CURRENT STATUS OF THE MISSOURI UNIVERSITY RESEARCH REACTOR HEU TO LEU CONVERSION FEASIBILITY STUDY

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ABSTRACT

The Missouri University Research Reactor (MURR) is a pressurized, reflected, open pool-type, light-water moderated and cooled, heterogeneous system designed for operation at a maximum steady-state power level of 10 Megawatts thermal. It is the highest powered University-owned research reactor in the U.S. that provides a broad range of analytical and irradiation services to the research community and the commercial sector.

Presently, the MURR is one of five U.S. high performance research reactors that use HEU fuel that is actively collaborating with the U.S. Department of Energy to find a suitable LEU fuel replacement. A conversion feasibility study, which involves both the U-7Mo and U-10Mo monolithic LEU fuels, is currently being performed. In order to determine core excess reactivity, MCNP broad scoping studies were conducted in which the water-to-metal ratio was varied by decreasing either the plate cladding or fuel meat thickness while increasing the width of the coolant channel gap. Over the next year studies will include: fuel requirements, including design limitations; experimental facility performance; fuel cycle; and safety analyses, which include the determination of power peaking factors, reactivity transient analysis, loss of flow and coolant accidents, and steady state safety limits.

I. Introduction

The Missouri University Research Reactor (MURR) is a multi-disciplinary research and education facility providing a broad range of analytical and irradiation services to the research community and the commercial sector. The facility is situated on a 7.5-acre (3.0-hectare) lot in the central portion of the University Research Park, an 84-acre (34.0-hectare) tract of land approximately one mile (1.6 km) southwest of the Missouri University main campus. The campus is located in the southern portion of Columbia, the county seat and largest city in Boone County, Missouri.

Scientific programs include research in archaeometry, epidemiology, health physics, human and animal nutrition, nuclear medicine, radiation effects, radioisotope studies, radiotherapy, and nuclear engineering; and research techniques including neutron activation analysis, neutron scattering, and neutron interferometry. MURR staff generate and nurture extensive collaborations with outside researchers. Research groups are made up of both MURR staff and researchers from the University of Missouri System's four main campuses. The breadth and quality of the research programs and the available facilities and equipment are comparable to those found in the U.S. National Laboratories.

The MURR has six types of experimental facilities designed to support these services and research programs: the Center Test Hole (Flux Trap); the Pneumatic Tube System; the Graphite Reflector Region; the Bulk Pool Area; the (six) Beamports; and the Thermal Column. The first four types provide areas for the placement of sample holders or carriers in different regions of

the reactor core assembly for the purposes of material irradiation. Some of the material irradiation services include transmutation doping of silicon, isotope production for the development of radiopharmaceuticals and other life-science research, and neutron activation analysis. The six beamports channel neutron radiation from the reactor core to experimental equipment which is used primarily to determine the structure of solids and liquids through neutron scattering. The graphite thermal column is designed for the purpose of performing neutron radiographs and large sample irradiations.

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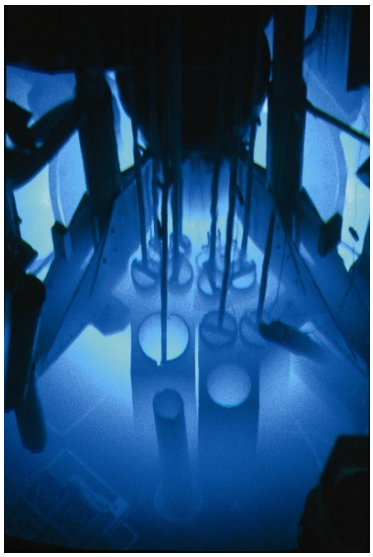


MURR Core

The MURR also participates in a U.S. Department of Energy (DOE) program to provide the availability of university reactor facilities to non-reactor-owning colleges and universities. The MURR also provides support to institutions with reactors, but which operate at power levels too low to adequately perform required experiments. Reactor sharing projects include work in fields such as anthropology, archaeology, animal science, crystallography, analytical epidemiology-nutrition, geology, materials science, physics, nuclear analysis development, and biochemistry.

