PRELIMINARY SAFETY ANALYSIS REPORT FOR CONVERSION OF THE UNIVERSITY OF MISSOURI-COLUMBIA RESEARCH REACTOR TO LOW-ENRICHED URANIUM FUEL

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ABSTRACT

The University of Missouri Research Reactor (MURR®) is one of five U.S. high performance research and test reactors that are actively collaborating with the U.S. National Nuclear Security Administration (NNSA) Office of Material Management and Minimization (M^3) to find a suitable low-enriched uranium (LEU) fuel replacement for the currently required highly enriched uranium (HEU) fuel. In January 2017, MURR received a new 20-year license for operation with HEU fuel from the U.S. Nuclear Regulatory Commission (NRC). More recently, a Preliminary Safety Analysis Report (PSAR) for a proposed core loaded with U-10Mo monolithic LEU fuel that is currently being tested for qualification has been completed and submitted to the NRC for review. The PSAR was prepared in a format consistent with NUREG-1537 and describes all changes to the fuel element design and core operating conditions due to conversion. Transient accident analyses for postulated positive reactivity insertion accidents (RIAs), loss of coolant accidents (LOCAs), and loss of flow accidents (LOFAs) were completed under conditions established during relicensing of the HEU core and consistent with guidelines in NUREG-1537. The models include both fresh and irradiated fuel assemblies and the associated fresh and irradiated fuel thermo-physical properties, which may affect the severity of the accidents. Also, analyses of accidents with radiological consequences, including the maximum hypothetical accident (MHA), which is a fuel handling accident with LEU fuel, have been revised and are discussed. Detailed analyses at steady-state and accident conditions demonstrate that following conversion with the proposed fuel form and a power uprate there are sufficient safety margins and operational performance is maintained. All accident scenarios demonstrate an acceptable margin to potential fuel damage or acceptable dose consequences in the case of the MHA.

1. Introduction

1.1 General Background

Because of its compact core design (33 liters), which requires a very high loading density of ^{235}U, the University of Missouri Research Reactor (MURR) could not perform its mission with any previously qualified low-enriched uranium (LEU) fuels. However, in 2006 with the prospect of the National Nuclear Security Administration (NNSA) Global Threat Reduction Initiative (GTRI) Reactor Conversion (RC) Program validating the performance of U-10Mo monolithic LEU foil fuels, MURR began actively collaborating with what is currently the NNSA Material Management and Minimization (M^3) Office's RC Program, along with four other U.S. high-performance research and test reactors that use highly enriched uranium (HEU) fuel, to find a suitable LEU fuel replacement. It was concluded that the proposed LEU fuel assembly design, in conjunction with an increase in power level from 10 to 12 MW_{th}, will (1) maintain safety margins during steady-state and transient conditions, (2) allow operating fuel cycle lengths to be maintained for efficient and effective use of the facility, and (3) preserve an acceptable level and spectrum of key neutron fluxes to meet the scientific mission of the facility.

A preliminary version of the LEU Conversion Safety Analysis Report (SAR) was submitted to the U.S. Nuclear Regulatory Commission (NRC) on August 18, 2017. The report follows the recommended format and content of NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Chapter 18, "Highly Enriched to Low-Enriched Uranium Conversions." The emphasis in any conversion SAR is to explain the differences between the LEU and HEU cores and to show the acceptability of the new design; there is no need to repeat information regarding the current reactor that will not change upon conversion. Hence, as seen in the report, the bulk of the SAR is devoted to Chapter 4, "Reactor Description," and Chapter 13, "Accident Analyses."

2. MURR Fuel Element

The current MURR 775-gram HEU fuel element is a product of the UAl_x dispersion fuel system development. The UAl_x dispersion fuel system was developed at Idaho National Engineering Laboratory (INEL) for the high flux, high power Advanced Test Reactor (ATR) and subsequently used at the Materials Test Reactor (MTR) and Engineering Test Reactor (ETR) prior to its use at MURR [1, 2]. Several features of the UAl_x dispersion fuel system increase the fuel performance capability in high flux reactors [3, 4, 5, 6]. One of these features is that the powder dispersion fuel matrix provides some inherent void space, which helps reduce fission product swelling. The UAl_x structure has exceptional tolerance for fission gas retention and burnable poisons can readily be dispersed in the fuel matrix. Due to the compact core size, the fuel uses HEU to obtain sufficient excess reactivity to be able to operate at 10 MW.

