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Performance and Fabrication Status of TREAT LEU Conversion Conceptual Design Concepts.

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ABSTRACT

Resumption of transient testing at the TREAT facility was approved in February 2014 to meet U.S. Department of Energy objectives. The National Nuclear Security Administration's Global Threat Reduction Initiative Convert Program has established a program to convert TREAT from its existing highly enriched uranium core to a new low enriched uranium (LEU) core using the existing core to test and qualify the LEU fuel. The screening decisions for the initial pre-conceptual designs are briefly described, followed by more-detailed discussions on current feasibility, qualification and fabrication approaches. Conversion feasibility considers LEU fuel element assembly designs that can meet TREAT design, performance, and safety requirements. Engineering challenges such as cladding oxidation, high temperature material properties, fuel block fabrication, along with physics challenges of neutronics performance, will be highlighted. Conceptual design evaluations have provided confidence that an acceptable design can be achieved.

1. Introduction

In 2010 the Department of Energy (DOE) proposed to re-establish the capability to conduct transient testing of nuclear fuels [1] at the Transient Reactor Test Facility (TREAT) reactor at Idaho National Laboratory (INL) and is currently preparing to do so. The TREAT reactor first achieved criticality in 1959 and successfully performed thousands of transient tests on nuclear

fuels until 1994, when its operations were suspended. Resumption of operations at the TREAT facility was approved in February 2014 for meeting the U.S. Department of Energy Office of Nuclear Energy's objectives in transient testing of nuclear fuels. The National Nuclear Security Administration's Global Threat Reduction Initiative (GTRI) Convert Program is evaluating the efforts that would be needed to convert TREAT from its existing highly enriched uranium (HEU) core to a new core containing LEU (i.e., with ^{235}U content less than 20% by weight) [2].

The existing TREAT fuel elements were designed in two primary campaigns (hereafter referred to as the "original" and "upgrade" designs). The original and upgrade cores were designed to support two distinct mission types within TREAT. They were both designed effectively as "lifetime" cores because TREAT fuel elements undergo irradiation at low power level during steady-state operations and at high power level for transient experiments for short durations of time. Therefore, burn-up is very low.

The original TREAT core was used for more than 6,000 reactor startups, of which nearly 3,000 were transient irradiations, having achieved approximately 0.7% burnup. These fuel elements remain either in the TREAT core or in storage locations today. When TREAT operations were suspended, the deterioration of these elements was slight and not even approaching end-of-life. Determining their present condition as a baseline for resumption of TREAT operations is underway.

Unlike many research reactors, the TREAT core is cooled by air. The reactor is capable of up to 120-kW steady-state power due to the heat removal capacity of the forced-air cooling system. Steady-state operations are sufficient to perform neutron radiography in the adjacent facility and for other system check outs. The "dry" nature of the core greatly simplifies the configuration of ex-core facilities that must penetrate into the core such as the fast-neutron hodoscope, neutron radiography facility, and experiment hardware of various configurations.

TREAT is capable of safely generating high power transients due to the design of the fuel elements. The large mass of graphite present in the fuel composite acts as a heat sink for the transient energy and also acts as the primary neutron moderating medium. The transient energy generated in the urania particles transfers quickly to the fuel blocks and elevates their temperature rapidly. The temperature feedback effect safely terminates the transients due to increased neutron thermal up-scattering and Doppler broadening, resulting in decreased fission cross sections, and increased neutron leakage. This characteristic of TREAT is essential to its safe operation.

During a transient, TREAT is capable of depositing high fission energies in test specimens without damage to the driver fuel. However, there is enough reactivity in the core to generate transients resulting in fuel temperatures high enough to damage the cladding of the fuel elements.

TREAT's automatic reactor control system provides the capability of functioning in various operational modes, including steady-state operation, simple, "single-pulse" transients, and a wide range of "shaped transients." The latter are typically used where specific power-time histories are needed to create test-fuel temperature-time simulations of off-normal or severe-accident simulations. Various mode schematics are depicted in Figure 1.

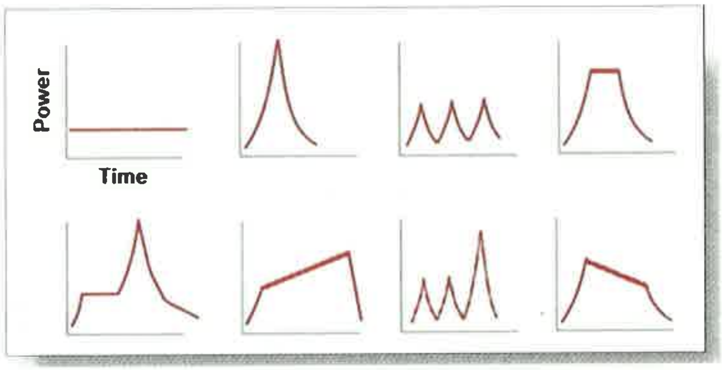

Figure 1. Example Operational Modes.

