Transition Core Planning and Safety Analyses in Support of LEU Fuel Conversion of the University of Missouri Research Reactor (MURR)

Nuclear Science & Engineering Division

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Transition Core Planning and Safety Analyses in Support of LEU Fuel Conversion of the University of Missouri Research Reactor (MURR)

prepared by

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Acronyms and Abbreviations

BOC	beginning-of-cycle
BOL	beginning-of-life
CHF/CHFR	critical heat flux/critical heat flux ratio
DDE	design demonstration element tested as an experiment in a test reactor
DOE	U.S. Department of Energy
EOC	end-of-cycle
EOL	end-of-life
HEU	highly enriched uranium with ≥ 20 weight% enrichment
LCO	limiting conditions for operation
LEU	low-enriched uranium with < 20 weight% enrichment
LSSS	limiting safety system settings
LTA	lead test assembly
M ³	NNSA Office of Material Management and Minimization
MOL	middle-of-life
MURR®	University of Missouri Research Reactor located in Columbia, Missouri
MW	megawatts thermal
NNSA	U.S. National Nuclear Security Administration
NRC	U.S. Nuclear Regulatory Commission
OFI	onset of flow instability
SAR	Safety Analysis Report
TS	Technical Specification
U-10Mo	Uranium – 10 wt% molybdenum alloy fuel being developed as a monolithic metallic alloy fuel
USHPRR	U.S. high-performance research reactor

Definition of Terms

Best estimate	Parameter value that is determined with the best available methods and models without including uncertainty.
Bounding parameter	Parameter value that has been technically determined to not be exceeded under given conditions, such as, for example, normal operating conditions.
Conservative	Method, or resulting parameter value, that is not best estimate and includes uncertainty or margin whether discretionary or due to conservative assumptions.
Conservative assumption	Parameter value or other consideration technically determined to be conservative as a consequence of intentional addition of margin or the use bounding data or other consideration.
Fission density	Number of nuclear fission events of all isotopes per unit volume of fuel.
Fuel cycle	Collection of all elements that are in active use at MURR from their time of initial insertion in the reactor core to reaching the fuel element discharge burnup.
Fuel qualification	Process of designing, conducting, and evaluating experiments to assure that the fuel is capable of performing without failure during reactor operations up to reported performance limits. Fuel qualification also includes measurements and reporting of fuel properties that can be used in performance and safety modeling.
Irradiation conditions	Conditions occurring in a reactor during operation at power and that, in the context of fuel design and qualification, refer to conditions experienced in fuel in a reactor driver position or in a fueled test article.
Licensed power	Power approved by the regulatory authority up to which the reactor can operate. For a reactor without a defined licensed duration this term is taken to be the equivalent of approved operating power.
Limiting conditions for operation (LCO)	Administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility. The LCOs are the lowest functional capability or performance level required for safe operation of the facility.
Limiting safety system settings (LSSS)	Limiting values for settings of the safety channels by which point protective action must be initiated. The LSSSs are chosen so that automatic protective action will terminate the abnormal situation before a safety limit is reached. The calculation of the LSSS shall include the process uncertainty, the overall measurement uncertainty, and transient phenomena of the process instrumentation.
Nominal value	Value of a parameter under normal operating conditions.
Normal operating conditions	Conditions that are reasonably expected to occur up to and including at the reactor licensed power. These do not include calculation or measurement uncertainties.
Operating cycle	Period of MURR operation from startup to shutdown with a group of eight elements selected by the reactor operator.
Power density	Heat generated by fission reactions deposited per unit fuel volume.

Prototypic condition	Conditions that are considered representative of normal operating conditions and matching key aspects of the fuel design geometry.
Prototypic fuel	Fuel produced with a commercial process consistent with the final specifications used for fabrication of fuel plates and elements.
Reactor design parameter	Best-estimate value from reactor analysis used as a basis in experiment design for fuel qualification and licensing tests. Each reactor stakeholder in RC Pillar activities identifies and documents reactor design parameter values.
Safety Analysis Report (SAR)	Report that describes a nuclear facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components of the facility as a whole.
Safety limit	Restriction or range placed upon important process variables that are necessary to reasonably protect the integrity of the physical barriers that guard against the uncontrolled release of radioactivity.
Safety margin	Difference between a regulatory limit and an appropriate value, that is either calculated or measured. Calculated values may include uncertainties to increase the conservatism of the safety margin.
Silicide fuel	Uranium silicide-aluminum (U_3Si_2 -Al) dispersion fuel.
Steady-state	Conditions that can be persistent and are not transient.
Target test value	Goal value based on a reactor design parameter to be achieved during testing, such as during an irradiation experiment to support fuel qualification or fuel demonstration. The Fuel Qualification (FQ) Pillar identifies and documents target test values in collaboration with other Pillars based on the reactor design parameters.
Technical Specification	Definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. Technical Specifications on reactor operations are approved by the regulator.

Executive Summary

The University of Missouri Research Reactor (MURR®) is one of six U.S. High Performance Research Reactors (USHPRR), including one critical facility, that is working with the National Nuclear Security Administration (NNSA) Office of Material Management and Minimization (M³) Reactor Conversion Program to convert from highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel. The M³ Reactor Conversion USHPRR Project objectives include the development of LEU fuel element designs that will ensure safe reactor operations and to maintain the existing experimental performance of each facility. The work is being conducted through many inter-related activities being completed by four Project Pillars: Fuel Qualification (FQ), Fuel Fabrication (FF), Reactor Conversion (RC), and Cross Cutting (CC).

A new type of LEU fuel based on an alloy of uranium-10 wt% molybdenum (U-10Mo) is expected to allow the conversion of those USHPRR, like MURR, requiring higher density fuels. The very-high-density LEU U-10Mo monolithic fuel is currently undergoing irradiation testing and post-irradiation examination under a planned and documented fuel qualification effort. The FQ Pillar will document fuel property and fuel performance data and qualify the fuel for use in these reactors. The FF Pillar is fabricating fuel for ongoing and future irradiation tests, as well as conducting fabrication demonstrations to validate or update preliminary fabrication assumptions. The FF Pillar is also working to develop and install commercial manufacturing capacity with the U-10Mo monolithic fuel to produce prototypic fuel.

Working with the RC Pillar at Argonne, MURR has progressed through a preliminary fuel element design using preliminary data for the proposed monolithic alloy of U-10Mo. Analyses were completed in previous work that found for typical equilibrium operations with the preliminary LEU fuel element design, in conjunction with a power uprate to 12 MW and appropriate changes to the MURR Limiting safety system settings (LSSS), MURR will have adequate margins to safety for steady-state operations and postulated transient accidents and will have experimental performance in key locations that meets or exceeds current operations with HEU fuel.

The purpose of this work is to develop a sequence of transition cycles that will enable MURR to transition from operation with the reactor core loaded with fresh LEU fuel elements only to typical equilibrium operations with mixed-burnup cores following conversion while meeting operational requirements on safety and experimental performance. It is expected that the use of fresh LEU fuel at conversion and subsequent low burnup of the LEU fuel elements that will initially be available for use following conversion will result in critical control blade positions that will substantially change the axial power distribution in the core and the neutron flux available in key experimental locations relative to equilibrium LEU operations.

Given the constraints of MURR safety margins, operational practices, and production and research, a novel method has been developed to identify a transition sequence that minimizes the time MURR operates atypically compared to the current prototypic cycles using HEU fuel. The proposed transition sequence moves quickly to the same sort of equilibrium cycles for the LEU fuel that have already been evaluated in documented preliminary safety analyses. Although shifting the neutron flux peak to the lower half of the core during initial cycles with LEU at 12 MW reduces the experiment performance in some key locations relative to current HEU operations at 10 MW, all LEU cores provide an average performance that meets or exceeds that of HEU. An LEU cycle is reached that meets or exceeds the level of experimental performance predicted for current HEU and equilibrium LEU operations in more than 450 key locations identified by a reactor specialist at MURR by the 23rd cycle.

following conversion and that afterwards will enable MURR to consistently meet its experimental performance requirements.

The proposed transition sequence only requires the fabrication of 34 fresh LEU elements in the first year of operation and does not exceed the anticipated availability of fresh elements that can be produced by the fuel fabricator. By the third year after conversion, 22 fresh LEU elements will be required each year, which is the same as expected for equilibrium LEU operations and the same as current operations with HEU fuel. The proposed transition sequence thus combines a relatively short time period before equilibrium burnup is achieved, a temporary increase of fuel elements needed annually relative to typical operations that are within the production capabilities of the fuel fabricator, and demonstrates comparable experimental performance of the LEU cores relative to current HEU operations. Further measures may be taken to reduce any initial experimental performance penalty even further, where possible, by repositioning certain experiments to leverage the increased performance in the lower axial experimental positions in the initial cycles following conversion or leaving the experiments in the irradiation facilities longer in order to achieve the required neutron fluence. This analysis may require refinement depending on the experimental facilities in use at the time of conversion. Nonetheless, the results presented here, including the experimental performance, core burnup, and critical control blade positions throughout the transition cycles, show that the proposed transition cycle fuel management patterns are consistent with what is expected and desired for MURR operation with LEU U-10Mo fuel.

Detailed core power distributions from the neutronics models were also used to evaluate safety margins during steady-state operations for the selected transition cycles and the equilibrium LEU core. It is shown that there are adequate safety margins for both steady-state operations and postulated accident scenarios. For the steady-state operations with the preliminary LEU fuel element design the analysis predicts at least 2.49 MW margin to the onset of flow instability at the LSSS power of 15 MW. Considering the LSSS power is 125% of full license power, the margin to OFI is sufficient. In addition, the critical heat flux ratio at LSSS power is well above the requirement of $CHFR > 2.0$ from NUREG-1537 for all considered cases. For postulated transient accidents, the minimum margin to the fuel temperature safety limit is at least 109 °C.

In summary, the proposed sequence of core loadings for MURR operations following conversion to LEU fuel and a power uprate to 12 MW provides sufficient safety margins for both steady-state operations and postulated transient accidents during a proposed sequence of transition cycles to equilibrium operations. Analysis has shown that there are some local experimental performance penalties during the initial cycles. Although there are local shifts in the experimental performance, on average all LEU cores at 12 MW have equal or higher performance than HEU at 10 MW. Temporary adjustments are being planned that will produce suitable experimental performance during these cycles. The results indicate that for the equilibrium LEU core the experimental performance exceeds that of current HEU operations in all key locations while also demonstrating sufficient safety margins.

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1 INTRODUCTION

The international non-proliferation community aims to minimize the amount of highly enriched uranium (HEU) that could be diverted and used for nuclear weapons. In this context, most research and test reactors, both domestic and international, have started a program of conversion to the use of low-enriched uranium (LEU) fuel. For decades, the U.S. Department of Energy (DOE) National Nuclear Security Administration (NNSA) Office of Material Management and Minimization (M³) Reactor Conversion Program has sponsored efforts to develop and qualify fuel, design and fabricate fuel elements, model reactor operating cycles and reactor operations, perform reactor safety analyses, maintain experimental performance and irradiation services, and support regulatory submissions to meet the challenges of reactor conversion [1]. To date, more than 70 reactors worldwide, including 20 in the U. S., have completed conversion to LEU in cooperation with the M³ Conversion Program [2,3].

The University of Missouri-Columbia Research Reactor (MURR[®]) located in Columbia, Missouri is one of six U.S. high performance research reactors (USHPRR), including one critical facility, that is actively collaborating with the M³ Conversion Program to find a suitable LEU fuel to replace the current HEU fuel. A new type of very high density LEU fuel based on an alloy of uranium and 10 weight percent molybdenum (U-10Mo) is expected to allow conversion to LEU of USHPRR [1] that have been found to be unable to convert with previously qualified uranium silicide-aluminum (U₃Si₂-Al) dispersion fuel [4].

Working with the NNSA M³ Conversion Program USHPRR Project Reactor Conversion (RC) Pillar led by Argonne National Laboratory, MURR has completed performance and safety analyses for prototypic equilibrium fuel cycle operations with the current HEU fuel and following conversion to this LEU fuel and a power uprate from 10 MW to 12 MW [5]. Performance and safety analyses have also been completed for HEU and for the preliminary LEU fuel element design [6] which demonstrate satisfactory experimental performance and margins to safety following a major facility upgrade [7].

The very high-density monolithic fuel is currently undergoing irradiation and post-irradiation testing under a planned and documented fuel qualification effort under the direction of the Fuel Qualification (FQ) Pillar to document fuel properties and fuel performance data and qualify the fuel for use in these reactors [8]. The Fuel Fabrication (FF) Pillar is fabricating fuel for ongoing and future irradiation tests, as well as conducting fabrication demonstrations to validate or update preliminary fabrication assumptions. The FF Pillar is also working to develop and install commercial manufacturing capacity with the U-10Mo monolithic fuel. The Cross-Cutting (CC) Pillar address cross-program activities including, but not limited to, transport and back-end planning.

Presented in this report is the analysis of a crucial step in the conversion process for MURR, which is the sequence of transition cores from HEU to equilibrium LEU operations. MURR typically operates with a relatively fixed total core burnup using a mixture of eight elements ranging from at or near beginning-of-life (BOL) to near or at end-of-life (EOL). This approach provides consistent and predictable experimental performance in all irradiation positions in the facility. At conversion from HEU to LEU fuel, MURR will operate with only fresh (unirradiated) LEU elements, which will likely require a brief period of adjustment to the typical position and/or loading of experiments in order to fulfill needed irradiation services. These mitigation strategies, which are currently being understood and developed, will continue throughout the transition cores beginning with the initial insertion of eight fresh LEU elements to operations where there are a sufficient number and variety of partially depleted LEU elements available to allow MURR to operate in a manner that is typical for current HEU

operations. Given the constraints of MURR operation and experiments, a transition sequence with LEU has been developed from a number of candidates that aimed to limit the period during which MURR operates atypically compared to the prototypic cycles currently run with HEU fuel while meeting all operational and safety requirements.

Section 2 of this report describes the approach for analysis of transition cores and presents a novel methodology that has been prepared to develop candidate transition sequences from all-fresh LEU cores to MURR's typical operating conditions. The aim of the transition sequence is to minimize the number of cycles during which MURR will operate atypically in order to fulfill the needs of its customers for research and irradiation services. This is crucial to maintaining MURR's mission as a leading supplier of radioisotopes and radiopharmaceuticals. A proposed sequence of transition cores and annual fresh fuel requirements that satisfy constraints imposed by the reactor operator is also provided. Detailed core neutron physics models and results are presented in Section 3, including predicted reactor design parameters during the operating period leading up to, and including, prototypic equilibrium operations with LEU. Also presented are evaluations throughout the transition sequence of nearly 500 experimental performance metrics, identified by specialists at MURR, which are key to meeting the scientific mission of the facility. The results of thermal hydraulics analysis of the transition cores for steady-state operations and transient accident conditions are presented in Sections 4 and 5. Comparisons of the calculated safety margins for a reference equilibrium LEU core are also provided. All safety margins for the transition cores were found to be larger than, or equivalent to, those for equilibrium LEU operations. The conclusions of this study and important findings are summarized in Section 6.

2 APPROACH AND METHODOLOGY FOR ANALYSIS OF TRANSITION CORES

MURR is a multi-disciplinary research and education facility located near the campus of the University of Missouri-Columbia. Operations with HEU fuel began more than 50 years ago, and for more than 40 years the reactor has operated on a weekly schedule at about 150 hours per week to provide predictable and stable irradiation services.

The reactor itself is a light water moderated and cooled heterogeneous reactor with beryllium and graphite reflectors. MURR is currently licensed by the NRC to operate at a total core power level of 10 MW, making it the highest-powered university-operated research reactor in the United States. The core contains eight fuel elements which are placed in an annular ring inside a pressure vessel. During the brief weekly shutdown, the eight fuel elements in the core are removed and replaced with elements from either the active fuel cycle inventory or fresh elements. Elements are used in about 19 operating cycles before being discharged from the fuel cycle as spent fuel and not suitable for irradiation in any further fuel cycles.

The reactor and laboratory facilities provide a broad range of analytical, materials science, and irradiation services to the research community and the commercial sector. MURR processes and delivers more than 2,000 shipments of radioisotopes and radiopharmaceuticals each year from its laboratories. Consequently, demonstrating adequate safety margins and maintaining existing experimental performance are key to the regulatory and operational decisions to convert to LEU.

MURR began collaboration with the USHPRR conversion project in 2006 to address unique challenges in converting to LEU fuel. Much of the insight gained from UHSPRR work activities informs the decisions made by MURR in its fuel conversion effort. A conversion feasibility design based on U-10Mo monolithic LEU fuel was completed in 2009 [9].

The proposed LEU element design was subsequently optimized based on feedback from the FF Pillar and the fuel fabricator [10]. The optimized LEU fuel element design retains the same element and plate configuration as HEU, but with a decrease of the number of fuel plates per element (23 with LEU vs. 24 with HEU). For this analysis, 70 performance measures identified by specialists were evaluated using tallies in the MCNP5 (Monte Carlo N-Particle transport code) [11] models for the HEU and LEU reference cores at beginning-of-cycle (BOC) and equilibrium xenon conditions. These metrics were based on irradiation facilities and experiment loadings that were typical for operations at MURR in 2008. A detailed comparison of all experimental performance metrics for HEU and LEU reference cores at comparable conditions found that a minimum reactor power of 11.6 MW with LEU will be necessary in order to maintain the same performance as HEU at 10 MW for all performance measures. Consequently, a power uprate to 12 MW for LEU, along with changes to the reactor limiting safety systems settings (LSSS) for the minimum core coolant flow rate and maximum coolant inlet temperature, was assumed for safety analyses with the preliminary LEU fuel element design [6]. These analyses confirmed that for equilibrium operations with the proposed LEU fuel design, a core power increase, and changes to the MURR Technical Specifications, MURR will maintain sufficient safety margins during steady-state and transient conditions, operating cycle length, and level and spectrum of key neutron fluxes and reaction rates to meet the scientific mission of the facility [5,7,12,13].

Work has continued toward conversion of MURR with a particular interest at this stage in identifying the impacts of conversion during the sequence of transition cores from fresh LEU to equilibrium LEU

operations. These impacts may include both safety margins and experimental performance. During the MURR operating cycle, the positioning of the control blades needed to compensate for the core excess reactivity and maintain a critical condition affects the axial and radial neutron flux distribution in the core and in the various experimental locations in the reactor. With HEU fuel in a typical MURR operating cycle, the critical control blade position for the core with no xenon (at startup) is approximately 17 inches withdrawn (out of a full blade travel length of 26 inches), and approximately 23 inches withdrawn at equilibrium xenon. The excess reactivity of a core fully loaded with fresh LEU fuel elements will require the control blades to be more deeply inserted than what is typical, such that the neutron flux will be compressed toward the bottom of the core.

The impact of such changes to the core power distribution on predicted safety margins during the transition cycles must be evaluated and understood. It is likewise expected that, owing to the deeper insertion of the control blades, experimental performance in irradiation positions above and below the core midplane will be substantially affected during the transition cycles relative to the typical MURR HEU operating cycle. As a result, work has been performed to identify strategies to mitigate reductions in experimental performance at key locations relative to current HEU operations while also maintaining adequate safety margins.

2.1 Previous Transition Cycle Analysis

In previous analysis of transition cycles from all-fresh to equilibrium LEU operations [14], transition cycles were evaluated for the same set of irradiation facilities and experiment loading as modeled in the preliminary Safety Analysis Report (SAR) [5]. The use of LEU “transition elements” with the addition of boron as a burnable absorber in the aluminum alloy 6061 (AA6061) side plates were considered as a mechanism for reactivity hold-down in the initial cores in order for MURR to operate with control blade positions that are similar to prototypic HEU operations. Other than this change to the side plate material, the analyses assumed that the side plates, end fittings, and fuel plates of the LEU transition elements could be fabricated to be dimensionally the same as the preliminary LEU fuel element design [6]. It was found through analysis that borated side plates with approximately 1,300 ppm boron in the transition LEU fuel elements would result in critical control blade positions for an all-fresh LEU core that are nearly the same as for current HEU operations.

Using a fuel management scheme for the transition cycles developed and proposed by an experienced reactor operator, it was found that during the first year of operation 12 transition LEU fuel elements (elements with borated side plates) would be needed along with 24 standard LEU fuel elements (elements like the preliminary LEU fuel element design with no borated side plates). Thus, a total of 36 fresh LEU fuel elements would be needed during the first year after conversion compared with 22 fresh fuel elements (either HEU or LEU) needed annually during typical equilibrium operations [5]. It was also found that it would take 49 cycles of operations (nearly one year) with these transition elements before MURR would operate with an equilibrium core burnup and no transition elements.

Core k_{eff} calculations predicted that with this strategy the core excess reactivity during the first 10 cycles following conversion was about 0.75% $\Delta k/k$ greater than a reference prototypic core. This would mean that the critical control blade position during these initial cycles with specialized transition elements would be about 1.5 inches lower than typical operations. This would cause a slight decrease of the experimental performance relative to the reference core in locations above the core midplane. The magnitude of the performance penalties was not evaluated with detailed analysis. From the tenth cycle after conversion until there were no more specialized transition elements in use (the 50th cycle), the difference in the core excess reactivity relative to the reference core was only about 0.01% $\Delta k/k$. Thus, during much of the transition to equilibrium LEU operations with the

specialized transition elements, the critical control blade positions, and thus the experimental performance, would be expected to be similar to the reference core.

Safety margins for cores loaded with the LEU transition elements during this transition period were not calculated explicitly. However, it was observed that because of the presence of the boron absorber in the side plates of the transition element, the most important power peaking factors affecting safety margins along the transverse edge of the fuel were reduced relative to those for equilibrium LEU cores. Consequently, given the adequate safety margins found through analysis of the equilibrium LEU core, and the reduced power peaking factors for the transition cores, it was judged that there would be adequate margin to safety for the cores loaded with the LEU transition elements, pending confirmation from future detailed analysis.

2.2 Approach for the Current Analysis of Transition Cycles

Subsequent to the transition core sequence with transition elements proposed in Reference 14, a determination was made to simplify the USHPRR material qualification effort and explore alternate approaches to the sequence of MURR transition cycles that would not require fabrication of special LEU transition fuel elements. The determination to explore alternative strategies, in conjunction with the need to perform more detailed analysis of safety margins and experimental performance for the transition cores, motivated the present work to develop and evaluate a proposed transition sequence that exclusively uses the preliminary LEU fuel element design.

In order to plan a sequence of transition cycles for MURR with LEU fuel, it is helpful to understand the current HEU fuel management scheme. A typical HEU core consists of eight elements with a range of burnups. The target for the total core burnup at BOC is 572 ± 20 MWd. This is a facility-specific practice that maintains appropriate excess reactivity in the core and thus allows the control blades to be withdrawn to an optimal position during the operating cycle which provides the ideal flux profile that MURR experimenters require. The ± 20 MWd range on the core burnup provides sufficient operational flexibility while also maintaining stable and predictable performance for MURR's irradiation services.

The operating time each week is approximately 6.3 days, with the rest of the week (approximately 18 hours) dedicated to Technical Specification surveillance tests, maintenance, refueling, and experiment change out. After the reactor is shutdown, all eight elements are removed from the core and placed in storage baskets in the reactor pool. These elements are not available for use in the next operating cycle due to the xenon content that is still present in the element at startup of the next cycle. Rather, the core loading is selected from the remaining elements in the active fuel cycle, which typically consists of about 32 elements with a range of burnup. HEU elements are typically discharged from the fuel cycle after achieving a total burnup of 150 MWd, which corresponds to approximately 19 cycles of operation for the element [5]. As elements are discharged, fresh elements are introduced into the fuel cycle and loaded in the reactor as a pair, and are kept paired together throughout their lifetime. MURR typically requires 22 new fuel elements from a fabricator each year to maintain its operations.

The LEU transition cycle fuel management in the current work was modeled so as to mimic the current HEU fuel management as much as possible. This is consistent with the approach used for the equilibrium LEU fuel cycle analysis completed to support the preliminary safety analysis [5]. In that work, it was found that a power uprate to 12 MW will be needed for equilibrium LEU operations to provide experimental performance that meets or exceeds the performance for typical HEU cores at 10 MW. A power uprate to 12 MW was also assumed for the transition cores.

In addition to fuel management considerations, experimental performance and safety margins for the LEU transition cores must also be evaluated. In order to assess the experimental performance during the transition cycles relative to current operations with HEU, 454 key experimental flux and reaction rate metrics (total and energy dependent) were identified by a reactor specialist at MURR. This represents a substantial increase in detail relative to the 70 metrics evaluated in preliminary SAR work [5]. These metrics, which are identified in the reactor cross-section shown in Figure 2.1, are based on irradiation facilities and experiment loadings that were typical for operations at MURR in 2018 and used in previous analyses of equilibrium LEU operations following a major prospective facility upgrade [7]. In the current work these performance metrics provide an opportunity to evaluate LEU transition cycles and the LEU core after typical equilibrium operating conditions are reached relative to values calculated for a typical HEU core. Detailed comparisons of LEU performance for selected transition cores and the equilibrium LEU core relative to HEU are provided in Section 3.2.4. Safety margin evaluations for selected cores in the transition sequence up to and including the equilibrium LEU core are provided in Sections 4 and 5.

Figure 2.1. Location and Type of Key Experimental Performance Metrics in MURR Reactor.

It was confirmed in previous work that for a prototypic equilibrium LEU core using elements with a mixture of burnup levels and operating at 12 MW, the experimental performance meets or exceeds the performance for a typical HEU core at 10 MW [5,7]. For the LEU core transition cycles, however, experimental performance that will enable MURR to maintain current irradiation services is expected to be a key challenge. In the first LEU transition cycle, MURR will operate with eight fresh LEU elements. Because of the excess reactivity of the core loaded with eight fresh elements, the control blades will be more deeply inserted than is typical for MURR operations with HEU, which substantially changes the axial shape of the neutron flux. This can be seen in Figure 2.2, which plots

the performance for the fresh LEU core relative to typical HEU operations for all key metrics by their axial position in the reactor. Even with a 20% uprate in power, the experimental performance is substantially reduced in the upper part of the reactor. It was found that for 183 metrics, mostly those above the core midplane, the experimental performance in the fresh LEU core is lower than that for the typical HEU core by as much as 34%.