Conversion of MURR from an HEU fuel system to an LEU fuel system has required redesign of the MURR fuel element. Reducing the enrichment to 19.75 wt.% ^{235}U led to changing the fuel-bearing region of the MURR plates from a dispersion fuel form to an alloy of uranium and 10 wt.% molybdenum, referred to as the U-10Mo monolithic fuel foil. This high-density U-10Mo monolithic alloy fuel is currently under development and qualification [7].

The change in fuel form also required adjustments to the fuel-bearing region thicknesses in the fuel plates and a higher metal-to-water ratio by increased coolant channel thickness, resulting in the reduction of the number of fuel plates in an element from 24 to 23. Additionally, a reactor power uprate to 12 MW is required to allow the reactor to continue to meet the facility's mission. The plate-specific calculated peak plate burnup in the 12 MW LEU fuel system is $< 3.4 \times 10^{21}$ fissions/cm^3, compared to the $< 2.3 \times 10^{21}$ fissions/cm^3 in the 10 MW HEU dispersion fuel system.

2.1 LEU Fuel Element Description

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). The LEU fuel element is designed to be similar to the HEU fuel element. The LEU element has identical side plates and end fitting geometry, which is compatible with the existing support structure. As discussed above, the number of fuel plates in the element has been reduced from 24 plates in the HEU element to 23 plates in the LEU element. There are also changes to the thickness and arc width of the fuel zone in the plates, the overall thickness of the fuel plates, the thickness of the coolant channels between plates, and the thickness of the coolant channels between the outermost plates and the inner and outer reactor pressure vessels.

The LEU fuel element was developed through an extensive series of scoping studies, followed by optimization of a design that meets the conversion goals of reactor safety, fuel utilization, and experimental performance. The proposed fuel element has been designated "CD35," and is constructed with 23 fuel plates (instead of 24 plates as in the HEU elements). The element uses various U-10Mo foil thicknesses to flatten the radial heat flux profile. The thinnest nominal aluminum alloy AA6061 cladding is 0.011 inches (0.2794 mm) and the thinnest total plate thickness is 0.044 inches (1.1176 mm). There is a thin zirconium layer that is nominally 0.001 inches (0.0254 mm) on the largest surfaces of the fuel foil that serves as an interaction barrier layer between the U-10Mo and AA6061 cladding. The other portions of the element construction (e.g., side plate length, width, and thickness; and end fittings) will be identical to the HEU fuel element. A drawing of the MURR LEU fuel element and a schematic of the element cross-section are shown in Figure 1. Eight (8) such fuel elements comprise the fixed MURR core.

