent HFIR HEU fuel cycle (April-May 2004) having a cycle length of 24.33 days was modeled (actual control/safety element position as a function of time was included). The end-of-life calculated k-effective was 0.9990±0.0002 indicating excellent ability to predict cycle length. Two HFIR critical experiments, conducted in 1965, were modeled and spatially dependent local power densities were calculated and compared to measured values. Essentially all calculated values agreed with measured values to within the uncertainty of the experimental measurements. (Documentation of both of these studies is in progress.) Other studies [7], [8] have shown that similar MCNP models accurately calculate reflector and target flux levels and spectra and experiment reactivity worth for the HFIR. Collectively, these studies indicate excellent ability to calculate the power profile in the HFIR core (the heat source term for thermal hydraulic analyses) and fluxes at target locations (the indicators of reactor performance).

The MCNP model, modified to include LEU fuel, was used to calculate the impact on beginning-of-life multiplication factor due to increasing the uranium loading of the reactor core. Fig. 2 shows that stepwise, equivalent reactivity gains for beginning of life conditions require increasing larger changes in uranium loadings. The phenomenon is simply an example of self-shielding.

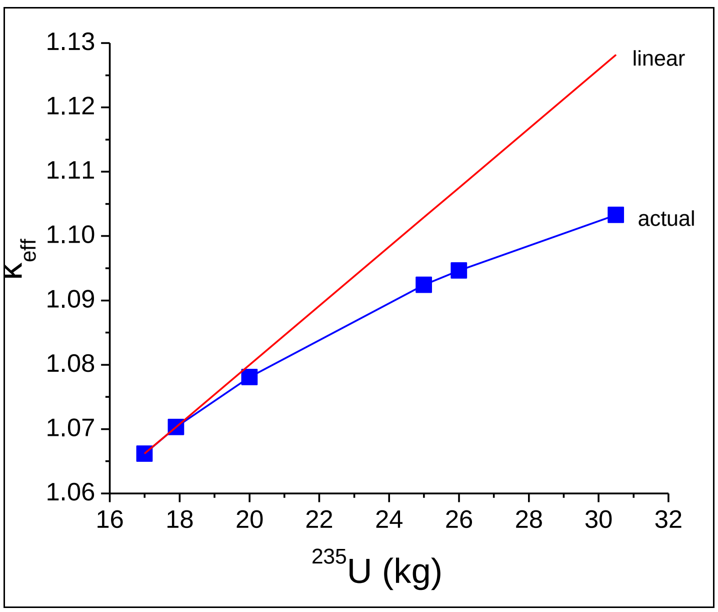

Fig. 2. Effective multiplication constant at BOC vs. ^{235}U loading for LEU fuel

Studies documented in previous reports demonstrated the need to increase reactor power in order to maintain the same flux performance for LEU as for HEU. An LEU fuel cycle would require an operating power of 100MW to achieve the same flux values at target locations as the current HEU core achieves with 85 MW. However, in past studies [1], [2], the calculated loading for equivalent calendar day operation in an LEU fuel cycle (17.9 kg ^{235}U in LEU) was based on the current, HEU-fuelled power level of 85 MW. Maintaining flux performance for the same calendar day cycle length with LEU as for the current fuel requires the addition of extra fuel to accommodate the 18% increase in energy produced during a cycle. That is, the end-of-life burnup for LEU fuel would be expected to be 2600 MWD versus the current, HEU end-of-life burnup of 2200 MWD; both fuel cycles having the same calendar day length but operating at different powers. From these studies, the significant increase in estimated LEU core uranium loading from values presented in earlier papers (125 kg versus 85 kg) is believed due to the requirement for increased reactor power for LEU fuel for the same calendar time as the current HEU cycle coupled with declining incremental reactivity worth of added uranium as the uranium inventory of the reactor core increases.

Using MCNP, a graded LEU profile was established for the inner and outer fuel element plates (IFE and OFE, respectively; HFIR has only two types of fuel plates). The width/thickness profile shown in Fig. 3 provides the minimal spatial variation in the power density distribution ("flattest" power profile).