Figure 2.2. MURR LEU Cycle 1 Total Cycle Performance at 12 MW Relative to Prototypic HEU Performance at 10 MW.

Consequently, the approach adopted in the current work is to develop a sequence of transition cycles that will arrive at LEU core burnup conditions that are comparable to LEU equilibrium cycle core burnup as quickly as possible during the transition to equilibrium operations to limit the duration of any experimental performance penalties following conversion. To accomplish this, some LEU elements will be used in back-to-back cycles. While this is not typical for current operations at MURR with HEU fuel, the LEU elements will initially have relatively low burnup and are expected to have enough excess reactivity to overcome the xenon poisoning at startup. Detailed predictions of the experimental performance for these cycles are described in Section 3.2.4.

This strategy, rather than attempting to mitigate the performance penalty at particular locations that may occur during the transition sequence with LEU relative to current HEU operations, was adopted after consultation with the reactor operator. MURR is exploring additional strategies to meet the requirements of experimenters while local performance losses exist during transition to equilibrium LEU operations. For example, it may be possible to leverage the increased performance in the lower section of the reactor by repositioning certain experiments to compensate for the lower flux in the upper section, or experiments could remain in the reactor longer in order to achieve the neutron fluence they require before discharge.

2.3 Methodology to Evaluate Candidate Transition Sequences

In this work, a methodology has been developed to prepare candidate sequences of transition cycles that move from fresh to equilibrium LEU operations with fuel management practices similar to those of typical HEU operations at MURR, with the exception of impacts related to the power uprate. In order to efficiently plan candidate transition sequences, a novel methodology using a Python code was developed which selected the optimal loading of LEU elements for a given cycle from an available fuel cycle inventory. Each candidate cycle was evaluated against the imposed constraints developed with the input of the reactor operator. Specifically, the Python code employed the following set of constraints:

1. LEU fuel elements are not used in back-to-back cycles to ensure a xenon-free condition at reactor startup, except as discussed below.
2. Fresh LEU fuel elements are introduced in pairs whenever optimal, except as noted below, up to a limit of 32 LEU elements in the active fuel inventory. This is consistent with current MURR operating practices that typically have 32 HEU elements in the active fuel inventory to allow for flexibility in cycle planning.
3. The target total LEU core burnup at BOC was $687 \text{ MWd} \pm 24 \text{ MWd}$. The target burnup for the LEU core is based on the reference equilibrium cores obtained from fuel cycle analyses performed in References 7 and 10, and confirmed in this work for a different experiment loading but similar fuel management. The reference LEU cores at 12 MW were found to preserve experimental performance measures and safety margins relative to prototypic HEU cores at 10 MW. The BOC core burnup range of $\pm 24 \text{ MWd}$ is selected as a 20% increase from the HEU target core burnup band of $\pm 20 \text{ MWd}$.
4. In order to ensure that the core will have sufficient excess reactivity for MURR's 6.3-day cycle, the core burnup was not allowed to exceed 711 MWd ($687 + 24$). The total core burnup was allowed to drop below 663 MWd ($687 - 24$) if no suitable collection of eight elements could be found that was within the designated band.
5. The LEU discharge burnup was assumed to be 180 MWd per element. This is the same as the LEU discharge burnup found in the equilibrium fuel cycle analysis completed in the preliminary SAR, and 20% greater than the HEU element discharge burnup of 150 MWd based on the 20% power uprate for LEU relative to HEU operations (10 MW to 12 MW).
6. A fabrication limit of 36 fresh LEU elements annually was assumed to allow flexibility in planning and to meet the needs of the reactor during the transition cycles. The assumed fabrication limit was based on input from the FF Pillar and is expected not to exceed the anticipated availability of fresh elements that can be produced by the fuel fabricator.

A more detailed description of the Python code can be found in Reference 15.

Items 1 and 2, above, place constraints on the loading of elements in the core that require certain exceptions during transition cycles. Notably, it is assumed that the eight LEU elements loaded in the first transition cycle following conversion will be re-used in each of the next five cycles of the transition sequence rather than removing the elements from the core to allow xenon to decay before re-use. This is subject to confirmation from detailed neutronics analysis that each cycle will have sufficient excess reactivity at startup with xenon present in the LEU elements. After the sixth transition cycle, elements will not be used in back-to-back cycles in order to be consistent with current operations. Due to this constraint, the seventh transition cycle was also loaded with eight fresh LEU elements just like the first cycle since there will be no previously irradiated xenon-free elements available for use. Whenever fresh elements are needed in any cycle after the seventh cycle of operations using LEU, two fresh elements are loaded.

The Python code written for this work did not perform any neutronics analysis. Instead, the burnup of each element was estimated by assuming equal power sharing among the eight elements in the core. For an LEU core power of 12 MW, this means that an element accumulates a burnup of 9.45 MWd ($= 12 \text{ MW} \div 8 \text{ elements} \times 6.3 \text{ days}$) for each cycle of use. This eliminated the need for completing detailed neutronics and depletion analysis for the initial evaluation of the candidate cycles. The validity/accuracy of this assumption was verified after the completion of neutronics analysis (see Section 3.2.1).

2.4 Proposed Transition Sequence for MURR Conversion

In the progressive development of this work, a number of alternate transition sequences with differing constraints and approaches were devised. Given the constraints of MURR operations and the goals of the USHPRR project as summarized in Section 2.3, above, a sequence of transition cycles was predicted by the Python code. Figure 2.3 provides an overview of estimated key parameters for the LEU operating cycles over the first two years following conversion of MURR if this sequence of transition cycles is adopted.

The estimated core burnup at BOC is shown to generally increase from 0 MWd for the first fresh core to near the target core LEU core burnup of 687 MWd. Based upon results obtained from previous work with LEU equilibrium cores [5,7], it is expected that cores at or near this burnup operating at 12 MW will yield experimental performance at all key locations that meets or exceeds the performance for typical HEU cores at 10 MW. The proposed sequence of transition cycles has cores within the target burnup band of 663 to 711 MWd beginning first in cycle 20. Beginning at cycle 30, the BOC burnup for the proposed LEU cores is quite stable and within the target band for a total of 30 cycles during the first year of operations and a total of 42 cycles during the second year of operations. It is expected that the anomalous cycles with a total core burnup < 663 MWd can likely be accommodated through alternate core loadings devised by the reactor operator or short-term adjustments to mitigate any expected experimental performance loss.

Figure 2.3 illustrates two other key parameters that must be considered for the MURR fuel management. First are the periodic needs for fresh LEU fuel elements to support operations. Those cycles in which fresh fuel elements are loaded are indicated by symbols that are either red- (eight fresh elements required) or yellow-filled (two fresh elements required). The timing of cycles with fresh fuel is an important consideration, particularly because of the planning required for fresh fuel procurement and delivery from the fabricator. Cycles 1 and 7 (red-filled symbols) will be loaded with eight fresh LEU elements. Cycles represented by a yellow-filled symbol in Figure 2.3 will require two fresh LEU elements.

MURR normally procures 22 fresh HEU elements annually. Since MURR will begin operations following conversion with LEU fuel only, a larger procurement rate of fresh LEU fuel elements will be needed initially. A constraint was applied in the Python code used to develop the proposed transition sequence so as to not exceed the expected availability of fresh fuel from the fabricator. The sequence proposed in this work remains within this constraint, using 34 fresh LEU elements in the first year, 30 fresh elements in year two, and 22 elements in year three. Thus, the proposed transition sequence will initially require a higher rate of fuel element fabrication than typical operations, but the fabrication requirements are within the availability anticipated by the fabricator. By the third year following conversion, it is expected that MURR will be operating with a fresh fuel shipment frequency similar to current operations. On average, two fresh fuel elements will be needed every three cycles

Figure 2.3. Core and Fuel Cycle Parameters During Proposed Transition Sequence for MURR.

during the first year of LEU operations, every four cycles during the second year, and every five cycles during year three and beyond.

Previous analysis of the transition cycles required 36 LEU fuel elements in the first year of LEU operations, 12 of which were specialized LEU transition elements with borated side plates [14]. Compared to the previous work, the transition sequence proposed here represents a two element savings to the program by requiring only 34 fuel elements to be fabricated for MURR during the first year of operations with LEU fuel. Furthermore, the proposed does not use any specialized elements which would require a material qualification program.

Figure 2.3 also shows the number of LEU fuel elements in the MURR fuel inventory. Elements are counted to be part of the available inventory in the first cycle in which they are loaded and up until they have reached the assumed discharge burnup of 180 MWd. MURR typically has 32 fuel elements in the active fuel cycle which are available for use in the core. It has been found through many years of operational experience that this number of elements gives adequate flexibility to plan each cycle at the facility, as well as keeping on hand a “backup” core that is occasionally needed in the event of unanticipated shutdown that may require a mid-cycle core reload. While this preferred operating condition with 32 elements in the fuel cycle is not reached until the 75th cycle of this proposed sequence, with a planning goal to reach the target core burnup as quickly as possible, the proposed sequence demonstrates the soonest that this condition can be reached. Tradeoffs could be explored in future planning, but the approach presented here demonstrates the viability of a transition sequence that requires no specialized elements with burnable absorbers.

In addition to planning the core loading strategy for transition cycles, the Python program was written to identify a location for each fuel element, whether in the core or pool storage, throughout its lifetime. This is useful for planning purposes, as well as to prepare model inputs for the detailed neutronics analysis that must be completed. Operators follow certain loading practices to limit power peaking in the core, support a uniform radial flux distribution for experiments, and optimize fuel element usage. These practices were incorporated in the Python program as best as possible,

although with some simplifications that are considered to have a negligible impact on the overall analysis.

As indicated above, fresh elements are loaded as a pair, and are kept paired together throughout their lifetime. It is the practice at MURR to load fresh and low burnup elements in the core positions identified as X1 and X5 in Figure 2.1, such that they are positioned on “opposite” sides of the core. As the burnup of the elements increases through middle-of-life (MOL) to EOL, they are loaded in other alternating locations. The algorithm was programmed so that with increasing burnup the elements were loaded in the X3 and X7 positions, followed by the X4 and X8 positions. Elements near or at EOL were placed in the X2 and X6 positions.

Following these practices, Table 2.1 through Table 2.4 were prepared for planning purposes to provide the details of the anticipated LEU fuel element loading for each cycle by quarter during the first year of the proposed transition sequence. This sequence of loading patterns was also used for detailed neutronics and depletion analysis discussed in Section 3.2. The MURR LEU elements will be identified by the fabricator by markings as MO-001, MO-002, etc. If the element is in the core, the proposed location is indicated in a yellow-shaded cell in the tables. A location identifier of “Out” means the element is not in use and is stored in one of the storage baskets in the reactor pool. When an element is predicted to reach its discharge burnup limit of 180 MWd, this is indicated by the entry of “DIS” after the last cycle of irradiation.

Lastly, it should be noted that the proposed transition sequence is based on certain assumptions, which are identified below.

1. The proposed transition sequence assumes equal power sharing among the elements in the core.
2. The proposed transition sequence assumes that there is sufficient excess reactivity available for reactor startup for each cycle and that there is enough reactivity in each core to support the 6.3-day operating cycle.
3. It must be confirmed that there are adequate safety margins for steady-state operations and transient accident conditions for the cycles in the proposed transition sequence.
4. Detailed neutronics analysis will be performed to calculate experiment performance metrics at key locations for selected LEU transition cores for comparison to typical operations with HEU. It is expected that this information will provide helpful guidance for specialists at MURR to devise additional strategies to mitigate performance reductions at particular locations.
5. The proposed transition sequence assumes an idealized series of cycles at MURR, with no shutdowns which might necessitate a mid-cycle refueling from the available inventory of fuel elements. These occasional unplanned shutdowns cannot be predicted at this time, but will be accommodated by the MURR operations staff at the time of their occurrence. Therefore, they will not be evaluated here.

Subject to confirmation of the validity of these assumptions which will be documented in the remainder of this report, this proposed transition sequence presents a strategy to accommodate the needs of MURR operations and experiment performance/isotope production following conversion, and is in keeping with the M³ conversion goal to maintain existing HEU operating practices at the facility wherever practical. While it is expected that there will be some temporary reductions in experimental performance at certain key locations for selected cycles of operation with LEU relative to current HEU operations, the duration of these reductions will be limited. Based on the estimated

core burnup, it is further anticipated that any performance reduction will generally diminish over the first 20 cycles of LEU operations, after which experimental performance for LEU, in conjunction with a power uprate from 10 to 12 MW, will meet or exceed current HEU performance metrics. Finally, specialists at MURR are working to devise additional strategies in order to continue to provide the kinds of irradiation services required by experimenters.

Given the constraints of MURR operation, experiments, and the USHPRR conversion program, this transition scheme minimizes the time MURR operates atypically compared to the prototypic cycles currently run with HEU fuel, and moves quickly to the same sort of equilibrium cycles for the LEU fuel. This transition scheme also requires the fabrication of 34 fresh LEU elements in the first year of operation, but does not exceed the anticipated availability of fresh elements that can be produced by the fuel fabricator. Combining the short time period before equilibrium burnup is achieved, the small number of fuel elements used, and the expected ability to remain near the ideal BOC burnup target, this transition scheme appears suitable for use when MURR converts to LEU fuel.

Table 2.1. First Quarter MURR LEU Fuel Element Loading for Proposed Transition Sequence.

Quarter Cycle	1												
	1	2	3	4	5	6	7	8	9	10	11	12	13
Element													
MO-001	X1	X1	X1	X1	X1	X1	Out	X2	Out	X3	Out	X3	Out
MO-002	X5	X5	X5	X5	X5	X5	Out	X6	Out	X7	Out	X7	Out
MO-003	X2	X2	X2	X2	X2	X2	Out	X4	Out	X4	Out	X4	Out
MO-004	X6	X6	X6	X6	X6	X6	Out	X8	Out	X8	Out	X8	Out
MO-005	X3	X3	X3	X3	X3	X3	Out	X3	Out	X2	Out	X2	Out
MO-006	X7	X7	X7	X7	X7	X7	Out	X7	Out	X6	Out	X6	Out
MO-007	X4	X4	X4	X4	X4	X4	Out	X1	Out	Out	X2	Out	X2
MO-008	X8	X8	X8	X8	X8	X8	Out	X5	Out	Out	X6	Out	X6
MO-009							X3	Out	X3	Out	X3	Out	X3
MO-010							X7	Out	X7	Out	X7	Out	X7
MO-011							X4	Out	X4	Out	X1	Out	X1
MO-012							X8	Out	X8	Out	X5	Out	X5
MO-013							X1	Out	X1	Out	X4	Out	X4
MO-014							X5	Out	X5	Out	X8	Out	X8
MO-015							X2	Out	X2	Out	Out	X1	Out
MO-016							X6	Out	X6	Out	Out	X5	Out
MO-017										X1	Out	Out	Out
MO-018										X5	Out	Out	Out
MO-019													
MO-020													
MO-021													
MO-022													
MO-023													
MO-024													
MO-025													
MO-026													
MO-027													
MO-028													
MO-029													
MO-030													
MO-031													
MO-032													
MO-033													
MO-034													

Legend:	
X1	Position element loaded in core
Out	Element out of core
DIS	Element discharged from fuel cycle

Table 2.2. Second Quarter Cycle MURR LEU Fuel Element Loading for Proposed Transition Sequence.

Quarter Cycle	2												
	14	15	16	17	18	19	20	21	22	23	24	25	26
Element													
MO-001	X3	Out	X2	Out	X2	Out	X3	Out	Out	X2	Out	X2	Out
MO-002	X7	Out	X6	Out	X6	Out	X7	Out	Out	X6	Out	X6	Out
MO-003	X4	Out	X4	Out	X4	Out	X4	Out	X4	Out	Out	X4	Out
MO-004	X8	Out	X8	Out	X8	Out	X8	Out	X8	Out	Out	X8	Out
MO-005	X2	Out	X3	Out	X3	Out	X2	Out	X2	Out	X2	Out	Out
MO-006	X6	Out	X7	Out	X7	Out	X6	Out	X6	Out	X6	Out	Out
MO-007	Out	X2	Out	X2	Out	X2	Out	X2	Out	Out	X4	Out	X2
MO-008	Out	X6	Out	X6	Out	X6	Out	X6	Out	Out	X8	Out	X6
MO-009	Out	X4	Out	X4	Out	X4	Out	X3	Out	X3	Out	Out	X3
MO-010	Out	X8	Out	X8	Out	X8	Out	X7	Out	X7	Out	Out	X7
MO-011	Out	Out	Out	X1	Out	Out	Out	Out	X3	Out	X3	Out	Out
MO-012	Out	Out	Out	X5	Out	Out	Out	Out	X7	Out	X7	Out	Out
MO-013	Out	X3	Out	X3	Out	X3	Out	X4	Out	X4	Out	Out	X4
MO-014	Out	X7	Out	X7	Out	X7	Out	X8	Out	X8	Out	Out	X8
MO-015	X1	Out	X1	Out	Out	X1	Out	X1	Out	X1	Out	X3	Out
MO-016	X5	Out	X5	Out	Out	X5	Out	X5	Out	X5	Out	X7	Out
MO-017	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out
MO-018	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out
MO-019		X1	Out	Out	X1	Out	Out	Out	X1	Out	X1	Out	X1
MO-020		X5	Out	Out	X5	Out	Out	Out	X5	Out	X5	Out	X5
MO-021							X1	Out	Out	Out	Out	Out	Out
MO-022							X5	Out	Out	Out	Out	Out	Out
MO-023												X1	Out
MO-024												X5	Out
MO-025													
MO-026													
MO-027													
MO-028													
MO-029													
MO-030													
MO-031													
MO-032													
MO-033													
MO-034													

Legend:

X1 Position element loaded in core

Out Element out of core

DIS Element discharged from fuel cycle

Table 2.3. Third Quarter Cycle MURR LEU Fuel Element Loading for Proposed Transition Sequence.

Quarter Cycle	3												
	27	28	29	30	31	32	33	34	35	36	37	38	39
Element													
MO-001	Out	X4	Out	Out	X2	Out	Out	Out	Out	Out	Out	Out	X2
MO-002	Out	X8	Out	Out	X6	Out	Out	Out	Out	Out	Out	Out	X6
MO-003	Out	X2	Out	X2	Out	Out	X2	Out	Out	X2	DIS		
MO-004	Out	X6	Out	X6	Out	Out	X6	Out	Out	X6	DIS		
MO-005	X2	Out	X2	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out
MO-006	X6	Out	X6	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out
MO-007	Out	Out	Out	Out	Out	X2	Out	Out	X2	Out	Out	X2	Out
MO-008	Out	Out	Out	Out	Out	X6	Out	Out	X6	Out	Out	X6	Out
MO-009	Out	Out	X4	Out	X4	Out	Out	X2	Out	Out	X2	Out	Out
MO-010	Out	Out	X8	Out	X8	Out	Out	X6	Out	Out	X6	Out	Out
MO-011	X3	Out	X3	Out	Out	X3	Out	X3	Out	Out	Out	X4	Out
MO-012	X7	Out	X7	Out	Out	X7	Out	X7	Out	Out	Out	X8	Out
MO-013	Out	Out	Out	X4	Out	X4	Out	X4	Out	X4	Out	Out	X4
MO-014	Out	Out	Out	X8	Out	X8	Out	X8	Out	X8	Out	Out	X8
MO-015	X4	Out	Out	X3	Out	Out	X4	Out	X4	Out	X4	Out	Out
MO-016	X8	Out	Out	X7	Out	Out	X8	Out	X8	Out	X8	Out	Out
MO-017	X1	Out	X1	Out	X1	Out	Out	Out	Out	Out	Out	X3	Out
MO-018	X5	Out	X5	Out	X5	Out	Out	Out	Out	Out	Out	X7	Out
MO-019	Out	X3	Out	Out	X3	Out	X3	Out	X3	Out	X3	Out	Out
MO-020	Out	X7	Out	Out	X7	Out	X7	Out	X7	Out	X7	Out	Out
MO-021	Out	Out	Out	Out	Out	X1	Out	Out	Out	X1	Out	Out	X3
MO-022	Out	Out	Out	Out	Out	X5	Out	Out	Out	X5	Out	Out	X7
MO-023	Out	X1	Out	Out	Out	Out	Out	X1	Out	X3	Out	X1	Out
MO-024	Out	X5	Out	Out	Out	Out	Out	X5	Out	X7	Out	X5	Out
MO-025				X1	Out	Out	X1	Out	Out	Out	Out	Out	X1
MO-026				X5	Out	Out	X5	Out	Out	Out	Out	Out	X5
MO-027									X1	Out	X1	Out	Out
MO-028									X5	Out	X5	Out	Out
MO-029													
MO-030													
MO-031													
MO-032													
MO-033													
MO-034													

Legend:	
X1	Position element loaded in core
Out	Element out of core
DIS	Element discharged from fuel cycle

Table 2.4. Fourth Quarter Cycle MURR LEU Fuel Element Loading for Proposed Transition Sequence.

Quarter Cycle	4												
	40	41	42	43	44	45	46	47	48	49	50	51	52
Element													
MO-001	Out	Out	Out	X2	DIS								
MO-002	Out	Out	Out	X6	DIS								
MO-003													
MO-004													
MO-005	Out	Out	Out	Out	X2	Out	Out	Out	Out	Out	Out	X2	DIS
MO-006	Out	Out	Out	Out	X6	Out	Out	Out	Out	Out	Out	X6	DIS
MO-007	Out	Out	Out	Out	Out	Out	Out	X2	DIS				
MO-008	Out	Out	Out	Out	Out	Out	Out	X6	DIS				
MO-009	Out	X4	Out	Out	Out	X4	Out	Out	Out	Out	X2	Out	X2
MO-010	Out	X8	Out	Out	Out	X8	Out	Out	Out	Out	X6	Out	X6
MO-011	X4	Out	Out	X4	Out	Out	X2	Out	Out	X2	Out	Out	Out
MO-012	X8	Out	Out	X8	Out	Out	X6	Out	Out	X6	Out	Out	Out
MO-013	Out	X2	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out	X4
MO-014	Out	X6	Out	Out	Out	Out	Out	Out	Out	Out	Out	Out	X8
MO-015	X2	Out	X2	Out	Out	X2	Out	Out	X2	Out	Out	Out	Out
MO-016	X6	Out	X6	Out	Out	X6	Out	Out	X6	Out	Out	Out	Out
MO-017	Out	Out	X1	Out	Out	Out	Out	X3	Out	X1	Out	X4	Out
MO-018	Out	Out	X5	Out	Out	Out	Out	X7	Out	X5	Out	X8	Out
MO-019	X3	Out	X4	Out	X4	Out	X4	Out	X4	Out	X4	Out	Out
MO-020	X7	Out	X8	Out	X8	Out	X8	Out	X8	Out	X8	Out	Out
MO-021	Out	X3	Out	Out	X3	Out	X3	Out	Out	X4	Out	Out	Out
MO-022	Out	X7	Out	Out	X7	Out	X7	Out	Out	X8	Out	Out	Out
MO-023	Out	Out	X3	Out	Out	Out	Out	X4	Out	X3	Out	X3	Out
MO-024	Out	Out	X7	Out	Out	Out	Out	X8	Out	X7	Out	X7	Out
MO-025	Out	X1	Out	X3	Out	X3	Out	X1	Out	Out	Out	Out	Out
MO-026	Out	X5	Out	X7	Out	X7	Out	X5	Out	Out	Out	Out	Out
MO-027	Out	Out	Out	Out	X1	Out	X1	Out	X3	Out	X3	Out	Out
MO-028	Out	Out	Out	Out	X5	Out	X5	Out	X7	Out	X7	Out	Out
MO-029	X1	Out	Out	X1	Out	Out	Out	Out	Out	Out	Out	X1	Out
MO-030	X5	Out	Out	X5	Out	Out	Out	Out	Out	Out	Out	X5	Out
MO-031						X1	Out	Out	X1	Out	Out	Out	X3
MO-032						X5	Out	Out	X5	Out	Out	Out	X7
MO-033											X1	Out	X1
MO-034											X5	Out	X5

Legend:

X1	Position element loaded in core
Out	Element out of core
DIS	Element discharged from fuel cycle

3 CORE NEUTRON PHYSICS ANALYSIS

3.1 Neutronics Methods and Modeling

Simulations of the transition cycles and a reference equilibrium fuel cycle were accomplished using two neutronics analysis codes. For the fuel cycle simulation with detailed fuel management and fuel depletion, the software REBUS-PC v. 1.5 was used [16]. As one solution option, the user may select the DIF3D multi-dimensional, multi-group neutron diffusion code to quickly simulate the neutronic environment for each state point of the transition sequence. Deterministic diffusion codes generally execute much faster than Monte Carlo neutronics codes, which is preferred for calculating the fuel burnup over the many state points needed for simulating multiple operating cycles. Furthermore, the θ -R-z geometry option in DIF3D utilizing broad-group cross sections for the fuel, fission products, and structural components prepared with WIMS-ANL [17] was found to be well-suited for modeling the MURR fuel cycle with either HEU or LEU fuel and to provide good agreement for predictions of the core reactivity compared to higher-fidelity MCNP5 [11] models [5,10]. The same library of broad-group cross-section data was utilized in the current analysis since there was no change to the nominal HEU or LEU fuel element dimensions or fuel plate loadings.