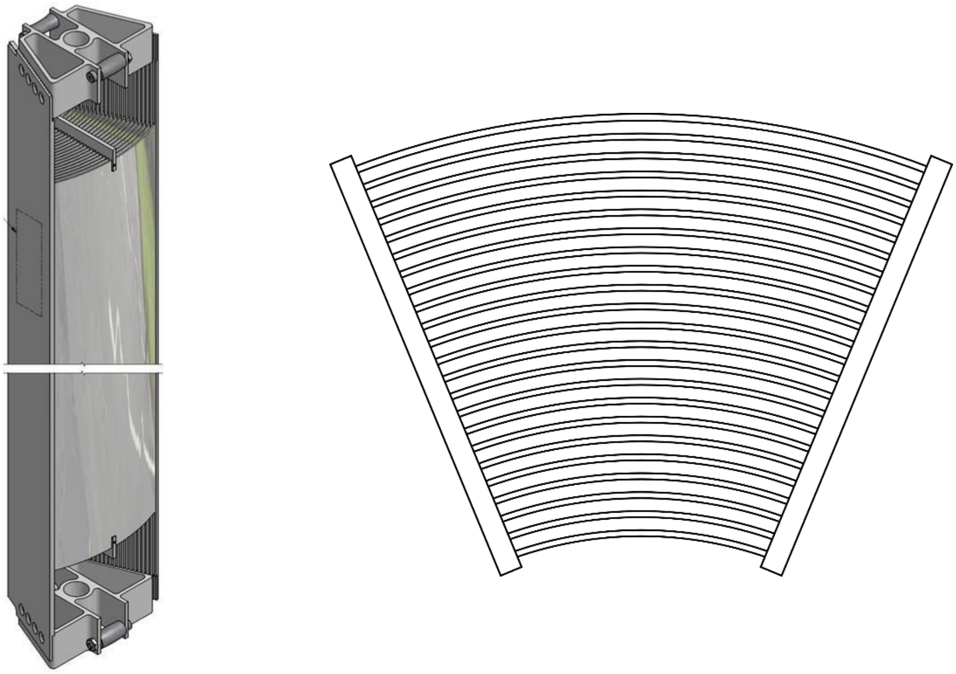

Figure 1. LEU Fuel Element

Table 1 shows a comparison of the summary data for the HEU and LEU fuel elements. Table 2 provides a summary of the LEU fuel system nominal specifications alongside the prior HEU specifications for comparison. The proposed design changes required a thinner fuel plate thickness and one fewer plate in order to provide adequate neutron moderation. Otherwise, the LEU fuel element design is of identical construction to the HEU fuel element without changes to the side plates, combs, or end fittings, except for changes to accommodate the number and thickness of the fuel plates and the thickness of the coolant channels.

Description	HEU Nominal Value	LEU Nominal Value
Fuel Content (grams ^{235}U per element)	775	1507
Type of Fuel	Aluminide-UAl_x mostly UAl_3 Phase	U-10Mo Monolithic Alloy
Fuel Density (grams of ^{235}U loaded per cubic centimeter)	1.43	3.03
Boron Content (natural boron per element)	Trace Impurities	Trace Impurities
Cold Excess Reactivity, Δk/k (clean core – control blades full out)	8.6%	7.0%
Control Blade Worth for Fresh / Mixed Cores (Δk/k)	20.3% / 21.5%	17.0% / 17.7%
Peak Burnup Density (fissions per cubic centimeter)	$< 2.3 \times 10^{21}$	$< 3.4 \times 10^{21}$
Energy per Element at Peak Fission Density	≈ 150	180

Description	HEU Nominal Value	LEU Nominal Value
(MWd per element)		

Table 1. Summary of the HEU and LEU Fuel Elements

Description	HEU Nominal Value	LEU Nominal Value
Fuel Material		
Enrichment ^{235}U	93%	19.75%
Thickness	0.020 inches (0.508 mm)	0.009 – 0.020 inches (0.2286 – 0.5080 mm)
Interlayer		
Material	N/A	Zirconium
Thickness	N/A	0.001 inches (0.0254 mm)
Cladding		
Material	Aluminum AA6061	Aluminum AA6061
Thickness	0.015 inches (0.381 mm)	0.011 – 0.0165 inches (0.2794 – 0.4191 mm)
Fuel Assembly		
Number of Fuel Plates	24	23
Innermost Fuel Plate Center Radius	2.795 inches (7.099 cm)	2.778 inches (7.056 cm)
Outermost Fuel Plate Center Radius	5.785 inches (14.694 cm)	5.780 inches (14.681 cm)
Overall Fuel Assembly Length	32.5 inches (82.550 cm)	32.5 inches (82.550 cm)
Overall Fuel Plate Length	25.5 inches (64.770 cm)	25.5 inches (64.770 cm)
Overall Active Fuel Length	24.0 inches (60.960 cm)	24.0 inches (60.960 cm)
Fuel Plate Thickness	0.050 inches (1.270 mm)	0.044 – 0.049 inches (1.118 – 1.225 mm)
Distance Between Fuel Plates	0.080 inches (2.032 mm)	0.092 – 0.093 inches (2.337 – 2.362 mm)
Nominal ^{235}U Loading	775 grams	1507 grams
Element Weight	6.25 kg	12.50 kg

Table 2. Summary of the HEU and LEU Fuel Element Specifications

3. Accident Analyses

3.1 Maximum Hypothetical Accident

The Maximum Hypothetical Accident (MHA) postulates conditions leading to consequences worse than those from any credible accident. For the HEU core, the failure of a fueled experiment results in the greatest dose to a member of the general public outside the exclusion area. For the LEU core, it has been found that the dose to a member of the general public is greatest for a postulated fuel-handling accident (FHA).

