Neutron Physics of an LEU U-Mo Fueling Study
for the High Flux Isotope Reactor

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INTRODUCTION

The U.S. nonproliferation policy "to minimize, and to the extent possible, eliminate the use of highly enriched uranium (HEU) in civil nuclear programs throughout the world" has resulted in the conversion (or scheduled conversion) of many U.S. research reactors from HEU to low-enriched uranium (LEU). In support of this activity, a study was initiated in 2005 to investigate the feasibility of converting the High Flux Isotope Reactor (HFIR) to LEU fuel.

One important activity under the Reduced Enrichment for Research and Test Reactors (RERTR) Program has been the development of high-density LEU fuels. Recent efforts have focused on the development of dispersed (in aluminum) and monolithic uranium-molybdenum (U-Mo) alloy fuels.

The HFIR is a pressurized light-water-cooled and -moderated, flux-trap-type reactor that currently uses fuel highly enriched in ^{235}U (93 wt %) and is currently operating at 85 MW. The reactor core (Fig. 1) consists of two annular fuel elements, each approximately 61 cm high (fueled height is 51 cm). At the center of the core is a 12.70-cm-diam cylindrical hole, referred to as the "flux trap target" (FTT) region, which contains 37 vertical experimental target sites.

The HFIR fuel elements contain vertical, curved plates extending in the radial direction. The fuel elements are separated by a narrow water gap. The inner element contains 171 involute-shaped fuel plates, and the outer element contains 369 involute-shaped fuel plates, as shown in Fig. 2. The fuel plates are a sandwich-type construction with a fuel-bearing cermet bonded to a cladding of type-6061 aluminum. In the current HEU core, the HEU oxide is distributed (graded) along the arc of the involute fuel plate.

Control plates, in the form of two thin europium/tantalum-bearing concentric cylinders, are located in an annular region between the outer fuel element and the beryllium reflector. Reactivity is increased by downward motion of the inner cylinder, which is used only for shimming and regulation, and an upward motion of the outer control plates.

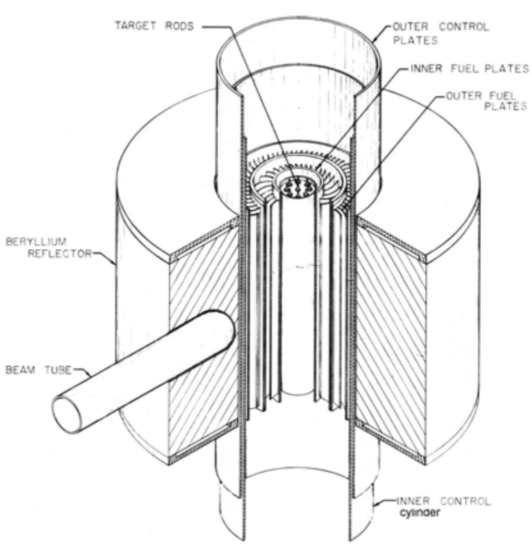

Fig. 1. Schematic of HFIR core configuration.

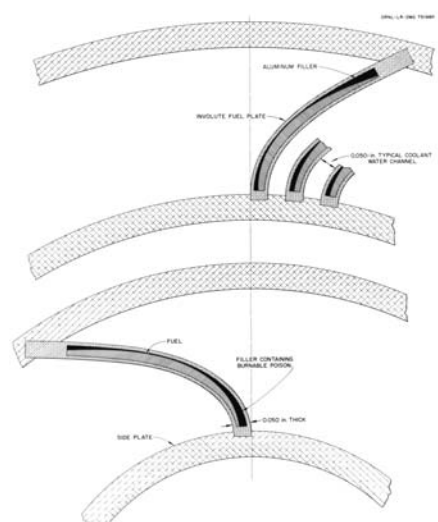

Fig. 2. Schematic of current HFIR HEU fuel plates.

The control plates and fuel elements are surrounded by a concentric ring of beryllium that serves as a reflector and is approximately 30 cm thick. The beryllium is surrounded by a light-water. In the axial direction, the reactor is reflected by light water.

DESCRIPTION OF THE ANALYSES

Earlier studies [1] investigated monolithic and dispersion fuel designs; this study considers U-7 wt % Mo dispersion fuel (in aluminum) with a net uranium density of 8.7 gU/cm^3, named "DISP87." (The 19.75 wt % enrichment U-7wt % Mo fuel material itself has a density of about 17 g/cm^3.) The dispersion fuel has a sufficient uranium density to allow the ^{235}U loading of the core for critical reactor operation at the target 26 days at the 85-MW power level. This LEU core represents an increase in ^{235}U loading by a factor of approximately 1.7 over the current HEU core (its fuel being U_3O_8 mixed with aluminum).

Key top-level assumptions and constraints guide the HFIR studies at ORNL.[2] For example, only changes in the fuel meat region (between the cladding) of the fuel plates are allowed. The performance of HFIR with LEU fuel was analyzed using the standard set of computational tools that are currently used at ORNL to support the operation of the reactor. The computer codes used for these analyses included SCALE 5 [4], BOLD VENTURE [5], and MCNP5 [6] for reactor physics calculations and assessments. The nuclear data libraries used with the neutronics codes were based on ENDF/B-V and -VI nuclear data. Ref. 7 describes the preparation of group cross sections used in the BOLD VENTURE calculations.

