Design of an LEU core for the MIT Reactor

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Abstract

A design of the MIT Reactor core using monolithic U-7Mo LEU fuel has been developed with the goal of maintaining thermal and fast neutron fluxes as well as increasing the flexibility for meeting the needs of in-core experiments. An optimum core was sought by varying the core materials, and fuel plate numbers and thicknesses, but maintaining the outside dimensions of a fuel element. A full-core model of the MITR by the Monte-Carlo transport code MCNP was used to calculate the neutron fluxes, reactivity and neutron spectrum available for experiments. The optimum reactor design consisted of the use of half-sized fuel elements made up of nine U-7Mo LEU fuel plates of 0.55 mm thickness with 0.25 mm finned aluminum cladding. This design also utilized solid beryllium fuel elements (dummies) with boron fixed absorbers or solid lead dummies, depending on the in-core experiment flux and spectrum needs. Because the new core design contains twice the amount of ^{235}U as does the existing HEU core, and produces much more Pu, its fuel cycle length is twice as long at the same power level.

Preliminary thermal-hydraulic and neutronic safety evaluations indicate superior performance to the current HEU fuel.

KEYWORDS: Research Reactor, HEU to LEU conversion, MCNP modeling

1. Introduction

A study of the feasibility of using low enriched uranium (LEU) fuel in the MIT Reactor is currently underway. The fuel chosen for the initial evaluation is monolithic U-Mo fuel with a concentration of 7 w/o Mo (U-7Mo), with a uranium density of 16.3 g/cm^3. This fuel material is currently under evaluation by the RERTR program at Argonne National Laboratory [1] and shows promise for use in research reactors with high power densities.

This initial study was made with the constraint that the LEU core design would be made to fit within the existing reactor core structure. The goal, in addition to meeting of nuclear and heat transfer safety parameters, was to deliver neutron fluxes to experimental facilities that were equivalent to the fluxes in the current HEU design, as well as maintaining an equivalent or greater fuel cycle length.

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2. Reactor description

The current MIT Reactor (MITR-II), which began operation in 1975, contains a hexagonal core that uses fuel elements in up to twenty-seven positions, as shown in Fig. 1. Each rhomboid-shaped fuel element contains fifteen aluminum-clad fuel plates using 93% enriched uranium in an aluminide cermet matrix with a fuel thickness of 0.76 mm (0.030 in.) and a length of 610mm (24 inches). Each fuel plate surface has 0.25 mm fins to increase heat transfer to the coolant. The fuel originally used had a fuel density of 3.4 g/cm^3, with a total loading of 445 g ^{235}U in each fuel element. Higher density fuel (3.7 g/cm^3) was used in cores after 1990 with a total fuel loading of 506 g ^{235}U per element.

The core is light water moderated and cooled and is surrounded by a D_2O reflector. Boron-stainless steel control blades are present at the periphery of the core at each of the sides of the hexagon. In addition, fixed absorbers of cadmium were originally installed in the upper twelve inches of the core in a hexagonal configuration between the inner and second fuel rings as well as in three radial arms extending to the edge of the core. These absorbers were removed after the first configuration because of swelling concerns. Fixed absorbers of boron-impregnated stainless steel were later installed in only the radial arms.

Several reentrant thimbles are installed inside the D_2O reflector, allowing a larger neutron flux to be delivered to the beam ports outside the core region. Beyond the D_2O reflector, a secondary reflector of graphite exists in which several horizontal and vertical facilities are present for thermal neutron irradiation facilities. In addition, the MITR Fission Converter Facility is installed outside the D_2O reflector. This facility contains eleven partially spent MITR fuel elements for a delivery of a beam of primarily epithermal neutrons to the medical facility for use in Boron Neutron Capture Therapy. Fig. 2 shows a view of the reactor and reflector regions.

3. Neutronic and Thermal/Hydraulic Models

A model of the MITR-II core using the monte-carlo transport code MCNP was first made by Redmond, *et al.* [2] The model (of HEU core #2), used for comparative purposes for LEU design, was extensively validated against measured values of reactivity and fluxes. [3] The model was then modified for the optimization of LEU fuel and core design.

In addition to the MCNP model, the MCNP-ORIGEN linkage code MCODE, developed by Xu, *et al*. [4], was used for burnup calculations. Because of the limitation on the number of regions that can be tallied in MCNP (99), each of the fuel elements in core was initially modeled so that all plates within an element were assumed to have identical material compositions.

For thermal/hydraulic modeling, the multichannel analysis code MULCH-II, developed and benchmarked by Hu, Bernard, and McGuire [5] was used. This code allowed development of a model of the MITR-II, coupling power distributions with momentum and energy conservation equations, to obtain system design parameters and safety limits.