FHAs cover a class of accident caused by events or scenarios that could cause a breach in the fuel cladding that result in the release of radioactive fission products. Events, which could cause an accident in this category, include a fuel-handling event where a fuel plate is damaged or scratched severely enough to breach the cladding or the simple failure of the fuel cladding due to a manufacturing defect or corrosion.

For the MURR FHA, it is conservatively assumed that the cladding on a fuel plate is damaged and a section of fuel meat that is one-inch square and 5 mils thick is exposed. The mass of fuel contained in this volume is 0.117 and 0.248 grams of ^{235}U for the HEU and LEU fuel element, respectively. Only the innermost and outermost plates of a fuel element are likely to receive any potential damage in the FHA. The innermost plate of the element (plate 1) is assumed to be damaged in the FHA analysis, and it is assumed that the damage occurs at the location of the peak power density. While the outermost plate of a fuel element has the most surface area, and therefore a greater likelihood of being damaged, it has a lower peak power density, and consequently, will have a lower inventory of fission products per gram of fuel than the innermost plate.

The whole-core radioactive material source term, as provide in Table 3, was calculated by assuming full-power operation of the MURR core in twelve 10-day cycles over a 300-day period to simulate multiple cycles of fuel irradiation. The total irradiation time in the calculation of the source term was 120 full-power days, which is comparable to the typical number of full-power days for a MURR HEU element prior to discharge (19 cycles of operation at 6.3 full-power days per cycle). The HEU core power level is 10 MW, so that the fission product inventory assumed in the accident analysis corresponds to a total core burnup of 1,200 MWd. This gives a very conservative radioactive source term, since the total burnup of all elements in a typical HEU core is about 650 MWd.

		HEU (SAR [9])	HEU	LEU
Iodines	^{131}I	1.7E+05	2.20E+05	2.70E+05
	^{132}I	3.3E+05	3.08E+05	3.77E+05
	^{133}I	5.1E+05	5.42E+05	6.60E+05
	^{134}I	6.3E+05	6.11E+05	7.42E+05
	^{135}I	5.2E+05	5.06E+05	6.16E+05
Noble Gases	^{85}Kr	4.7E+02	4.63E+02	5.46E+02
	$^{85}Kr_m$	1.1E+05	1.31E+05	1.54E+05
	^{87}Kr	2.1E+05	2.05E+05	2.43E+05
	^{88}Kr	3.0E+05	2.91E+05	3.44E+05
	^{89}Kr	3.8E+05	3.69E+05	4.36E+05
	^{90}Kr	3.8E+05	3.68E+05	4.34E+05
	^{133}Xe	4.2E+05	3.85E+05	4.67E+05
	^{135}Xe	9.6E+04	7.56E+04	1.64E+05
	$^{135}Xe_m$	9.4E+04	3.62E+04	1.03E+05
	^{137}Xe	4.9E+05	4.81E+05	5.84E+05
	^{138}Xe	5.2E+05	5.01E+05	6.04E+05
	^{139}Xe	4.2E+05	4.07E+05	4.89E+05

Table 3. Whole-Core Activity of Radioiodines and Noble Gases (Curies)

A fuel cycle simulation for the MURR operating with the proposed LEU fuel demonstrated that the weekly operations cycle for the converted core would be the same as for the current HEU fuel [8]. Therefore, the radioactive source term in the LEU core is also computed based on twelve 10-day cycles, although the core power level in the calculation is increased to 12 MW, which is the uprated power necessary to maintain the MURR experimental performance with the LEU fuel. Consequently, the fission product inventory for the accident analysis corresponds to a core burnup of 1,440 MWd, while the anticipated total burnup of all elements in the LEU core in typical weekly operations will be approximately 750 MWd [8].