II. Facility History

The MURR first achieved criticality on October 13, 1966. The reactor was originally designed for 10-MW operation, but was initially licensed to operate at only 5 MW until reactor utilization and operating experience were sufficient to justify full power operation. In 1974, additional cooling equipment was added and the process instrumentation and safety systems were modified as required to facilitate operation of the reactor at the full design power of 10 MW.

In 1986, the University of Missouri requested that a determination be made by the U.S. Nuclear Regulatory Commission (NRC) that the MURR has a *Unique Purpose*, as defined by 10 CFR 50.2, and is therefore exempt from the conversion from HEU to LEU fuel [3]. The MURR has a compact core design (33 liters) that can not perform its intended function with any currently-qualified LEU uranium densities. A BOLD VENTURE 3D model, benchmarked against the only MURR destructively analyzed fuel element, was used to demonstrate that a new silicide

LEU core loaded to density of 7.2 g U/cm^3 and with no fission product inventory would result in a K_{eff} of less than 1.0 [3].

Since achieving initial criticality, the reactor has operated safely for over 272,000 full power hours in support of the primary mission of providing the maximum flux, or current, of neutrons to the maximum number of users without endangering the health and safety of facility workers and the general public. In August 2006, the MURR submitted a re-licensing application to the U.S. Nuclear Regulatory Commission for a twenty-year renewal to the facility's operating license, R-103.

III. Basic Reactor Description

The MURR is a pressurized, reflected, heterogeneous, open pool-type, which is light-water moderated and cooled. The reactor is designed and licensed to operate at a maximum thermal power level of 10 MW with forced cooling, or up to 50 kW in the natural convection mode.