The coupled point kinetics-hydrodynamic-heat transfer code PARET was used for reactivity transient calculations. This code has been validated by comparison with experimental results from the Spert-III experiments [6]. Although there are some differences between the Spert III C core and the MITR-II, the fuel plate dimensions were similar to those of the MITR-II and should fairly closely represent the thermal-hydraulic and heat transfer conditions of the MITR-II.

Figure 1: Plan view of the MITR-II core. Fuel is typically contained in 22 to 24 of the 27 available fuel element positions

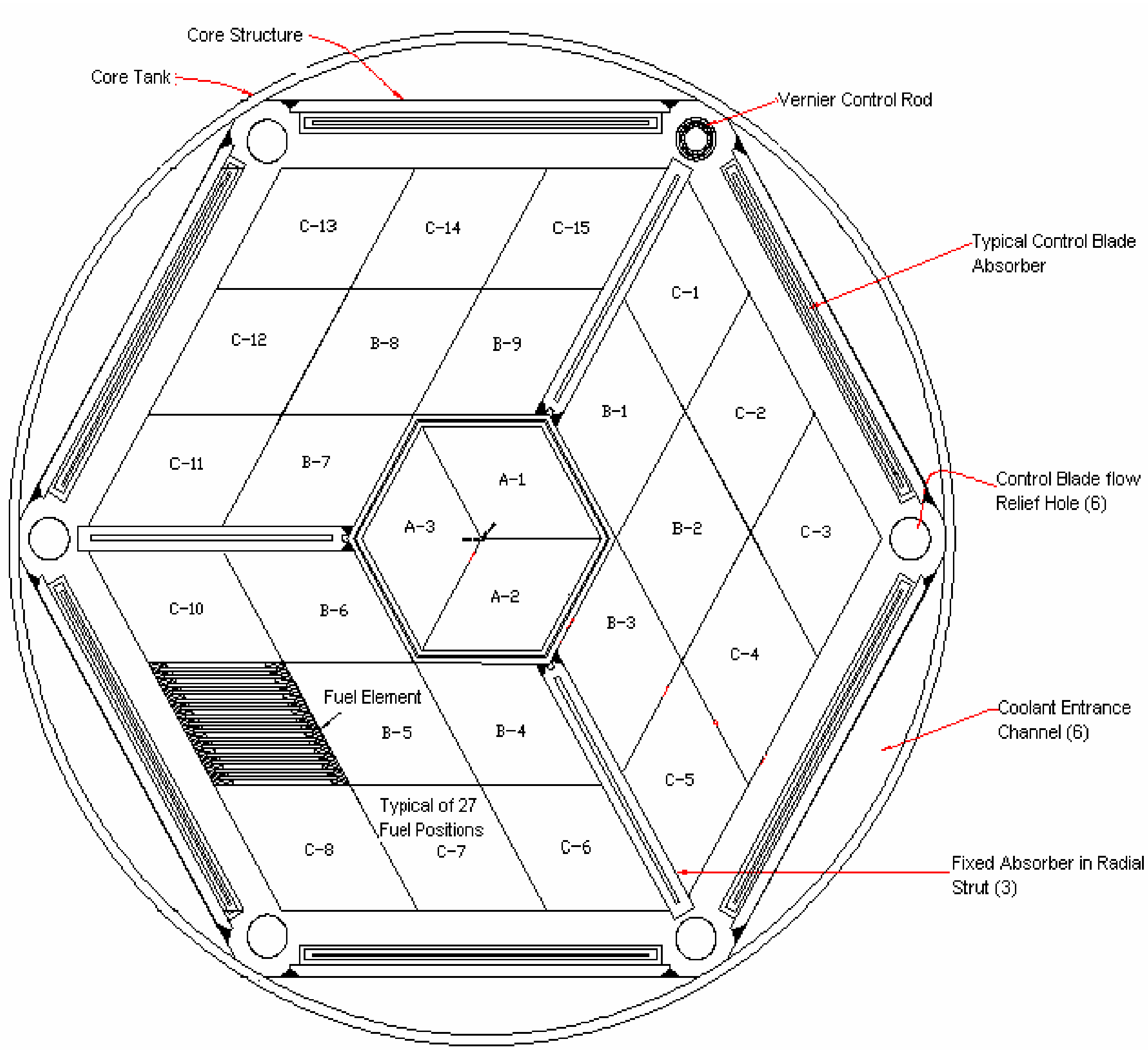
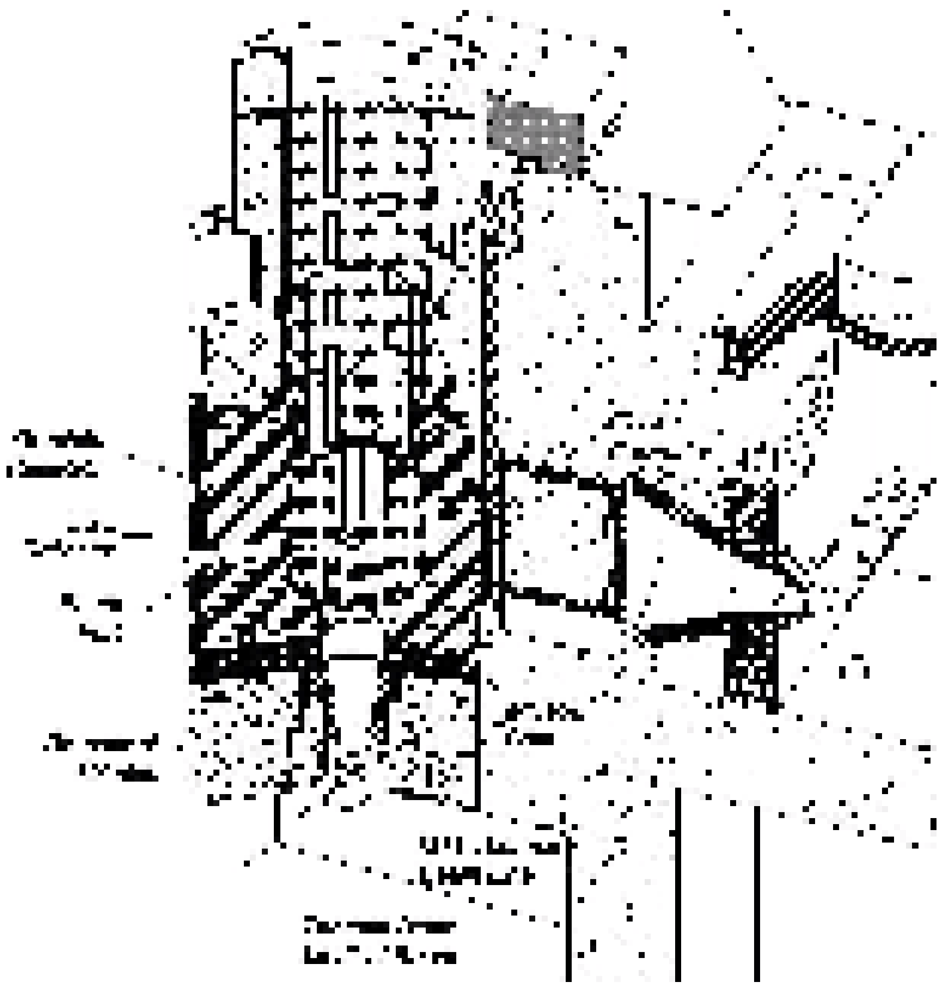