The whole-core fission product inventory for the HEU core analysis in the HEU SAR [9] was calculated using the ORIGEN code [10]. For the current HEU and LEU core FHA analysis, the whole-core fission product inventory is calculated using the MONTEBURNS code [11].

3.1.1 Occupational Dose

For calculating the dose as a result of the FHA, it is conservatively assumed that the noble gases in the exposed part of the fuel plate are released directly into the reactor containment building from the reactor pool, while the radioiodine mixes uniformly into the 20,000 gallons of pool water. When the reactor containment building ventilation system is in operation, the evaporation rate from the reactor pool is approximately 80 gallons of water per day, or 0.055 gallons/minute. For the purpose of the FHA, the assumption is made that there is an evaporation rate of 4 gallons/minute of pool water containing the radioiodines released from the damaged portion of the fuel. The noble gases (krypton and xenon) and radioiodines released from the fuel into the containment structure are assumed to uniformly mix into the containment building air free volume of 225,000 ft^3. None of the assumptions related to the pool evaporation rate or the exposure time for Operations personnel will be affected by the conversion to LEU fuel.

The occupational dose to a worker in the containment building for 5 minutes following the accident was calculated to be 447 mrem for the HEU FHA, and 827 mrem for the LEU FHA. These doses are well within the published regulatory occupational limit of 5,000 mrem (5 rem) for the total effective dose equivalent (TEDE). These dose values are also lower than the occupational dose of 1,180 mrem calculated for the fueled experiment failure accident [12].

3.1.2 Public Dose

The calculated dose for the general public resulting from the HEU FHA is 0.0083 mrem. For the LEU FHA, the public dose is calculated to be 0.0153 mrem. The dose to the general public from the fueled experiment failure accident was calculated in Reference 12 to be 0.0112 mrem. Since the public dose for the LEU FHA is greater than the fueled experiment failure accident, the FHA will be classified as the MURR MHA following conversion to LEU.

3.2 Insertion of Excess Reactivity

Following the guidance provided in NUREG-1537, two different accident scenarios for an insertion of positive reactivity were evaluated for the selected LEU core (CD35 element design) and the results are compared against those for the current HEU core. First, several positive step reactivity insertions, based upon the various maximum reactivity limits listed in the MURR Technical Specifications (TSs), were considered. Second, a continuous ramp insertion of positive reactivity, based on the simultaneous continuous withdrawal of MURR's four shim control blades and the regulating blade, was analyzed. The exact mechanisms or events that could cause these reactivity insertions can vary, but could include an inadvertent rapid insertion or removal of an experiment from the center test hole, reactor or equipment malfunction, or operator error. The analyses were performed using the PARET/ANL code [13].

3.2.1 Step Reactivity Insertion Accident

Results for a 0.6% Δk/k step Reactivity Insertion Accident (RIA) in a reference core loaded with LEU fuel elements are presented in Table 4. The reference core selected for this analysis (Core 7A) has eight elements with a mixture of burnups, which are typical for MURR operations. The loading of experiments in the center flux trap and reflector are also typical for MURR operations. It is conservatively assumed that the bank of four shim control blades is inserted by a reactor scram from their fully withdrawn positions (26 inches withdrawn). The reference core selected for this analysis (Core 7A) has eight elements with a mixture of burnups, which are typical for MURR operations. The loading of experiments in the center flux trap and graphite reflector region are the same as assumed in the HEU reference core and are typical for MURR operations. Several branch cases were also run to estimate the effect of potential non-conservative variations of the core operating conditions on the severity of the accident.

Results of Accident Analysis		Reactor Power, MW	Peak Fuel Temperature, °C (Plate)
Base Case		46.69	289.4 (EOL)
Branch01	Decrease thermal conductivity $\kappa_{U10\text{-}Mo}$ by 20%	46.68	297.9 (MOL)
Branch02	Increase oxide layer thickness by 23%	46.77	297.4 (EOL)
Branch03b	No coolant channel restriction with burnup	46.84	290.9 (MOL)
Branch04f	Pressurizer pressure = 90 psia; Pressurizer level = +4 inches	46.68	293.8 (MOL)
Branch05	Mode II LSSS Operating Conditions	25.41	241.1 (MOL)
Branch06b	$t_{insertion}$ = 0.1 s	41.47	285.2 (EOL)