For the simulation of the transition cycles and the equilibrium cycle, it was necessary to model MURR within the limits of the DIF3D geometry capabilities. Some of the more significant simplifications in the DIF3D model of MURR are: homogenized experiments in the experimental positions, homogenization of the fuel and cladding, and banking of the control blades at a position of 58.42 cm (23 inch), which is the typical position of the control blades at equilibrium xenon conditions.

While the DIF3D model is not used for detailed critical power distributions, the model must still accurately represent the neutronic environment at each state point for the calculation of the fuel element compositions during use. It was found that even with the simplified geometry of the DIF3D model, the predicted k_{eff} is in good agreement with a more detailed MCNP5 [11] model. For example, Table 3.1 compares the k_{eff} from the DIF3D model with homogenized experiments and a more detailed MCNP5 model for a reference core in Reference 7 with fresh LEU fuel and the control blades at a position of 58.4 cm (23.0 inch). Likewise, comparisons of DIF3D and MCNP5 models have demonstrated differences of less than 1% for predictions of the element powers in fresh HEU and fresh LEU cores [7]. Comparisons of the axial power profiles in limiting plates have also demonstrated good agreement between simplified DIF3D models and more explicit MCNP5 models; differences of less than 2.5% were observed for the peak power density in the fuel plates [7]. Finally, a database of depleted HEU fuel element compositions generated by REBUS-DIF3D is routinely used at MURR for predicting BOC and hot-startup estimated critical control blade positions [18].

Table 3.1. Comparison of DIF3D and MCNP5 Models for MURR with Fresh LEU Fuel and Control Blade Position of 58.4 cm (23.0 inch).

DIF3D	MCNP5	Difference
1.07919	1.07914 $\pm$ 0.00008	0.0043% $\Delta k/k$

Prior to the current analysis for the MURR transition cores, significant work was done to update the HEU and LEU REBUS-DIF3D models from the experiment loading used in the preliminary SAR analysis based on conditions in 2008 to the experiment conditions typical for MURR operations in 2018 [7]. There were four significant modifications to the experiment layout from 2008 to 2018:

1. The installation of the Flux-trap Irradiations Reactivity Safety Trip (FIRST) device in the Center Flux Trap (CFT). This modification enabled the loading of a larger combined reactivity worth for the flux trap experiment holder plus experiments in the CFT.
2. A single 60° wedge (“combined GHJ wedge”) replaced two 30° wedges in the graphite reflector in support of silicon irradiation.
3. Two additional irradiation positions in the “G6” graphite wedge
4. New experiment loadings in the “G3” and “G4” graphite wedges

These facility modifications are shown in the models depicted in Figure 3.1. More details on the experiment loading are provided in Reference 7.

Figure 3.1. 2008 and 2018 MURR Core Models.

In order to model the experiment facilities in the DIF3D model for MURR at 2018 conditions, a sequence of calculations comparing DIF3D to an explicit MCNP5 model was used. First, a clean core base case with fresh fuel and no experiments was established for both core models. Then, experiments in different locations were sequentially modeled with MCNP5 in explicit detail. The reactivity change of each modification to the MCNP5 model was calculated by comparison to the prior case. Since DIF3D cannot model the core experiments in detail, boron was homogenized in each experiment position with a modeling criterion that the predicted core eigenvalue change matched the *a priori* result from MCNP5 within 0.01% $\Delta k/k$.

Models for HEU and LEU cores were developed. The HEU core model was used to establish a reference equilibrium core that is prototypic for current operations as a basis for experiment performance comparisons. The LEU core model was used for predicting experimental performance and completing safety analyses for the sequence of transition cores and the equilibrium core that are expected following conversion to LEU.

Table 3.2 provides a summary of the series of MCNP and DIF3D cases that were completed to develop the HEU MURR_2018 models. The modifications to the MURR experimental facilities implemented between 2008 and 2018 result in a 0.587% $\Delta k/k$ increase in the predicted HEU core excess reactivity (based on fresh fuel and control blades at 58.4 cm). The agreement between the DIF3D and MCNP5 models for the HEU core with the experiments that were typical for 2018 is 0.159% $\Delta k/k$. These models form the basis for the experimental performance comparisons completed in this work or the transition and equilibrium cores loaded with LEU.

Table 3.2. Preparation of HEU MURR_2018 DIF3D Model.

Case	Description	MCNP				DIF3D			Diff. DIF3D vs. MCNP (pcm ^a)
		k-eff	σ (pcm ^a)	Experiment Reactivity Worth (pcm ^a)		k-eff	Experiment Reactivity Worth (pcm ^a)		
				Ind.	Total		Ind.	Total	
0	Base case. All-fresh HEU fuel. Blades at 23" withdrawn; reg blade at 10" withdrawn. Empty holder in flux trap. Graphite-filled aluminum boxes with no experiment positions (other than beam tubes) in all wedge positions. Remove hanging wedge.	1.10158	5			1.10362			168
2	MCNP5. Change L and N wedges in graphite reflector to fast-flux irradiation facilities.	1.09312	5	-703	-703				
2a	DIF3D. Fast-flux irradiation facilities installed. Smear 873 ppm boron into graphite.					1.09516	-700	-700	170
4	MCNP5. Change combined GHJ and hanging wedges to 2018 conditions.	1.08517	5	-670	-1373				
4e	DIF3D. 664 ppm boron added to graphite of combined GHJ wedge.					1.08718	-671	-1370	170
5	MCNP5. Change 3-wedge to 2018 conditions.	1.08446	5	-60	-1433				
5a	DIF3D. 55 ppm boron added to graphite of 3-wedge.					1.08646	-61	-1431	170
6	MCNP5. Change 4-wedge to 2018 conditions.	1.08383	5	-54	-1487				
6b	DIF3D. 90 ppm boron added to graphite of 4-wedge.					1.08584	-53	-1484	171
7	MCNP5. Change 6-wedge to 2018 conditions.	1.08321	5	-53	-1540				
7a	DIF3D. 90 ppm boron added to graphite of 6-wedge.					1.08509	-63	-1547	160
8-3	MCNP5. Change center test hole between axial planes -7.493 to +12.327 cm from empty-holder to 2018 conditions.	1.08759	5	+372	-1168				
8-3c	DIF3D. Aluminum spacers with 106 ppm boron in center test hole sample positions between axial mesh 25-33 (roughly equiv. to MCNP axial positions -7.5 to +12.3 cm)					1.08948	+371	-1176	160
8-4	MCNP5. Change entire holder to 2018 conditions.	1.09054	5	+249	-919				
8-4c	DIF3D. 113 ppm boron smeared in center test hole sample positions between axial mesh 10-25. 113 ppm boron in center test hole sample positions between axial mesh 33-38.					1.09423	+248	-928	159

^a Percent-mille. 1 pcm = 0.001% $\Delta k/k$.

Table 3.3 provides a summary of the series of MCNP and DIF3D cases that were completed to develop the MURR_2018 models used for evaluation of the transition and equilibrium cores with LEU. The agreement between the DIF3D and MCNP5 models for the LEU core with the experiments that were used for typical operations in 2018 is -0.038% $\Delta k/k$. A comparison of results presented in Table 3.2 and Table 3.3 shows that the reactivity worth of the experiments for the LEU fuel is similar to that for the HEU fuel, with differences of less than 0.1% $\Delta k/k$ for experiments in specific locations and less than 0.02% $\Delta k/k$ for all experiments that are typically loaded. The predicted excess reactivity of the fresh LEU core (based on control blades at 58.4 cm) is 1.504% $\Delta k/k$ lower than that for the fresh HEU core loaded with the same experiments. Subsequent depletion analyses have found that for typical equilibrium cores loaded with fuel elements that have a mixture of burnup, the difference in the predicted excess reactivity for LEU relative to HEU is not as large because of the production of Pu-239 in the LEU fuel. For the equilibrium LEU core, the excess reactivity is only 0.310% $\Delta k/k$ lower than that for the equilibrium HEU core.

The results presented in Table 3.2 for HEU and Table 3.3 for LEU show that the MURR_2018 DIF3D models for HEU and LEU cores with all experiments modeled closely match the excess reactivity predicted by the more detailed MCNP5 models. Thus, the DIF3D models are considered suitable for depletion analyses.

In the simulations performed as part of this work, the REBUS-PC code used the neutron flux solution computed by DIF3D to calculate the compositions of the depleted fuel elements which are loaded in the reactor core for each cycle. A predictor-corrector method was employed to improve the accuracy of the neutron flux solution and depletion analysis over the time step. The fuel depletion analysis used a transmutation chain with the same set of nuclides and broad-group cross sections as in previous work [7,10], which included eleven actinides, four fission products, and a lumped fission product to account for the reactivity effects of depletion. REBUS-PC can also model radioactive isotope decay during the time the elements are out of the reactor, which is important for modeling the effects of xenon poisoning. The fuel elements were depleted on plate-by-plate basis, with 12 axial zones per plate, resulting in 276 compositions per fuel element in the model. To limit the number of depletion zones and improve computational efficiency, the DIF3D flux solution and REBUS-PC depletion neglected lateral peaking in the fuel plates. This will have only a small impact on the depletion analysis. It was found in previous work for equilibrium cores at MURR [7,10] that lateral peaking is small ($\sim 10\%$) relative to axial peaking ($\sim 60\%$) for the most limiting plates. The same trend was observed in the current work. In addition, for the initial transition cores which will have lower burnup/excess reactivity and critical blade positions that are more deeply inserted than typical equilibrium operations at MURR, the axial power peaking has been found to be much greater relative to the lateral peaking. Lateral power peaking is included in the detailed three-dimensional power profiles calculated by MCNP5 for the safety analysis. Finally, the control blades were modeled at a fixed position of 58.4 cm (23.0 inch) throughout the depletion analysis, which is the modeling approach employed in previous work [7,10]. While the critical control blade positions are adjusted throughout the weekly operating cycle to compensate for the buildup of fission products and fuel depletion, this control blade position is typical for most of the MURR weekly operating cycle. As discussed above, the critical control blade positions during the initial transition cores will be more deeply inserted. The impact of the critical control blade position on fuel burnup on a local level, which is a key reactor design parameter, is discussed in more detail in Section 3.2.5. By using this model, the fuel element burnup from a sequence of operating cycles was calculated with minimal computational effort and acceptable accuracy.

Table 3.3. Preparation of LEU MURR_2018 DIF3D Model.

Case	Description	MCNP				DIF3D			Diff. DIF3D vs. MCNP (pcm ^a)
		k-eff	σ (pcm ^a)	Experiment Reactivity Worth (pcm ^a)		k-eff	Experiment Reactivity Worth (pcm ^a)		
				Ind.	Total		Ind.	Total	
0	Base case. All-fresh HEU fuel. Blades at 23" withdrawn; reg blade at 10" withdrawn. Empty holder in flux trap. Graphite-filled aluminum boxes with no experiment positions (other than beam tubes) in all wedge positions. Remove hanging wedge.	1.08344	5			1.08294			-42
2	MCNP5. Change L and N wedges in graphite reflector to fast-flux irradiation facilities.	1.07577	5	-658	-658				
2a	DIF3D. Fast-flux irradiation facilities installed. Smear 900 ppm boron into graphite.					1.07528	-658	-658	-42
4	MCNP5. Change combined GHIJ and hanging wedges to 2018 conditions.	1.06860	5	-624	-1282				
4e	DIF3D. 593 ppm boron added to graphite of combined GHIJ wedge.					1.06815	-621	-1279	-40
5	MCNP5. Change 3-wedge to 2018 conditions.	1.06796	5	-56	-1338				
5a	DIF3D. 55 ppm boron added to graphite of 3-wedge.					1.06749	-57	-1337	-41
6	MCNP5. Change 4-wedge to 2018 conditions.	1.06742	5	-47	-1385				
6b	DIF3D. 80 ppm boron added to graphite of 4-wedge.					1.06699	-44	-1381	-38
7	MCNP5. Change 6-wedge to 2018 conditions.	1.06678	5	-56	-1441				
7a	DIF3D. 90 ppm boron added to graphite of 6-wedge.					1.06631	-59	-1440	-41
8-3	MCNP5. Change center test hole between axial planes -7.493 to +12.327 cm from empty-holder to 2018 conditions.	1.07062	5	+336	-1105				
8-3c	DIF3D. Aluminum spacers with 106 ppm boron in center test hole sample positions between axial mesh 25-33 (roughly equiv. to MCNP axial positions -7.5 to +12.3 cm)					1.07018	+339	-1102	-39
8-4	MCNP5. Change entire holder to 2018 conditions.	1.07294	5	+202	-903				
8-4c	DIF3D. 112 ppm boron smeared in center test hole sample positions between axial mesh 10-25. 112 ppm boron in center test hole sample positions between axial mesh 33-38.					1.07250	+202	-899	-38

^a Percent-mille. 1 pcm = 0.001% $\Delta k/k$.

Following the REBUS-PC analysis, more detailed evaluations of the core power distributions and experimental performance metrics during the transition cycles was completed using MCNP5. This was done by taking the composition data by plate and axial level computed by REBUS-PC for each fuel element in a given core to prepare a detailed MCNP5 model. The MCNP5 model also contained the full detailed description of specific experiments that were typical for operations at MURR in 2018. This model was used to calculate the critical control blade positions, detailed three-dimensional power distributions for safety analyses, and experimental performance measures for comparison to typical operations with HEU. A Python script was written to automate the critical control blade position search for each cycle of interest. This program adjusted the control blades based on predictions until the k_{eff} calculated by MCNP5 was within 0.005% $\Delta k/k$ of 1.0. Once the calculated k_{eff} fell in this range, the reactor was assumed to be critical.

3.2 Neutronics Analysis

3.2.1 Neutronics Depletion Simulation of Transition and Equilibrium LEU Cycles

A transition sequence proposed for MURR operations following conversion to LEU was prepared using a Python program that incorporated constraints identified by a senior reactor operator. This sequence is described in Section 2.4. The REBUS-PC software [16] was used to complete a neutronics depletion simulation of this sequence with the MURR_2018 DIF3D model described in the previous section. The cycle-by-cycle fuel management pattern was modeled for the first 49 cycles of weekly operations. As illustrated in Figure 2.3, this is more than sufficient to establish a core loading condition that is typical for MURR operations. From Cycle 30 throughout the remainder of the first year of the proposed sequence of operations the core burnup is within the target band of $687 \pm \text{MWd}$.

An alternate set of simulations in REBUS-PC, similar to what was completed as part of the preliminary SAR [5], was used to identify reference HEU and LEU equilibrium cores. The fuel management employed in the simulation of several MURR “pseudo-equilibrium” cycles was planned by an experienced reactor engineer at MURR. The fuel loading pattern for each cycle was designed “by hand” based on the engineer’s many years of experience and employed many of the same principles and constraints as those coded into the Python program in this work. The bases for the HEU and LEU simulations were the MURR_2018 DIF3D models which are documented by Table 3.2 and Table 3.3, respectively. A reference equilibrium HEU core identified by the selection of one of the pseudo-equilibrium cycles was used in a subsequent MCNP5 calculation to establish values for a set of 454 performance metrics in key locations. These performance values were normalized to a reactor power of 10 MW and provided a point of comparison to evaluate LEU cores in this work. Safety margins for steady-state and transient accident conditions in the reference equilibrium LEU core were used as a point of comparison for the safety margins of the LEU transition cores.

Figure 3.2 charts the core burnup and eigenvalue predicted by REBUS-PC for the LEU transition cycles derived from the Python program (displayed as chart “a”). Alongside the results of the transition cycles are results for a number of the pseudo-equilibrium cycles from the alternate simulation used to identify the reference equilibrium LEU core (displayed as chart “b”). Both simulations reach roughly the same equilibrium state, but it is observed that the simulation based on core loadings developed by the Python program provides a sequence of cycles that has much less variability upon reaching the equilibrium state. At one level, this comparison provides a partial validation of the Python program to predict a sequence of core loadings that closely matches the practical experience of a reactor engineer at MURR. Furthermore, the comparison illustrates the value of the Python

program to efficiently and effectively identify an optimized core loading sequence given a set of identified constraints that are considered most relevant for MURR to maintain its mission as a stable provider of irradiation services. It took only seconds for the Python program to design a multi-year sequence of operating cycles, compared with many hours of careful planning by an experienced engineer.

Figure 3.2. Results of MURR LEU Core Neutronics Depletion Simulations with REBUS-PC.

The control blade position in the DIF3D calculations was fixed at 58.4 cm (23.0 inch), causing the predicted core k_{eff} to vary by a wide amount during the simulation of each operating cycle. In actual operations, the control blade positions would be adjusted throughout the cycle to maintain $k_{\text{eff}} = 1.0$. For those cores that have a BOC burnup in the desired range of $687 \pm \text{MWd}$, the core k_{eff} ranges from ~ 1.035 to ~ 0.998 from BOC to EOC. While the predicted EOC core k_{eff} is less than 1.0, the control blades were modeled at 88% of their fully withdraw position of 66.0 cm (26.0 inch), so further withdrawal of the control blades is possible to ensure that MURR can meet its typical 6.3-day operating cycle length.

The reference equilibrium LEU core selected from among the set of pseudo-equilibrium cycles is identified in chart b of Figure 3.2. This core was selected as a reference core for the safety margin analysis based on an inspection of the plate heat flux values for the pseudo-equilibrium cores predicted by REBUS-PC, as illustrated in Figure 3.3. The maximum heat flux was selected as a metric for the safety margins in steady-state and transient accident conditions, with larger values for the heat flux implying a smaller safety margin. For each pseudo-equilibrium cycle, the maximum plate surface heat flux in the core is plotted by the core burnup at four state points in the cycle: BOC, 1 day, 2 days (assumed to be equilibrium xenon conditions), and EOC (6.3 days). The heat flux is predicted by DIF3D for each plate in the core at 12 axial levels, but the model does not include lateral peaking and the control blades are fixed at a position of 58.4 cm (23.0 inch) throughout the cycle. Nonetheless, all predictions are at similar conditions, so this is considered a valid comparison for identifying the most limiting core for more detailed safety analyses which rely on detailed three-dimensional power distributions predicted by an explicit MCNP5 model that includes lateral peaking and critical control blade positions.

Figure 3.3. Maximum Heat Flux in LEU Pseudo-Equilibrium Cores Predicted by REBUS-PC Neutronics Depletion Simulation.

Each of the pseudo-equilibrium cycles in the neutronics depletion simulation has a core that is loaded with fuel elements with a mixture of burnup that is typical for MURR operations. Each cycle generally has two elements at or near BOL, four elements in the middle of their operating life, and two elements at or near EOL. It is observed in Figure 3.3 that those cycles that are loaded with two fresh fuel elements generally have a larger maximum heat flux, and so are expected to have smaller safety margins compared with other cores. Among these, the cycle identified as the “Reference equilibrium LEU cycle” in Figure 3.3 (and also in Figure 3.2) has the largest local heat flux predicted by DIF3D at each state point in the cycle. For the current work, the BOC, eq. xenon, and EOC core state points for this cycle are identified as 7X, 8X, and 9X, respectively. The total burnup of the eight elements in this reference core with typical fuel management and experiments that are typical for 2018 is 687 MWd, which matches the equilibrium LEU cycle burnup from previous neutronics depletion simulations with the same preliminary LEU fuel element design and experiment loadings that were typical for MURR operations in 2008 [5] or with a major facility upgrade [7].

A similar approach was used to identify a reference equilibrium HEU cycle. For the HEU reference core, the total burnup of the elements in the core is 593 MWd, which is marginally outside the typical BOC burnup range for HEU of 572 ± 20 MWd. The reference core states at BOC, eq. xenon, and EOC are identified as 3X, 4X, and 0X, respectively. The safety analysis for the reference equilibrium HEU core will not be performed as part of this work, but prior work confirmed that the safety margins for steady-state operations with HEU (margin to onset of flow instability (OFI) at LSSS conditions) are maintained for equilibrium operations with the preliminary LEU fuel element design, with a corresponding change to the LSSS conditions for LEU [5,7]. Detailed analysis of the reference equilibrium HEU core at 10 MW is used to establish the experimental performance in key locations for comparison with LEU at 12 MW for transition cores and the reference equilibrium LEU core.

Finally, a vital assumption in the Python program used to identify the proposed transition sequence is that, no matter the burnup of the elements in the core, each element contributes an equal fraction of the total core power. In other words, of the 12 MW of the total core power, each element in the core was assumed to contribute 1.5 MW (one-eighth of 12 MW) in the Python program. This was a crucial assumption that provided a way to efficiently develop candidate transition sequences without the initial computational expense of neutronics analysis.

The neutronics depletion simulation of the proposed transition sequence completed using REBUS-PC calculated the power and burnup of each fuel element in the fuel cycle. The element burnup from this neutronics analysis was compared with that obtained from the equal power sharing assumption in the Python program. Figure 3.4 summarizes the difference in element burnup for each element utilized during the first 49 cycles of LEU operation (identified as MO-001 to MO-032). Results are presented for each cycle in which the LEU element is loaded in the core and show that the difference between the calculated element burnup and the burnup based on the assumption of equal power sharing is $< 4\%$ (small exceptions are noted for elements MO-021 and MO-029). It is further observed that for the earliest transition cycles which are loaded with elements that all have similar burnup, the difference between the estimated and actual burnup is smaller in magnitude and varies from $+3\%$ to -1% . The difference decreases as the elements are used over multiple cycles and is no more than 1.7% for an element that has reached a discharge burnup of 180 MWd, which is the burnup limit for a MURR LEU fuel element assumed in the analysis. This brief analysis verifies that the assumption of equal power sharing among the elements in the MURR core is reasonable for planning purposes.

Figure 3.4. Difference between Estimated and Actual BOC Burnup of LEU Elements used During First 49 Cycles of Operations.

3.2.2 Critical Control Blade Positions and Power Distributions

For the remaining evaluations of the proposed transition sequence, it is necessary to consider the critical control blade positions during the transition cycles because of the impact on safety margins and experimental performance. Reactivity control during the MURR weekly operating cycle is provided by four control blades and one stainless-steel regulating blade that move vertically in a channel located between the outer pressure vessel and beryllium reflector shown in Figure 2.1. The control blades contain BORAL® as a poison material with an AA6061 cladding that extends 0.32 cm (0.13 inch) below the bottom of the poison material. The most deeply inserted blade position measured from the bottom of the control blade cladding is 4.76 cm (1.88 inch) below the bottom of fuel. The poison material is a little over 86 cm (34 inch) long, so in the fully inserted position the poison material extends 20 cm (8 inch) above the fuel. The blades have a total travel length of 66 cm (26 inch), so that in the fully withdrawn position of the control blades the bottom of the blade is aligned with at the top of the fuel. The control blades are designated as A, B, C, and D and can move independently, but are usually ganged together at the same axial position. Fine adjustments of the core reactivity are provided by the stainless-steel regulating blade that is adjacent to core driver element position 5.

For the current work, the four control blades and the beryllium reflector surrounding the core were modeled at fresh conditions. While in actual operations these core components have some prior operating history, the assumptions of fresh control blades and beryllium were also employed in previous analyses for HEU and LEU cores [5,7]. The impact of prior control blade use on the critical control blade positions and the core safety margins is considered for analysis of selected cases in Section 4.2.

Calculations of the critical control blade positions were completed using an explicit MCNP5 model with the fuel compositions derived from the neutronics depletion simulation completed with REBUS-PC. Figure 3.5 shows the results of these predictions during several transition cycles and for the equilibrium LEU core. Predictions at BOC, equilibrium xenon (eq. Xe) conditions (assumed to exist by two days into the cycle), and end-of-cycle (EOC) (6.3 days) were completed. Note that although straight lines are used to connect the data points in the figure, the actual control blade movement with time will not be linear. The figure does show that, as expected, the critical control blade positions during the cycle are affected by both the core burnup and xenon-state of the core. It is observed, for example, that the critical control blade position at reactor startups with xenon-free conditions is more fully withdrawn as the burnup of the core increases. The control blades are further withdrawn throughout the remainder of the cycle, with the greatest change in the critical control blade position occurring during the build-in of xenon to equilibrium levels. There is a notable exception to this pattern for Cycle 6, which is discussed in more detail below.

It can be seen in Figure 3.5 that for most of the cores evaluated during the first 20 cycles of the transition sequence (results for Cycles 1, 6, 9, and 15 are shown), the critical control blade positions at startup and throughout the cycle are more deeply inserted (less withdrawn) than is typical for MURR equilibrium operations (identified as “Eq. LEU” in Figure 3.5). These blade positions suppress the neutron flux in the upper part of the reactor relative to typical operations while increasing the flux in the lower part of the reactor. As has been previously discussed, this will reduce the neutron flux available for experiments in the upper part of the reactor. Furthermore, the variability observed for the critical control blade positions at a given state point from cycle-to-cycle is not preferred for experiments, because this will cause a variation in the experimental performance in a given position, making it challenging to provide consistent flux levels in all experiment positions. This makes it difficult for researchers to predict their experimental performance in advance. Finally, the critical control blade position affects the core axial and radial power distribution (as discussed below), which can impact safety margins for steady-state and transient accident conditions.

Figure 3.5. Critical Control Blade Positions during Selected Transition Cycles.

However, for most cycles after Cycle 20, and virtually all cycles after Cycle 30, the core burnup is within the target burnup range for equilibrium operations. Consequently, the critical control blade positions at startup and throughout the cycle are similar to what is predicted for equilibrium LEU operations and for typical HEU operations. At Cycle 30, for example, the critical control blade positions at BOC, eq. Xe, and EOC are within 0.2 cm (0.1 inch) of those for the equilibrium LEU core and within 1.2 cm (0.5 inch) of those for the typical HEU core under the same assumptions of control blade and beryllium reflector burnup.

