Safety Analysis of the Mo-99 Production Upgrade to the University of Missouri Research Reactor (MURR) with Highly Enriched and Low-Enriched Uranium Fuel

Nuclear Science and Engineering Division

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Safety Analysis of the Mo-99 Production Upgrade to the University of Missouri Research Reactor (MURR) with Highly Enriched and Low-Enriched Uranium Fuel

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Executive Summary

The University of Missouri (MU) has been working in conjunction with Argonne National Laboratory (ANL) in the National Nuclear Security Administration (NNSA) Material Management and Minimization (M3) Reactor Conversion Program to support conversion of the University of Missouri-Columbia Research Reactor (MURR®) from highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel. MURR is one of five U.S. High Performance Research Reactors (USHPRR) plus a critical facility that plans to convert to the use of LEU fuel.

In the framework of non-proliferation policies, the international community has worked on minimizing the use of and, when possible, eliminating weapons-usable nuclear material. In this geopolitical context, most research and test reactors, both domestic and international, have started a program of conversion to the use of LEU fuel. Conversion of civilian research reactors from HEU to LEU, and the return of the HEU to the country of origin, is an important component of the NNSA non-proliferation program. Worldwide, 71 reactors have been converted to the use of LEU fuel, and an additional 28 have been confirmed to be permanently shut down. These 99 reactors include conversion of 20 U.S. reactors among the 39 countries on six continents where conversions have occurred. With recent conversions in Ghana and Nigeria, this marks an important milestone in completing conversion of all reactors on the continent of Africa to LEU fuel. This marks the third continent to have completed all HEU to LEU reactor conversions, following Australia and South America.

A new type of LEU fuel based on an alloy of uranium-10wt% molybdenum (U-10Mo) is expected to allow the conversion of U.S. domestic high performance reactors where higher density fuels are required. The M3 conversion objectives for the USHPRR are to develop an LEU fuel element design that will ensure safe reactor operations, as well as maintain the existing experimental performance of the facility. To accomplish this, the very high-density U-10Mo monolithic fuel is currently being tested and qualified for use. An LEU fuel element specifically optimized to allow conversion of MURR with the U-10Mo monolithic fuel has been designed by staff members at ANL and MURR as a part of the USHPRR Reactor Conversion (RC) Pillar in the M3 Reactor Conversion Program.

The Preliminary Design for the LEU fuel element design specific to MURR has been completed based on required core performance analyses, fuel cycle performance analyses, steady-state thermal-hydraulic safety analyses, and accident safety analyses. After completing this documented technical basis, a Preliminary Safety Analysis Report (PSAR) was prepared and submitted to the U.S. Nuclear Regulatory Commission (NRC). The analysis performed as part of the PSAR demonstrated that the preliminary LEU fuel element design provides similar steady-state safety margins as HEU, adequate margin to the burnup-dependent fuel temperature safety limit for transient accidents, and meets or exceeds the HEU experimental performance with a power uprate to 12 MW for LEU relative to a licensed operating power level of 10 MW for HEU. Completion of this Preliminary Design stage allows for MURR-specific inputs into the irradiation qualification experiments and fabrication demonstrations. Due to the reactor-specific fuel element designs in the USHPRR, confirmation of, and updates to, the assumptions made in Preliminary Design is critical. Preliminary Design thus provides a basis for requirements in advance of fuel fabrication demonstrations and irradiation qualification testing that will lead to final fuel element specifications and SARs for conversion.

MURR is a highly reliable, public research facility that provides irradiation services and isotope production for a variety of uses. With active support from NNSA, MURR worked with commercial partners to develop a reactor-based technique to produce a reliable domestic supply of fission-based

molybdenum-99 (Mo-99). Although the Mo-99 production device would be located outside the core driver fuel region, analysis has shown that the proposed upgrade to the facility affects the three-dimensional power distribution in the core for both HEU and LEU fuel. This will affect the documented margins to safety, as well as could potentially impact the experimental performance of LEU relative to HEU following conversion. This assessment is necessary at this stage in the program, even though the plans for Mo-99 production are in a state of flux, since this device could impact the viability of the preliminary LEU fuel element design that is beginning fabrication demonstrations and irradiation qualification testing.

The purpose of this report is to document the safety margins for both HEU and LEU cores with the Mo-99 production upgrade proposed, designated as the 2017-RBM-99 device in this report. In addition to the 2017-RBM-99 device, the neutronics models also include descriptions of the experimental facilities and experiment loading that are currently typical for MURR and which are different from the experiments that were modeled in the analysis performed for the PSAR. Lastly, there have been changes to the LEU fuel element design with regard to the coolant channel thickness tolerances for the innermost and outermost coolant channels based on feedback from the fuel fabricator. This design change did not change the nominal dimensions of the coolant channels/fuel plates or fuel loading of the fuel plates in the element. Thus, for the purpose of the neutronics analysis for the 2017-RBM-99 upgrade, which is based on nominal dimensions and fuel loadings, the components of the LEU element are modeled the same as in the PSAR analysis. However, the design tolerance changes were incorporated in the calculations of the steady-state safety margins for both HEU and LEU. While the use of fewer significant figures has simplified fabrication, yet there were no negative impacts, including on safety margins, since the end channels were not limiting locations.

Detailed neutronics analyses for reference pseudo-equilibrium HEU and LEU cores with the same fuel management pattern as employed in the PSAR analysis found that the 2017-RBM-99 upgrade causes a shift in the radial power distribution in the core towards the driver fuel elements that are closest to the production device, as expected. Additionally, the core excess reactivity is higher with the upgrade, which results in critical control blade positions that are more deeply inserted than in the PSAR analysis. Consequently, the axial power peaking factor is higher when the upgrade is installed. Although the upgrade influences the radial and axial power peaking factors in the HEU and LEU cores, the maximum calculated discharge burnup of the HEU and LEU elements is only marginally affected relative to previous analysis without the upgrade and reported in References 1 and 2. Lastly, the neutronics analysis found that a power uprate to 12 MW with LEU would result in experimental performance in MURR that meets or exceeds HEU at 10 MW.

NUREG-1537 establishes the regulatory criteria for safety margins for non-power reactors under steady state and transient accident conditions. For steady-state operations, it must be shown that there is an adequate margin to the onset of flow instability (OFI) and a minimum critical heat flux ratio (CHFR) that is greater than 2.0 at the reactor Limiting System Safety Settings (LSSS), while also taking into account uncertainties. The margins to flow instability were calculated for 40 reference HEU cores with the 2017-RBM-99 upgrade, and also for 40 corresponding reference LEU cores. The analysis with the upgrade found a minimum flow instability margin of 1.78 MW relative to an LSSS power of 12.5 MW for HEU and 1.88 MW relative to an LSSS power of 15.0 MW for LEU. The calculated CHFR at steady state LSSS conditions is greater than 2.0 for all reference cores. For the PSAR analysis, the margin to flow instability for both HEU and LEU was predicted to be 2.1 MW. Thus, as a result of the 2017-RBM-99 upgrade, the minimum margins to flow instability for both HEU and LEU are decreased relative to the PSAR analysis, but the margin for LEU following conversion was found to be larger than HEU when the upgrade is included.

For transient accidents, NUREG-1537 requires that the fuel temperature remain below the fuel temperature safety limit. MURR (and other research reactors) has used the lower 95% confidence of the measured fuel blister threshold temperature as the basis for the temperature safety limit to meet the safety criterion of no fuel damage or fission product release as a result of transients. For all transient accidents, the initial steady-state conditions for the primary coolant flow rate, reactor coolant inlet temperature, and pressurizer pressure were set to the LSSS conditions, and the pool was set at the limiting condition for operations temperature. For reactivity insertion accidents (RIA), the operating power at the onset of the accident was assumed to be at the rod run-in set point of 115% of the full licensed power. For loss-of-coolant and loss-of-flow accidents (LOCA and LOFA, respectively), the accidents are assumed to begin from the full licensed power. These initial conditions are more stringent compared to the conservative limits of the normal operating band for MURR, but are in keeping with the requirements set forth in NUREG-1537 and are consistent with the recent HEU relicensing of MURR.

With the 2017-RBM-99 upgrade, the margin to the fuel temperature safety limit for the most extreme RIA in a reference HEU core is more than 250 °C, and occurs in a beginning-of-life (BOL) plate. For the same accident in a reference LEU core, the margin to the fuel temperature safety limit is 102 °C and occurs in an end-of-life (EOL) plate. The 2017-RBM-99 upgrade causes a small decrease in the calculated margins to the fuel temperature safety limits for HEU and LEU relative to the PSAR analyses. For HEU, the upgrade causes a decrease in the margin to the fuel temperature safety limit for the most limiting RIA of less than 3 °C. For LEU, the margin decreases less than 8 °C. All RIA transient results are considered to have very adequate margins to the burnup-dependent fuel temperature safety limit.

Four postulated LOCAs were analyzed for reference HEU and LEU cores. The effects of fuel burnup were included in the analysis when such modeling led to conservative results. The most limiting LOCA is initiated from a doubled-ended break in the core inlet pipe (cold-leg). For the reference HEU core with the 2017-RBM-99 upgrade, the minimum margin to the fuel temperature safety limit is 200 °C and occurs in a BOL plate. The effects of fuel burnup on the coolant channel thicknesses in the depleted elements in the reference core, which would cause a coolant flow redistribution and increase the cooling of the BOL element, was neglected in this case as it led to more conservative results. Relative to the PSAR analysis of this LOCA, the 2017-RBM-99 upgrade decreases the HEU margin to the fuel temperature safety limit by 9 °C. For the reference LEU core with the 2017-RBM-99 upgrade, LOCAs were evaluated both with and without coolant flow redistribution due to fuel element burnup in the core. The maximum predicted fuel temperature is 348 °C in a BOL plate and occurs in the double-ended cold-leg break LOCA without coolant flow redistribution. The margin to the fuel temperature safety limit in this case is 102 °C. This is a slightly smaller margin to the fuel temperature safety limit than predicted in the PSAR analysis, which was 115 °C and occurred in an EOL plate in a case with coolant flow redistribution. As in the RIAs, all transient LOCA results are considered to have very adequate margins to the burnup-dependent fuel temperature safety limit.

Three postulated LOFAs were analyzed for the reference HEU and LEU cores with the 2017-RBM-99 upgrade. For the HEU core, the analysis of the limiting LOFA found that the maximum fuel temperature is 148 °C and occurs in a BOL plate for the case without coolant flow redistribution due to burnup-related effects. The margin to the HEU fuel temperature safety limit in this case is 327 °C. This is the same maximum HEU fuel temperature and margin to the safety limit that were calculated for the limiting LOFA in the PSAR analysis. However, in the PSAR analysis, the maximum fuel temperature occurred in fuel plate 24 in a BOL element in core position 1, whereas in the analysis with the 2017-RBM-99 upgrade the maximum fuel temperature occurs in fuel plate 2 in a BOL element in core position 5. This element position is adjacent to the 2017-RBM-99 device when it is

installed. Thus, the upgrade causes a change in the location of the maximum HEU fuel temperature resulting from the limiting LOFA, which is consistent with the shift in the core power distribution that occurs due to the upgrade. For the LEU core, the minimum margin to the fuel temperature safety limit for the limiting LOFA with the 2017-RBM-99 upgrade is 232 °C and occurs in an EOL fuel plate. This is the same margin as was predicted for the limiting LOFA for the PSAR analysis. The location of the minimum margin moves from plate 23 of the fuel element in core position 8 in the PSAR analysis to plate 23 of the fuel element in core position 4 with the upgrade. With more margin than the other scenarios, all LOFA transient results are considered to have very adequate margins to the burnup-dependent fuel temperature safety limit.

Because there is no change to the HEU or LEU nominal fuel element loadings and an overall negligible effect on the fuel burnup as a result of the 2017-RBM-99 upgrade, the radiological consequences of the maximum hypothetical accident and fuel handling accident were not evaluated in this work. As found in the PSAR analysis, for any event where a release is considered in the current HEU SAR, the radiological consequences after conversion remain lower than the regulatory limits.

In summary, the Mo-99 production upgrade to MURR with the 2017-RBM-99 device has been found to cause a shift in the core power distribution for both HEU and LEU cores relative to the PSAR analysis. The predicted steady-state margins to flow instability with the upgrade are adequate for both HEU and LEU cores. The margins with the upgrade are lower than those predicted in the PSAR analysis for both HEU and LEU, but the margin for LEU is larger than that for HEU. For postulated transient accidents, the upgrade causes a slight decrease in the minimum margin to a fuel temperature safety limit based on the fuel blister threshold temperature. There is a 9 °C reduction in the safety margin for HEU for the most limiting accident evaluated under conditions specified in NUREG-1537. For a reference LEU core, the minimum safety margin for the most limiting transient accident with the 2017-RBM-99 upgrade is 102 °C, which is 7 °C less than the margin predicted in the PSAR analysis. Together with the preliminary blister measurements, results indicate that fuel conversion to a preliminary U-10Mo monolithic fuel element design in cores with the 2017-RBM 99 upgrade will result in no damage to fresh or depleted fuel plates for any credible accident postulated for MURR.

Finally, the preliminary fuel element design has been found to be robust even if a major modification to the core, such as the 2017-RBM-99 upgrade, were to occur. This provides confidence that current and future changes at MURR that are of a similar, or lesser scale, along with the required power uprate to 12 MW for LEU, will result in adequate safety margins and experimental performance in MURR that meets or exceeds the performance for HEU at 10 MW. Thus the Preliminary Design of the MURR LEU fuel element remains viable for fabrication demonstration and irradiation testing.

Table of Contents

Executive Summary	i
Table of Contents	v
List of Figures	vii
List of Tables	x
1 INTRODUCTION.....	1
1.1 Overview of Previous MURR Safety Analysis without Mo-99 Upgrade	3
1.2 Overview of 2017-RBM-99 Upgrade.....	4
1.3 Technical Objective of the Current Work	5
2 CORE NEUTRON PHYSICS ANALYSIS.....	7
2.1 Changes to Core Neutron Physics Models since the PSAR Analysis	7
2.2 Fuel Cycle Simulation for HEU and LEU Cores with the 2017-RBM-99 Upgrade	10
2.3 Reference Core States.....	20
2.4 Predicted Experimental Performance in LEU Cores Relative to HEU Cores	34
2.5 Shutdown Margin and Core Physics Parameters	35
3 STEADY-STATE THERMAL HYDRAULICS ANALYSIS	38
3.1 Introduction	38
3.2 Results	40
3.3 Summary and Conclusion.....	45
4 TRANSIENT ANALYSES.....	46
4.1 Initial Operating Conditions for Transient Analyses	46
4.1.1 Selection of Initial Steady-State Conditions for Transient Analyses.....	46
4.1.2 Selection of Core States for Transient Analyses	47
4.1.3 Modeling of Fuel Elements in Transient Analyses	47
4.2 Reactivity Insertion Accidents (RIA)	51
4.2.1 RIA to Be Considered.....	52
4.2.2 Results from HEU Core RIA Analyses	53
4.2.3 Results from LEU Core RIA Analyses	54
4.2.4 Summary of RIA Analyses	59
4.3 Loss-of-Coolant Accidents (LOCA).....	59
4.3.1 LOCA to be Considered	59
4.3.2 Results from HEU Core LOCA Analyses	61
4.3.3 Results from LEU Core LOCA Analyses	64
4.3.4 Summary of LOCA Analyses.....	69
4.4 Loss-of-Flow Accidents (LOFA).....	71
4.4.1 LOFA to Be Considered	72

4.4.2	Results from HEU Core LOFA Analyses	72
4.4.3	Results from LEU Core LOFA Analyses	73
4.4.4	Summary of LOFA Analyses.....	74
5	CONCLUSIONS	76
	ACKNOWLEDGEMENTS	79
	REFERENCES.....	80
	APPENDIX A – Effect of Change in End Channel Tolerances for Preliminary LEU Design on Thermal-Hydraulic Results	83
	REFERENCES – Appendix A	86
	APPENDIX B – Consideration of Lateral Heat Conduction Phenomena in MURR Fuel Plates Using CFD-COMSOL	87
B.1	Introduction	87
B.2	Previous StarCD Study of FSD Fuel Element.....	87
B.3	Model of Preliminary LEU Fuel Element Design	93
B.4	COMSOL Model of FSD	104
B.4.1	Description of Model	104
B.4.2	Comparisons of COMSOL and StarCD Simulations.....	106
B.5	Consideration of Parameters that Affect Lateral Heat Conduction	107
B.5.1	Effects of Cladding Configuration and Contact Gap	107
B.5.2	Effects of Thermo-Physical Properties	108
B.5.3	Effects of Fuel Burnup.....	109
B.6	Analysis of Preliminary LEU Fuel Element Design.....	117
B.7	Conclusion	120
	REFERENCES – Appendix B.....	121
	APPENDIX C – Effect of Beryllium Reflector Age on Power Distributions and Predicted Safety Margins	122
	APPENDIX D – Updated Accident Analyses for LOCA and LOFA Scenarios Presented for Preliminary Safety Analysis Report.....	125
D.1	Introduction	125
D.2	Loss-of-Coolant Accidents (LOCA).....	125
D.2.1	Results from HEU Core LOCA Analysis	125
D.2.2	Results from LEU Core LOCA Analyses	128
D.2.3	Anti-Siphon System Test.....	131
D.3	Loss-of-Flow Accidents (LOFA).....	132
D.3.1	Results from HEU Core LOFA Analysis	132
D.3.2	Results from LEU Core LOFA Analysis	133
	REFERENCES – Appendix D.....	135

List of Figures

Figure 1.1. MURR Core Layout as Modeled in Preliminary Safety Analysis Report [14].	5
Figure 2.1. MURR Core Models.	8
Figure 2.2. Radial Cross Section of Center Test Hole in MCNP5 MURR_2018 Model.	9
Figure 2.3. Axial Cross Sections of Center Test Hole in MCNP5 MURR_2018 Model.	12
Figure 2.4. Axial Heat Flux Profile in Plate 1 of Element Position 5 in All Fresh HEU Core with 2017-RBM-99 Upgrade.	16
Figure 2.5. Axial Heat Flux Profile in Plate 23 of Element Position 5 in All Fresh LEU Core with 2017-RBM-99 Upgrade.	17
Figure 2.6. Maximum Heat Flux in LEU Pseudo-Equilibrium Cores with the 2017-RBM-99 Upgrade.	21
Figure 2.7. Control Blade Perturbations Considered in Analysis of Minimum Safety Margin.	24
Figure 2.8. Hot-Stripe Heat Flux at Eq. Xenon Conditions in Reference Mixed-Burnup MURR LEU Core with 2017-RBM-99 Upgrade (Core 8C) and in PSAR Model (Core 8A).	30
Figure 2.9. Maximum Local Heat Flux by Plate at BOC in Reference Mixed-Burnup MURR LEU Core with 2017-RBM-99 Upgrade (Core 7C) and in PSAR Model (Core 7A).	31
Figure 4.1. U-10Mo Thermal Conductivity with Burnup at $T = 200\text{ }^{\circ}\text{C}$.	48
Figure 4.2. PARET/ANL Coolant Channel Model.	57
Figure 4.3. Schematic View of MURR Primary Loop.	60
Figure 4.4. Cold-Leg Double-Ended Break LOCA Transients for HEU Core 3C BOL Element (Element 5) without Flow Redistribution.	63
Figure 4.5. Cold-Leg Double-Ended Break LOCA Transients for LEU Core 7C BOL Element (Element 5) with Flow Redistribution.	67
Figure 4.6. Cold-Leg Double-Ended Break LOCA Transients for LEU Core 7C BOL Element (Element 5) without Flow Redistribution.	68
Figure 4.7. Transient Fuel Temperatures for Cold-Leg Double-Ended Break LOCA Transients for LEU Core 7C BOL Element (Element 5) without Flow Redistribution in Plate 3, Node 3.	69
Figure A.1. Schematic of MURR Fuel Element Showing Thicknesses of End Channels.	83
Figure B.1. Drawing of Adjacent MURR Fuel Elements and Modeling in StarCD.	88
Figure B.2. Power Density Profile in FSD Plate 3 in Element 1 in Core 7B.	89
Figure B.3. Axially-Averaged Local-to-Average Azimuthal Power Density Profile for Channel 4 of FSD Element in Position 1 in Core 7B.	90
Figure B.4. Azimuthally-Averaged Axial Power Density Profile for Channel 4 of FSD Element in Position 1 in Core 7B.	90
Figure B.5. Axially-Averaged Azimuthal Power Density Profile in Plate 3 of FSD Element in the Element 1 Position in Core 7B.	92

Figure B.6. Heat Flux from Plate 3 into Coolant Channel 4 in Element 1 in Core 7B at Location of Axial Peak for Heat Flux (0.236 m from bottom of plate).....	92
Figure B.7. Power Density Profile for Plate 3 of BOL Element (0 MWd), Core 7A.	94
Figure B.8. Azimuthally-Averaged Axial Power Density Profile in Plates 3 and 4 of Channel 4 of BOL Element (0 MWd) in Core 7A.	94
Figure B.9. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 3 and 4 of Channel 4 of BOL Element (0 MWd) in Core 7A.....	95
Figure B.10. Power Density Profile for Plate 22 of BOL Element (0 MWd), Core 7A.	95
Figure B.11. Azimuthally-Averaged Axial Power Density Profile in Plates 22 and 23 of Channel 23 of BOL Element (0 MWd) of Core 7A.	96
Figure B.12. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 22 and 23 of Channel 23 of BOL Element (0 MWd) in Core 7A.....	96
Figure B.13. Power Density Profile for Plate 3 of MOL Element (96 MWd), Core 7A.	97
Figure B.14. Azimuthally-Averaged axial Power Density Profile in Plates 3 and 4 of Channel 4 of MOL Element (96 MWd) in Core 7A.....	97
Figure B.15. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 3 and 4 of Channel 4 of MOL Element (96 MWd) in Core 7A.	98
Figure B.16. Power Density Profile for Plate 22 of MOL Element (96 MWd), Core 7A.	99
Figure B.17. Azimuthally-Averaged Axial Power Density Profile in Plates 22 and 23 of Channel 23 of MOL Element (96 MWd) in Core 7A.....	99
Figure B.18. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 3 and 4 of Channel 4 of MOL Element (96 MWd) in Core 7A.	100
Figure B.19. Power Density Profile for Plate 3 of EOL Element (170 MWd), Core 7A.....	101
Figure B.20. Azimuthally-Averaged Axial Power Density Profile in Plates 3 and 4 of Channel 4 of EOL Element (170 MWd) in Core 7A.	101
Figure B.21. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 3 and 4 of Channel 4 of EOL Element (170 MWd) in Core 7A.....	102
Figure B.22. Power density Profile for Plate 22 of EOL Element (170 MWd) in Core 7A.	102
Figure B.23. Azimuthally-Averaged Axial Power Density Profile in Plates 22 and 23 of Channel 23 of EOL Element (170 MWd) in Core 7A.	103
Figure B.24. Axially-Averaged Local-to-Average Azimuthal Power Density Distribution in Plates 22 and 23 of Channel 23 of EOL Element (170 MWd) in Core 7A.	103
Figure B.25. COMSOL Model Simulating Channel 4 of FSD Fuel Assembly.	104
Figure B.26. Coordinate Systems Used for Power Density Correlations.....	105
Figure B.27. Normal Total Surface Heat Flux – FSD Plate 3.	106
Figure B.28. Channel and Fuel Plate Geometry.	107
Figure B.29. Normal Total Surface Heat Flux – FSD Plate 3.	107
Figure B.30. Normal Total Surface Heat Flux for Preliminary LEU Fuel Element.	109
Figure B.31. Effect of Thermal Conductivity on Surface Heat Flux.	111
Figure B.32. Effect of Flowrate on Surface Heat Flux.....	112

Figure B.33. Effect of Coolant Inlet Temperature and Velocity on Surface Heat Flux.....	113
Figure B.34. Effect of Power Distribution on Surface Heat Flux.....	113
Figure B.35. Effect of Oxide Layer on Surface Heat Flux.....	114
Figure B.36. Effect of Oxide Layer on Heat Flux.....	115
Figure B.37. Total Effect of Burnup.....	116
Figure B.38. Axially-Averaged Hot-Stripe Heat Flux of All Fuel Plates in Core 7A for a 5 mm Hot-Stripe Width.....	117
Figure B.39. Azimuthal Power Density Distribution for Limiting Plates in Core 7A.....	118
Figure B.40. Surface Heat Flux for Limiting Plates in Core 7A.....	118
Figure B.41. Determination of 1D Representative Hot-Stripe Width using Data of Plate 4.....	120
Figure D.1. Cold-Leg Double-Ended Break LOCA Transients for HEU Core 3A BOL Element (Element 5) without Flow Redistribution.....	127
Figure D.2. Cold-Leg Double-Ended Break LOCA Transients for LEU Core 7A EOL Element (Element 8) with Flow Redistribution.....	130

List of Tables

Table 2.1. Preparation of HEU MURR_2018_Mo DIF3D Model	13
Table 2.2. Preparation of LEU MURR_2018_Mo DIF3D Model	15
Table 2.3. Driver Element Power Calculated by MCNP5 and DIF3D for HEU and LEU Cores.	16
Table 2.4. Summary of Results from Pseudo-Equilibrium Cycles in Fuel Cycle Simulation for HEU and LEU Cores with 2017-RBM-99.	19
Table 2.5. BOC Burnup Elements in HEU and LEU Reference Cores with the 2017-RBM-99 Upgrade.	22
Table 2.6. Maximum Hot-Stripe and Local Heat Flux for HEU Cores with the 2017-RBM-99 Upgrade.	26
Table 2.7. Comparison of Maximum Hot-Stripe and Local Heat Flux for HEU Cores with the 2017-RBM-99 Upgrade and from PSAR Analysis.	27
Table 2.8. Maximum Hot-Stripe and Local Heat Flux for LEU Cores with the 2017-RBM-99 Upgrade.	28
Table 2.9. Comparison of Maximum Hot-Stripe and Local Heat Flux for LEU Cores with the 2017-RBM-99 Upgrade and from PSAR Analysis.	29
Table 2.10. Burnup by Plate for Discharged LEU Element with the 2017-RBM-99 Upgrade.	33
Table 2.11. LEU Performance at 12 MW Relative to HEU Performance at 10 MW with the 2017-RBM-99 Upgrade.	35
Table 2.12. Kinetics Parameters and Reactivity Feedback Coefficients for MURR Core with the 2017-RBM-99 Upgrade.	36
Table 2.13. Control Bank Work ($\Delta k/k/\text{inch}$) in LEU Mixed-Burnup Cores at BOC.	37
Table 3.1. LSSS Conditions Assumed for Steady-State Thermal-Hydraulic Analysis.	40
Table 3.2. Flow Instability Results for HEU Core.	41
Table 3.3. Flow Instability Results for Preliminary LEU Core.	42
Table 3.4. Reactor Power for OFIR = 1.0 and CHFR = 2.0 for Preliminary LEU Core.	43
Table 3.5. Minimum CHFR for HEU Core at LSSS Power.	44
Table 3.6. Minimum CHFR for Preliminary LEU Core at the LSSS Power.	45
Table 4.1. Initial Steady-State Conditions for Transient Analyses.	47
Table 4.2. Oxide Thickness on MURR Fuel Plates.	50
Table 4.3. Results for Step Reactivity Insertion in Reference HEU Cores.	53
Table 4.4. Results for Step Reactivity Insertion in Reference LEU Cores.	54
Table 4.5. Results for 0.6% $\Delta k/k$ Step Reactivity Insertion in Reference LEU Core with the 2017-RBM-99 Upgrade.	56
Table 4.6. Results for Ramp Reactivity Insertion in Reference LEU Cores.	58
Table 4.7. HEU Core 3C LOCA Analyses Results without Flow Redistribution.	62
Table 4.8. LEU Core 7C LOCA Analyses Results with Flow Redistribution.	65
Table 4.9. LEU Core 7C LOCA Analyses Results without Flow Redistribution.	66

Table 4.10. HEU Core 3C LOCA Analyses Results without Flow Redistribution for MOL-1 (73 MWd) Pair.	71
Table 4.11. HEU Core 3C LOFA Analyses Results without Flow Redistribution.....	72
Table 4.12. LEU Core 7C LOFA Analyses Results with Flow Redistribution.	73
Table 4.13. LEU Core 7C LOFA Analyses Results without Flow Redistribution.	74
Table 4.14. HEU Core 3C LOFA Analyses Results without Flow Redistribution for MOL-1 (73 MWd) Pair.	75
Table A.1. HEU Flow Instability Results with New End Channel Tolerances.	85
Table A.2. LEU Flow Instability Results with New End Channel Tolerances.	85
Table B.1. Design Parameters for the FSD Fuel Element Design.	89
Table B.2. LEU Element Power in Core 7A (MW).....	93
Table B.3. Plate Power by Element in Core 7A (W).....	93
Table B.4. Design Parameters for the Preliminary LEU Fuel Element Design.	104
Table B.5. Fuel and Coolant Channel Properties for Preliminary LEU Fuel Element.....	104
Table B.6. Flow Condition and Material Thermal Conductivity of FSD.....	105
Table B.7. Comparison between COMSOL and StarCD.	106
Table B.8. Comparison between Different Configuration of Cladding and Contact Gap.....	108
Table B.9. Material Thermal Conductivity Assumed in Analysis (W/m-K).....	108
Table B.10. Comparison of Simulation Results for Different Sets of Thermal Conductivity Data.	109
Table B.11. Effect of Fuel Thermal Conductivity on Surface Heat Flux.....	111
Table B.12. Effect of Coolant Flowrate on Surface Heat Flux.....	112
Table B.13. Effect of Power Distribution on Surface Heat Flux.....	114
Table B.14. Effect of Oxide Layer on Surface Heat Flux.....	114
Table B.15. Models with Different Burnup Level.	116
Table B.16. Total Effect of Burnup.	116
Table B.17. Maximum Surface Heat Flux for Limiting Plates in Core 7A.	119
Table B.18. Determination of Representative Stripe Width for 1D Heat Transfer Analysis.....	119
Table C.1. Effect of Aged Beryllium Reflector on Flow Instability Power.	123
Table D.1. HEU Core 3A LOCA Analyses Results without Flow Redistribution.	126
Table D.2. LEU Core 7A LOCA Analyses Results with Flow Redistribution.....	128
Table D.3. LEU Core 7A LOCA Analyses Results without Flow Redistribution.....	129
Table D.4. Results of Anti-Siphon System Test.....	132
Table D.5. HEU Core 3A LOFA Analyses Results without Flow Redistribution.....	133
Table D.6. LEU Core 7A LOFA Analyses Results with Flow Redistribution.	134
Table D.7. LEU Core 7A LOFA Analyses Results without Flow Redistribution.	135

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1 INTRODUCTION

In the framework of non-proliferation policies, the international community has worked on minimizing the use of and, when possible, eliminating weapons-usable nuclear material. 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. A new type of LEU fuel based on an alloy of uranium-10wt% molybdenum (U-10Mo) is expected to allow the conversion of U.S. domestic high performance reactors where higher density fuels are required.

The National Nuclear Security Administration (NNSA) Material Management and Minimization (M3) program has developed an integrated approach to address the persistent threat posed by unintentional proliferation of nuclear materials. M3 reduces the risk of highly enriched uranium (HEU) and plutonium falling into the hands of non-state actors by minimizing the use of and, when possible, eliminating weapons-usable nuclear material around the world [3].

One of the M3 missions is reactor conversion. The M3 Conversion Office works with partners around the world to convert (or verify shutdown of) civilian facilities that use or produce weapon-usable nuclear materials. Since the U.S. Department of Energy Reduced Enrichment for Research and Test Reactor (RERTR) program began in 1978 [4], Argonne National Laboratory has been involved in reactor conversions and associated fuel development activities along with many domestic and international partners. Following the formation of NNSA, the domestic RERTR program has been integrated into the missions of M3.

Conversion of civilian research reactors from HEU to LEU, and the return of the HEU to the country of origin, is an important component of the NNSA non-proliferation program. Worldwide, 71 reactors have been converted to the use of LEU fuel, and an additional 28 have been confirmed to be permanently shut down. These 99 reactors include conversion of 20 U.S. reactors among the 39 countries on six continents where conversions have occurred. With recent conversions in Ghana and Nigeria, this marks an important milestone in completing conversion of all reactors on the continent of Africa to LEU fuel. This marks the third continent to have completed all HEU to LEU reactor conversions, following Australia and South America.

Within M3 Reactor Conversion, the goal of U.S. High Performance Research Reactor (USHPRR) conversion is to convert the six domestic high performance reactors (including one critical facility) that still use, and regularly refuel with, HEU fuel. Based on demonstrated favorable irradiation behavior, the USHPRR project has focused on the development of the LEU U-10Mo monolithic alloy fuel [5,6]. A Preliminary Design milestone has shown that the U-10Mo monolithic fuel system would allow the conversion of all the USHPRR [7].

The University of Missouri has been working in conjunction with the NNSA M3 Reactor Conversion Program at Argonne National Laboratory (ANL) to perform fuel element design and fuel cycle performance analyses [1], steady-state thermal-hydraulic safety analyses [8], and accident safety analyses [9] to support conversion of the University of Missouri Research Reactor (MURR) from HEU to LEU fuel. The conversion objectives are to develop an LEU fuel element design that will ensure safe reactor operations and acceptable shutdown and safety margins, as well as maintain the existing experimental performance of the facility.

The current HEU fuel element consists of 24 curved fuel plates of uranium aluminide dispersion fuel. An LEU fuel element for conversion of MURR has been designed by staff members at ANL and MURR

as a part of the USHPRR Reactor Conversion (RC) Pillar, led by ANL, in the M3 Reactor Conversion Program. The preliminary LEU fuel element has 23 curved fuel plates of a monolithic U-10Mo alloy fuel that is currently being qualified for use in USHPRR by the Fuel Qualification (FQ) Pillar, led by Idaho National Laboratory. The Fuel Fabrication (FF) Pillar, led by Pacific Northwest National Laboratory, is working to develop and install manufacturing capacity with the U-10Mo monolithic fuel. The FF Pillar is fabricating fuel for ongoing and future irradiation qualifications. The FF Pillar is also beginning work to perform fuel plate and fuel element demonstrations. This is required to validate or update preliminary fabrication assumptions, including critical safety/operational tolerances for the fuel element specifications under development by the RC Pillar. Final irradiation and fabrication data are needed to perform analysis for the Final SAR for conversion to be submitted to the regulator, as well as prior to finalizing fuel specifications needed to commence production of initial core loads of LEU fuel elements that will be needed prior to conversion.

The preliminary design analysis has found that, following conversion to LEU using this element design and a proposed power uprate to 12 MW, MURR will be able to continue to operate safely and reliably, with a equivalent experimental performance relative to HEU operations.

Part of MURR's mission is to provide irradiation services and radioisotope production for a variety of uses, including radiopharmaceuticals. Recently, MURR, in a partnership with General Atomics (GA) and Nordion [10,11], evaluated an upgrade to MURR for commercial-scale production of molybdenum-99 (Mo-99) for medical diagnostic imaging. The design of an LEU-fueled reactor-based production of Mo-99, designated "2017-RBM-99" in this report, was to be inserted in a reflector position outside the active core. Also, HEU core neutron physics models developed as part HEU to LEU conversion analysis reported in Reference 1, were updated by MURR and others outside of the scope of this report to represent changes to the experimental facilities subsequent to the preliminary design analysis. Based on that work, MURR submitted Part 1 of a License Amendment Request (LAR) to the U.S. Nuclear Regulatory Commission (NRC) to gain regulatory approval to operate with the 2017-RBM-99 [12].

No prior work has evaluated the impact of the 2017-RBM-99 upgrade on HEU element burnup or safety margins. Likewise, no analysis of the impact of the 2017-RBM-99 upgrade on LEU element burnup, power distributions, or safety margins has been performed. This report presents the results of detailed analyses of the impact of the 2017-RBM-99 upgrade on fuel burnup, power distributions, and predicted safety margins in steady-state operations and accident scenarios for both HEU- and LEU-fueled cores.

Subsequent to the work in this report being performed, MURR's partners withdrew from the project for commercial, but not technical, reasons [13]. MURR has since withdrawn its LAR to operate with the 2017-RBM-99. However, since there are no technical obstacles to commercial-scale reactor-based Mo-99 production in MURR and future partnerships may be developed that would revive interest in the 2017-RBM-99 or a similar reactor-based Mo-99 production capability at MURR, the detailed safety analyses for the proposed 2017-RBM-99 upgrade in HEU and LEU cores have been completed. Despite any specific commercial events, the work in this report remains critically important to understand that the preliminary MURR LEU fuel element design remains viable. This viability is what allows USHPRR fabrication demonstrations and irradiation qualification to continue with the preliminary fuel element design. Since uranium-fueled Mo-99 targets have not been present in the previous MURR conversion analysis, this work introduces these targets, and envelopes changes expected, at up to this scale, for any other projected introduction of Mo-99 targets at MURR.

1.1 Overview of Previous MURR Safety Analysis without Mo-99 Upgrade

This section provides an overview of previous neutronics and safety analyses for MURR HEU and LEU cores without an upgrade to the facility for Mo-99 production, which was the modeling basis for all preliminary design and analysis. This overview is provided since the methodologies and models used in the preliminary design neutronics analysis served as the starting point for the HEU core power distribution analysis reported in the proprietary reports. Furthermore, the impact on safety margins from the 2017-RBM-99 upgrade in the current work will proceed from the previous safety analyses [9,14], with appropriate modifications of the models that reflect a design change to the LEU element unrelated to the 2017-RBM-99 and software updates that meet current quality assurance (QA) standards that are required under the M3 Reactor Conversion Program.

Following the completion of preliminary design, a Preliminary Safety Analysis Report (PSAR) was prepared following the guidelines of NUREG-1537 [15,16] for conversion of non-power reactors. The work reported in References 1, 8, and 9 served as the technical basis for preparing the PSAR [14] which has been submitted to the NRC for review. The PSAR analysis for the HEU cores was based on current reactor operating parameters, typical fuel management, and typical experimental loadings that existed in MURR in 2008. The analysis for LEU cores was based on the same operating conditions, fuel management, and experiment loadings, with the exception of a proposed reactor power uprate to 12 MW and changes to the Limiting System Safety Settings (LSSS) following conversion. The power uprate for LEU was proposed to maintain experimental performance.

The typical fuel management used at MURR was modeled in a fuel cycle simulation. The fuel element shuffling and depletion were performed with the REBUS-PC software [17] using the DIF3D neutronics solver and broad-group cross sections generated by WIMS/ANL [18,41] to obtain HEU and LEU element burnups that are typical for MURR's weekly operations. Critical control blade positions and detailed three-dimensional power distributions were calculated for reference HEU and LEU cores using MCNP5 [19].

For the steady-state thermal-hydraulic analysis, a set of 24 reference HEU cores and 24 reference LEU cores were identified for predicting the minimum steady-state safety margins. These cores contained elements that were either all fresh, which is not a typical core loading in MURR but would be the case immediately following conversion, or with a mixture of burnups, which is typical for MURR operations. The reference cores also included variations on the core xenon poisoning, control blade positioning, and experimental loading.

For steady-state HEU operations at the extreme of LSSS conditions, the minimum margin to coolant flow instability among all reference cores was predicted to be 2.11 MW. The analysis was completed using the PLTEMP/ANL software [20]. For the LEU reference cores operating at LSSS conditions, a similar minimum margin of 2.10 MW to coolant flow instability was predicted. A detailed description of the steady-state thermal hydraulic analysis performed in the prior work without the 2017-RBM-99 upgrade can be found in References 8 and 14.

Analyses of accidents following the guidelines of NUREG-1537 were also completed to assess margins to the fuel temperature safety limits for HEU and LEU fuel. For reactivity insertion accidents (RIA), the initial conditions at the start of the transient assumed an operating power at the rod run-in set point of 115% of full licensed power, and reactor core coolant inlet temperature and flow rate at the extreme of LSSS settings. Reference HEU and LEU cores that included fuel elements with a mixture of burnups were considered. Burnup dependent effects, such as degradation of the fuel thermal

conductivity, coolant channel gap closure due to burnup-related effects, and the formation of an oxide layer on the surface of the fuel plates, were included in the modeling. In the PSAR analysis, the set of RIAs, including both step and ramp insertion of reactivity, was evaluated with the PARET/ANL software [21]. A margin of more than 250 °C to the HEU fuel temperature safety limit based on measurements of aluminide fuel [22] was predicted for the most limiting accident. The highest predicted HEU fuel temperature was in plate 1 (adjacent to the center test hole) in a beginning-of-life (BOL) element. For the LEU core, the predicted margin to a burnup-dependent fuel temperature safety limit for all RIAs was greater than 109 °C. The LEU fuel temperature safety limit was based on the lower 95% confidence interval for preliminary blister anneal tests performed for U-10Mo monolithic fuel plates, as described in Chapter 13 of the PSAR [14]. The smallest margin to the LEU fuel temperature safety limit was predicted for a 0.6% $\Delta k/k$ step insertion and occurred in fuel plate 23 of an end-of-life (EOL) element.