The TREAT upgrade core was designed to accommodate a larger test section in an 8-in. diameter advanced test loop. The upgrade core was designed to retain the original TREAT fuel elements in the core's periphery with the new upgrade elements in the center 11 × 11 ring of the reactor grid. Unlike the original elements, the standard upgrade elements contained a longer 5-ft fuel section, composed of a 4 × 4 array of 1-in.-square HEU urania-in-graphite fuel blocks, with graphite reflectors on top and bottom, all canned in Inconel 625. The TREAT upgrade fuel elements were successfully fabricated in the 1980s; however, the program for which the upgrade fuel was required was suspended before the fuel was loaded into the core. These elements currently remain in storage at the TREAT facility. Although resumption of TREAT operations is currently focused on use of the original core, the long-term need for the upgrade core to support higher power testing of large test pieces is being evaluated.

2. LEU TREAT Fuel Functional Requirements

The TREAT Conversion Project's objectives are to perform the design work necessary to generate an LEU replacement fuel element assembly, restore the capability to fabricate the assemblies, fabricate replacement LEU fuel element assemblies, and implement the physical and operational changes required to convert the TREAT facility to use LEU fuel element assemblies. In support of these objectives, computational modeling and analysis efforts are being performed.

The fundamental functions of the TREAT LEU design are listed below. These provided the basis for the feasibility assessment that was performed:

- Performance capabilities corresponding to TREAT's mission needs:
  - Core neutronic configuration as capable of depositing the fission energies versus time response in the test specimen(s) as is the current (original-design) HEU core
  - Rapid and predictable negative temperature feedback for transient operation
  - Predictable fuel can oxidation rate, which allows for adequate transient energy deposition and fuel element service life
  - Proper heat conduction and adequate capacity for transient-generated energy deposited in the driver fuel (thermo-mechanical properties of fuel meat)
  - Steady-state operation capability sufficient for system check-out, calibrations, and neutron radiography.
- Dimensional stability of fuel element assemblies (retain steady-state coolability and predictable nuclear configuration):
  - Fuel meat resistance to fracture driven by thermal shock, fission damage, and gas evolution

- Hermeticity of fuel encapsulation (including any welds and penetrations) to prevent ingress of atmosphere and bulging or pillowing
- Adequate retention of gases evolved during irradiation and high-temperature service to prevent bulging or pillowing
- Manageable thermal creep, elongation, swelling, and other environmentally driven structural behaviors.

- Chemical stability of fuel element assemblies:
  - Hermeticity of fuel encapsulation (including any welds and penetrations) to prevent oxidation of fuel meat, reflectors, and any other reactive material contained therein
  - Inertness or benign reactions between dissimilar materials (can to fuel block contact, fuel particle to graphite matrix, can to end caps).
- Fission gas retention:
  - Hermeticity of fuel encapsulation (including any welds and penetrations) to prevent fission gas release.
- Operability:
  - Compatibility with existing reactor infrastructure and methods for handling, fixturing, and storage with low risk for mechanical damage during such evolutions.

3. LEU Fuel Design Approach and Pre-conceptual Screening Decisions

Pre-conceptual low-enriched uranium (LEU) fuel system concepts and designs for the TREAT facility were successfully established and down selected and have formed the basis of development work during the current conceptual design phase. Functional and operational requirements provided inputs on design categories and options, while scoping performance testing provided justification for the screening status of several concepts. Some key aspects considered were LEU materials, fuel can materials, fuel system design and configuration alternatives, cladding oxidation management, and fuel block fabrication feasibility. Down selection decisions were based on evaluations and analytic modeling performed by a team of subject matter experts from Argonne National Laboratory, Idaho National Laboratory, and Los Alamos National Laboratory. The general design approach was to "keep it simple" to facilitate qualification and fabrication, which will contribute toward maintaining the aggressive schedule of this conversion program.

Because concept selection often has a tremendous influence on the eventual success of the design, a trade study was completed to serve as the basis for robust and defensible concept down-selection decisions followed by more-detailed evaluations of the candidates that appeared to represent the most balanced technical solutions.**Error! Reference source not found.** The design options, categorical breakdown, and screening status of each component and configuration are summarized in Figure 2.

Several design variants were evaluated. The most-promising variant will require an engineering solution that maintains a gap between the fuel meat and the cladding. Design optimization and further engineering development will occur in the conceptual design phase. As with the original fuel design, this design variant does not make provisions for any insulation between fuel meat and cladding, which tends to keep fabrication simple. However, use of an insulating material between the fuel meat and fuel element assembly cladding is a design option that will be considered as an alternative during the conceptual design phase.