As noted above, Figure 3.5 illustrates one complication that must be briefly accommodated by MURR operations during the proposed transition sequence. The transition sequence was planned to use elements in consecutive cycles for the first six cycles. Following the brief shutdown interval between these cycles for maintenance, substantial xenon poisoning will exist in the elements at the next reactor startup. Consequently, as can be seen in Figure 3.5 for Cycle 6, the control blades are positioned about 18 cm (7 inch) higher than the control blade position of 43.2 cm (17.0 inch) that is typical at BOC (see the curve for the equilibrium LEU core) and then are gradually inserted as the core reaches eq. Xe conditions. While the approach of loading elements in consecutive cycles has some benefit of reducing the experimental performance penalties in certain key locations relative to using a collection of elements with lower burnup during the early transition cycles (see further discussion in Section 3.2.4), this represents an atypical pattern for MURR operators to accommodate during Cycles 2 through 6, which must be factored into planning and procedures. The analysis shows that there is sufficient excess reactivity for startup of the core even with xenon poisoning at the beginning of Cycle 6, but it is not feasible to load these same elements for a 7th cycle because there will not be enough excess reactivity for the reactor startup. Thus, after Cycle 6 of the proposed transition sequence, elements are never used in consecutive cycles, which is typical for normal operations at MURR.

The BORAL control blades used to compensate for the core excess reactivity also have a strong influence on the core power distribution. This can be observed in Figure 3.6, which shows the axial profile of the heat flux for plates 1 and 23 of the LEU fuel element design calculated using MCNP5 for the BOC condition of selected transition cycles and the reference equilibrium LEU core. The profiles presented for each plate are for a fresh LEU fuel element in the X1 position of the core. For these presentations, the heat flux is calculated as a “simple” heat flux by assuming that energy from fission deposited in the fuel, which has been normalized to a reactor power of 12 MW for LEU, conducts evenly out the two large radial surfaces of the fuel plate, neglecting any lateral heat conduction to the side plates. The heat flux values at each axial level in Figure 3.6 are averaged over the lateral dimension of the fuel width of the fuel plate; the lateral peaking at the edge of the fuel plate is similar for both plates 1 and 23. Finally, while the heat flux profiles for the plates in a fresh LEU element loaded in the X1 core position are shown, the trends are exhibited for the other fuel elements, as well.

Figure 3.6. Axial Heat Flux Profiles for Plates 1 and 23 of Fresh LEU Element at BOC of Selected Cycles.

As expected, the heat flux profile is more peaked towards the bottom of the fuel for those cores with a lower burnup which have a critical control blade position that is lower than typical operations in the reference equilibrium LEU core. This effect is most pronounced for plate 23, which is the outermost plate of the LEU fuel element design and located nearest to the control blades. The control blade position has a less pronounced impact on the axial heat flux profile for plate 1 since it is positioned furthest away from the blades. It is observed that for Cycle 30 of the transition sequence the axial heat flux profiles in both plates 1 and 23 are nearly the same as the corresponding plate heat flux profiles in the equilibrium LEU core.

Figure 3.7 presents the “hot-stripe” heat flux by plate for a fresh LEU element in core position X1 for selected transition cycles and the reference equilibrium LEU core at BOC. The hot-stripe heat flux is the axially-averaged heat flux in a narrow 0.5 cm (0.2 inch) stripe along the edge of the fuel zone in the plate. The width of this stripe was determined in previous work to yield conservative predictions of the heat flux for use in thermal hydraulics models without lateral heat conduction in the fuel plates [7]. The hot-stripe heat flux is especially relevant for predictions of the margin to OFI during steady-state operations. These profiles show the influence of the critical control blade position on the radial power distribution in the fuel element. With fresh LEU fuel only loaded in the core in Cycle 1, the control blades are most deeply positioned, which reduces the heat flux for the outermost plates of the fuel element. As the core burnup increases through successive transition cycles, the radial profile becomes similar to that for the reference equilibrium LEU core. As the steady-state core power at BOC is increased, it is expected that OFI will occur first in coolant channels adjacent to fuel plate 4 because of its greater hot-stripe heat flux.

Figure 3.8 presents the hot-stripe heat flux by plate for the same LEU element as in Figure 3.7, but at equilibrium xenon conditions of the cycles. Because of the decreased excess reactivity due to xenon poisoning, the critical control blade positions are more fully withdrawn than at xenon-free conditions at reactor startup. This substantially reduces the suppression of the hot-stripe heat flux in the outermost fuel plates, so that OFI is expected to occur first in coolant channels adjacent to fuel plate 23, and at lower reactor power levels for cores at equilibrium xenon and EOC conditions. Detailed analyses of the steady-state are described in Section 4.

3.2.3 Reference Transition Cycles for Thermal Hydraulics Analysis

Steady-state safety analyses are used to determine the margin to OFI and critical heat flux ratio (CHFR) at the LSSS settings for the inlet coolant temperature, core coolant flow rate, and the reactor power level (for CHFR only). Transient accident safety analyses involve determining the margin of the maximum temperature of the fuel to a burnup dependent fuel temperature safety limit (FTSL) during postulated accident scenarios following guidelines provided in NUREG-1537 [19,20].

Previous work evaluated the safety margins for reference equilibrium LEU cores loaded with the preliminary LEU fuel element design and with different experiment loading configurations than considered in this work [5,7]. Those prior analyses found that the steady-state safety margins for the LEU cores with a power uprate to 12 MW and a proposed change to the reactor LSSS settings are comparable to the margins for the current HEU core. For postulated transient accidents in the equilibrium LEU cores, the analyses predicted that there was adequate margin to a FTSL based on preliminary U-10Mo blister threshold data.

Predictions of the safety margins for the reference equilibrium LEU core with the experiment loadings typical for 2018 are evaluated as part of this work. Safety margins are also calculated for the transition cycles, but it is not considered necessary to evaluate the safety margins for every cycle in the proposed transition sequence. Rather, cycles that cover the range of core burnup from fresh LEU to equilibrium operations plus intermediate cycles that may have more limiting safety margins were selected for detailed analysis. These cycles were selected based on conditions that are expected to lead to decreased margin.

Figure 3.7. Hot-Stripe Heat Flux by Plate in Fresh LEU Element at BOC of Selected Cycles.

Figure 3.8. Hot-Stripe Heat Flux by Plate in Fresh LEU Element at Eq. Xenon of Selected Cycles.

Experience gained from previous work with the preliminary LEU fuel element design and the U-10Mo fuel system has identified two conditions that may lead to reduced margins during steady-state operations and transient accidents. First, fuel loading and/or operating conditions that result in increased power peaking tends to decrease safety margins, especially for transient cases. Greater power peaking causes an increase in the local heat flux, which leads to larger maximum fuel temperatures during postulated transient accident scenarios. As illustrated in Figure 3.6, the peak-to-average plate power is expected to be greater for cores with lower total burnup, especially in plate 23. Additionally, it has been observed that when there is a greater disparity in the burnup of elements loaded adjacent to each other the power peaking tends to be greater, leading to reduced safety margins. This condition exists, for example, when a fresh fuel element is loaded adjacent to an element near its EOL burnup. A second condition which has been found to result in decreased safety margins is when the burnup of individual fuel elements is increased. Increasing burnup reduces the coolant channel gap thickness between fuel plates because of the effects of irradiation-induced fuel swelling, irradiation-induced creep, and oxide layer growth on the surface of the fuel plates. The coolant channel thickness reduction decreases coolant flow in the channel because of flow redistribution, which may reduce the rate of heat removal and potentially decrease the CHFR and/or increase the fuel temperature. The decreased coolant flow rate may also increase the enthalpy deposited per unit mass of the coolant, which can cause a boiling crisis that may lead to OFI in the coolant channel and further reduce the CHFR. Fuel burnup also decreases the U-10Mo thermal conductivity, which causes an increase in the fuel temperature and reduced margins to the burnup dependent FTSL.

The proposed transition cycles were reviewed to identify particular cores for detailed steady-state and transient accident safety analyses. As previously discussed and shown in Figure 3.2, the BOC core burnup is near the target BOC burnup for the LEU equilibrium core for nearly all transition cycles following Cycle 30, indicating the end of the transition period. To determine which transition cycles in the period from Cycle 1 to Cycle 30 may have decreased safety margin, two criteria, which are shown in Figure 3.9, were evaluated for each cycle. These criteria are based on the insights gained from previous experience described above. In particular, Figure 3.9 plots the maximum burnup difference between adjacent elements and the total core burnup for each cycle. The transition cycles with lower core burnup are expected to have greater power peaking in plate 23 due to more deeply inserted critical control blade positions. Cycles with a larger burnup difference between adjacent elements are also expected to have greater power peaking, as well as greater fuel element burnup which will decrease the coolant channel thicknesses and increase the fuel temperature in the depleted elements. Out of the thirty cycles considered as part of the transition period, the eleven cycles identified by large red circles (and by the notation “C n”, where n is the cycle number) were selected for detailed safety analyses based on these criteria, as well as efforts to cover the ranges of core and element burnup conditions that will exist during the transition cycles. The reference equilibrium LEU core was also evaluated. The burnup conditions of the elements loaded in each of these cycles are presented in Table 3.4, along with the predicted critical control blade positions. The results of the safety analyses are presented in Sections 4 and 5.

Note that for the steady-state thermal hydraulics analyses, core states from BOC to EOC were evaluated for each cycle. While experience has found that the CHFR is lowest at BOC conditions when the blades are most deeply inserted and the peak local heat flux is greatest, the predicted margins to OFI are smaller at equilibrium xenon and EOC conditions due the effect of withdrawing the control blades on the hot-stripe heat flux (see Figure 3.7 and Figure 3.8). For postulated transient accidents, it is expected that the maximum fuel temperature is greatest, and the corresponding margin to the FTSL is lowest, at BOC when the control blades are more deeply inserted. This control blade

Figure 3.9. Maximum Burnup Difference between Adjacent Elements and Total Core Burnup for Transition Cycles and Reference Equilibrium LEU Core.

Table 3.4. Fuel Burnup Conditions at BOC for Transition Cycles and Reference Equilibrium LEU Core Selected for Evaluation in Steady-State and Transient Accident Analyses.

Cycle	Burnup, MWd ^a				Core Burnup, MWd	Maximum Difference of Burnup of Adjacent Elements, MWd	Critical Control Blade Position, cm (inch) ^b
	X1/X5	X2/X6	X3/X7	X4/X8			
1	0.0	0.0	0.0	0.0	0	0.0	32.0 (12.6)
9	9.3	9.5	9.5	9.5	76	0.3	34.4 (13.5)
11	19.0	66.3	19.0	18.6	246	47.6	37.0 (14.6)
15	0.0	85.0	37.7	38.1	322	85.2	37.4 (14.7)
17	37.7	94.3	47.2	47.6	454	56.8	39.8 (15.7)
19	47.5	103.6	56.8	57.2	530	56.5	41.0 (16.1)
22	19.2	122.9	47.1	123.0	624	103.9	42.2 (16.6)
23	66.3	121.7	76.3	75.9	680	55.5	43.0 (16.9)
27	9.6	141.5	66.3	85.1	605	132.1	41.7 (16.4)
28	9.7	141.6	47.8	140.3	679	132.3	42.9 (16.9)
30	0.0	150.9	94.6	95.0	681	151.1	42.3 (16.6)
Eq. LEU	0.0	96.0	77.0	170.4 ^c	687	170.5	42.3 (16.7)

^a Average burnup of elements in core positions noted at BOC. Elements are loaded in pairs throughout their lifetime and have similar burnup.

^b Control blades assumed to be fresh and are ganged together at the same axial position. Regulating blade position at 31.8 cm (12.5 inch).

^c For added conservatism in the accident analyses, this element modeled at burnup conditions representative of the EOC core state, or an element burnup of 180 MWd.

position results in greater local power peaking than occurs later in the same cycle when the control blades are withdrawn to compensate for the build-in of xenon. Consequently, for the postulated accidents, the analyses were conducted primarily only at BOC of the selected cycles, which is the approach also employed in previous work [5,7]. The EOC core states for two transition cycles were also evaluated to confirm that the BOC state is most limiting.

3.2.4 Experimental Performance during Transition Cycles

As stated in Section 2.2, 454 key experimental flux and reaction rate metrics at key locations were identified by reactor specialists at MURR to evaluate the performance of the proposed transition cycles and the equilibrium LEU core relative to HEU. These metrics, which are identified in Figure 2.1, were calculated for the LEU cores at 12 MW and compared with values calculated for a typical equilibrium HEU core at 10 MW. The flux or reaction rate tallies predicted by MCNP5 at each location were integrated over the 6.3-day operating cycle length to calculate cycle totals for the performance metrics. The fluence or reaction total values for selected LEU transition cycles were compared with those for the equilibrium HEU core to predict the performance penalties for all locations during the transition period. Figure 3.10 shows a summary of the calculated performance penalties for the 454 metrics for selected transition cycles over the first 23 cycles. While there are performance penalties relative to current HEU operations in some locations of 20% or more during the first 15 cycles of the proposed transition sequence, the penalties are generally diminishing. In Cycle 23 of the transition sequence, there is no penalty on performance for all 454 metrics. It is noted that although shifting the neutron flux peak to the lower half of the core during initial cycles with LEU at 12 MW reduces the experiment performance in some key locations relative to current HEU operations at 10 MW, all LEU cores provide an average performance that meets or exceeds that of HEU.

It is observed that Cycle 1 has the largest performance penalty of the six selected cycles, with over 100 metrics having a greater than 20% performance penalty relative to HEU performance. However, the relative performance of the LEU core for all metrics is not uniform throughout all locations. As seen in Figure 3.11 the performance losses are concentrated in the regions above the core midplane, which is due to the deep insertion of the control blades to compensate for the large excess reactivity in the core with fresh LEU elements. Below the core midplane, there is a significant experiment performance gain when compared to HEU because the flux is compressed to the bottom of the core.

One of the unique aspects of the proposed transition sequence was the determination, after consultation with the reactor operator, that the LEU elements initially loaded in the core could be burned in back-to-back cycles (with a brief 18-hour outage between cycles that is routine for MURR) to rapidly increase the core burnup. While this is not typical or feasible for current operations at MURR with HEU fuel, the LEU elements will initially have relatively low burnup and enough excess reactivity to overcome the xenon poisoning in these elements at startup. Preliminary analysis found that the LEU fuel element design can operate in this fashion for six back-to-back cycles. This results in critical control blade positions that will cause less compression of the neutron flux towards the bottom of the core. This can be seen in Figure 3.12, which shows the distribution of the relative performance metrics across all locations in the core during Cycle 6 of the transition sequence. There are still some performance losses at locations above the core midplane, but the maximum penalty is only 13%, compared to a 34% maximum penalty in Cycle 1. As shown in Figure 3.13 and Figure 3.14, there is no experimental performance penalty in any of the key locations for Cycle 23 and the reference equilibrium LEU core, respectively. All 454 experiment metrics meet or exceed the experiment performance relative to HEU.

Performance Penalty of LEU at 12 MW Relative to Prototypic HEU at 10 MW

Figure 3.10. Experimental Performance Penalties in Key Locations during Transition Cycles.

Figure 3.11. Experiment Performance for LEU Cycle 1 Relative to Prototypic HEU Core.

Figure 3.12. Experiment Performance for LEU Cycle 6 Relative to Prototypic HEU.

Figure 3.13. Experiment Performance for LEU Cycle 23 Relative to Prototypic HEU Core.

Figure 3.14. Experiment Performance for Reference Equilibrium LEU Core Relative to Prototypic HEU Core.

The experimental performance penalty in certain key locations in the LEU transition cycles relative to a prototypic HEU core has been found to be as much as 34% in Cycle 1 with fresh LEU fuel (which will also exist in Cycle 7). However, the maximum performance penalties in these locations generally diminish throughout the transition sequence. This can be seen in Figure 3.15(a), which shows the maximum performance penalty relative to the prototypic HEU core for all key locations in the LEU transition cycles. For all cycles after Cycle 20, the performance penalty in any location is < 5%; after Cycle 30, and for the prototypic equilibrium core, no performance penalty is expected in any location. Figure 3.15(b) also presents the LEU performance at 12 MW relative to HEU performance at 10 MW as an average of all locations. Considered in this way, it is observed that the average experimental performance for all locations in the LEU cores with the power uprate to 12 MW exceeds the performance for the current HEU core at 10 MW by ~5%. The average was calculated by a simple averaging all values without any weighting. This indicates that with some planning, performance losses may be mitigated during the brief period of the transition cycles following conversion.

Figure 3.15. Largest LEU Experimental Performance Penalty in All Locations (a) and Average LEU Performance for All Locations (b) in Transition Cycles Relative to Prototypic HEU Core.

As expected, the performance penalties, when they do exist, have been found to be concentrated in locations above the core midplane. In order to mitigate performance penalties above the core midplane, two main strategies will be employed by experimenters at MURR. First, for experiments that utilize the center flux trap region, samples can be shifted lower in the experiment holder in order to take advantage of the performance gain in the lower elevations of the core. However, because of the limited space in the flux trap, there is still likely to be some performance loss at times for experiments. The second mitigation strategy is simply leaving experiments in the reactor for longer durations of time. Many experiments in the graphite reflector region require a specified fluence instead of a flux target and reach this fluence target usually before the end of the cycle. By simply leaving these experiments in the reactor longer, the desired fluence target can still be met, such that there is no net performance loss. Conversely, some experiments may reach their fluence target sooner if they are below the core midplane. In this scenario, these experiments can be withdrawn from the reactor, leaving higher flux areas for other experiments to be positioned in these vacated areas.

3.2.5 Reactor Physics Parameters

The MURR Technical Specifications (TS) place requirements on reactor physics parameters that are considered relevant for safety [21]. Certain physics parameters were calculated for selected transition cores and the equilibrium LEU core for initial confirmation that the cores loaded for operations following conversion to LEU will satisfy the Technical Specifications. At the time of

conversion, a series of startup tests and predictions will be performed to ensure the Technical Specifications are met [22].

Technical Specification 3.1(b) requires that the cold minimum shutdown margin be greater than 2% $\Delta k/k$. The shutdown margin was calculated for the fresh LEU core loaded in Cycle 1 and the equilibrium LEU core for comparison against this requirement. The Cycle 1 core is expected to be the most limiting with regard to shutdown margin because it will have fresh fuel only. The following assumptions were used in the calculations:

- All reactor materials were at isothermal conditions at 38 °C (100 °F). This is the typical pool water temperature and is 17 °C (30 °F) less than the typical primary coolant temperature at normal full power.
- The most reactive control blade and the regulating blade were in the fully withdrawn position of 66.0 cm (26.0 inch); the other three control blades were in a fully inserted position of 0.0 cm (0.0 inch).
- The beryllium reflector was conservatively assumed to be at fresh conditions without any prior operating history.
- The core was at xenon-free conditions.
- The experiment loading in the center flux trap experiment holder and the graphite reflector were typical for operations at MURR in 2018.

Calculations were performed for the control blades at two conditions for their prior operating history. For the base case, the control blades were assumed to be at fresh conditions with no prior operating history. In normal operations, the BORAL control blades are typically resident for about 2 to 8 years, with an average life of about 4 years, before being retired. During this time, the boron-10 in the bottom few inches of the control blade poison section has been observed to deplete significantly, as this portion of the blade almost always resides in the range of the active core height. The control blades are also shuffled to different positions to balance the effect of burnup for the control blades that are in use on the core power distribution. As a conservative analysis of the shutdown margin, a case with all four control blades at their maximum life of 8 years was considered. Note that MURR never operates with all four control blades at their maximum burnup. The composition of the materials in the control blades was calculated in previous work [23].

Table 3.5 summarizes the calculations of the LEU core cold shutdown margin. The effect of the extreme blade depletion reduces the shutdown margin by 0.7% $\Delta k/k$ for both the fresh LEU and equilibrium LEU cores. It can be seen, nonetheless, that the shutdown margin is > 2.0% $\Delta k/k$ even for the extreme case with the fresh LEU core loaded in Cycle 1 and all four BORAL control blades at their maximum burnup. A larger shutdown margin is observed for the equilibrium LEU core which has some previously irradiated fuel elements and an initial total core burnup of 687 MWd.

Technical Specification 3.1(b)(2) also requires that the minimum margin be determined with “irradiation facilities and experiments in place and the total worth of all experiments in their most reactive state.” The experiment loading assumed in this analysis is typical for typical operations at MURR in 2018.

Table 3.3 showed that the worth of the experiments loaded in the experiment holder in the center flux trap in this configuration is 0.538% $\Delta k/k$. Technical Specification 3.8(b) limits the maximum worth of all experiments in the center flux trap to an absolute value of 0.6% $\Delta k/k$. If the worth of the experiments loaded in the center flux trap was increased to this amount, the cold minimum shutdown margin is estimated to be 2.30% $\Delta k/k$, which meets the Technical Specification 3.1(b) requirement.

Table 3.5. Cold Shutdown Margin and Cold Excess Reactivity for LEU Transition Cycles and Equilibrium Core.

	Cycle 1 (Fresh LEU Core)	Equilibrium LEU Core
Cold Minimum Shut Down Margin ($\Delta k/k$) ^a [TS: > 2.0% $\Delta k/k$]		
All Control Blades Fresh	3.04%	6.74%
All Control Blades 8-years depleted	2.36%	6.00%
Cold Excess Reactivity ($\Delta k/k$) [TS: > 9.8% $\Delta k/k$]		
All Control Blades Fresh	7.68%	4.30%
All Control Blades 8-years depleted	7.89%	4.53%

^a Shutdown margin based on regulating blade and highest worth control blade "stuck out." Found through analysis of all blade combinations to be blade "C".

Table 3.5 also provides the results of the calculation of the cold core excess reactivity. Technical Specification 3.1(a) requires that the cold excess reactivity of the MURR core be less than 9.8% $\Delta k/k$ as additional assurance that the shutdown margin specification is met. The cold excess reactivity was calculated under the same conditions as the shutdown margin, except that all four control blades were positioned at 66.0 cm (26.0 inch) and the regulating blade was positioned at 25.4 cm (10.0 inch). Even for the extreme case of the fresh LEU core loaded in Cycle 1 and the control blades at their maximum burnup that the core excess reactivity requirement is met.

The kinetics parameters and reactivity coefficients for the LEU cores are needed for the reactivity insertion accident (RIA) analysis that was performed in Section 5.3. The core physics parameters needed for the accident analysis were calculated using MCNP5 for selected LEU cores at BOC (xenon-free), eq. Xe, and EOC core states. To cover the range of conditions expected throughout the transition sequence, the core physics parameters were calculated for Cycle 1 (fresh LEU), Cycle 6 (378 MWd core burnup), and the reference equilibrium LEU core (687 MWd core burnup). The results are summarized in Table 3.6. For each parameter, the values presented in Table 3.6 are those which are most conservative for accident conditions, which are the minimum values for the effective delayed neutron fraction (β_{eff}) and the prompt neutron lifetime (Λ) and the least negative (smallest absolute value) for the coolant void, coolant temperature, and fuel temperature coefficients of reactivity. The coolant temperature coefficients reported in Table 3.6 were calculated by accounting for only the temperature change of the primary coolant without any corresponding change to the coolant density. This was done because the PARET/ANL software [24] used for the RIA analysis assumes a separability of the coolant temperature and density changes when calculating the reactivity feedback from transient accidents. For the analyses of a RIA in the transition cycles and the equilibrium LEU core completed in Section 5.3, additional conservatism was introduced by using the most conservative values for the core physics parameters among all the selected cycles and core states, which are the values in bold-type face in Table 3.6.

Table 3.6. Kinetics Parameters and Reactivity Feedback Coefficients for MURR Transition Cycles and Reference Equilibrium LEU Core.

Core	β_{eff}	Λ (μs)	Coolant Void Coefficient (\$/%-void)	Coolant Temperature Coefficient ^a (\$/°C)	Fuel Temperature Coefficient (\$/°C)
Cycle 1	0.782%	41.5	-0.319	-1.566E-03	-8.124E-04
Cycle 6	0.769%	44.0	-0.287	-3.086E-03	-7.968E-04
Equilibrium LEU	0.758%	42.7	-0.306	-2.881E-03	-8.426E-04

^a Reactivity feedback for temperature changes of the coolant only. The effect of coolant density changes on reactivity due to temperature increase or decrease is accounted for in the coolant density coefficient.

The MURR Technical Specifications place requirements on the values of some core physics parameters, which will be confirmed through a series of low-power reactor startup tests and predictions that will be completed when the core is initially loaded with fresh LEU at conversion [22]. As an initial confirmation of these requirements, the minimum absolute value of the core void coefficient of reactivity has been calculated as -0.287 \$/%-void, which occurs at BOC of Cycle 6. This value is equivalent to $-2.20 \times 10^{-3} \Delta k/k/\text{%-void}$ in the primary coolant. MURR Technical Specification 5.3(b) requires that the average core void coefficient be more negative than $-2.0 \times 10^{-3} \Delta k/k/\text{%-void}$. Therefore, it is expected that the requirement on the core void coefficient will be satisfied for the LEU fuel at all conditions from initial conversion through transition to equilibrium LEU operations.

MURR Technical Specification 5.3(a) also specifies a requirement that the core temperature coefficient of reactivity must be more negative than $-6.0 \times 10^{-5} \Delta k/k/^\circ\text{F}$. The coolant temperature coefficient values reported in Table 3.6 do not include the effect of coolant density changes as a result of temperature fluctuations, and so are not an appropriate point of comparison to the Technical Specification requirement. Most the negative reactivity feedback resulting from changes in the primary coolant temperature are due to the corresponding density change of the coolant. Analyses completed in Reference 22 predicted the primary coolant temperature coefficient for the fresh LEU core including both temperature and density changes. The predicted value of $-6.44 \times 10^{-5} \Delta k/k/^\circ\text{F}$ over the range of the analysis satisfies the Technical Specification requirement. At the time of the low-power startup tests, the value of the primary temperature coefficient will be confirmed through measurement and detailed analysis.