For loss-of-coolant and loss-of-flow plant transient accidents (LOCA and LOFA, respectively), the initial conditions at the time of the accident were assumed to be full-licensed power and coolant inlet temperature and flow rate at the LSSS settings for MURR. The same reference cores considered for the RIAs were used for all LOCA and LOFA analyses. The set of LOCAs and LOFAs evaluated for the PSAR were modeled with RELAP5/MOD3.3 [24]. The predicted margin to the HEU fuel temperature safety limit for the most limiting HEU accident, a cold-leg double-ended break LOCA, was greater than 200 °C. The peak HEU fuel temperature was predicted for fuel plate 2 in a BOL element. For the LEU core, the smallest predicted margin to the monolithic fuel temperature safety limit was 115 °C. This was predicted for the cold-leg double-ended break LOCA modeled with coolant flow redistribution among the elements in the core due to burnup effects and occurred in fuel plate 22 of an EOL element.

In summary, the PSAR analysis predicted that adequate safety margins exist for HEU operations at 10 MW and LEU operations with a power uprate to 12 MW for both steady-state and postulated accident scenarios. Also, the predicted experimental performance in MURR for LEU at 12 MW was slightly improved relative to HEU. The impact of the 2017-RBM-99 upgrade on reactor operations or predicted safety margins was not evaluated in the PSAR and is the subject of this report.

1.2 Overview of 2017-RBM-99 Upgrade

Based on the proposed licensing, the 2017-RBM-99 upgrade related to the LAR in HEU cores is an experiment and would not impact the total licensed power of MURR. This report covers the safety and performance of the core and other experiments in HEU and LEU cores with the 2017-RBM-99 upgrade. Consequently, the work of others on experimental performance or safety margins of the 2017-RBM-99 experimental device were not considered in this report.

Each 2017-RBM-99 is designed to replace one of the 30° graphite reflector wedges that surround the core. It has been proposed to replace the two fast-flux irradiation facilities which are noted in the depiction of the MURR reactor layout illustrated in Figure 1.1. These positions are directly adjacent to core position 5 that is typically loaded with fresh, or nearly fresh, fuel. Because the 2017-RBM-99 contains fissile material, it is expected that the power distribution among the fuel plates in the reactor core will shift towards the outermost plates of the element in position 5 and its neighbors. This could improve, or degrade, the safety margin for HEU, as some power is pulled away from the currently limiting HEU innermost plates and towards outermost plates near the 2017-RBM-99.

Figure 1.1. MURR Core Layout as Modeled in Preliminary Safety Analysis Report [14].

The impact of the 2017-RBM-99 on the MURR HEU core results in critical control blade positions that are 2 to 3 inches more deeply inserted relative to the critical blade positions for the same core in the reactor with the experimental configuration shown in Figure 1.1. Note that the MURR control blades and regulating blade are inserted from the top of the core. The more deeply inserted control blade position affects both the axial and radial power profiles in the core, increasing the maximum local heat flux towards the bottom of the core and towards fuel plate 1 adjacent to the center test hole. Additionally, the radial power profile across the core is affected by neutron leakage from the 2017-RBM-99 to the core, increasing the power peaking in core positions closest to the device and decreasing power peaking in core positions furthest from the 2017-RBM-99. As a result, a maximum increase in power peaking of approximately 10% was observed relative to the same core without the device installed. Thus, prior to undertaking the current work there was evidence that a significant change such as this required evaluation whether or not local power increases impacted limiting locations and safety margins.

1.3 Technical Objective of the Current Work

Overall, the local power distribution for any reactor impacts whether there is sufficient safety margin to avoid a boiling crisis in the coolant channels between fuel plates and/or fuel damage. It has been found through previous analysis [8,14] that the inner plates/channels of a BOL HEU element in MURR, i.e., those closest to the center test hole and furthest from the reflector, are limiting with respect to flow instability and safety margin. For the preliminary LEU fuel element design for conversion, the outer plates/channels of an EOL element, i.e., those closest to the reflector, have been found to be limiting. Although the limiting LEU element at EOL is not as immediately adjacent to the proposed location of the 2017-RBM-99, it is another near neighbor, and the movement of additional power towards the outer plates, where the LEU element is already limited, is expected to reduce the safety margin.

In this work, HEU and LEU cores with the 2017-RBM-99 upgrade will be analyzed using the same methods and analogous assumptions as were used to complete the PSAR analysis for HEU and LEU cores with the 2008 configuration and without the 2017-RBM-99. Models for the current analysis will begin with models developed during the PSAR analysis and will include subsequent changes to the

experimental configuration of MURR. These changes include changes to the center test hole configuration and nominal sample loading, as well as subsequent changes to the experimental facilities in the outer graphite reflector region and the proposed change to install the 2017-RBM-99 in the graphite reflector.

A fuel cycle simulation based on typical fuel management for MURR weekly operations was performed to obtain appropriate HEU and LEU plate-level compositions that include the effect of the 2017-RBM-99 upgrade on the core power distribution. This is described in Section 2. Detailed, three-dimensional critical core power distributions were calculated for a set of reference cores. Experimental performance measures were also calculated to compare LEU relative to HEU and to confirm if the proposed LEU power uprate to 12 MW is still appropriate with the 2017-RBM-99. Lastly, steady-state (Section 3) and accident (Section 4) thermal-hydraulic analyses were performed to assess the safety margins for the reference cores with the 2017-RBM-99 upgrade. An LEU fuel element design change unrelated to the 2017-RBM-99 upgrade was necessary when preparing the LEU fuel element specification and drawings for initial demonstration element trials [25]. Since this may affect the safety margins, this design change was included in all analyses reported in Section 3. The effect of the design change on steady-state thermal-hydraulic analysis relative to the PSAR analysis is described in Appendix A.

The safety margins obtained from this analysis for LEU cores with the 2017-RBM-99 upgrade will be compared against analogous HEU safety margins with the 2017-RBM-99 and LEU safety margins obtained from prior analysis without the 2017-RBM-99 (i.e., the PSAR analysis).

It is possible that the safety margin analysis will conclude that the LEU fuel element design itself may be impacted by the 2017-RBM-99 upgrade, and that some reduction of the fuel core thickness in, for example, the outermost plates would also be required. In the case of a change in the element design, a significant portion of the safety analyses would need to be repeated in order to understand the impact to safety margin, including peak temperatures during transient accidents, and whether the ATR/BR-2 irradiation testing parameters also impact the design of FQ experiments already in design at INL (MP-2, FSP-1, and DDE MURR). For this reason, this activity will clarify associated conversion impacts, which are primarily anticipated to be to the RC and FQ Pillars at this point.

Lastly, some software used in the analysis performed as part of this work have been qualified following software evaluation procedures in keeping with NQA-1 standards. In some instances, the use of a new software version produces results for cores without the 2017-RBM-99 upgrade that are different than those that were reported in the PSAR analysis [14] using earlier versions of the software. Where differences relative to the PSAR analysis are obtained with newer versions of the software, the differences are described in this report. However, no changes to the software were substantial enough to lead to different conclusions regarding margin to safety.

2 CORE NEUTRON PHYSICS ANALYSIS

This section describes the core neutron physics analysis that was completed for MURR reference HEU and LEU cores with the 2017-RBM-99 upgrade. The neutron physics analysis includes modeling of the reactor core and experimental facilities that affect the fuel element compositions with burnup, critical control blade positions, three-dimensional power distributions, experimental performance, and reactivity feedback coefficients.

The modeling and results of the neutron physics analysis for HEU and LEU cores with the experimental loading that was typical in 2008 were reported in detail in References 1 and 14. As a result of the previous analysis, an LEU fuel element design was developed that met the conversion objectives of adequate safety margins and comparable or improved experimental performance relative to HEU. This design was labeled as the “CD35” element design in reference reports. The same LEU element design is assumed for the core neutron physics analysis with the 2017-RBM-99 upgrade.

This element design will be tested by the FQ Pillar through a full demonstration element irradiation test in a test reactor under prototypic conditions for MURR known as the Design Demonstration Element (DDE) experiment. The DDE-MURR experiment will be supplemental to monolithic U-10Mo fuel qualification experiments with mini- and full-plate tests under bounding conditions. The MURR LEU fuel element design specification for the initial element trial is documented in Reference 25. This specification reflects a design change to the manufactured coolant channel tolerances for the innermost and outermost coolant channels based on feedback from the fuel fabricator (BWXT). However, this design change did not change the nominal dimensions of the coolant channels/fuel plates or fuel loading of the fuel plates in the element. Thus, for the purpose of the neutronics analysis for the 2017-RBM-99 upgrade, which is based on nominal dimensions and fuel loadings, the components of the LEU element are modeled the same as in the PSAR analysis.

Lastly, design parameters for the DDE-MURR experiment under prototypic conditions for MURR based on the preliminary LEU element design and the 2008 experimental loadings were reported in Reference 2. These design parameters will be used by the FQ Pillar for planning the irradiation campaign. If changes to the design parameters are required due to any element design change or facility upgrade, the design parameters report will be revised.

2.1 Changes to Core Neutron Physics Models since the PSAR Analysis

The neutronics model for the PSAR analysis was based on typical experimental facilities/loadings in MURR in 2008. This configuration is depicted in Figure 2.1 and is labeled as “MURR_2008.” An MCNP5 model was developed to calculate detailed power distribution analysis of both all fresh and typical mixed burnup cores. The MCNP5 model had been benchmarked for a core loaded with all fresh HEU fuel against startup tests. [1] A REBUS-DIF3D model for the MURR_2008 configuration was developed for calculating burned fuel compositions from a fuel cycle simulation that replicated typical fuel management over an extended number of weekly cycles. Reference 1 describes how the DIF3D model was benchmarked against the MCNP5 model for fresh HEU and LEU fuel by matching the reactivity worth¹ of the experiments. Furthermore, fuel element compositions derived from the REBUS-DIF3D analysis at select burnups were used in MCNP5 calculations to benchmark the calculated critical control blade heights against core critical blade measurements from weekly startups with good agreement. [26]

¹ Throughout this report, reactivity worth is calculated as $\Delta k/k$, or $(k_2 - k_1)/(k_1 \times k_2)$.

Figure 2.1. MURR Core Models.

Since 2008, there have been four changes to typical experimental facilities and experiment loading at MURR that are included in the neutronics model, and are illustrated in the “MURR_2018” depiction of the reactor in Figure 2.1. Note that the MURR_2018 model depicted in Figure 2.1 does not show the 2017-RBM-99 upgrade, but rather depicts the experimental facilities and experiment loading in the current (2018) typical configuration for MURR operations.

1. Center Test Hole. A radial cross section of the center test hole in the MCNP5 MURR_2018 model is provided in Figure 2.2. A six-tube canister holder assembly is installed in the center test hole. This canister is similar in design to the three-tube canister that was modeled in the MURR_2008 model, with the exception that the new canister design has three small diameter tubes for moveable experiments and there is increased capacity for samples in the vertical direction in the three larger tubes. Additionally, the typical collection of samples loaded in the holder apparatus in 2018 has a greater total reactivity worth than in 2008. This is allowed because of the installation of the Flux-trap Irradiations Reactivity Safety Trip (FIRST) device that has been approved by an operating license amendment [27].
2. A single 60° wedge (“combined GHIJ wedge” identified in Figure 2.1) in the graphite reflector adjacent to fuel element position 1 for silicon irradiation in the MURR_2018 model replaces the 30° “G10” (“yellow-green”) and the 30° “G11” (“blue-yellow”) wedge positions in the MURR_2008 model.
3. Two additional irradiation positions are included in the “G6” graphite wedge (see identifier in Figure 2.1) in the MURR_2018 model relative to MURR_2008.
4. The “G3” and “G4” graphite wedges (identified in Figure 2.1) have the same irradiation positions between the two models, but the samples loaded in these positions are different between the MURR_2018 and MURR_2008 models.

Figure 2.2. Radial Cross Section of Center Test Hole in MCNP5 MURR_2018 Model.

There have also been two changes to the neutronics model since the PSAR analysis with the MURR_2008 model unrelated to changes in experimental facilities or loading.

1. Sample materials in the center test hole and graphite reflector are modeled with ENDF/B-VII cross sections in the MURR_2018 model. Samples were modeled with ENDF/B-VI cross sections in the MURR_2008 model.
2. In the MURR_2018 model, the graphite in the combined GHIJ wedge is modeled at 40 °C (103 °F), while all other graphite in the reflector is modeled at 149 °C (300 °F). All graphite was modeled at a temperature of 171 °C (340 °F) in the MURR_2008 model. The graphite temperature change has been found through MCNP5 analysis to have a negligible effect (< 5 pcm, or $0.005\% \Delta k/k$) on the core reactivity calculated by the two models.

The graphite material in the graphite reflector is modeled with a density of 1.723 g/cm^3 in both the MURR_2008 and MURR_2018 models. This modeling choice is based on an analysis of the graphite wedges manufactured for MURR and performed by SGL Carbon in 2002. The impurities for the graphite samples were measured to have a boron equivalence of 0.304 ppm boron. The MCNP models used for the LAR analysis of the 2017-RBM-99 upgrade (described briefly in Section 1.2, above) used a graphite density of 1.6 g/cm^3 in all wedges except the combined GHIJ wedge, which used the SGL measured value of 1.723 g/cm^3 . The graphite density assumed in the LAR analysis was empirically-adjusted to match the neutron spectrum and reaction rates in the graphite reflector. This modeling choice employed was made without regard to the impact on core excess reactivity, but it was found that the graphite density change affects the MCNP5-calculated core excess reactivity by $< 0.08\% \Delta k/k$ (80 pcm). This difference is small compared to the MCNP5 model bias of $0.63\% \Delta k/k$, which was determined in Reference 1 by comparing MCNP5 model results against critical core measurements of fresh HEU core startup physics tests. Consequently, a graphite density of 1.723 g/cm^3 and impurity

level of 0.304 ppm boron were selected for all graphite in the models used in this work since they are based on experimentally measured data. Furthermore, this empirical adjustment was not used in any of the work for LEU conversion Preliminary Design or for the PSAR.

The final neutronics modeling change relative to the PSAR analysis considered in this work was the 2017-RBM-99 upgrade, including the fueled cartridges for reactor-based Mo-99 production. These were installed in place of the fast-flux irradiation facilities in the “L and “N” wedge positions depicted in Figure 2.1 in a “MURR_2018_Mo” model that is not shown due to the proprietary nature of the cartridge design, but which was used for all neutron physics analysis conducted as part of this work.

2.2 Fuel Cycle Simulation for HEU and LEU Cores with the 2017-RBM-99 Upgrade

In order to obtain appropriate HEU and LEU depleted fuel element compositions, the typical fuel management used at MURR was modeled in a fuel cycle simulation for HEU and LEU cores with the 2017-RBM-99 upgrade. The fuel element shuffling and depletion were performed with REBUS-PC v.1.5 [17]. The same computational method was employed for the PSAR analysis. The simulation modeled more than 80 weeks of operations to arrive at a “pseudo-equilibrium” fuel cycle. Due to the large number of flux calculations needed for the analysis, the DIF3D neutronics solver was selected because of its computational efficiency. The neutron flux was calculated in 10 energy groups, which was then used for the fuel depletion by REBUS on a plate level with 12 axial regions.

The MURR core and experimental facilities were modeled in DIF3D with the three-dimensional θ -R-Z geometry option since it is well-suited to represent the curved shape of the fuel plates, control blades, and beryllium and graphite reflectors. However, the DIF3D solver cannot model the exact geometric details of the experiments in MURR. Instead, a homogeneous mixture of boron was added to the various experimental locations in the DIF3D model in order to match the reactivity worth calculated by MCNP5 for all experiments in the graphite reflector and center test hole in all fresh cores of HEU or LEU fuel. The homogenized boron was not depleted in the REBUS-DIF3D analysis. Likewise for the 2017-RBM-99, exact geometric modeling of the device or its fueled cartridges is not possible with the DIF3D geometry options, but the details of modeling the cartridge geometry are not necessary in order to capture the effect of the 2017-RBM-99 on the core reactivity or its influence on the core power distribution in DIF3D for the purpose of the fuel cycle simulation. Later comparison (described below) of the DIF3D model results against the results of the MCNP5 model which represents the geometric details of the experiments in the center test hole and the 2017-RBM-99 was performed to verify the DIF3D model.

A broad-group cross section library for the HEU and LEU fuel and core structural materials was previously generated by WIMS/ANL v.5.10 [18,41] as part of the preliminary design analysis [1]. This same library of cross section data was utilized in the current DIF3D analysis since there was no change to the nominal HEU or LEU fuel element dimensions or fuel plate loadings and the 2017-RBM-99 will have little effect on the core neutron spectrum. Broad-group cross sections for the structural components of the 2017-RBM-99 and its fueled cartridges were generated using WIMS/ANL and added to the previously generated cross section library.

To develop an appropriate DIF3D model for the fuel cycle simulation, a series of cases were run to match the reactivity worth of all experiments in the reactor. The MCNP5 MURR_2018_Mo model described above was altered to remove all experiments and experiment positions in the graphite reflector. This “clean” configuration had an empty six-tube canister holder apparatus in the center

test hole, solid graphite in AA6061 cans in all positions surrounding the beryllium reflector, no “hanging wedge” (which is adjacent to the combined GHIJ wedge), and all fresh HEU or LEU fuel. Beam tubes and their penetrations through the graphite reflector wedges were modeled. The 2017-RBM-99 and all typical experiments were successively added to the model and MCNP5 calculations were performed to calculate the individual and cumulative experiment worth. The four BORAL® control blades in the MCNP5 calculations were positioned at 23 inches withdrawn, which is the typical blade position at equilibrium xenon conditions (control blades at about 90% of their full travel from all-in to all-out, so the control blade tip is near the top of the core). The stainless steel regulating blade was positioned at 10 inches withdrawn for these calculations.

The experimental facilities and typical experiment loadings were included in the model in the following order to restore the MURR_2018_Mo model to its complete state:

1. Insert two 2017-RBM-99 devices and fueled cartridges in place of the fast-flux irradiation facilities in the L and N wedges of the graphite reflector.
2. Replace solid graphite cans in the combined GHIJ 60° wedge with silicon irradiation experiments and restore the hanging wedge to the model.
3. Insert the irradiation positions and experiments in the G3 graphite wedge.
4. Insert the irradiation positions and experiments in the G4 graphite wedge.
5. Insert the irradiation positions and experiments in the G6 graphite wedge.
6. Insert typical sample loadings in the center test hole holder apparatus in the axial range from -7.5 cm to +12.3 cm relative to the core midplane. This axial range covers roughly one-third of the active fuel height in the core. This range was chosen by inspection of the detailed experiment loading in the MCNP5 model, which is shown in Figure 2.3. The blue-shaded portions of the tubes and small barrels in the center test hole contain all or mostly water. This modeling choice divides the DIF3D model of the center test hole into three axial segments that appropriately represent the axial details of the typical experiment loading.
7. Insert remaining samples in the MCNP5 model in the axial ranges below -7.5 cm and above +12.3 cm in the model.

All MCNP5 calculations were completed with v1.6 of the software [19].

A similar set of calculations were completed with the DIF3D model, again starting with a core loaded with all fresh HEU or LEU fuel, the control blades and regulating blade at the same position as in the MCNP5 calculations, an empty holder configuration in the center test hole, and no experiments in either the center test hole or the graphite reflector. The details of the 2017-RBM-99 were approximated in the DIF3D model, as discussed briefly above. The reactivity worth of any experiments in the graphite reflector wedges was modeled by a homogenized addition of boron in the graphite in that wedge. The amount of boron in each wedge was empirically derived by adjustment that matched the reactivity worth of the experiment calculated by MCNP5. A criterion of ± 10 pcm for matching the reactivity worth of each experiment was selected. For the experiments in the three axial segments of the center test hole, solid aluminum spacers were placed inside the holder apparatus and boron was added to the aluminum to match the reactivity worth of the experiments to within ± 10 pcm.

Figure 2.3. Axial Cross Sections of Center Test Hole in MCNP5 MURR_2018 Model.

Table 2.1 provides a summary of the series of MCNP and DIF3D cases that were completed to develop the HEU MURR_2018_Mo DIF3D model. It can be seen that the reactivity worth of the two 2017-RBM-99 devices in the HEU core is predicted by MCNP5 to be -42 pcm relative to solid graphite cans with no experiments. For the DIF3D model which approximates the geometry of the fueled cartridges and surrounding structure, the reactivity worth is reasonably well predicted to be -140 pcm. The excess reactivity calculated with the DIF3D model for an all fresh HEU core with the 2017-RBM-99 upgrade and typical experiments in the graphite reflector and center test hole agrees with the detailed MCNP5 model to within 221 pcm. The current experimental configuration and experiment loading plus the 2017-RBM-99 upgrade results in a fresh HEU core excess reactivity that is 1177 pcm greater than with the typical experimental configuration and experiment loading in 2008.

Table 2.1. Preparation of HEU MURR_2018_Mo DIF3D Model.

Case	Description	MCNP				DIF3D			Diff. DIF3D vs. MCNP (pcm)
		k-eff	σ (pcm)	Experiment Reactivity Worth (pcm)		k-eff	Experiment Reactivity Worth (pcm)		
				Ind.	Total		Ind.	Total	
0	Base case. All fresh HEU fuel. Blades at 23" withdrawn; reg blade at 10" withdrawn. Empty holder in center test hole. Graphite-filled aluminum boxes with no experiment positions (other than beam tubes) in all wedge positions. Remove hanging wedge.	1.10095	8			1.10362			220
2	MCNP5. Insert 2017-RBM-99 in L and N positions in graphite reflector.	1.10044	8	-42	-42				
2a	DIF3D. 2017-RBM-99 installed in L- and N-wedges.					1.10191	-140	-140	121
4	MCNP5. Change combined GHIJ and hanging wedges to 2018 conditions.	1.09320	8	-602	-644				
4e	DIF3D. 315 ppm B added to graphite of combined GHIJ wedge.					1.09585	-502	-642	221
5	MCNP5. Change 3-wedge to 2018 conditions.	1.09242	8	-65	-709				
5b	DIF3D. 60 ppm B added to graphite of 3-wedge.					1.09497	-74	-716	213
6	MCNP5. Change 4-wedge to 2018 conditions.	1.09145	8	-81	-791				
6b	DIF3D. 80 ppm B added to graphite of 4-wedge.					1.09402	-79	-795	216
7	MCNP5. Change 6-wedge to 2018 conditions.	1.09052	8	-78	-869				
7a	DIF3D. 70 ppm B added to graphite of 6-wedge.					1.09301	-85	-879	209
8-3	MCNP5. Change center test hole between axial planes - 7.5 to +12.3 cm from empty-holder to 2018 conditions.	1.09526	8	+397	-472				
8-3a	DIF3D. Aluminum spacers with 90 ppm B in center test hole sample positions between axial mesh 25-33 (roughly equiv. to MCNP axial positions -7.5 to +12.3 cm)					1.09784	+402	-477	214
8-4	MCNP5. Change entire holder to 2018 conditions.	1.09760	8	+195	-277				
8-4c	DIF3D. 130 ppm B smeared in center test hole sample positions between axial mesh 10-25. 130 ppm B in center test hole sample positions between axial mesh 33-38.					1.10026	+201	-276	221

Table 2.2 provides a summary of the series of MCNP and DIF3D cases that were completed to develop the LEU MURR_2018_Mo DIF3D model. The reactivity worth the two 2017-RBM-99 devices in the LEU core is predicted by MCNP5 to be -49 pcm relative to solid graphite cans with no experiments. For the DIF3D model which approximates the geometry of the fueled cartridges and surrounding structure, the reactivity worth is predicted to be -44 pcm, which is in very good agreement with the MCNP5 model. The excess reactivity calculated with the DIF3D model for an all fresh LEU core with the 2017-RBM-99 upgrade and typical experiments in the graphite reflector and center test hole agrees with the detailed MCNP5 model to within 4 pcm. The current experimental configuration and experiment loading plus the 2017-RBM-99 upgrade results in an all fresh LEU core excess reactivity that is 1064 pcm greater than with the typical experimental configuration and experiment loading in 2008.

The results presented in Table 2.1 for HEU and Table 2.2 for LEU show that the MURR_2018_Mo DIF3D models for HEU and LEU cores with all experiments modeled closely match the excess reactivity predicted by the more detailed MCNP5 models. Table 2.3 provides a comparison of the power produced in each driver fuel element in the all fresh HEU and LEU cores predicted by the MCNP5 and DIF3D models. The power in MCNP5 is calculated from an f7 tally, which conservatively assumes that all energy is deposited locally in the fuel. Work presented in Reference 28 showed that 94.0% and 96.4% of the energy from fission and gamma-heating is deposited in the primary system for HEU and LEU cores, respectively, while the balance of the total energy is deposited outside the primary system (e.g., reflectors and pool). This is taken into account in the thermal-hydraulics analysis when calculating the steady-state margin to flow instability (Section 3) and the calculation of the margin to the fuel temperature safety limit during accidents (Section 4) to avoid being overly conservative. In the DIF3D calculation, the energy from fission and capture of excess neutrons is calculated for each fueled and non-fueled region in the model. To put the DIF3D results in Table 2.3 on the same basis as the MCNP5 results, the driver element powers were renormalized to a total power of 10 or 12 MW in the fueled regions for HEU or LEU, respectively. The results show that the DIF3D MURR_2018_Mo model used in the fuel cycle simulation predicts the driver element power to within 0.8% of that predicted by the more detailed MCNP5 model.

Figure 2.4 additionally compares the axial heat flux profile calculated by the DIF3D and MCNP5 MURR_2018_Mo models for fuel plate 1 in fuel element position 5 for the all fresh HEU core. Fuel plate 1 is closest to the center test hole and is near the location of the limiting plate for HEU cores. Fuel element position 5 is adjacent to the 2017-RBM-99. The calculations are for a core loaded with all fresh HEU fuel, the control blades at 23 inches withdrawn, the regulating blade at 10 inches withdrawn, and typical experimental loading conditions in the MURR_2018_Mo model (i.e., center test hole, graphite reflector, and 2017-RBM-99). The heat flux values are normalized to a total power of 10 MW produced by the eight HEU elements and assume all heat is deposited inside the fuel plates and is conducted evenly out the two radial surfaces of the fuel plate. The comparison shows that the DIF3D model provides a good prediction of the axial power profile for the HEU core compared to the more detailed MCNP5 model. The difference between DIF3D and MCNP5 for the peak heat flux in plate 1 is 2.2%.

Table 2.2. Preparation of LEU MURR_2018_Mo DIF3D Model.

Case	Description	MCNP				DIF3D			Diff. DIF3D vs. MCNP)
		k-eff	σ (pcm)	Experiment Reactivity Worth (pcm)		k-eff	Experiment Reactivity Worth (pcm)		
				Ind.	Total		Ind.	Total	
0	Base case. All fresh LEU fuel. Blades at 23" withdrawn; reg blade at 10" withdrawn. Empty holder in center test hole. Graphite-filled aluminum boxes with no experiment positions (other than beam tubes) in all wedge positions. Remove hanging wedge.	1.08287	8			1.08295			7
2	MCNP5. Insert 2017-RBM-99 in L and N positions in graphite reflector.	1.08229	7	-49	-49				
2a	DIF3D. 2017-RBM-99 installed in L- and N-wedges.					1.08243	-44	-44	12
4	MCNP5. Change combined GHIJ and hanging wedges to 2018 conditions.	1.07556	8	-578	-628				
4f	DIF3D. 570 ppm B added to graphite of combined GHIJ wedge.					1.07567	-580	-625	9
5	MCNP5. Change 3-wedge to 2018 conditions.	1.07477	7	-68	-696				
5b	DIF3D. 60 ppm B added to graphite of 3-wedge.					1.07486	-70	-695	8
6	MCNP5. Change 4-wedge to 2018 conditions.	1.07413	7	-55	-751				
6b	DIF3D. 55 ppm B added to graphite of 4-wedge.					1.07421	-56	-751	7
7	MCNP5. Change 6-wedge to 2018 conditions.	1.07304	7	-95	-846				
7a	DIF3D. 92 ppm B added to graphite of 6-wedge.					1.07304	-101	-852	0
8-3	MCNP5. Change center test hole between axial planes - 7.5 to +12.3 cm from empty-holder to 2018 conditions.	1.07721	7	+361	-485				
8-3c	DIF3D. Aluminum spacers with 90 ppm B in center test hole sample positions between axial mesh 25-33 (roughly equiv. to MCNP axial positions -7.5 to +12.3 cm)					1.07725	+363	-489	3
8-4	MCNP5. Change entire holder to 2018 conditions.	1.07914	8	+166	-319				
8-4j	DIF3D. 128 ppm B in center test hole sample positions between axial mesh 10-25. 128 ppm B in center test hole sample positions between axial mesh 33-38.					1.07919	+167	-322	4

Table 2.3. Driver Element Power Calculated by MCNP5 and DIF3D for HEU and LEU Cores.

Driver Fuel Element Position	HEU			LEU		
	Element Power (MW)		Difference	Element Power (MW)		Difference
	MCNP5	DIF3D		MCNP5	DIF3D	
1	1.228	1.235	0.6%	1.469	1.465	-0.3%
2	1.244	1.252	0.6%	1.491	1.492	0.1%
3	1.244	1.252	0.6%	1.494	1.504	0.7%
4	1.265	1.259	-0.5%	1.517	1.523	0.4%
5	1.242	1.233	-0.7%	1.494	1.496	0.1%
6	1.273	1.263	-0.8%	1.528	1.525	-0.2%
7	1.255	1.250	-0.4%	1.508	1.498	-0.7%
8	1.250	1.256	0.5%	1.498	1.497	-0.1%
Core	10.0	10.0	0.0%	12.0	12.0	0.0%

Figure 2.4. Axial Heat Flux Profile in Plate 1 of Element Position 5 in All Fresh HEU Core with 2017-RBM-99 Upgrade.

Control blades at 23 inches withdrawn, regulating blade at 10 inches withdrawn, and all experiments for "MURR_2018_Mo" conditions modeled.

Figure 2.5 compares the axial heat flux profile calculated by the DIF3D and MCNP5 MURR_2018_Mo models for fuel plate 23 in fuel element position 5 for the all fresh LEU core. Fuel plate 23 is closest to the reflector and the 2017-RBM-99 and is near the location of the limiting plate for LEU cores. The calculations are for a core loaded with all fresh LEU fuel, the control blades at 23 inches withdrawn, the regulating blade at 10 inches withdrawn, and typical experimental loading conditions in the MURR_2018_Mo model. The heat flux values are normalized to a total power of 12 MW produced by the eight LEU elements and assume all heat is deposited inside the fuel plates and is conducted evenly out the two radial surfaces of the fuel plate. A flattening of the axial power profile in plate 23 at the axial position near the tip of the regulating blade (which is inserted from above) can be seen in the

results from the more detailed MCNP5 model. This is not as apparent in the DIF3D result because of the coarser axial mesh. The comparison shows that the DIF3D model provides a good prediction of the axial power profile for the outermost fuel plate closest to the 2017-RBM-99. The difference between DIF3D and MCNP5 for the calculated peak heat flux in plate 23 is 2.5%.

Figure 2.5. Axial Heat Flux Profile in Plate 23 of Element Position 5 in All Fresh LEU Core with 2017-RBM-99 Upgrade.

Control blades at 23 inches withdrawn, regulating blade at 10 inches withdrawn, and all experiments for “MURR_2018_Mo” conditions modeled.

Using the DIF3D model described above, a fuel cycle simulation was performed which modeled more than 80 weeks of operations in MURR with typical fuel management in HEU- and LEU-fueled cores. The analysis assumed the same shuffling operations as modeled for the preliminary design and PSAR analysis with the MURR_2008 model [1,14] and is comparable to current MURR HEU operations. The simulation utilizes each MURR fuel element for about 19 different cycles of operation at a core power level of 10 MW for HEU and 12 MW for LEU before discharge at an element average burnup of around 150 and 180 MWd for HEU and LEU, respectively.

The effect of the 2017-RBM-99 on the core power distribution and the plate-level burnup of the driver fuel elements was evaluated in this work. It was, however, beyond the scope of this work to evaluate the burnup or Mo-99 production levels of the fueled cartridges in the 2017-RBM-99 during reactor operations. It is anticipated that the fueled cartridges would be replaced in typical operation each week or every other week, which is a much shorter residence time than the driver fuel elements. Thus, the amount of fuel depletion in the fueled cartridges will be relatively small compared with the driver element depletion. Therefore, the fueled cartridges in the 2017-RBM-99 were not depleted in the fuel cycle analysis performed in this work. This assumption is also conservative with regard to assessing the impact of the 2017-RBM-99 on the driver fuel element power distribution since the fresh cartridges will have the strongest influence on the core power distribution.

The HEU and LEU fuel cycle simulations assumed an initial core loading of all fresh fuel. While this is not typical for MURR operations, it was necessary in order to prepare a collection of elements that have some prior burnup that are typically selected from in loading the fuel for weekly operations. The simulations progressed with weekly fuel shuffling that followed typical fuel management practices in MURR. After about 50 weeks, all of the elements in the simulation that had been utilized in the “transition cycles” had reached their discharge burnup. The remaining cycles in the simulation were considered “pseudo-equilibrium cycles” that could be evaluated for deriving suitable reference core states.

The neutron flux calculation performed by DIF3D and used in the fuel depletion analysis in REBUS is based on a power normalization to a user-specified total reactor power. When the fueled cartridges of the 2017-RBM-99 are included in the model, the power produced in the cartridges is included as a part of the total reactor power in the DIF3D power normalization. In order to maintain a power of 10 or 12 MW produced by the eight driver fuel elements in the HEU or LEU cores, respectively, the total power in the DIF3D model must be increased by the power produced by the 2017-RBM-99 devices. The exact power sharing between the eight driver elements in the core and the 2017-RBM-99 during the pseudo-equilibrium cycles is not known prior to the simulation and must be determined using an iterative approach in the analysis.

The power produced by the 2017-RBM-99 devices for a core loaded with all fresh fuel was calculated from the MCNP5 model described above and added to the total core power in the DIF3D model. As the fuel in the driver elements depletes, the core neutron flux that is needed to maintain the specified core power increases because of parasitic absorption in the fission products. This, in turn, causes an increase in the power produced by the 2017-RBM-99 because there is a greater leakage of neutrons from the core with burned driver elements to the subcritical fueled cartridges. Likewise, there is an increase in the core neutron flux and 2017-RBM-99 power from the beginning-of-cycle (BOC) core state with no xenon to the core state with the equilibrium xenon concentration. Consequently, the calculated power produced by the core during the pseudo-equilibrium cycles was under predicted by a small amount, 51 kW and 48 kW, respectively, for the HEU and LEU fuel cycle simulations. The total reactor power assumed in HEU and LEU REBUS-DIF3D models was increased by these amounts and the simulation was repeated. In the second iteration, the total power produced by the eight driver elements in the core is under predicted by < 1 kW (< 0.01%), which is considered acceptable.

A summary of results from the HEU and LEU fuel cycle simulations is presented in Table 2.4. Results presented for the MURR_2008 model that are based on experimental facilities and loadings that were typical in 2008 and are the basis for the reference cores utilized for the safety analysis performed as part of the PSAR [14]. The MURR_2018_Mo model includes experimental facilities and loadings that are currently typical for MURR with the 2017-RBM-99 upgrade. The fuel management pattern used for the 2017-RBM-99 upgrade analysis was assumed to be the same as in the PSAR analysis, so the element burnups are not significantly affected by the upgrade. Each of the pseudo-equilibrium cycles is loaded with eight driver fuel elements with a mixture of burnups, ranging from BOL to an EOL burnup of 150 or 180 MWd for HEU or LEU, respectively. The BOC core burnup varies from 500 to 644 MWd in the 33 HEU pseudo-equilibrium cycles. For the 32 LEU pseudo-equilibrium cycles, the BOC core burnup varies from 595 to 822 MWd. The average values presented in Table 2.4 are the average for all pseudo-equilibrium cycles. The average core burnup of the LEU pseudo-equilibrium cycles is about 20% greater than that for the HEU cores, reflecting the power uprate from 10 to 12 MW.

For each cycle in the simulation, the control blades were fixed at 23 inches withdrawn and the regulating blade was positioned at 10 inches withdrawn. This control blade position is typical for

most of the weekly fuel cycle at equilibrium xenon conditions, so it is relevant for the fuel cycle simulation to calculate depleted fuel compositions. With the control blades at this position, the calculated core reactivity at end of cycle state averages about 765 and 505 pcm above critical for the HEU and LEU pseudo-equilibrium cycle cores, respectively. Critical blade positions at all points in the weekly cycle (beginning of cycle, equilibrium xenon, and end of cycle) and detailed three-dimensional power distributions needed for steady-state and transient accident analyses were calculated with MCNP5 once the reference core was selected (see Section 2.3).

Table 2.4. Summary of Results from Pseudo-Equilibrium Cycles in Fuel Cycle Simulation for HEU and LEU Cores with 2017-RBM-99.

		HEU		LEU	
		MURR_2008	MURR_2018_Mo	MURR_2008	MURR_2018_Mo
Core Power, MW		10	10	12	12
Number of Pseudo-Equilibrium Cycles Evaluated		33	33	32	32
Total Core Burnup at BOC, MWd	Minimum	500	500	594	595
	Maximum	644	644	822	822
	Average	577	576	689	689
Core k_{eff} at EOC¹	Maximum	0.99905	1.01310	0.99671	1.00887
	Minimum	0.98843	1.00279	0.98704	0.99935
	Average	0.99332	1.00765 (1431 pcm greater than MURR_2008 model with HEU)	0.99282	1.00505 (1226 pcm greater than MURR_2008 model with LEU)
Average Element Discharge Burnup, MWd		150	150	180	180
Elements/year		21.9	21.9	21.9	21.9

¹Control blades at 23 inches withdrawn, regulating blade at 10 inches withdrawn. Control blades contain fresh BORAL[®] material. Fresh beryllium reflector.

The current target discharge burnup for HEU elements in MURR is currently limited to 150 MWd by the requirements on using the BMI-1 spent fuel shipping cask [23]. This burnup is reached in HEU elements after about 19 weeks of typical operations. The fuel cycle simulation completed for the PSAR analysis using the MURR_2008 model and the current analysis with the 2017-RBM-99 upgrade (MURR_2018_Mo model) assumed the same fuel management for both HEU and LEU cores to provide a relevant comparison to current operations. At an HEU element burnup of 150 MWd, the peak local fission density in HEU plates is $< 1.6 \times 10^{21}$ fissions/cm³ [1], which is substantially lower than the Technical Specification limit of 2.3×10^{21} fissions/cm³ for HEU aluminide fuel. Thus, the current MURR HEU elements do not reach the Technical Specification limit and could be discharged at a higher burnup if supported by core safety analysis and the acquisition of a different spent fuel shipping container.

A peak local LEU burnup of 3.4×10^{21} fissions/cm³ at an element burnup of 180 MWd following use in 19 weekly fuel cycles was calculated in the PSAR analysis and in the current work with the

2017-RBM-99 upgrade. A discharge burnup limit for U-10Mo monolithic LEU fuel will be established through fuel qualification experiments.

The PSAR analysis completed with the MURR_2008 model found that the reactivity-limited burnup for HEU and LEU elements with the experimental facilities and experiment loadings that existed in 2008 was ~150 and ~180 MWd, respectively. The average EOC reactivity calculated for the pseudo-equilibrium cycles with the control blades at 23 inches withdrawn was ~700 pcm below critical for both HEU and LEU, resulting in a critical control blade position near the 26 inch control blade withdrawal limit. Note that the analysis was performed assuming all control blades containing fresh BORAL® material, so that the critical position for control blades with prior use, which is typical for MURR, would be more deeply inserted than the 26 inches withdrawn position.