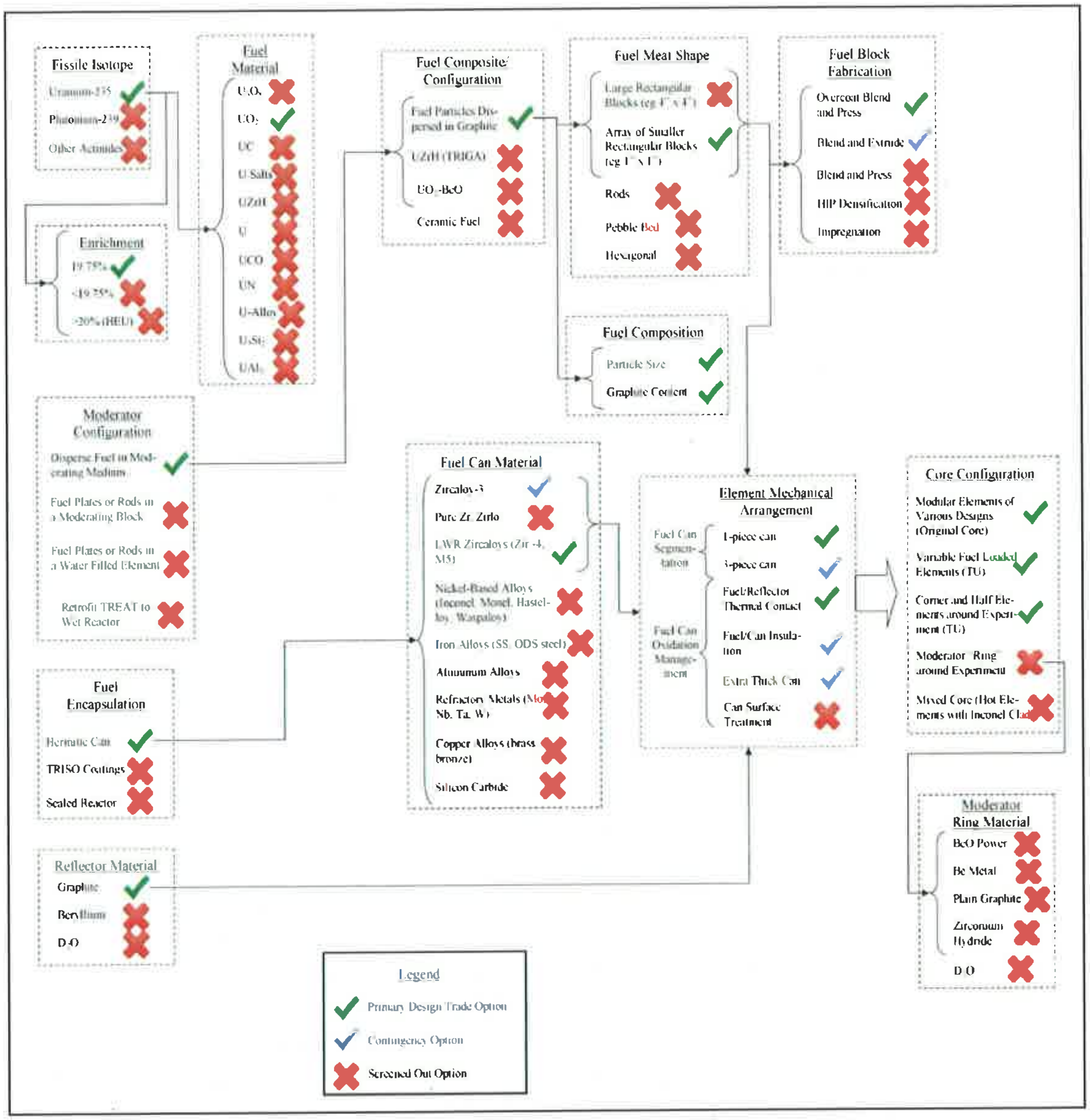

Figure 2. Concept Screening Flow Sheet.

UO_2, U_3O_8 and UC were considered for the fuel fissile particles. U_3O_8 was used to make the original core because it appeared to become more uniformly dispersed than fuel made with UO_2, and it was understood at the time that the U_3O_8 would reduce during heat treatment to UO_2. Microscopic evaluation of samples showed no signs of reaction between oxide particles and the matrix, leading to voids or cracks. Energy-dispersive spectroscopy, however, now confirms that U_3O_8 particles are less stable in the graphite matrix than the UO_2 particles and tend to reduce. With modern processing techniques, adequate dispersion of UO_2 in graphite is now achieved with confidence. Although UC is more stable in the graphite matrix than the oxide fuel particles, it is unstable in air, requiring processing of the fuel blocks in inert atmospheres. Production and characterization of material compatibility samples have effectively enabled down selection to the UO_2 graphite system.

Two methods for fabricating fuel blocks were investigated, including compaction and extrusion. Two LEU fuel blocks, patterned after the existing TREAT cores, were fabricated for the TREAT conversion project by each process. Babcock & Wilcox Nuclear Operations Group was asked to

develop and demonstrate feasibility of a compaction process to make LEU fuel blocks. Los Alamos National Laboratory was asked to recreate the extrusion process used to fabricate TREAT upgrade fuel blocks, a request that was later broadened to allow development of the process using modern resins and to increase the graphite content of the samples from approximately 59% to 72%. The fabrication feasibility study recommended pursuit of the compaction process for the fabrication of LEU fuel blocks for the TREAT core. Although the extrusion process has been demonstrated as feasible, the compaction process has a significant advantage by producing fuel blocks of higher graphitic content (graphite to total carbon ratio of 0.91 vs. 0.72)., higher density, and slightly better thermal properties. The compaction process appears capable of producing fuel blocks with a highly-uniform dispersion of surrogate fuel in a high-density graphite matrix. The compaction technique will be further developed as needed to minimize or prevent cracking to maximize the thermal diffusivity (and hence thermal conductivity) and to help ensure mechanical stability.