Termination of reactor transients from engineered safety systems (i.e., control blade scram) that are tripped from core power, coolant flow, or reactor period settings are modeled in the transient analyses. The control bank (all four control blades ganged together) differential reactivity worth was calculated with MCNP5 for the BOC core state of Cycles 1 and 6, as well as the BOC, eq. Xe, and EOC states for the equilibrium LEU core (Cores 7X, 8X, and 9X, respectively) along the entire range of control blade movement. The control blades in the analysis were assumed to be fresh, with no prior use. The results for all cores are shown in Figure 3.16. The 95% confidence (2σ) of the MCNP5 predictions is 0.06% $\Delta k/k/\text{inch}$; the upper and lower 95% confidence intervals about the average of all cases are shown as the dotted lines in the figure. All predicted values for the differential control bank worth are generally within the 95% confidence interval of the mean, indicating that core burnup or presence of xenon has only a small effect on the reactivity worth of the control blades.

Figure 3.16. Differential Control Bank Worth for LEU Transition Cycles and Equilibrium LEU Core.

MURR Technical Specification 3.2(e) specifies that the maximum rate of reactivity insertion from withdrawal of the control bank must be less than $3.0 \times 10^{-4} \Delta k/k/\text{second}$. The control blades can be withdrawn at a maximum speed of 2.54 cm/minute (1.0 inch/minute). The maximum predicted differential worth for the control bank for all cases examined is $1.35 \times 10^{-2} \Delta k/k/\text{inch}$, which results in a maximum predicted reactivity insertion rate of $2.24 \times 10^{-4} \Delta k/k/\text{second}$ that satisfies the MURR Technical Specifications. At the time of the low-power startup tests, the control blade worth will be measured through a thorough series of calibration tests and confirmed from detailed analysis that will account for the prior use of the control blades and the beryllium reflector.

3.2.6 Reactor Design Parameters

Irradiation and flow testing of fuel plates and fuel elements fabricated with a new very high density U-10Mo monolithic fuel are being completed as part of a fuel qualification effort leading to regulatory decisions to license MURR with the new fuel. This testing to qualify the fuel will be conducted to confirm the geometric stability of fuel plates and fuel elements and the stable and predictable behavior of the fuel system [25].

Information produced to date that is relevant for the design of these experiments has been integrated into a report [26]. Among the irradiation tests that will be conducted in a test reactor are mini-plate, large-size plate, and fuel element tests with test articles fabricated based on one or more key features of the MURR preliminary LEU fuel element design. The plate-level fuel qualification tests are being designed to bound prototypic irradiation conditions for plates that envelope the conditions of the

USHPRR that will use the fuel. Some of the plate-level tests will irradiate test articles to conditions that bound prototypic (equilibrium) MURR irradiation conditions. In general, these tests are being designed with experiment margins of 10% and 15% above the prototypic maximum fission density and maximum power density, respectively, for these key plates and are expected to bound the conditions of brief off-normal operations such as will exist during the transition cycles. The MURR design demonstration element (DDE) will be an irradiation test of a full-size element (with minor adjustments to the design of the end fitting to accommodate insertion in the test reactor) at conditions that are as close to prototypic (equilibrium operations) as possible [8].

To support this testing effort, reactor design parameters for MURR operating with the preliminary LEU fuel element design at prototypic operating conditions have been documented to serve as a basis for target test values for experiment designers in the FQ Pillar [27]. The reactor design parameters were derived from an equilibrium LEU fuel cycle analysis with a model based on the experiment configuration typical for MURR operations in 2008. Analysis has been completed in this work to calculate reactor design parameters corresponding to conditions that will exist during the LEU transition cycles and prototypic equilibrium operations with models based on the 2018 experiment configuration. This has been done to confirm that the documented reactor design parameters in Reference 27 remain appropriate for the goals of the irradiation tests.

One of the key reactor design parameters is the fission density of the fuel because of its documented impact on fuel property and performance data [28]. The maximum local fission density in each plate of an LEU fuel element discharged during equilibrium operations in MURR was reported as reactor design parameter data in Reference 27. A similar approach was used to calculate the maximum local fission density by plate for LEU elements discharged from the equilibrium cycle modeled in this work. A comparison of the predicted fission density values for these two cases is provided in Table 3.7. The results show that the documented reactor design parameters remain appropriate. The maximum fission density values in Reference 27 are 0.8% to 3.4% greater than those predicted for an LEU element discharged from the equilibrium fuel cycle with the 2018 experiment conditions. Thus, the reactor design parameters documented in Reference 27 remain appropriate as the basis for target test values for the MURR DDE test element.

Also included in Table 3.7 are the plate-by-plate maximum discharge fission density values for six fuel elements that are loaded in the proposed transition sequence and reach the same discharge element burnup of ~180 MWd as the elements in the equilibrium fuel cycle. The transition cycles in which these elements are loaded are identified in Table 2.1 through Table 2.4. Each of these elements is discharged at some point during the first 49 cycles of operations that were modeled in the fuel cycle simulation completed with REBUS-PC. The predicted fission density values include the effects of both axial and lateral peaking in the fuel plates. During the cycles in which these elements are loaded in the core, the control blade positions are more deeply inserted than during prototypic equilibrium operations because of the lower total burnup of the elements loaded in these cycles. As a result, the axial peak-to-average plate burnup for these elements is somewhat greater than for an element that is used during equilibrium operations. The greatest impact on the axial peaking occurs for those plates closest to the control blades (e.g., plate 23).

As shown in Table 3.7, the maximum fission density always occurs in plate 1, which has a thin 0.23 mm (0.009 inch) fuel core thickness. The maximum fission density in plate 1 of element MO-008 is 2.3% greater than that for an element discharged from the equilibrium LEU fuel cycle based on the 2018 experiment conditions and only 0.5% greater than the reactor design parameter value of 3.37×10^{21} fissions/cm³. The maximum fission density in plate 23 of element MO-008 is 7.8% greater

Table 3.7. Maximum Local Fission Density by Plate in LEU Elements Discharged from Equilibrium and Transition Cycles.

Plate	Maximum Local Fission Density (10^{21} fissions/cm ³)							
	Reactor Design Parameter ^a	Eq. LEU	Elements Loaded in Transition Cycles ^b					
			MO-001	MO-002	MO-003	MO-004	MO-007	MO-008
1	3.37	3.31	3.37	3.37	3.37	3.38	3.39	3.39
2	2.79	2.74	2.78	2.78	2.79	2.79	2.79	2.80
3	2.25	2.22	2.25	2.24	2.25	2.26	2.26	2.26
4	1.85	1.83	1.86	1.85	1.86	1.87	1.87	1.87
5	1.62	1.60	1.62	1.62	1.63	1.63	1.63	1.63
6	1.48	1.46	1.48	1.48	1.48	1.49	1.49	1.49
7	1.35	1.34	1.37	1.36	1.37	1.37	1.37	1.37
8	1.27	1.26	1.28	1.28	1.28	1.29	1.29	1.29
9	1.22	1.20	1.22	1.22	1.22	1.23	1.22	1.23
10	1.17	1.16	1.18	1.18	1.18	1.19	1.19	1.19
11	1.14	1.13	1.15	1.15	1.15	1.15	1.15	1.15
12	1.13	1.11	1.13	1.13	1.13	1.13	1.13	1.13
13	1.13	1.12	1.13	1.13	1.13	1.14	1.14	1.14
14	1.13	1.12	1.14	1.13	1.14	1.14	1.14	1.14
15	1.14	1.12	1.14	1.14	1.14	1.15	1.15	1.15
16	1.16	1.14	1.16	1.15	1.16	1.16	1.16	1.16
17	1.20	1.17	1.20	1.19	1.20	1.20	1.20	1.20
18	1.26	1.22	1.25	1.25	1.26	1.26	1.26	1.26
19	1.35	1.32	1.35	1.34	1.36	1.36	1.36	1.36
20	1.48	1.45	1.49	1.48	1.50	1.51	1.50	1.51
21	1.70	1.65	1.70	1.69	1.72	1.73	1.73	1.73
22	2.07	2.00	2.08	2.06	2.11	2.13	2.12	2.13
23	2.59	2.51	2.63	2.60	2.67	2.70	2.70	2.71

^a Results based on calculations for equilibrium LEU fuel cycle with MURR_2008 model. Values reported in Table 4-3 through Table 4-5 of Reference 27.

^b Results based on fuel cycle simulation with MURR_2018 model.

than that for the element discharged from the equilibrium LEU fuel cycle and 4.5% greater than the reactor design parameter value for the same plate. Thus, the discharge fission density of the elements used during the MURR transition cycles will be bounded by the irradiation conditions of the plate-level tests which are being designed with an experiment margin of 10% relative to the reactor design parameter values for fission density. Finally, the fuel burnup during the transition cycles will be monitored, and adjustments to reactor fuel management may be employed as needed, to ensure that the maximum fission density does not exceed the license limit of the fuel that will be established by the regulator.

In addition to fission density, the power density at a local level is also a fundamental reactor design parameter for designing irradiation tests of the fuel. Plate-level tests are planned to test the fuel at

conditions which bound the prototypic operating conditions of MURR plates 1 and 22, and plates 1, 22, and 23 of the MURR preliminary LEU fuel element design have been identified as key plates for the MURR DDE tests [26]. Table 3.8 provides a summary of the maximum power density of these key plates for cores loaded during the LEU transition and equilibrium cycles. The reactor design parameter data from Reference 27 are also provided. The results show that the maximum power density in each plate generally decreases from the largest value in the fresh LEU core loaded in Cycle 1 through the transition to the equilibrium LEU core. The reactor design parameter values were all predicted from models for an equilibrium fuel cycle with the loading of experiments typical for MURR operations in 2008. The current analysis for the transition cycles and the equilibrium LEU core is based on the experiment loading that was typical in 2018. The largest power density values observed during the transition cycles are all bounded by the reactor design parameter values plus a 15% margin on the power density which is planned for the plate-level tests [26]. The reactor design parameter values for the maximum power density are larger than those for the MURR_2018 model of the equilibrium LEU core for all plates except plate 2. The difference for plate 2 is not large, however, so it is expected that the reactor design parameters documented in Reference 27 remain appropriate as the basis for the power density target test values for the MURR DDE test element.

Table 3.8. Maximum Power Density in Key Plates in LEU Elements in Transition and Equilibrium LEU Cycles.

	Max Power Density (kW/cm ³)			
	Plate 1	Plate 2	Plate 22	Plate 23
Cycle 1 ^a	14.83	11.17	7.38	9.85
Cycle 15 ^a	14.60	11.00	7.18	9.33
Cycle 30 ^a	14.30	10.70	7.07	9.41
Equilibrium LEU ^a	14.37	<u>10.84</u>	7.09	9.40
Reactor Design Parameter ^b	14.40	10.61	7.40	9.69

^a Results based on fuel cycle simulation with MURR_2018 model.

^b Results based on calculations for equilibrium LEU fuel cycle with MURR_2008 model. Values reported in Table 4-3 (plates 1 and 2) and Table 4-5 (plates 22 and 23) of Reference 27.

4 STEADY-STATE THERMAL HYDRAULICS ANALYSIS

4.1 Thermal Hydraulics Methods and Modeling

Steady-state thermal hydraulic calculations were performed to determine safety margins during the normal operations, i.e., margins to OFI and CHF. PLTEMP/ANL v. 4.3 [29] was used as an analysis tool. The software has undergone verification and validation (V&V) process to be NQA-1 compliant [30].

The PLTEMP/ANL model of the MURR reactor core used for previous analyses [7,12] consisted of fuel elements with all coolant channels and fuel plates. The same model was used for the current analysis, but with the core power distributions for the transition cycles and the equilibrium LEU core with the 2018 experiment loadings as discussed in Section 3. Hot channel factors were applied to the PLTEMP/ANL calculations in order to account for factors such as manufacturing tolerances and the uncertainty of relevant numerical inputs based on other calculations (e.g., the neutronics calculation to obtain the power distribution). The application of hot channel factors, as well as built-in conservatism using power peaking factors discussed above, ensure that the calculated safety margins are assuming the most severe conditions.

The preliminary LEU fuel element design was used for the current work. Following work completed as part of the preliminary SAR [5] the manufacturing tolerances for the end channel dimensions of this design were changed based on the recommendation of the fuel manufacturer. There were no changes to the nominal dimensions of the fuel plates or any other components of the fuel element. In analysis completed as part of Reference 7, it was found that the impact of the end channel tolerance change on the steady-state safety margins is negligible.

Changes to the fuel plate thickness dimensions on a local level can occur due to a number of burnup-related effects which are discussed below. The fuel plate thickness changes reduce the coolant channel thickness and cause the coolant flow to decrease in the affected channels. These may consequently increase the rise of the coolant enthalpy in the channels and decrease the margins to OFI and CHF. However, directly modeling the channel thickness reduction in the PLTEMP/ANL model results in non-conservative flow conditions in the other channels which are bounded by plates with less burnup. Because the total core flow rate is fixed, reducing the coolant flow rate to certain flow channels inevitably increases the flow rate to others. Instead, the coolant channel reduction was treated as a manufacturing tolerance, where the reduction of each channel is isolated and treated locally, so that it does not affect the coolant flow rate in the other coolant channels. This maintains conservatism in the analysis.

The maximum coolant channel thickness reduction in the LEU element due to burnup related effects was conservatively determined to be 0.008 inch at the maximum element burnup of 180 MWd, which is consistent with previous analyses [5,7]. This value was chosen to bound the expected fuel plate thickness changes at a local level from fuel swelling, irradiation-induced creep, and oxide layer growth. The value is based on a 0.004 inch maximum plate thickness increase from fuel swelling and irradiation-induced creep and a 0.002 inch maximum oxide thickness on each surface of the fuel plate. It was further assumed that the coolant channel reduction increases linearly with burnup. For the end channels, the reduction was assumed to be half of the internal channel reduction (0.004 inch maximum reduction). A complete discussion of the details of PLTEMP/ANL model of the MURR reactor can be found in Section 13 of Reference 12.

The safety margins for normal steady-state operation were calculated for each of the eleven transition cycles identified in Section 3.2.3 as being potentially more limiting for margins to safety. Each transition core evaluated was labeled as “A-BBB-C”. The identifier “A” refers to the particular cycle during the transition period (integer 1 through 30). The identifier “BBB” is the core state, which can be “BOC”, “EqXe”, or “EOC” for beginning-of-cycle, equilibrium-xenon, and end-of-cycle conditions, respectively. The “-C” identifier may or may not be present. The absence of “-C” identifier indicates that the core was evaluated with the assumption that the control blades are ganged together, which is typical for MURR operations. As discussed above, there was an additional assumption in this work that the control blades have no prior operating history (fresh BORAL composition). In selected cases, the impact of different control blade conditions were considered. For these cases, the identifier “C” with a value of “1”, “2”, or “3” is used, with each representing a specific control blade burnup and position mismatch (skew) at the limits of the MURR Technical Specifications, as indicated in the table below. As discussed in Section 2.3 of Reference 7, these conditions are uncommon, but were evaluated as conservative limits of operations.

C=Blank	Blades A, B, C, and D at fresh nominal conditions and ganged.
C=1	Blades A and B fresh, blades C and D at 8 years prior use. Blades C and D positioned 1 inch higher than A and B, per Technical Specification.
C=2	Blades B and C fresh, blades A and D at 8 years prior use. Blades A and D positioned 1 inch higher than B and C, per Technical Specification.
C=3	Blades A and D fresh, blades B and C at 8 years prior use. Blades B and C positioned 1 inch higher than A and D, per Technical Specification.

These blade depletion and position perturbations are illustrated in Figure 4.1. The pairing of two fresh blades adjacent to each other causes a radial power tilt towards the two highly depleted blades on the opposite side of the core. Positioning the depleted blades up to the Technical Specification limit of 2.54 cm (1 inch) more withdrawn than the fresh blades creates an even greater power tilt. A case with blades C and D being fresh with blades A and B at their maximum burnup was not considered because it was not expected to produce more limiting conditions than the other cases.

Figure 4.1. Control Blade Perturbations Considered in Analysis of Minimum Safety Margin.

The safety margins were also calculated for the reference LEU equilibrium core to provide a point of comparison for the transition cycles. Each core was evaluated at BOC, equilibrium xenon, and EOC

conditions in the same manner as for the transition cycles. For the equilibrium cores, each core is labeled as “nX-C”. The identifier “n” is “7”, “8”, or “9”, referring to BOC, EqXe, or EOC, respectively, and the identifier “C” is the same as noted above for the transition cores.

The critical control blade positions and corresponding three-dimensional power distributions were calculated using MCNP5 for each case. The critical control blade positions for the fresh and depleted control blades (for cases with $C \neq 0$) were determined through a search algorithm. The stainless steel regulating blade was positioned at 10 inches withdrawn for BOC core states (xenon-free cores at the beginning of the cycle) and at 15 inches withdrawn for equilibrium xenon and EOC core states.

4.2 Reactor Operating Conditions for Steady-State Analysis

Safety margins for normal steady-state operation were calculated at the conditions shown in Table 4.1, unless indicated otherwise in the description of the analysis. As discussed above, the power distributions for each core state were evaluated with MCNP5. Previous analysis found that 96.4% of the energy from fission and gamma-heating is deposited in the primary system [31], with the balance of the total energy deposited outside the primary system (e.g., reflectors and pool). Since only the primary system of MURR was modeled with PLTEMP/ANL, the LSSS power of 15 MW was scaled accordingly for PLTEMP/ANL calculations. Note that the total reactor power is reported in all results presented below.

Table 4.1. Conditions for Steady-State Thermal Hydraulic Analysis of LEU Cores.

Condition	Value
LSSS Reactor Power, MW	15.0
Fraction of Total Reactor Power Deposited in Primary System	96.4%
LSSS Core Coolant Inlet Temperature, °F (°C)	145 (62.8)
LSSS Primary Coolant Flowrate, gpm (m ³ /s)	3300 (0.208)
LSSS Pressurizer Pressure, psig (kPa)	60.7 (419)
Pressurizer Level, inch (m)	-16 (-0.41)

4.3 Results of Steady-State Thermal Hydraulics Analysis

The safety margins for normal steady-state operations for the transition cycles and the reference equilibrium LEU core are presented in this section based on two figures of merit: reactor power at which OFI occurs under LSSS coolant flow conditions (OFI power) and the CHFR at LSSS power and coolant flow conditions. For predictions of the OFI power, calculations were conducted based on an iterative search procedure in PLTEMP/ANL to find the minimum reactor power at which OFI occurs in the core. In this case, all conditions from Table 4.1 were specified as part of the code input except reactor power. For the calculation of the CHFR, a single calculation at the LSSS power was conducted. Therefore, all conditions from Table 4.1 were used.

Table 4.2 reports the OFI power for each considered core. The results were obtained by conducting PLTEMP/ANL calculations with an iterative solution option to predict the lowest reactor power when OFI occurs. The most limiting case among the transition cycles is for Core 28-EOC-1, which has a predicted OFI power of 17.49 MW, giving a margin of 2.49 MW above the LSSS power of 15 MW (16.6% margin to the LSSS power). For equilibrium cores (which are shaded in gray), Core 9X-1 has the lowest OFI power of 17.52 MW, which gives 2.52 MW margin above the LSSS power (16.8% margin to the LSSS power).

Table 4.2. Results for OFI Power.

Core	OFI Reactor Power, MW	Element	Channel	Core	OFI Reactor Power, MW	Element	Channel
1-BOC	19.58	X5	4	23-EOC-2	17.87	X8	23
1-EqXe	20.57	X8	4	23-EOC-3	17.78	X6	23
1-EOC	20.52	X3	4	27-BOC	19.30	X2	4
9-BOC	19.78	X5	4	27-EqXe	19.27	X8	23
9-EqXe	20.50	X4	4	27-EqXe-1	17.68	X7	23
9-EOC	20.48	X7	4	27-EqXe-2	17.96	X8	23
11-BOC	19.57	X2	4	27-EqXe-3	18.10	X7	23
11-EqXe	20.40	X2	4	27-EOC	19.05	X7	23
11-EqXe-1	18.56	X7	23	27-EOC-1	17.73	X7	23
11-EOC	20.33	X2	4	27-EOC-2	17.72	X8	23
11-EOC-1	18.49	X6	23	27-EOC-3	18.01	X6	23
15-BOC	19.54	X2	4	28-BOC	19.21	X8	4
15-EqXe	20.24	X2	4	28-EqXe	18.89	X7	23
15-EqXe-1	18.63	X6	23	28-EqXe-1	17.60	X7	23
15-EqXe-2	18.77	X8	23	28-EqXe-2	17.87	X8	23
15-EqXe-3	19.17	X6	23	28-EqXe-3	18.00	X7	23
15-EOC	20.31	X6	4	28-EOC	18.64	X7	23
15-EOC-1	18.38	X7	23	28-EOC-1	17.49	X7	23
15-EOC-2	18.69	X8	23	28-EOC-2	17.81	X8	23
15-EOC-3	18.90	X4	23	28-EOC-3	17.81	X7	23
17-BOC	19.49	X2	4	30-BOC	19.23	X2	4
17-EqXe	19.78	X7	23	30-EqXe	18.95	X7	23
17-EqXe-1	18.12	X6	23	30-EqXe-1	17.71	X7	23
17-EOC	19.73	X7	23	30-EqXe-2	17.83	X8	23
17-EOC-1	17.84	X7	23	30-EqXe-3	18.12	X6	23
19-BOC	19.52	X2	4	30-EOC	18.88	X7	23
19-EqXe	19.57	X7	23	30-EOC-1	17.55	X7	23
19-EOC	19.28	X6	23	30-EOC-2	17.82	X8	23
22-BOC	19.30	X8	4	30-EOC-3	17.87	X7	23
22-EqXe	19.13	X6	23	7X	19.13	X8	4
22-EqXe-1	17.65	X7	23	7X-1	18.57	X8	4
22-EOC	18.97	X7	23	7X-2	18.71	X8	4
22-EOC-1	17.53	X7	23	7X-3	18.70	X4	4
23-BOC	19.32	X2	4	8X	18.98	X7	23
23-EqXe	18.85	X6	23	8X-1	17.61	X7	23
23-EqXe-1	17.51	X6	23	8X-2	17.90	X2	23
23-EqXe-2	17.90	X8	23	8X-3	17.91	X6	23
23-EqXe-3	17.83	X6	23	9X	18.61	X6	23
23-EOC	18.70	X6	23	9X-1	17.52	X6	23
23-EOC-1	17.57	X6	23	9X-2	17.86	X2	23
				9X-3	17.72	X6	23

A few trends regarding the OFI power are notable. Previous thermal hydraulics analysis for core states that exist during equilibrium operations found that OFI is generally most limiting at equilibrium xenon and EOC core states when the control blades are more fully withdrawn [5,7]. The more withdrawn control blade position results in a greater axially-averaged hot-stripe heat flux and a corresponding greater enthalpy rise in the coolant channel (compare the equilibrium LEU core hot-stripe heat flux provided in Figure 3.7 and Figure 3.8). For the initial transition cycles, the OFI power is generally lower at the BOC core state. This is because the initial cycles have greater excess reactivity and deeper control blade positions which pushes the radial power profile towards the innermost plates. By Cycle 19 the total core burnup is comparable to normal equilibrium operations so that the OFI power tends to be lower later in the cycle.

Previous analyses also generally demonstrated that the location where OFI occurs shifts from channel 4 (between plates 3 and 4) to channel 23 (between plates 22 and 23) as the cycle progresses from BOC to EOC. This is for similar reasons as noted above—the withdrawal of the control blades during the cycle results in a larger hot-stripe heat flux in the outermost plates later in the cycle. This trend is not observed in the results presented in Table 4.2 for the initial transition cycles with the control blades ganged together; the limiting channel for OFI tends to remain in channel 4 throughout the cycle. Variations are observed for initial transition cycles with different control blade skew patterns. For Cycle 17 and later, the previously observed trend occurs, and the lower OFI power occurs in channel 23 later in the cycle.

The results in Table 4.2 also show that the control blade skew pattern for “C=1” tends to result in a lower OFI power than the other blade skew patterns (C=2 and C=3, as well as the case with the control blades all-fresh and ganged together) and that OFI tends to occur first in elements in positions X6, X7, or X8. The fuel management employed in the fuel cycle simulation placed the highest burnup elements in core positions X6 for the transition cycles and X8 for the equilibrium LEU core and position X7 would be loaded with an element at an MOL burnup state. For the case with “C=1,” the power distribution in the core is uncharacteristically pushed towards the elements in the core with greater burnup.

Finally, the OFI power for particular core states tends to decrease as elements with greater burnup are loaded in the core. This is because of the assumed reduction of the coolant channel thickness on a local level due to burnup-related effects (fuel swelling, irradiation-induced creep, and oxide). The lowest OFI power never occurs in a fresh element except for Cycle 1, which is loaded only with fresh LEU elements.

Figure 4.2 presents the calculated OFI power for each of the 69 transition cycle cases from Table 4.2 in a chart form. It can be seen that the OFI power for the initial cores following conversion to LEU is much larger than for those later in the transition sequence, resulting in OFI margins to the LSSS power of more than 4.5 MW. As the burnup of the fuel elements (and the total core burnup) increases, there is a general reduction of the OFI power. The minimum OFI power of 17.49 MW predicted for the transition cycles is very close to the OFI power of 17.52 MW for the reference equilibrium LEU core. It is therefore concluded that the safety margin to OFI for the transition cycles will not be significantly less than that for the equilibrium LEU cores. In addition, there is adequate margin to OFI in all cases.

Figure 4.2. OFI Powers for Transition LEU Cores.