Changes to the MURR experimental facilities and experiment loadings since 2008, which are discussed above in Section 2.1 and illustrated in Figure 2.1, have caused an increase of the core excess reactivity for a given HEU core burnup. This reactivity increase is largely the result of the installation of the FIRST device which allows for greater positive reactivity worth of experiments loaded in the center test hole. Results from calculations for the MURR_2018 model are not provided in Table 2.4 since it is beyond the scope of the current work, but analysis with MCNP5 indicates that the reactivity-limited burnup for HEU and LEU is greater than 150 and 180 MWd, respectively. It is shown in Table 2.4 that the average EOC reactivity for the 23 inches withdrawn control blade position calculated for the HEU pseudo-equilibrium cycles with the 2017-RBM-99 upgrade is 1431 pcm greater than that calculated with the control blades at the same position in the MURR_2008 model used in the PSAR analysis. The average EOC reactivity for the LEU pseudo-equilibrium cycles with the 2017-RBM-99 upgrade is 1226 pcm greater than that calculated with the MURR_2008 model. This confirms that the reactivity-limited element discharge burnup with the 2017-RBM-99 upgrade is greater than 150 and 180 MWd for HEU and LEU, respectively. However, since the goal of the current analysis was to investigate the impact of the 2017-RBM-99 upgrade on the safety margins relative to the analysis performed for the PSAR, analysis of the burnup extension that is possible as a result of the upgrade or the current facility experiment loading was not completed as part of this work.

2.3 Reference Core States

The power distributions calculated by REBUS-DIF3D in the fuel cycle simulation for each of the pseudo-equilibrium cycles with fuel at a mixture of burnups, which is typical for MURR, were examined to find the cycle which yielded the highest local heat flux on a fuel plate. Heat flux values were calculated for 12 axial segments (two-inch segment height) for each fuel plate. For computational efficiency, the DIF3D model neglected any azimuthal power peaking in the fuel plates. Azimuthal power peaking was found from previous analysis [1] to be small (~10%) relative to axial peaking (~60%) for the most limiting plates. Azimuthal power peaking is included in the detailed three-dimensional power profiles calculated by MCNP5. Figure 2.6 provides a summary of the maximum local heat flux on a fuel plate for each of the 32 pseudo-equilibrium cycles for the LEU core with the 2017-RBM-99 upgrade. Results are presented for four different time points in each cycle, at 0, 1, 2, and 6.3 days of the cycle. The heat flux values are normalized to a total power of 12 MW produced by the eight LEU driver elements and assume all heat is deposited within the fuel and is conducted evenly out the two radial surfaces of the fuel plate.

Figure 2.6. Maximum Heat Flux in LEU Pseudo-Equilibrium Cores with the 2017-RBM-99 Upgrade.

It is observed from Figure 2.6 that, in general, the maximum heat flux on a fuel plate in a cycle decreases with the total core burnup. Most cycles loaded in the core have eight elements with some previous burnup, represented by the open circles in Figure 2.6. MURR fully utilizes about 22 driver elements/year, so about every fifth week (~10 times a year) two elements reach their end of life and are discharged from the fuel cycle and two BOL elements are added to the fuel cycle. The simulation was designed so that the BOL and most highly depleted elements are loaded in the same cycle and are placed in core positions adjacent to each other in order to maximize the power peaking that occurs in the BOL fuel elements. As expected, the heat flux is greater in those cycles in which a BOL element is loaded next to an element that is near its discharge burnup, which are represented by the solid circles in Figure 2.6. Among all the pseudo-equilibrium cycles, the cycle labeled as “Week 71” was found to have the highest local heat flux. This core is loaded with two BOL elements, two elements at EOL, and four elements at middle-of-life (MOL). The maximum heat flux was found for plate 23 of the BOL element in core position 5, which is adjacent to the location of the 2017-RBM-99 upgrade. This core was selected for more detailed analysis with MCNP5.

A similar approach was used to identify the HEU core with the 2017-RBM-99 upgrade that had the highest local heat flux. Based on the same selection criteria as used for LEU, the “Week 73” core from the HEU fuel cycle simulation was selected for more detailed evaluation. Similar to the LEU reference core, this core is loaded with two BOL elements, two EOL elements, and four MOL elements. The maximum heat flux calculated by the DIF3D analysis occurs in fuel plate 1 of the fresh element in core position 5.

The burnups of each HEU and LEU driver element in these “reference cores” are tabulated in Table 2.5.

Table 2.5. BOC Burnup Elements in HEU and LEU Reference Cores with the 2017-RBM-99 Upgrade.

Core Driver Position	Element Burnup (MWd)	
	HEU	LEU
1	0.0	0.0
2	81.0	95.8
3	73.4	77.3
4	142.5	171.0
5	0.0	0.0
6	81.1	96.3
7	73.3	76.9
8	142.1	169.5
Total	593.3	686.9

The detailed compositions by fuel plate and axial position for all eight elements in the reference cores were extracted from the REBUS-DIF3D results and used to prepare material composition cards for the MCNP5 analysis of critical blade height and detailed, three-dimensional power distributions. This was done for the LEU Week 71 reference core with the 2017-RBM-99 upgrade at BOC (no xenon conditions), equilibrium xenon (2 days), and EOC (6.3 days) conditions. Material compositions by plate and axial level were also extracted from the REBUS-DIF3D results for the HEU Week 73 reference core with the 2017-RBM-99 upgrade at BOC, equilibrium xenon, and EOC. Depleted LEU fuel material compositions were extracted for 2,208 fuel compositions (8 elements x 23 plates x 12 axial zones). For HEU, 2,304 fuel compositions (8 x 24 x 12) were prepared for the MCNP5 analysis.

Power distributions were calculated for several variations of the reference cores in order to identify the conditions that yield the smallest margin to the onset of flow instability (OFI) during steady-state operations with the 2017-RBM-99 upgrade. Power peaking which may limit safety margins in the MURR core is dependent upon the mixture of burnup states among the elements in the core, the core xenon buildup state, critical control blade compositions and positions, and the experiment/sample loadings, particularly in the center test hole. A number of cases that covered the range of expected variations in these conditions were examined. Cases were labeled with a nomenclature *cfb*, where *c* indicates the fuel type and its burnup condition, *f* indicates the center test hole loading, and *b* indicates the control blade conditions. These conditions are discussed below.

The core fuel type and burnup conditions (“*c*” in the case name label) examined are as follows:

HEU		LEU	
<i>c</i> =1	All elements fresh, BOC (no Xe)	<i>c</i> =5	All elements fresh, BOC (no Xe)
<i>c</i> =2	All elements fresh, day 2 (eq. Xe)	<i>c</i> =6	All elements fresh, day 2 (eq. Xe)
<i>c</i> =3	Reference mixed-burnup core (“Week 73”), BOC (no Xe)	<i>c</i> =7	Reference mixed-burnup core (“Week 71”), BOC (no Xe)
<i>c</i> =4	Reference mixed-burnup core (“Week 73”), day 2 (eq. Xe)	<i>c</i> =8	Reference mixed-burnup core (“Week 71”), day 2 (eq. Xe)
<i>c</i> =0	Reference mixed-burnup core (“Week 73”), day 6.3 (EOC)	<i>c</i> =9	Reference mixed-burnup core (“Week 71”), day 6.3 (EOC)

In addition to evaluating the power distribution for the reference mixed-burnup cores derived from the fuel cycle simulation, cores loaded with eight fresh LEU elements were also evaluated. The first core following conversion to LEU will contain all fresh LEU elements. Low-power physics testing will

also be conducted for this initial LEU core. The case with eight fresh HEU elements was also evaluated to complement the LEU case. Transition from cores loaded with all fresh LEU to LEU cores loaded with elements with a mixture of burnup that are typical for MURR are being evaluated in a separate activity. The transition core analysis is being performed without the 2017-RBM-99 upgrade but for experimental facilities and experiment loading that currently exist at MURR. It is expected that the thermal-hydraulic margins for any transition cores will be bounded by the margins calculated for all fresh LEU and the reference mixed-burnup core. This will be confirmed in the transition core analysis.

Two center test hole loading conditions were considered. The nominal case, indicated by $f=C$ in the case name label (e.g., 3C, 7C), has the center test hole loaded with samples that are typical for current MURR operations and the 2017-RBM-99 upgrade. The complementary cases evaluated for the PSAR analysis used the nomenclature $f=A$ (e.g., 3A, 7A). These models included the fast-flux irradiation facilities in the L and N graphite wedge positions and experimental facilities and experiment loadings that were typical in 2008. An off-nominal “empty trap” case, in which the center test hole is emptied of samples and the holder apparatus, is indicated by $f=D$ with the 2017-RBM-99 upgrade.

Reactivity control during the MURR weekly operating cycle is provided by four movable control blades located between the outer pressure vessel and beryllium reflector that contain BORAL® as a poison material with an AA6061 cladding that extends 1/8 inch below the bottom of the poison material. The most deeply inserted blade position measured from the bottom of the control blade cladding is about two inches (1.875 inches) below the bottom of fuel. The poison material is a little over 34 inches long, so in the fully inserted position the poison material extends 8 inches above the fuel. The blades have a total travel length of 26 inches, so that the fully withdrawn position of the control blades is with the bottom of the blade at the top of the fuel. The control blades are designated as A, B, C, and D and can move independently, but are usually banked together at the same axial position. Fine adjustments of the core reactivity are provided by a stainless steel regulating blade that is adjacent to core driver element position 5.

The control blade condition assumed in the fuel cycle simulation was that all four BORAL® blades are at a nominal fresh composition and banked in position. The effect on the core power distribution of perturbations of the effective control blade ^{10}B concentration due to poison material depletion and mismatched positioning of the blades was examined since variations of this sort can exist in MURR. The four control blades in MURR will usually be at different points in their lifetime. The range of megawatt-days for an operational control blade, measured as the core burnup accumulated during the blade residence, is from 0 MWd up to 33,000 MWd, after which it is retired. Blades are typically utilized in two-year increments, which allows for required routine inspection and cooling of the blades. As discussed in Section 2.2.4 of Reference 1 and in Reference 26, the poison material near the bottom four inches of the poison material is significantly depleted during the blade lifetime. Since the control blade position is measured from the bottom of the blade cladding, this means that the poison material in a blade near the end of its life is effectively positioned four inches higher relative to a fresh blade. The differences in the ^{10}B concentration at the tip of the control blades can create a radial tilt in the power distribution across the core. Furthermore, the MURR Technical Specifications allow for the height of the blade tips to be mismatched by up to one inch. As extreme scenarios, perturbed heat flux profiles were calculated with two of the blades at their end of life (33,000 MWd) and positioned one inch above the other two blades, which were assumed to be fresh.

The cores for each fuel type (HEU and LEU) were evaluated under three conditions with regard to blade depletion and positioning, which is designated by the character “ b ” in the case name, indicated as follows:

$b=\text{Blank}$	Blades A, B, C, and D at fresh nominal conditions and banked.
$b=1$	Blades A and B fresh, blades C and D at 33,000 MWd. Blades C and D positioned 1 inch higher than A and B, per Technical Specification.
$b=2$	Blades B and C fresh, blades A and D at 33,000 MWd. Blades A and D positioned 1 inch higher than B and C, per Technical Specification.
$b=3$	Blades A and D fresh, blades B and C at 33,000 MWd. 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 2.7. 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 1 inch more withdrawn than the fresh blades creates an even greater power tilt. Calculations for a reference LEU core in the PSAR analysis without the 2017-RBM-99 upgrade found that the power peaking is greatest in the fresh element located in core position 1 near its interface with an element that is near its discharge burnup located in core position 8. The $b=1$ and $b=2$ perturbations were considered because they were expected to cause the greatest increase of the power peaking at the interface between these two elements. With the 2017-RBM-99 upgrade, the additional case with blades B and C at their maximum burnup and withdrawn 1 inch more than the fresh blades A and D ($b=3$) was included to increase the radial power tilt toward elements in core positions 4, 5, and 6 which are adjacent to the device. A case with blades C and D being fresh with blades A and B at their maximum burnup was not considered because it would not produce more limiting power distributions than the other cases.

Figure 2.7. Control Blade Perturbations Considered in Analysis of Minimum Safety Margin.

Detailed three-dimensional power distributions were calculated using MCNP5 for all of the HEU and LEU cores with the 2017-RBM-99 upgrade. The critical positions for the fresh and depleted control blades (for cases with $b \neq 0$) 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.

The power distributions were calculated with MCNP5 by tallying the fission power (f7 tally) within 24 axial, 9 azimuthal, and either 24 (HEU) or 23 (LEU) radial segments (plate-by-plate) of the fuel

meat in the entire core of eight elements (i.e., either 5,184 (HEU) or 4,968 (LEU) segments per element). While the power distribution varies most significantly in the axial and radial (plate-by-plate) dimensions of the fuel element, the power density also varies along the width of the fuel in the plate, and can peak sharply at the edge of the fueled section of the plate due to an increase in the local water-to-fuel ratio near the edge of the fuel. Lateral conduction of the heat generated by fission to the unfueled portion of the fuel plate is not modeled in the one-dimensional steady-state thermal-hydraulics analysis employed for MURR. Rather, an appropriately-sized finite “hot-stripe” width was selected for the power tally segment width that yields conservative estimates of the fuel temperature and the heat conduction into the coolant channel for the flow instability analysis.

A three-dimensional computational fluid dynamics (CFD) analysis performed as a part of the PSAR analysis concluded that the MCNP power distributions should be tallied by azimuthally subdividing the fuel of each fuel plate into nine tally stripes; three stripes of 0.5 cm (5 mm) each along both edges of the fuel, and three stripes of equal angle in the middle portion of the fuel to cover the balance of the fuel width [29]. However, that analysis was conducted with an earlier feasibility study LEU element design. As part of the present work, the three-dimensional heat conduction analysis was completed for the preliminary LEU element design using the COMSOL 5.3a software [30]. The modeling and results of that analysis are described in detail in Appendix B. The study concluded that the 0.5 cm stripe width along the edge of the fuel is an appropriately conservative “hot-stripe” width.

The MCNP5 tallies were normalized by a post-processor to facilitate studies of different core power levels and different levels of tally detail. It should be noted that the heat flux results presented in this section assume that all heat is deposited in the fuel and is conducted evenly out the two radial surfaces of the fuel plate. These are both conservative assumptions. The safety margin analyses presented in Sections 3 and 4 account for the uneven heat conduction from the two radial surfaces of the fuel plate that is caused by the asymmetric coolant channel conditions on each side of a fuel plate, particular for plates 1 and 24 of the HEU element and plates 1 and 23 of the LEU element which are cooled on one side by coolant channels that are heated by only a single fuel plate. Additionally, the heat flux in the safety margin analysis was reduced by the fraction of energy deposition that occurs outside the primary system.

Table 2.6 provides a summary of the MCNP5 calculations for each of the HEU cores with the 2017-RBM-99 upgrade evaluated in this work. As described above, the identifier *cfb* for each core indicates the core burnup/xenon state, center test hole loading, and control blade conditions. The hot-stripe heat flux is the maximum axially-averaged heat flux among all azimuthal stripes in all plates in the core. The hot-stripe heat flux is a figure-of-merit for comparing core states that may be limited by the onset of flow instability since it represents the total heat that must be transferred to the water in the coolant channel. The maximum hot-stripe heat flux of 144.1 W/cm² occurs in Core 3D3 for fuel plate 1 in the BOL element in core position 5 that is adjacent to the 2017-RBM-99. Table 2.7 compares the maximum calculated hot-stripe heat flux for HEU cores with the 2017-RBM-99 upgrade with the results from the PSAR analysis (without the 2017-RBM-99 upgrade). For the PSAR analysis, the maximum hot-stripe heat flux is slightly lower (141.3 W/cm²) and occurred in core 3B2 for fuel plate 1 in the BOL element in core position 1. The increase of the maximum hot-stripe heat flux and the change in the element in which it occurs for the set of HEU cores with the 2017-RBM-99 upgrade relative to the results of the PSAR analysis gives some indication of the impact of the upgrade on the core power distribution. Complete results of the impact of the upgrade on the HEU margins to steady-state flow instability are provided in Section 3.

Table 2.6. Maximum Hot-Stripe and Local Heat Flux for HEU Cores with the 2017-RBM-99 Upgrade.

Core	Critical Blade Position (inches withdrawn)		Reg. Blade Position (in. withdrawn)	Hot-Stripe Heat Flux (W/cm ²)		Local Heat Flux (W/cm ²)	
	Fresh	Depleted		Maximum	Location ¹	Maximum	Location ²
1C	11.89	N/A	10	131.7	X5 S1 P1	189.0	X5 A17 S1 P1
1C1	9.17	10.17	10	134.6	X7 S9 P1	189.8	X7 A17 S9 P1
1C2	9.44	10.44	10	133.0	X2 S1 P1	187.5	X2 A17 S1 P1
1C3	9.04	10.04	10	136.8	X5 S1 P1	193.1	X5 A17 S1 P1
2C	15.24	N/A	15	126.5	X5 S1 P1	179.8	X5 A16 S1 P1
2C1	12.79	13.79	15	128.6	X8 S1 P1	176.4	X8 A16 S1 P1
2C2	13.06	14.06	15	127.4	X1 S9 P1	175.6	X1 A16 S9 P1
2C3	15.59	16.59	15	130.5	X5 S1 P1	180.1	X5 A16 S1 P1
3C	16.35	N/A	10	136.6	X5 S1 P1	191.9	X5 A16 S1 P1
3C1	13.94	14.94	10	136.2	X5 S9 P1	186.4	X5 A16 S9 P1
3C2	14.17	15.17	10	136.7	X1 S1 P1	185.8	X1 A16 S1 P1
3C3	13.86	14.86	10	140.6	X5 S1 P1	192.0	X5 A16 S1 P1
4C	21.12	N/A	15	130.3	X5 S1 P1	171.2	X5 A15 S1 P1
4C1	19.02	20.02	15	130.0	X5 S1 P1	169.6	X5 A15 S1 P1
4C2	19.24	20.24	15	129.9	X1 S1 P1	169.3	X5 A14 S1 P1
4C3	18.96	19.96	15	132.9	X5 S1 P1	172.3	X5 A14 S1 P1
0C	21.74	N/A	15	128.8	X5 S1 P1	167.4	X5 A14 S1 P1
0C1	19.73	20.73	15	128.0	X5 S1 P1	165.2	X5 A14 S1 P1
0C2	19.96	20.96	15	127.8	X1 S1 P1	165.1	X5 A14 S1 P1
0C3	19.66	20.66	15	130.8	X5 S1 P1	167.8	X5 A13 S1 P1
1D	12.39	N/A	10	134.9	X5 S1 P1	193.8	X4 A17 S9 P1
1D1	9.71	10.71	10	139.4	X6 S9 P1	195.8	X7 A16 S1 P1
1D2	9.99	10.99	10	136.3	X2 S1 P1	192.3	X8 A17 S9 P1
1D3	9.59	10.59	10	141.0	X5 S9 P1	197.8	X5 A16 S9 P1
2D	15.92	N/A	15	129.9	X5 S9 P1	183.7	X5 A16 S9 P1
2D1	13.48	14.48	15	132.8	X6 S9 P1	183.2	X6 A15 S9 P1
2D2	13.77	14.77	15	130.1	X8 S9 P1	179.1	X8 A16 S9 P1
2D3	13.37	14.37	15	133.8	X5 S1 P1	183.3	X4 A16 S9 P1
3D	17.08	N/A	10	139.8	X5 S1 P1	194.8	X5 A15 S1 P1
3D1	14.72	15.72	10	139.9	X5 S9 P1	190.7	X5 A15 S9 P1
3D2	14.94	15.94	10	140.7	X1 S1 P1	191.3	X1 A15 S1 P1
3D3	14.64	15.64	10	144.1	X5 S1 P1	194.5	X5 A15 S1 P1
4D	22.41	N/A	15	133.4	X5 S1 P1	174.0	X5 A14 S1 P1
4D1	20.46	21.46	15	132.9	X5 S1 P1	172.2	X5 A13 S1 P1
4D2	20.68	21.68	15	132.3	X1 S1 P1	171.6	X5 A13 S1 P1
4D3	20.41	21.41	15	135.7	X5 S1 P1	174.6	X5 A13 S1 P1
0D	23.17	N/A	15	131.4	X5 S1 P1	170.0	X5 A13 S1 P1
0D1	21.35	22.35	15	131.1	X5 S1 P1	168.8	X5 A13 S1 P1
0D2	21.57	22.57	15	130.5	X1 S1 P1	167.7	X5 A13 S1 P1
0D3	21.30	22.30	15	133.2	X5 S1 P1	170.6	X5 A13 S1 P1

¹ Xi Sj Pk. i is element number. j is azimuthal stripe in plate, numbered 1-9 moving clockwise. k is plate, numbered 1-24 moving from inner pressure vessel to outer pressure vessel.

² Xi Az Sj Pk. i is element number. z is axial level, numbered 1-24 from top of fuel meat. j is azimuthal stripe in plate, numbered 1-9 moving clockwise. k is plate, numbered 1-24 moving from inner pressure vessel to outer pressure vessel.

The calculated maximum local heat flux on the surface of the HEU fuel plates is also provided in Table 2.6. It can be expected that there will be a higher fuel centerline temperature for core states with a larger maximum local heat flux. Consequently, the maximum local heat flux is a figure-of-merit for comparing core states that may be limited by the margin to the fuel temperature safety limit during accident scenarios. Among all the HEU cores with the 2017-RBM-99 upgrade, the maximum local heat flux of 197.8 W/cm² occurs in core 1D3 in fuel plate 1 of the BOL element in core position 5. This all fresh fuel core is not prototypic for MURR operations and has a critical control blade position that is much lower than is typical for MURR cores at BOC. Table 2.7 shows that for the PSAR analysis without the 2017-RBM-99 upgrade, the maximum local heat flux is similar, 197.6 W/cm², occurring in fuel plate 1 in a BOL element in core position 7 in core 1B1. The results of the impact of the 2017-RBM-99 upgrade on the HEU margins to the fuel temperature safety limit for accident scenarios in a core with fuel elements with a mixture of burnups that is typical for MURR are provided in Section 4.

Table 2.7. Comparison of Maximum Hot-Stripe and Local Heat Flux for HEU Cores with the 2017-RBM-99 Upgrade and from PSAR Analysis.

Model	Model	Core	Location	Heat Flux
Maximum Hot-Stripe Heat Flux	With 2017-RBM-99 Upgrade (MURR_2018_Mo)	3D3	X5 S1 P1	144.1
	PSAR (MURR_2008)	3B2	X1 S1 P1	141.3
Maximum Local Heat Flux	With 2017-RBM-99 Upgrade (MURR_2018_Mo)	1D3	X5 A16 S9 P1	197.8
	PSAR (MURR_2008)	1B1	X7 A16 S1 P1	197.6

Table 2.8 provides a summary of the MCNP5 calculations for each of the LEU cores with the 2017-RBM-99 upgrade that were evaluated in this work. Among all the cases that were evaluated, the maximum hot-stripe heat flux of 150.8 W/cm² occurs in core 8C3 for fuel plate 23 in the BOL element in core position 5 that is adjacent to the 2017-RBM-99. As shown in Table 2.9, this is lower than the maximum hot-stripe heat flux of 156.8 W/cm² for the PSAR analysis (without the 2017-RBM-99 upgrade), which occurs in core 8A2 for fuel plate 23 in the BOL element in core position 1. The change in the location and the *decrease* of the maximum hot-stripe heat flux with the 2017-RBM-99 relative to the PSAR analysis is due to the combined effects of the upgrade and the critical control blade position (as discussed above, the core excess reactivity is increased with the 2017-RBM-99 upgrade, so the control blades must be more deeply inserted) on the core power distribution. Complete results of the impact of the upgrade on the LEU margins to steady-state flow instability are provided in Section 3.

The calculated maximum local heat flux on the surface of the LEU fuel plates for all the variations of the reference cores is also provided in Table 2.8. The maximum local heat flux of 238.5 W/cm² occurs in core 5C3 in fuel plate 23 of the BOL element in core position 6 (all elements are fresh in this core). This all fresh fuel core has a critical control blade position that is much lower than is typical for MURR cores at BOC, which results in greater axial power peaking in plate 23 relative to the mixed-burnup cores. Table 2.9 shows that the maximum local heat flux was about 2% lower for the PSAR analysis (without the 2017-RBM-99 upgrade), 234.0 W/cm², and occurred in the same fuel plate and element (plate 23, element 6) in core 5B1. The results of the impact of the 2017-RBM-99 upgrade on the LEU margins to the fuel temperature safety limit for accident scenarios in a core with fuel at a mixture of burnups that is typical for MURR are provided in Section 4.

Table 2.8. Maximum Hot-Stripe and Local Heat Flux for LEU Cores with the 2017-RBM-99 Upgrade.

Core	Critical Blade Position (inches withdrawn)		Reg. Blade Position (in. withdrawn)	Hot-Stripe Heat Flux (W/cm ²)		Local Heat Flux (W/cm ²)	
	Fresh	Depleted		Maximum	Location ¹	Maximum	Location ²
5C	12.15	N/A	10	128.7	X5 S9 P4	228.2	X6 A18 S1 P23
5C1	9.47	10.47	10	132.3	X7 S1 P4	233.8	X6 A17 S9 P23
5C2	9.73	10.73	10	130.6	X1 S9 P4	223.5	X8 A18 S9 P23
5C3	9.34	10.34	10	135.1	X4 S9 P4	238.5	X6 A17 S1 P23
6C	15.68	N/A	15	125.3	X5 S9 P23	220.8	X6 A16 S1 P23
6C1	13.24	14.24	15	138.9	X6 S9 P23	225.7	X6 A16 S9 P23
6C2	13.52	14.52	15	132.3	X8 S9 P23	213.6	X7 A16 S9 P23
6C3	13.13	14.13	15	139.1	X5 S9 P23	225.7	X6 A15 S1 P23
7C	15.93	N/A	10	130.9	X5 S1 P4	206.5	X5 A17 S9 P23
7C1	13.52	14.52	10	139.5	X5 S9 P23	222.6	X1 A17 S1 P23
7C2	13.76	14.76	10	139.1	X1 S9 P23	216.4	X1 A15 S9 P23
7C3	13.42	14.42	10	143.7	X5 S9 P23	214.9	X5 A17 S9 P23
8C	20.93	N/A	15	140.8	X5 S9 P23	199.7	X5 A14 S9 P23
8C1	18.82	19.82	15	148.0	X5 S9 P23	208.3	X5 A14 S9 P23
8C2	19.05	20.05	15	148.6	X1 S1 P23	198.2	X1 A14 S1 P23
8C3	18.78	19.78	15	150.8	X5 S9 P23	204.8	X5 A14 S9 P23
9C	21.45	N/A	15	140.3	X5 S9 P23	196.8	X5 A14 S9 P23
9C1	19.40	20.40	15	146.5	X5 S9 P23	203.8	X5 A14 S9 P23
9C2	19.63	20.63	15	147.8	X1 S1 P23	194.1	X1 A14 S1 P23
9C3	19.37	20.37	15	150.0	X5 S9 P23	202.1	X5 A14 S9 P23
5D	12.68	N/A	10	130.1	X5 S1 P4	227.5	X6 A18 S1 P23
5D1	10.03	11.03	10	134.2	X7 S1 P4	233.9	X6 A17 S9 P23
5D2	10.31	11.31	10	130.9	X1 S1 P4	224.1	X8 A17 S9 P23
5D3	9.91	10.91	10	136.3	X5 S9 P4	237.4	X6 A17 S1 P23
6D	16.42	N/A	15	126.4	X5 S9 P23	217.4	X6 A16 S1 P23
6D1	14.03	15.04	15	141.6	X6 S9 P23	224.9	X6 A15 S9 P23
6D2	14.32	15.32	15	134.5	X8 S9 P23	213.9	X7 A16 S9 P23
6D3	13.92	14.92	15	140.3	X5 S9 P23	221.2	X6 A15 S1 P23
7D	16.66	N/A	10	130.7	X5 S1 P4	206.1	X1 A16 S1 P23
7D1	14.29	14.29	10	140.1	X5 S9 P23	220.9	X1 A16 S1 P23
7D2	14.54	14.54	10	140.4	X1 S1 P23	213.2	X1 A15 S1 P23
7D3	14.22	15.22	10	144.9	X5 S9 P23	212.7	X5 A17 S9 P23
8D	22.23	N/A	15	141.6	X5 S9 P23	195.7	X5 A14 S9 P23
8D1	20.28	21.28	15	147.6	X5 S9 P23	203.3	X5 A14 S9 P23
8D2	20.51	21.51	15	149.9	X1 S1 P23	193.8	X1 A13 S1 P23
8D3	20.19	21.19	15	150.3	X5 S9 P23	200.7	X5 A13 S9 P23
9D	22.87	N/A	15	141.2	X5 S9 P23	193.0	X5 A14 S9 P23
9D1	21.00	22.00	15	147.2	X5 S9 P23	200.7	X5 A13 S9 P23
9D2	21.24	22.24	15	149.4	X1 S1 P23	190.4	X1 A13 S1 P23
9D3	20.93	21.93	15	149.3	X5 S9 P23	197.9	X5 A14 S9 P23

¹ Xi Sj Pk. i is element number. j is azimuthal stripe in plate, numbered 1-9 moving clockwise. k is plate, numbered 1-23 moving from inner pressure vessel to outer pressure vessel.

² Xi Az Sj Pk. i is element number. z is axial level, numbered 1-24 from top of fuel meat. j is azimuthal stripe in plate, numbered 1-9 moving clockwise. k is plate, numbered 1-23 moving from inner pressure vessel to outer pressure vessel.

Table 2.9. Comparison of Maximum Hot-Stripe and Local Heat Flux for LEU Cores with the 2017-RBM-99 Upgrade and from PSAR Analysis.

Model	Model	Core	Location	Heat Flux
Maximum Hot-Stripe Heat Flux	With 2017-RBM-99 Upgrade (MURR_2018_Mo model)	8C3	X5 S9 P23	150.8
	PSAR (MURR_2008 model)	8A2	X1 S1 P23	156.8
Maximum Local Heat Flux	With 2017-RBM-99 Upgrade (MURR_2018_Mo model)	5C3	X6 A17 S1 P23	238.5
	PSAR (MURR_2008 model)	5B1	X6 A17 S9 P23	234.0

A comparison of the hot-stripe heat flux in each LEU fuel plate in reference cores with a mixture of fuel burnup in the driver elements at equilibrium xenon conditions is provided in Figure 2.8. It should be noted that the plate ordering for the radial profiles for elements 1-3 and 8 is displayed in reverse order because it illustrates how the radial power profile changes as a result of the 2017-RBM-99 upgrade. The hot-stripe heat flux profiles for Core 8C with the 2017-RBM-99 upgrade and for Core 8A evaluated as part of the PSAR analysis are provided. These cores are comparable in configuration (comparable fuel element burnup, experiments in the center test hole, fresh and banked control blades). These cores were selected for comparison because the results in Table 3.3 in Section 3 show that, although the minimum margin to steady-state flow instability in LEU cores occurs in a reference core configuration other than Cores 8A and 8C, the smallest margins do occur at equilibrium xenon conditions. Consequently, these cores were selected for the comparison to show the impact of the 2017-RBM-99 upgrade on the MURR core power profile and the calculated margin to steady-state flow instability.

Likewise, a comparison of the maximum local heat flux in each LEU fuel plate in reference cores with a mixture of fuel burnup in the driver elements at BOC (no xenon conditions) for the PSAR analysis (7A) and with the 2017-RBM-99 upgrade (7C) is provided in Figure 2.9. Just as for Figure 2.8, the plate ordering for the radial profiles of the maximum heat flux for elements 1-3 and 8 is displayed in reverse order, because it illustrates how the radial power profile changes as a result of the 2017-RBM-99 upgrade. Reference cores 7A and 7C are comparable in configuration (comparable fuel element burnup, experiments in the center test hole, fresh and banked control blades). These cores were selected for comparison because they are the cores evaluated in the transient accident analyses for the calculation of the minimum margin to the fuel temperature safety limit. (See Section 4) The control blades are most deeply inserted at BOC (no xenon) conditions, which results in the greatest axial power peaking and the highest fuel temperatures. Consequently, these cores were selected for the comparison to show the impact of the 2017-RBM-99 upgrade on the MURR core power profile and the calculated margin to the fuel temperature safety limit.

The MURR LEU fuel element is designed with variable fuel foil thicknesses in the fuel plates to flatten the radial heat flux profile, particularly for the fuel plates near the center test hole. Fuel plate 1, which is closest to the center test hole, has the thinnest fuel foil (0.009 inch thick, excluding the zirconium interlayer). Plates 4-22 are designed with full-thickness fuel foils (0.020 inch, excluding the zirconium). Plate 23, which is closest to the beryllium reflector and the control blades, has a fuel foil that is 0.017 inch thick (excluding the zirconium) to reduce power peaking in the plate. The flattening of the radial heat flux profile as a result of the element design can be seen in Figure 2.8, particularly for the BOL elements in core positions 1 and 5.

Figure 2.8. Hot-Stripe Heat Flux at Eq. Xenon Conditions in Reference Mixed-Burnup MURR LEU Core with 2017-RBM-99 Upgrade (Core 8C) and in PSAR Model (Core 8A).

Figure 2.9. Maximum Local Heat Flux by Plate at BOC in Reference Mixed-Burnup MURR LEU Core with 2017-RBM-99 Upgrade (Core 7C) and in PSAR Model (Core 7A).

As discussed in Section 3.2, the minimum margin to steady-state flow instability for equilibrium xenon cores with the 2017-RBM-99 upgrade and for the PSAR analysis (without the upgrade) occurs in coolant channel 23, which is located between fuel plates 22 and 23. The total power and axial power profile in these plates is strongly influenced by the control blades, which are withdrawn throughout the cycle to maintain a critical operating condition. The control blades at equilibrium xenon are positioned in the upper 20% of the core. Consequently, the radial power profile shifts towards these fuel plates, so there is a smaller margin to flow instability in this coolant channel at equilibrium xenon relative to BOC conditions when the blades are positioned about two inches above the core midplane.

As shown in Figure 2.8 for cores 8C and 8A, which differ primarily in whether or not the 2017-RBM-99 upgrade is in place, the hot-stripe heat flux in fuel plates 22 and 23 of the BOL element in core position 1 is less with the upgrade than that in the reference core evaluated for the PSAR (9.2% reduction for plate 23 with the upgrade). There is also a slight increase (2.6%) in the hot-stripe heat flux in fuel plate 1 of this element with the upgrade. Likewise, the upgrade causes an increase in the hot-stripe heat flux in plates 1-4 (4.2% in plate 1) and plates 22 and 23 (3.4% in plate 23) in the BOL element in core position 5, which is closest to the 2017-RBM-99. This is because the radial power profile shifts towards the fueled cartridges of the 2017-RBM-99 when it is installed. This is the reason for the increase of the hot-stripe heat flux in plates 22 and 23 of the BOL element in core position 5 relative to the PSAR analysis. It is also noted that with the upgrade the hot-stripe heat flux in plates 1-4 is increased relative to the PSAR analysis for the EOL element in position 4 and the MOL element in position 6, while the hot-stripe heat flux in fuel plates 22 and 23 of these elements are less with the upgrade relative to the PSAR analysis. This is because of differences in the critical control blade position for the cores without and with the upgrade.

As discussed in Section 2.2 above, the upgrade increases the core excess reactivity for the mixed burnup LEU core by ~1200 pcm. The critical control blade position in Core 8A evaluated for the PSAR was 24.3 inches withdrawn. With the upgrade, the critical control blade position for the comparable Core 8C is 20.9 inches withdrawn. To compensate for the increase in the core excess reactivity, the control blades must be positioned 3.4 inches lower with the upgrade, which pushes the power distribution away from the plates that are closest to the control blades and towards the fuel plates that are near the center test hole.

Figure 2.9 displays the maximum local heat flux for all plates in reference cores with (7C) and without (7A) the 2017-RBM-99 upgrade. Under xenon-free conditions at BOC, the critical control blade position in Core 7C is about two inches above the core midplane, which is 1.6 inches lower than the critical control blade position in Core 7A evaluated for the PSAR. While this causes some radial power shift away from plate 23 and towards the plates near the center test hole, it also causes a larger axial power peak near the bottom of the core. As a result, the maximum local heat flux in this fuel plate is decreased by only 2.2% with the 2017-RBM-99 upgrade relative to the PSAR analysis. The maximum local heat flux in all fuel plates in the elements in core positions 4, 5, and 6 is increased by between 2.4% and 6.4% as a result of the 2017-RBM-99 upgrade. This is because of shifts in the radial and axial power profiles as a result of the device installation and the deeper critical control blade position. The location of the maximum local heat flux in a BOL element in the reference LEU cores moves from element position 1 to 5 due to the upgrade. Likewise, the location of the maximum local heat flux in an EOL element in the reference LEU cores moves from element position 8 to 4 due to the upgrade.

It is also important to note the fuel plate burnup, particularly for the LEU fuel element, because of the impact of burnup on monolithic fuel thermo-physical properties and performance. The thermal

conductivity and fuel temperature safety limit of the U-10Mo monolithic fuel are dependent on fuel burnup, both of which decrease monotonically with increasing burnup. It is important to include these phenomena in the modeling of the transients presented in Section 4, because they will affect the maximum fuel temperature and the margin to the safety limit.

Table 2.10 provides the burnup for each fuel plate in the LEU fuel element discharged from the pseudo-equilibrium fuel cycle evaluated with the 2017-RBM-99 upgrade. The average burnup and axial peaking factor for each plate was derived from the results of the REBUS-DIF3D fuel cycle simulation. In that analysis, the azimuthal power peaking in the fuel plates was neglected. The azimuthal peaking factor provided in Table 2.10 was derived from the detailed, three-dimensional power distributions calculated by MCNP5. The maximum local burnup in the discharged element with the 2017-RBM-99 upgrade is 3.32×10^{21} fissions/cm³ and occurs in fuel plate 1. For the LEU element discharged from the MURR core with the experimental configuration and experimental loadings that were typical for MURR in 2008, and which was modeled in the PSAR analysis, the maximum local burnup in plate 1 is 3.37×10^{21} fissions/cm³. For fuel plate 23, the maximum local burnup in the discharged element is 2.52×10^{21} and 2.59×10^{21} fissions/cm³ for the case with the 2017-RBM-99 upgrade and the PSAR analysis, respectively.

Table 2.10. Burnup by Plate for Discharged LEU Element with the 2017-RBM-99 Upgrade.

Plate	Plate Average Burnup (fissions/cm ³)	Axial Peaking Factor	Azimuthal Peaking Factor	Maximum Local Burnup (fissions/cm ³)
1	2.67E+21	1.198	1.035	3.32E+21
2	2.08E+21	1.218	1.085	2.74E+21
3	1.58E+21	1.227	1.145	2.22E+21
4	1.24E+21	1.235	1.198	1.83E+21
5	1.05E+21	1.238	1.233	1.60E+21
6	9.44E+20	1.238	1.250	1.46E+21
7	8.66E+20	1.239	1.255	1.35E+21
8	8.15E+20	1.240	1.255	1.27E+21
9	7.79E+20	1.241	1.249	1.21E+21
10	7.53E+20	1.242	1.245	1.16E+21
11	7.35E+20	1.243	1.242	1.13E+21
12	7.22E+20	1.244	1.241	1.11E+21
13	7.24E+20	1.246	1.245	1.12E+21
14	7.18E+20	1.248	1.250	1.12E+21
15	7.17E+20	1.251	1.258	1.13E+21
16	7.21E+20	1.254	1.270	1.15E+21
17	7.32E+20	1.258	1.282	1.18E+21
18	7.56E+20	1.263	1.293	1.24E+21
19	8.08E+20	1.269	1.297	1.33E+21
20	8.85E+20	1.276	1.289	1.46E+21
21	1.03E+21	1.282	1.253	1.66E+21
22	1.31E+21	1.287	1.189	2.01E+21
23	1.78E+21	1.279	1.106	2.52E+21

Lastly, it should be noted that a fresh beryllium reflector was assumed for the HEU and LEU neutronics analysis, both in the PSAR analysis and with the 2017-RBM-99 upgrade. The MURR beryllium reflector is currently replaced on an 8-year cycle (about 26,000 MWd of HEU core burnup). The loss of ^9Be atoms and poisoning from ^6Li produced from fast neutron capture causes a decrease in the core excess reactivity, which has been measured for HEU to be about 500 pcm when the reflector is near the end of its lifetime. The scheduled replacement of the beryllium reflector also precludes beryllium cracking, which has occurred once in MURR's history. Since implementing the 8-year replacement schedule, no failures have occurred. The beryllium lifetime with LEU is being investigated under a separate activity.