Table 4. Peak Reactor Power and Fuel Temperature
for 0.6% Δk/k Step Reactivity Insertion Accident in LEU Reference Core

The step insertion of 0.6% Δk/k reactivity in the critical, steady-state core results in a sharp reactor power rise from the TS Limiting Conditions for Operations (LCO) power level (115% of full licensed power) of 13.8 MW to 46.7 MW before the transient is curtailed by negative reactivity feedback effects from changes in the fuel temperature and coolant conditions (density and temperature). The transient is terminated by a reactor scram 0.151 seconds after the reactivity insertion. The predicted maximum steady-state LEU fuel temperature is in an EOL plate at 167.5 °C, and reaches a peak of 289.4 °C in the accident. The calculated margin to the fuel temperature safety limit in this case is 109 °C.

3.2.2 Control Blade Withdrawal Accident

The second class of reactivity accidents analyzed to assess the impact of fuel conversion is a ramp reactivity insertion accident resulting from continuous control blade withdrawal initiated from the reactor operating at steady-state. Even though the reactivity addition resulting from the simultaneous withdrawal of all four shim control blades follows the typical differential rod worth curve behavior, it was conservatively assumed that the maximum value for reactivity addition from control blade withdrawal allowed by the MURR TSs occurs during the analysis of this accident. It was also previously found that the maximum rate of reactivity addition from control blade withdrawal in the LEU core is less than the TS requirement. Thus, a conservative positive ramp reactivity insertion at the rate of 0.03% Δk/k/second in the LEU core was considered. Table 5 summarizes the results of the analysis of this accident scenario in the reference mixed-burnup core loaded with LEU fuel. The calculated peak fuel temperature is much lower than that predicted for the 0.6% Δk/k step RIA, and is well below the fuel temperature safety limit.

Results of Accident Analysis		Reactor Power, MW	Peak Fuel Temperature, °C (Plate)
Base Case		15.10	176.6 (MOL)
Branch01	Decrease thermal conductivity $\kappa_{U10\text{-}Mo}$ by 20%	15.11	180.5 (MOL)
Branch02	Increase oxide layer thickness by 23%	15.11	180.9 (EOL)
Branch03b	No coolant channel restriction with burnup	15.10	178.5 (MOL)
Branch04f	Pressurizer pressure = 90 psia; Pressurizer level = +4 inches	15.10	176.6 (MOL)
Branch05	Mode II LSSS Operating Conditions	7.52	149.6 (MOL)
Branch07	Withdraw regulating blade simultaneous with shim control blades for first 24 seconds	15.13	176.6 (MOL)

Table 5. Peak Reactor Power and Fuel Temperature
for Control Blade Withdrawal Accident in LEU Reference Core

3.3 Loss of Primary Coolant

Historically, the most serious accident considered in the safety analyses of most reactors is the postulated loss of coolant accident (LOCA) from the primary coolant system, frequently initiated, in theory, by the double-ended rupture in a section of main coolant piping. The use of Engineered Safety Features (ESFs) greatly helps to mitigate the effects of this type of accident; however, the consequences of such an accident should still be considered.

The following four LOCAs were analyzed:

1. Cold leg LOCA – double-ended 12-inch diameter break at the core side of primary coolant isolation valve V507B (cold leg isolation valve).
2. Hot leg LOCA – double-ended 12-inch diameter break at the core side of primary coolant isolation valve V507A (hot leg isolation valve).
3. Cold leg LOCA – 8-inch diameter breach in the 8-inch cold leg piping at one of the two flow control diaphragm valves.
4. Cold leg LOCA – 2-inch diameter breach in the 2-inch cold leg piping that connects the pressurizer to the 8-inch cold leg piping.

The peak fuel temperature during the double-ended cold leg LOCA for the HEU core occurs in plate number-2 of element 5, which is a fresh element. It was previously calculated for a core with HEU fuel that 6.0% of the reactor power is deposited outside the primary coolant system [14]. Consequently, the steady-state HEU core power in the RELAP5 model of the LOCA is 9.4 MW. Since the radial geometry of all plates and coolant channels is represented in the RELAP5 model, any uneven heat conduction into the coolant channels adjacent to the two radial surfaces of the HEU fuel plate is accounted for directly in the RELAP5 analysis.