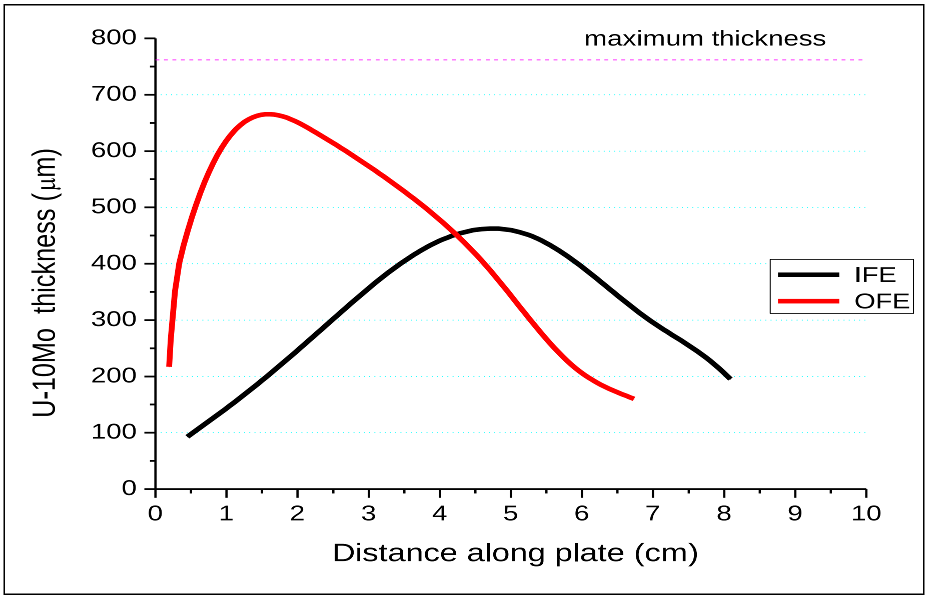

Fig. 3. Plate fuel region thickness profiles

Calculated flux performance for the current HEU HFIR and the proposed LEU HFIR is presented in Tables 1 and 2. The design goal of maintaining the current level of reactor performance appears to have been achieved. Predicted end-of-cycle isotopic inventory for the current, HEU fuel and the proposed LEU fuel is shown in Table 3. These data are the source terms for shipping case certification and accident dose consequence studies.

Table 1. Neutron fluxes at beginning-of-cycle

Location	Fuel	Thermal flux (n/cm^2s)	Epithermal flux (n/cm^2s)	Fast flux (n/cm^2s)
Central target	HEU LEU	2.2×10^{15} 2.3×10^{15}	1.3×10^{15} 1.2×10^{15}	1.1×10^{15} 1.0×10^{15}
Cold source edge	HEU LEU	6.8×10^{14} 8.1×10^{14}	2.4×10^{14} 2.8×10^{14}	9.0×10^{13} 1.0×10^{14}
Reflector r=27cm	HEU LEU	6.0×10^{14} 7.0×10^{14}	6.5×10^{14} 7.7×10^{14}	4.1×10^{14} 4.8×10^{14}

Table 2. Neutron fluxes at end-of-cycle

Location	Fuel	Thermal flux (n/cm^2s)	Epithermal flux (n/cm^2s)	Fast flux (n/cm^2s)
Central target	HEU LEU	2.3×10^{15} 2.5×10^{15}	1.1×10^{15} 1.2×10^{15}	9.9×10^{14} 1.0×10^{15}
Cold source edge	HEU LEU	8.3×10^{14} 8.3×10^{14}	2.4×10^{14} 2.7×10^{14}	8.9×10^{13} 9.9×10^{13}
Reflector r=27cm	HEU LEU	8.1×10^{14} 7.2×10^{14}	6.5×10^{14} 7.3×10^{14}	4.0×10^{14} 4.5×10^{14}

Table 3. End-of-cycle inventory data for equivalent cycle time
(Cycle length of 24.33 days; HEU burnup is 2068 MWD; LEU burnup is 2433 MWD)

Nuclide	HEU (g)	LEU (g)	Nuclide	HEU (g)	LEU (g)
B-10	0.203	0.746	Pm-147	11.960	15.390
B-11	12.480	10.280	Pm-148	0.257	0.257
Kr-86	15.840	18.140	Pm-148m	0.088	0.151
Zr-93	53.480	61.840	Pm-149	2.059	2.403
Mo-97	51.440	60.220	Sm-149	0.382	1.876
Tc-99	43.580	51.620	Sm-150	13.200	14.080
Ru-101	46.950	55.670	Sm-151	1.133	3.329
Ru-103	24.280	29.860	Sm-152	7.005	7.147
Rh-103	5.121	6.466	Sm-153	0.646	0.618
Rh-105	0.530	0.989	U-234	88.040	232.100
I-135	1.263	1.357	U-235	6785.0	22250.0
Xe-131	18.640	22.660	U-236	502.300	740.300
Xe-133	23.270	27.860	U-238	532.0	101700.0
Xe-135	0.054	0.271	Np-237	6.188	9.369
Cs-133	50.180	60.310	Np-238	0.134	0.121
Cs-134	1.531	1.266	Np-239	2.777	76.170
Cs-135	2.910	12.290	Pu-238	0.273	0.624
Ce-141	58.760	68.790	Pu-239	11.410	390.900
Pr-143	40.940	48.120	Pu-240	1.429	25.440
Nd-143	26.340	32.160	Pu-241	0.612	8.070
Nd-145	49.380	58.250	Pu-242	0.049	2.799
Nd-147	14.060	17.450			