As a result of the decrease of the effectiveness of the irradiated beryllium as a reflector material, the critical control blade positions for typical cores with an aged beryllium reflector are more fully withdrawn relative to a core with a fresh reflector. For a reflector with the equivalent of 26,000 MWd of burnup, the control blades are about 1 inch more fully withdrawn at the beginning of cycle and about 2 inches more fully withdrawn at the end of cycle. For HEU, which is limiting near the center test hole (coolant channel 2 for margin to flow instability in steady-state operations and fuel plate 2 for the fuel temperature in accidents), the assumption of a fresh beryllium reflector for the analysis of safety margins is likely conservative because the lower critical control blade position that occurs with a fresh reflector increases the power peaking toward fuel plates 1 and 2. For LEU, which tends to be limited in the outermost coolant channels that are close to the control blades, the predicted margin to flow instability may be smaller in the aged beryllium case, because the more fully withdrawn control blades will increase the total power in the outer plates. However, the local power peaking in the outermost plates will be less when the control blades are more fully withdrawn in the aged beryllium case. Therefore, it is expected that LEU fuel temperatures reached during accidents will be lower in the case of aged beryllium. The effect of the assumption of fresh beryllium on the HEU and LEU core power distributions and safety margins are discussed in more detail in Appendix C.

2.4 Predicted Experimental Performance in LEU Cores Relative to HEU Cores

Conversion from HEU to LEU fuel will affect the experimental performance of MURR. The effect of conversion to LEU with the 2017-RBM-99 upgrade was examined by calculating neutron flux and/or reaction rate predictions in a number of important experimental locations for reference HEU and LEU cores with fuel elements at a mixture of burnups and at different critical states during the operating cycle. The current typical experiment configuration and experimental loading for MURR, but with the 2017-RBM-99 upgrade in place of the fast-flux irradiation facilities, was modeled. Previous work for a typical HEU core case at 10 MW without the 2017-RBM-99 upgrade found that the one-group (n, γ) capture reaction rates for various isotopes (i.e., thermal and resonance monitors) and fluxes calculated by an MCNP5 model were within 5% of measurements by Peters et al. [31].

For the PSAR analysis, a total of 70 performance measures were evaluated by tallies in the MCNP5 models for the HEU and LEU reference cores at BOC and equilibrium xenon conditions. The comparison of HEU and LEU performance for these reference cores found that in order to maintain the HEU performance at 10 MW for all performance measures, a minimum LEU power of 11.56 MW was necessary. A power uprate to 12 MW for LEU was assumed for safety analyses in the PSAR.

For the current work with the 2017-RBM-99 upgrade, 432 performance measures were evaluated for the HEU and LEU reference cores at BOC, equilibrium xenon, and EOC conditions. The additional performance measures were included to evaluate the experimental performance in locations that

cover an extended axial range relative to what was evaluated for the PSAR. A comparison of the LEU experimental performance at 12 MW relative to HEU performance at 10 MW is provided in Table 2.11. The results show that the HEU core experimental performance at 10 MW is met or exceeded with the LEU core at 12 MW for all performance measures except for the beam tube E net outbound current at BOC. The neutron current was calculated for a 47-group neutron spectrum that was defined by a key experimentalist at MURR. Upon closer inspection, it was found that the HEU performance is met for the neutron current in beam tube E at all energies with the exception of neutrons with $13.2 \leq E_n < 15.8$ MeV. The neutrons in this energy range were calculated to account for less than 0.002% of the total current in beam tube E. Consequently, the degradation of the experimental performance in beam tube E for LEU at 12 MW relative to HEU at 10 MW is negligible. Neglecting this one performance measure, it was found that with the upgrade for both HEU and LEU cores, a minimum LEU core power of 11.8 MW will meet all HEU performance measures at 10 MW.

Table 2.11. LEU Performance at 12 MW Relative to HEU Performance at 10 MW with the 2017-RBM-99 Upgrade.

Performance Measure		BOC			Eq. Xe			EOC		
Location	Number of Tallies	Min	Max	Avg.	Min	Max	Avg.	Min	Max	Avg.
2017-RBM-99 (Power)	1	105%	105%	105%	105%	105%	105%	105%	105%	105%
Center Test Hole	76	102%	114%	109%	103%	113%	108%	103%	113%	108%
Rabbit Irradiation	6	104%	111%	108%	105%	111%	108%	105%	111%	108%
Si Irradiation in Combined GHIJ Wedge	4	106%	110%	107%	106%	110%	108%	107%	110%	108%
"G1" Flux Wire in Graphite Reflector	99	101%	113%	106%	102%	113%	107%	102%	116%	107%
"H1" Flux Wire in Graphite Reflector	99	101%	115%	106%	102%	112%	107%	102%	112%	106%
"N1" Flux Wire in Graphite Reflector	99	102%	113%	106%	103%	112%	107%	102%	112%	107%
Beam Tube E Net Outbound Current	48	99%	124%	112%	102%	117%	111%	103%	125%	112%

2.5 Shutdown Margin and Core Physics Parameters

The MURR Technical Specifications require a minimum shutdown margin of 2% $\Delta k/k$. The shutdown margins for the HEU and LEU cores with the 2017-RBM-99 upgrade were not calculated in this work. The minimum cold shutdown margins for the all fresh HEU and LEU cores for a case with all four control blades at their maximum burnup were calculated in the PSAR analysis to be 3.4% and 3.1% $\Delta k/k$, respectively. In Section 2.2 it was found that with the 2017-RBM-99 upgrade, the excess reactivity for the all fresh HEU and LEU cores is 1.18% and 1.06% $\Delta k/k$ greater, respectively, relative

to the core configuration considered in the PSAR analysis. Thus, the cold minimum shutdown margin for the all fresh HEU and LEU cores with the 2017-RBM-99 upgrade is estimated to be 2.22% and 2.04% $\Delta k/k$, respectively, which exceeds the MURR Technical Specifications requirement. Note that this calculation is for an extreme case with all fresh fuel and with all four control blades at their maximum burnup. MURR never operates with all four control blades at their maximum burnup.

For the reference mixed burnup cores, the cold minimum shutdown margin with all four control blades at their maximum burnup was calculated in the PSAR analysis to be 8.5% and 6.9% $\Delta k/k$ for the HEU and LEU cores at no xenon conditions, respectively. With the 2017-RBM-99 upgrade, the core excess reactivity for the reference mixed burnup cores at BOC is 1.33% and 1.13% $\Delta k/k$ greater for the HEU and LEU core, respectively. Thus, the cold minimum shutdown margin for the reference mixed burnup HEU and LEU cores is estimated to be greater than 7.2% and 5.8% $\Delta k/k$, respectively.

The kinetics parameters and reactivity coefficients for the HEU and LEU cores with the 2017-RBM-99 upgrade are needed for the reactivity insertion accident analysis that will be performed in Section 4.2. These core physics parameters were calculated for all fresh (no xenon) and mixed burnup (no xenon and equilibrium xenon) HEU and LEU cores. The kinetics parameters were calculated using the kinetics option in MCNP5. The reactivity coefficients were calculated with MCNP5 by perturbing the coolant density, coolant temperature, or fuel temperature by a small amount relative to a base case at nominal conditions. The smallest value calculated for all of the fuel burnup and xenon conditions evaluated for each fuel type was selected as a conservative value for modeling the reactivity insertion accidents. The values for these core physics parameters with the 2017-RBM-99 upgrade are summarized in Table 2.12. The calculated β_{eff} and reactivity feedback coefficients for the HEU and LEU cores with the 2017-RBM-99 upgrade are within the statistical uncertainty ($< 1-\sigma$) of their corresponding values calculated without 2017-RBM-99 upgrade for the PSAR analysis reported in Reference 14.

Table 2.12. Kinetics Parameters and Reactivity Feedback Coefficients for MURR Core with the 2017-RBM-99 Upgrade.

Parameter	HEU	LEU
β_{eff}	$0.781\% \pm 0.003\%$	$0.762\% \pm 0.002\%$
Λ (μs)	50.0 ± 0.04	38.9 ± 0.03
Coolant Density Coefficient, $\$/\% \text{-void}$	-0.310 ± 0.011	-0.299 ± 0.009
Coolant Temperature Coefficient, ^a $\$/^\circ\text{C}$	$-5.15 \times 10^{-3} \pm 9.4 \times 10^{-5}$	$-1.50 \times 10^{-3} \pm 9.5 \times 10^{-5}$
Fuel Temperature Coefficient $\$/^\circ\text{C}$	N/A	$-0.867 \times 10^{-3} \pm 5.2 \times 10^{-5}$

^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.

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 transient scenarios evaluated in Section 4 include additional conservatism by assuming the scram initiates from the control blades at the fully withdrawn position. The control bank reactivity worth was calculated for the LEU Core 7C with the 2017-RBM-99 upgrade for control bank positions from 16 to 26 inches withdrawn. Table 2.13 provides a comparison of the bank worth for the reference LEU core with the upgrade and as modeled for the PSAR. The control bank worth calculated for each core is within the statistical uncertainty of the MCNP5 calculation, and so, is not

affected by the upgrade. Consequently, the scram shutdown reactivity worth curves calculated for the HEU and LEU cores from the PSAR analysis were used in the reactivity insertion and loss-of-coolant and loss-of-flow transient accident analyses in this work.

Table 2.13. Control Bank Work ($\Delta k/k/\text{inch}$) in LEU Mixed-Burnup Cores at BOC.

Control Bank Position (inches withdrawn)	With 2017-RBM-99 Upgrade (Core 7C)	Without 2017-RBM-99 Upgrade (Core 7A)
26	0.0018	0.0017
25	0.0023	0.0026
24	0.0031	0.0031
23	0.0037	0.0038
22	0.0043	0.0039
21	0.0050	0.0052
20	0.0055	0.0049
19	0.0062	0.0064
18	0.0068	0.0068
17	0.0076	0.0075
16	0.0083	0.0082

3 STEADY-STATE THERMAL HYDRAULICS ANALYSIS

3.1 Introduction

Steady-state thermal-hydraulics were performed with the PLTEMP/ANL code, version 4.3 [32]. All coolant channels in the eight elements in the MURR core are modeled in a one-dimensional analysis. Each coolant channel is modeled by three azimuthal stripes, two unheated stripes representing the unfueled portion of the fuel plates and associated coolant adjacent to the element side plates and a single heated stripe that represents the fueled portion of the plate and associated coolant channel. As discussed in Section 2.3, lateral heat conduction from the fueled to the unfueled portion of the fuel plate is neglected in the analysis. To be conservative, the axial power density profile derived from the MCNP5 analysis for an appropriately-sized azimuthal stripe that represents the portion of the fuel plate with the maximum power density was utilized in the PLTEMP/ANL analysis.

The core neutron physics analysis subdivided the fuel in each plate into nine vertical stripes. To simplify the thermal-hydraulics analysis, the PLTEMP/ANL model for the MURR core used multipliers on the local hot channel factors to represent the most severe thermal conditions of the limiting stripe for each fuel plate. This is explained in more detail in the second and third paragraphs of Section 13 of Reference 8.

When steady-state thermal-hydraulics safety calculations are performed, hot-channel factors are applied to account for factors such as statistical variations in manufacturing of the fuel element and the uncertainty of the method used for the neutronics calculation, which would affect the calculated heat flux. This methodology for the MURR analysis is described in detail in Section 8 of Reference 8. The application of hot-channel factors in the analysis ensures that the calculated minimum thermal-hydraulic safety margins are maintained under even the most extreme conditions.

The end channel thicknesses and their tolerances for both the HEU and the LEU CD35 elements that were assumed in the PSAR analysis are defined in Figure 2 and Table 2 of Reference 8. After References 8 and 14 were completed, the tolerances on the end channel dimensions, indicated as “C1” and “C-L” in Figure 2 of Reference 8, were changed from the LEU CD35 element, which became the preliminary LEU fuel design. This tolerance is shown in Section A-A of Sheet 3 of Reference 25. The tolerance was increased based on the recommendation of the fuel manufacturer and is preferred, because it facilitates the element manufacturing and will lead to fewer rejected elements. The larger end channel tolerances were assumed in the PLTEMP/ANL analysis for the LEU core states in this work. The same range for the end channel thickness was also assumed in the analysis of the HEU cores states, and is consistent with the current manufacture of HEU elements. A more detailed discussion and a comparison of the results for cores without the 2017 RBM-99 upgrade with the old (Reference 8 results) and new end channel tolerances is provided in Appendix A.

The impact of the coolant channel restriction that may occur with depletion of the fuel under normal operating conditions is also incorporated into the analysis. The reduction in channel thickness due to fuel plate swelling, material creep, and oxide buildup on the surface of the fuel plate reduces the flow through the channel. This adversely affects the bulk coolant temperature rise and the film temperature rise at the surface of the fuel plate. Since the total core flow rate is controlled, the flow that is diverted from one channel must go to another channel. A bounding approach to the flow reduction in one channel could cause an overly optimistic assessment of the flow in the other channels. This problem is avoided by treating the channel thickness reduction as if it were the result of a local manufacturing tolerance. The narrowing of the coolant channel due to fuel plate swelling

and oxide buildup is represented in the modeling as an additional reduction in the channel thickness and is included with the reduction in channel thickness due to manufacturing tolerances.

The coolant channel thicknesses in selected HEU elements in MURR are monitored at the end of the element lifetime with a 62-mil (0.062 inch) welding rod that is used as a gauge to ensure there is an adequate coolant channel gap. This thickness is 10 mil less than the minimum as-built HEU internal channel thickness based on manufacturing tolerances. In the history of MURR, no HEU element has ever been removed from service as a result of failing to meet this criterion. Based on this criterion, the maximum reduction of the HEU coolant channel gap due to burnup-related effects for channels bounded by two fuel plates is assumed to be 10 mil for the steady-state thermal hydraulic analysis.

For the modeling of the LEU elements at maximum burnup this value was assumed to be 8 mil to bound expected fuel swelling and creep for U-10Mo monolithic fuel plates and oxide growth on LEU fuel plates. This total reduction of the coolant channel between two LEU fuel plates is based on an assumption of a 4 mil maximum plate thickness increase from fuel swelling and creep and a maximum of 2 mil of oxide on each surface of the fuel plates. The plates on each side of the coolant channel are assumed to increase in thickness uniformly. Fuel swelling and creep analysis for LEU fuel plates at maximum burnup and prototypic conditions confirmed that the assumption of a 4 mil plate thickness increase is conservative [33]. Calculations of the oxide layer thickness on LEU fuel plates at EOL based on an empirical relationship developed from measured data for HEU fuel plates [34] found the oxide thickness for a maximum burnup element to be < 1 mil (see Section 4.1.3 and Table 4.2).

The channel thickness reduction is assumed to be zero for BOL fuel elements and to increase linearly proportional to the burnup of the elements in the core, which are given in Table 2.5. For the elements in core positions 4 and 8, which are at EOL, the element burnup used for the channel gap restriction is assumed to be at the maximum, 150 MWd for HEU and 180 MWd for LEU. Lastly, for both HEU and LEU elements, the reduction of the end channel thickness is taken to be half of the internal channel values, i.e., 5 mil for the HEU and 4 mil for LEU.

The steady-state safety margins were calculated at the extreme condition of the LSSS settings for MURR. These conditions are summarized in Table 3.1. The power distribution results presented in Section 2.3 were prepared from MCNP5 results using a post-processor which normalized the results by assuming that all heat is deposited in the fuel and is conducted evenly out the two radial surfaces of the fuel plate. These are both conservative modeling assumptions because 1) neutron and photon transport deposit a portion of the total energy release from fission in materials outside of the fuel and 2) conduction of the heat deposited in the fuel will be greater towards the side of the fuel plate with the lower surface temperature. In order to eliminate over-conservatism of the analysis where defensible, the power density used in the PLTEMP/ANL analysis was normalized to the total energy deposited in the primary coolant system. Previous analysis found that the fraction of the reactor power deposited outside the primary system is 6.0% for HEU and 3.6% for LEU. Thus, the power density data used in the PLTEMP/ANL models were normalized to a total core power of 94.0% and 96.4% of the total reactor power for HEU and LEU, respectively. Additionally, PLTEMP/ANL accounts for the uneven heat conduction from the two radial surfaces of the fuel plate that is caused by the asymmetric coolant channel conditions on each side of a fuel plate.

Lastly, the analysis in Reference 8 was completed using beta versions of the PLTEMP/ANL v.4.2 software [20]. The PLTEMP/ANL software, version 4.3 was used for the present analysis, which includes repeating all safety-basis cases modeled in Reference 8. The safety-basis results presented in this Section replace those results presented in Reference 8. There is also a set of operational limit curves for LEU cores without the 2017-RBM-99 reported in References 8 and 14. These curves are

not considered part of the MURR safety basis, and so these calculations were not repeated with PLTEMP/ANL v. 4.3 as a part of this work. However, the calculations for the operational limit curves will be performed with a qualified version of the PLTEMP/ANL software as a part of the Final Safety Analysis Report for MURR conversion. Finally, it is noted that the operational limit curves for HEU cores are not based on PLTEMP/ANL analysis, and so are not affected by software changes in PLTEMP/ANL.

Table 3.1. LSSS Conditions Assumed for Steady-State Thermal-Hydraulic Analysis.

	HEU	LEU
LSSS Reactor Power [MW]	12.5	15.0
Fraction of Total Reactor Power Deposited in Primary System	94.0%	96.4%
Core Coolant Inlet Temperature [°F]	155	145
Primary Coolant Flowrate [gpm]	3200	3300
Pressurizer Pressure [psig]	60.7	60.7
Pressurizer Level [in]	-16	-16

3.2 Results

With the 2017-RBM-99 device installed, 40 core states (5 burnup conditions $\times$ 2 center test hole loadings $\times$ 4 control blade conditions for each HEU and LEU), which are described in Section 2.3, were considered for each fuel type. The thermal-hydraulic effect of this device is that it affects the power distribution within each core state.

The flow instability powers with the new end channel tolerance changes described in Section 3.1 and in Appendix A for the 24 reference HEU core states in Table 16 of Reference 8 without the 2017-RBM-99 were calculated. These results are presented along with those for the 40 HEU core states with the 2017-RBM-99 upgrade in Table 3.2. Similarly, the flow instability powers for the 24 reference LEU CD35 core states without the 2017-RBM-99 upgrade in Table 18 of Reference 8 with the new end channel tolerances were calculated. These results are presented along with the 40 reference LEU core states with the 2017-RBM-99 upgrade in Table 3.3.

In Table 3.2 and Table 3.3, the flow instability power in red indicates the lowest power within the group, where all of the core states without 2017-RBM-99 form one group and all of the core states with the upgrade form the other group. The yellow background indicates the lowest power for both groups taken together. It is helpful in keeping track of which cases include the 2017-RBM-99 device and which do not. Those designated with “A” or “B” in the case column of the results tables, as in “3A1”, for example, do not include the device, and those designated with “C” or “D” includes the device.

Core state 3D3 has the lowest HEU flow instability power, 14.56 MW, resulting in a margin of 2.06 MW above the HEU LSSS power of 12.5 MW. Core state 8D1 has the lowest LEU flow instability power, 17.35 MW, resulting in a margin of 2.35 MW above the LEU LSSS power of 15.0 MW.

Table 3.2. Flow Instability Results for HEU Core.

Without 2017-RBM-99 Device								With 2017-RBM-99 Device							
Loaded Flux Trap				Empty Flux Trap				Loaded Flux Trap				Empty Flux Trap			
Core	Reactor Power, MW	Element	Channel	Core	Reactor Power, MW	Element	Channel	Core	Reactor Power, MW	Element	Channel	Core	Reactor Power, MW	Element	Channel
1A	16.47	2	2	1B	15.76	5	2	1C	15.84	5	2	1D	15.51	5	2
1A1	15.76	7	2	1B1	15.12	7	2	1C1	15.52	7	2	1D1	15.04	7	2
1A2	15.92	1	2	1B2	15.30	1	2	1C2	15.70	2	2	1D2	15.38	1	2
								1C3	15.25	5	2	1D3	14.86	5	2
2A	17.12	5	2	2B	16.47	5	2	2C	16.55	5	2	2D	16.18	5	2
2A1	16.58	7	2	2B1	15.96	7	2	2C1	16.30	8	2	2D1	15.84	6	2
2A2	16.75	1	2	2B2	16.10	1	2	2C2	16.42	2	2	2D2	16.19	8	2
								2C3	16.03	5	2	2D3	15.70	5	2
3A	15.91	5	2	3B	15.33	5	2	3C	15.27	5	2	3D	15.01	5	2
3A1	15.89	1	2	3B1	15.13	1	2	3C1	15.32	5	2	3D1	14.98	5	2
3A2	15.56	1	2	3B2	14.89	1	2	3C2	15.28	1	2	3D2	14.91	1	2
								3C3	14.82	5	2	3D3	14.56	5	2
4A	16.72	5	2	4B	16.13	1	2	4C	16.06	5	2	4D	15.79	5	2
4A1	16.72	1	2	4B1	16.06	1	2	4C1	16.14	5	2	4D1	15.85	5	2
4A2	16.64	1	2	4B2	16.05	1	2	4C2	16.14	1	2	4D2	15.92	1	2
								4C3	15.76	5	2	4D3	15.51	5	2
								0C	16.24	5	2	0D	15.99	5	2
								0C1	16.35	5	2	0D1	16.06	5	2
								0C2	16.39	1	2	0D2	16.14	1	2
								0C3	16.00	5	2	0D3	15.78	5	2
Minimum Margin Compared to the 12.5 MW LSSS Power															
Ratio		14.89/12.5 = 1.191						14.56/12.5 = 1.164							
MW		14.89 – 12.5 = 2.39						14.56 – 12.5 = 2.06							

Table 3.3. Flow Instability Results for Preliminary LEU Core.

Without 2017-RBM-99 Device								With 2017-RBM-99 Device							
Loaded Flux Trap				Empty Flux Trap				Loaded Flux Trap				Empty Flux Trap			
Core	Reactor Power, MW	Element	Channel	Core	Reactor Power, MW	Element	Channel	Core	Reactor Power, MW	Element	Channel	Core	Reactor Power, MW	Element	Channel
5A	20.00	5	4	5B	19.54	4	3	5C	19.38	5	4	5D	19.17	5	4
5A1	19.26	8	4	5B1	18.66	7	4	5C1	18.94	7	4	5D1	18.57	7	4
5A2	19.21	1	4	5B2	18.91	2	4	5C2	19.15	1	4	5D2	18.99	1	4
								5C3	18.56	5	4	5D3	18.34	5	4
6A	20.96	2	4	6B	20.36	5	3	6C	20.30	5	4	6D	20.12	6	4
6A1	19.14	6	23	6B1	19.07	6	23	6C1	19.65	6	23	6D1	19.38	6	23
6A2	19.39	8	23	6B2	19.29	8	23	6C2	20.18	1	4	6D2	20.07	1	4
								6C3	19.55	5	23	6D3	19.45	5	4
7A	19.29	8	4	7B	18.87	4	4	7C	18.74	4	4	7D	18.57	4	4
7A1	18.50	6	23	7B1	18.32	8	4	7C1	18.40	8	4	7D1	18.25	8	4
7A2	18.47	1	23	7B2	18.25	8	4	7C2	18.62	8	4	7D2	18.40	8	4
								7C3	18.16	4	4	7D3	17.99	4	4
8A	18.53	8	23	8B	18.21	8	23	8C	19.28	6	23	8D	18.93	6	23
8A1	17.70	7	23	8B1	17.86	6	23	8C1	17.55	6	23	8D1	17.35	6	23
8A2	17.73	1	23	8B2	17.89	8	23	8C2	18.51	8	23	8D2	18.39	8	23
								8C3	17.96	6	23	8D3	17.76	6	23
								9C	19.20	6	23	9D	18.73	6	23
								9C1	17.55	6	23	9D1	17.36	6	23
								9C2	18.52	8	23	9D2	18.39	8	23
								9C3	17.96	6	23	9D3	17.69	6	23
Minimum Margin Compared to the 15.0 MW LSSS Power															
Ratio		17.70/15.0 = 1.180						17.35/15.0 = 1.157							
MW		17.70 – 15.0 = 2.70						17.35 – 15.0 = 2.35							

For all 64 HEU core states in Table 3.2 (24 core states without the 2017-RBM-99 upgrade and 40 cores states with the upgrade), the critical heat flux ratio (CHFR) is greater than 2.0 at the flow instability power. For 38 of the 64 LEU core states presented in Table 3.3, the CHFR is greater than 2.0 at the flow instability power. The 26 cores states with CHFR < 2.0 at the flow instability power are shown in Table 3.4 with orange background. All 64 cores in Table 3.3 were evaluated to find the reactor powers for CHFR = 2.0. Table 3.4 compares the power levels at OFI (labeled as OFIR = 1.0) and the power level for CHFR=2.0 for the LEU core states. Even though some cores reach CHFR = 2.0 at a lower reactor power level than OFI (cores states indicated with an orange background), the lowest power at which OFI occurs for all the LEU cores evaluated is 17.35 MW (Core 8D1 with the upgrade), which is less than the lowest power at which CHFR=2.0 (17.77 MW for Core 5C3 with the upgrade). Likewise, considering only the cases without the upgrade, the minimum OFI power of 17.70 MW (for Core 8A1) is less than the lowest power at which CHFR=2.0 (18.15 MW for Core 5B1).

Table 3.4. Reactor Power for OFIR = 1.0 and CHFR = 2.0 for Preliminary LEU Core.

Without 2017-RBM-99 Device						With 2017-RBM-99 Device					
Loaded Flux Trap			Empty Flux Trap			Loaded Flux Trap			Empty Flux Trap		
Core	OFIR = 1.0 Power, MW	CHFR = 2.0 Power, MW	Core	OFIR = 1.0 Power, MW	CHFR = 2.0 Power, MW	Core	OFIR = 1.0 Power, MW	CHFR = 2.0 Power, MW	Core	OFIR = 1.0 Power, MW	CHFR = 2.0 Power, MW
5A	20.00	19.53	5B	19.54	19.59	5C	19.38	18.88	5D	19.17	18.92
5A1	19.26	18.18	5B1	18.66	18.15	5C1	18.94	18.23	5D1	18.57	18.00
5A2	19.21	18.35	5B2	18.91	18.32	5C2	19.15	18.92	5D2	18.99	18.96
						5C3	18.56	17.77	5D3	18.34	17.80
6A	20.96	20.05	6B	20.36	20.32	6C	20.30	19.35	6D	20.12	19.52
6A1	19.14	18.93	6B1	19.07	19.15	6C1	19.65	18.68	6D1	19.38	18.68
6A2	19.39	19.29	6B2	19.29	19.63	6C2	20.18	19.68	6D2	20.07	19.74
						6C3	19.55	18.66	6D3	19.45	18.94
7A	19.29	19.95	7B	18.87	19.88	7C	18.74	19.90	7D	18.57	19.87
7A1	18.50	18.30	7B1	18.32	18.43	7C1	18.40	18.57	7D1	18.25	18.73
7A2	18.47	18.75	7B2	18.25	18.88	7C2	18.62	19.09	7D2	18.40	19.33
						7C3	18.16	18.62	7D3	17.99	18.75
8A	18.53	20.83	8B	18.21	20.86	8C	19.28	20.45	8D	18.93	20.75
8A1	17.70	19.99	8B1	17.86	20.88	8C1	17.55	19.57	8D1	17.35	19.94
8A2	17.73	20.06	8B2	17.89	20.68	8C2	18.51	20.30	8D2	18.39	20.61
						8C3	17.96	19.69	8D3	17.76	20.00
						9C	19.20	20.73	9D	18.73	21.11
						9C1	17.55	19.85	9D1	17.36	20.32
						9C2	18.52	20.68	9D2	18.39	20.88
						9C3	17.96	20.06	9D3	17.69	20.43
Minimum Margin Compared to the 15.0 MW LSSS Power											
Criteria		OFIR = 1.0		CHFR = 2.0		OFIR = 1.0		CHFR = 2.0			
Ratio		17.70/15.0 = 1.180		18.15/15.0 = 1.210		17.35/15.0 = 1.157		17.77/15.0 = 1.185			
MW		17.70 – 15.0 = 2.70		18.15 - 15.0 = 3.15		17.35 – 15.0 = 2.35		17.77 – 15.0 = 2.77			

The PLTEMP/ANL code was also used to determine the minimum critical heat flux ratio at the LSSS power for both HEU and LEU at each of the 64 core states. The LSSS power is 12.5 MW for the HEU core and 15.0 MW for the LEU core. Since 6.0% of the reactor power is deposited in the reflector when the HEU fuel is used and 3.6% when the LEU fuel is used, 94.0% of 12.5 MW (11.75 MW) was assumed in the CHFR analysis of the HEU core and 96.4% of 15.0 MW (14.46 MW) was assumed in the CHFR analysis of the LEU core.

Table 3.5 and Table 3.6 provide the minimum CHFRs for the HEU and LEU cores, respectively. The value in red indicates the lowest CHFR in the group. All of the core states without the 2017-RBM-99 device form one group and all of the core state with the device form the other. The yellow background indicates the lowest CHFR for both groups taken together. The CHFR is greater than 2.0 for all HEU and LEU cases. For HEU cores, the minimum CHFR always occurs on the concave side of plate 1. For LEU cores, the minimum CHFR always occurs on the convex side of plate 23, which is the outermost plate of the fuel element adjacent to the beryllium reflector and the control blades.

Table 3.5. Minimum CHFR for HEU Core at LSSS Power.

Without 2018 RBM-99 Device								With 2018 RBM-99 Device							
Loaded Flux Trap				Empty Flux Trap				Loaded Flux Trap				Empty Flux Trap			
Core	CHFR	Element	Axial ¹	Core	CHFR	Element	Axial ¹	Core	CHFR	Element	Axial ¹	Core	CHFR	Element	Axial ¹
1A	2.97	2	18	1B	2.79	2	17	1C	2.83	5	18	1D	2.76	5	18
1A1	2.93	7	17	1B1	2.73	7	17	1C1	2.82	7	18	1D1	2.74	7	18
1A2	2.96	2	16	1B2	2.77	1	17	1C2	2.86	1	18	1D2	2.80	2	18
								1C3	2.79	4	18	1D3	2.70	5	18
2A	3.19	5	15	2B	3.04	5	17	2C	3.02	4	17	2D	2.97	4	17
2A1	3.14	7	15	2B1	3.03	6	16	2C1	3.04	8	17	2D1	2.97	6	17
2A2	3.19	7	15	2B2	3.04	2	16	2C2	3.06	1	17	2D2	3.03	8	17
								2C3	2.99	4	17	2D3	2.94	6	17
3A	2.97	5	15	3B	2.87	5	16	3C	2.80	5	17	3D	2.78	5	16
3A1	3.03	5	15	3B1	2.90	1	16	3C1	2.88	5	17	3D1	2.84	5	16
3A2	3.01	1	15	3B2	2.84	1	16	3C2	2.87	1	17	3D2	2.82	1	16
								3C3	2.80	5	17	3D3	2.76	5	16
4A	3.27	5	14	4B	3.29	1	14	4C	3.17	5	15	4D	3.14	5	14
4A1	3.35	5	14	4B1	3.28	1	14	4C1	3.21	5	15	4D1	3.18	5	15
4A2	3.32	5	14	4B2	3.29	1	14	4C2	3.23	5	16	4D2	3.18	5	15
								4C3	3.17	5	15	4D3	3.14	5	15
								0C	3.26	5	15	0D	3.24	5	15
								0C1	3.32	5	15	0D1	3.27	5	14
								0C2	3.31	5	15	0D2	3.26	5	14
								0C3	3.25	5	15	4D3	3.22	5	15

¹ Axial node (out of 24) from top of fuel plate.

Table 3.6. Minimum CHFR for Preliminary LEU Core at the LSSS Power.

Without 2018 RBM-99 Device								With 2018 RBM-99 Device							
Loaded Flux Trap				Empty Flux Trap				Loaded Flux Trap				Empty Flux Trap			
Core	CHFR	Element	Axial ¹	Core	CHFR	Element	Axial ¹	Core	CHFR	Element	Axial ¹	Core	CHFR	Element	Axial ¹
5A	2.67	8	18	5B	2.67	6	18	5C	2.57	6	19	5D	2.57	6	18
5A1	2.48	6	18	5B1	2.47	6	17	5C1	2.48	6	18	5D1	2.45	6	18
5A2	2.50	8	18	5B2	2.50	8	18	5C2	2.58	8	18	5D2	2.59	8	18
								5C3	2.41	6	18	5D3	2.42	6	18
6A	2.75	6	16	6B	2.79	6	16	6C	2.65	6	17	6D	2.67	6	17
6A1	2.59	6	16	6B1	2.65	6	16	6C1	2.56	6	16	6D1	2.56	6	16
6A2	2.67	8	17	6B2	2.73	7	16	6C2	2.71	7	17	6D2	2.71	7	17
								6C3	2.55	6	17	6D3	2.60	6	16
7A	2.75	1	17	7B	2.75	1	17	7C	2.76	5	18	7D	2.77	5	18
7A1	2.51	1	16	7B1	2.53	1	17	7C1	2.57	5	18	7D1	2.60	1	17
7A2	2.58	1	16	7B2	2.63	1	16	7C2	2.63	1	17	7D2	2.68	1	17
								7C3	2.59	5	18	7D3	2.62	5	18
8A	2.94	1	14	8B	2.99	1	15	8C	2.87	5	16	8D	2.91	5	15
8A1	2.83	1	14	8B1	2.97	1	13	8C1	2.74	5	15	8D1	2.80	5	15
8A2	2.88	1	14	8B2	2.96	1	14	8C2	2.87	1	15	8D2	2.93	1	15
								8C3	2.76	5	15	8D3	2.83	5	14
								9C	2.92	5	16	9D	2.98	5	14
								9C1	2.78	5	15	9D1	2.86	5	15
								9C2	2.93	1	14	9D2	2.98	1	15
								9C3	2.83	5	15	9D3	2.87	5	14

¹ Axial node (out of 24) from top of fuel plate.

3.3 Summary and Conclusion

Steady-state thermal-hydraulic analyses of 64 reference HEU core states and 64 reference LEU core states without and with the 2017-RBM-99 upgrade were performed. For both fuel types, the limiting thermal safety criterion is the margin to onset of flow instability. The requirement that the CHFR be at least 2.0 at the LSSS power is easily met. For each fuel type, the 2017-RBM-99 upgrade reduces the margin to flow instability. For the HEU core, this margin was reduced from 2.39 MW above the HEU LSSS power of 12.5 MW to 2.06 MW above the LSSS power. For the LEU core, this margin was reduced from 2.70 MW above the LEU LSSS power of 15.0 MW to 2.35 MW above the LSSS power. Thus, the margin to flow instability is reduced for both HEU and LEU with the 2017-RBM-99 upgrade, but both HEU and LEU are calculated to have adequate margin. With the upgrade, the margin for LEU is 0.29 MW greater than the margin for HEU.

Table 3.5 shows that at the LSSS power, the minimum CHFR among all 64 HEU core states is 2.70. Table 3.6 shows that at the LSSS power, the minimum CHFR among all 64 LEU core states is 2.41. Thus, the regulatory requirement that the minimum CHFR be at least 2.0 at the LSSS power is easily met for all 64 core states of each fuel type.

4 TRANSIENT ANALYSES

NUREG-1537 Part 1 guidelines [15] identify nine (9) postulated accidents to be evaluated for non-power reactors. Five of those accidents (Maximum Hypothetical Accident, Insertion of Excess Reactivity, Loss of Primary Coolant, Loss of Primary Coolant Flow, Mishandling or Malfunction of Fuel) are affected by conversion to LEU, and were evaluated in the MURR PSAR [14] for HEU and LEU cores with the experiment configuration and experimental loading which was typical for MURR operations in 2008. Subsequent changes to the MURR experiments and the 2017-RBM-99 upgrade are modeled in this work. Since there is no change to either the HEU or LEU driver fuel element design (except for the change to the end channel tolerances, as discussed in Section 3.1 and Appendix A), the Maximum Hypothetical Accident and Mishandling or Malfunction of Fuel accidents will not be affected by the upgrade. Only transients initiated from the Insertion of Excess Reactivity, Loss of Primary Coolant, and Loss of Primary Coolant Flow scenarios are evaluated here.

4.1 Initial Operating Conditions for Transient Analyses

Reactivity insertion accidents (RIAs), loss-of-coolant accidents (LOCAs), and loss-of-flow accidents (LOFAs) are of the most extreme accidents conceived within the design basis of the reactor. The most extreme of these accidents, which have the potential to be the most detrimental to the reactor plant, are not expected to occur during the life of the reactor and have not occurred during MURR's over 50 years of operation. Nevertheless, the transient behavior of these accidents must be investigated and understood to ensure proper safety margins in reactor designs.

Two important inputs for transient accident analyses are: 1) the initial steady-state conditions of the reactor, i.e., reactor power, primary coolant flow rate, core coolant inlet temperature, etc., and 2) the core state, i.e., burnup of the fuel elements in the core, ^{135}Xe loading, control rod skew, flux trap loading, etc., when the accident is initiated. These are provided in the following sections.

4.1.1 Selection of Initial Steady-State Conditions for Transient Analyses

The initial conditions for all accidents were chosen based on guidance in NUREG-1537 Part 1 [15]. For all accidents, the initial steady-state conditions for the primary flow, reactor inlet temperature, pressurizer pressure, and pool temperature are the LSSS conditions. The pressurizer level is set at the scram set point. 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 because the anti-siphon tank is submerged in the pool. For the RIAs, following the guidance of Reference 15, the initial operating power at the onset of the accident is assumed to be at the rod run-in set point of 115% of the full licensed power. For the LOCAs and LOFAs, again following the guidance of Reference 15, the accidents are assumed to begin from the full licensed power. These initial conditions, outlined in Table 4.1, are more stringent compared to the conservative limits of the normal operating band for MURR and are the same as assumed in the previous MURR safety analyses without the 2017-RBM-99 device reported in Appendix B of Reference 9 and in Chapter 13 of Reference 14.

Table 4.1. Initial Steady-State Conditions for Transient Analyses.

	HEU		LEU	
Accident	RIA	LOCA/LOFA	RIA	LOCA/LOFA
Reactor Power (MW)	11.5	10	13.8	12
Core Coolant Inlet Temperature (°F)	155		145	
Primary Coolant Flowrate (gpm)	3200		3300	
Pressurizer Pressure (psig)	60.7		60.7	
Scram Set Point Pressurizer Level (in)	-16		-16	
Anti-Siphon Tank Pressure (psig)	-	36	-	36
Anti-Siphon Tank Temperature (°F)	-	120	-	120
Pool Temperature (°F)	-	120	-	120

4.1.2 Selection of Core States for Transient Analyses

The core state includes fuel element burnup, flux trap loading, control blade prior history and position, etc. In general, core states significantly affect the transient behavior of accident scenarios in terms of core power distribution. Additionally, the fuel burnup affects the thermo-physical properties of the fuel material and the coolant channel dimensions as they change over the life of the element due to several irradiation-induced phenomena, including, but not limited to, fuel swelling and oxide layer growth on the fuel plate surfaces.

In the steady-state safety analyses performed for the HEU and LEU cores without the 2017-RBM-99 [1,8], a set of 24 reference core states were identified that represent the range of possible conditions that could exist in MURR. These core states include variations in the burnup of the fuel elements (mixed-burnup cores and all fresh fuel), ^{135}Xe loading (no xenon at beginning of cycle, and equilibrium xenon conditions), center test hole loading (loaded with samples typical for MURR, and empty test hole), and critical control blade configuration (fresh blades that are banked, and blades that are highly depleted and mismatched in axial position). However, many of these are core states that the reactor will rarely, if never, achieve. Previous transient accident analysis without the 2017-RBM-99 device [9] identified a more typical core state for each HEU and LEU core, i.e., mixed-burnup cores and flux trap loaded with samples typical for MURR. The identified core states do not have ^{135}Xe , so the control blades must be inserted more deeply than when the core is at equilibrium xenon conditions, which results in a greater maximum local heat flux, and consequently, should be more limiting.