For the LEU core cold leg double-ended LOCA, the peak fuel temperature occurs in plate number-3 of fuel element 5, which is a fresh element. It was previously calculated for a core with LEU fuel that 3.6% of the reactor power is deposited outside the primary coolant system [14]. Consequently, the steady-state LEU core power in the RELAP5 model of the LOCA is 11.57 MW. Since the radial geometry of all plates and coolant channels is represented in the RELAP5 model, any uneven heat conduction into the coolant channels adjacent to the two radial surfaces of the fuel plate is accounted for directly in the RELAP5 analysis.

Table 6 below is a comparison of the most limiting LOCA for both the HEU and LEU cores (beginning and end of life); the double-ended rupture of the cold leg piping. The smallest margin to the LEU fuel temperature safety limit for this LCOA was calculated to be 115 °C and occurs in an EOL plate 22 in a case that assumed flow redistribution between the fuel elements due to a channel gap thickness reduction due to burnup.

Core	HEU	LEU	
Burnup	BOL	BOL	EOL
Channel gap thickness reduction due to burnup?	no	no	yes
Limiting Plate	2	3	22
Temperature Limit, °C (°F)	475 (887)	450 (842)	417 (783)
Peak Temperature, °C (°F)	266 (510)	303 (578)	301 (574)
Temperature Margin, °C (°F)	209 (376)	147 (265)	115 (207)

Table 6. Comparison of HEU and LEU Core Double-Ended Cold Leg Loss of Coolant Accident

3.4 Loss of Primary Coolant Flow

Any one, or a combination, of the following anomalies, can initiate a loss of flow accident (LOFA) for the primary coolant system:

(a) Loss of facility electrical power (or coolant circulation pump power);
(b) Inadvertent closure of coolant loop isolation valve(s);
(c) Inadvertent loss of pressurizer pressure;
(d) Locked rotor in a coolant circulation pump; and
(e) Failure of a coolant circulation pump coupling.

As shown in Section B-6.2 of Reference 15, the most limiting LOFA for the HEU core is accident (a) listed above, which is the LOFA caused by a loss-of-site power. The peak fuel temperature, 299 °F (148 °C), occurs in element 5, which is a fresh element, at 22.3 seconds after the start of the transient, in axial node 3 (just below the core midplane) of plate number-24. The analysis assumed no flow redistribution due to burnup, which would have increased flow to this fuel element. The temperature history at the location of the peak is provided in Figure 2.

As indicated in Section B-6.2 of Reference 15, the most limiting LOFA for the LEU core is accident (d) listed above, which is the LOFA caused by the locked rotor (i.e., seizure) of one of two primary pumps. The RELAP5 simulation that assumed no flow redistribution due to burnup yielded the most limiting LEU LOFA result, as explained in the last paragraph of Section B-6.2 of Reference 15. The smallest margin to the fuel temperature safety limit in this case, was a ΔT of 232 °C (418 °F), which occurred in element 8, which is an end-of-life element with 170 MWd of burnup, in axial node 3 of plate 23. It is noted that the peak fuel temperature in this element, 171°C (339 °F), occurred in axial node 3 of plate 22 at 0.34 s after the start of the transient. The locations of the peak fuel temperature and the smallest margin to the fuel temperature safety limit are different because of differences in the peak fuel burnup in each plate, which affects the burnup-specific safety limit. Figure 3 shows the peak fuel temperature history at the node of the peak fuel temperature.