Figure 2: Isometric view of the MIT Reactor



4. Design methods

Fluxes were evaluated at four ex-core experimental facilities and within an in-core sample assembly tube located near the center of the reactor. The ex-core facilities primarily use thermal neutrons for their experimental work (neutron activation analysis, neutron scattering, and silicon doping), whereas materials studies using the in-core facility usually require fast neutrons. All evaluations were made using the same control blade height (21.5 cm), fuel and moderator temperatures, and neutron cross section libraries where possible.

An optimum configuration of fuel plates was achieved by varying the plate number and thicknesses and using the full-core MCNP model of the MITR to determine the effect on flux and reactivity. Since monolithic fuel may allow thinner cladding, a thickness of 0.25 mm (as compared with the reference 0.38 mm cladding) was also evaluated. In addition, the use of different moderator and fuel dummy materials as well as fixed absorbers was evaluated to optimize the neutron fluxes, reactivity and neutron spectrum available for experiments. Changes in the core configuration arrangement and fuel design were also evaluated for neutronic attributes using MCNP as well as for power generation and heat transfer adequacy using MCNP and MULCH-II. A full description of the design options considered is given in [7].

After an LEU design basis core was chosen, MCODE was used to evaluate burnup reactivity. In addition, PARET was used to compare HEU and LEU cores in determining the maximum safe step reactivity insertion.