The identified core states – so-called Core 3A and Core 7A for HEU and LEU cores, respectively – have been used for the previous transient accident analysis without the 2017-RBM-99 device [9]. The current analyses with the device use the equivalent core states, which are labeled as 3C and 7C to indicate the presence of the 2017-RBM-99 device. As discussed in Section 2.1, there are a few changes in core configurations assumed for PSAR transient accident analyses (3A and 7A) in addition to the 2017-RBM-99 device, e.g., center test hole apparatus, irradiation samples, etc. However, the impact of these changes on the transient accident results is minimal compared to the significance of the 2017-RBM-99 device.

4.1.3 Modeling of Fuel Elements in Transient Analyses

The burnup of each of the eight elements at BOC for the reference Cores 3C (HEU) and 7C (LEU) is provided in Figure 2.5. The elements in these cores consist of four pairs of elements with similar burnup, which is typical for MURR operations. In the reference cores, the BOL pair is in core positions

1 and 5, an MOL-1 pair is in core positions 3 and 7, an MOL-2 pair is in core positions 2 and 6, and an EOL pair is in core positions 4 and 8. The burnups of these element pairs in the HEU transient analyses were modeled as 0, 73, 81, and 150 MWd for the BOL, MOL-1, MOL-2, and EOL elements, respectively. For the LEU transient analyses, the element burnups were modeled as 0, 77, 96, and 180 MWd for the BOL, MOL-1, MOL-2, and EOL elements, respectively. Note that the EOL elements were modeled at the discharge burnup that would occur at EOC of the reference cores for added conservatism in the analysis.

The transient accident analysis models take into account the impact of fuel burnup on fuel temperature. Three phenomena were considered. First, increasing fuel burnup results in a decrease in the fuel thermal conductivity relative to fresh material for both HEU and LEU. The models for HEU and LEU fuel thermal conductivity used in this work are described in Section 2 of Reference 9. The model for the U-10Mo thermal conductivity reported in Reference 9 was based on available measurements of the thermal conductivity of irradiated U-10Mo [35]. More recent data were reported in References 36 and 6, and future measurements of prototypic material are expected as a part of ongoing fuel qualification work by the FQ Pillar. It is observed in Figure 4.1 that at temperatures and fission densities relevant to MURR transient analyses, the model for the burnup- and temperature-dependent U-10Mo thermal conductivity derived from the data in Reference 35 and used in Reference 9 (solid line) and in the present work is conservative relative to a model developed from data reported in References 36 and 6 (dashed line).

Figure 4.1. U-10Mo Thermal Conductivity with Burnup at T = 200 °C.

The coolant channel restriction that may occur with depletion of the fuel under normal operating conditions was also modeled for selected cases. The modeling of decreased thicknesses of the coolant channels between the fuel plates with element burnup is described in Section 3.1. For HEU cases with coolant flow redistribution between elements of different burnup in the core due to the channel thickness reduction (considered only for reactivity insertion accidents described in Section 4.2.2), it

was assumed that the coolant channel thickness reduces linearly with burnup, up to a maximum reduction of 10 mil at an element burnup of 150 MWd. For LEU cases with coolant flow redistribution between elements of different burnup in the core, it was assumed that the maximum coolant channel thickness reduction is 8 mil at an element burnup of 180 MWd.

The third burnup-related effect considered in the transient analyses is the increased thermal resistance across the fuel plate cladding due to the growth of an oxide layer on the surface of the fuel plates. Griess [37] developed an empirical relationship for the oxide layer thickness as a function of core residence time and plate surface temperature for AA6061 based on experimental measurements in the Advanced Test Reactor. Griess found that the pH of the coolant affects the rate of oxide layer growth. Therefore, the specific condition for the MURR water chemistry is important to consider in estimating the oxide layer thickness for both HEU and LEU fuel plates. Measurements of the oxide layer thickness on the convex surface of plate 24 of several HEU fuel elements were conducted as a part of prior work at MURR in connection with a requested license amendment. [34,38] PLTEMP/ANL was used to perform calculations of the plate surface temperature under nominal conditions over the life of an HEU element. Based on the measured oxide thickness data and the calculated temperatures, the Griess correlation was adjusted by an empirically-determined multiplier of 0.822 that takes into account the MURR water chemistry. This “modified Griess correlation” was used to estimate the oxide layer thickness for HEU and LEU fuel plates.

Table 4.2 provides the estimated oxide thickness for MURR fuel plates. For HEU, the oxide layer thickness on the convex surface of fuel plate 1 was predicted to have the maximum oxide layer thickness for given element burnup because this fuel plate surface has the highest temperature in the element throughout its lifetime. The maximum oxide thickness on this plate surface was conservatively assumed over the entire surface of all fuel plates in an HEU element of a given burnup. For LEU, the oxide layer thickness on the concave side of fuel plate 23 was predicted to have the maximum oxide layer thickness for a given element burnup. Again, this maximum oxide thickness was conservatively assumed over the entire surface of all fuel plates in an LEU element of a given burnup. It can also be seen in Table 4.2 that the 2017-RBM-99 upgrade causes a slight increase in the predicted oxide layer thickness for some elements because of the change in the core power distribution and/or changes to the burnup (residence time) of the elements in the reference HEU cores.

For the RIAs in the MURR core with the 2017-RBM-99 upgrade, a four-channel PARET/ANL v. 7.6 [39] model was utilized that represented the average fuel plate in the core, plus three additional plates that were predicted to be the hottest among the BOL, MOL-1/MOL-2, and EOL elements. The details of this model and the complete results for all RIAs with the 2017-RBM-99 upgrade are provided in Section 4.2.

For HEU, the maximum local heat flux in an element always occurs in fuel plate 1, regardless of the element burnup. Among the BOL, MOL, and EOL elements in HEU Core 3C, those in core positions 5, 3, and 4, respectively, have the maximum local heat flux and were selected for modeling in the RIA analysis. It is noted that fuel plate 1 in the MOL-1 element with a burnup of 73 MWd was selected for analysis rather than the MOL-2 element with a burnup of 81 MWd, because the maximum local heat flux for fuel plate 1 is about 4% greater in the MOL-1 element relative to MOL-2. However, the higher burnup of MOL-2 will result in a thermal conductivity that is slightly lower than that in the MOL-1 element, a narrower coolant channel thickness, and a slightly thicker oxide layer, which could result in a higher fuel temperature. The impact of this modeling choice will be explored in Section 4.2.2.

Table 4.2. Oxide Thickness on MURR Fuel Plates.

Core	Parameter	Element				
		BOL	MOL-1	MOL-2	EOL	Core Average
HEU without 2017-RBM-99 Upgrade	Burnup, MWd (Residence time, hours)	0 (0)	65 (1361)	81 (1663)	150 (2873)	72 (1493)
	Maximum oxide thickness for convex surface of plate 1, mil	0.00	0.34	0.39	0.54	0.36
HEU with 2017-RBM-99 Upgrade	Burnup, MWd (Residence time, hours)	0 (0)	73 (1512)	81 (1663)	150 (2873)	74 (1531)
	Maximum oxide thickness for convex surface of plate 1, mil	0.00	0.36	0.39	0.54	0.36
LEU without 2017-RBM-99 Upgrade	Burnup, MWd (Residence time, hours)	0 (0)	77 (1361)	96 (1663)	180 (2873)	85.9 (1503)
	Maximum oxide thickness for concave surface of plate 23, mil	0.00	0.55	0.63	0.93	0.59
LEU with 2017-RBM-99 Upgrade	Burnup, MWd (Residence time, hours)	0 (0)	77 (1361)	96 (1663)	180 (2873)	85.9 (1503)
	Maximum oxide thickness for concave surface of plate 23, mil	0.00	0.56	0.64	0.94	0.60

For LEU Core 7C, the maximum local heat flux in an element always occurs in fuel plate 23, regardless of the element burnup. Among the BOL, MOL, and EOL elements in HEU Core 7C, those in core positions 5, 6, and 4, respectively, have the maximum local heat flux and were selected for modeling in the RIA analysis. It is noted that fuel plate 23 in the MOL-2 element with a burnup of 96 MWd was selected for analysis rather than the MOL-1 element with a burnup of 77 MWd. The maximum local heat flux for fuel plate 23 is about 5% greater in the MOL-2 element relative to MOL-1, which also has a higher burnup. Therefore, MOL-2 will be limiting relative to MOL-1 for the RIA analysis.

For LOCAs and LOFAs, a one-element RELAP5 model is used for accident simulations. In other words, each simulation is conducted on one element instead of simulating all eight elements at the same time. (More detailed discussion on the model construction can be found in Reference 9.) To simplify the analysis, one element of each of the four pairs of elements was selected to represent the specific burnup state. To determine which element to be simulated, maximum local heat flux was used as a figure of merit. Based on MCNP5 simulation of the selected core state (Core 3C for HEU and Core 7C for LEU), a maximum local heat flux for each plate was computed. An element that contains a plate with the highest maximum local heat flux was selected to represent a specific burnup state to be used in the one-element model for accident simulations. As a result, elements in core positions 5, 3, 2, and 4 were selected for HEU core, and elements in core positions 5, 7, 6, and 4 for LEU core (for BOL, MOL-1,

MOL-2, and EOL, respectively). The complete results of the LOCA and LOFA analyses with the 2017-RBM-99 upgrade are provided in Sections 4.3 and 4.4, respectively.

4.2 Reactivity Insertion Accidents (RIA)

All the analyses in this section were performed using the qualified version of PARET/ANL code, v.7.6 [39]. PARET/ANL is a 1-dimensional reactor transient analysis code that models plate, pin, and nested tube fuel geometries. The collection of fuel meat, clad, 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., the hot-stripe of the fuel plate) which are expected to reach the highest temperature as a result of the transient. In the current analysis, three fuel plate/coolant channels were modeled to represent the fuel plates most likely to have maximum fuel temperatures. The channels to be modeled will be 1) a BOL, unirradiated fuel plate with the maximum local heat flux of all BOL fuel plates in the core, 2) an EOL plate with the maximum local heat flux of all EOL plates in the core, and 3) an MOL plate with the maximum local heat flux of all MOL plates in the core. Appropriate degradation of the HEU U-Al_x or LEU U-10Mo thermal conductivity with fuel burnup, the formation of an oxide layer on irradiated fuel plates, and reduction of the coolant channel thickness due to oxide growth, fuel swelling, and fuel creep will be modeled in the MOL, EOL, and average fuel channels. The details of this modeling are provided in Section 4.1.3.

There is no calculated heat transfer between the channels in PARET/ANL, but the channels are coupled through reactivity feedback effects. 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 kinetics parameters and reactivity feedback coefficients for the HEU and LEU cores with the 2017-RBM-99 upgrade are provided in Table 2.12.

The user provides a description of the fuel geometry, material thermo-physical properties, initial steady-state thermal-hydraulic conditions (see Table 4.1), and the transient initiating condition (e.g., insertion of reactivity) as a time-dependent table. The thermal conductivity and volumetric heat capacity of the plate materials can be supplied as input to PARET/ANL as temperature dependent functions. The coolant properties are generated by the code using built-in functions. A point-kinetics solution is used to calculate the overall core reactivity, total core power, and local heat generation rates as a function of time. The thermal-hydraulics models in PARET/ANL are used to compute the heat conduction, material temperatures (e.g., fuel meat, clad, coolant), and inherent reactivity feedbacks. Lastly, termination of the reactor transient from engineered safety systems (i.e., control blade scram) that are actuated from core power, coolant flow, or reactor period settings can be modeled. As discussed in Section 2.5, the reactivity worth of the control blades for HEU and LEU cores is not affected by the upgrade, so the same values as used in Reference 9 are used in this work.

PARET/ANL uses a user-supplied axial power profile for each channel to compute the thermal-hydraulic conditions and temperatures. It is assumed that while the reactivity insertion will affect the total core power, the power shape relative to the steady-state power distribution will not be affected by the transient. Detailed, 3-dimensional power distribution data for the HEU and LEU reference cores were calculated using MCNP5. These so-called Cores 3C (HEU) and 7C (LEU) are loaded with eight elements with a mixture of burnup, the flux trap is loaded with experiments typical for MURR operations, the 2017-RBM-99 upgrade is installed in the graphite reflector, and the four control blades are banked at the same axial position. For the peak BOL, MOL, and EOL channels in the PARET/ANL models, the 1-D axial heat source was calculated for the plate and azimuthal strip in the core that generates the highest local heat flux. The calculated axial heat source for these channels

accounted for the reduced heat flux into the hot channel due to the uneven power split between the hot and cold surfaces of the plate. For the average channel in the PARET/ANL models, the axial heat source description was calculated by averaging the power density of all plates in the core. Lastly, the total core power at the start of the RIAs was normalized to the fraction of the reactor power that is transferred to the primary coolant system. For HEU, the normalized total core power was 10.81 MW (94.0% of the rod run-in set point of 115% of the full licensed power). For LEU, the total core power was normalized to 13.30 MW (96.4% of the rod run-in set point).

4.2.1 RIA to Be Considered

Insertion of uncontrolled and/or unanticipated positive reactivity either to a critical reactor or during a reactor startup is one of the nine (9) postulated accident-initiating scenarios required to be analyzed for potential radiological consequence per the NUREG-1537 guidelines [15,16]. The exact mechanisms or events that could cause these reactivity insertions can vary, but could include an inadvertent rapid insertion or removal of an experiment from the center test hole, reactor or equipment malfunction, or operator error.

Positive step reactivity insertions, based upon the various maximum reactivity limits given in the MURR Technical Specifications, were evaluated for the selected HEU and LEU cores with the 2017-RBM-99 upgrade. The initial steady-state conditions assumed in the RIA analysis followed the guidance provided in NUREG-1537, and are summarized in Table 4.1.

- 0.60% $\Delta k/k$ step insertion of reactivity. This is the Tech Spec limit for a) total reactivity worth of *all unsecured experiments* in the reactor, b) reactivity worth of *any secured removable experiment*, and c) total reactivity worth of *all experiments in the center test hole*.
- 0.25% $\Delta k/k$ step insertion of reactivity. This is the Tech Spec limit for the reactivity worth of any *unsecured experiment*.
- 0.10% $\Delta k/k$ step insertion of reactivity. This is the Tech Spec limit for the reactivity worth of any *moveable experiment*.

Also, transients initiated by the continuous ramp insertion of positive reactivity based on the simultaneous continuous withdrawal of MURR's four shim control blades were considered for the LEU core.

- *Blade withdrawal accident*. 0.03% $\Delta k/k$ /second ramp insertion of reactivity from full power. This is the Tech Spec limit for the reactivity insertion rate from control blade withdrawal.
- *Startup accident*. 0.03% $\Delta k/k$ /second ramp insertion of reactivity from low power (1.0 W) and a shutdown reactivity of -4.2% $\Delta k/k$. This will simulate a blade withdrawal accident at startup without operator intervention.

Termination of reactor transients from engineered safety systems (i.e., control blade scram) are modeled in the analyses of the step insertion and blade withdrawal accidents. The startup accident was analyzed for 150 seconds after the start of the blade withdrawal, assuming no reactor trip or operator intervention to terminate the transient.

The MURR reactor overpower trip setting is 125% of full-licensed power and was modeled in the PARET/ANL analyses of the step and blade withdrawal reactivity insertion accidents. MURR also has an automated trip setting at a minimum reactor period of 8 seconds. However, it was conservatively assumed in the transient accident analysis that the minimum reactor period trip setting (which is reached almost simultaneously with the step reactivity insertions) fails to initiate a reactor scram.

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 assumed to be 150 milliseconds for the modeling of the RIAs in the current work. The transient analyses include additional conservatism by assuming the scram initiates with the control blades at the fully withdrawn position.

The step insertion of reactivity in HEU and LEU cores results in the highest fuel temperatures and smallest margins to the fuel temperature safety limits among all the RIAs considered. Since the ramp insertion of reactivity RIAs are less severe and are not limiting, they were not evaluated for the HEU core. However, since these transients are relevant to the analyses completed for the MURR PSAR [14] they were repeated here for the LEU core with the 2017-RBM-99 upgrade.

4.2.2 Results from HEU Core RIA Analyses

Results for step reactivity insertion accidents in the reference HEU cores with and without the 2017-RBM-99 upgrade are summarized in Table 4.3. For the 0.6% $\Delta k/k$ step insertion, the maximum HEU fuel temperature for the core with the 2017-RBM-99 upgrade is 223.9 °C and occurs in fuel plate 1 in the BOL element in core position 5. The maximum reactor power in this case reaches 38.18 MW. The cases with a step insertion of 0.25% and 0.10% $\Delta k/k$ yield much lower maximum fuel temperatures.

Table 4.3. Results for Step Reactivity Insertion in Reference HEU Cores.

HEU Core 3C, With 2017-RBM-99 Upgrade

Reactivity Inserted		0.60% $\Delta k/k$	0.25% $\Delta k/k$	0.10% $\Delta k/k$
Maximum Reactor Power, MW		38.18	16.41	13.05
Maximum Fuel Temperature, °C	BOL	223.9	163.0	147.3
	MOL	223.2	156.6	142.2
	EOL	202.5	146.7	134.4

HEU Core 3A, Without 2017-RBM-99 Upgrade

Reactivity Inserted		0.60% $\Delta k/k$	0.25% $\Delta k/k$	0.10% $\Delta k/k$
Maximum Reactor Power, MW		38.08	16.57	13.61
Maximum Fuel Temperature, °C	BOL	221.3	160.3	145.4
	MOL	220.9	156.7	142.5
	EOL	196.8	144.2	132.4

The maximum fuel temperature for the HEU core without the 2017-RBM-99 upgrade is 221.3 °C, and also occurs in a BOL plate. Thus, the upgrade results in a small increase, 2.6 °C, in the maximum predicted HEU fuel temperature for the most limiting step reactivity insertion transient. The largest temperature change due to the 2017-RBM-99 upgrade for the 0.6% $\Delta k/k$ step insertion transient in the HEU core is 5.7 °C, and occurs in the EOL plate.

The fuel safety temperature limit for HEU U-Al_x fuel has been assessed to be about 475 °C (887 °F) [Section 2.6 of Reference 9]. There is a margin of more than 250 °C to the HEU fuel temperature safety limit for all fuel plates.

It is noted that the PSAR analysis was performed with an earlier version of the PARET/ANL code [21], using the controlled source code file PARETANL_7.5_r82_160809. Modifications for the release of the qualified PARET/ANL v.7.6 software have resulted in a slight change in the calculated results, but the differences between the results predicted by the earlier and currently qualified versions of PARET/ANL are inconsequential (≤ 0.01 °C).

The HEU MOL element selected for analysis for the reactivity insertion transients was the MOL-1 (73 MWd) element in core position 3 in Core 3C. The other potentially limiting MOL element is the MOL-2 (81 MWd) element loaded in core position 2. The maximum local heat flux in the MOL-1 element is about 4% greater than for the MOL-2 element, but as discussed above, the greater burnup of the MOL-2 element could result in a higher maximum fuel temperature. A case for the 0.6% $\Delta k/k$ step insertion in the HEU core was evaluated to confirm which of these two MOL elements is most limiting. The maximum fuel temperature for fuel plate 1 in the MOL-2 element was calculated to be 219.2 °C, which is less than the maximum fuel temperature of 223.2 °C for fuel plate 1 in the MOL-1 element (see Table 4.3). Thus, the most limiting MOL element with the 2017-RBM-99 upgrade was evaluated in the cases summarized in Table 4.3.

4.2.3 Results from LEU Core RIA Analyses

Results for step reactivity insertion accidents in reference LEU cores with and without the 2017-RBM-99 upgrade are summarized in Table 4.4. For the 0.6% $\Delta k/k$ step insertion, the maximum LEU fuel temperature for the core with the 2017-RBM-99 upgrade is 299.2 °C and is predicted to occur in fuel plate 23 in the EOL element in core position 4. The maximum reactor power in this case reaches 47.23 MW. Much lower maximum fuel temperatures are predicted for cases with a step insertion of 0.25% and 0.10% $\Delta k/k$.

Table 4.4. Results for Step Reactivity Insertion in Reference LEU Cores.

LEU Core 7C, With 2017-RBM-99 Upgrade

Reactivity Inserted		0.60% $\Delta k/k$	0.25% $\Delta k/k$	0.10% $\Delta k/k$
Maximum Reactor Power, MW		47.23	19.85	15.71
Maximum Fuel Temperature, °C	BOL	256.2	187.0	166.4
	MOL	298.2	207.8	184.1
	EOL	299.2	205.6	182.6

LEU Core 7A, Without 2017-RBM-99 Upgrade

Reactivity Inserted		0.60% $\Delta k/k$	0.25% $\Delta k/k$	0.10% $\Delta k/k$
Maximum Reactor Power, MW		46.69	19.98	16.32
Maximum Fuel Temperature, °C	BOL	256.7	191.0	170.7
	MOL	289.2	204.8	182.7
	EOL	289.4	203.7	182.1

The maximum fuel temperature for the LEU core without the 2017-RBM-99 upgrade is 289.4 °C, and also occurs in an EOL plate 23, although in the element in core position 8 rather than 4. The difference in element location is because of shift in the core radial power distribution as a result of the upgrade. The results show that the upgrade causes an increase of 9.7 °C for the predicted maximum LEU fuel temperature for the most limiting step reactivity insertion transient. There is a slight decrease in the maximum fuel temperature of the BOL fuel plates as a result of the 2017-RBM-99 upgrade.

A burnup-dependent fuel temperature safety limit for U-10Mo monolithic fuel is given in Equation 13-1 of Reference 14. The safety limit was derived as the lower bound of the 95% confidence interval (-2σ) of measured blister threshold temperature data for about 30 irradiated U-10Mo samples. The safety limit for U-10Mo monolithic fuel is formulated as a constant value of 450 °C for unirradiated and low-burnup U-10Mo, and decreases monotonically for fuel burnups above 1.5×10^{21} fissions/cm³.

Plate number 23 in the BOL, MOL, and EOL elements was selected for analysis in PARET/ANL because this plate has the highest maximum heat flux among all other plates in the fuel element. However, other plates could be potentially more limiting with regard to the fuel temperature safety limit. For example, plate 1 has the highest discharge burnup of all plates, and so the lowest U-10Mo thermal conductivity relative to other plates and the smallest safety limit. Plate 4 has the highest heat flux among the plates near the center test hole, especially for depleted elements, as can be seen in Figure 2.9. Plate 22 has the second-highest maximum heat flux among the plates in the fuel element (plate 23 being the highest) and is cooled by two coolant channels that are heated on both sides by fuel plates. By comparison, fuel plate 23 is cooled on one side by a one-sided heated end channel.

The margin between the maximum fuel temperature and the safety limit was calculated for each of the limiting BOL, MOL, and EOL plates reported in Table 2.4, plus these additional plates that could be potentially more limiting. The results for the maximum fuel temperature and margin to the safety limit for the 0.6% $\Delta k/k$ step insertion in LEU reference Core 7C are provided in Table 4.5. The maximum fuel temperature of 307 °C occurs in EOL plate 22. This is 8 °C greater than the EOL plate 23 maximum fuel temperature. Likewise, the MOL plate 22 maximum fuel temperature is 2 °C greater than plate 23 of the same element. The minimum margin to the fuel temperature safety limit for the most limiting step insertion accident, 102 °C, occurs in the EOL plate 23. The safety margin for the LEU core with the 2017-RBM-99 upgrade is 7 °C less than the safety margin for the core without the upgrade evaluated in the PSAR [14].

The PARET/ANL code, v. 7.6, allows that the channels in the model may have differing values for the thicknesses of the fuel plate components (e.g., fuel, Zr interlayer, cladding, and oxide) and coolant channel with the IGEOM=2 option. This allows for an average channel that represents all fuel plates in the reactor core and one or more “hot channels” with differing dimensions that represent various fuel plates that are anticipated to be limiting. In the analysis performed here and in Reference 7, the hot channels in the PARET/ANL model represented the hottest stripe of the potentially limiting fuel plate 23 in BOL, MOL, and EOL fuel elements.

The average channel in the PARET/ANL model represented an average fuel plate, which was calculated for the LEU element to have a fueled portion with a fuel width of 2.868 inches. The width of the coolant channels in MURR are defined by the spacing between the element side plates. For the average coolant channel in the LEU element, this spacing, which is identified as the plate width in the model, is 3.008 inches. This results in a nominal 0.070 inch unfueled portion between the fuel and the side plate along the two vertical edges of each fuel plate.

Table 4.5. Results for 0.6% $\Delta k/k$ Step Reactivity Insertion in Reference LEU Core with the 2017-RBM-99 Upgrade.

	BOL Element				MOL Element				EOL Element			
Plate	1	4	22	23	1	4	22	23	1	4	22	23
Fuel thickness (mil)	9	20	20	17	9	20	20	17	9	20	20	17
Maximum Local Burnup ($\times 10^{21}$ fissions/cm ³)	0	0	0	0	1.93	0.98	1.06	1.37	3.32	1.83	2.01	2.52
Maximum Fuel Temperature (°C)	231	252	253	256	244	288	300	298	229	296	307	299
Fuel Temperature Safety Limit (°C)	450	450	450	450	422	450	450	450	380	427	419	401
Margin to Safety Limit (°C)	219	198	197	194	178	162	150	152	151	131	112	102

The heat source descriptions for the hot channels were calculated from the MCNP5 power density tallies for the 0.5 cm 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. With this assumption in mind, it is appropriate to model the plate width as the same size as the fuel width for the hot channels. However, PARET/ANL v. 7.6 has a limitation that does not allow the ratio between the fuel width and plate width of the hot channels to be different from the average channel. Consequently, the coolant channel flow area for the hot channels is 4.9% ($=3.008/2.868-1$) too large in the PARET/ANL models used in the analysis for the LEU fuel element. This limitation will be corrected in a future version of the software.

Two test cases were developed that each set the plate width size the same as the fuel width for the hot channels to understand the impact of this software limitation on the predicted maximum fuel temperature. The test cases were run for the limiting 0.6% $\Delta k/k$ step reactivity insertion accident in the LEU core. In the first test case, the plate width for the average channel was set equal to the fuel width area (plate width = fuel width = 2.868 inches in the model). With this modeling change, the hot channels will be modeled with the correct coolant channel flow. However, this is a non-conservative modeling assumption, because the average channel flow area will be smaller than nominal, which will result in greater reactivity feedback from coolant temperature and voiding during the transient. As expected in this case, the maximum reactor power as a result of the transient is predicted to be 47.09 MW, which is lower than the maximum reactor power of 47.23 MW for the case presented in Table 4.4 with the correct plate width and fuel width for the average channel. The maximum fuel temperature in this case is 298.2 °C in the EOL plate, which is 1.0 °C lower than the maximum fuel temperature in Table 4.4, and is consistent with the lower maximum power predicted for the transient.

In the second test case, no adjustment was made to the plate width for the hot channels or the average channel. Instead, the coolant channel thicknesses for the hot channels were adjusted in the model to provide the correct coolant channel flow areas. Figure 4.2 illustrates the coolant channel model in PARET/ANL. The plate width and fuel width for the average and hot channels are noted as PW and FW, respectively. The parameter C is half of the coolant channel thickness (half of the distance from

the outer surface of the fuel plate to the adjacent fuel plate), RN is the half-thickness of the model from the center of the fuel to the center of the coolant channel, and RS is the plate half-thickness. It should be noted that the values for C, RN, and RS are affected by the assumptions of burnup-dependent fuel swelling and oxide thickness, so they are different for each of the hot channels which represent limiting channels in BOL, MOL, and EOL elements. The coolant channel flow area represented in the model is calculated as $C \times PW$, but this is too large for the hot channels since it includes the unfueled portion of the fuel plate. This results in an incorrect coolant temperature for the hot channels. The correct coolant channel flow area for the hot channels is shown by dark blue in Figure 4.2 and is calculated as $A = C \times FW$. Since the width of all coolant channels in PARET/ANL v.7.6 are modeled as PW, adjusted values for C' and RN' were calculated that preserve the coolant flow area, A. The adjusted coolant flow area is shown in light blue. Lastly, it is necessary to ensure that the adjustment to RN' in the model to obtain the correct coolant channel flow area and coolant temperature does not affect the correct equivalent hydraulic diameter used for heat transfer correlations. This diameter is calculated by applying the following formula to the dark blue flow area and used as an input parameter in the PARET model:

$$DE = 4 \cdot \text{flow area} / \text{wetted perimeter}.$$

Figure 4.2. PARET/ANL Coolant Channel Model.

The 0.6% $\Delta k/k$ step reactivity insertion transient was modeled for the reference LEU core with the thicknesses of the hot channels represented by the adjusted values of RN' for the hot channels. The maximum reactor power in this case is 47.23 MW, which is the same as the result reported in Table 4.4 for this transient because no change was made to the modeling of the average channel. For the hot channels, the maximum fuel temperature in an EOL plate is 299.4 °C when the correct coolant channel flow area is modeled. This is an increase of 0.2 °C relative to the results in Table 4.4. Thus, the software limitation has a small non-conservative but nearly negligible effect on the maximum fuel temperature.

In addition to step insertion of reactivity transients, transients initiated by the continuous ramp insertion of positive reactivity were also evaluated for LEU cores with the 2017-RBM-99 upgrade. These transients were modeled in PARET/ANL assuming the maximum reactivity insertion rate allowed by the MURR Tech Spec limit from continuous withdrawal of MURR's four control blades. The insertion limit is 0.03% $\Delta k/k$ /second. Analysis performed in Reference 14 confirmed that the simultaneous withdrawal of all four control blades, which were assumed to be at a fresh composition, inserts reactivity at a lower rate than the Tech Spec limit.

A ramp reactivity insertion from a continuous blade withdrawal without operator intervention in a core operating at the rod run-in set point of 115% of the full-licensed power and the primary coolant flow rate and inlet temperature at the LSSS conditions was evaluated. The reactor overpower trip setting is at 125% of the full-licensed power.

Likewise, the startup accident assumes the reactor at an initial shutdown reactivity of $-4.2\% \Delta k/k$ and low power of 1 W. The startup accident is conservatively modeled assuming 150 seconds before any operator intervention or scram occurs. The primary coolant flow rate and inlet temperature are set at the LSSS conditions in analysis of the startup accident.

Results for these limiting ramp reactivity insertion accidents in the LEU core with the 2017-RBM-99 upgrade are presented in Table 4.6. For the blade withdrawal accident, the maximum LEU fuel temperature for the core with the 2017-RBM-99 upgrade is $181.0\text{ }^{\circ}\text{C}$ and is predicted to occur in plate 23 in the MOL element in core position 6. The maximum reactor power in this accident reaches 15.06 MW, which is just above the reactor overpower trip setting of 15 MW. For the LEU core without the 2017-RBM-99 upgrade the maximum fuel temperature is $176.6\text{ }^{\circ}\text{C}$, which also occurs in an MOL plate 23. The results show that the upgrade causes an increase of $4.4\text{ }^{\circ}\text{C}$ for the predicted maximum LEU fuel temperature for the most limiting ramp reactivity insertion transient. As for the step insertion accident, there is a slight decrease in the maximum fuel temperature of the BOL fuel plates as a result of the 2017-RBM-99 upgrade, because of the shift in the power distribution.

Table 4.6. Results for Ramp Reactivity Insertion in Reference LEU Cores.

LEU Core 7C, With 2017-RBM-99 Upgrade

Accident		Blade Withdrawal Accident	Startup Accident (results at 150 seconds; no scram or operator intervention)
Maximum Reactor Power, MW		15.06	11.77 W
Maximum Fuel Temperature, $^{\circ}\text{C}$	BOL	163.5	62.8
	MOL	181.0	62.8
	EOL	179.8	62.8

LEU Core 7A, Without 2017-RBM-99 Upgrade

Accident		Blade Withdrawal Accident	Startup Accident (results at 150 seconds; no scram or operator intervention)
Maximum Reactor Power, MW		15.04	11.77 W
Maximum Fuel Temperature, $^{\circ}\text{C}$	BOL	164.6	62.8
	MOL	176.6	62.8
	EOL	176.3	62.8

For the startup accident in LEU cores with and without the 2017-RBM-99 upgrade, the reactor power remains low throughout the 150 seconds modeled in the PARET/ANL analysis. During this time, it is assumed that there is no operator intervention or reactor scram, which would normally occur as a result of the minimum reactor period trip setting, to terminate the transient. (The reactor period drops below the minimum period trip setting of 8 seconds at around 145 seconds after the beginning of the reactivity insertion, when the reactor power is less than 6 W, but it is assumed that no scram occurs as a result of the period trip.) For the startup accident with no actuation of the engineered

safety systems and a delay of operator intervention for 150 seconds (2.5 minutes), the power increases to 11.8 W in the LEU core with the 2017-RBM-99 upgrade, and the fuel temperature does not rise above 62.8 °C, which is the temperature of the reactor coolant. Except for a small increase in the maximum reactor power (see the results for Core 7A in Table 4.6), the upgrade does not affect the results of the startup accident.

4.2.4 Summary of RIA Analyses

The effect of the 2017-RBM-99 upgrade on the consequences of several RIAs in HEU and LEU cores has been evaluated. For the step insertion of 0.6% $\Delta k/k$ in the reference HEU core, the maximum fuel temperature increases less than 3 °C as a result of the upgrade relative to the MURR experimental loading assumed in the PSAR. The calculated margin to the HEU fuel temperature safety limit for the most limiting accident is more than 250 °C.

For the step insertion of 0.6% $\Delta k/k$ in the reference LEU core, the maximum fuel temperature increases 9.7 °C as a result of the upgrade relative to the MURR experimental loading assumed in the PSAR. The maximum fuel temperature occurs in an EOL fuel plate and the margin to the fuel temperature safety limit is 102 °C for the most limiting RIA.

A blade withdrawal accident was evaluated for LEU cores with and without the 2017-RBM-99 upgrade. The upgrade results in a 4.4 °C increase in the maximum fuel temperature in an MOL plate relative to the MURR experimental loading assumed in the PSAR. The margin to the fuel temperature safety limit in this case is nearly 270 °C. Lastly, a reactor startup accident was modeled without reactor scram or operator intervention. The upgrade does not affect the results of the startup accident.

4.3 Loss-of-Coolant Accidents (LOCA)

The presence of 2017-RBM-99 device, while it does not dramatically affect the performance of the core, is predicted to cause a slight change in neutron flux distribution, and thereby alters the core power distribution. The qualified software RELAP5/MOD3.3 Patch 05 [40] was used as the simulation tool for the present analysis. The same modeling methods were used in the present analysis as for the PSAR analysis (Appendix B of Reference 9). The only differences in the models are those to reflect differences in power shape and burnup that resulted from the inclusion of the 2017-RBM-99 device. Table 4.1 summarizes the set of initial conditions for the transient accident analyses.

In the following sections, comparisons with the results without the 2017-RBM-99 device are made in order to gauge the impact of the 2017-RBM-99 device on the safety margin. The original set of calculations for the accident analysis without the 2017-RBM-99 device was previously conducted with RELAP5/MOD3.3 Patch 03 and was reported in Reference 14 and Appendix B of Reference 9. In order to address the software version change, the previous calculations were repeated with RELAP5/MOD3.3 Patch 05 (see Appendix D) in order to facilitate direct comparison of results with and without the 2017-RBM-99 device.

4.3.1 LOCA to be Considered

The following four LOCA scenarios were selected based on the previous transient accident analysis without the 2017-RBM-99 device [9]:

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).
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 schematic view of the MURR primary system loop, shown in Figure 4.3, indicates the potential failure locations in the model. Each potential pipe break/breach is marked by green arrows and is modeled as an initially fully closed valve connected to a fixed-pressure air sink at an atmospheric pressure (14.3 psia) via a pipe with the same diameter as the original pipe, which can instantaneously become fully open.

Figure 4.3. Schematic View of MURR Primary Loop.

These four LOCAs have been simulated with RELAP5 for each HEU and LEU reference core (Core 3C and 7C, respectively), for each burnup condition (four different burnups: BOL, two MOLs and EOL) using the one-element model discussed in Reference 9. For the LEU core, simulations have been conducted with and without considering flow redistribution due to coolant channel reduction due to

oxide layer growth and fuel swelling. Thus, a total of 48 simulations have been conducted to cover the LOCA analysis (16 for the HEU core and 32 for LEU).

Flow redistribution in the LEU core due to coolant channel reduction was considered via a separate hydraulics model using spreadsheet calculations, in the same manner as the previous transient accident analyses [9]. The magnitude of the coolant channel reduction was calculated assuming a linear reduction as a function of burnup with a maximum reduction of 8 mil at the discharge burnup (180 MWd for the LEU element). Flow redistribution for the HEU elements was not considered, because it has previously been observed that the most limiting element for the HEU core is the BOL element and therefore simulating accidents without the flow redistribution would give more conservative estimates. For LEU elements, however, it was observed that the high burnup elements could be more limiting, which is the reason for including simulations with flow redistribution.

4.3.2 Results from HEU Core LOCA Analyses

Transient behavior of each LOCA scenario was simulated with RELAP5. The results for HEU with the 2017-RBM-99 device, core state 3C, are summarized in Table 4.7. These simulation results have been analyzed based on the margin to the fuel temperature safety limit, which for HEU aluminide fuels has been assessed to be about 475 °C (887 °F) [9]. In the table, the highest maximum fuel temperature and the lowest minimum margin to fuel temperature safety limit, which are the most limiting conditions, are bolded. The location where the maximum fuel centerline temperature occurs is also noted. For the HEU core, there are a total 24 fuel plates per fuel element. They are numbered from inside out, i.e., the innermost and outermost plates are numbered as 1 and 24, respectively. Within each plate, there are four axial nodes in the RELAP5 model that are numbered from the top to bottom of the fuel plate, i.e., the uppermost and bottommost nodes are numbered as 1 and 4. It is also worth mentioning that the coolant flow in the core is downward.

For the HEU reference core (3C), the most limiting LOCA scenario occurs in the BOL element (core position 5) and is the cold-leg double-ended break. The maximum fuel temperature for this scenario is 291 °C (556 °F) in plate 2, which gives a minimum margin to the fuel temperature safety limit of 184 °C (331 °F) in the same location. The accident analysis without the 2017-RBM-99 device predicts that the most limiting LOCA scenario for HEU core is also the cold-leg double-ended break. The maximum fuel temperature and minimum margin to fuel temperature safety limit for this case are 246 °C (474 °F) and 229 °C (413 °F), respectively, and are both in plate 2 of the BOL element (core position 5).

More detailed transient behavior of the cold-leg double-ended break LOCA is outlined in Figure 4.4 for the HEU core without considering flow redistribution due to burnup.

Table 4.7. HEU Core 3C LOCA Analyses Results without Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Cold-leg LOCA, Double-ended	BOL (0)	128	291 (556)	163.16	2, 3	184 (331)	2, 3
	MOL-1 (73)	125	182 (360)	57.92	2, 3	293 (527)	2, 3
	MOL-2 (81)	123	177 (350)	53.78	2, 3	298 (537)	2, 3
	EOL (150)	118	191 (377)	73.32	2, 3	284 (510)	2, 3
Cold-leg LOCA, 8"	BOL (0)	131	140 (284)	8.71	1, 3	335 (603)	1, 3
	MOL-1 (73)	125	134 (273)	9.59	2, 3	341 (614)	2, 3
	MOL-2 (81)	123	132 (270)	9.23	2, 3	343 (617)	2, 3
	EOL (150)	118	126 (259)	7.96	2, 3	349 (628)	2, 3
Cold-leg LOCA, 2"	BOL (0)	131	132 (269)	0.17	1, 3	343 (618)	1, 3
	MOL-1 (73)	125	130 (267)	5.77	2, 3	345 (620)	2, 3
	MOL-2 (81)	123	130 (266)	7.24	2, 3	345 (621)	2, 3
	EOL (150)	118	130 (265)	11.47	2, 3	345 (622)	2, 3
Hot-leg LOCA, Double-ended	BOL (0)	131	131 (269)	0.00	1, 3	344 (618)	1, 3
	MOL-1 (73)	125	125 (257)	0.00	1, 3	350 (630)	1, 3
	MOL-2 (81)	123	123 (253)	0.00	1, 3	352 (634)	1, 3
	EOL (150)	118	118 (245)	0.00	2, 3	357 (642)	2, 3

Figure 4.4. Cold-Leg Double-Ended Break LOCA Transients for HEU Core 3C BOL Element (Element 5) without Flow Redistribution.