RESULTS

BOLD VENTURE simulations were performed with the HFIR critical for 26 days at the 85-MW full-power level; the control absorbers were adjusted in the simulation to be within ~0.2% reactivity of critical. MCNP5 models of the HFIR core were used to analyze the HEU and LEU ("DISP87") cores; the results compared favorably with the beginning of cycle (BOC) BOLD VENTURE k_{eff} determinations. The results of the calculations indicate that the required ^{235}U fuel loading increases from 9.4 kg for HEU to 16.2 kg for the LEU case.

For illustrative purposes, Fig. 3 is a comparison of the uncontrolled k_{eff} curve for the HEU case and the "DISP87" LEU case. These results show the effect of the fissile plutonium generation in the LEU cycle resulting in a reduced slope in the k_{eff} curve with time compared to the HEU core. The excess reactivity at 26 days is similar for the HEU and LEU cases.

The primary performance parameters evaluated include the thermal neutron flux in the central flux trap and the outer beryllium reflector. A comparison of the flux values shows that there is a reduction of the thermal neutron fluxes at the end of cycle (EOC) conditions by ~ 8% in the central flux trap region and by ~12% in the outer beryllium reflector.

CONCLUSION

These results indicate that with the conversion from HEU to LEU using the U-7Mo dispersion fuel and an acceptable cycle duration can be obtained. However, the use of LEU does result in a reduction of the thermal-neutron-flux level in the central flux trap region by approximately 8% and in the outer beryllium reflector region by approximately 12% at a power level of 85 MW. Ongoing work is being performed to improve upon initial LEU core designs to further minimize the impact of the conversion. For example, 2-D fuel grading in the fuel plates is being implemented to flatten the power density distributions to allow HFIR to operate at a higher power level to reduce the degradation in the reflector thermal-neutron-flux level.

ACKNOWLEDGMENT

This research is sponsored by the DOE National Nuclear Security Administration (NNSA) Reduced Enrichment for Research and Test Reactors (RERTR) program and performed at Oak Ridge National Laboratory (ORNL) operated by UT-Battelle, LLC for the U.S. DOE under contract No. DE-AC05-00OR22725.

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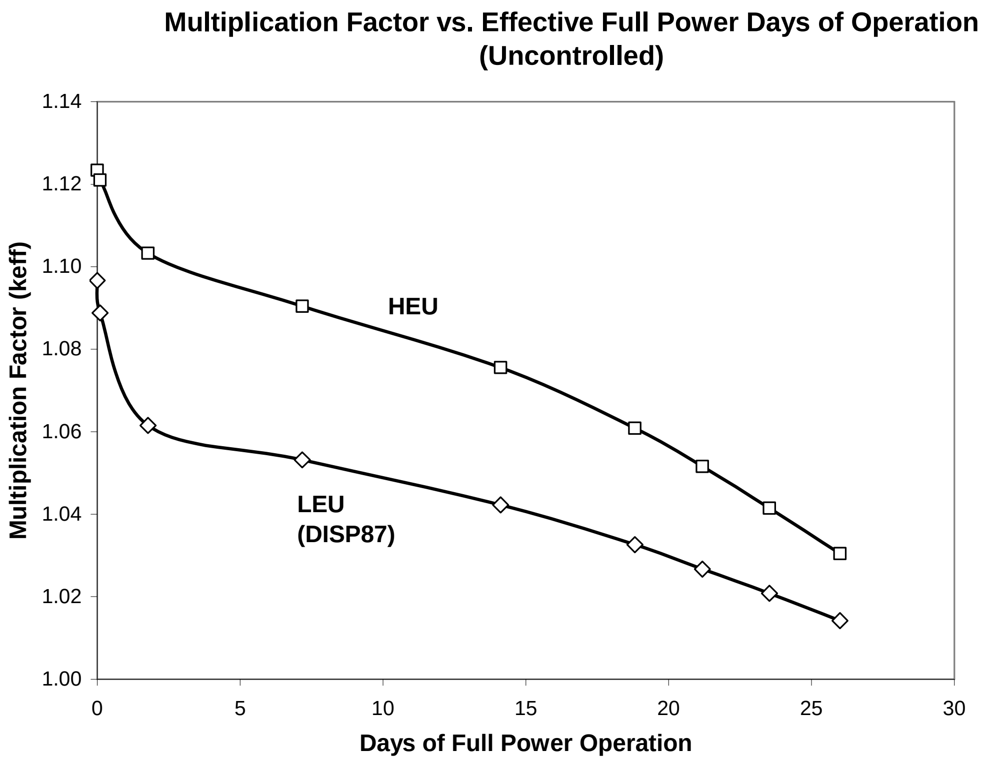

Fig. 3. Comparison of k_{eff} as a function of days of full-power (85-MW) operation in the simulation of the ("DISP87") LEU and HEU HFIR operational history (with no control absorber insertion).

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