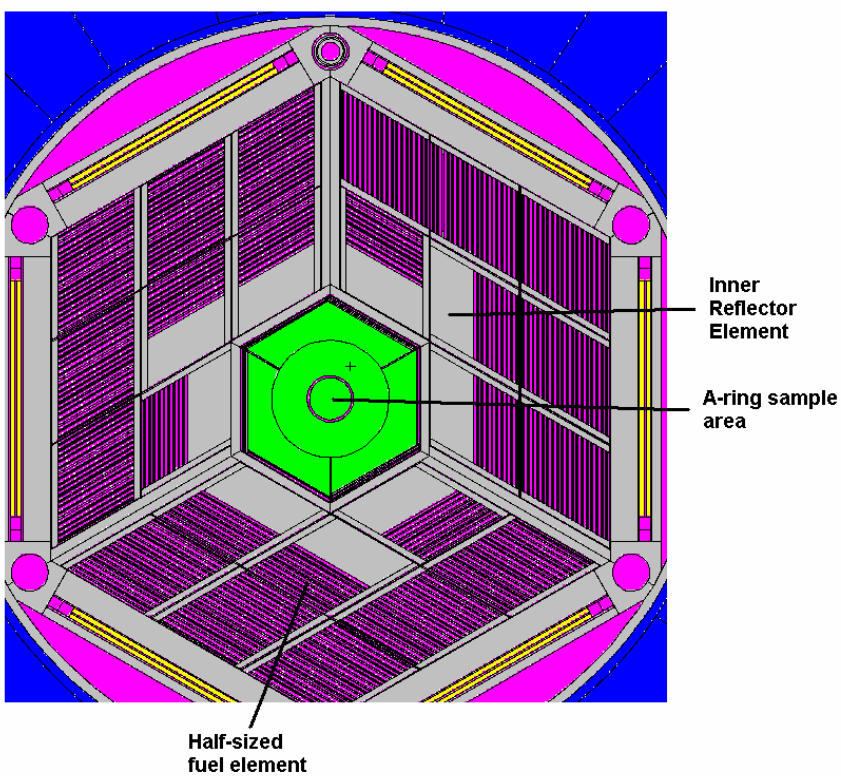
5. Results

5.1 LEU Core

Direct replacement of LEU fuel in the same geometry as the HEU reference core resulted in a 20% loss in in-core fast flux, a 10% loss in ex-core experimental thermal flux, and a reactivity change of -1.2 % $\Delta K/K$. Optimizing the number and thickness of plates increased the ex-core thermal fluxes somewhat, but had little effect on in-core fast flux. The use of thinner (0.25 mm) cladding increased the in-core fast flux at most by about 10% over that of LEU fuel using 0.38 mm cladding. A number of cladding, fixed absorber, and solid dummy materials were evaluated, and although some combinations showed improvement in fluxes delivered to experimental facilities, no combination using the existing core configuration was identified which delivers the equivalent HEU fluxes to experiments.

In order to deliver higher fluxes to experiments, the core was redesigned into the configuration shown in Fig. 3. This design used half-sized fuel elements made up of nine U-7Mo LEU fuel plates of 0.55 mm thickness with 0.25 mm finned aluminum cladding. This design also utilized solid dummy elements of materials chosen to best meet the experiment flux and spectrum needs.

Figure 3: Proposed LEU core design using half-sized fuel elements and solid dummies.