The CHFR at the LSSS power for each considered core was also calculated using PLTEMP/ANL. A summary of the results is provided in Table 4.3. The values marked in red indicate the lowest CHFR at the LSSS power among the transition cores and among the equilibrium LEU cores. The minimum CHFR at LSSS power for all cores evaluated is highlighted in green. For the transition cores, Core 1-BOC is predicted to have the lowest CHFR, with a value of 2.67. For the equilibrium cores, Core7X-1 has the lowest CHFR, with a value of 2.53. Note that for all cases the minimum CHFR always occurs on the convex side of plate 23, which is the last (radially outmost) plate. The local heat flux for this plate is strongly influenced by the control blade positions, which is also another reason why the CHFR for a given core loading and control blade skew pattern occurs at BOC. Although the power density is greatest for plate 1 (see, for example, Table 3.8), this plate has a very thin fuel zone. The thicker fuel in plate 23 thus results in a greater heat flux on the surface of the fuel plate, and so a smaller CHFR. In addition, the smallest CHFR tends to occur in fresh or lower burnup elements loaded in the core.

Regarding the CHFR at LSSS safety criterion, the safety margins of the transition cycles are bounded by the equilibrium LEU core. Also, the regulatory requirement that the minimum CHFR be at least 2.0 at the LSSS power is always met with significant margins during the transition period and for the LEU equilibrium core.

Table 4.3. Results for CHFR at LSSS Power of 15 MW.

Core	CHFR at LSSS Power	Element	Axial Level ^a	Core	CHFR at LSSS Power	Element	Axial Level ^a
1-BOC	2.67	X6	18	23-EOC-2	3.15	X7	14
1-EqXe	2.75	X6	16	23-EOC-3	3.15	X7	14
1-EOC	2.81	X6	17	27-BOC	2.83	X1	17
9-BOC	2.73	X6	18	27-EqXe	3.01	X1	15
9-EqXe	2.81	X6	17	27-EqXe-1	2.80	X1	15
9-EOC	2.85	X6	17	27-EqXe-2	2.89	X1	15
11-BOC	2.74	X8	18	27-EqXe-3	2.97	X7	14
11-EqXe	2.90	X4	16	27-EOC	2.96	X1	15
11-EqXe-1	2.77	X1	16	27-EOC-1	2.83	X1	15
11-EOC	2.93	X4	16	27-EOC-2	2.89	X1	15
11-EOC-1	2.81	X8	15	27-EOC-3	3.06	X7	15
15-BOC	2.80	X8	18	28-BOC	2.81	X1	17
15-EqXe	2.88	X1	16	28-EqXe	2.99	X1	14
15-EqXe-1	2.69	X1	16	28-EqXe-1	2.82	X1	14
15-EqXe-2	2.80	X1	16	28-EqXe-2	2.89	X1	15
15-EqXe-3	2.89	X3	15	28-EqXe-3	2.95	X7	13
15-EOC	2.93	X1	16	28-EOC	3.00	X1	15
15-EOC-1	2.72	X1	15	28-EOC-1	2.89	X1	15
15-EOC-2	2.84	X1	15	28-EOC-2	2.95	X1	14
15-EOC-3	2.89	X7	15	28-EOC-3	2.96	X7	13
17-BOC	2.83	X8	18	30-BOC	2.76	X1	17
17-EqXe	3.03	X4	16	30-EqXe	2.93	X1	14
17-EqXe-1	2.89	X1	15	30-EqXe-1	2.79	X1	15
17-EOC	3.06	X4	15	30-EqXe-2	2.86	X1	14
17-EOC-1	2.89	X1	16	30-EqXe-3	3.03	X5	15
19-BOC	2.86	X8	17	30-EOC	2.97	X1	15
19-EqXe	3.06	X4	15	30-EOC-1	2.80	X1	15
19-EOC	3.12	X1	15	30-EOC-2	2.91	X1	15
22-BOC	2.85	X1	18	30-EOC-3	3.07	X1	15
22-EqXe	3.04	X1	15	7X	2.76	X1	17
22-EqXe-1	2.88	X1	15	7X-1	2.53	X1	17
22-EOC	3.04	X1	15	7X-2	2.60	X1	16
22-EOC-1	2.90	X1	14	7X-3	2.72	X5	16
23-BOC	2.97	X8	17	8X	2.96	X1	16
23-EqXe	3.20	X4	14	8X-1	2.79	X1	15
23-EqXe-1	3.05	X1	15	8X-2	2.85	X1	14
23-EqXe-2	3.09	X7	14	8X-3	3.03	X5	16
23-EqXe-3	3.06	X7	14	9X	3.01	X1	14
23-EOC	3.24	X1	14	9X-1	2.84	X1	14
23-EOC-1	3.11	X1	15	9X-2	2.89	X1	15
				9X-3	3.08	X1	15

^a Axial level of 24 nodes in PLTEMP/ANL model, with 1 being at the top of the fuel and 24 being at the bottom of the fuel.

5 ACCIDENT ANALYSES

5.1 Introduction

Among the nine postulated accidents identified by NUREG-1537 to be evaluated for non-power reactors [19,20], only reactivity insertion accidents (RIAs), loss-of-coolant accidents (LOCAs), and loss-of-flow accidents (LOFAs) were evaluated as part of the transition cores safety analysis. The predicted outcomes of these accidents are specifically dependent on the cycle and core state (i.e., burnup of the fuel elements in the core, ^{135}Xe concentration, prior control blade burnup and skew, experiment loading, and so on) which affect the core power distribution. The remaining postulated accidents from NUREG-1537 are independent of the specific core states, and so are not uniquely affected by the conditions that will exist during the transition cycles. These accidents were analyzed in the preliminary SAR [5].

The transition cycles that are expected to have smaller safety margins to the FTSL during RIAs, LOCAs, and LOFAs were selected for analysis based on the total core burnup at BOC and the burnup difference between adjacent elements in the core, as discussed in Section 3.2.3. Accident analyses were also completed for the reference equilibrium LEU core. Unlike the steady-state thermal hydraulics analysis which considered core states from BOC through EOC for each cycle and various control blade burnup and control blade position skews, the accident analyses only evaluated the BOC core states and with fresh control blades that are ganged together. The BOC core states are expected to be the most limiting because the axial power peaking is largest at BOC due to the deeper control blade positions. Two cycles were evaluated at both BOC and EOC in order to confirm the conservatism of this assumption. The patterns of prior control blade burnup and control blade position skews were selected as extreme states for evaluation in the steady-state thermal hydraulics analysis, although these are conditions that the reactor is not expected to ever achieve practically. There seems to be no reason to include the added conservatism that a RIA or a severe LOCA will occur while such an extreme off-normal reactor state exists. Even so, initial conditions that are limiting at full licensed power, limiting conditions for operation (LCO), or LSSS conditions have been evaluated, as described below.

5.2 Modeling and Assumptions

The initial steady-state conditions before the accident begins are an important factor affecting the outcome of the postulated event. A summary of the initial conditions assumed for the accident analyses completed in this work is provided in Table 5.1. For all accidents, the initial steady-state conditions for the primary flow, reactor inlet temperature, pressurizer pressure, and pool temperature were assumed to be the LSSS conditions. The pressurizer level was assumed to be set at the scram set point. The pressure and temperature of the anti-siphon tank was set at the midrange of the operating band and temperature of the pool (because the anti-siphon tank is submerged in the pool), respectively. Based on guidance provided in NUREG-1537 [19], the initial operating power for the RIAs was assumed to be at the rod run-in set point LCO, which is 115% of the full licensed power, while the LOCAs and LOFAs were assumed to begin from the full licensed power. These conditions are more stringent compared to any conservative limits of the normal operating band for MURR and are maintained from the previous safety analysis [7,13].

The same types of impacts of fuel burnup considered in previous transient accident analyses [7,13] were maintained in this work. Specifically, the effects of burnup/in-reactor use on the U-10Mo thermal conductivity, FTSL, oxide layer growth, and coolant channel restriction were modeled. For the U-10Mo fuel thermal conductivity, the same combined fit of the thermal conductivity as a function

Table 5.1. Initial Steady-State Conditions for Accident Analyses.

Accident	RIA	LOCA/LOFA
Reactor Power, MW	13.8	12
Core Coolant Inlet Temperature, °F (°C)	145 (62.8)	
Primary Coolant Flowrate, gpm (m ³ /s)	3300 (0.208)	
Pressurizer Pressure, psig (kPa)	60.7 (419)	
Scram Set Point Pressurizer Level, inch (cm)	-16 (-0.41)	
Anti-Siphon Tank Pressure, psig (kPa)	-	36 (250)
Anti-Siphon Tank Temperature, °F (°C)	-	120 (48.9)
Pool Temperature, °F (°C)	-	120 (48.9)

of fission density and temperature as used in previous work was used [7,13]. This fit was derived from preliminary measured data [32]. Future measurements of prototypic fuel are expected as a part of ongoing fuel qualification work by the FQ Pillar [26]. The thermal conductivity of the U-10Mo monolithic fuel decreases with increasing fission density. A summary of the methodology used to estimate the maximum fission density for the plates in the LEU fuel elements in the transition cycles is provided in Appendix A. For added conservatism in the evaluation of the fuel temperature during the postulated accidents, the U-10Mo thermal conductivity for the fuel plates in each element represented in the models was determined based on the maximum fission density in the fuel plate.

The safety margin for the RIAs, LOCAs, and LOFAs is the difference between a burnup-dependent FTSL and the maximum predicted fuel temperature. The same burnup dependent safety limit for the U-10Mo fuel as used in previous transient accident analyses [5,7,13] was used in this work. This safety limit, based on the lower 95% prediction bound for preliminary blister anneal tests performed for U-10Mo monolithic fuel plates, is described in more detail in Reference 13. In the preliminary work, the burnup dependent FTSL was determined as a constant value of 450 °C (842 °F) for fission density values below 1.5×10^{21} fissions/cm³, and monotonically decreases for fission density values greater than 1.5×10^{21} fissions/cm³. For added conservatism in the accident analyses, the FTSL for each plate in the accident models was determined based on the maximum fission density of the fuel in the plate.

The oxide layer on the surface of the fuel plates impacts the fuel temperature during steady-state operations and in postulated accidents in two ways. As a direct effect, the oxide results in a higher fuel temperature because of the thermal resistance of the oxide to heat conduction from the fuel plate to the coolant. As a second-order effect, the oxide causes some restriction of the coolant channel and reduces the coolant flow rate through channels in elements with greater burnup relative to fresh elements.

The thickness of the oxide layer on the surface of the fuel plates as part of in-reactor use was modeled in a manner similar to previous safety analyses [5,7]. To predict the oxide layer thickness with element use, an empirically modified form of the Griess correlation [33] was prepared based on measured data for HEU fuel elements in use at MURR [34]. The details of this modeling effort are discussed in Appendix B. The “MURR modified Griess correlation” was used to predict the maximum oxide thickness on the inner (concave) surface of LEU plate 23 as a function of the number of hours of use for elements in the transition cycles and the equilibrium LEU core. For a given number of hours of use, the oxide thickness is predicted to be 7% to 10% larger for elements loaded in the transition cycles relative to elements loaded in an equilibrium cycle. This is because the lower burnup of the

elements in the transition cycles requires that the control blades positions must be more deeply inserted, which results in greater power peaking and greater maximum plate surface temperatures, which in turn increases the oxide thickness.

For the LEU fuel elements, the maximum oxide layer thickness is expected to occur on the inner surface of plate 23, where the plate surface temperature during reactor operations is expected to be the greatest. This assumption was preliminarily based on the heat flux profiles in Figure 3.8, which show that the heat flux for plate 23 is greatest among all the plates in the LEU element for all but Cycle 1, and was also confirmed through PLTEMP/ANL analysis. The plate surface temperature is greater on the inner surface of the fuel plate because that surface is adjacent to a coolant channel heated on two sides (between fuel plates 22 and 23); the outer (convex) surface is adjacent to a coolant channel heated on only one side. For added conservatism in the analysis of the postulated transient accidents in the transition cycles and the reference equilibrium core, the maximum oxide thickness predicted with the MURR modified Griess correlation for the inner surface of plate 23 was assumed to occur on all fuel plates in the element.

The oxide thickness on the fuel plates was included in the software models used to evaluate the postulated transient accidents. While the REBUS-PC results for the fuel cycle simulations provide the burnup of each element loaded in the core in transition cycles and the equilibrium cycle, the correlations provided in Appendix B were fit with the number of hours of use in the reactor core rather than the core burnup as this is the general form of the Griess correlation. The increase of element burnup with time is not a constant value over the life of the element. Rather, elements with lower burnup accumulate burnup at a faster rate than those with greater burnup due to the somewhat unequal power sharing in the core (see discussion in Section 3.2.1). The burnup following each cycle of use for each element used during the first 49 cycles of weekly operations was compiled from the REBUS-PC results. These data showed that for elements at or near BOL it takes 15.7 hours to accumulate 1 MWd of additional burnup. For elements at or near EOL, it takes 16.6 hours to accumulate 1 MWd of burnup, or 5.2% longer. This change in the rate of burnup accumulation was accounted for when predicting the oxide thickness on the plates in LEU fuel elements using correlations derived from work documented in Appendix B. The predicted oxide thicknesses evaluated for elements with burnups ranging from 0 to 180 MWd (0 to ~2,900 hours of operation) loaded in transition cycles and the equilibrium fuel cycle are summarized in Table 5.2 and Figure 5.1.

Finally, the same assumptions for the coolant channel restriction as used in the steady-state analysis were applied in the transient accident analyses. That is, it was assumed that there is a linear relationship between coolant channel thickness reduction and fuel burnup, with a maximum reduction of 0.008 inch at the discharge burnup of 180 MWd. Note that, as described in Section 4.1, the coolant channel reduction in the models includes the thickness of an oxide layer, which is conservatively estimated as 0.002 inch thick at the EOL of the LEU fuel element for the purpose of predicting the channel reduction with burnup. As shown in Figure 5.1, the actual thickness of the oxide layer on the surface of the LEU fuel plates is predicted not to exceed 0.001 inch (1.0 mil) for fuel element burnup that remains less than the discharge burnup of 180 MWd.

5.3 Reactivity Insertion Accidents

For the analysis of a postulated RIA during the transition cycles and equilibrium operations, a 0.6% $\Delta k/k$ step insertion was assumed. This step insertion is the Technical Specification limit for either a) all unsecured experiments in the reactor; b) the reactivity worth for any secured removable experiment, or c) the total reactivity worth of all experiments in the center flux trap. While other RIA

Table 5.2. Predicted Oxide Thickness on Fuel Plates in LEU Elements Loaded in Transition Cycles and Equilibrium Cycle.

Element Burnup, MWd	In-Use Hours ^a	Oxide Thickness, mil ^b	
		Transition Cycles	Equilibrium Cycles
0	0	0.000	0.000
10	158	0.091	0.082
20	316	0.174	0.157
30	474	0.251	0.227
40	633	0.322	0.291
50	792	0.387	0.351
60	952	0.447	0.406
70	1112	0.502	0.458
80	1273	0.554	0.505
90	1434	0.601	0.550
100	1596	0.646	0.592
110	1758	0.688	0.632
120	1920	0.728	0.671
130	2083	0.767	0.708
140	2247	0.806	0.745
150	2411	0.844	0.782
160	2575	0.882	0.819
170	2740	0.921	0.857
180	2905	0.963	0.897

^a Based on fit of in-use hours with element burnup compiled from REBUS-PC results.^b 1 mil = 0.001 inch. **Figure 5.1. Predicted Oxide Thickness on Fuel Plates in LEU Elements Loaded in Transition Cycles and Equilibrium Cycle.**

with either smaller step reactivity insertions or ramp reactivity insertions were evaluated in previous safety analyses for equilibrium LEU cores [5,7], the 0.6% $\Delta k/k$ step RIA is the most limiting because it results in the highest fuel temperature of all the RIAs analyzed. In the analysis completed for the MURR LEU preliminary SAR, the minimum margin to the FTSL for this accident in an equilibrium LEU core with the MURR_2008 model was determined to be 109 °C [5]. For the safety analysis for an equilibrium LEU core with the experimental facilities in use in 2018 but modified by a major experimental facility upgrade to produce Mo-99, the minimum margin to the FTSL was predicted to be 102 °C [7].

The RIA analysis for both the transition cycles and the reference equilibrium core with the MURR_2018 model was completed in this work using v. 7.6 of the PARET/ANL software [24]. PARET/ANL is a one-dimensional reactor transient analysis code that models plate, pin, and nested tube fuel geometries. The collection of fuel, zirconium interlayer, cladding, and associated coolant is called a “channel” in PARET/ANL. An average channel is typically used to represent the core and bulk reactivity feedback effects from the transient, while one or more “peak” channels are used to represent those components (e.g., a fuel plate) which are expected to reach the highest temperature as a result of the transient. There is no heat transfer between the channels in PARET/ANL, but the channels are coupled through reactivity feedback effects (i.e., the reactivity feedbacks from each channel in the model are volume-weighted and summed to compute the total core reactivity and power as a function of time in the transient). The heat source descriptions for each channel in the PARET/ANL models were calculated from the MCNP5 power density tallies for the 0.5 cm (0.2 inch) hot-stripe along the edge of the fuel that is adjacent to the side plates. The analysis is conservative because it neglects any coolant mixing that would occur between the portions of the coolant channel that are adjacent to the fueled and unfueled parts of the fuel plate.

As described in Table 5.1, the RIA for the LEU core is assumed to be initiated from a steady state operating power of 13.8 MW, which is the rod run-in set point of 115% of the expected full licensed power for LEU operations of 12 MW. The coolant flow rate and inlet temperature were assumed to be at the LSSS conditions. The scram set point is 125% of the full licensed power, or 15 MW. There is also a minimum reactor period trip setting of 8 seconds, which will be reached before the power trip in the case of a 0.6% $\Delta k/k$ step RIA, but for added conservatism it is assumed that the period trip fails in the analysis of the accident. From the time that the scram signal is sent to the start of the control blade drop there is a short delay time. Measurements of the safety system trip functions timing performed at MURR in May and June of 2014 observed the delay time to be 75 milliseconds. The delay time was conservatively modeled to be 150 milliseconds in the current work. While the heat source distributions for each case were calculated with the control blades at their critical positions, the transient analyses assumed the scram initiates with the control blades at the fully withdrawn position. The differential blade worth at the fully withdrawn position of the control blades is much lower than that when the blades are near their critical positions of 32 to 42 cm (13 to 17 inch) for the cores that were evaluated (see Figure 3.16). Therefore, this assumption adds conservatism to the analysis because the initial rate of negative reactivity insertion from the control blade scram is lower.

Eleven transition cycles plus the reference equilibrium LEU core were selected for analysis. Each core was evaluated at BOC because the xenon-free core state will have greater power peaking, and consequently greater maximum fuel temperature, than later in the cycle because the control blade positions are more deeply inserted. The EOC core state was evaluated for two of the transition cycles (Cycle 1 and 30) to confirm this assumption.

The burnup conditions of the elements loaded in the transition cycles and the reference equilibrium LEU core are summarized in Table 3.4. In addition to an average channel to represent the core, four

peak channels representing plate 23 in each of the LEU element pairs loaded in the cores were modeled. Plate 23 was selected for analysis because this plate has the greatest local heat flux among all the plates in the LEU fuel element and, consequently, is expected to be the most limiting during the RIA. Previous analysis found that at some LEU element burnup conditions plate 22 may exhibit a larger fuel temperature than plate 23 during the 0.6% $\Delta k/k$ step RIA, but the margin to the FTSL is always smallest for plate 23 in all elements at all burnup conditions [5,7]. Consequently, only plate 23 was evaluated as part of this work to identify the most limiting core states among cores from initial conversion to LEU through the transition to equilibrium LEU operations. Finally, although plate 23 is cooled by channels on each side that are different in nominal fabricated thickness (0.093 inch on the inboard side and 0.0955 inch on the outboard side) and enthalpy rise (the inboard channel is heated on both sides by plates 22 and 23 while the outboard channel is heated only on one side by plate 23), the model assumes a channel that is heated on both sides by two identical plates with the heat source of plate 23, which adds conservatism to the analysis.

The oxide layer thickness on the surface of the fuel plates has a direct impact on the predicted maximum fuel temperature. The methodology used in this work for predicting the oxide layer thickness for elements loaded in the transition cycles and the equilibrium LEU core is described in Section 5.2 and Appendix B. Given the possible variations of the operating conditions that may exist during the transition cycles, a set of branch cases with a 10% increase of the oxide thickness in the models for the 0.6% $\Delta k/k$ step RIA was evaluated. The impact of the increased oxide thickness on the maximum fuel temperature was less than 3 °C.

In addition to the modeling of an explicit oxide layer on the surface of the fuel plates in elements with non-zero burnup, the PARET/ANL model also included a reduction of the coolant channel thickness that increases linearly with element burnup from 0 inch at 0 MWd to 0.008 inch at 180 MWd. The channel reduction relative to the nominal coolant channel gap can result in a flow redistribution among the elements in the core when they have different burnup, with less flow in those elements with greater burnup and more of a channel reduction. The corresponding flow redistribution was included in the PARET/ANL model for the channels that represent elements with differing burnup. Except for the case of the fresh core in Cycle 1, the maximum fuel temperature/minimum margin to the FTSL always occurs in an element that has some prior burnup. While a case without flow redistribution (as was considered for the LOCA and LOFA analyses along with cases with flow redistribution) would result in greater temperatures in the fuel plates in elements with lower burnup, the elements in the core with low burnup were always predicted to have a wide margin to the FTSL compared to those elements with greater burnup. Consequently, the flow redistribution case is the most conservative for the analysis of the 0.6% $\Delta k/k$ step RIA.

The results for all cases evaluated for the 0.6% $\Delta k/k$ step RIA are summarized in Table 5.3. The maximum fuel temperature for the transition cycles is 295.0 °C and occurs in plate 23 in an element with a burnup of 151 MWd loaded in Cycle 30 at BOC. The minimum predicted margin to the FTSL for the transition cycles is 110 °C and occurs in this same element and core state.

The EOC core states for Cycles 1 and 30 were evaluated to confirm the assumption that the BOC core state is the most limiting in terms of maximum fuel temperature and margin to the FTSL during the operating cycle. The BOC core state is the most limiting for Cycle 30, but the results in Table 5.3 show that this is not the case for Cycle 1. The maximum fuel temperature for the 0.6% $\Delta k/k$ step RIA is greater and the margin to the FTSL is smaller for the EOC core state of Cycle 1 compared to the BOC

Table 5.3. Results of 0.6% $\Delta k/k$ Step RIA for Transition Cycles and Reference Equilibrium LEU Core.

Core ^a	Maximum Fuel Temperature, °C				Margin to Fuel Temperature Safety Limit (FTSL), °C			
	X1/X5	X2/X6	X3/X7	X4/X8	X1/X5	X2/X6	X3/X7	X4/X8
1-BOC	249.1	263.8	254.4	261.0	201	186	196	189
1-EOC	257.7	268.0	263.4	268.1	192	182	187	182
9-BOC	260.0	271.6	265.0	271.7	190	178	185	178
11-BOC	264.2	289.9	268.8	275.8	186	160	181	174
15-BOC	253.8	294.9	275.5	283.6	196	155	175	166
17-BOC	272.4	293.4	278.2	286.2	178	150	172	164
19-BOC	273.9	294.6	282.2	288.0	176	141	168	162
22-BOC	264.8	296.6	280.7	295.7	185	125	169	126
23-BOC	277.6	296.3	285.4	290.6	172	126	165	159
27-BOC	262.4	295.6	284.3	292.3	188	115	166	158
28-BOC	262.8	295.4	281.1	294.3	187	115	169	117
30-BOC	253.1	295.0	287.2	293.1	197	110	156	150
30-EOC	253.7	281.8	279.9	279.5	196	119	155	155
7X (BOC)	253.5	291.3	282.1	291.8	197	159	168	109

^a Cores during transition cycles are identified as N-XXX, where N is the cycle number and XXX is either BOC or EOC. The core labeled 7X is the reference equilibrium LEU core at BOC. The fuel element burnup conditions for all cores at BOC are summarized in Table 3.4.

core state. While the plate 23 maximum local heat flux during all cycles is greatest at BOC when the control blades are most deeply inserted, the very deeply inserted control blade positions at BOC for Cycle 1 (32.0 cm (12.6 inch)) result in less energy deposition and lower enthalpy rise along the length of the coolant channel compared with the EOC core state. Consequently, the coolant temperature and the corresponding fuel temperature is greater for the EOC core state of Cycle 1. Nonetheless, the margin to the FTSL is much larger at all core states during Cycle 1 in comparison to cycles later in the transition sequence and for the reference equilibrium LEU core.

For the reference equilibrium LEU core (cells shaded in gray), the maximum fuel temperature during the 0.6% $\Delta k/k$ step RIA is 291.8 °C and occurs in plate 23 in an element with a burnup of 180 MWd. Although the maximum fuel temperature in this case is 3.2 °C lower than that for Cycle 30, the greater fuel burnup of the element and maximum fission density in plate 23 in the equilibrium LEU core reduces the FTSL. The margin to the FTSL for the equilibrium LEU core is 109 °C. Thus, the equilibrium LEU core results in the smallest margin to the FTSL among all the cores evaluated in this work from initial conversion to LEU through the transition cycles to equilibrium operations. The minimum margin of 109 °C to the FTSL for the limiting 0.6% $\Delta k/k$ step RIA is considered an adequate margin to safety for all postulated RIA.

5.4 Loss-of-Coolant and Loss-of-Flow Accidents

Safety margins for postulated LOCAs and LOFAs during the transition period were evaluated using RELAP5/MOD3.3, Patch 05 [35]. The software has undergone a verification and validation (V&V) process to be NQA-1 compliant [36].