Zircaloy-3 was used as the cladding material in the original TREAT core. Zircaloy-3 and Zircaloy-4 alloys were both identified as viable cladding material options based on oxidation rate information from the TREAT Final Safety Analysis Report (FSAR), Nuclear Regulatory Commission oxidation studies [3], and from the TREAT conversion scoping materials performance test data [2]. Zircaloy-4 is a commercially available alloy; therefore, it is immediately available for out-of-pile qualification testing. M5 (an Areva, Inc zirconium-niobium alloy), while not tested during scoping efforts, was also identified as a potential cladding material due to its potentially lower oxidation rates compared with Zircaloy-4.

Of the various cladding/can material characteristics, oxidation is of prime concern for the project. Data sets for materials of interest were identified. These data sets were limited and employed different test parameters between them, however, resulting in their being an unsatisfactory basis for selecting the zirconium alloy best suited for TREAT. To address this data gap, thermo-gravimetric analysis testing of zirconium-based alloys in air was performed to augment the existing data, using a consistent set of test parameters. To support such testing, legacy Zircaloy-3 pieces were found in the TREAT warehouse, Zircaloy-4 sheet stock was obtained from ATI Wah-Chang, pure zirconium sheet was obtained from existing sheetstocks at INL, and Zirlo sheet stock was obtained from LANL. Oxidation-rate data were taken on the first three types of zirconium alloy samples at four different temperature regimes. Such data on Zirlo were measured at only one temperature (600°C). The results for post-breakaway corrosion are presented in Table 1. Although the sample sets used for the scoping studies is not statistically representative, it is used to form an initial comparison between various Zircaloy-based alloys for cladding down selection. The newly measured Zircaloy-3 scoping oxidation rates were found to be generally higher than the measured values reported in the TREAT Final Safety Analysis Report (FSAR). The new Zircaloy-4 oxidation rates are lower than the prior measurements at 600°C and above. The lower Zircaloy-4 oxidation rates may be due to improved heat treatment controls during the beta-phase quenching step. This needs to be confirmed with a larger sample set. However, care should be taken when comparing the different data sets tabulated in Table 1, because different experimental protocols were used. Comparing only the data set from the scoping studies, the oxidation rate of Zircaloy-4 appears to be roughly 50 to 100% higher than those of Zircaloy-3 over the range 600 – 800 °C.

From these estimated oxidation calculations, both Zircaloy-3 and Zircaloy-4 cladding appear to be feasible design options for TREAT conversion fuel element assembly design. Further oxidation rate statistical data will be acquired to verify the values used in this evaluation during the conceptual design phase. Additionally, the alloy M5 needs to be considered due to the low air

oxidation rate reported in the NRC 2004 report [3]. Additional mechanical testing may be required if M5 is identified as the ideal candidate material because less test data exist over the temperature range relevant to the TREAT core environment. If needed as a potential design option, cladding protecting layers applied by, e.g., atomic layer deposition, chemical vapor deposition, laser deposition, will be considered during future design phases. Rigorous investigation needs to be conducted on the effect of temperature on phase transitions for the candidate materials. Thermal analyses showed that cladding temperatures during accident conditions without cooling may cause the material to undergo phase transition.

Fabrication of the cladding/can, using one of a couple of zirconium-based material candidates, appears to be feasible. Standard forming processes should be adequate to address this component of the LEU fuel element assembly.

Table 1. Oxidation Rates Benchmark Study.

	Scoping Studies (New)[2] Rate ($mg/cm^2/hr$)				**NRC 2004** [3] Rate ($mg/cm^2/hr$)			**TREAT FSAR** (Table 15.3-2 page 15-217) [S3942-0001-YT] **[4]** Rate ($mg/cm^2/hr$)		
Temp (°C)	Zr	Zr-3	Zr-4	Zirlo	Zr-4	Zirlo	M5	Zr[a]	Zr-2[a]	Zr-3[b]
500	0.0114	0.0105	0.0109	*	0.0069	0.0084	0.006	0.00279	0.0147	*
600	0.0391	0.0448 0.0744	0.0679 0.0864	0.1408	0.12	0.12	0.077	0.0257	0.148	0.041
700	0.0879	0.3970	0.6970	*	0.74	1.0	0.48	0.15	0.928	0.24
800	0.1939	1.930	4.576	*	6.4	6.9	3.0	0.631	4.13	2

*Not measured

a. Data for Zr-2 and pure zirconium were only obtained at 500°C, 600°C, and 700°C, with other values extrapolated using Arrhenius relationship. Rates obtained after transition to linear corrosion.

b. Zr-3 data were obtained using transient tests from room temperature to testing value 35 times over 69 hours. The specimen was a used piece of a TREAT fuel element. The 800°C data were obtained on a specimen already tested at 600°C.

