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Summary of Properties and Fuel Performance Data for USHPRR U-10Mo LEU Conversion Analysis

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ABSTRACT

Ongoing efforts to convert research and test reactors (RTR) to operate with low-enriched uranium (LEU) rather than high-enriched uranium (HEU) fuel requires analysis for each reactor to be converted. This report presents the technical basis for several key material properties utilized in the LEU analyses, as well as a discussion of fuel performance parameters, most notably blister temperature. Collection and assessment of the data available in literature was conducted, the results of which were compared to the current values utilized for analysis in each of the U.S. High Power Research Reactors (USHPRR). There is general agreement amongst the USHPRR and literature, although key data gaps were identified where additional experimental data would be valuable. In addition to confirmation or revision of available data during qualification, of particular importance is adding to the current limited data with collection of density and thermal conductivity values from prototypic samples over the temperature and burnup ranges pertinent to the USHPRR.

1 Introduction

As part of the effort to convert USHPRR operating with HEU to LEU, analysis of the safety and performance parameters of the proposed LEU core are required. For each of the USHPRR facilities the scope of these conversion analyses is mostly limited to safety basis analysis regarding the fuel elements that will be modified for conversion. Thus, this report will present a summary of the technical basis data for key materials properties for utilization in neutronics, steady-state mechanics/thermal hydraulics, and transient analyses of proposed LEU fuel cores.

This work is appropriate where the USHPRR Conversion program maturity has transitioned by completing reactor conceptual fuel element designs as well as initial fuel fabrication development and irradiation testing [1]. With established fabrication processes, the USHPRR program is now beginning qualification irradiations with plates in 2018-19 that allow a progression to full LEU conversion fuel element irradiations in subsequent years. This qualification effort, led within the Fuel Qualification Pillar area of the USHPRR Conversion program, is now concluding a Preliminary U-Mo report [2] that provides a comprehensive compilation of best available information on U-Mo monolithic fuel, and allows assessment of areas that remain to be completed.

In tandem with the Reactor Conversion Pillar area of the program where conceptual designs for the USHPRR reactors are finished, or nearing completion, this preliminary UMo report will serve as a basis for comparison to the body of knowledge that the program will complete, or confirm, during qualification testing. The broader work underlying this paper helped not only with completion of the Preliminary U-Mo report, but provides a more comprehensive basis for updating Reactor Conversion Pillar work as new information becomes available.

Thus, a critical review of available literature data was conducted for comparison to the assumptions made in available conversion reports in order to assess the consistency and validity of these assumptions. The key materials properties that will be discussed in this summary report are density and thermal conductivity of U-Mo monolithic alloy. Additionally, data and assumptions for the blister temperature of U-Mo fuel plates will be examined.

2 Density of U-Mo Alloy Fuel

A literature review was conducted to determine the dependence of the density of U-Mo monolithic alloy fuels on Mo content. This review was limited to the data that were collected via immersion density measurements, rather than x-ray diffraction measurements. X-ray diffraction measurements determine the idealized theoretical density, while immersion density measurements include the effects of as-fabricated defects such as voids and dislocations. The results of this literature review are shown in Figure 1. The limited data include information gathered by McGeary [3], Meyer [4], and Lee [5], although the samples from Lee have high porosity, resulting in a low density value compared to the rest of the available data. The enrichment of the samples used in these experiments is not always known, but the effect of enrichment is minor, as shown by the theoretical density lines for natural uranium (NU) and low-enriched uranium (LEU). Thus, samples with a mixture of enrichments can be used to determine a correlation.