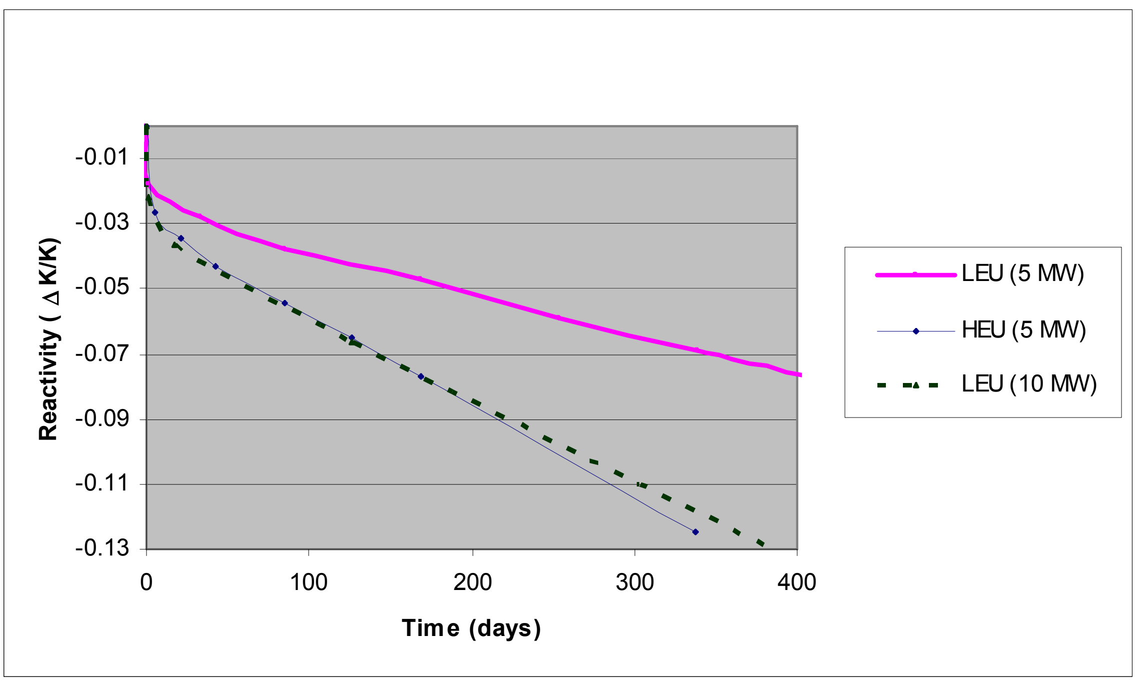


When using solid beryllium dummies and boron-stainless steel fixed absorbers, the ex-core fluxes exceeded those of the HEU reference core by about 5% at a reactor power of 5 MW. Elimination of the fixed absorbers and using solid lead dummies in core caused the LEU fast flux in the in-core experimental assembly to exceed that of the reference HEU case by about 5%. Unfortunately, no design was identified that could simultaneously deliver ex-core thermal flux and in-core fast flux to experimental facilities equivalent to the HEU reference core. Therefore, flexibility was built into the design to deliver higher fluxes to the experimental facilities when needed.

5.2 Burnup

Because the LEU core design contains a larger ^{235}U loading than the reference HEU core, the reactivity loss rate due to burnup calculated using MCODE at the existing 5 MW, is less than that of the HEU core, as shown in Fig. 4. In addition, the reactivity loss rate lessens in the LEU fuel later in life because of the buildup of ^{239}Pu. The two combined effects result in a reactivity loss rate half of that of the HEU core. Also shown is the LEU core burnup rate at twice the existing reactor power (10 MW), which is roughly equivalent to that of the HEU core at 5 MW.

Figure 4: Burnup comparisons of current HEU and proposed LEU cores



5.3 Thermal/Hydraulics

MCNP results of the proposed LEU core showed that power peaking (power produced in the hottest plate compared with the average plate was about 25% higher than that of the reference HEU core. However, because of the larger number of plates (and thus lower power produced per plate), the heat flux in the peak plate is only about 10% higher in the LEU core. In addition, because of the larger coolant channels as a result of the thinner fuel plates, thermal-hydraulic calculations using the MULCH-II code indicate that the core will remain adequately cooled at 5 MW under normal operating and loss-of-flow conditions.

An increase in primary coolant flow is also more easily attained with the LEU fuel design since primary flow rate is normally limited by the pressure on the reactor core tank and larger coolant channels reduce the pressure drop across the core. Because of this and decreased reactivity burnup rate, an upgrade to higher power levels is more easily achieved with the LEU core design.

5.4 Reactivity Calculations

Results from MCNP indicate that the overall and local moderator reactivity coefficients of the proposed LEU core are negative in all cases. In addition, calculations of control blade worths and shutdown margins (reactor 1% subcritical with most reactive blade out) meet all current licensing requirements.

The PARET code was used to determine the maximum safe step reactivity insertion possible in the LEU core without exceeding the clad softening point of 450 ^{o}C. These calculations indicate that, primarily due to Doppler broadening of ^{238}U resonances in the LEU fuel, the allowed step reactivity limit could be as high as \$3.69. This is a 60% increase over the current limit using HEU fuel, and may allow in-core experiments with larger reactivity worths to be installed.

6. Summary

A proposed core using monolithic U-7Mo LEU fuel for the MIT Reactor meets the goal of delivering equivalent neutron fluxes to both ex-core and in-core experiments, although not simultaneously. The core design was made flexible so as to meet anticipated experimental needs as necessary. In addition, refueling intervals will be twice as long as with the HEU core at a given power level.

Thermal/hydraulic calculations indicate that, although power peaking is higher in the LEU core, a larger number of fuel plates and larger coolant channels allow the core to be adequately cooled under current conditions, as well as in the case of loss of flow. An increase in power is also facilitated by a lower pressure drop across the core as well as decreased burnup reactivity loss.

Reactivity calculations indicate that all reactivity coefficients in the LEU core are negative and can withstand a 60% increase in step reactivity over the current HEU core.

Overall, the LEU core design has been shown to meet fundamental safety criteria and will meet current and future anticipated operational and experimental needs.

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