4. Current LEU Fuel Conceptual Design Options

During the pre-conceptual design phase and engineering efforts, eight candidate LEU fuel element designs were identified. During the refining process after the initial design down-selection, additional design variants have been identified. The objective of the initial selection process for the conceptual phase series (numbered 2x) was to identify the design with the most effective thermal performance (using similar neutronics input for each). Follow-on selection efforts focused on neutronics performance and will be discussed in Section 5. The key parameters of the current conceptual design variant designs are summarized in Table 2. All

variants have the same flat-to-flat distance (3.96 in.) as the original HEU fuel assembly.

Table 2. Parameters Used for the Current TREAT LEU Conceptual Options.

Parameters	Baseline 2.0	Variant 2.1	Variant 2.2	Variant 2.3
Can wall corner/side properties	Chamfered	Chamfered	Chamfered	Rounded
Clad Thickness	25 mil	67.3 mil	67.3 mil	67.3 mil
Fuel/Clad gap (before gas evacuation)	55 mil	35.5 mil	70.5 mil	70.5 mil
Fuel/Clad gap (after gas evacuation)	Direct contact[a)]	20 mil	55 mil	55 mil
Fuel length (4 ft)	122 cm	122 cm	122 cm	122 cm
Fuel volume	11120 cm^3	10865 cm^3	10479 cm^3	10477 cm^3
Corner section open flow area	2.98 cm^2	2.98 cm^2	2.98 cm^2	3.38 cm^2
Corner section surface perimeter	6.35 cm	6.35 cm	6.35 cm	11.05 cm
Side section open flow area	1.59 cm^2	1.59 cm^2	1.59 cm^2	1.33 cm^2
Side section surface perimeter	31.25 cm	31.25 cm	31.25 cm	26.16 cm
Coolant flow rate per assembly	7.48 cfm	7.48 cfm	7.48 cfm	7.48 cfm
Corner/Side mass-flow ratio	22/1	22/1	22/1	20/1

a) Following an evacuation of the fuel assembly, 43% of the side clad wall is in direct contact with the fuel block.

As part of the LEU fuel conceptual design process, thermal analysis using the COMSOL multi-physics finite-element code was performed to test the effect of the cladding configurations identified as design Variants 2.1, 2.2, and 2.3. The design of the original fuel cladding utilized chamfered corners, while the upgrade core used rounded corners. Both forming shapes were considered for the LEU fuel element cladding design. Preliminary rounded forming efforts were executed on fabricating Zircodyne 702 cladding. Wall thickness measurements made on the rounded corners showed thinning of the cladding material at the outside corners. Forming chamfered corners on the other hand does not present the same challenge because the amount of material formed is reduced. Initial demonstration of chamfered and rounded cladding specimen capabilities are shown in Figure 3.

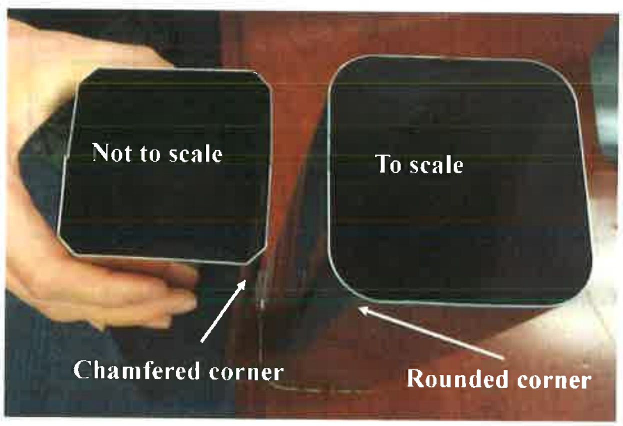

Figure 3. Initial demonstration of chamfered and rounded cladding/can specimen capabilities.

Variants 2.2 and 2.3 have a similar fuel volume, but the flow and surface areas between clad and air coolant differ between the two cases (see Table 2). Assuming a 300 ms, bell-shaped transient power profile and for total assembly energy of 12.7 MJ, the calculated peak fuel temperature was almost the same for both cases (632°C for Variant 2.2 and 636°C for Variant 2.3). However, the peak side wall temperature of the cladding was about 20°C cooler for the case with rounded corners (533°C for Variant 2.2 and 514°C for Variant 2.3). For the same coolant air flow rate (i.e., 7.48 cfm per assembly), the cooling is slightly more effective in the case of rounded corners. This is partly due to an increase in heat transfer surface area at the corner section as well

as a reduction in conduction pathway from the side region to the side/corner region of the cladding. Variant 2.1 had a calculated peak fuel temperature of 617°C and a peak cladding temperature of 517°C.

The conceptual LEU fuel assembly design with chamfered corners was therefore identified as the most viable from a thermal analysis and fabrication perspective. Variants 2.1 and 2.2 both featured chamfered corners, but preliminary neutronics analyses found that the smaller fuel volume of Variant 2.2 would not provide sufficient excess reactivity, given the current constraints on the fuel properties. Engineering evaluations needs to be completed to determine the most feasible method to ensure alignment so that the required gap can be maintained. This variant design has a cladding thickness and fuel cladding gap both large enough to prevent the cladding from bowing inward enough to touch the fuel after the sealed cladding is evacuated during fuel assembly manufacture and subsequently heated during transient operation in the core. This design includes a vacuum between fuel and cladding that is maintained during the lifetime of the assembly at a pressure low enough that conduction heat transfer from fuel to cladding is negligible compared to radiative heat transfer. (Use of ultra-low-conductivity microporous insulation and pyrolytic graphite were considered as an alternative to the vacuum gap, but were not included in the reference design due to the added complexity involved in incorporating those additional materials in the fuel element assembly process, as well as the added costs of qualifying the more complex design.)