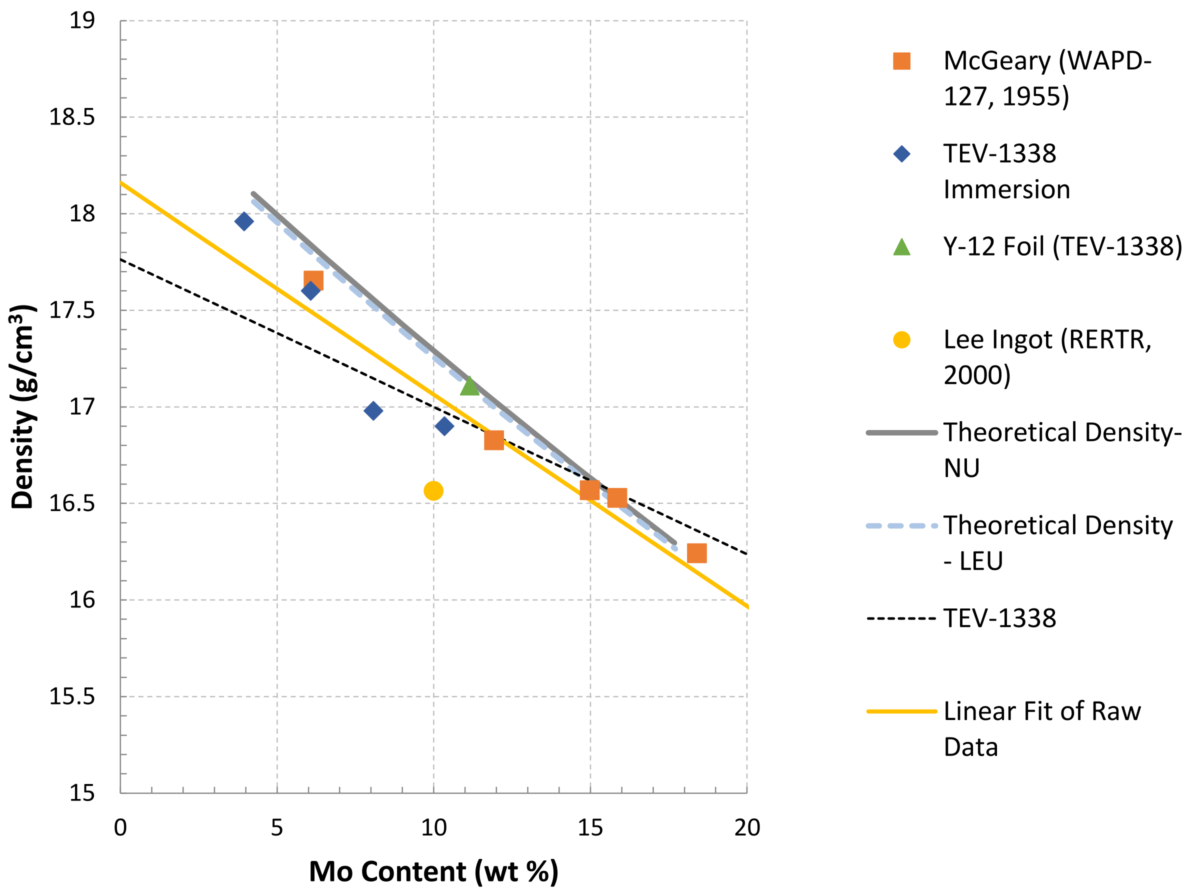

Figure 1: Dependence of the density of U-Mo alloys on Mo content.

A comparison of the assumptions made for conversion reports for all five USHPRR are shown in Figure 2, alongside the data originally shown in Figure 1 (in grey). Although not all reports assumed the same values for the density of U-10Mo fuel at room temperature, all values fall within the range of the literature values.