The same modeling approaches were used in this analysis as those for the previous LOCAs and LOFAs analyses for the preliminary SAR [5] and the Mo-99 upgrade safety analysis [7]. Detailed descriptions of the modeling of these accidents are provided in those references. The only differences from the previous models compared to the current work are those caused by the differences in power distribution and fuel element burnup/plate fission density. The power distribution directly affects the local heat flux. The detailed power distribution used as an input to the transient simulation for each core was obtained from the critical MCNP5 neutronics models described in Section 3. Fuel element burnup and plate fission density influence the thermal hydraulics by oxide layer growth, coolant channel reduction, decreased fuel thermal conductivity, and decreased FTSL. The RELAP5 models for the LOCAs and LOFAs represented all fuel plates in the elements in the core. For simplicity of the model development and for added conservatism in the evaluation of the fuel temperature during the postulated accidents, the U-10Mo thermal conductivity for all fuel plates in an element was determined based on the maximum fission density for fuel plate 1 in the element. This plate has the largest fission density among all the plates in an LEU element, and consequently the smallest thermal conductivity.

As discussed in Section 5.1, the effect of flow redistribution due to coolant channel reduction caused by fuel swelling, irradiation-induced creep, and oxide layer growth was also considered for LOCA and LOFA analyses. Because flow redistribution among the elements with different amounts of burnup/channel reduction may actually predict higher safety margins for those fuel elements with lower burnup, the safety analysis for LOCAs and LOFAs was completed twice to assess the minimum safety margin; first with flow redistribution caused by burnup-related effects and second without flow redistribution.

For each considered accident scenario, the maximum fuel temperature during the accident transients was evaluated. The safety margin for each scenario was then determined by calculating the margin to the FTSL from the maximum fuel temperature achieved during the accident. The FTSL limit was conservatively determined for each plate in each element using the maximum fission density in the fuel plate using a conservative empirical relation discussed in Section 2 of Reference 13. The maximum LEU fuel temperature safety limit is 450 °C (842 °F) for fresh fuel and decreases as the fuel burnup increases.

5.4.1 Accidents to Be Considered

Figure 5.2 provides a schematic view of the MURR primary system loop. The figure also shows the locations of postulated failures leading to transient accident conditions; green arrows for LOCAs and yellow highlights for LOFAs. Each pipe break point is modeled as a hypothetical valve connected to a fixed-pressure air sink at the local atmospheric pressure (14.3 psia). The valve is initially fully closed and becomes instantaneously fully open at the onset of the accident transient, simulating a pipe break to an ambient condition.

The same set of LOCA and LOFA scenarios that were analyzed in previous accident analyses [5,7] were evaluated in this work. For the LOCAs, the following four postulated scenarios were evaluated:

1. Cold-leg LOCA – double-ended 12-inch diameter break at the core side of the cold-leg isolation valve (V507B).
2. Cold-leg LOCA – 8-inch diameter breach in the 8-inch cold-leg piping at the flow control diaphragm valve (near V540).

Figure 5.2. Schematic View of MURR Primary Loop.

3. 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 (near V527).
4. Hot-leg LOCA – double-ended 12-inch diameter break at the core side of the hot-leg isolation valve (V507A).

The postulated LOFA scenarios that were evaluated are as follows:

1. Loss of facility electrical power (or coolant circulation pump power).
2. Inadvertent closure of coolant loop isolation valves (V507A and V507B).
3. Locked rotor in one of two primary coolant circulation pumps.

A summary of the initial conditions for LOCA and LOFA simulations can be found in Table 5.1. The initial steady-state conditions for the primary coolant flow rate, reactor inlet temperature, pressurizer pressure, and pool temperature prior to the LOCAs and LOFAs were assumed to be the LSSS conditions. The pressurizer level was set at the scram set point, and the pressure of the anti-siphon tank was assumed to be at the midrange of the operating band. The temperature of the anti-siphon tank was assumed to equal that of the pool. Lastly, the reactor power was assumed to be at the full licensed power. From the initial steady-state conditions, null transients were simulated for

200 seconds before the failure was initiated at $t = 0$ s (i.e., $-200 \text{ s} \leq t < 0 \text{ s}$ for null transients and $t \geq 0 \text{ s}$ for the actual accident transients).

The same set of cores as evaluated for the RIA were evaluated for the LOCAs and LOFAs. For each core there will be four pairs of fuel elements. For each pair, the fuel elements share practically the same burnup, resulting in four different element burnups present in a given core. As described in Reference 13, a one-element model was used to simulate LOCAs and LOFAs with RELAP5. Thus, four simulations must be conducted to represent each core. Then, the same set of simulations was repeated with and without flow redistribution due to burnup-related effects. Therefore, total of 784 scenarios were modeled and simulated with RELAP5 (7 accidents $\times$ 14 cores $\times$ 4 elements $\times$ 2 flow redistribution options).

Flow redistribution due to burnup was addressed via a separate hydraulics model in the same manner as the previous transient accident analyses [7,13]. The magnitude of the coolant channel reduction was modeled as a linear function of burnup starting from no reduction for a fresh fuel (0 MWd burnup) reaching 0.008 inch reduction at the maximum burnup (discharge burnup of 180 MWd). For scenarios without the effect of flow redistribution, no coolant channel reduction was assumed for all fuel elements, regardless of their burnup.

LOCA and LOFA transients initiate from one or more system failures that cause loss of coolant or flow. Following the failure(s) and the reactor trip settings initiate a control blade scram, the reactor power is reduced rapidly, which is input to RELAP5 as table of the power versus time. Figure 5.3 shows the relative reactor power versus time after scram for the fresh LEU and reference equilibrium LEU cores. The curves presented in Figure 5.3 were calculated based on PARET/ANL analyses of cores operating at steady-state followed by scram. In addition, the residual power after scram is also affected by photoneutrons coming from the interaction of fission-product gammas with the beryllium reflector. In previous work, the residual power with time after scram with the increased delayed neutron fraction from the photoneutrons was compared with the power with time after scram without photoneutrons [13]. Based on that comparison, the relative power after scram calculated by PARET/ANL was conservatively increased by 2% to account for the effect of photoneutrons [37] plus a small calculational uncertainty, but not allowing the relative reactor power to exceed 100% of the steady-state power. The results presented in Figure 5.3 show that the post-scram power decreases more rapidly for the transition cores than for the equilibrium LEU core. Therefore, the post-scram power for Core 7X (equilibrium LEU core at BOC) was used for the LOCA and LOFA analysis of all BOC core states and the for Core 9X (equilibrium LEU core at EOC) curve was used for all EOC core states.

5.4.2 LOCA and LOFA Analyses Results

From the RELAP5 simulation results for the LOCA and LOFA transients, it was observed that the cold-leg double-ended LOCA is always the most limiting accident scenario among all seven postulated accidents of these types. For a given core and fuel element, the safety margin for cold-leg double-ended LOCA scenario is always at least 100 °C lower than the margins for the other six accident scenarios. This observation agrees with previous LOCA and LOFA analyses in Reference 7 and 13. Therefore, only the results for cold-leg double-ended LOCA scenario are presented in the tables below.

Table 5.4 summarizes the results for cold-leg double-ended LOCA for the selected transition cycles and the equilibrium LEU core with the effect of flow redistribution due to burnup-related effects (14 cores $\times$ 4 elements = 56 cases). The results for the same accident scenarios without the effect of flow redistribution due to burnup (also 56 cases) are summarized in Table 5.5. In both tables, the results for the equilibrium LEU core are shaded gray.

Figure 5.3. Reactor Relative Power After Scram from Steady-State for Fresh LEU and Reference Equilibrium LEU Cores.

Table 5.4. Cold-Leg Double-Ended LOCA Results with Flow Redistribution due to Burnup.

Core	Element Pair (Burnup, MWd)	Steady-State Temp. at Maximum Location, °C (°F)	Maximum Fuel Temp, °C (°F)	Location of Maximum Temp., Plate/Axial	Minimum Margin to FTSL, °C (°F)	Location of Minimum Margin, Plate/Axial
1-BOC	X1/X5 (0)	135 (275)	310 (590)	3/3	140 (252)	3/3
	X2/X6 (0)	137 (278)	328 (623)	3/3	122 (219)	3/3
	X3/X7 (0)	136 (277)	296 (564)	3/3	154 (278)	3/3
	X4/X8 (0)	136 (276)	285 (545)	3/3	165 (297)	3/3
1-EOC	X1/X5 (10)	136 (277)	318 (604)	3/3	132 (238)	3/3
	X2/X6 (10)	136 (276)	300 (572)	3/3	150 (270)	3/3
	X3/X7 (10)	136 (276)	309 (589)	3/3	141 (253)	3/3
	X4/X8 (10)	136 (277)	301 (573)	3/3	150 (269)	3/3
9-BOC	X1/X5 (9)	138 (280)	315 (598)	3/3	135 (244)	3/3
	X2/X6 (10)	138 (281)	290 (554)	3/3	160 (288)	3/3
	X3/X7 (10)	138 (280)	319 (606)	3/3	131 (236)	3/3
	X4/X8 (10)	138 (281)	280 (536)	3/3	170 (306)	3/3
11-BOC	X1/X5 (19)	138 (281)	302 (575)	3/3	148 (267)	3/3
	X2/X6 (19)	140 (283)	298 (568)	3/3	152 (274)	3/3
	X3/X7 (19)	140 (284)	313 (596)	3/3	137 (246)	3/3
	X4/X8 (66)	148 (298)	297 (566)	4/3	153 (276)	4/3
15-BOC	X1/X5 (0)	135 (275)	329 (624)	3/3	121 (218)	3/3
	X2/X6 (85)	150 (302)	315 (598)	4/3	135 (244)	4/3
	X3/X7 (38)	142 (287)	279 (535)	3/3	171 (307)	3/3
	X4/X8 (38)	142 (287)	317 (602)	3/3	133 (240)	3/3

Core	Element Pair (Burnup, MWd)	Steady-State Temp. at Maximum Location, °C (°F)	Maximum Fuel Temp, °C (°F)	Location of Maximum Temp., Plate/Axial	Minimum Margin to FTSL, °C (°F)	Location of Minimum Margin, Plate/Axial
17-BOC	X1/X5 (38)	144 (291)	304 (580)	4/3	146 (262)	4/3
	X2/X6 (48)	146 (295)	323 (614)	4/3	127 (228)	4/3
	X3/X7 (94)	150 (303)	312 (594)	4/3	138 (248)	4/3
	X4/X8 (47)	146 (294)	320 (607)	4/3	131 (235)	4/3
19-BOC	X1/X5 (48)	146 (294)	327 (621)	4/3	123 (221)	4/3
	X2/X6 (57)	147 (297)	312 (594)	4/3	138 (248)	4/3
	X3/X7 (104)	150 (303)	300 (572)	4/3	150 (270)	4/3
	X4/X8 (57)	146 (295)	318 (604)	4/3	132 (238)	4/3
22-BOC	X1/X5 (19)	138 (280)	293 (559)	3/3	157 (283)	3/3
	X2/X6 (123)	152 (306)	302 (576)	4/3	148 (266)	4/3
	X3/X7 (123)	151 (304)	315 (600)	4/3	135 (242)	4/3
	X4/X8 (47)	145 (293)	311 (592)	4/3	139 (250)	4/3
23-BOC	X1/X5 (66)	147 (296)	301 (573)	4/3	149 (269)	4/3
	X2/X6 (122)	151 (304)	307 (585)	4/3	143 (257)	4/3
	X3/X7 (76)	148 (298)	301 (573)	4/3	149 (269)	4/3
	X4/X8 (76)	150 (303)	299 (571)	22/3	151 (271)	22/3
27-BOC	X1/X5 (10)	137 (278)	319 (606)	3/3	131 (236)	3/3
	X2/X6 (85)	149 (299)	303 (578)	4/3	147 (264)	4/3
	X3/X7 (142)	153 (308)	307 (585)	4/3	136 (245)	4/3
	X4/X8 (66)	147 (296)	314 (597)	4/3	136 (245)	4/3
28-BOC	X1/X5 (10)	136 (276)	305 (581)	3/3	145 (261)	3/3
	X2/X6 (142)	156 (312)	300 (572)	22/3	131 (236)	22/3
	X3/X7 (48)	145 (293)	303 (577)	4/3	147 (265)	4/3
	X4/X8 (140)	153 (307)	314 (597)	4/3	136 (245)	4/3
30-BOC	X1/X5 (0)	134 (273)	296 (565)	3/3	154 (277)	3/3
	X2/X6 (151)	154 (310)	305 (581)	4/3	133 (240)	4/3
	X3/X7 (95)	150 (301)	312 (594)	4/3	138 (248)	4/3
	X4/X8 (95)	150 (302)	321 (610)	4/3	129 (232)	4/3
30-EOC	X1/X5 (10)	138 (281)	291 (556)	22/3	159 (286)	22/3
	X2/X6 (104)	147 (296)	264 (507)	22/3	186 (335)	22/3
	X3/X7 (160)	153 (307)	298 (568)	22/3	123 (222)	22/3
	X4/X8 (104)	148 (298)	263 (505)	22/3	187 (337)	22/3
7X (BOC)	X1/X5 (0)	135 (274)	306 (582)	3/3	144 (260)	3/3
	X2/X6 (96)	149 (299)	305 (580)	4/3	145 (262)	4/3
	X3/X7 (77)	147 (297)	301 (574)	4/3	149 (268)	4/3
	X4/X8 (180)	156 (312)	316 (600)	4/3	111 (200)	4/3

Table 5.5. Cold-Leg Double-Ended LOCA Results without Flow Redistribution due to Burnup.

Core	Element Pair (Burnup, MWd)	Steady-State Temp. at Maximum Location, °C (°F)	Maximum Fuel Temp., °C (°F)	Location of Maximum Temp., Plate/Axial	Minimum Margin to FTSL, °C (°F)	Location of Minimum Margin, Plate/Axial
1-BOC	X1/X5 (0)	135 (275)	309 (587)	3/3	141 (255)	3/3
	X2/X6 (0)	137 (278)	321 (609)	3/3	129 (233)	3/3
	X3/X7 (0)	139 (282)	304 (580)	4/3	146 (262)	4/3
	X4/X8 (0)	136 (276)	299 (570)	3/3	151 (272)	3/3
1-EOC	X1/X5 (10)	136 (277)	307 (585)	3/3	143 (257)	3/3
	X2/X6 (10)	136 (277)	300 (572)	3/3	150 (270)	3/3
	X3/X7 (10)	141 (285)	303 (577)	22/3	147 (265)	22/3
	X4/X8 (10)	136 (277)	303 (577)	3/3	147 (265)	3/3
9-BOC	X1/X5 (9)	138 (280)	282 (539)	3/3	168 (303)	3/3
	X2/X6 (10)	141 (287)	267 (513)	4/3	183 (329)	4/3
	X3/X7 (10)	141 (286)	303 (577)	4/3	147 (265)	4/3
	X4/X8 (10)	138 (281)	286 (548)	3/3	164 (294)	3/3
11-BOC	X1/X5 (19)	139 (283)	289 (552)	3/3	161 (290)	3/3
	X2/X6 (19)	140 (285)	276 (529)	3/3	174 (313)	3/3
	X3/X7 (19)	141 (285)	313 (595)	3/3	137 (247)	3/3
	X4/X8 (66)	148 (298)	311 (592)	4/3	139 (250)	4/3
15-BOC	X1/X5 (0)	139 (282)	297 (567)	4/3	153 (275)	4/3
	X2/X6 (85)	150 (302)	312 (593)	4/3	139 (249)	4/3
	X3/X7 (38)	142 (288)	273 (524)	3/3	177 (318)	3/3
	X4/X8 (38)	143 (289)	307 (585)	3/3	143 (257)	3/3
17-BOC	X1/X5 (38)	145 (293)	309 (588)	4/3	141 (254)	4/3
	X2/X6 (48)	148 (298)	327 (621)	4/3	123 (221)	4/3
	X3/X7 (94)	151 (304)	303 (578)	4/3	147 (264)	4/3
	X4/X8 (47)	147 (296)	318 (605)	4/3	132 (237)	4/3
19-BOC	X1/X5 (48)	147 (297)	324 (614)	4/3	126 (228)	4/3
	X2/X6 (57)	148 (299)	320 (608)	4/3	130 (234)	4/3
	X3/X7 (104)	151 (304)	295 (564)	4/3	155 (278)	4/3
	X4/X8 (57)	148 (298)	314 (597)	4/3	136 (245)	4/3
22-BOC	X1/X5 (19)	140 (284)	315 (599)	3/3	135 (243)	3/3
	X2/X6 (123)	153 (308)	298 (569)	4/3	152 (273)	4/3
	X3/X7 (123)	156 (313)	288 (551)	22/3	162 (291)	22/3
	X4/X8 (47)	147 (296)	308 (587)	4/3	142 (255)	4/3
23-BOC	X1/X5 (66)	149 (299)	297 (567)	4/3	153 (275)	4/3
	X2/X6 (122)	156 (313)	294 (561)	22/3	156 (281)	22/3
	X3/X7 (76)	150 (301)	298 (569)	4/3	152 (273)	4/3
	X4/X8 (76)	152 (306)	299 (570)	22/3	151 (272)	22/3
27-BOC	X1/X5 (10)	139 (282)	325 (617)	3/3	125 (225)	3/3
	X2/X6 (85)	150 (302)	293 (559)	4/3	157 (283)	4/3
	X3/X7 (142)	154 (309)	287 (549)	4/3	157 (282)	4/3
	X4/X8 (66)	148 (299)	313 (596)	4/3	137 (246)	4/3

Core	Element Pair (Burnup, MWd)	Steady-State Temp. at Maximum Location, °C (°F)	Maximum Fuel Temp., °C (°F)	Location of Maximum Temp., Plate/Axial	Minimum Margin to FTSL, °C (°F)	Location of Minimum Margin, Plate/Axial
28-BOC	X1/X5 (10)	139 (281)	314 (597)	3/3	136 (245)	3/3
	X2/X6 (142)	157 (314)	282 (539)	22/3	150 (270)	22/3
	X3/X7 (48)	147 (297)	312 (593)	4/3	138 (249)	4/3
	X4/X8 (140)	156 (313)	280 (537)	22/3	152 (273)	22/3
30-BOC	X1/X5 (0)	137 (279)	321 (610)	3/3	129 (232)	3/3
	X2/X6 (151)	155 (311)	295 (562)	4/3	143 (258)	4/3
	X3/X7 (95)	151 (304)	318 (605)	4/3	132 (237)	4/3
	X4/X8 (95)	151 (304)	303 (577)	4/3	147 (265)	4/3
30-EOC	X1/X5 (10)	141 (286)	303 (578)	22/3	147 (264)	22/3
	X2/X6 (104)	148 (298)	244 (471)	22/3	206 (371)	22/3
	X3/X7 (160)	153 (308)	274 (525)	22/3	147 (265)	22/3
	X4/X8 (104)	150 (301)	258 (497)	22/3	192 (345)	22/3
7X (BOC)	X1/X5 (0)	138 (280)	335 (635)	3/3	115 (207)	3/3
	X2/X6 (96)	150 (302)	299 (570)	4/3	151 (272)	4/3
	X3/X7 (77)	149 (300)	315 (598)	4/3	135 (244)	4/3
	X4/X8 (180)	156 (313)	304 (579)	4/3	123 (221)	4/3

Each case for the transition cycles is labeled as N-XXX, where N represents the specific cycle during the transition period and XXX is either BOC or EOC. The reference equilibrium LEU core at BOC conditions is identified as Core 7X. MURR loads eight elements in four pairs, each pair containing two elements with same burnup (see Table 3.4) in core positions that are opposite each other (i.e., core position X1 is opposite X5, X2 opposite X6, X3 opposite X7, and X4 opposite X8). Finally, the core power distribution for each case was derived from the detailed three-dimensional MCNP5 analysis at the critical control blade position.

In both Table 5.4 and Table 5.5, the highest maximum fuel temperature and the lowest minimum margins to fuel temperature safety limit for the transition cycles and for the equilibrium LEU core are marked in red. The lowest margins for all cases for each flow redistribution option (with and without flow redistribution) are highlighted in green in each table. The locations where the maximum fuel centerline temperatures occur are also noted. There are total of 23 fuel plates per fuel element, and those plates are numbered sequentially from 1 to 23 for the innermost and outermost plates, respectively. Within each plate, there are four axial segments in the RELAP5 model that are numbered from the top to bottom of the fuel plate (i.e., the highest and lowest axial nodes are numbered as 1 and 4, respectively. Figure 5.10 from Reference 13 illustrates the one-element RELAP5 model by showing a schematic view of the discretization of the fuel plates and coolant channels.

For the transition cycles, the lowest safety margin to the FTSL occurs in the X1/X5 pair (0 MWd element burnup) in Core 15-BOC with the effect of flow redistribution due to burnup. The maximum fuel temperature for this scenario is 329 °C (624 °F) in a fresh plate 3, which gives a minimum margin to the fuel temperature safety limit of 121 °C (218 °F) in the same location. For the equilibrium LEU core, the lowest safety margin occurs in the X4/X8 pair (180 MWd element burnup) with the effect of flow redistribution due to burnup. The maximum fuel temperature for this scenario is 316 °C (600 °F) in plate 4, which gives a minimum margin to the FTSL of 111 °C (200 °F) in the same location. Note

that the minimum margin can occur in a location different from that of the maximum fuel temperature because the FTSL is burnup dependent. A fuel element with greater burnup may have a lower maximum fuel temperature than an element with a lower burnup, but will also have a lower FTSL, which may give a smaller safety margin than the element with lower burnup. More details of the behavior of the cold-leg double-ended break LOCA are presented in Section 4.3.3 of Reference 7.

As described in Section 5.4.1, a total of 784 LOCA and LOFA scenarios were modeled and simulated with RELAP5. As previously discussed, the cold-leg double-ended LOCA is by far the most severe among the seven LOCA and LOFA scenarios considered. The maximum fuel temperature for this accident without flow redistribution due to channel reduction is 335 °C (635 °F) in a fresh plate 3. The maximum fuel temperatures for this accident with flow redistribution is 329 °C (624 °F) in a fresh plate 3. The maximum fuel temperature for a fuel plate in an element with non-zero burnup is 327 °C (621 °F) in plate 4 of an element with a burnup of 48 MWd (Core 17-BOC; without flow redistribution). For accidents other than the double-ended cold-leg LOCA, the maximum fuel temperatures are significantly lower than these values for fresh and irradiated fuel.

The margin to the FTSL is a key figure-of-merit for assessing reactor safety in the event of a postulated accident. The safety margin results for all 784 cases considered for the LOCAs and LOFAs are shown in Figure 5.4. For each transition cycle and the reference equilibrium LEU core, the minimum margins to the FTSL are the smallest for the cold-leg double-ended LOCA. All other accidents result in significantly larger margins. Also, as shown in Figure 5.4, the minimum margin to the FTSL for the equilibrium LEU core is the lowest of all cases, confirming that the safety margins for the transition cycles are bounded by the equilibrium LEU core.

Figure 5.4. Minimum Margin to Fuel Temperature Safety Limit for all 784 Cases Considered for the LOCAs and LOFAs.

6 CONCLUSIONS

Working with the RC Pillar at Argonne, MURR is completing analyses that are expected to support conversion from current operations with HEU fuel at 10 MW to LEU operations with a very high density U-10Mo monolithic fuel and a preliminary LEU fuel element design. Previous work found that for equilibrium LEU operations with the preliminary LEU fuel element design, in conjunction with a power uprate to 12 MW and appropriate changes to the MURR LSSS conditions, MURR will have adequate margins to safety for steady-state operations and postulated transient accidents and will also meet or exceed experimental performance relative to current prototypic operations with HEU fuel [5,7]. The purpose of this work was to develop a sequence of transition cycles that will enable MURR to transition from the operation with the reactor core loaded with fresh LEU fuel elements only to typical equilibrium operations with mixed-burnup cores following conversion while meeting operational requirements on safety and experimental performance. It is expected that the use of fresh LEU fuel at conversion and subsequent low burnup of the LEU fuel elements that will initially be available for use following conversion will result in critical control blade positions that will substantially change the axial power distribution in the core and the neutron flux available in key experimental locations relative to equilibrium LEU operations.

Given the constraints of MURR safety margins, operational practices, and production and research, a novel method was developed to identify a transition sequence that minimizes the time MURR operates atypically compared to the current prototypic cycles currently using HEU fuel. The proposed transition sequence moves quickly to the same sort of equilibrium cycles for the LEU fuel that have already been evaluated in documented preliminary safety analyses [5]. Although shifting the neutron flux peak to the lower half of the core during initial cycles with LEU at 12 MW reduces the experiment performance in some key locations relative to current HEU operations at 10 MW, all LEU cores provide an average performance that meets or exceeds that of HEU. An LEU cycle is reached that meets or exceeds the level of experimental performance predicted for current HEU and equilibrium LEU operations in more than 450 key locations by the 23rd cycle following conversion. Following this, the results show that the core burnup for future idealized cycles is within a narrow band of a target burnup based on the equilibrium LEU core for all but a few cycles that will enable MURR to consistently meet its experimental performance requirements.

The proposed transition sequence only requires the fabrication of 34 fresh LEU elements in the first year of operation and does not exceed the anticipated availability of fresh elements that can be produced by the fuel fabricator. By the third year after conversion, 22 fresh LEU elements will be required each year, which is the same as expected for equilibrium LEU operations and the same as current operations with HEU fuel. The proposed transition sequence developed in this work is an improvement over prior work that developed a transition sequence that required 36 fresh LEU elements in the first year, 12 of which were specialized elements with borated side plates that would require an additional material qualification effort [14].

This analysis has shown that there will be an experimental performance penalty in some key locations during the early transition cycles until the core burnup reaches the equivalent core burnup for equilibrium cycles. Measures may be taken to reduce this experimental performance penalty even further, where possible. For example, it may be possible to compensate for the decreased neutron flux in the upper axial positions of the core by repositioning certain experiments to leverage the increased performance in the lower axial positions of the reactor. Alternatively, some experiments could remain in the irradiation facilities longer in order to achieve the required neutron fluence. Note that the experiments modeled in this work are based on typical experiment facilities and loadings in 2018 and may not be the experiments in the reactor when MURR converts to LEU fuel. Consequently, this

analysis may require refinement depending on the experimental facilities in use at the time of conversion. Nonetheless, the results presented here, including the experimental performance, core burnup, and critical control blade positions throughout the transition cycles, show that the proposed transition cycle fuel management patterns are consistent with what is expected and desired for MURR operation with LEU U-10Mo fuel. The proposed transition sequence combines a relatively short time period before equilibrium burnup is achieved, a temporary increase of fuel elements needed annually relative to typical operations that are within the production capabilities of the fuel fabricator, and demonstrates comparable experimental performance of the LEU cores relative to current HEU operations.