3. Reactor Performance

Having met lifetime and flux performance goals, the proposed LEU fuel was evaluated for thermal hydraulic performance. Current HFIR analysis techniques are based on a special purpose computer program documented in [9]. This program only models heat transfer in one dimension (from plate surface to adjacent coolant). With the LEU fuel graded radially as shown in Fig. 3, the maximum operating power of the HFIR would be significantly less than 100 MW if the same margin-to-incipient-boiling is maintained as exists for the current, HEU fuel cycle.

HFIR employs down-flow of coolant. As with all reflected reactors, a local peak in the thermal flux, and thus the power density, occurs at the edge of the core. This local peaking at the base of the core was the source of the limit in operating power for LEU fuel. By reducing the thickness of the uranium foil to 125 μm for the bottom 2 cm of the fuelled plate – both inner and outer plates – the predicted maximum operating power of the HFIR with LEU fuel was increased to 103 MW with the same margin-to-incipient-boiling as exists for the current, HEU fuel cycle.

While the LEU fuel plate designs have the radial profiles for the fuel regions as shown in Fig. 3 for the entire height of the fuel plate except for the bottom-most 2 cm – in that lower region the fuel profile is flat - it is likely that lowest-cost fabrication techniques will lead to a thickness gradient in the lower, 2 cm region rather than a step change in thickness. Certainly, fabrication of LEU fuel plates would be simplified and therefore less expensive if the bevel (step change in thickness) at the lower edge of the fuel foil could be removed. Investigating this possibility requires developing multidimensional thermal hydraulic analysis methods/models to replace those in [9].

During the past year, development of a new, thermal hydraulic/structural analysis model for HFIR was instigated. The COMSOL, commercial finite element solver is the basis for the development of the new analysis tool. The new methodology will model heat conduction in three dimensions rather than one; both inside the fuel plate and turbulent, transverse conduction in the coolant. Preliminary studies have shown that impact of fully modeling heat conduction leads to significant reduction in the expected hottest temperature in the coolant at the core exit. Reduction in this value leads to an increase in allowable operating power without reduction from the existing margin-to-incipient-boiling that is the current safety basis for HFIR operation [10]. (These studies are currently being documented.)

4. Response of LEU core to transients

In the HFIR Safety Analysis Report [SAR,10], considerably more analysis is required to predict/resolve the consequences of potential reactor transients – though the potential may be very small – than for prediction of steady-state operation. To prepare for the eventual revision of the SAR for LEU fuel, two limiting transients have been analyzed. Safety studies were performed by preparing both HEU and LEU fuel plate models for the transient analysis computer program PARET [11, 12]. PARET is a computer code which iteratively solves for the neutronic-hydrodynamic-heat transfer aspects of the reactor under steady state and transient behavior. The first transient, a primary coolant pump shaft seizure, is defined as follows:

A mechanical failure within the primary pumps could cause a pump shaft to shear or seize. The transient resulting from this failure is similar to a single pump coast-down except that the reduction in flow is immediate. Also, a mechanical pump failure prevents operation of the associated pony motor.
[10, p. 15.3.3-2]

Table 4 summarizes the parameters used in this problem.

Table 4. Primary coolant pump seizure initial and boundary conditions

Parameter	**Value**
Transient time	100.0 s
Reactivity insertion	None from control system; no SCRAM
Boundary conditions	\$0.0 at 0.0 s \$0.0 at 100 s
Flow rate (kg/m^2s)	$2.044(10^4)$ from 0.0 s – 10.0 s $1.589(10^4)$ from10.0 s – 100.0 s
Time step (s)	10^{-3} (0.0 s- 9.0 s) 10^{-4} (9.0 s – 29.0 s) 10^{-3} (29.0 s – 100.0 s)

The initial conditions considered in these analyses are:

- Reactor power = HEU 87.6 MW, LEU 103 MW
- Reactor inlet temperature = 126.2°F (52.33°C)
- Reactor inlet pressure = 406.0 psig (420.7 psia [2.9Mpa])
- Secondary basin temperature = 88.2°F
- Primary flow rate = 15,840 gpm
- Letdown flow rate = ~120 gpm

[10, p. 15.3.3-21, -22]

Fig. 4 shows the fuel centerline temperatures in LEU fuel that result from the transient. Temperature values are similar to HEU fuel and far below concern for the aluminum clad.