The current TREAT LEU fuel design option selected for development during the conceptual design phase is UO_2 fuel particles in a graphite matrix, encapsulated with zirconium-based clad and graphite reflectors at both ends. Figure 4 shows a schematic presentation of the main characteristics of the current conceptual LEU design compared with the previous original and upgrade HEU designs as is documented in the TREAT Conversion LEU Fuel Design Trade Study, INL/LTD-14-31704 [2]. After this fuel geometry was selected, neutronics analyses were performed to determine a fuel composition which both provides sufficient excess reactivity and meets shutdown margin requirements (using the existing control rods).

5. Status of Neutronic Analyses regarding LEU Fuel Conversion Feasibility

For an LEU core to be feasible in TREAT, it must be able to match the performance capability of the HEU core while meeting all of the safety requirements. Because there will be a harder neutron spectrum in an LEU-fueled core, it will be necessary to operate at higher core powers in order to achieve the same total energy deposition (TED) as the HEU core in a given test sample. These higher core powers will translate into higher peak fuel temperatures.

The limit on TREAT power is dictated by the temperature constraints of the TREAT fuel cladding, which are in place to limit cladding oxidation. In the Zircaloy-3-cladded HEU fuel elements, the cladding must remain below 600°C during a planned transient and below 820°C in an accident scenario. The HEU fuel is assumed to be in contact with the cladding, so these temperature limits are imposed as a limit on fuel temperature. LEU core analysis was performed using the M8CAL half-slotted core loading, with the M8CAL fuel pins as the test sample, for the current fuel design. In TREAT operations, the relationship between TREAT core energy and energy in a test sample is expressed in terms of the power coupling factor (PCF), as Joules per gram of test fuel per MJ of TREAT core energy, or the equivalent watts per gram of test fuel per MW of TREAT core power. For the M8CAL test pins, there is a 26% decrease in the PCF in the LEU core compared to the HEU core. This means that an equivalent increase in core power is needed to generate the same TED in the pins.

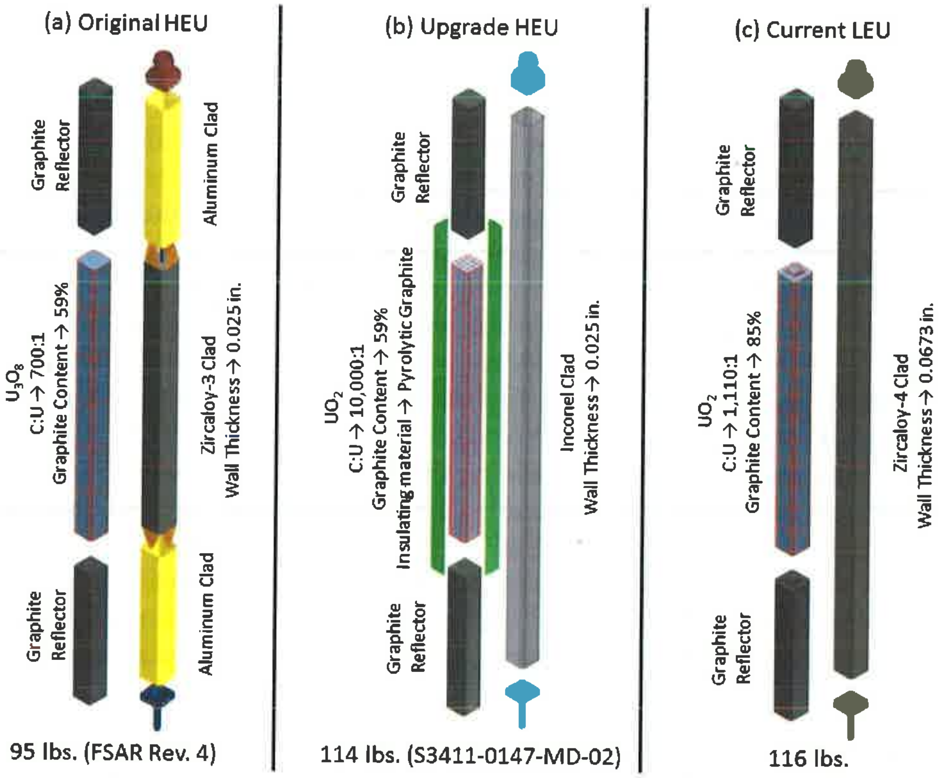

Figure 4. Comparative Schematic Presentation of the Main Characteristics of the (a) Original and (b) Upgrade HEU Fuel Element Assembly and the (c) Draft Conceptual LEU Fuel Element Assembly Design.