The first subplot in Figure 4.4 shows the maximum fuel temperature history at the hottest location (corresponding to the highest maximum fuel temperature; bolded in Table 4.7), which is node 3 of plate 2. The four vertical lines in all subplots indicate major events in the LOCA: transient initiation, scram, anti-siphon activation, and when the maximum fuel temperature occurs. The second subplot represents heat transfer mode for both surfaces of the same plate and node. The heat transfer mode is strictly integer and each value signifies a different heat transfer regime at a given time and location. RELAP5 defines the heat transfer regime associated with each heat transfer mode verbatim as follows [40]:

0. Convection to noncondensable-steam-water mixture or superheated liquid
1. Convection at supercritical pressure or superheat wall with negative heat flux due to superheated gas
2. Single-phase liquid convection at subcritical pressure, subcooled wall and low void fraction
3. Subcooled nucleate boiling
4. Saturated nucleate boiling
5. Subcooled transition boiling
6. Saturated transition boiling
7. Subcooled film boiling
8. Saturated film boiling
9. Single-phase vapor or supercritical two-phase convection

When noncondensable gases are present, RELAP5 uses the same mode numbers, but adds 20 to each number. At about 1 second after the beginning of the transient, the anti-siphon system valve opens and injects air, which is a noncondensable gas, into the reactor vessel. Therefore, the heat transfer mode number increases by 20 shortly after anti-siphon activation, and is indicated by the axis scale on the right hand side and corresponds to the dashed lines in Figure 4.4. It should be noted that the heat transfer modes of 4 and 5 (24 and 25 with noncondensable gases) are saturated nucleate boiling and subcooled transition boiling, respectively. Consequently, it can be expected that the CHF occurs when the heat transfer mode changes from 4 to 5 (24 to 25 with noncondensable gases), which occurs between scram and anti-siphon activation for this case. Also, all heat transfer modes above 5 (25 with noncondensable gases) are without liquid contact, indicating a significantly less efficient mode of heat transfer.

The third subplot shows the coolant flow on either side of the hottest fuel plate. It is shown that, from the point of accident transition initiation, the coolant flow on both sides of the plate goes down to essentially zero. As previously mentioned, the anti-siphon system injects air into the reactor vessel, pushing the flow upward. This is indicated by a significant dip in coolant flow rate. Negative flow rate indicates upward flow because the coolant flow is downward during normal operation. The pressure in the reactor vessel follows a similar pattern as shown in the last subplot. After a significant depressurization, the pressure in the reactor vessel spikes due to the activation of the anti-siphon system which follows the dip in coolant flow rate. The induced upward coolant flow then sufficiently cools the fuel plates as shown in the first subplot.

4.3.3 Results from LEU Core LOCA Analyses

Transient behavior of each LOCA scenarios was simulated with RELAP5. The results for LEU with and without flow redistribution due to burnup are summarized in Table 4.8 and Table 4.9, respectively. Unlike HEU aluminide fuels, the fuel temperature safety limit for LEU U-10Mo monolithic fuels is dependent on burnup. The fuel temperature safety limit for LEU fuel was calculated using the formulation provided by Equation 13-1 of Reference 14. The LEU element has 23 fuel plates, which

Table 4.8. LEU Core 7C LOCA Analyses Results with Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Cold-leg LOCA, Double-ended	BOL (0)	136	328 (622)	191.79	3, 3	122 (220)	3, 3
	MOL-1 (77)	148	299 (571)	151.71	4, 3	151 (271)	4, 3
	MOL-2 (96)	154	312 (593)	157.08	22, 3	138 (249)	22, 3
	EOL (180)	155	322 (611)	167.08	4, 3	105 (188)	4, 3
Cold-leg LOCA, 8"	BOL (0)	139	147 (296)	8.04	4, 3	303 (546)	4, 3
	MOL-1 (77)	150	157 (315)	6.73	23, 3	293 (527)	23, 3
	MOL-2 (96)	156	162 (324)	6.34	23, 3	287 (517)	1, 3
	EOL (180)	158	165 (329)	7.36	22, 3	239 (431)	23, 3
Cold-leg LOCA, 2"	BOL (0)	139	139 (282)	0.17	4, 3	311 (560)	4, 3
	MOL-1 (77)	150	150 (303)	0.17	23, 3	300 (539)	23, 3
	MOL-2 (96)	156	156 (313)	0.17	23, 3	294 (529)	23, 3
	EOL (180)	158	158 (316)	0.22	22, 3	247 (444)	23, 3
Hot-leg LOCA, Double-ended	BOL (0)	139	139 (282)	0.00	4, 3	311 (560)	4, 3
	MOL-1 (77)	150	150 (302)	0.00	23, 3	300 (540)	23, 3
	MOL-2 (96)	156	156 (312)	0.00	23, 3	294 (530)	23, 3
	EOL (180)	158	158 (316)	0.00	22, 3	247 (444)	23, 3

are numbered from inside out, i.e., the innermost and outermost plates are numbered as 1 and 23, respectively. Within each plate, there are four axial nodes in the RELAP5 model that are numbered from the top to the bottom of the fuel plate, i.e., the uppermost and bottommost nodes are numbered as 1 and 4. Coolant flow in the core is downward.

For the LEU reference core (7C), which includes the 2017-RBM-99 device and flow redistribution due to burnup, the most limiting LOCA is the cold-leg double-ended break. The maximum fuel temperature for this scenario occurs in fuel plate 3 of a BOL element and is 328 °C (622 °F). The minimum margin to the fuel temperature safety limit of 105 °C (188 °F) occurs in fuel plate 4 of an EOL element. The minimum margin can occur in an EOL element, while the maximum fuel temperature is in a BOL element, because EOL elements have a higher fuel burnup and therefore have a lower safety limit. The accident analysis without the 2017-RBM-99 device predicts that the most limiting LOCA scenario for LEU core with flow redistribution is also the cold-leg double-ended break. The maximum fuel temperature and minimum margin to the fuel temperature safety limit are 299 °C (569 °F) in plate 4 of the EOL element and 118 °C (212 °F) in the same location, respectively.

Table 4.9. LEU Core 7C LOCA Analyses Results without Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Cold-leg LOCA, Double-ended	BOL (0)	139	349 (660)	209.92	3, 3	101 (182)	3, 3
	MOL-1 (77)	150	314 (597)	164.39	22, 3	136 (245)	22, 3
	MOL-2 (96)	156	309 (588)	152.90	4, 3	141 (254)	4, 3
	EOL (180)	155	294 (561)	138.94	4, 3	133 (239)	4, 3
Cold-leg LOCA, 8"	BOL (0)	142	149 (301)	7.30	4, 3	301 (541)	4, 3
	MOL-1 (77)	152	158 (316)	5.38	23, 3	292 (526)	23, 3
	MOL-2 (96)	158	163 (325)	4.97	23, 3	287 (516)	1, 3
	EOL (180)	158	164 (326)	5.40	22, 3	240 (432)	23, 3
Cold-leg LOCA, 2"	BOL (0)	142	142 (288)	0.20	4, 3	308 (554)	4, 3
	MOL-1 (77)	152	153 (307)	0.18	23, 3	297 (535)	23, 3
	MOL-2 (96)	158	158 (316)	0.17	23, 3	292 (526)	23, 3
	EOL (180)	158	158 (317)	0.18	22, 3	246 (442)	23, 3
Hot-leg LOCA, Double-ended	BOL (0)	142	142 (288)	0.00	4, 3	308 (554)	4, 3
	MOL-1 (77)	152	152 (306)	0.00	23, 3	298 (536)	23, 3
	MOL-2 (96)	158	158 (316)	0.00	23, 3	292 (526)	23, 3
	EOL (180)	158	158 (317)	0.00	22, 3	246 (442)	23, 3

For the LEU reference core (7C) with the 2017-RBM-99 device and without flow redistribution due to burnup, the most limiting LOCA is the cold-leg double-ended break. The maximum fuel temperature for this scenario occurs in fuel plate 3 of the BOL element and is 349 °C (660 °F). The minimum margin to the fuel temperature safety limit of 101 °C (182 °F) occurs in the same location. The accident analysis without the 2017-RBM-99 device predicts that the most limiting LOCA scenario for LEU core without flow redistribution is also the cold-leg double-ended break. The maximum fuel temperature and minimum margin to the fuel temperature safety limit are 289 °C (552 °F) in plate 3 of BOL element and 139 °C (250 °F) in plate 4 of EOL element, respectively.

Detailed transient behavior of the cold-leg double-ended break LOCA is shown in Figure 4.5 and Figure 4.6 for the LEU core with and without flow redistribution due to burnup, respectively. As discussed in the previous section, each subplot represents the maximum fuel temperature, heat transfer mode, coolant flow rate at the hottest location, and primary system pressure, respectively. It is worth noting that the heat transfer mode changes from 4 to 5 between transient initiation and scram as shown in the second subplot. This indicates that both sides of the fuel plate reach CHF before scram and remain in the poor heat transfer modes of 5 (25 with noncondensable gases) or above until

Figure 4.5. Cold-Leg Double-Ended Break LOCA Transients for LEU Core 7C BOL Element (Element 5) with Flow Redistribution.

Figure 4.6. Cold-Leg Double-Ended Break LOCA Transients for LEU Core 7C BOL Element (Element 5) without Flow Redistribution.

well beyond the maximum fuel temperature has been reached. It can therefore be deduced that what causes the fuel temperature to stop rising and decline is not a return to an efficient heat transfer mode, but rather the activation of the anti-siphon system.

It is noted that the maximum fuel temperature for the cold-leg double-ended break LOCA in the reference LEU core with the 2017-RBM-99 upgrade is 349 °C (660 °F) and occurs in fuel plate 3 of the BOL element in core position 5 when there is no coolant flow redistribution modeled. For the PSAR analysis, the maximum BOL element fuel temperature for this LOCA without flow redistribution is 289 °C (552 °F) and also occurs in fuel plate 3 in the BOL element in core position 1. Thus, the upgrade causes the maximum BOL fuel temperature to increase by 60 °C (108 °F) for the cold-leg double-ended LOCA without flow redistribution due to burnup.

Figure 4.7 compares the maximum fuel temperature for the limiting BOL plates during the LOCA without flow redistribution for the reference LEU cores in the PSAR analysis and with the 2017-RBM-99 upgrade. The steady-state maximum fuel temperature in fuel plate 3 is 2 °C higher with the upgrade, because the maximum volumetric heat generation rate is 5.5% greater with the upgrade. During the LOCA, the fuel temperature increases for both reference cores, but increases to a much higher value with the upgrade. This is because, as shown in Figure 4.6, the heat transfer mode oscillates between transition boiling (heat transfer modes 5 and 6) and film boiling (heat transfer modes 7 and 8) on both surfaces of the fuel plate during the LOCA with the 2017-RBM-99 upgrade, whereas in the case without the 2017-RBM-99 device, film boiling tends to occur on only one surface of the fuel plate for a significant duration (see Figure D.2). Since film boiling inhibits heat transfer, the maximum fuel temperature is significantly greater with the upgrade.

Figure 4.7. Transient Fuel Temperatures for Cold-Leg Double-Ended Break LOCA Transients for LEU Core 7C BOL Element (Element 5) without Flow Redistribution in Plate 3, Node 3.

4.3.4 Summary of LOCA Analyses

Transient behavior for four LOCA scenarios has been analyzed. As is true for the earlier analysis without the 2017-RBM-99 device in the PSAR analysis, among all four LOCAs, the most limiting LOCA

is the cold-leg double-ended break scenario. For both HEU and LEU reference cores with the 2017-RBM-99 upgrade, the most limiting element is the BOL element.

For the HEU core, plate 2 is the most limiting with a maximum fuel temperature of 291 °C (556 °F) and the minimum margin to the fuel temperature safety limit is 184 °C (331 °F). Compared to previous analyses without the 2017-RBM-99 device, the maximum fuel temperature is approximately 45 °C (81 °F) higher and the minimum margin to fuel safety temperature limit is 45 °C (82 °F) lower when the Mo-99 production upgrade is installed.

For the LEU core, flow redistribution due to burnup was considered. Because the coolant channel flow area is reduced in the depleted elements and the magnitude of the reduction increases with increasing burnup, the coolant flowrate through the more burned elements will be reduced, pushing more flow to the less burned elements. Therefore, including flow redistribution is expected to lower the peak fuel centerline temperatures as well as the initial steady state temperatures in BOL elements and increase those in EOL elements. Both pairs of simulation results (the pair with the 2017-RBM-99 upgrade and the pair evaluated in the PSAR analysis without the upgrade) confirm this, as shown by comparing Table 4.8 with Table 4.9, and Table D.6 with Table D.7.

Calculation results in Appendix D without the 2017-RBM-99 device indicate that the LEU element with the smallest safety margin for the LOCA is the EOL element with and without flow redistribution. When the 2017-RBM-99 device is included, the most limiting element with flow redistribution remains an EOL element. However, when the 2017-RBM-99 device is included without flow redistribution, BOL element 5 becomes the most limiting. This is due to not having the beneficial effect of flow distribution in this BOL element, in addition to the power being pulled towards the 2017-RBM-99 device, which is located near BOL element 5.

Regarding the analysis of the LEU core with flow redistribution, the most limiting location is plate 4 of the EOL element with minimum margin to the fuel temperature safety limit of 105 °C (188 °F) even though the maximum fuel temperature of 328 °C (622 °F) occurs in the BOL element. This is because EOL elements have greater fuel burnup, which results in a lower fuel temperature safety limit. Without considering flow redistribution, the most limiting location is also plate 3 of the BOL element with the maximum fuel temperature of 349 °C (660 °F) and minimum margin to the fuel temperature safety limit of 101 °C (182 °F). Compared to the calculation results without the 2017-RBM-99 device, the maximum fuel temperature for the entire set of LEU LOCA scenarios is 50 °C (90 °F) higher and the minimum margin to the fuel temperature safety limit is 17 °C (31 °F) lower.

Because all LOCA simulations were conducted with the one-element model [9], one element from each burnup pair was selected based on the maximum local heat flux rather than the hot-stripe heat flux, as discussed in Section 2.3 and 4.1.3. Because the focus in transient accident analyses is the maximum fuel temperature and whether it is below a specific fuel temperature safety limit, selecting an element that contains a plate with the highest maximum local heat flux would more likely represent a case with a higher maximum fuel temperature and consequently lower margin to the safety limit.

The same element would be selected no matter which figure of merit is used for 7 of the 8 burnup pairs in the reference HEU and LEU cores (BOL, MOL-1, MOL-2, and EOL for each fuel type). Only the MOL-1 pair (~73 MWd) for the reference HEU core gives a different element when the hot-stripe heat flux is used as the figure of merit instead of the maximum local heat flux. Element 3 has the highest maximum local heat flux while element 7 has the highest hot-stripe heat flux. Full LOCA analyses have been conducted for both of these HEU elements in order to confirm that selecting elements with the

highest maximum local heat flux indeed provides more conservative results. The results are summarized in Table 4.10. Note that the results for element 3 in Table 4.10 are the same as presented in Table 4.7.

Table 4.10. HEU Core 3C LOCA Analyses Results without Flow Redistribution for MOL-1 (73 MWd) Pair.

Accident	Core Position	Initial Temp. at Peak Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Cold-leg LOCA, Double-ended	3	125	182 (360)	57.92	2, 3	293 (527)	2, 3
	7	124	182 (359)	57.24	2, 3	294 (528)	2, 3
Cold-leg LOCA, 8"	3	125	134 (273)	9.59	2, 3	341 (614)	2, 3
	7	126	134 (273)	7.59	2, 3	341 (614)	2, 3
Cold-leg LOCA, 2"	3	125	130 (267)	5.77	2, 3	345 (620)	2, 3
	7	124	130 (266)	6.00	2, 3	345 (621)	2, 3
Hot-leg LOCA, Double-ended	3	125	125 (257)	0.00	1, 3	350 (630)	1, 3
	7	124	124 (256)	0.00	1, 3	351 (631)	1, 3

It is expected that an element with the highest maximum local heat flux is more likely to represent a case with highest maximum fuel temperature. The maximum fuel temperatures for HEU element 3 are equal or greater than those for element 7 for all four LOCA scenarios. Also, the same trend is observed regarding the minimum margin to the fuel temperature safety limit. The minimum safety margins for element 3 are equal to or less than those for element 7 for all 4 scenarios. This comparison confirms that the element with highest maximum local heat flux indeed experiences higher maximum fuel temperatures during the LOCAs, as well as lower minimum margin to the fuel temperature safety limit, thus validating the modeling approaches to provide conservative results for the transient accident analyses.

4.4 Loss-of-Flow Accidents (LOFA)

The modeling approach for LOFAs is almost identical to LOCAs. LOCAs occur when the coolant itself is lost due to pipe break/breach, whereas LOFAs arise when the coolant flow is lost without losing the actual coolant, e.g., pump failure or valve closure. The initial conditions used for the analysis of postulated LOFAs are summarized in Table 4.1 and are the same as for the LOCAs.

As discussed in Section 4.3, the original calculations for the accident analysis without the 2017-RBM-99 device were repeated with the same version of the RELAP5 software (RELAP5/MOD3.3 Patch 05) and the results are documented in Appendix D.

4.4.1 LOFA to Be Considered

The following three LOFA scenarios were selected based on the previous transient accident analysis without the 2017-RBM-99 device in the PSAR [9]:

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 a coolant circulation pump (marked as primary pump in Figure 4.3)

The initial failure point for each LOFA is marked as a yellow highlight in Figure 4.3, except for loss of facility electrical power. Similar to LOCA analyses, these three LOFAs have been simulated for each HEU and LEU reference core (Core 3C and 7C, respectively), for each burnup condition (four pairs of elements at different burnups: one BOL, two MOL, and one EOL). For the LEU core, simulations have been conducted with and without considering flow redistribution, as well. Therefore, a total of 36 simulations (12 for HEU and 24 for LEU) have been conducted to cover the LOFA analysis. Simulation and analysis methodologies are identical to those for the LOCAs in Section 4.3.

4.4.2 Results from HEU Core LOFA Analyses

Transient behavior of each LOFA scenarios was simulated with RELAP5. The results for HEU are summarized in Table 4.11.

Table 4.11. HEU Core 3C LOFA Analyses Results without Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Loss of facility electrical power	BOL (0)	113	148 (298)	36.78	2, 2	327 (589)	2, 2
	MOL-1 (73)	110	145 (293)	34.75	2, 2	330 (594)	2, 2
	MOL-2 (81)	109	143 (290)	33.86	2, 2	332 (597)	2, 2
	EOL (150)	112	139 (283)	27.78	22, 3	336 (604)	22, 3
Inadvertent closure of isolation valves	BOL (0)	113	145 (294)	32.19	2, 2	330 (593)	2, 2
	MOL-1 (73)	110	144 (291)	33.21	2, 2	331 (596)	2, 2
	MOL-2 (81)	109	143 (290)	33.69	2, 2	332 (597)	2, 2
	EOL (150)	104	140 (285)	36.46	3, 2	335 (602)	3, 2
Locked rotor in a coolant circulation pump	BOL (0)	131	145 (294)	13.88	1, 3	330 (593)	1, 3
	MOL-1 (73)	125	136 (276)	10.90	1, 3	339 (611)	1, 3
	MOL-2 (81)	123	134 (272)	10.64	2, 3	341 (615)	2, 3
	EOL (150)	118	127 (261)	9.21	2, 3	348 (626)	2, 3

For the reference HEU core with the 2017-RBM-99 upgrade, the most limiting LOFA scenario is the loss of facility electrical power. The maximum fuel temperature for this scenario is 148 °C (298 °F) in

plate 2 of the BOL element, which gives a minimum margin to the fuel temperature safety limit of 327 °C (589 °F) in the same location. The most limiting LOFA scenario for HEU core without the 2017-RBM-99 device was determined to be the loss of facility electrical power, as well, for which the maximum fuel temperature and minimum margin to the fuel temperature safety limit are 148 °C (299 °F) and 327 °C (588 °F), respectively, both in plate 24 of the BOL element. The slight difference in the temperature in Fahrenheit is due to rounding up.

4.4.3 Results from LEU Core LOFA Analyses

Transient behavior of each LOFA scenario in the reference LEU Core 7C was simulated with RELAP5. The results with and without flow redistribution due to burnup are summarized in Table 4.12 and Table 4.13, respectively.

Table 4.12. LEU Core 7C LOFA Analyses Results with Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Loss of facility electrical power	BOL (0)	121	155 (311)	34.42	4, 2	295 (531)	4, 2
	MOL-1 (77)	148	154 (310)	6.25	22, 3	296 (532)	22, 3
	MOL-2 (96)	154	156 (312)	1.35	22, 3	286 (515)	1, 3
	EOL (180)	158	158 (316)	0.00	22, 3	240 (432)	2, 3
Inadvertent closure of isolation valves	BOL (0)	139	149 (301)	10.41	4, 3	301 (541)	4, 3
	MOL-1 (77)	150	150 (302)	0.00	23, 3	300 (540)	23, 3
	MOL-2 (96)	156	156 (312)	0.00	23, 3	284 (512)	1, 3
	EOL (180)	158	158 (316)	0.01	22, 3	246 (443)	2, 3
Locked rotor in a coolant circulation pump	BOL (0)	139	154 (309)	15.08	4, 3	296 (533)	4, 3
	MOL-1 (77)	150	164 (328)	14.22	23, 3	286 (514)	23, 3
	MOL-2 (96)	156	170 (339)	14.71	23, 3	280 (503)	23, 3
	EOL (180)	158	172 (341)	13.95	22, 3	234 (420)	23, 3

For the reference LEU core with the 2017-RBM-99 upgrade, the most limiting LOFA scenario with flow redistribution due to burnup is a locked rotor in a coolant circulation pump for the EOL element (element 4). The maximum fuel temperature for this scenario is 172 °C (341 °F) in fuel plate 22, and the minimum margin to the fuel temperature safety limit is 234 °C (420 °F) in fuel plate 23 of the same element. The most limiting LOFA scenario for a reference LEU core without the 2017-RBM-99 upgrade and with flow redistribution was determined to be a locked rotor in a coolant circulation pump for the EOL element, as well, for which the maximum fuel temperature and minimum margin to the fuel temperature safety limit are 170 °C (338 °F) in plate 22 and 234 °C (420 °F) in plate 23, respectively.

Table 4.13. LEU Core 7C LOFA Analyses Results without Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Loss of facility electrical power	BOL (0)	115	155 (311)	40.41	11, 3	295 (531)	11, 3
	MOL-1 (77)	150	154 (310)	4.31	22, 3	296 (532)	22, 3
	MOL-2 (96)	158	158 (316)	0.00	23, 3	287 (516)	1, 3
	EOL (180)	158	158 (317)	0.00	22, 3	245 (440)	2, 2
Inadvertent closure of isolation valves	BOL (0)	142	149 (299)	6.53	4, 3	301 (543)	4, 3
	MOL-1 (77)	152	153 (307)	0.01	23, 3	298 (536)	23, 3
	MOL-2 (96)	158	158 (316)	0.00	23, 3	285 (513)	1, 3
	EOL (180)	158	158 (317)	0.01	22, 3	246 (442)	23, 3
Locked rotor in a coolant circulation pump	BOL (0)	142	158 (316)	15.91	4, 3	292 (526)	4, 3
	MOL-1 (77)	152	167 (333)	14.70	23, 3	283 (509)	23, 3
	MOL-2 (96)	158	173 (343)	15.12	23, 3	277 (499)	23, 3
	EOL (180)	158	172 (342)	13.87	22, 3	232 (418)	23, 3

For the reference LEU core (7C) with the 2017-RBM-99 upgrade, the most limiting LOFA scenario without considering flow redistribution due to burnup is a locked rotor in a coolant circulation pump. The maximum fuel temperature for this scenario is 173 °C (343 °F) in plate 23 of MOL-2 element (96 MWd burnup) in fuel plate 23, and the minimum margin to the fuel temperature safety limit is 232 °C (418 °F) in fuel plate 23 in the EOL element. The most limiting LOFA scenario for the reference LEU core without the 2017-RBM-99 upgrade and without considering flow redistribution was found to be the locked rotor in a coolant circulation pump for the EOL element, for which the maximum fuel temperature and minimum margin to the fuel temperature safety limit are 171 °C (339 °F) in fuel plate 22 and 232 °C (418 °F) in fuel plate 23, respectively.

4.4.4 Summary of LOFA Analyses

Transient behavior for three LOFA scenarios was analyzed. The most limiting LOFA for a reference HEU core is the loss of facility electrical power. For the reference LEU core the most limiting LOFA is the locked rotor in a coolant circulation pump. For both HEU and LEU cores, the 2017-RBM-99 upgrade does not significantly affect the maximum fuel temperatures or the margin to the fuel temperature safety limits during the LOFAs. For all LOFAs, the safety margins are significantly larger than those of LOCAs. The minimum margins to the fuel temperature safety limit in the LEU core with the 2017-RBM-99 upgrade are 101 °C (182 °F) and 232 °C (418 °F) for the most limiting LOCA and most limiting LOFA, respectively. Therefore, it is reasonable to conclude that the LOCAs are generally more limiting than the LOFAs. For the LOFAs, the margin to the fuel temperature safety limit is significantly large and the potential for fuel damage has not been significantly affected by the 2017-RBM-99 upgrade.

Similar to the LOCA analysis, the set of LOFA analyses was conducted regarding the pair of MOL-1 elements in the reference HEU core to assess the differences between an element with highest maximum local heat flux (element 3) and that with highest maximum hot-stripe heat flux (element 7). The results are presented in Table 4.14. Note that the results for element 3 in Table 4.14 are the same as those presented in Table 4.11. The results confirm that the maximum fuel temperatures for element 3 (highest maximum local heat flux) are equal or greater than those for element 7 (highest hot-stripe heat flux). Likewise, the minimum margins to the fuel temperature safety limit for element 3 are equal or less than those for element 7, thus confirming the modeling approaches used for the LOFA analyses.

Table 4.14. HEU Core 3C LOFA Analyses Results without Flow Redistribution for MOL-1 (73 MWd) Pair.

Accident	Element	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Loss of facility electrical power	3	110	145 (293)	34.75	2, 2	330 (594)	2, 2
	7	111	145 (293)	34.48	2, 2	330 (594)	2, 2
Inadvertent closure of coolant loop isolation valve	3	110	144 (291)	33.21	2, 2	331 (596)	2, 2
	7	111	144 (291)	32.94	2, 2	331 (596)	2, 2
Locked rotor in a coolant circulation pump	3	125	136 (276)	10.90	1, 3	339 (611)	1, 3
	7	124	135 (276)	10.84	1, 3	340 (611)	1, 3

5 CONCLUSIONS

The current MURR HEU fuel element consists of 24 curved fuel plates of uranium-aluminide dispersion fuel. An LEU fuel element for conversion of MURR has been designed by staff members at ANL and MURR as a part of the RC Pillar in the M3 USHPRR LEU conversion program. The preliminary LEU fuel element design has 23 curved fuel plates with a monolithic U-10Mo alloy fuel. FF Pillar efforts are underway to fabricate this preliminary design for MURR. The FQ Pillar will subsequently perform plate and element-level irradiation tests in order to qualify the fuel for use in USHPRR. Both the final fabrication and irradiation data will allow the RC Pillar to finalize fuel element specifications. These final fuel specifications are needed to perform the analysis required for the final SARs to be submitted to the regulator of each USHPRR reactor to allow conversion. Due to the critical safety and operational-related tolerances, final fuel element specifications based on demonstrated adequate fabrication throughput and process capability/yield are required in order to perform SAR analyses that finalizes the assumptions made in the Preliminary Design stage. Similarly, finalized irradiation performance and fuel data will also be required.

It has been shown in a Preliminary SAR (PSAR) [14] that, following conversion to LEU using this element design and a proposed power uprate to 12 MW, MURR will be able to continue to operate safely and reliably, with a slight improvement to experimental performance relative to HEU operations. The PSAR analysis was based on a reactor neutronics model that included experimental facilities and experiment loading that were prototypic for MURR in 2008.

A proposed upgrade to the MURR facility for a reactor-based production of a reliable domestic supply of Mo-99 has been developed through a commercial partnership involving MURR, Nordion, and GA and has been described in a series of proprietary reports. The device, which is labeled as the 2017-RBM-99 upgrade in this report, would be located outside the core driver fuel region. Analysis has shown that the proposed upgrade to the facility affects the three-dimensional power distribution in the core for both HEU and LEU fuel. This will affect the documented margins to safety, as well as could potentially impact the experimental performance of LEU relative to HEU following conversion.

This report evaluates the impact of the 2017-RBM-99 upgrade on the neutronic and thermal-hydraulic behavior of HEU and LEU cores in steady-state operation and transient accident scenarios. The analysis was performed assuming the nominal fuel element dimensions and fuel loading for the current HEU element design and the preliminary LEU element design in the PSAR. Changes to the end channel tolerances for both HEU and LEU unrelated to the upgrade were suggested by the fuel manufacturer and were also included in the analysis, but were not found to affect the calculated safety margins.

Reactor neutronics models based on the current HEU and preliminary LEU element designs, experimental facilities and experiment loading that are typical for current (2018) operations in MURR, and with the 2017-RBM-99 upgrade were developed using similar modeling techniques as used in the PSAR analysis. A fuel cycle simulation found that for reference core loadings of HEU or LEU that are typical for MURR operations with the upgrade, there is a change in the three-dimensional core power distribution relative to prototypic cores that were developed as a part of the PSAR analysis. The power distributions for both HEU and LEU reference cores change due to the combined effects of 1) the shift of the core radial power profile towards the driver elements and fuel plates closest to the fueled cartridges of the 2017-RBM-99 when it is installed and 2) an increase in the core excess reactivity that results in a more deeply inserted critical control blade position that affects the axial power shape.

The MURR Technical Specifications require a minimum shutdown margin of 2% $\Delta k/k$. The shutdown margins for the HEU and LEU cores with the 2017-RBM-99 upgrade were not calculated in this work but are estimated to be 2.22% and 2.04% $\Delta k/k$, respectively, which exceed the Technical Specifications requirement. Note that this estimate is for an extreme case with all fresh fuel and with all four control blades at their maximum burnup. MURR never operates with all four control blades at their maximum burnup and it is anticipated to have all fresh fuel in the core only for a single operating cycle following conversion.

More than 400 experimental performance measures, which were identified by experimentalists at MURR, were evaluated for HEU and LEU reference cores with the 2017-RBM-99 upgrade. The analysis found that the performance for LEU following a power uprate to 12 MW meets or exceeds the performance for HEU operating at 10 MW.

Detailed core power distributions derived from the neutronics models with the 2017-RBM-99 upgrade were used to evaluate the steady-state safety margins for reference cores operating at LSSS conditions following guidance in NUREG-1537 [15,16]. The minimum margin to the onset of flow instability was predicted in the PSAR analysis (without the upgrade) to be 2.11 and 2.10 MW, respectively for HEU and LEU cores. Calculations with the qualified PLTEMP/ANL v.4.3 software in the current analysis predict that the minimum margin to the onset of flow instability for the HEU and LEU cores without the 2017-RBM-99 upgrade are 2.39 and 2.70 MW, respectively. Greater margins are predicted, because of the addition of two hot-channel factors in the analysis that are available with the new version of the software. With the 2017-RBM-99 upgrade, the margins are calculated to be 2.06 and 2.35 MW for HEU and LEU, respectively. Although the margins to flow instability are reduced with the upgrade, there is still adequate safety margin and the margin is greater for LEU relative to HEU following conversion. In all cases with the upgrade, the critical heat flux ratio at the LSSS power is greater than 2.0.

Transient accidents were also modeled and evaluated for the margin to a fuel temperature safety limit based on the fuel blister threshold temperature for HEU aluminide and LEU U-10Mo monolithic fuels. A class of transients known as reactivity insertion accidents (RIA) postulate the unplanned insertion of positive reactivity in cores operating at the rod run-in set point of 115% of full licensed power and coolant flow rate and coolant inlet temperature at LSSS conditions. These conditions are more stringent compared to the conservative limits of the normal operating band for MURR, but are in keeping with the requirements set forth in NUREG-1537 [16] and are consistent with the recent HEU relicensing for MURR. For the most limiting RIA in a reference HEU core with the 2017-RBM-99 upgrade, the margin to the HEU fuel temperature safety limit is more than 250 °C and only 3 °C lower than for a reference core from the PSAR analysis without the upgrade. For the reference LEU core with the 2017-RBM-99 upgrade the minimum margin to the LEU fuel temperature safety limit for the most limiting RIA is 102 °C, which is 7 °C lower than for a reference core from the PSAR analysis.

The same selection of loss-of-coolant accidents (LOCA) and loss-of-flow accidents (LOFA), which were evaluated in the PSAR analysis, were also considered in this work with the 2017-RBM-99 upgrade. The LOCAs and LOFAs were modeled with the core operating at the full licensed power of 10 MW for HEU or 12 MW for LEU and coolant flow rate and coolant inlet temperature at LSSS conditions. These initial conditions are in keeping with the requirements specified in NUREG-1537 [16]. The most limiting accident evaluated is the double-ended cold-leg break LOCA. For the reference HEU core with the 2017-RBM-99 upgrade, the smallest margin to the HEU fuel temperature safety limit was predicted to be 184 °C. Relative to the analysis of this LOCA without the 2017-RBM-99 device, the 2017-RBM-99 upgrade decreases the HEU margin to the fuel temperature safety limit by 45 °C.

For the reference LEU core with the 2017-RBM-99 upgrade, LOCAs were evaluated both with and without coolant flow redistribution due to fuel element burnup in the core. The maximum predicted fuel temperature is 349 °C in a BOL plate and occurs in the double-ended cold-leg break LOCA without coolant flow redistribution. The margin to the fuel temperature safety limit in this case is 101 °C. This is a smaller margin to the fuel temperature safety limit than predicted in the analysis without the 2017-RBM-99 device, which was 118 °C and occurred in an EOL plate in a case with coolant flow redistribution.

Because there is no change to the HEU or LEU nominal fuel element loadings and only a minimal effect on the fuel burnup as a result of the 2017-RBM-99 upgrade, the radiological consequences of the maximum hypothetical accident and fuel handling accident were not evaluated in this work. As found in the PSAR analysis, for any event where a release is considered in the current HEU SAR, the radiological consequences after conversion remain lower than the regulatory limits.

In summary, it has been found that the predicted steady-state margins to flow instability and the critical heat flux ratio with the upgrade are adequate for both HEU and LEU cores. The predicted margins to the onset of flow instability with the 2017-RBM-99 device in both HEU and LEU cores are less than those without the upgrade, but are still greater than 2.0 MW. The minimum steady-state safety margin to the onset of flow instability for LEU with the 2017-RBM-99 upgrade is larger than that for HEU. The critical heat flux ratio at the LSSS power is greater than 2.0 for both HEU and LEU cores with the 2017-RBM-99 upgrade.

For postulated transient accidents, the upgrade causes a decrease in the minimum margin to a temperature safety limit based on the fuel blister threshold temperature. For HEU, there is a 45 °C reduction in the safety margin for the most limiting accident evaluated under conditions specified in NUREG-1537, but the maximum fuel temperatures for all considered HEU scenarios stay below the safety limit by at least 184 °C. For a reference LEU core, the minimum safety margin for the most limiting transient accident with the 2017-RBM-99 upgrade is 101 °C, which is 8 °C less than the margin predicted without the presence of 2017-RBM-99 device. Together with the preliminary blister measurements, results indicate that fuel conversion with a preliminary U-10Mo monolithic fuel element design in cores with the 2017-RBM-99 upgrade will result in no damage to fresh or depleted fuel plates for any credible accident postulated for MURR.

Finally, the preliminary LEU fuel element design has been found to be robust even when a major modification to the core occurs, such as the 2017-RBM-99 upgrade. This provides confidence that current and future changes at MURR that are of a similar, or lesser scale, along with the required power uprate to 12 MW for LEU, will result in adequate safety margins and experimental performance that meets or exceeds the existing performance in MURR with HEU at 10 MW.

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APPENDIX A –Effect of Change in End Channel Tolerances for Preliminary LEU Design on Thermal-Hydraulic Results

The end channel thicknesses and their tolerances for both the HEU and the LEU CD35 elements were defined in Figure 2 and Table 2 of Reference A-1. For convenience, Figure 2 of Reference A-1 is reproduced as Figure A.1, below.

After Reference A-1 was completed, the tolerances on the dimensions C1 and C-L assumed in the thermal-hydraulics analysis were increased for both the HEU and the preliminary LEU elements. The tolerance for C1 was increased from ± 0.007 inches to a new value of ± 0.013 inches. The tolerance for C-L was increased from ± 0.010 inches to a new value of ± 0.013 inches. It is important to note that the larger tolerance of ± 0.013 inches was used in the manufacture of past HEU elements. However, the smaller values were used for the PSAR analysis, because they were considered to be feasible and would potentially lead to improved thermal safety margins for LEU. However, since changes not essential to conversion are to be avoided, a design change to the original CD35 design has been made for manufacturing to be consistent with the HEU manufacturing tolerance range on the end channels, resulting in the preliminary LEU design. The larger tolerance is also preferred because it is expected to improve the element manufacturing yield, and has been found to result in no reduction of the minimum thermal safety margins, as described below.

Figure A.1. Schematic of MURR Fuel Element Showing Thicknesses of End Channels.

As explained in Section 1 of Reference A-1, there is a total of 0.030-inch nominal clearance between the outermost edges of the element rollers and the inner and outer vessel walls. This clearance is needed to facilitate insertion of the element into the reactor pressure vessel without an undue interference between the element and the vessel walls. In the analysis, it is assumed that the element is nominally centered between the inner and outer vessel walls, but can move ± 0.015 inches (half of the 0.030-inch nominal clearance) if the rollers press up against either pressure vessel wall. Thus, the tolerance of each HEU end channel, the dimensions E1 and E-L in Figure A.1, is ± 0.028 inches (± 0.013 inches for dimensions C1 and C-L, and ± 0.015 inches for the clearance between the rollers and the pressure vessel walls).

For the preliminary LEU element design, the nominal reference dimension from the outermost edge of the rollers to the nearest fuel plate surface, C1 and C-L in Figure A.1, is 0.0805 inches. To facilitate the manufacturing and inspection costs of the fuel element, the fuel manufacturer recommended that the drawings should not use inspection dimensions of greater precision than 0.001 inches. In order to accommodate this recommendation, the LEU drawings in Reference A-2 show that C1 is between 0.067 and 0.093 inches and C-L is between 0.068 and 0.094 inches thick. Thus, each has a range of 0.026 inches, which is equivalent to a tolerance ± 0.013 inches. It should be noted that reference dimensions on the drawing, which are for information only, are not measured and do not affect the acceptability of the element per the manufacturing specification. The reference dimensions remain compatible with the design information.

Thus, the LEU analysis was performed with the innermost end channel (E1) at a low-end, i.e. minus-side, tolerance of $(0.0805 - 0.067) + 0.015 = 0.0135 + 0.015 = 0.0285$ inches. The outermost end channel (E-L) has a low-end tolerance of $(0.0805 - 0.068) + 0.015 = 0.0125 + 0.015 = 0.0275$ inches. Only the low-end tolerances are used in the analysis. It was decided to put the smaller low-end tolerance on the outer end channel because it is expected to be more limiting than the inner one.

The results with the new end channel tolerances and without the 2017-RBM-99 upgrade are shown in Table A.1 and Table A.2 for the HEU and LEU cores, respectively. In Table A.1 and Table A.2, the values in red indicate the lowest power.

It should be noted that the counterparts of Table A.1 and Table A.2 (Table 16 and 18 of Reference A-1) were obtained with a beta version of the PLTEMP/ANL v.4.2 software. The results in Table A.1 and Table A.2 were produced with PLTEMP/ANL v.4.3. PLTEMP/ANL v.4.3 includes two new hot channel factors (f_{dhtd} and f_{velc}) that the previous versions of the software did not have. Therefore, in order to provide consistency in the calculations performed here relative to those performed in Reference A-1 and to observe the effect of the change in the end channel tolerances, the cases in Table A.1 and Table A.2 were modeled with PLTEMP/ANL v.4.3 but without utilizing the two new hot channel factors. All PLTEMP/ANL calculations in Section 3 were conducted with PLTEMP/ANL v.4.3 with the two new hot channel factors. More discussions on the additional hot channel factors can be found in References A-1 and A-3.