Core power distributions from the neutronics models were also used to evaluate safety margins during steady-state operations for the selected transition cycles and the equilibrium LEU core. The margin to OFI at LSSS conditions for the coolant flow rate and inlet temperature was evaluated. The minimum margin to OFI for the transition cycles is 2.49 MW greater than an LSSS power of 15 MW for Cycle 28 at EOC with an extreme control blade configuration at the limits of the MURR Technical Specifications. The minimum margin for the transition cycles is similar to the 2.52 MW margin to OFI for the equilibrium LEU core at EOC conditions. The CHFR was evaluated at the LSSS conditions for power, coolant flow rate, and coolant inlet temperature. For all cases, the CHFR is predicted to exceed 2.0, which is a minimum requirement from NUREG-1537 [20]. The minimum CHFR for the transition cycles and the equilibrium LEU cores is 2.67 and 2.53, respectively. From these results, it can be concluded that the safety margins during steady-state operations for the transition cycles are at least very close and for most cases larger than those for the equilibrium LEU core.

Safety margins were also evaluated for postulated transient accidents (RIA, LOCA, and LOFA) based on a burnup dependent FTSL. The FTSL is based on the lower 95% prediction bound for preliminary blister anneal tests performed for U-10Mo monolithic fuel plates [13]. For the RIAs, a 0.6% $\Delta k/k$ step insertion from 115% of full licensed power was assumed. For the LOCA and LOFA, the same set of accidents evaluated in the previous analyses [7,13] were evaluated (four LOCA and three LOFA scenarios initiated from full licensed power). A summary of the initial conditions for the accident analyses can be found in Table 5.1.

For the RIAs, the minimum margin to the FTSL for all transition cycles is 110 °C and occurs in Cycle 30. For the equilibrium LEU core, the minimum margin to the FTSL is 109 °C. As discussed in Section 5.4.2, the cold-leg double-ended LOCA is by far the most severe accident among the seven postulated LOCA and LOFA scenarios considered. For the transition cycles, the minimum margin is 121 °C and occurs in Cycle 15. The minimum margin to the FTSL is 111 °C for the equilibrium LEU core. For all postulated accidents scenarios, whether RIA, LOCA, or LOFA, it is observed that the safety margins for the transition cycles are bounded by those for the equilibrium LEU core.

In summary, it has been shown that there are adequate safety margins for both steady-state operations and postulated accident scenarios. For the steady-state operations with the preliminary LEU fuel element design the analysis predicts at least 2.49 MW margin to OFI at the LSSS power of 15 MW. Considering the LSSS power is 125% of full license power, the margin to OFI is sufficient. In addition, CHFR at LSSS power is well above the 2.0 requirement from NUREG-1537 for all considered cases. For postulated transient accidents, the minimum margin to the FTSL is at least 109 °C. Given the level of conservatism included in modeling the accidents, safety margins over 100 °C can be considered sufficient.

In summary, the proposed sequence of core loadings for MURR operations following conversion to LEU fuel and a power uprate to 12 MW, and transitioning to equilibrium operations, is not only

suitable to produce satisfactory experimental performance, despite some local penalties for the initial cycles, but also provides sufficient safety margins for both steady-state operations and postulated transient accidents. The results also indicate that the experimental performance for the equilibrium LEU core exceeds that of current HEU operations while showing sufficient safety margins. This again provides confidence that the transition to LEU operations can be achieved with minimal or no impact to the performance and safety.

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APPENDIX A: Plate Maximum Fission Density for Elements Loaded in Transition Cycles and Reference Equilibrium LEU Core

Table 3.4 summarizes the burnup of the elements loaded in each of the transition cycles and the reference equilibrium LEU core selected for detailed transient accident analysis. The fuel burnup and time of in-reactor use affects fuel property and fuel performance data, which impact the fuel temperature and the margin to the fuel temperature safety limit (FTSL) during postulated transient accidents. This appendix summarizes the methodology used for estimating the maximum fission density for each plate in the elements in the accident models. The fission density is used for predictions of the FTSL and the fuel thermal conductivity needed for the accident analyses.

The fuel cycle simulation completed with REBUS-PC relied on models that represented each plate in the fuel elements loaded in the core with 12 axial divisions to account for influence of axial peaking on the plate burnup. Lateral peaking in the fuel plates was not included in the REBUS-PC models because this is a small effect relative to the axial power peaking (see discussion in Section 3.1).

The axial burnup profile in the MURR fuel plates is affected by the critical control blade position during operation. In the REBUS-PC models, the control blades were held at a fixed position of 58.4 cm (23.0 inch), which is the typical position for the control blades from equilibrium xenon conditions to EOC, and is representative of more than 70% of the period during which the elements are used in the core. For the modeling of the plate fission density for the elements in the equilibrium LEU fuel cycle, the axial burnup profile in each plate from the REBUS-PC model was multiplied by a lateral peaking factor derived from an explicit MCNP5 model of the fuel elements in the reactor core. Table A.1 provides the maximum fission density calculated in this way at BOC of the equilibrium LEU core by plate for each element in the core. The fission density values for the element in core positions X4/X8 at the EOL burnup of 180 MWd are the same as those in Table 3.8 under the heading “Eq. LEU.”

During the transition cycles prior to reaching equilibrium operations with the LEU fuel, the control blade positions will be more deeply inserted than during prototypic equilibrium operations because of the lower total burnup of the elements loaded in these cycles. As a result, the axial peak-to-average plate burnup for these elements is somewhat greater than for an element that is used during equilibrium operations. The greatest impact on the axial peaking occurs for those plates closest to the control blades (e.g., plate 23). This can be seen in Figure A.1, which presents the axial power profile in plate 23 for the elements loaded in Cycle 1, Cycle 15, and the reference equilibrium LEU core at equilibrium xenon conditions. The “average maximum peaking factor” for each core presented in Figure A.1 is the average of plate 23 peaking factors for all eight elements in the core. The total BOC burnup of the elements loaded in these cores is 0, 322, and 687 MWd for Cycle 1, Cycle 15, and the equilibrium LEU core, respectively, which has a direct impact on the critical control blade positions and the axial peaking factors. The axial peaking factor for plate 23 in Cycle 1 is 26% greater than that for the equilibrium LEU core. For Cycle 15, the axial peaking factor for plate 23 is 13% greater than the equilibrium LEU core. For all other plates in the fuel elements, the differing critical positions of the control blades in these transition cycles has less of an effect on the axial peaking factor relative to the equilibrium LEU core, but the differences are still on the order of > 10% for Cycle 1 and > 5% for Cycle 15. The critical positions of the control blades have a relatively smaller impact ($\leq 1\%$) on the lateral peaking factors for the fuel plates in the elements.

The results presented in Table A.1 indicate that the maximum fission density for each plate in an element with a given burnup which has been used during the transition cycles will be greater than for an element that has reached the same burnup while loaded in the equilibrium fuel cycle because of the increased axial peaking factors. The first elements loaded following conversion will be used in cores with a total burnup ranging from 0 to 687 MWd (the equilibrium core burnup). Consequently, with a core burnup of 322 MWd, Cycle 15 represents an approximate average of the conditions these

elements are expected to experience during their use. The axial peaking factors for each plate in Cycle 15 relative to those for the equilibrium cycle are summarized in Table A.2.

Table A.1. Maximum Fission Density for Each Plate in LEU Elements Loaded in Reference Equilibrium LEU Core.

Core Positions of Elements	X1/X5	X2/X6	X3/X7	X4/X8	
Element Burnup, MWd	0.0 ^a	96.0 ^a	77.0 ^a	170.4 ^a	180.0 ^b
Plate	Maximum Fission Density, 10 ²¹ fissions/cm ³				
1	0.00	1.92	1.57	3.17	3.31
2	0.00	1.52	1.23	2.61	2.74
3	0.00	1.21	0.971	2.11	2.22
4	0.00	0.978	0.785	1.73	1.83
5	0.00	0.849	0.681	1.51	1.59
6	0.00	0.777	0.624	1.38	1.46
7	0.00	0.716	0.574	1.27	1.34
8	0.00	0.671	0.538	1.19	1.26
9	0.00	0.642	0.515	1.14	1.20
10	0.00	0.619	0.497	1.10	1.16
11	0.00	0.601	0.483	1.07	1.13
12	0.00	0.591	0.475	1.05	1.11
13	0.00	0.597	0.480	1.06	1.12
14	0.00	0.595	0.478	1.06	1.11
15	0.00	0.597	0.479	1.06	1.12
16	0.00	0.605	0.486	1.08	1.14
17	0.00	0.624	0.500	1.11	1.17
18	0.00	0.650	0.521	1.16	1.22
19	0.00	0.699	0.559	1.25	1.32
20	0.00	0.763	0.611	1.37	1.45
21	0.00	0.872	0.697	1.56	1.65
22	0.00	1.06	0.844	1.89	1.99
23	0.00	1.36	1.09	2.38	2.51

^a Values presented for element at BOC core state.

^b Values presented for element at EOC core state, which is EOL for the elements in core positions X4 and X8. Although accident analysis was completed for the BOC core state, the EOL fission density was used for determining the thermal conductivity and fuel temperature safety limit of this fuel element for added conservatism.

To estimate the maximum fission density by plate for elements loaded in the transition cycles, the maximum fission density for each plate in Table A.1 was first fit as a second-order polynomial as a function of fuel element burnup. The fit for each plate in each element in the selected transition cycle was evaluated at the element burnup (element burnup values for each element in the transition cycles selected for analysis are summarized in Table 3.4) and then multiplied by the plate axial peaking factor from Table A.2 to account for the effect of the control blade position in the transition cycles on the maximum fission density. The application of this methodology yields a conservative estimate of the maximum fission density by plate for elements loaded in the transition cycles for the modeling of the transient accidents. For example, the estimated maximum fission densities for plates 1 and 23 in an element that reaches an element burnup of 180 MWd during the transition cycles are 3.52×10^{21} and 2.82×10^{21} fissions/cm³, respectively. These values are greater than the plate 1 and 23 fission

densities calculated for the plate burnup in elements discharged from the transition cycles that accounts for the axial peaking in each cycle the elements are used (see Table 3.8).

Figure A.1. Axial Peaking Factor in LEU Plate 23 at Equilibrium Xenon Conditions in Cycle 1 (a), Cycle 15 (b), and Reference Equilibrium LEU Core (c).

Table A.2. Axial Peaking Factors by Plate for LEU Elements in Cores at Equilibrium Xenon Conditions.

Plate	Cycle 15 Core (322 MWd)	Eq. LEU Core (687 MWd)	Cycle 15 rel. to Eq. LEU
1	1.326	1.250	1.061
2	1.335	1.267	1.054
3	1.337	1.273	1.050
4	1.337	1.281	1.044
5	1.347	1.287	1.047
6	1.349	1.285	1.050
7	1.348	1.285	1.049
8	1.350	1.285	1.051
9	1.353	1.286	1.052
10	1.353	1.289	1.049
11	1.358	1.294	1.050
12	1.361	1.289	1.056
13	1.362	1.293	1.054
14	1.371	1.294	1.059
15	1.375	1.294	1.063
16	1.380	1.306	1.057
17	1.387	1.306	1.062
18	1.401	1.307	1.072
19	1.416	1.316	1.076
20	1.431	1.326	1.079
21	1.459	1.338	1.090
22	1.488	1.344	1.107
23	1.515	1.345	1.126

APPENDIX B: MURR Modified Griess Correlation

Griess, Savage, and England conducted a series of tests to observe the oxide layer growth on heated plates fabricated with aluminum alloy 6061 (AA6061) and aluminum alloy X-8001 [33]. A correlation of the oxide layer thickness as a function of time and temperature was correlated as follows:

$$T_{oxide} = 443 \times \theta^{0.778} \times \exp\left(\frac{-4600}{K}\right) \quad (B.1)$$

where T_{oxide} is the oxide thickness in mil (1 mil = 0.001 inch), θ is the time in hours that the plate surface is heated, and K is the plate surface temperature in degrees Kelvin. The correlation given in Equation B.1 is known as the “Griess correlation” and was found to be applicable to both aluminum alloys tested in Reference 33 and to aluminum alloy 1100 which was tested in previous work by Griess [38]. MURR fuel plates are fabricated with AA6061 cladding.

The Griess correlation was developed for tests with water at a coolant pH of 5.0. At pH values in the range of 5.7 to 7.0, the oxide growth was found to be greater than that predicted by Equation B.1 by a factor of up to 2.7 [33]. Regular measurements of the coolant pH at MURR have observed values in the range of 5.3 to 5.5 [34].

The effect of heat flux on the plate surface temperature is factored into Equation B.1. Griess found that the oxide growth rate was insensitive to heat flux for heat flux values in the range 300 to 600 W/cm², while observing smaller oxide growth rates when measurements were normalized as a function of temperature for heat flux values less than 300 W/cm². In Reference 10 it was reported that the maximum plate surface heat flux for a fresh fuel element in a typical MURR core is in the range of 170 to 185 W/cm² during the weekly operating cycle. For an element near its discharge burnup, the maximum plate surface heat flux is in the range of 105 to 115 W/cm².

Measurements of the oxide thickness on MURR HEU fuel plates were used to determine an empirical correction factor to the Griess correlation that accounts for the water chemistry and heat flux that are typical for MURR [34]. Measurements were taken on 17 different HEU fuel elements that were in use in 1987. Measurements were taken on the outer (convex) surface of plate 24 of the fuel elements at axial locations 6 inch, 12 inch, and 18 inch from the top of the fuel plate. The maximum measured oxide thickness was recorded for each element. For all but one of the fuel elements/plates the largest oxide thicknesses were measured at the 12 inch or 18 inch positions.

Oxide thickness measurement results collected in Reference 34 are summarized in Table B.1. For each measurement, the fuel element burnup, hours of operation, and maximum measured oxide thickness are provided. The number of hours for the element use is estimated assuming uniform power sharing among the eight HEU elements in a core operating at a nominal power of 10 MW, resulting in a burnup accumulation of 1.25 MWd/day (=10/8).

The Griess correlation relates the oxide thickness on the surface of heated plates with AA6061 cladding to the number of hours of use and the plate surface temperature. The PLTEMP/ANL v. 4.3 software [29] was used to calculate the temperature on the outer surface of plate 24 of the HEU elements in a core operating at a nominal power of 10 MW. MURR typically operates using a collection of eight elements with a mixture of burnups from at or near BOL to at or near EOL. A reference core with two fresh elements, four MOL elements, and two EOL elements was modeled to provide the range of conditions that are expected during typical operations. Figure B.1 presents the plate 24 surface temperatures for this reference Core 4A at equilibrium xenon conditions, which is representative of the conditions during more than 70% of the in-service use of the elements. The

plate surface temperature is observed to decrease by about 10 °C over the life of the element as the plate heat flux decreases with fuel burnup, which will affect the oxide growth rate.

Table B.1. Measurements of Oxide Thickness on Plate 24 of MURR HEU Elements [34].

Fuel Element	MWd	Hours ^a	Maximum Oxide Thickness (mil)
MO-133 ^b	148	2842	0.36
MO-135 ^b	148	2842	0.47
MO-143	100	1920	0.20
MO-145 ^b	148	2842	0.44
MO-146 ^b	149	2861	0.34
MO-146 ^{b,c}	149	2861	0.27
MO-147 ^b	148	2842	0.46
MO-148 ^b	149	2861	0.49
MO-200	150	2880	0.61
MO-201	148	2842	0.71
MO-203	148	2842	0.87
MO-203 ^c	148	2842	0.85
MO-206 ^b	155	2976	0.44
MO-216	110	2112	0.28
MO-218	110	2112	0.30
MO-226	100	1920	0.19
MO-228	74	1421	0.17
MO-239	70	1344	0.16
MO-246	9	173	0.06

^a Estimated based on burnup accumulation of 1.25 MWd/day.

^b Measurements considered most reliable and used for determining empirical adjustment to Griess correlation for oxide layer growth at typical operating conditions for MURR.

^c Measurement repeated on this element.

Figure B.1. Temperature on Outer Surface of Plate 24 in HEU Reference Core.

The Griess correlation (Equation B.1) was used to predict the maximum oxide thickness on the outer surface of plate 24 in a discharged HEU element for comparison with the measured data. The linear fit of the bounding plate surface temperature as a function of hours of operation shown in Figure B.1 was used. Because the plate surface temperature (parameter K in Equation B.1) is time-dependent, the calculation of the oxide thickness was completed over multiple time intervals from 0 to 2873 hours (the equivalent of 19 cycles of use to reach the element discharge burnup of 150 MWd). The results of this calculation are summarized in Table B.2. At the end of each interval, the equivalent number of hours to reach the predicted oxide thickness at the temperature corresponding to the next interval was calculated and used for the next step. For example, after 200 hours of operation at a temperature of 360.9 K, Equation B.1 predicts an oxide thickness of 0.080 mil. For the next interval of use from 200 to 400 hours, the plate surface temperature is 360.1 K. It would require 207.0 hours at 360.1 K to achieve an oxide thickness of 0.080 mil. The oxide thickness of 0.135 mil at the end of the second interval was thus calculated using a temperature of 360.1 K for 407 hours.

This methodology for evaluating the Griess correlation with time-dependent temperature on the plate surface is complex. As a check, if the plate surface temperature at 0 hours of 361.4 K (88.3 °C) was held constant for 2873 hours, the predicted oxide thickness from Equation B.1 would be 0.65 mil. If the expected EOL plate surface temperature of 350.5 K (77.4 °C) was held constant for 2873 hours, the predicted oxide thickness would be 0.43 mil. Based on the methodology described above and the results presented in Table B.2, the maximum oxide thickness on the outer surface of plate 24 in a discharged HEU element is predicted to be 0.53 mil, which is between these two extreme values.

Table B.2. Prediction of Oxide Thickness on Outer Surface of HEU Plate 24.

Interval, hours		Average Temperature over Interval, K	Oxide Thickness, mil	Eq. Time at temperature for next interval, hours
Start	End			
0	200	360.9	0.080	207.0
200	400	360.1	0.135	421.4
400	600	359.4	0.182	643.5
600	800	358.6	0.225	873.5
800	1000	357.9	0.264	1111.9
1000	1200	357.1	0.300	1359.1
1200	1400	356.3	0.334	1615.4
1400	1600	355.6	0.366	1881.2
1600	1800	354.8	0.396	2157.0
1800	2000	354.1	0.424	2443.2
2000	2200	353.3	0.451	2740.3
2200	2400	352.5	0.477	3048.8
2400	2600	351.8	0.501	3391.6
2600	2873	350.9	0.532	N/A

As discussed above, the Griess correlation is based on the conditions of a coolant pH of 5.0 and a plate heat flux between 300 and 600 W/cm². Other than using the maximum plate 24 surface temperature for typical operating conditions at MURR, the results presented above are calculated from the Griess correlation without any adjustment to account for the actual conditions of the MURR coolant pH or the plate heat flux. An adjustment factor on the Griess correlation to account for the conditions at MURR was empirically determined by comparing the oxide thickness predicted from Equation B.1 to the oxide thickness values measured at MURR.

Among the twelve measurements for those elements that were at the typical MURR HEU element discharge burnup of 150 MWd, the maximum measured oxide thickness ranged from a low value of 0.27 mil (element MO-146) to a maximum of 0.87 mil (element MO-203). In Reference 34, no relationship was found between the variations of the measured oxide thickness and any characteristics of the element use other than time of use in the operating core. The three discharged elements with the greatest measured oxide thickness in the range of 0.61 to 0.87 mil (MO-200, MO-201, and MO-203) were among the first to be fabricated at Babcock and Wilcox (B&W). All other elements fabricated by B&W do not exhibit a trend of having larger oxide thickness with use than the other elements which were fabricated at Atomics International. Consequently, a decision was made to exclude five measurements of the oxide thickness for the discharged elements as outliers, either because of uncharacteristically low or high oxide thickness measurements. Seven measurements were considered most reliable for determining an empirical adjustment to the Griess correlation to account for operating conditions in MURR, which are noted by gray shading in Table B.1. The average measured oxide thickness on the outer surface of plate 24 for these seven elements near the MURR typical discharge burnup is 0.429 mil.

An empirical adjustment factor to the Griess correlation of 0.806 was calculated by taking the ratio of the average measured oxide thickness and that predicted using Equation B.1. Thus, the MURR modified Griess correlation is

$$T_{oxide} = 0.806 \times 443 \times \theta^{0.778} \times \exp\left(\frac{-4600}{K}\right) \quad (B.2)$$

where T_{oxide} is the oxide thickness in mil, θ is the time in hours that the plate surface is heated, and K is the plate surface temperature in degrees Kelvin. The predicted oxide thickness as function of hours of operation from Equation B.2 is plotted alongside the measured data from Reference 34 in Figure B.2. Also included in Figure B.2 is the plate surface temperature derived from the PLTEMP/ANL analysis presented in Figure B.1. The predicted oxide thickness on the outer surface of plate 24 matches the average of those measurements selected as representative of typical operations at MURR. For elements with burnup lower than the discharge burnup, which include several measurements for elements fabricated by B&W, the predicted oxide thicknesses are conservative relative to measurements. The “Measured data considered as outliers” were not used in calculating the empirical adjustment factor of 0.806. If these measurement values were included in the derivation of the empirical adjustment factor, then the adjustment factor would be larger but the predicted oxide thicknesses could be overly conservative.

The thickness of the oxide layer on the surface of LEU fuel plates as a result of in-reactor use can also be predicted using the MURR modified Griess correlation (Equation B.2). The preliminary LEU fuel element design retains the same AA6061 cladding as the HEU fuel element design. The coolant pH is also expected to remain the same as current HEU operations. However, the preliminary LEU element design has one less fuel plate than the HEU element (23 fuel plates in the LEU element versus 24 in the HEU element) and a power uprate to 12 MW is planned that will allow MURR to meet or exceed its current experimental performance with HEU and 10 MW operations. These changes are expected to result in an increase in the maximum plate heat flux and plate surface temperature, which directly affect the oxide thickness growth rate. It has been shown through analysis for a reference LEU core representative of equilibrium operations with fresh and depleted elements that the maximum plate heat flux in a fresh element is in the range of 195 to 210 W/cm² during the weekly operating cycle, which is about 15% greater than for the equilibrium HEU core. For an LEU element near its discharge burnup, the maximum plate surface heat flux is in the range of 155 to 170 W/cm². For cores that will be loaded beginning at conversion with fresh LEU, the maximum plate heat flux is predicted to be

Figure B.2. Measured and Predicted Oxide Thickness on Outer Surface of HEU Plate 24.

somewhat greater than that for equilibrium operations. For Cycle 1, the maximum heat flux for fresh elements will be in the range of 210 to 220 W/cm². Safety analyses have been completed in this work that demonstrate that, along with appropriate changes to the MURR LSSS settings, there are adequate safety margins for steady-state operations and postulated transient accidents.

Figure B.3 shows the maximum cladding surface temperature predicted by PLTEMP/ANL on the inner (concave) surface of LEU plate 23 over the operating life of a fuel element under nominal steady-state conditions. This location was selected because it is where the greatest cladding surface temperatures and oxide thickness are anticipated. The temperature predictions are provided for elements loaded in selected transition cycles and in the equilibrium LEU core. The maximum cladding surface temperatures are greater for elements loaded in the transition cycles because the control blades are more deeply inserted, which increases the axial peaking of the plate heat flux. It is observed that the difference in the cladding surface temperature between the transition and equilibrium cycles decreases with time due to the increasing total core burnup which occurs as the elements loaded in the transition cycles increase in burnup. With increasing core burnup, the operating conditions for the transition cycles, especially the control blade positions which affect the axial power profile in plate 23, are more similar to that for the equilibrium core.

The peak plate surface temperatures over the life of a fuel element from the linear trendline fits in Figure B.3, in combination with the MURR modified Griess correlation (Equation B.2), were used to predict the thickness of the oxide layer on the inner surface of plate 23 over the life of an LEU fuel element in transition and equilibrium cycles. The predicted oxide layer thicknesses are shown in Figure B.4. The thickness of the oxide layer is greater for elements used in the transition cycles than for elements in equilibrium operation because the peak cladding temperatures are greater for transition cycles. For an element in equilibrium operations that has reached an EOL burnup of ~180 MWd (which occurs after about 2,900 hours of use), the oxide thickness on the inner surface of

Figure B.3. Peak Plate 23 Inner Surface Temperature over the Life of LEU Fuel Element.

Figure B.4. Thickness of the Oxide Layer on Inner Surface of Plate 23 Over the Life of LEU Fuel Element.

plate 23 is estimated to be 0.89 mil. For an element that reaches a similar burnup within the sequence of transition cycles the predicted oxide thickness is 0.95 mil. Note that these values are similar to predictions of the oxide thickness used in previous analyses. For example, as shown in Figure B.4, the oxide thickness on plates in an LEU element at an EOL burnup of ~180 MWd was predicted to be 0.93 mil in the analysis completed for the preliminary SAR with the MURR_2008 model [5].

The third-order polynomial fits in Figure B.4 of oxide thickness as a function of the number of in-use hours were used for modeling of accident analyses for cores in transition cycles and the reference equilibrium LEU core. The oxide thickness on the inner surface of plate 23 is used for modeling of all plates in the LEU fuel elements in these cores. This is a conservative modeling choice because the plate surface temperatures and resulting oxide thicknesses on all other plates are expected to be less than for the inner surface of LEU plate 23.

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