The second transient examined was a control element ejection accident. This accident is one of a family of transients that refer to the sudden insertion or removal of reactivity into the core of the reactor starting from steady state condition with consequent change in power and temperature as well as changes in other properties of the fuel and the moderator. The analysis was performed to determine the probability of fuel damage during rapid transients. Fuel damage may occur if sufficiently high temperatures are achieved for metal softening to occur. A description of the transient is provided in Table 5.

The peak power achieved in a control element transient was found to be less for LEU fuel than for HEU. The total energy released in the transient was also found to be less. No fuel damage would be expected. (Further documentation of these studies is currently in preparation.)

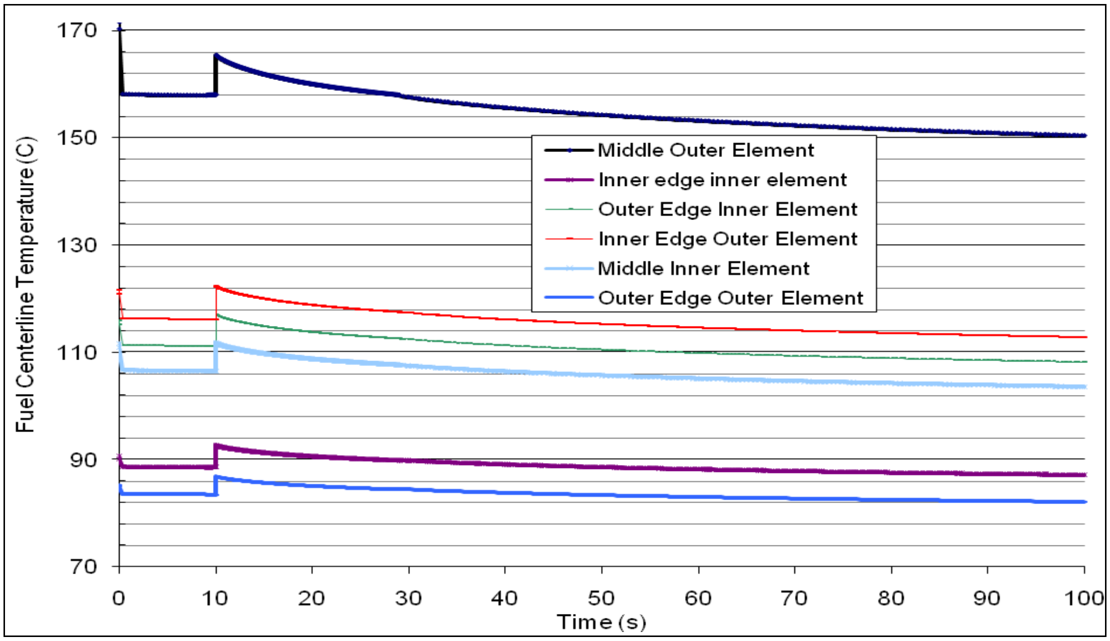

Fig. 4. Fuel temperature profile for several core regions for LEU for pump seizure transient

Table 5. Sequence of events for a control element ejection transient

Simulation time(s) for HEU	Event
0.000	Control cylinder ejection started; reactor power at 100 kW
0.89	Power rate exceeded (Rate > 20 MW/s).
0.931	Power exceeded (magnitude > 5 MW).
0.931	Reactor scram initiated.
0.941	Four safety plates insertion started.
0.946	Flux-to-flow ratio exceeded (Ration > 1.3).
1.10	Maximum reactor power occurs.
5.000	transient model terminated.

5. Conclusions

A conceptual design for an LEU fuel that maintains the current level of reactor performance has been developed. Behavior of the fuel under two, severe transient conditions has been studied. Fuel behavior is bounded by the current, HEU fuel. Refinements in plate design that would lead to lower manufacturing cost while maintaining reactor performance are under study. Fabrication and irradiation experience are needed to aid design development.

6. References

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