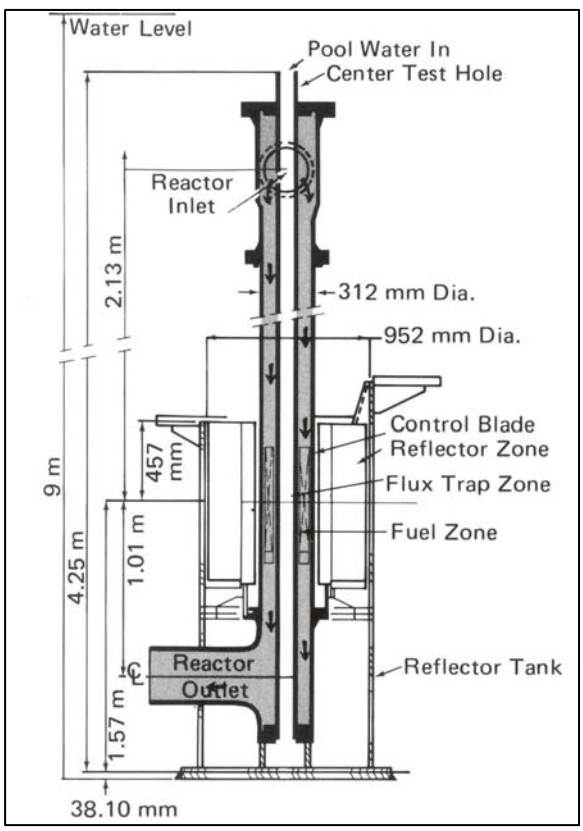


Figure 1
Reactor Core Assembly – 2D View

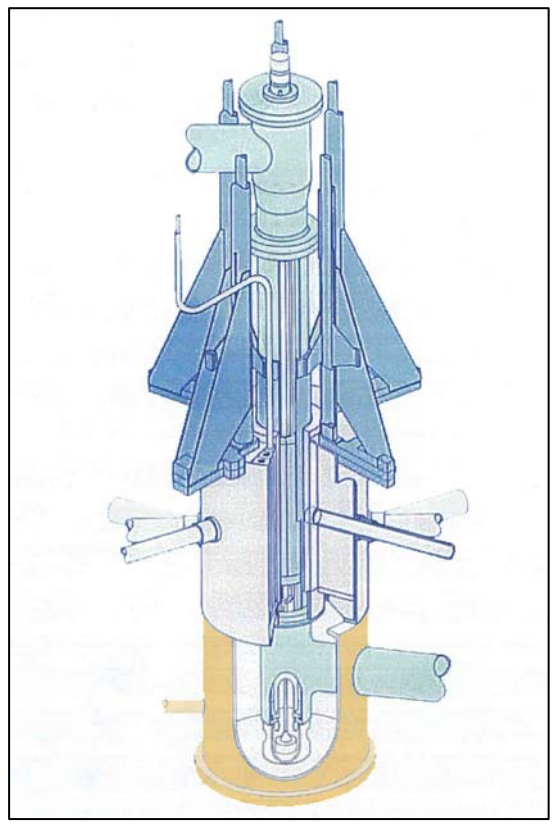
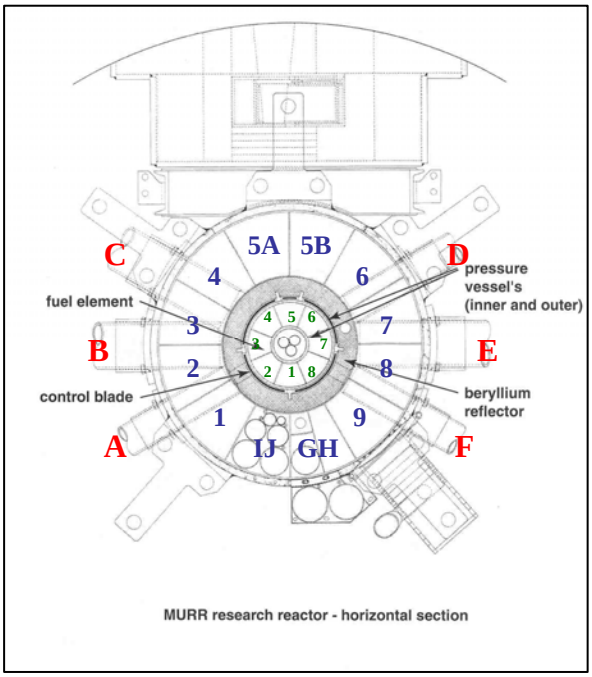


Figure 2
Reactor Core Assembly – 3D View

The reactor core assembly is located eccentrically within a cylindrically-shaped, aluminum-lined pool, approximately 10 feet (3.0 m) in diameter and 30 feet (9.1 m) deep. The reactor core consists of three major regions: fuel, control blade, and reflector. Two and three-dimensional views of the reactor core assembly are shown in Figures 1 and 2, respectively. A plan view is shown in Figure 3.

The fuel region has a fixed geometry consisting of eight (8) fuel elements having identical physical dimensions placed vertically around an annulus in between two cylindrical aluminum reactor pressure vessels. Each fuel assembly is comprised of 24 circumferential plates containing uranium enriched to approximately 93% in the isotope ^{235}U as the fuel material. The control blade region is an annular gap between the outer pressure vessel and the inner reflector annulus, so that no penetration of the pressure vessels is required. Five (5) control blades - four (4) boral and one (1) stainless steel - operate vertically within this gap, controlling reactor power by varying neutron reflection. The reflector region consists of two concentric right circular annuluses surrounding the control blade region. The inner reflector annulus is a 2.71-inch (6.9 cm) thick solid sleeve of beryllium metal. The outer reflector annulus consists of vertical elements of graphite canned in aluminum, having a total thickness of 8.89 inches (22.6 cm).



Graphite Reflector Elements (12):
1
2
3
4
5A
5B
6
7
8
9
GH
IJ

Fuel Elements (8):
1
2
3
4
5
6
7
8

Beamports (6):
A
B
C
D
E
F

Figure 3
Reactor Core Assembly – Plan View

IV. Current Fuel System and Design

The fuel material at the time of initial startup was a uranium-aluminum alloy with each fuel assembly loaded to a maximum of 650 grams of ^{235}U. This type of fuel system had performed very reliably in the Materials Test Reactor (MTR) and the Engineering Test Reactor (ETR) at the Idaho National Engineering Laboratory (INEL), as well as in other reactors throughout the world. However, in order to reduce the fuel cycle cost and the amount of ^{235}U needed per MWD of energy produced at the MURR, a conversion was performed in 1971 to switch to a uranium-aluminide dispersion UAl_x fuel material with a maximum loading of 775 grams of ^{235}U per assembly. The MURR 775-gram fuel element is a product of the UAl_x dispersion fuel system development. The UAl_x dispersion fuel system was developed at INEL for the high flux, high power Advanced Test Reactor (ATR) and subsequently used at the MTR and ETR prior to its use at the MURR [1] [2]. A drawing of the MURR fuel element is shown in Figure 4.