Table A.1. HEU Flow Instability Results with New End Channel Tolerances.

Typical Experiments in Center Test Hole				No Experiments in Center Test Hole			
Case	Reactor Power, MW	Element	Channel	Case	Reactor Power, MW	Element	Channel
1A	16.15	2	2	1B	15.46	5	2
1A1	15.46	7	2	1B1	14.83	7	2
1A2	15.61	1	2	1B2	15.00	1	2
2A	16.79	5	2	2B	16.16	5	2
2A1	16.26	7	2	2B1	15.66	7	2
2A2	16.44	1	2	2B2	15.80	1	2
3A	15.60	5	2	3B	15.04	5	2
3A1	15.58	1	2	3B1	14.84	1	2
3A2	15.26	1	2	3B2	14.61	1	2
4A	16.40	5	2	4B	15.82	1	2
4A1	16.40	1	2	4B1	15.76	1	2
4A2	16.32	1	2	4B2	15.74	1	2
Minimum Margin Compared to the 12.5 MW LSSS Power							
Ratio				14.61/12.5 = 1.169			
MW				14.61 – 12.5 = 2.11			

Table A.2. LEU Flow Instability Results with New End Channel Tolerances.

Typical Experiments in Center Test Hole				No Experiments in Center Test Hole			
Case	Reactor Power, MW	Element	Channel	Case	Reactor Power, MW	Element	Channel
5A	19.63	5	4	5B	19.18	4	3
5A1	18.91	8	4	5B1	18.32	7	4
5A2	18.86	1	4	5B2	18.56	2	4
6A	20.57	2	4	6B	19.94	3	1
6A1	18.79	6	23	6B1	18.73	6	23
6A2	19.05	8	23	6B2	18.94	8	23
7A	18.58	8	4	7B	18.17	4	4
7A1	17.99	6	23	7B1	17.65	8	4
7A2	17.95	8	23	7B2	17.58	8	4
8A	17.87	8	23	8B	17.56	8	23
8A1	17.10	8	23	8B1	17.37	6	23
8A2	17.14	8	23	8B2	17.25	8	23
Minimum Margin Compared to the 15.0 MW LSSS Power							
Ratio				17.10/15.0 = 1.140			
MW				17.10 – 15.0 = 2.10			

The Table A.1 results for the core states without the 2017-RBM-99 upgrade but with the new end channel tolerances were compared with those for their counterparts in Table 16 of Reference A-1 for

the PSAR analysis. No differences were observed. Although the end channel tolerances are smaller in the Reference A-1 analysis, the limiting channel in all cases is channel 2, which is not an end channel. The minimum margin to flow instability for all of the reference HEU core states evaluated without the 2017-RBM-99 upgrade, which is 2.11 MW above the LSSS power, is the same with the new, larger end channel tolerances as with the assumed smaller HEU end channel tolerances found in the PSAR analysis.

For all 24 cores states in Table A.1, the critical heat flux ratio (CHFR) is greater than 2.0 at the flow instability power shown. This is also true in Table 16 of Reference A-1.

The results for the LEU core states without the 2017-RBM-99 upgrade but with the new end channel tolerance in Table A.2 were compared with their counterparts in Table 18 of Reference A-1 for the PSAR analysis. The reactor powers for Cores 7A, 8A2, 5B, and 6B are shown in green in Table A.2 to indicate a difference from PSAR results. The differences are less than 0.01 MW for Cores 7A, 8A2, and 5B, and can be attributed to round-off since the limiting channel has not changed and is not an end channel. For Core 6B the new limiting location is an end channel and the flow instability power is 0.05 MW lower than shown for the same core in Table 18 of Reference A-1 due to the new larger end channel tolerance. However, this small difference is of no consequence since the limiting core state is Core 8A1 and has a flow instability power that is more than 2.8 MW lower than Core 6B. The minimum margin to flow instability for all of the reference LEU core states evaluated, which is 2.10 MW above the LSSS power, is the same with the new, larger end channel tolerances as with the smaller end channel tolerances found in the PSAR analysis.

For all 24 cores states in Table A.2, the critical heat flux ratio (CHFR) is greater than 2.0 at the flow instability power shown. This is also true in Table 18 of Reference A-1.

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APPENDIX B – Consideration of Lateral Heat Conduction Phenomena in MURR Fuel Plates Using CFD-COMSOL

B.1 Introduction

The lateral heat conduction in MURR fuel plates is considered using a three dimensional computational fluid dynamics (CFD) model in COMSOL 5.3a [B-1] to determine a representative hot-stripe width that can be conservatively used for one-dimensional (1D) steady-state thermal hydraulic analysis. This follows the method documented by Brookhaven National Laboratory in 2010 [B-2].

Previous analysis of the lateral heat conduction with StarCD for a MURR LEU feasibility study design (FSD) [B-3] was reported in Reference B-4. This modeling and analysis are described in Section B.2. Data needed for a COMSOL model of the preliminary LEU fuel element is provided in Section B.3. To verify the COMSOL simulation, a model of an FSD coolant channel was built with the same specifications as the previous analysis with StarCD. A comparison of the results from COMSOL and StarCD is provided in Section B.4. In Section B.5, parameters that may affect the lateral heat conduction are evaluated. The parameters studied in the previous StarCD analysis for the FSD design were how the cladding material and any contact gap located between fuel plate and side plate affect lateral heat conduction. This analysis is repeated with the COMSOL FSD model. The parameteric study was expanded with the COMSOL model to assess the effects of different thermo-physical property data assumptions on the lateral heat conduction. The FSD analysis considered only the case of a fresh fuel element for evaluating the lateral heat conduction. Since MURR is typically loaded with multiple elements with a range of burnups, a base COMSOL model developed for the preliminary LEU fuel element is used to study how fuel element burnup may effect the lateral heat conduction. Finally, detailed results presented in Section B.6 show that a hot-stripe width up to 7 mm could be conservatively used for 1D steady-state thermal-hydraulics analysis of all plates in the preliminary LEU fuel element for elements at all burnups relevant to MURR.

B.2 Previous StarCD Study of FSD Fuel Element

A three-dimensional heat conduction analysis of MURR LEU fuel plates was conducted in 2011 using the StarCD code [B-4]. The purpose of the analysis was to assess the impact of lateral heat conduction from the fuel plates to the side plates of the element, thus reducing the convective heat transfer into the coolant channel. At the time, PLTEMP/ANL did not have the capability to model lateral heat conduction. Instead, a “hot-stripe” without lateral heat conduction or coolant channel mixing was modeled for steady-state flow instability and critical heat flux analysis. The three-dimensional heat conduction analysis in StarCD was used to determine the width of the hot-stripe that conservatively estimated the convective heat transfer into the limiting coolant channel(s).

The StarCD analysis modeled a single coolant channel with two half-plates on either side of the channel. The analysis was performed for the so-called Feasibility Study Design (FSD) [B-3]. In that work, coolant channel 4, which is between fuel plates 3 and 4, was selected for analysis because of the relatively high heat flux for these two plates which bound the same coolant channel. Figure B.1 shows the details of two adjacent MURR fuel elements in the core and the dimensional modeling. (Note that the image is for a 23-plate element design, but the principles are the same regardless of the number of fuel plates in the element.) When loaded in the reactor core, there is a 0.040-inch spacing between adjacent fuel elements. The x-y origin for the radial dimensions in the fuel element

design is formed by the vertex of two lines that run halfway between adjacent elements, as shown in red in the figure. For the StarCD model, the origin was shifted 0.020 inches to line up with the outside edge of the side plate, as shown by the blue line in Figure B.1.

Figure B.1. Drawing of Adjacent MURR Fuel Elements and Modeling in StarCD.

The angular position of the inside edge of the side plate (where the fuel plates are swaged into the side plate) is $\alpha_{sp,n}$, and is angle from the model origin to a point that is in the middle of coolant channel n between two adjacent fuel plates. Likewise, the angular position of the start of the fuel meat, $\alpha_{fuel,n}$, is from the model origin to the center of the coolant channel, and traversing into the coolant channel the distance of the unfueled width of the fuel plate that extends into the coolant channel. The formulas for calculating these angular positions and design dimensions are shown in Figure B.1, and the values are summarized in Table B.1 for the FSD element design.

The MURR fuel plates are 25.5 inches long with a 24.0-inch fuel meat length. Nominally, there is 0.75 inches of unfueled plate above and below the fuel meat. The “z” origin for the StarCD model is the bottom of the fuel plate. In the StarCD model, the fuel was modeled as starting at 19.728 mm (0.7767 inches) from the bottom of the fuel plate. Although the unfueled length of 0.75 inches was not used in the StarCD model (the prior work), this should have little effect on the results.

An MCNP model of the MURR core loaded with the FSD element design was used to calculate a detailed three-dimensional power distribution. Reference cores loaded with all fresh elements and elements with a mixture of burnups were evaluated. For the StarCD analysis, a reference core identified as Core 7B was selected. This core has two fresh elements and six other elements with a range of burnups. The center test hole was modeled as empty, which increases the power peaking in plates 3 and 4 relative to a core with typical samples in the test hole.

Table B.1. Design Parameters for the FSD Fuel Element Design.

Plate/ Channel (n)	Fuel Meat Thickness (mm)	AA6061 Cladding Thickness (mm)	Zr Interlayer Thickness (mm)	Plate Width (mm)	Fuel Meat Width (mm)	Coolant Channel Thickness (mm)	R_n (mm)	$\alpha_{sp,n}$ (degrees)	$\alpha_{fuel,n}$ (degrees)
1	0.2286	0.4572	0.0254	50.32	42.95	2.413	70.599		
2	0.3048	0.2794	0.0254	53.02	45.66	2.337	74.041		
3	0.4572	0.2032	0.0254	55.62	48.25	2.337	77.343		
4	0.4572	0.2032	0.0254	58.21	50.84	2.337	80.645	2.7645	4.0565
5	0.4572	0.2032	0.0254	60.80	53.44	2.337	83.947		
18	0.4572	0.2032	0.0254	94.52	87.15	2.337	126.873		
19	0.4572	0.2032	0.0254	97.11	89.74	2.337	130.175		
20	0.4572	0.2032	0.0254	99.70	92.34	2.337	133.477		
21	0.4572	0.2032	0.0254	102.30	94.93	2.337	136.779		
22	0.4572	0.2032	0.0254	104.89	97.52	2.337	140.081		
23	0.4572	0.2032	0.0254	107.48	100.12	2.337	143.383		
24	0.4318	0.3556	0.0254	110.19	102.82	2.337	146.825		

Energy deposition tallies (f7 tallies) were used to calculate the power density in each fuel plate. The fuel meat of each plate was divided into 24 axial tallies of one inch each. Nine tallies were set up to calculate the azimuthal power distribution. The arc width of the fuel meat was divided into equally-sized tally zones within each plate. Figure B.2 shows the power density for plate 3 in the fresh fuel element in core position 1 in Core 7B.

Figure B.2. Power Density Profile in FSD Plate 3 in Element 1 in Core 7B.

The power density profiles in plates 3 and 4 were correlated as a function of azimuthal and axial position for use in the StarCD analysis. The axially-averaged local-to-average azimuthal power density

profile was fit using a 4th order polynomial fit. This is shown in Figure B.3 for the half of the arc width of the fuel plate that was modeled in StarCD. The power density profile is plotted as a function of angular position, with 0.0 degrees being positioned at the outside edge of the fuel element side plate, as described above. The dashed line shows the location of the start of the fuel meat in the model, which is 4.057 degrees. The azimuthally-averaged axial power density profiles for plates 3 and 4 are shown in Figure B.4. A 6th-order polynomial fit was used to capture the axial shape.

Figure B.3. Axially-Averaged Local-to-Average Azimuthal Power Density Profile for Channel 4 of FSD Element in Position 1 in Core 7B.

Figure B.4. Azimuthally-Averaged Axial Power Density Profile for Channel 4 of FSD Element in Position 1 in Core 7B.

The azimuthal correlation fits for the local-to-average power density for plates 3 and 4 are provided in Equations 1 and 2, respectively. The axial correlation fits for plates 3 and 4 are provided in Equations 3 and 4, respectively.

$$P03 = 3.2456E-06\phi^4 - 2.9336E-04\phi^3 + 9.9303E-03\phi^2 - 1.4912E-01\phi + 1.7670E+00 \quad (1)$$

$$P04 = 4.2641E-06\phi^4 - 3.8269E-04\phi^3 + 1.2824E-02\phi^2 - 1.9022E-01\phi + 1.9657E+00 \quad (2)$$

where P03 and P04 are the axially-averaged local-to-average power density for plates 3 and 4, respectively, and ϕ is the angular position in degrees.

$$PD3 = 6.24E+12z^5 - 1.32E+13z^4 + 1.12E+13z^3 - 4.71E+12z^2 + 9.36E+11z - 6.30E+10z + 6.06E+09 \quad (3)$$

$$PD4 = 6.02E+12z^5 - 1.27E+13z^4 + 1.07E+13z^3 - 4.48E+12z^2 + 8.91E+11z - 6.31E+10z + 5.26E+09 \quad (4)$$

where PD3 and PD4 are the azimuthally-averaged power density in W/m³ for plates 3 and 4, respectively, and z is the axial position from the bottom of the fuel plate in m.

The azimuthal and axial profile correlations can be multiplied together to obtain the power density in the fuel plates, as shown in Equations 5 and 6.

$$\text{Power density } (3,\phi,z) = P03 (\phi) * PD3 (z) \quad (5)$$

$$\text{Power density } (4,\phi,z) = P04 (\phi) * PD4 (z) \quad (6)$$

The RMS error of the correlation fit of the power density is 2.6% and 3.4% in plates 3 and 4, respectively.

An additional case was run with MCNP with narrower azimuthal tally stripes near the edge of the fuel meat. A comparison of the axially-averaged azimuthal power density profile for plate 3 in element position X1 of Core 7B with this more detailed tally striping and that used to generate the above correlation is provided in Figure B.5. This was done to assess the accuracy of the correlation to capture the shape of the power density increase near the edge of the fuel meat. Near the edge of the fuel meat, the correlation under predicts the axially-averaged power density calculated by MCNP in the detailed stripe model by 6.2%. For all other azimuthal positions, the correlation over predicts the axially-averaged power density by less than 4%. The larger under prediction at locations near the edge of the fuel meat is not considered problematic, since nearly all of the heat generated in this position will be conducted laterally towards the side plate, so that the convective heat flux into the coolant channel will be very small.

Figure B.6 shows the results of a heat flux correlation along the plate width derived from MCNP5 analysis and assuming no lateral heat conduction. This heat flux profile is compared to the results from the 3D StarCD analysis for the convective heat flux into coolant channel 4 from plate 3. Three cases were evaluated in StarCD. In the first case, the Zr interlayer is treated as a part of the AA6061 cladding. In the second case, an explicit layer of 1.25 mil of Zr (the nominal Zr thickness is 1.0 mil) is modeled. In the final case, there is a small gap modeled between the side plate and the fuel plate to assess the impact if there is imperfect contact between the side plate and fuel plate. The results are practically the same regardless of how the components of the plate are modeled.

Figure B.5. Axially-Averaged Azimuthal Power Density Profile in Plate 3 of FSD Element in the Element 1 Position in Core 7B.

Figure B.6. Heat Flux from Plate 3 into Coolant Channel 4 in Element 1 in Core 7B at Location of Axial Peak for Heat Flux (0.236 m from bottom of plate).

The results also clearly show that the lateral heat conduction near the edge of the fuel meat has a significant effect on the convective heat flux into the coolant channel. In this case, the convective heat transfer decreases rapidly starting at about 3 mm from the edge of the fuel. It was found that averaging the azimuthal heat flux profile predicted from the MCNP5 analysis over a 5 mm stripe near the edge of the fuel would result in a conservative estimate of the heat flux conducted into the coolant channel for use in the 1D PLTEMP-ANL analysis of the FSD element.

B.3 Model of Preliminary LEU Fuel Element Design

A hot-stripe width of 5 mm, calculated based on lateral heat conduction analysis of the FSD element described in the previous section, was utilized for 1D steady-state thermal-hydraulic analyses of reference cores loaded with the LEU CD35 element reported in References B-5, and accident analyses reported in Reference B-6. This CD35 element design has become the preliminary LEU fuel element design after a change in the end channel tolerances (see Appendix A). It is necessary to confirm through analysis that the hot-stripe width of 5.0 mm is conservative for all fuel plates in the preliminary LEU fuel element design and for a range of element burnups that are relevant for MURR.

This section presents data for the preparation of a COMSOL model of the preliminary LEU fuel element design. A core with a mixture of fuel elements at different burnups is typical for MURR operations. A xenon-free core (BOC) was selected for analysis because this core state tends to have greater power peaking (due to the more deeply inserted control blades), which was thought to produce more conservative results for the lateral heat conduction analysis. Power by element and plate for the BOC reference Core 7A described in Section 2.3 are provided in Table B.2 and Table B.3, respectively, for BOL, MOL, and EOL elements. Plots of the axial and azimuthal power density profiles similar to those generated for the FSD design (Section B.2) are shown in Figure B.7 to Figure B.24. The power density correlations were obtained by polynomial fitting and are provided after each corresponding figure. Geometry data and other relevant operational parameters are provided in Table B.4 and Table B.5.

Table B.2. LEU Element Power in Core 7A (MW).

BOL Element (0 MWd)	MOL Element (96 MWd)	EOL Element (170 MWd)
1.580	1.491	1.438
	-5.6%	-9.0%

Table B.3. Plate Power by Element in Core 7A (W).

Plate 3			Plate 4		
BOL Element (0 MWd)	MOL Element (96 MWd)	EOL Element (170 MWd)	BOL Element (0 MWd)	MOL Element (96 MWd)	EOL Element (170 MWd)
6.29E+04	5.80E+04	5.61E+04	6.47E+04	6.15E+04	6.01E+04
	-7.8%	-10.8%		-5.1%	-7.2%
Plate 22			Plate 23		
BOL Element (0 MWd)	MOL Element (96 MWd)	EOL Element (170 MWd)	BOL Element (0 MWd)	MOL Element (96 MWd)	EOL Element (170 MWd)
1.21E+05	1.16E+05	1.11E+05	1.47E+05	1.33E+05	1.22E+05
	-4.5%	-8.1%		-9.7%	-17.4%

Figure B.7. Power Density Profile for Plate 3 of BOL Element (0 MWd), Core 7A.

Figure B.8. Azimuthally-Averaged Axial Power Density Profile in Plates 3 and 4 of Channel 4 of BOL Element (0 MWd) in Core 7A.

$$\begin{aligned} \text{AX3} = & 3.92220\text{E}+12\text{xZ}^{**6} - 8.56316\text{E}+12\text{xZ}^{**5} + 7.69934\text{E}+12\text{xZ}^{**4} - 3.49534\text{E}+12\text{xZ}^{**3} + \\ & 7.43800\text{E}+11\text{xZ}^{**2} - 5.27240\text{E}+10\text{xZ} + 5.98655\text{E}+09 \end{aligned}$$

$$\begin{aligned} \text{AX4} = & 4.99255\text{E}+12\text{xZ}^{**6} - 1.06511\text{E}+13\text{xZ}^{**5} + 9.15487\text{E}+12\text{xZ}^{**4} - 3.93058\text{E}+12\text{xZ}^{**3} + \\ & 8.05886\text{E}+11\text{xZ}^{**2} - 5.97987\text{E}+10\text{xZ} + 5.18327\text{E}+09 \end{aligned}$$

Figure B.9. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 3 and 4 of Channel 4 of BOL Element (0 MWd) in Core 7A.

$$AZ3 = 3.0942E-06xA^{**4} - 2.6900E-04xA^{**3} + 8.8909E-03xA^{**2} - 1.3183E-01xA + 1.6683E+00$$

$$AZ4 = 3.7141E-06xA^{**4} - 3.3221E-04xA^{**3} + 1.1100E-02xA^{**2} - 1.6486E-01xA + 1.8429E+00$$

Figure B.10. Power Density Profile for Plate 22 of BOL Element (0 MWd), Core 7A.

Figure B.11. Azimuthally-Averaged Axial Power Density Profile in Plates 22 and 23 of Channel 23 of BOL Element (0 MWd) of Core 7A.

$$AX22 = -6.74389E+12xZ^{**6} + 1.17570E+13xZ^{**5} - 6.96943E+12xZ^{**4} + 1.53620E+12xZ^{**3} - 1.00011E+11xZ^{**2} + 7.96609E+09xZ + 3.93697E+09$$

$$AX23 = -1.36736E+13xZ^{**6} + 2.47618E+13xZ^{**5} - 1.59951E+13xZ^{**4} + 4.40655E+12xZ^{**3} - 5.44825E+11xZ^{**2} + 4.36455E+10xZ + 4.60105E+09$$

Figure B.12. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 22 and 23 of Channel 23 of BOL Element (0 MWd) in Core 7A.

$$AZ22 = 3.3854E-06xA^{**4} - 2.9805E-04xA^{**3} + 9.2254E-03xA^{**2} - 1.1763E-01xA + 1.4796E+00$$

$$AZ23 = 1.3667E-06xA^{**4} - 1.2372E-04xA^{**3} + 3.9929E-03xA^{**2} - 5.4168E-02xA + 1.2385E+00$$

Figure B.13. Power Density Profile for Plate 3 of MOL Element (96 MWd), Core 7A.

Figure B.14. Azimuthally-Averaged Axial Power Density Profile in Plates 3 and 4 of Channel 4 of MOL Element (96 MWd) in Core 7A.

$$\text{AX3} = 1.23125\text{E}+12\text{xZ}^{**6} - 3.77223\text{E}+12\text{xZ}^{**5} + 4.49746\text{E}+12\text{xZ}^{**4} - 2.51555\text{E}+12\text{xZ}^{**3} + 6.18974\text{E}+11\text{xZ}^{**2} - 5.05276\text{E}+10\text{xZ} + 5.82992\text{E}+09$$

$$\text{AX4} = 2.23090\text{E}+12\text{xZ}^{**6} - 5.41709\text{E}+12\text{xZ}^{**5} + 5.37587\text{E}+12\text{xZ}^{**4} - 2.64518\text{E}+12\text{xZ}^{**3} + 6.04345\text{E}+11\text{xZ}^{**2} - 4.84703\text{E}+10\text{xZ} + 4.86989\text{E}+09$$

Figure B.15. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 3 and 4 of Channel 4 of MOL Element (96 MWd) in Core 7A.

$$AZ3 = 2.6280E-06xA^{**4} - 2.2548E-04xA^{**3} + 7.4122E-03xA^{**2} - 1.1097E-01xA + 1.5754E+00$$

$$AZ4 = 3.1964E-06xA^{**4} - 2.8669E-04xA^{**3} + 9.6800E-03xA^{**2} - 1.4671E-01xA + 1.7702E+00$$

Figure B.16. Power Density Profile for Plate 22 of MOL Element (96 MWd), Core 7A.

Figure B.17. Azimuthally-Averaged Axial Power Density Profile in Plates 22 and 23 of Channel 23 of MOL Element (96 MWd) in Core 7A.

$$\begin{aligned}
 AX22 &= -9.21665E+12xZ^{**6} + 1.64856E+13xZ^{**5} - 1.03828E+13xZ^{**4} + 2.69461E+12xZ^{**3} - \\
 &\quad 2.88375E+11xZ^{**2} + 2.13429E+10xZ + 3.46409E+09 \\
 AX23 &= -1.64656E+13xZ^{**6} + 3.01714E+13xZ^{**5} - 1.99961E+13xZ^{**4} + 5.81589E+12xZ^{**3} - \\
 &\quad 7.81579E+11xZ^{**2} + 5.95825E+10xZ + 3.80334E+09
 \end{aligned}$$

Figure B.18. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 3 and 4 of Channel 4 of MOL Element (96 MWd) in Core 7A.

$$AZ22 = 2.7933E-06xA^{**4} - 2.4450E-04xA^{**3} + 7.5363E-03xA^{**2} - 9.5542E-02xA + 1.3805E+00$$

$$AZ23 = 1.0901E-06xA^{**4} - 9.8148E-05xA^{**3} + 3.1582E-03xA^{**2} - 4.2129E-02xA + 1.1704E+00$$

Figure B.19. Power Density Profile for Plate 3 of EOL Element (170 MWd), Core 7A.

Figure B.20. Azimuthally-Averaged Axial Power Density Profile in Plates 3 and 4 of Channel 4 of EOL Element (170 MWd) in Core 7A.

$$\begin{aligned}
 \text{AX3} &= 4.02503\text{E}+12\text{xZ}^{**6} - 8.67644\text{E}+12\text{xZ}^{**5} + 7.59944\text{E}+12\text{xZ}^{**4} - 3.34281\text{E}+12\text{xZ}^{**3} + \\
 &\quad 6.97365\text{E}+11\text{xZ}^{**2} - 5.12654\text{E}+10\text{xZ} + 5.68137\text{E}+09 \\
 \text{AX4} &= 3.93606\text{E}+12\text{xZ}^{**6} - 8.43032\text{E}+12\text{xZ}^{**5} + 7.31807\text{E}+12\text{xZ}^{**4} - 3.19002\text{E}+12\text{xZ}^{**3} + \\
 &\quad 6.65248\text{E}+11\text{xZ}^{**2} - 5.04243\text{E}+10\text{xZ} + 4.82211\text{E}+09
 \end{aligned}$$

Figure B.21. Axially-Averaged Local-to-Average Azimuthal Power Density Profile in Plates 3 and 4 of Channel 4 of EOL Element (170 MWd) in Core 7A.

$$AZ3 = 2.2822E-06xA^{**4} - 1.9829E-04xA^{**3} + 6.6281E-03xA^{**2} - 1.0078E-01xA + 1.5276E+00$$

$$AZ4 = 2.8891E-06xA^{**4} - 2.6286E-04xA^{**3} + 9.0185E-03xA^{**2} - 1.3789E-01xA + 1.7194E+00$$

Figure B.22. Power density Profile for Plate 22 of EOL Element (170 MWd) in Core 7A.

Figure B.23. Azimuthally-Averaged Axial Power Density Profile in Plates 22 and 23 of Channel 23 of EOL Element (170 MWd) in Core 7A.

$$\begin{aligned}
 AX22 &= -8.76660E+12xZ^{**6} + 1.57695E+13xZ^{**5} - 1.00271E+13xZ^{**4} + 2.65598E+12xZ^{**3} - \\
 &\quad 2.97696E+11xZ^{**2} + 2.16745E+10xZ + 3.48760E+09 \\
 AX23 &= -1.50299E+13xZ^{**6} + 2.77576E+13xZ^{**5} - 1.86163E+13xZ^{**4} + 5.52962E+12xZ^{**3} - \\
 &\quad 7.69476E+11xZ^{**2} + 5.83219E+10xZ + 3.89208E+09
 \end{aligned}$$

Figure B.24. Axially-Averaged Local-to-Average Azimuthal Power Density Distribution in Plates 22 and 23 of Channel 23 of EOL Element (170 MWd) in Core 7A.

$$\begin{aligned}
 AZ22 &= 2.8338E-06xA^{**4} - 2.4938E-04xA^{**3} + 7.7161E-03xA^{**2} - 9.7422E-02xA + 1.3794E+00 \\
 AZ23 &= 1.0296E-06xA^{**4} - 9.5571E-05xA^{**3} + 3.1941E-03xA^{**2} - 4.4188E-02xA + 1.1835E+00
 \end{aligned}$$

Table B.4. Design Parameters for the Preliminary LEU Fuel Element Design.

Plate/ Channel	Fuel Thickness (mm)	AA6061 Cladding Thickness (mm)	Zr Interlayer Thickness (mm)	Plate Width (mm)	Fuel Width (mm)	Coolant Channel Thickness (mm)	R _n (mm)	$\alpha_{\text{fuel},n}$ (degrees)
3	0.4064	0.3302	0.0254	55.75	48.38	2.362	77.508	
4	0.5080	0.2794	0.0254	58.48	51.11	2.362	80.988	4.0434
22	0.5080	0.2794	0.0254	107.39	100.03	2.362	143.269	
23	0.4318	0.3810	0.0254	110.18	102.81	2.362	146.749	2.2085

Table B.5. Fuel and Coolant Channel Properties for Preliminary LEU Fuel Element.

	BOL Element (0 MWd)	MOL Element (96 MWd)	EOL Element (170 MWd)
Ratio of thermal conductivity relative to fresh fuel	1.00	0.75	0.65
Coolant channel gap (mm)	2.36	2.25	2.16
Oxide thickness on surface of fuel plate (mm) [Oxide thermal conductivity assumed 2.25 W/m-K]	0.00	0.0160	0.0236

B.4 COMSOL Model of FSD

B.4.1 Description of Model

A COMSOL model was first developed for the FSD design in order to compare results of the COMSOL analysis against the previous StarCD results [B-4] as a means of model validation. The COMSOL model shown in Figure B.25 represents half of the channel 4 of the FSD fuel assembly. The same modeling approach was used for the StarCD analysis. Beside the geometry data and power distribution given in Section B.2, other input parameters including the flow conditions and material thermal conductivity are taken from Reference B-4, and are presented in Table B.6. The k- ϵ turbulent model was used in the COMSOL analysis.

Figure B.25. COMSOL Model Simulating Channel 4 of FSD Fuel Assembly.

Table B.6. Flow Condition and Material Thermal Conductivity of FSD.

Input parameters		Value
Inlet flow rate, m/s		11.19
Inlet Temperature, °C		68.33
Reference pressure for water properties, MPa		0.44
Thermal conductivity, W/m-K	Fuel	10.0
	AA6061	180.0
	Zirconium	19.0

It should be noted that the correlation of power density versus location were developed for the StarCD geometry, which has a different coordinate system compared to that of COMSOL geometry. As shown in Figure B.26 the origin of COMSOL model is located at point C, while the origin of StarCD model is located at point O. Hence, a coordinate conversion was performed to use the previously developed correlations in the COMSOL models.

Figure B.26. Coordinate Systems Used for Power Density Correlations.

A brief discussion about the coolant inlet conditions in Table B.6 is needed. For typical operations, the coolant inlet temperature is 48.9 °C (120 °F) and the coolant flow rate through the core is 3,750 gpm. In the analysis with StarCD performed for the FSD in Reference B-4, the MURR HEU LSSS conditions of 68.3 °C (155 °F) for the coolant inlet temperature and 3,200 gallons per minute for the whole core volumetric coolant flow rate were assumed.² The LSSS conditions were chosen for the StarCD analysis because they represent the most limiting operating conditions, but no analysis was performed in Reference B-4 to assess how these coolant conditions affect lateral heat conduction relative to typical operating parameters.

A further note is needed regarding the coolant velocity in Table B.6. Reference B-4 reported that the mass flow rate through the FSD channel 4 is 0.68 kg/s, which corresponds to the whole core volumetric flow rate of 3,200 gpm. When preparing the initial COMSOL model inputs for the present work, the mass flow rate in Reference B-4 was interpreted as applying to the half-channel width which was modeled, resulting in the calculated coolant velocity through the channel of 11.19 m/s.

² Subsequent to the work performed in Reference B-4, and based in part on the hot-stripe width of 5 mm derived from that work, the LSSS conditions for the FSD and (CD35) LEU operations was set to 62.8 °C (145 °F) for the coolant inlet temperature and 3,300 gpm for the whole core flow rate. This adjustment to the MURR Technical Specifications was based on the results of the steady-state thermal-hydraulics analysis performed in Reference B-5.

This is about 60% larger than MURR's typical coolant flow velocity through the channels of about 7 m/s, owing to the use of the incorrect channel cross-sectional area when converting from the coolant mass flow rate to velocity. The impact of using the elevated inlet coolant temperature and coolant velocity in the COMSOL analysis on the analysis of the lateral heat conduction is provided in Section B.5.3.

B.4.2 Comparisons of COMSOL and StarCD Simulations

A verification of the COMSOL model used for 3D lateral heat conduction analysis was performed by comparing the COMSOL results with those from the previous StarCD analysis for the FSD [B-4]. The same dimensions for the fuel zone, zirconium thickness, and cladding as in the previous StarCD analysis were utilized in the COMSOL model. A comparison of the results from COMSOL to the StarCD results is provided in Figure B.27 and Table B.7. The COMSOL model predicts a maximum surface heat flux that is 2.2% greater than that predicted by StarCD. The location of the maximum heat flux from the edge of the fuel is 6.7% less in the COMSOL results. Understanding that the two codes use different calculation methods and the models have different meshing systems, the differences between the two simulations are considered acceptable. Furthermore, given that the COMSOL results are more conservative, the COMSOL model is considered suitable for determination of the appropriate hot-stripe width for the preliminary LEU design.

Figure B.27. Normal Total Surface Heat Flux – FSD Plate 3.

Table B.7. Comparison between COMSOL and StarCD.

Code	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
COMSOL	192.62	2.64	25.25
StarCD	188.48	2.83	23.56

B.5 Consideration of Parameters that Affect Lateral Heat Conduction

B.5.1 Effects of Cladding Configuration and Contact Gap

Three test configurations that were evaluated for the FSD design were repeated with the COMSOL model of this design to confirm the previous conclusions on the effects of modeling the fuel plate fabrication:

- A simplified plate model with LEU fuel and the zirconium interlayer replaced by AA6061 cladding.
- A model with LEU fuel and separate AA6061 and zirconium layers in the fuel plate.
- A model with LEU fuel, separate AA6061 and zirconium layers in the fuel plate, and the presence of a 0.0127 mm contact gap between the fuel plate and AA6061 side plate (Figure B.28).

Figure B.28. Channel and Fuel Plate Geometry.

The results from the COMSOL analysis for plate 3 of the FSD are presented in Figure B.29. Table B.8 shows that modeling only the AA6061 cladding in the fuel plate (treating the 0.0254 mm Zr interlayer as AA6061) is non-conservative (the maximum convective heat flux is lower in this case), but that the effect of the contact gap between the fuel plate and side plate on the heat flux profiles is negligible. Consequently, modeling the AA6061 cladding and zirconium interlayer as distinct components of the fuel plate, but without the contact gap, was selected for all subsequent analysis.

Figure B.29. Normal Total Surface Heat Flux – FSD Plate 3.

Table B.8. Comparison between Different Configuration of Cladding and Contact Gap.

Configuration	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
Al only	191.53	2.77	24.50
Al+Zr	192.62	2.64	25.25
Al+Zr & Gap	192.63	2.64	24.50

B.5.2 Effects of Thermo-Physical Properties

As presented in Table B.9, three sets of thermal conductivity for the fuel, AA6061, and zirconium have been used for MURR thermal-hydraulics analysis as the analysis has matured. The COMSOL model for channel 4 of the preliminary LEU fuel element design was used to determine how these assumptions for material properties affect the calculated maximum surface heat flux and location relative to the edge of the fuel.

Table B.9. Material Thermal Conductivity Assumed in Analysis (W/m-K).

Material	Case		
	A (Reference B-4)	B (Reference B-5)	C ¹ (Reference B-6)
Fuel	10.0	16.2	$14.821 + 0.0309 \times T$
AA6061	180.0	150.0	$147.8 + 0.1792 \times T - 2.616E-04 \times T^2$
Zirconium	19.0	19.0	$8.8527 + 0.007082 \times T + 2.5329E-6 \times T^2 + 2992/T$

¹ Temperature-dependent correlations used in Case C are written assuming the material temperature in Kelvin.

The results presented in Figure B.30 and Table B.10 indicate that the three sets of material properties provide similar results for the steady-state maximum surface heat flux. It is observed that the increase of the fuel conductivity assumed in the analysis from set A to sets B and C seems to be compensated by the decrease of the AA6061 thermal conductivity assumed in the analysis for the different sets of data. Since data set C contains the most updated data, which were also employed in the MURR accident analyses [B-6], this set of data was used for the remaining simulations which are presented in the following sections.

Figure B.30. Normal Total Surface Heat Flux for Preliminary LEU Fuel Element.

Table B.10. Comparison of Simulation Results for Different Sets of Thermal Conductivity Data.

Plate #	Case	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
3	A	161.60	3.17	25.25
	B	162.61	2.91	25.25
	C	162.10	3.04	25.25
4	A	169.30	2.76	24.50
	B	170.38	2.76	24.50
	C	169.73	2.76	25.25

B.5.3 Effects of Fuel Burnup

Fuel burnup affects multiple thermal-hydraulics parameters that may have an impact on the lateral heat conduction in the LEU fuel plate. Increasing fuel burnup reduces the fuel thermal conductivity, decreases the coolant channel thickness due to fuel swelling, creep, and the growth of an oxide layer on the surface of the fuel plates, reduces the coolant flow rate through the channel due to restriction of the coolant channel, and generally causes a reduction of the power density in the fuel. Because the goal of the present analysis is to select one representative fuel width to be used in the 1D thermal-hydraulics analysis of LEU fuel elements at all burnup levels relevant for MURR, it is necessary to determine which condition (high burnup or low burnup) leads to the least amount of lateral heat

conduction and the smallest representative hot-stripe width for the 1D analysis. In the context of this work, this is identified as the most conservative condition.

The model developed in Section B.5.2, which, for present purposes, is called the “base model,” was based on the following input data:

- Power distribution data for an MOL LEU element from Section B.3;
- Fuel thermal conductivity data, coolant channel gap, and oxide layer for an MOL element from Table B.5;
- Coolant inlet temperature and velocity from Table B.6;
- Temperature-dependent thermal conductivity correlations for U-10Mo, AA6061, and zirconium from Case C in Table B.9.

To evaluate the effect of the influence of burnup on these parameters and on the lateral heat conduction in the LEU fuel plate, four test models were created by modifying the input parameters in the COMSOL analysis.

- Test case 1: Use thermal conductivity of fresh fuel (BOL)
- Test case 2: Use 115% of the coolant flowrate assumed in the base model
- Test case 3: Use the BOL power density distribution
- Test case 4: Eliminate the oxide layer on the surface of the fuel plates

Comparisons are provided below for the predictions of the convective heat flux with COMSOL for each of these test cases relative to the base model in order to determine which fuel element burnup conditions result in the most conservative results for the lateral heat conduction.

Test case 1 –Effect of Fuel Thermal Conductivity

The thermal conductivity of the U-10Mo monolithic fuel proposed for LEU conversion of MURR decreases with burnup. The base model uses the fuel thermal conductivity corresponding to MOL conditions, which was found in previous analysis to be about 25% lower than that of fresh U-10Mo.

A test case using the thermal conductivity of fresh U-10Mo was evaluated to assess the impact of the thermal conductivity of the fuel in the plate on the lateral heat conduction. The results presented in Figure B.31 and Table B.11 show that the fuel thermal conductivity has a negligible effect on the predicted surface heat flux or the location of the maximum heat flux relative to the edge of the fuel.

Figure B.31. Effect of Thermal Conductivity on Surface Heat Flux.

Table B.11. Effect of Fuel Thermal Conductivity on Surface Heat Flux.

Plate #	Case	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
3	LEU- MOL	139.68	5.30	26.00
	Test case 1	139.62	5.30	26.00
4	LEU- MOL	148.15	4.52	25.25
	Test case 1	148.04	4.52	25.25

Test case 2 –Effect of Coolant Flowrate

The coolant channel gap for the MOL element represented in the original model was reduced from a nominal gap thickness of 2.36 mm at BOL to 2.25 mm at 96 MWd of fuel burnup. This reduction of the channel thickness is assumed to be due to the thermo-mechanical effects of temperature- and irradiation-induced creep and fuel swelling, in addition to the formation of an oxide layer on the surface of the fuel plate during operation. The reduction of the coolant channel gap in the higher burnup elements in the MURR core will cause a flow redistribution towards the fresh and less depleted fuel elements.

To assess the impact of this phenomenon on the lateral heat conduction, a test case with the coolant velocity increased by 15% relative to the base model for channel 4 of the MOL element was evaluated to simulate the coolant flow redistribution in a BOL element coolant channel relative to an MOL element coolant channel. It can be seen in Figure B.32 and Table B.12 that increasing the coolant flowrate by 15% has a very small effect on the lateral heat conduction and the convective heat flux.

Figure B.32. Effect of Flowrate on Surface Heat Flux.

Table B.12. Effect of Coolant Flowrate on Surface Heat Flux.