Based on the experimentally-performed temperature-limited transients and linear extrapolation, the maximum-allowed reactivity had been determined during the M8CAL core operations as 5.95% for the HEU core. This reactivity would result in a peak fuel temperature of 820°C if it were inserted as a step. In the context of the feasibility analysis it was decided to study the resulting fuel and cladding peak temperature if the LEU core would match the TED of the bounding case. Table 3 summarizes key parameters of the bounding (5.95% available reactivity before the transient) shaped-transient case in the HEU core, along with the LEU core shaped transient necessary to generate the same TED in the test sample. The temperatures reported here are values computed with the point-kinetics code TREKIN. The time histories of core power and transient rod motion are shown in Figure 5(a) and (b), respectively. Figure 6 illustrates the temperature-limited transient power-time history that occurs when all of the initial reactivity available at the beginning of the LEU shaped transient is inserted as a step. This power-time history, which represents the accident case, was used for the thermal-hydraulic calculations of the cladding heating and cooling down temperatures of the bounding case.

The accident scenario power profile illustrated in Figure 6 was evaluated to determine the corresponding peak cladding temperatures in the extreme, hypothetical event where both of the TREAT blowers fail, and there is no natural convection. In this scenario, the only mode of heat dissipation is conduction, and a peak cladding temperature of 918°C was determined. . In the case of 3000 cfm forced air flow during this transient, the computations showed a peak cladding temperature of 876°C, which is approximately 40°C lower than the peak fuel temperature. Thus, for the LEU core to feasibly meet the reactor conversion performance objectives, the cladding will need to acceptably withstand such temperatures.

Table 3. Key Properties of the M8CAL Bounding Shaped Transient for Equal TED

	Initial Reactivity	Peak Power (MW)	Peak Core Temperature (°C)
HEU Core	5.95%	293	454
LEU Core	5.20%	392	533

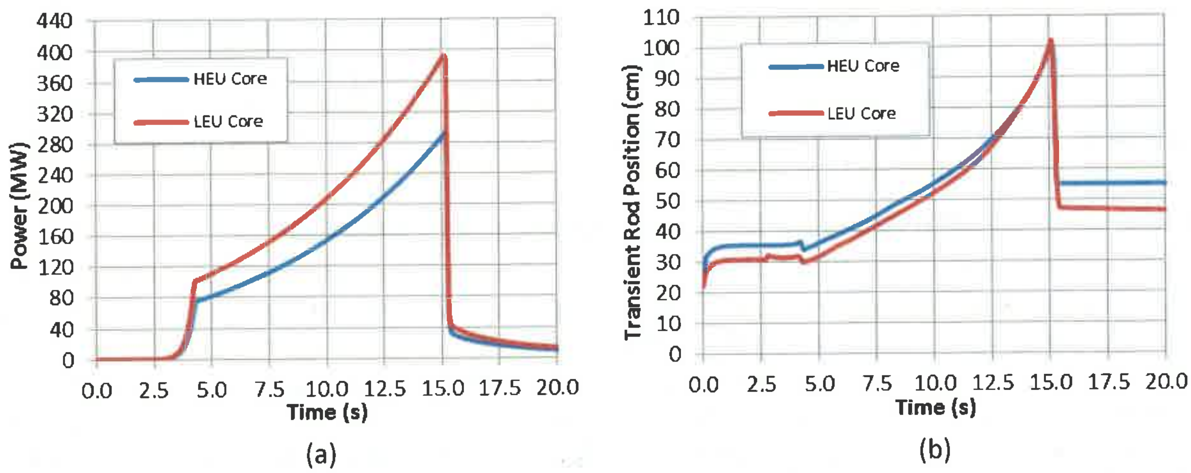

Figure 5. (a) Power-time History of the M8CAL HEU Core Bounding Shaped Transient Case (5.95% Reactivity) and Corresponding LEU Shaped Transient Needed for the Same TED and (b). Transient Rod Motion during the M8CAL HEU Core Bounding Shaped Transient (5.95% Reactivity) and Corresponding LEU Shaped Transient Needed for the Same TED.

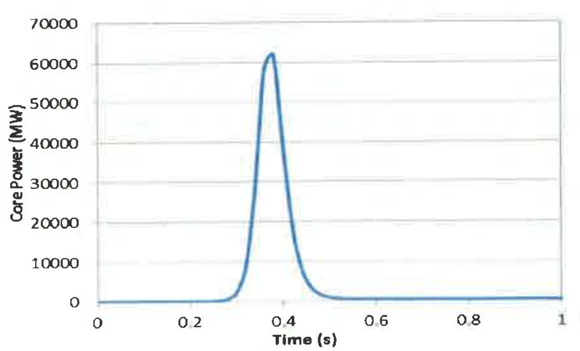

Figure 6. Power-time History of Reactivity Insertion Accident in LEU Core

As alluded to above, only the M8CAL core-and-experiment system has so far been used for the LEU/HEU comparison evaluation of this design. Results for several other core-and-experiment systems of different types will also need to be determined to ensure that the results from the M8CAL case apply to a sufficiently wide range of experiments.