The fuel elements have a radial dimension of 3.21 inches (8.15 cm) - as determined from radial length from the inside of the inner roller to the outside of the outer roller - and an overall length of 32.5 inches (82.55 cm). Each element is longitudinally-symmetrical with 24 fuel bearing plates. The fuel plates are segments of concentric circles 0.050 inches (1.27 mm) thick separated by a gap of 0.080 inches (2.03 mm). These gaps provide the coolant channels necessary for the removal of the heat generated during the fission process, and for neutron moderation. Additional coolant gaps exist between the innermost fuel plate (No. 1) and the island tube wall (inner pressure vessel), and the outermost fuel plate (No. 24) and the outer pressure vessel wall. These gaps are 0.095 inches (2.41 mm) and 0.075 inches (1.91 mm), respectively. The nominal radius to the innermost plate is 2.77 inches (7.04 cm), and 5.76 inches (14.63 cm) to the outermost plate.

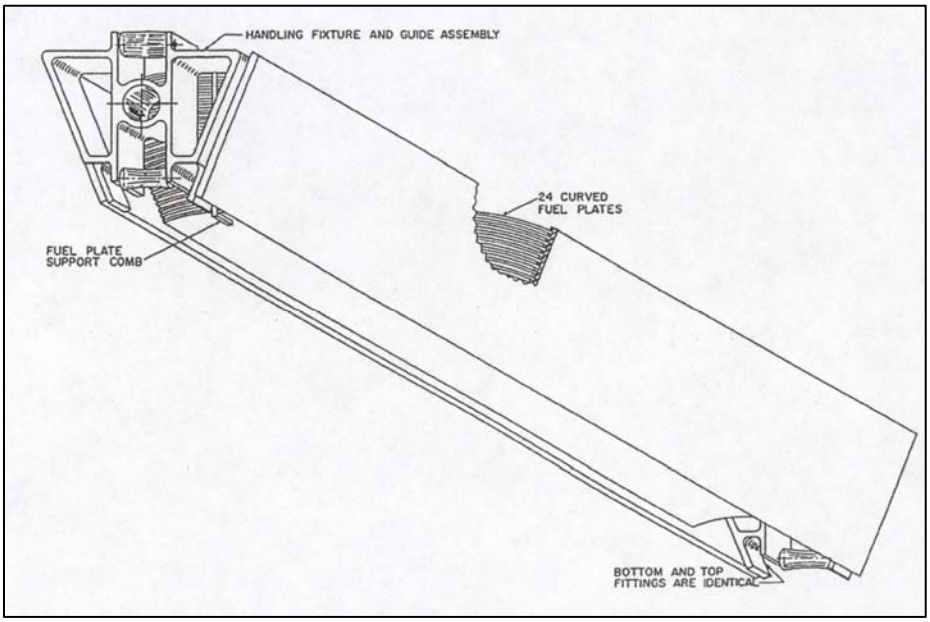


Figure 4
MURR Fuel Element – Pictorial View

The fuel meat in each plate is 0.020 inches (0.508 mm) thick. The fuel plates are supported along their vertical edge by slotted aluminum side plates. The fuel plates are permanently fastened into the side plates using a mechanical binding procedure that provides a tensile strength of greater than 150 pounds per linear inch of the side plate joint. This ensures a rigid assembly fully capable of withstanding the hydraulic forces imposed by the primary coolant design velocity of 23 ft/sec (7.01 m/sec). In fact, fuel assemblies of similar construction have withstood severe hydraulic tests at flow velocities up to 50 feet/sec (15.24 m/sec) without distortion, thus indicating an adequate design margin in this regard. The side plates are 31.75 inches (80.65 cm) long by 3.16 inches (8.03 cm) wide and 0.15 inches (3.81 mm) thick.