Plate #	Case	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
3	LEU- MOL	139.68	5.30	26.00
	Test case 2	139.94	5.14	26.00
4	LEU- MOL	148.15	4.52	25.25
	Test case 2	148.51	4.36	25.25

The base model for the COMSOL analysis of the LEU channel 4 used the coolant inlet conditions summarized in Table B.6. As described in Section B.4.1, these coolant inlet conditions do not represent typical operating conditions for MURR. Specifically, the coolant inlet temperature in the COMSOL analysis was about 20 °C (35 °F) higher than the typical inlet temperature and the coolant velocity through the channel in the COMSOL model was about 60% higher than what is realistic for MURR.

As a subset of the analysis performed for Test Case 2, the coolant inlet temperature and coolant velocity were set in the COMSOL analysis to values that are typical for MURR LEU operations (120 °F, 7 m/s, respectively). The results presented in Figure B.33 show that the setting the coolant inlet temperature to the lower value that is typical for MURR operations (Test Tinlet Case) has a negligible effect on the surface heat flux of the fuel plate, while using the typical coolant velocity in the channel (Test Vinlet Case) reduces the maximum surface heat flux by < 1.0%. It is further observed that the results for the base model, with an assumed coolant velocity of 11.19 m/s predict that less lateral heat conduction occurs in the fuel plates, which is more conservative for the purpose of the present analysis. Consequently, although the base model assumed coolant conditions that are quite different from that that are typical for MURR operations, these modeling assumptions have a small and conservative impact on the results.

MOL Case: $T_{\text{inlet}} = 68.3^{\circ}\text{C}$ (155°F), Channel coolant velocity = 11.19 m/s
 Test Tinlet Case: $T_{\text{inlet}} = 48.9^{\circ}\text{C}$ (120°F), Channel coolant velocity = 11.19 m/s
 Test Vinlet Case: $T_{\text{inlet}} = 68.3^{\circ}\text{C}$ (155°F), Channel coolant velocity = 7.00 m/s

Figure B.33. Effect of Coolant Inlet Temperature and Velocity on Surface Heat Flux.

Test case 3 – Effect of Power Density Distribution

This test case evaluated the effect the power density in the fuel on the lateral heat conduction. In this case, the power density distributions for plates 3 and 4 in a BOL element were utilized in the COMSOL analysis of channel 4. The average power density in plate Figure B.34 and Table B.13 show that with a higher power density in the fuel, the surface heat flux is greater and the distance between the maximum heat flux and the edge of the fuel is less. The shorter distance between the maximum heat flux and edge of the fuel indicates that the lateral heat conduction will be less in fuel elements with a higher power density, as at BOL. Therefore, the representative hot-stripe width obtained from analysis with the BOL power density is conservative.

Figure B.34. Effect of Power Distribution on Surface Heat Flux.

Table B.13. Effect of Power Distribution on Surface Heat Flux.

Plate #	Case	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
3	LEU- MOL	139.68	5.30	26.00
	Test case 3	151.95	4.97	25.25
4	LEU- MOL	148.15	4.52	25.25
	Test case 3	157.03	4.21	24.50

Test case 4 – Effect of Oxide Layer on the Fuel Plate

The simulation results presented in Figure B.35 and Table B.14 indicate that an oxide layer on the surface of the fuel plates in the element strongly affects lateral heat conduction and the surface heat flux. For the test case without oxide on the fuel plate, a larger maximum heat flux is predicted than for the original model with the oxide. Additionally, the distance from the edge of the fuel to the maximum heat flux is much smaller than when there is no oxide. This means that a more conservative hot-stripe width will be predicted for fuel plates without oxide (i.e., for fresh fuel plates).

Figure B.35. Effect of Oxide Layer on Surface Heat Flux.**Table B.14. Effect of Oxide Layer on Surface Heat Flux.**

Plate #	Case	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
3	LEU- MOL	139.68	5.30	26.00
	Test case 4	148.02	3.17	26.00
4	LEU- MOL	148.15	4.52	25.25
	Test case 4	159.66	2.76	25.25

These results indicate that the oxide layer has a beneficial effect with regard to increasing lateral heat conduction and decreasing the heat flux conducted into the coolant channel. A detailed depiction of heat transfer near the fuel edge is presented in Figure B.36 to explain these results. It can be seen that the lateral heat transfer is greater in the MOL plate with the oxide layer present (see the heat flux plots in the lower right-hand corner of the figure). This is because the presence of the oxide layer inhibits heat transfer from the fuel plate to coolant, so that a larger portion of the heat in the fuel is transferred laterally along the cladding toward the side plate. Consequently, the magnitude of the maximum heat flux conducted into the coolant channel at the surface of the fuel plate is decreased for the case with oxide on the fuel plate. Likewise, the location of the maximum heat flux moves away from the edge of the fuel in the azimuthal direction.

Figure B.36. Effect of Oxide Layer on Heat Flux.

The total effect of fuel burnup on the lateral heat conduction was also assessed by comparing the results for a COMSOL analysis of channel 4 in preliminary LEU elements at three different burnup states, which are summarized in Table B.15. The results presented in Figure B.37 and Table B.16 show that the representative hot-stripe width determined for the fresh fuel element case will be the more conservative value (greater maximum heat flux, smaller distance from the edge of the fuel to the maximum heat flux) to be used for 1D thermal-hydraulics analyses for elements at burnups relevant for MURR.

Table B.15. Models with Different Burnup Level.

Input parameters	BOL (0 MWd)	MOL (96 MWd)	EOL (170 MWd)
Ratio of thermal conductivity relative to fresh fuel	1.00	0.75	0.65
Coolant channel thickness, mm	2.36	2.25	2.16
Oxide layer thickness, mm	0.00	0.0160	0.0236
Power distribution	Correlations given in Section B.3

Figure B.37. Total Effect of Burnup.**Table B.16. Total Effect of Burnup.**

Plate #	Case	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
3	BOL	162.10	3.04	25.25
	MOL	139.68	5.30	26.00
	EOL	131.39	5.64	24.50
4	BOL	169.73	2.76	25.25
	MOL	148.15	4.52	25.25
	EOL	140.80	4.69	24.50

B.6 Analysis of Preliminary LEU Fuel Element Design

Figure B.38 presents the axially-averaged hot-stripe heat flux for all fuel plates in the reference Core 7A, calculated for a hot-stripe width of 5 mm. This figure indicates that the coolant channels of the preliminary LEU fuel design that are most likely to experience the minimum margin to flow instability are channel #4 (bounded by plates 3 and 4) and channel #23 (bounded by plates 22 and 23). Thus, these coolant channels were selected for evaluation in more detail. Based on the results of the parametric studies performed in Section B.5.3, a fresh element was chosen for the analysis since this provides the most conservative results for the lateral heat conduction over the range of element burnups relevant for MURR.

Figure B.38. Axially-Averaged Hot-Stripe Heat Flux of All Fuel Plates in Core 7A for a 5 mm Hot-Stripe Width.

The power distributions of the fresh plates 3, 4, 22, and 23 in Core 7A, plotted in Figure B.39 using the power density correlations provided in Section B.3, were used in the COMSOL model of the fuel plates. The design dimensions in Table B.4 and the fuel and coolant channel properities for the BOL element in Table B.5 were utilized. Additionally, the thermo-physical properties for Case C in Table B.9 were used.

Figure B.39. Azimuthal Power Density Distribution for Limiting Plates in Core 7A.

The results for the convective heat flux are presented in Figure B.40 and Table B.17. It can be seen that the maximum heat flux is greater for plates 22 and 23. Additionally, the flatter azimuthal power density profiles for plates 22 and 23 (Figure B.39) result in a larger distance between the edge of the fuel and the maximum of the heat flux.

Figure B.40. Surface Heat Flux for Limiting Plates in Core 7A.

Table B.17. Maximum Surface Heat Flux for Limiting Plates in Core 7A.

Plate #	Maximum Heat Flux, W/cm ²	Distance from Edge of Fuel to Max Heat Flux, mm	Distance from Plate Bottom, cm
3	162.10	3.04	25.25
4	169.73	2.76	25.25
22	182.47	3.31	25.25
23	208.99	4.31	25.25

To obtain a representative surface heat flux that will produce conservative results for 1D thermal-hydraulics analysis, a representative hot-stripe width needs to be selected so that averaging the heat flux over this stripe width is greater than or equal to the maximum heat flux obtained by the 3D CFD analysis. Table B.18 confirms that the previous 1D steady-state thermal-hydraulics analysis using a 5-mm hot-stripe width is conservative for all channels. Moreover, as the results in Table B.18 and Figure B.41 show, a hot-stripe width of 6 or 7 mm could also be used to obtain conservative results.

Table B.18. Determination of Representative Stripe Width for 1D Heat Transfer Analysis.

Plate #	COMSOL results of peak surface heat flux		1D surface heat flux calculated by integrating power density over various stripe widths, W/cm ²					
	Distance from Edge of Fuel to Max Heat Flux, mm	Maximum Heat flux, W/cm ²	5 mm	6 mm	7 mm	8 mm	9 mm	10 mm
3	3.04	162.10	170.01 4.9%	167.58 3.4%	165.36 2.0%	163.34 0.8%	161.50 -0.4%	159.83 -1.4%
4	2.76	169.73	176.34 3.9%	173.31 2.1%	170.54 0.5%	167.99 -1.0%	165.67 -2.4%	163.55 -3.6%
22	3.31	182.47	189.79 4.0%	187.83 2.9%	185.98 1.9%	184.23 1.0%	182.57 0.1%	181.00 -0.8%
23	4.31	208.99	214.37 2.6%	213.23 2.0%	212.15 1.5%	211.11 1.0%	210.12 0.5%	209.19 0.1%

Figure B.41. Determination of 1D Representative Hot-Stripe Width using Data of Plate 4.

B.7 Conclusion

A COMSOL model of a MURR LEU fuel element was set up and verified by comparing to previous results for the same element design with StarCD. A COMSOL model was then created to predict the lateral heat conduction and convective heat flux for fuel plates in the preliminary LEU fuel element design. The analysis has confirmed the following conclusions:

- The contact gap between fuel plate and side plate can be ignored in the thermal-hydraulic analyses.
- Three sets of thermal conductivity for materials in the LEU fuel plate have been used for MURR conversion analysis as the analysis has matured. The thermal conductivity values used in References B-4, B-5, and B-6 provide similar steady-state surface heat-flux distributions. The impact of increasing the fuel thermal conductivity from the values assumed in Reference B-4 to those assumed in References B-5 and B-6 appears to be compensated by the decreasing AA6061 thermal conductivity.
- The burnup of the fuel affects multiple thermal-hydraulic parameters, including the power density in the fuel and the oxide layer thickness on the fuel plate. The effect of increasing fuel burnup was found through a parametric study increase lateral heat conduction in the LEU fuel plate. Consequently, a representative hot-stripe width obtained through analysis of a fresh fuel element will produce conservative results for 1D thermal-hydraulic analyses for fuel elements at all burnups relevant for MURR.
- The 1D thermal-hydraulic analysis of the preliminary LEU fuel element for MURR conversion remains conservative for a representative hot-stripe width of up to 7.0 mm. The 1D thermal-hydraulic analyses reported in References B-5 and B-6 using a representative hot-stripe width of 5.0 mm are conservative.

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- B-3. J. C. McKibben, et al., *Feasibility Analyses for HEU to LEU Fuel Conversion of the University of Missouri Research Reactor (MURR)*, MURR Technical Data Report No. 0125, September 2009.
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- B-5. E. E. Feldman, et al., *Technical Basis in Support of the Conversion of the University of Missouri Research Reactor (MURR) Core from Highly-Enriched to Low-Enriched Uranium – Steady-State Thermal-Hydraulic Analysis*, ANL/RERTR/TM-12-37, Rev. 1, Argonne National Laboratory, January 2013.
- B-6. J. A. Stillman, et al., *Accident Analyses for Conversion of the University of Missouri Research Reactor (MURR) from Highly-Enriched to Low-Enriched Uranium*, ANL/GTRI/TM-14/5, Revision 1, February 2017.

APPENDIX C – Effect of Beryllium Reflector Age on Power Distributions and Predicted Safety Margins

A fresh beryllium reflector was assumed in the neutronics analysis from which power distributions for HEU and LEU cores were derived. As discussed in Section 2.1, the previous irradiation history of the beryllium reflector affects the core excess reactivity and critical control blade position. For an “aged beryllium” reflector that has experienced 26,000 MWd of core burnup, the core excess reactivity is reduced by ~500 pcm. Consequently, the critical control blade position at BOC (no xenon conditions) is about one inch more withdrawn than for a similar core loading with a fresh beryllium reflector. At EOC, the control blades are withdrawn about two inches more with a highly depleted reflector relative to the position with a fresh reflector. The critical control blade height has a direct effect on the radial and axial power distribution in the core.

Table C.1 presents the results of cases that simulate how “aged beryllium” may affect the predicted margins to flow instability for HEU and LEU cores. Cases were evaluated for the HEU Core 4C and LEU Core 8C, which have driver elements with a mixture of burnups that is typical for MURR operations, equilibrium xenon (two days into the weekly operating cycle), and the 2017-RBM-99 upgrade. The equilibrium xenon cases were selected because these tend to be more limiting for the flow instability power than BOC or EOC states. The critical control blade positions for these cores are 21.12 and 20.93 inches withdrawn for HEU and LEU, respectively. The results for the Cores 423 and 823 are for the same fuel element loading and xenon condition as Cores 4C and 8C, respectively, but with the control blades positioned at 23 inches withdrawn, which is about two inches higher than the corresponding critical case. As such, these cases simulate the effect of a beryllium reflector at its maximum burnup before the normally scheduled beryllium replacement in MURR (which occurs every 8 years). It should be noted that these cases do not account for any change in the neutron flux spectrum as a result of Be degradation, which can also influence the core power distribution. With the control blades more fully withdrawn, the predicted core k_{eff} is 889 and 840 pcm above critical for the HEU and LEU cores, respectively, because the beryllium is still modeled as fresh. The power distribution for these cores was calculated by MCNP5 using the same methodology as described in Section 2.3 for Cores 4C and 8C.

The flow instability power is affected by the hot-stripe power. In Table C.1, heat flux is used as a surrogate for power. In the table, the hot-stripe heat flux is defined as half of the stripe power divided by the heat transfer area of one face of the stripe and local heat flux is defined as the product of local volumetric heat generation rate in the fuel meat and half of the fuel meat thickness. For Core 4C, the limiting channel for steady-state flow instability is coolant channel 2 between fuel plates 1 and 2 in the BOL HEU element in core position 5. It can be seen in Table C.1 that the hot-stripe heat flux in Core 423 for both plates 1 and 2 in this element is lower when the control blades are more fully withdrawn. This is because the core power distribution shifts towards the outer fuel plates when the blades are positioned higher. The predicted flow instability power is 16.28 MW in Core 423, which is 0.22 MW higher than the corresponding Core 4C. Thus, the assumption of fresh beryllium is conservative for predicting the HEU margin to flow instability.

Table C.1. Effect of Aged Beryllium Reflector on Flow Instability Power.

HEU Core, Equilibrium Xenon Conditions

Beryllium Condition	Fresh	Fresh, but simulate 8-year aged Be by withdrawing control blades about 2 inches relative to fresh Be critical case
Core	4C	423
Control Blade position (inches withdrawn)	21.12	23
k_{eff}	0.99997	1.00889
Hot-Stripe Heat flux (affects OFI power)		
Element 5, Plate 1	130.3	128.6
Element 5, Plate 2	109.3	108.0
Maximum Local Heat Flux (affects maximum fuel temperature in transients)		
Element 5, Plate 1	171.2	165.7
Element 5, Plate 2	143.9	139.0
Limiting element	5	5
Limiting channel	2	2
OFI Power, Reactor (MW)	16.06	16.28

LEU Core, Equilibrium Xenon Conditions

Beryllium Condition	Fresh	Fresh, but simulate 8-year aged Be by withdrawing control blades about 2 inches relative to fresh Be critical case
Core	8C	823
Control Blade position (inches withdrawn)	20.93	23
k_{eff}	0.99998	1.00840
Hot-Stripe Heat flux (affects OFI power)		
Element 6, Plate 22	119.8	124.2
Element 6, Plate 23	128.8	133.7
Maximum Local Heat Flux (affects maximum fuel temperature in transients)		
Element 5, Plate 3	160.7	156.4
Element 5, Plate 4	162.3	157.9
Limiting element	6	6
Limiting channel	23	23
OFI Power, Reactor (MW)	19.28	18.63

Table C.1 also provides the maximum local heat flux in plates 1 and 2 of the BOL element in core position 5 as an indicator of how the prior irradiation of the beryllium may affect the maximum fuel temperature during transients, which are evaluated in Section 4. These plates were selected for comparison because the results in Table 4.7 show that the maximum HEU temperature during the

double-ended cold-leg LOCA occurs in plate 2 of this element. As for the hot-stripe heat flux, the maximum local heat flux is larger for the fresh beryllium case, indicating that this is a conservative assumption for calculating the minimum HEU margin to the fuel temperature safety limit during transients.

For Core 8C, the limiting channel for steady-state flow instability is coolant channel 23 between fuel plates 22 and 23 in the MOL LEU element in core position 6. The hot-stripe heat flux for these fuel plates is greater in Core 823 when the control blades are more fully withdrawn, which would be the case for the aged beryllium. The flow instability power in this case is 18.63 MW, which is 0.65 MW less than that predicted for Core 8C. However, as can be seen in Table 3.3, the effect of the aged beryllium reflector on the flow instability power is bounded by the extreme cases for this same core with a mixture of fresh and highly depleted control blades and blade-height mismatch. Thus, there remains confidence in the minimum predicted margin of 2.35 MW to flow instability relative to the LSSS power for LEU.

The maximum local heat flux for fuel plates 3 and 4 in the BOL LEU element in core position 5 is also summarized in Table C.1. These plates were selected because the results in Table 4.7 show that the minimum margin to the LEU fuel temperature safety limit in the double-ended cold-leg LOCA occurs in plate 3 of this element. It can be seen that the maximum local heat flux is greater for Core 8C, with the fresh beryllium, than for core 823, which simulates the aged beryllium. The more deeply inserted control blades in this case shifts the local power distribution towards the plates closest to the center test hole. This indicates that the power distributions calculated for the fresh beryllium will give conservative results for the minimum margin to the LEU fuel safety limit for the transients evaluated in Section 4.

APPENDIX D – Updated Accident Analyses for LOCA and LOFA Scenarios Presented for Preliminary Safety Analysis Report

D.1 Introduction

Accident analyses for HEU and LEU cores without the 2017-RBM-99 device with experiment loadings typical in 2008 (MURR_2008 in Section 2) were previously conducted for the Preliminary Safety Analysis Report (PSAR) as reported in Reference D-1 and Appendix B of Reference D-2 (referred as PSAR analyses). The identifications for those HEU and LEU cores are 3A and 7A, respectively. As discussed in Section 4, the accident analyses for cores with the 2017-RBM-99 device with experiment loadings typical in 2018 (MURR_2018 in Section 2) are conducted for reference HEU and LEU cores 3C and 7C, respectively. In order to gauge the impact of the presence of 2017-RBM-99 device, comparison of the results with and without the device is critical.

The LOCA/LOFA portion of the PSAR analyses was conducted with RELAP5/MOD3.3, Patch 03 on a PC platform. Since then, RELAP5/MOD3.3, Patch 05 has become available and has been evaluated for quality-affecting work. Consequently, the Mo-99 upgrade analyses reported in Section 4 of this document were conducted with Patch 05. In addition, the present Mo-99 upgrade calculations were conducted on a Linux platform, whereas the PSAR analyses were based on calculations performed on a PC. Consequently, it became necessary to repeat the LOCA and LOFA part of the PSAR analyses with RELAP5/MOD3.3, Patch 05 on a Linux platform. The repeated calculations serve two purposes: a) to provide accident analysis results with a software version and computation platform for cases without the 2017-RBM-99 device that allows direct comparison to the results predicted for cores with the 2017-RBM-99 device, and b) to document the updated results for the PSAR analyses with the newer software version and change in the computation platform.

The complete set of LOCA and LOFA calculations were repeated with identical input files. Minor changes in the results were observed, but the most limiting accident scenario as well as the location of minimum margin did not change (cold-leg double-ended LOCA for end-of-life element with flow redistribution due to burnup at plate 22, node 3). The tests for anti-siphon system were also repeated.

In the PSAR analyses, the fuel temperature safety limit for HEU aluminide fuels was not considered. Therefore, the margins to the fuel temperature safety limit were not calculated for the HEU core. In this section, the fuel temperature safety limit for HEU fuel is assumed to be 475 °C (887 °F) to be consistent with the Mo-99 upgrade analyses discussed in Section 4. The fuel temperature safety limit for LEU fuel is modeled to be burnup-dependent and the same approach was used to obtain the limits in the previous and current analyses.

D.2 Loss-of-Coolant Accidents (LOCA)

This section presents results for LOCA analysis as well as the anti-siphon system tests. The results presented in this section correspond to the results from Section B-5 of Reference D-2.

D.2.1 Results from HEU Core LOCA Analysis

The results for HEU are summarized in Table D.1. For the HEU reference core (3A), the most limiting LOCA scenario occurs in the BOL element and is the cold-leg double-ended break. The maximum fuel

temperature for this scenario is 246 °C (474 °F) in plate 2, which gives a minimum margin to the fuel temperature safety limit of 229 °C (413 °F) in the same location.

Table D.1. HEU Core 3A LOCA Analyses Results without Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Cold-leg LOCA, Double-ended	BOL (0)	127	246 (474)	119.03	2, 3	229 (413)	2, 3
	MOL-1 (65)	125	204 (400)	79.40	2, 3	271 (487)	2, 3
	MOL-2 (81)	123	183 (361)	60.03	2, 3	292 (526)	2, 3
	EOL (142)	118	158 (317)	40.89	2, 3	317 (570)	2, 3
Cold-leg LOCA, 8"	BOL (0)	130	138 (281)	8.39	1, 3	337 (606)	1, 3
	MOL-1 (65)	126	134 (274)	8.61	1, 3	341 (613)	1, 3
	MOL-2 (81)	123	131 (269)	8.51	1, 3	344 (618)	1, 3
	EOL (142)	118	125 (258)	7.78	2, 3	350 (629)	2, 3
Cold-leg LOCA, 2"	BOL (0)	127	131 (268)	4.27	2, 3	344 (619)	2, 3
	MOL-1 (65)	125	130 (267)	5.55	2, 3	345 (620)	2, 3
	MOL-2 (81)	123	130 (266)	7.33	2, 3	345 (621)	2, 3
	EOL (142)	115	129 (265)	14.55	2, 3	346 (622)	2, 3
Hot-leg LOCA, Double-ended	BOL (0)	130	130 (266)	0.00	1, 3	345 (621)	1, 3
	MOL-1 (65)	126	126 (258)	0.00	1, 3	349 (629)	1, 3
	MOL-2 (81)	123	123 (253)	0.00	1, 3	352 (634)	1, 3
	EOL (142)	118	118 (244)	0.00	2, 3	358 (644)	2, 3

Calculation results with RELAP5/MOD3.3 Patch 03 reported in Table B-5.3 of Reference D-2 predict the same most limiting scenario (axial node 3 of Plate 2 of the BOL element), but there is a decrease for the predicted maximum HEU fuel temperature of 20 °C (266 °C compared with 246 °C) when the new version of the software and a different computing platform is used.

More detailed transient behavior of the cold-leg double-ended break LOCA is outlined in Figure D.1 for the HEU core without considering flow redistribution due to burnup. As discussed in Section 4.3, each subplot illustrates a key parameter monitored during the accident transient.

Figure D.1. Cold-Leg Double-Ended Break LOCA Transients for HEU Core 3A BOL Element (Element 5) without Flow Redistribution.

D.2.2 Results from LEU Core LOCA Analyses

The results for LEU with and without flow redistribution due to burnup are summarized in Table D.2 and Table D.3, respectively. For the LEU reference core (7A) with flow redistribution due to burnup, the most limiting LOCA is the cold-leg double-ended break. The maximum fuel temperature for this scenario occurs in fuel plate 22 of the EOL element and is 299 °C (569 °F). The minimum margin to the fuel temperature safety limit of 118 °C (212 °F) occurs in the same location. For the LEU reference core (7A) without flow redistribution due to burnup, the most limiting LOCA is the cold-leg double-ended break. The maximum fuel temperature for this scenario occurs in fuel plate 3 of the BOL element and is 289 °C (552 °F). The minimum margin to the fuel temperature safety limit of 139 °C (250 °F) occurs in fuel plate 22 of the EOL element.

Table D.2. LEU Core 7A LOCA Analyses Results with Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Cold-leg LOCA, Double-ended	BOL (0)	139	272 (521)	132.31	22, 3	179 (321)	22, 3
	MOL-1 (77)	147	270 (517)	122.47	4, 3	180 (325)	4, 3
	MOL-2 (96)	152	276 (529)	123.54	22, 3	174 (313)	22, 3
	EOL (170)	156	299 (569)	142.27	22, 3	118 (212)	22, 3
Cold-leg LOCA, 8"	BOL (0)	144	149 (301)	5.79	23, 3	301 (541)	23, 3
	MOL-1 (77)	150	157 (315)	6.68	23, 3	293 (527)	23, 3
	MOL-2 (96)	154	160 (320)	6.37	23, 3	287 (517)	1, 3
	EOL (170)	156	164 (327)	7.56	23, 3	239 (430)	23, 3
Cold-leg LOCA, 2"	BOL (0)	144	144 (291)	0.16	23, 3	306 (551)	23, 3
	MOL-1 (77)	150	151 (303)	0.18	23, 3	299 (539)	23, 3
	MOL-2 (96)	154	154 (309)	0.18	23, 3	295 (530)	1, 3
	EOL (170)	156	157 (314)	0.22	23, 3	246 (444)	23, 3
Hot-leg LOCA, Double-ended	BOL (0)	144	144 (291)	0.00	23, 3	306 (551)	23, 3
	MOL-1 (77)	150	150 (303)	0.00	23, 3	300 (539)	23, 3
	MOL-2 (96)	154	154 (309)	0.00	23, 3	296 (533)	1, 3
	EOL (170)	156	156 (313)	0.00	23, 3	247 (444)	23, 3

Table D.3. LEU Core 7A LOCA Analyses Results without Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Cold-leg LOCA, Double-ended	BOL (0)	137	289 (552)	152.41	3, 3	161 (290)	3, 3
	MOL-1 (77)	149	281 (538)	132.53	4, 3	169 (304)	4, 3
	MOL-2 (96)	154	257 (494)	102.89	22, 3	193 (348)	22, 3
	EOL (170)	157	278 (532)	120.92	22, 3	139 (250)	22, 3
Cold-leg LOCA, 8"	BOL (0)	147	151 (304)	4.30	23, 3	299 (538)	23, 3
	MOL-1 (77)	153	158 (316)	5.44	23, 3	292 (526)	23, 3
	MOL-2 (96)	156	161 (321)	5.09	23, 3	287 (516)	1, 3
	EOL (170)	157	163 (325)	5.57	22, 3	239 (431)	22, 3
Cold-leg LOCA, 2"	BOL (0)	147	147 (297)	0.17	23, 3	303 (545)	23, 3
	MOL-1 (77)	153	153 (307)	0.18	23, 3	297 (535)	23, 3
	MOL-2 (96)	156	156 (313)	0.17	23, 3	294 (529)	1, 3
	EOL (170)	157	157 (315)	0.18	22, 3	245 (441)	22, 3
Hot-leg LOCA, Double-ended	BOL (0)	147	147 (296)	0.00	23, 3	303 (546)	23, 3
	MOL-1 (77)	153	153 (307)	0.00	23, 3	297 (535)	23, 3
	MOL-2 (96)	156	156 (312)	0.00	23, 3	294 (530)	23, 3
	EOL (170)	157	157 (314)	0.00	22, 3	245 (442)	22, 3

Results for the LOCAs with flow redistribution calculated with RELAP5/MOD3.3 Patch 03 were reported in Table B-5.4 of Reference D-2, where it was reported that the maximum fuel temperature of 301 °C (574 °F) occurred in fuel plate 22 of the EOL element. The minimum margin to the fuel temperature safety limit of 115 °C (239 °F) occurred in the same location. Thus, the change of the software version and computing platform results in a change of only a few degrees for the predicted fuel temperature and margin to the fuel temperature safety limit. For the LEU reference core (7A) without flow redistribution, the most limiting LOCA is the cold-leg double-ended break. With RELAP5/MOD3.3 Patch 03, the maximum fuel temperature occurs in fuel plate 3 of BOL element and is 303 °C (297 °F). Thus, the software change results in a decrease of 14 °C for the maximum fuel temperature. The minimum margin to the fuel temperature safety limit for this scenario occurs in the same location (fuel plate 22 of the EOL element), but is 8 °C lower with the new version of the software.

Detailed transient behavior of the most limiting LOCA for the LEU core without the 2017-RBM-99 upgrade is shown in Figure D.2.

Figure D.2. Cold-Leg Double-Ended Break LOCA Transients for LEU Core 7A EOL Element (Element 8) with Flow Redistribution.

D.2.3 Anti-Siphon System Test

The analysis of the effectiveness of the anti-siphon system that is provided in Sections 5.6 and 5.7.5 of Reference D-2, and then updated in Section B-5.12 of the same reference, assumed that the source of the incoming air used for siphoning and the exiting liquid were both at the same lowest elevation in the model. With this arrangement siphon is virtually impossible, even if the anti-siphon is not activated. This shortcoming in the analysis was recognized and Reference D-1 states, "The prediction of no siphoning, even with the anti-siphon system not activated, could almost be anticipated given that the water exit, which is located at the primary coolant outlet isolation valve, and the source of air needed for siphoning, which is located at the primary coolant inlet isolation valve, are both at the same elevation."

Further examination of the reactor system uncovered another shortcoming in the analysis. A break in the pressurized primary system boundary that is beyond the pool could allow pool water to drain into the pressurized system after the system depressurizes. This could come about because there are a pair of parallel check valves at the top of the inlet header to the reactor vessel. These valves are closed while the system is at pressure and can open after it depressurizes and thereby provide a source of water from the pool. Conceivably, over time, the pool water can drain through these valves and leak out through the break. When the pool level reaches these check valves, there would be a source of air at the top of the reactor vessel to promote siphoning. However, this source of air could not lead to siphoning, provided that at the outset of a depressurization event the in-pool heat exchanger valves (V546 in Figure 4.3) open as they are designed. The opening of the valves allows the in-pool heat exchanger path, which is marked as red in Figure 4.3, to connect the reactor inlet header to the reactor outlet header, thereby arresting any siphoning.

Proper modeling of the draining of the pool would require modeling the check valves, connecting a vertical volume of water above them to represent the pool water, and adding a source of air above this volume of water. (The cross sectional area of the added pool volume could be made much smaller than that of the actual pool in order to save on simulation time.) Then, to model the effectiveness of the anti-siphon system, it would be assumed that three faults, or sets of faults, occurred simultaneously: 1) a breach of the pressurized system occurred, initiating the upset, 2) the anti-siphon system, which has two parallel valves for redundancy, failed to activate, and 3) both parallel in-pool heat exchanger valves failed to open. Since this scenario is reasonably beyond credibility, no further analysis of the anti-siphon system should be needed. However, since the anti-siphon analysis was included in the PSAR that was submitted to the NRC, the analysis is repeated here using the qualified software RELAP5/MOD3.3 Patch 05 to replace the anti-siphon analysis performed in the PSAR with RELAP5/MOD3.3 Patch 03.

The results for the anti-siphon analysis presented here correspond specifically to the results from Section B-5.12 of Reference D-2. The results are shown in Table D.4, below.

There are significant changes in the results with newer patch version of RELAP5 and change in calculation platform (from Windows to Linux). The differences in the results from Section B-5.12 of Reference D-2 and those shown in Table D.4 can be attributed to the nature of the monitored variables. Three out of four (items 1, 2, and 4) variables presented in above table becomes very small and often show oscillatory behavior. For item 1, the time when check valve 502 permanently closes is determined by monitoring the flow area ratio for the valve (1 being fully open and 0 being fully closed). As the transient progresses, the flow area ratio decreases to very small values and oscillates, sometimes reaching zero and increasing to a very small value in the next time step, until at one timestep, it eventually reaches zero and stays zero. Similarly for item 3, the time for exit pipe 102 to

empty all liquid is determined by monitoring the liquid void fraction of the control volume and is assumed to be when the liquid void fraction drops below 0.001 for the last time. Because of the selection of cutoff value and oscillatory behavior of the monitored values, small perturbations in calculation scheme can result in seemingly significant discrepancies, especially for the types of variables dealt with in this section.

Table D.4. Results of Anti-Siphon System Test.

	Criterion	Anti-Siphon State:	HEU		LEU	
			active	inactive	active	inactive
1	Time for check valve 502 to permanently close, s		16	10	17	10
2	Minimum liquid fraction in inlet plenum 501		0.8704	0.8487	0.9995	0.8666
3	Time for exit pipe 102 to empty all liquid, s*		23	34	32	33
4	Time for flow at exit to go to zero, s**		17	34	17	33

*The volume is assumed void when liquid void fraction is less than 0.001.

**The flow rate is assumed zero when mass flow rate is less than 1 lb_m/s (≈ 0.4536 kg/s).

Nevertheless, as discussed in Section B-5.12 of Reference D-2, it should be noted that the important point is not the specific values or the differences between the values from previous and current results, but rather that the simulations predict that there is no siphoning for either HEU or LEU cores with the anti-siphon system activated or not activated. It should be noted that, as discussed in Section 6.3.8 of Reference D-1, it is expected that the siphoning does not occur even for the postulated scenario because “There is a very long, highly resistive flow path that would have to empty before the incoming air could reach the top of the reactor vessel.” Moreover, the siphoning would not occur even if the air does reach the top of the reactor vessel. At the outset of a depressurization event, the in-pool heat exchanger valves (V546 in Figure 4.3) are designed to open. Then the flow paths in the upper side of the pool (pipes marked as red in Figure 4.3) is opened, directing the inlet air to the top of the outlet header, arresting siphoning.

D.3 Loss-of-Flow Accidents (LOFA)

This section presents results for LOFA analysis. The results presented here correspond to the results from Section B-6 of Reference D-2.

D.3.1 Results from HEU Core LOFA Analysis

The results for HEU are summarized in Table D.5. For the HEU reference core (3A), the most limiting LOFA scenario occurs in the the loss of facility electrical power in the BOL element. The maximum fuel temperature for this scenario is 148 °C (299 °F) in plate 24, which gives a minimum margin to the fuel temperature safety limit of 327 °C (588 °F) in the same location. Calculation results with RELAP5/MOD3.3 Patch 03 reported in Reference D-2 predict the same most limiting scenario, maximum fuel temperature, and its location.

Table D.5. HEU Core 3A LOFA Analyses Results without Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Loss of facility electrical power	BOL (0)	116	148 (299)	32.55	24, 3	327 (588)	24, 3
	MOL-1 (65)	118	147 (297)	29.00	24, 3	328 (590)	24, 3
	MOL-2 (81)	117	147 (296)	29.33	24, 3	328 (591)	24, 3
	EOL (142)	107	138 (280)	31.26	2, 2	337 (607)	2, 2
Inadvertent closure of isolation valves	BOL (0)	127	147 (297)	20.44	2, 3	328 (590)	2, 3
	MOL-1 (65)	125	147 (296)	21.97	2, 3	328 (591)	2, 3
	MOL-2 (81)	107	146 (296)	39.14	3, 2	329 (591)	3, 2
	EOL (142)	110	145 (292)	34.18	22, 3	330 (595)	22, 3
Locked rotor in a coolant circulation pump	BOL (0)	130	143 (290)	13.50	1, 3	332 (597)	1, 3
	MOL-1 (65)	126	137 (278)	11.16	1, 3	338 (609)	1, 3
	MOL-2 (81)	123	133 (272)	10.45	1, 3	342 (615)	1, 3
	EOL (142)	118	127 (260)	9.11	2, 3	348 (627)	2, 3

D.3.2 Results from LEU Core LOFA Analysis

The results for LEU with and without flow redistribution due to burnup are summarized in Table D.6 and Table D.7, respectively. For the LEU reference core (7A) with flow redistribution due to burnup, the most limiting LOFA is the locked rotor in a coolant circulation pump. The maximum fuel temperature for this scenario occurs in fuel plate 22 of the EOL element and is 170 °C (338 °F). The minimum margin to the fuel temperature safety limit of 234 °C (421 °F) occurs in plate 23 of the same element. For the LEU reference core (7A) without flow redistribution due to burnup, the most limiting LOFA is also the locked rotor in a coolant circulation pump. The maximum fuel temperature for this scenario occurs in fuel plate 22 of the EOL element and is 171 °C (339 °F). The minimum margin to the fuel temperature safety limit of 232 °C (418 °F) occurs in fuel plate 23 of the same element. Calculation results with RELAP5/MOD3.3 Patch 03 reported in Reference D-2 predict the same most limiting scenarios, maximum fuel temperatures, minimum margins, and limiting locations.

Table D.6. LEU Core 7A LOFA Analyses Results with Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Loss of facility electrical power	BOL (0)	139	155 (310)	15.33	22, 3	295 (532)	22, 3
	MOL-1 (77)	149	154 (310)	5.73	22, 3	296 (532)	22, 3
	MOL-2 (96)	152	155 (310)	2.21	22, 3	286 (515)	1, 3
	EOL (170)	156	156 (313)	0.00	22, 3	240 (432)	2, 3
Inadvertent closure of isolation valves	BOL (0)	139	150 (302)	10.67	22, 3	300 (540)	22, 3
	MOL-1 (77)	149	151 (303)	1.86	22, 3	297 (534)	1, 3
	MOL-2 (96)	154	154 (309)	0.00	23, 3	283 (510)	1, 3
	EOL (170)	156	156 (313)	0.00	22, 3	244 (440)	2, 3
Locked rotor in a coolant circulation pump	BOL (0)	144	160 (319)	15.91	23, 3	290 (523)	23, 3
	MOL-1 (77)	150	165 (328)	14.26	23, 3	285 (514)	23, 3
	MOL-2 (96)	154	168 (335)	14.43	23, 3	282 (507)	23, 3
	EOL (170)	156	170 (338)	13.74	22, 3	234 (421)	23, 3

Table D.7. LEU Core 7A LOFA Analyses Results without Flow Redistribution.

Accident	Burnup (MWd)	Initial Temp. at Maximum Location, °C	Maximum Fuel Temperature			Fuel Safety Limit	
			Maximum Fuel Temp., °C (°F)	Temp. Rise at Maximum Location, °C	Location of Maximum - Plate, Node	Minimum Margin to Fuel Temp. Safety Limit, °C (°F)	Location of Minimum Margin - Plate, Node
Loss of facility electrical power	BOL (0)	142	155 (311)	12.57	22, 3	295 (531)	22, 3
	MOL-1 (77)	150	155 (310)	4.27	22, 3	295 (532)	22, 3
	MOL-2 (96)	156	156 (312)	0.00	23, 3	287 (516)	1, 3
	EOL (170)	157	157 (314)	0.00	22, 3	244 (439)	2, 2
Inadvertent closure of isolation valves	BOL (0)	142	150 (302)	7.86	22, 3	300 (540)	22, 3
	MOL-1 (77)	153	153 (307)	0.01	23, 3	297 (534)	1, 3
	MOL-2 (96)	156	156 (312)	0.00	23, 3	284 (510)	1, 3
	EOL (170)	157	157 (314)	0.00	22, 3	245 (441)	2, 3
Locked rotor in a coolant circulation pump	BOL (0)	147	164 (327)	16.74	23, 3	286 (515)	23, 3
	MOL-1 (77)	153	167 (333)	14.70	23, 3	283 (509)	23, 3
	MOL-2 (96)	156	171 (339)	14.80	23, 3	279 (503)	23, 3
	EOL (170)	157	171 (339)	13.69	22, 3	232 (418)	23, 3

REFERENCES – Appendix D

- D-1. *Low-Enriched Uranium Conversion Preliminary Safety Analysis Report for the University of Missouri Research Reactor*, University of Missouri Research Reactor, August 2017.
- D-2. J. A. Stillman, et al., *Accident Analyses for Conversion of the University of Missouri Research Reactor (MURR) from Highly-Enriched to Low-Enriched Uranium*, ANL/GTRI/TM-14/5, Revision 1, Argonne National Laboratory, February 2017.

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