In summary, the MCNP neutronics model of the TREAT M8CAL core, the COMSOL-based thermal-hydraulic model of the TREAT fueled assembly, and the TREKIN point-kinetics code, have been applied to a range of LEU fuel assembly design concepts, addressing several parameters in various combinations. These parameters have included fuel-cladding gap size, cladding thickness, cladding material, intervening materials between fuel and cladding, fuel

density, degree of carbon graphitization, fuel carbon-to-uranium ratio, fuel fissile height, numbers and arrangements of fuel assemblies of different fissile height. (In addition, utilization of non-fueled BeO or graphite assemblies near the core center was considered, although not mentioned above.) A concept that provides experiment-fuel total energy depositions (using the M8CAL fuel pins as a reference case) without significantly exceeding peak cladding temperatures allowed in HEU core operations was not found despite this extensive parameter study. The work described did, however, help to guide and evaluate design improvements leading to a design that presently appears to provide the required performance characteristics. Overall, the current LEU design concept has tentatively shown satisfactory neutronic characteristics, but it requires the core to generate higher transient energies than did the HEU core in order to produce the same TED in the M8CAL test fuel pins. The computed corresponding peak fuel temperature rise would be about 15-20% higher in the hottest LEU assembly than in the hottest HEU assembly.

6. Status of Structural Analysis LEU Fuel Conversion Feasibility

Structural integrity of the TREAT LEU fuel elements while in the reactor, are driven by gravity, pressure, vibration, and thermal loadings. In general gravity will apply compression onto the end fittings, can, and graphite contents (fuel and reflectors). The end fittings will experience very minimal pressures at their interface with the can (~20 psi) or with the graphite (~6 psi) but their long length makes them a candidate for buckling concerns. Although the proposed can design implements thicker walls than the original or upgrade design, the end geometry dictates a slightly reduced radius of gyration (1.55 in vs. 1.566 in), which is the primary buckling driver. As such, this modification of the can design from the original or upgrade designs as well as the implementation of a new material requires a can buckling calculation.

The buckling capacity of the graphite is being evaluated according to the ASME Section III Division 5 Subsection HH standard which also directs the means by which the material properties of the particular graphite being developed is to be measured to perform these structural calculations. Preliminary calculations implementing similar graphite materials show little buckling concern even given conservative fuel geometries. It is important to address can stress to ensure that it will not adversely deform, elastically or plastically, in a manner that would affect the fuel element assembly's function within the reactor. It is anticipated that these structural analyses will yield viable designs that can be further optimized to meet overall project design and performance requirements. Validation of structural model outcomes will be confirmed with experimental data.

7. LEU Fuel Element Qualification Approach

TREAT is a DOE-regulated research reactor and DOE does not have a prescribed regulatory framework for qualification of fuel for research reactors. Consequently, the TREAT Conversion Project's fuel qualification process will rely on information from previous qualification campaigns (i.e., historical precedence), along with individual reactor requirements and International Atomic Energy Agency guidance to formulate both high level and more specific lower-level requirements. Therefore internally generated, program-specific requirements provide the framework for this LEU fuel qualification plan and the LEU fuel qualification report [4]. The physical design of the LEU fuel element assembly is such that it will provide the same test capability as the original high-enriched uranium core, but specific engineering and performance challenges remain. These differences and challenges have required testing to demonstrate compliance with LEU fuel conversion requirements, because either the historic data does not exist or new materials or processes are used for the LEU fuel element assembly. Specifically, cladding material performance testing and fuel compact out-of-pile testing are required.

Performance-basis testing will consist of tests on samples rather than on complete fuel elements. Oxidation of the fuel cladding is a crucial consideration in TREAT reactor operations and performance, and qualification tests will therefore focus on material testing on smaller samples prior to fabrication of a Lead Test Assembly (LTA). Two assembly tests, an out-of-pile test and steady-state irradiation, are recommended for focusing on demonstrating dimensional stability, structural integrity, interface behavior and potential corrosion characteristics, thermal hydraulics, and neutronic performance of the assembly. These tests will be followed by the LTA campaign (three transient irradiation tests) to similarly demonstrate the characteristics mentioned above. As TREAT Conversion Project activities are pursued, the qualification approach will be revised as needed.

9. Conclusions

Based on the results of current fabrication feasibility evaluation, structural, thermal and neutronic analysis, it appears feasible to fabricate a TREAT LEU fuel element assembly that can meet TREAT design, performance, and safety requirements. The statement of feasibility recognizes that further development, analysis, and testing must be completed to refine the baseline design. Testing will focus on cladding oxidation and phase change, along with fuel element assembly material characterization supporting fuel element assembly qualification. Preliminary supply chain evaluation provided confidence that the conceptual designs can be achieved. The following design options will be used to address design and performance limitations as the fuel element assembly design matures:

- Tighter control on boron content (< 2ppm)
- Larger graphite content (>85%)
- Continue to optimize fuel C:U ratio as needed
- Use of insulation to reduce peak cladding temperature
- Fuel shuffling scheme during operations to manage cladding oxidation
- Replacement of fuel element assemblies (surveillance of cladding oxidation)
- Alternate fuel element cladding materials such as M5 from Areva
- Use of surface layers to enhance cladding oxidation performance
- C:U grading in radial and axial directions

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