Fuel element specifications are summarized in Table 1. Table 2 provides the current MURR fuel operating characteristics, including maximum fuel burnup, core MWDs with the control blades full out, core refuelings, and fuel cycle.

Table 1
Summary of MURR Fuel Element Specifications

Description	Nominal Value
Fuel Material	
Aluminide-UAl_x (mostly UAL_3 phase)	
Enrichment	93% ^{235}U
Thickness	0.020 inches (0.508 mm)
Cladding	
Material	Aluminum
Thickness	0.015 inches (0.381 mm)
Fuel assembly	
Number of Fuel Plates	24
Innermost Fuel Plate Center Radius	2.795 inches (7.099 cm)
Innermost Fuel Plate Center Radius	5.785 inches (14.694 cm)
Overall Fuel assembly Length	32.5 inches (82.550 cm)
Overall Fuel Plate Length	25.5 inches (64.770 cm)
Overall Active Fuel Length	24.0 inches (60.960 cm)
Fuel Plate Thickness	0.050 inches (1.270 mm)
Distance Between Fuel Plates	0.080 inches (2.032 mm)
Maximum ^{235}U Loading	775 grams
Fuel Density	1.5 to 1.6 grams/cm^3
Weight	~ 6 Kg

Table 2
Current MURR Fuel Operating Characteristics

Maximum Burn-Up:	150 MWD/element (1200 MWD/core) due to insufficient excess reactivity which achieves less than 1.8E+21 fissions/cm^3; Technical Specification limit is 2.3E+21 fissions/cm^3
Control Rods Full Out:	~670 MWD core with equilibrium xenon activity (56% of 1200 MWD)
Refuelings:	Weekly - replace all eight fuel elements; fuel elements are used in 18 to 20 core loadings to achieve 145 to 150 MWD burnup (24% burnup)
Fuel Cycle:	24 elements used per year; 32 fuel elements in active fuel cycle; 15 to 36 fuel elements in storage

V. HEU to LEU Fuel Conversion Feasibility Study – Current Status

The initial work of MURR's fuel conversion feasibility study included defining the fuel requirements, describing the HEU core, and defining experimental facility performance indicators. Some of the potential concerns in performing a conversion include: matching the performance capabilities of the current 775-gram ^{235}U fuel element; not increasing the fuel storage requirements; having sufficient excess reactivity in order to decrease the loading in fuel plates with high peaking factors; and maintaining or enhancing neutron flux in the flux trap, graphite reflector, and beamport regions. The flux needs of the experimental facilities are addressed in Section VI, while fuel performance and design limitations are covered in this section.

The first technical task that was required was a comparison of core excess reactivity between MURR's current UAl_x dispersion HEU fuel and the U-10Mo and U-7Mo LEU fuels. Using the MCNP Monte Carlo Code, broad scoping studies were performed. The first comparison consisted of replacing the 0.020-inch (0.508 mm) thick UAl_x HEU fuel meat with the two different U-Mo monolithic LEU fuels. No other physical changes were made to the core - the same fuel plate and coolant channel dimensions were maintained. Although this resulted in lower K_{eff}'s even with the ^{235}U loading more than doubling, the K_{eff}'s were still high enough to ensure sufficient excess reactivity with the core operating at 10 MW. However, the LEU cores would require a greater number of fuel elements in the fuel cycle, thus exceeding our fuel storage capacity. The results from the U-10Mo studies are listed first in Table 3, since U-10Mo has exhibited the least amount of swelling as a function of fuel burnup during irradiation testing, making it the preferred choice from an operational perspective. Table 4 provides the results from the U-7Mo studies.