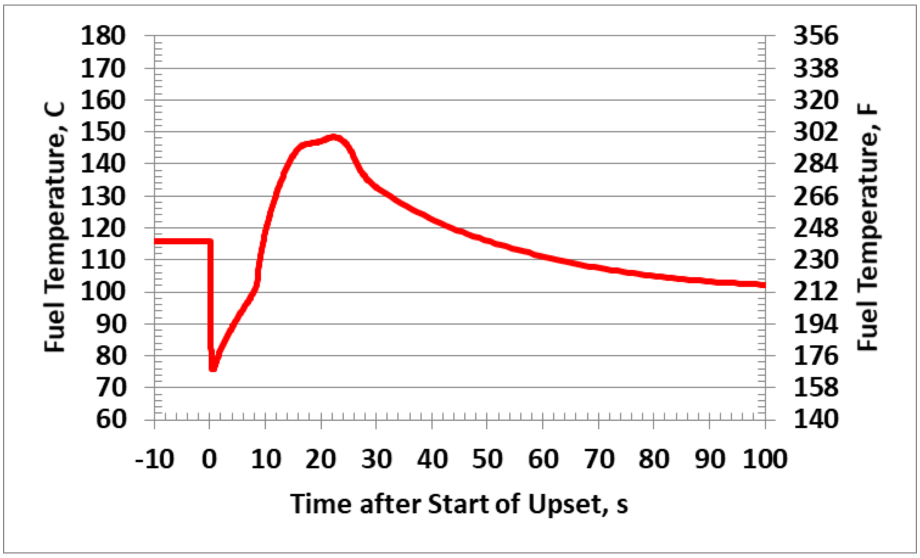

Figure 2. Peak Fuel Centerline Temperature during a Loss of Flow Accident – HEU Core

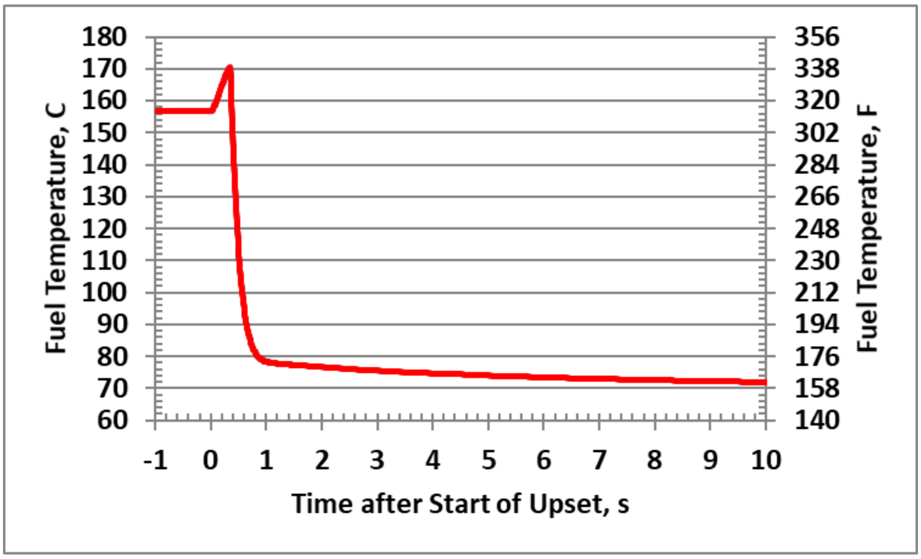

Figure 3. Peak Fuel Centerline Temperature during a Loss of Flow Accident – LEU Core

4. Conclusions

Safety analyses of the MURR core fueled with a monolithic alloy of uranium and 10 wt.% molybdenum (U-10Mo) were conducted to establish the steady-state safety basis of the reactor using a proposed LEU fuel element design and a power uprate from 10 MW to 12 MW. The thermal-hydraulic analyses of the reactor required detailed power distributions and geometrical considerations in order to identify regions of interest for postulated RIAs, LOCAs, and LOFAs. These accidents were analyzed at initial conditions established during relicensing of the HEU core and consistent with the guidelines in NUREG-1537. The minimum calculated margin to the fuel temperature safety limit occurs in a RIA, and remains below the safety limit by a margin of at least 109 °C for all accidents. Thus, the results of the analyses showed that the MURR fueled with LEU can withstand these postulated accidents without damage to the fuel.

Finally, the dose in the unrestricted area following the MHA, which is a postulated fuel handling accident for the LEU fuel, is 0.0153 mrem. This is well below the regulatory annual limit of 100 mrem for members of the general public.

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