The next step was to increase the water-to-metal ratio by decreasing either the plate cladding or fuel meat thickness while increasing the width of the coolant channel gap. First, the standard 0.015-inch (0.381-mm) thick cladding was decreased to 0.012 inches (0.305 mm), while the coolant channel gap was increased from 0.080 to 0.086 inches (2.032 to 2.184 mm). Next, a 0.010-inch (0.254-mm) cladding thickness was modeled with a 0.090-inch (2.286-mm) coolant channel gap. For the U-10Mo core, with a lower ^{235}U density than U-7Mo, the MCNP results indicate sufficient excess reactivity that could achieve even more MWD's per fuel element than the current design; assuming that fuel element performance can handle the higher burnup values. Finally, using the current 0.015-inch (0.381 mm) thick cladding, the fuel meat thickness was decreased to 0.018 inches (0.457 mm) and then to 0.016 inches (0.406 mm), while increasing the coolant channel gap to 0.0821 and 0.0842 inches (2.085 and 2.139 mm), respectively. The MCNP analyses indicate that equivalent or slightly higher K_{eff} values are obtained using less ^{235}U than with the standard 0.015-inch (0.381-mm) thick clad LEU core. The cores modeled do raise core and fuel plate design performance questions that would have to be addressed before they could be used. First is the question about sufficient cladding and plate thickness.

The current fuel design of 0.050-inch (1.270 mm) thick plates with 0.015-inch (0.381 mm) thick cladding is based on the initial tests for the ATR performed in the 1960's and early 1970's for improving the original MTR alloy fuel design to the current UAl_x dispersion fuels. The cladding is nominally 0.015 inches (0.381 mm) thick to ensure that the required minimal cladding thickness of greater than 0.010 inches (0.254 mm) is achieved. The possible fabrication techniques for the new U-Mo monolithic fuel plates should require less rolling. Additionally, the monolithic "foil fuel meat" should also reduce the chance of fuel particles being pressed up into the cladding. These combine to potentially allow the nominal cladding thickness to be closer to the required minimal thickness, which could permit a thinner nominal cladding thickness design. However, the fabrication methodologies must be further developed and tested before the thinnest allowable nominal cladding design thickness will be known. If either the fuel cladding or the fuel meat thickness is reduced, a thinner overall fuel plate will result. This raises the question on how thick a fuel plate must be in order to ensure sufficient stability and rigidness with design forced coolant flow. Because their contribution to stiffness can differ, plate stability may also vary depending on the relative thickness of the fuel meat and cladding. These values need to be determined to complete the optimum core design.

Table 3
UAl_x and U-10Mo Fuel Comparison

Fuel Type	No. of Plates	Fuel Meat (inches)	Total (inches)	Channel (inches)	^{235}U (gm/cm^3)	^{235}U Core Total (Kg)	K_{eff} [1]	K_{eff} [2]
UAl_x	24	0.020	0.050	0.0800	1.443	6.20	1.086	1.043
U-10Mo	24	0.020	0.050	0.0800	3.026	13.0	1.057	1.014
U-10Mo	24	0.020	0.044	0.0860	3.026	13.0	1.076	1.033
U-10Mo	24	0.020	0.040	0.0900	3.026	13.0	1.086	1.043
U-10Mo	24	0.018	0.048	0.0821	3.026	11.7	1.059	1.016
U-10Mo	24	0.016	0.046	0.0842	3.026	10.4	1.059	1.016

Table 4
UAl_x and U-7Mo Fuel Comparison

Fuel Type	No. of Plates	Fuel Meat (inches)	Total (inches)	Channel (inches)	^{235}U (gm/cm^3)	^{235}U Core Total (Kg)	K_{eff} [1]	K_{eff} [2]
UAl_x	24	0.020	0.050	0.0800	1.443	6.20	1.086	1.043
U-7Mo	24	0.020	0.050	0.0800	3.237	13.9	1.066	1.023
U-7Mo	24	0.020	0.044	0.0860	3.237	13.9	1.084	1.040
U-7Mo	24	0.020	0.040	0.0900	3.237	13.9	1.096	1.053
U-7Mo	24	0.018	0.048	0.0821	3.237	12.5	1.068	1.025
U-7Mo	24	0.016	0.046	0.0842	3.237	11.1	1.066	1.023

[1]Cold clean critical with 0 MWD.
[2]Equilibrium xenon = -0.035Δk/k, equilibrium samarium = -0.0093Δk/k.

Other fuel design limitation issues also need to be determined. A high minimum cladding blister temperature is required to ensure that the safety margins in the reactor safety analyses are not reduced. A peak fuel burnup limit equal to or greater than 2.3E+21 fissions/cm^3 is needed to meet current fuel performance limits. A limit of greater than 4.0E+21 fissions/cm^3 could provide some with the possibly of achieving significantly higher MWD's per fuel element.

VI. HEU to LEU Conversion Experimental Performance Indicators

Based on the current and projected MURR utilization, the following three experimental locations were selected for comparing the effect of an HEU to LEU fuel conversion.

1. Center Test Hole (Flux Trap) Irradiation Position:
   a. One of MURR's primary missions is to produce high specific activity isotopes for various applications including medical use. While the majority of isotopes are produced through thermal neutron reactions, a few require fast neutron interactions.
   b. The current unperturbed peak thermal flux in the flux trap region is 6.0E+14 n/cm^2-s, while the peak fast flux (> 1.0 MeV) is 6.0 E+13 n/cm^2-s.
   c. The goal is to achieve at least the same peak thermal and fast flux levels after the LEU conversion.

2. Graphite Reflector Region Irradiation Positions:
   The graphite reflector region has a number of sample positions that are used to irradiate various sample materials for research and commercial activities. For the purposes of benchmarking the conversion, the following three locations were selected:
   a. Graphite reflector element No. 3 - The row-1 pneumatic tube irradiation position is located in this graphite element. The average measured value of thermal flux in this location is 9.0E+13 n/cm^2-s, while the average epithermal flux (> Cadmium cutoff value) is approximately 3.0E+12 n/cm^2-s.
   b. Graphite reflector element No. GH - The Y-1 irradiation position is located in this graphite element. The average measured value of the thermal flux in this location will be used in the comparison.
   c. Graphite reflector elements No. 5A and 5B - Due to the amount of thermal neutron absorption material within these two elements, this location provides a good fast neutron irradiation position for benchmarking. The average computed fast flux (> 0.5 MeV) value for this location is 3.5E+12 n/cm^2-s.

3. Beamports:
   There are four (4) radial and two (2) radial-tangential beamports for extracting neutron beams for research at the MURR. Several future uses of the reactor propose to use these beamports for material studies as well as medical research. For comparison purposes, Beamport 'B,' which is a 6-inch (15.2-cm) diameter radial beam tube, was selected. The source end of the beam tube terminates at the surface of the beryllium reflector, with the center of the beam tube 7 inches (17.8 cm) below core centerline. The measured thermal and epithermal beam emerging from the beam tube are:
   a. Thermal flux at the end of an open beam tube is approximately 9.0E+9 n/cm^2-s, while the epithermal flux (> Cadmium cutoff value) is 2.7E+8 n/cm^2-s.
   b. With a 4-inch (10.2-cm) sapphire filter installed at the end of the beam, the above values change to 6.8E+9 and 2.3E+07 n/cm^2-s, respectively.

VII. Summary and Future Work

The completed broad scoping studies using MCNP indicate that U-10Mo and U-7Mo monolithic cores have the potential to provide an LEU fuel that could work in MURR.

Over the next year, additional feasibility studies will be performed. These include fuel depletion studies based on the optimal LEU core design identified in the MCNP broad scoping studies. Then safety analyses will be completed using worse case power peaking factors associated with mixed core loadings of fresh and end of life fuel elements. There are cladding and fuel plate thickness fabrication design limits that need to be determined to be able to finalize a fuel design.

VIII. References

[1] Gibson, G.W., Graber, M.J., and Francis, W.C., Annual Progress Report on Fuel Element Development for FY-1963, IDO-16934, 1963.

[2] Graber, M.J., et al., Performance Evaluation of Core II and III Advanced Test Reactor Fuel Elements, ANCR-1027, Aerojet Nuclear Company, 1971.

[3] Letter Request to USNRC, Application for Unique Purpose Exemption from Conversion from HEU Fuel, Facility License No. R-103, University of Missouri Research Reactor, September 1986; and supplemented by Letter Response to USNRC Request for Additional Information Supporting the Unique Purpose Exemption Request, February 1987.