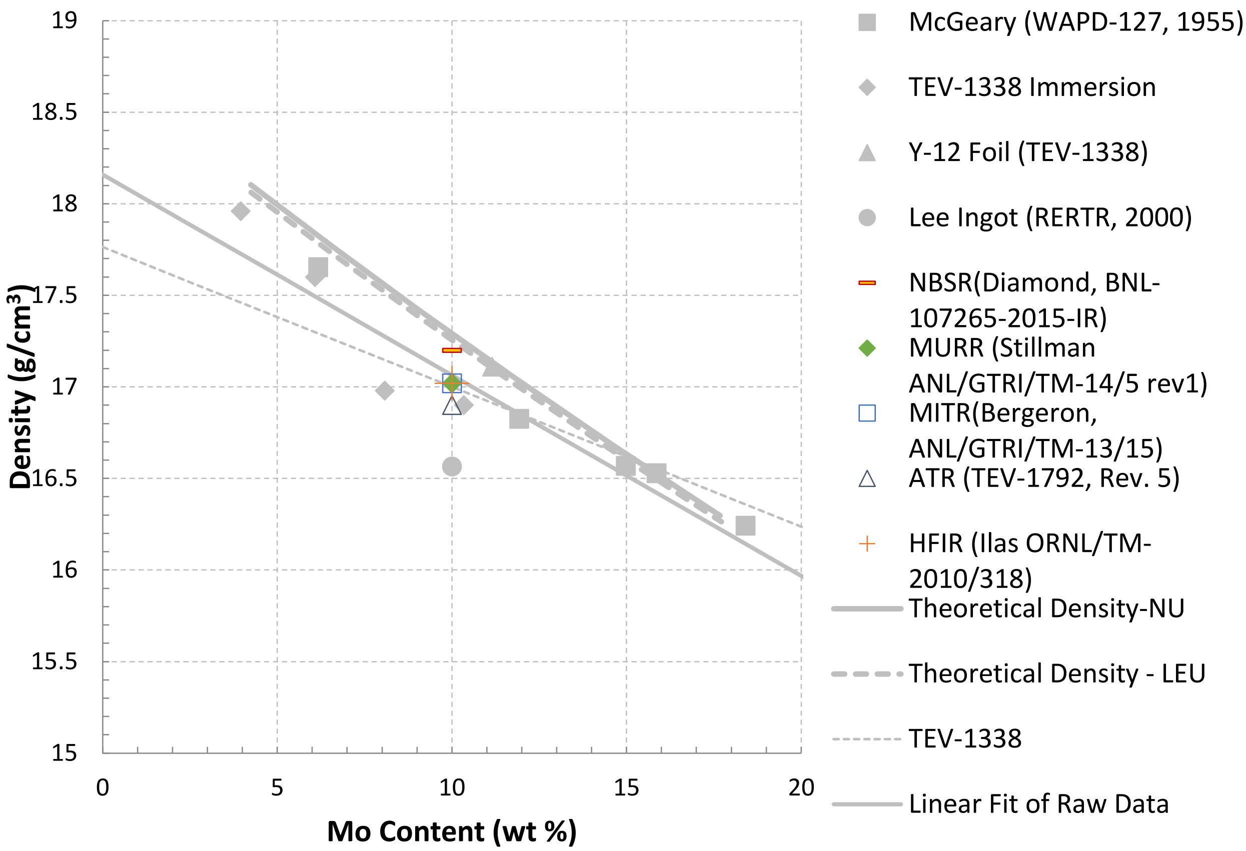

Figure 2: Comparison of U-10Mo density assumptions of the USHPRR with the experimental data.

It is notable that the reduction in density with irradiation is not directly utilized in the current safety analyses. This is since the swelling of the fuel is ordinarily directly correlated and allows analysis to account for dimensional changes to the coolant channels. Fuel swelling is an important phenomenon and significant recent progress has been made for U-10Mo monolithic alloy fuel [2n]. Future comparisons will be made to this recent work.

3 Thermal Conductivity of U-10Mo Fuel

The results of the literature review for U-10Mo fuel are shown in Figure 3. Data for thermal conductivity of U-Mo materials other than U-10Mo are shown in the figure (grey), as they were used to develop the correlation from the U-Mo handbook [6]. Although there are a fair number of data sets available for U-10Mo fuel, additional data collected on prototypic samples is desired, as the majority of the available data was collected prior to 2000, where the process knowledge of the samples is unknown. The amount of porosity and cold work in the sample affects the thermal conductivity, so collecting additional data on well-characterized samples prototypic of the current fabrication methodologies would be of great value. As noted in Section 2, the samples examined by Lee [5] contained atypically high porosity (evidenced by low density), so correction of the thermal conductivity data to account for this is required prior to incorporation of this data into any correlations.

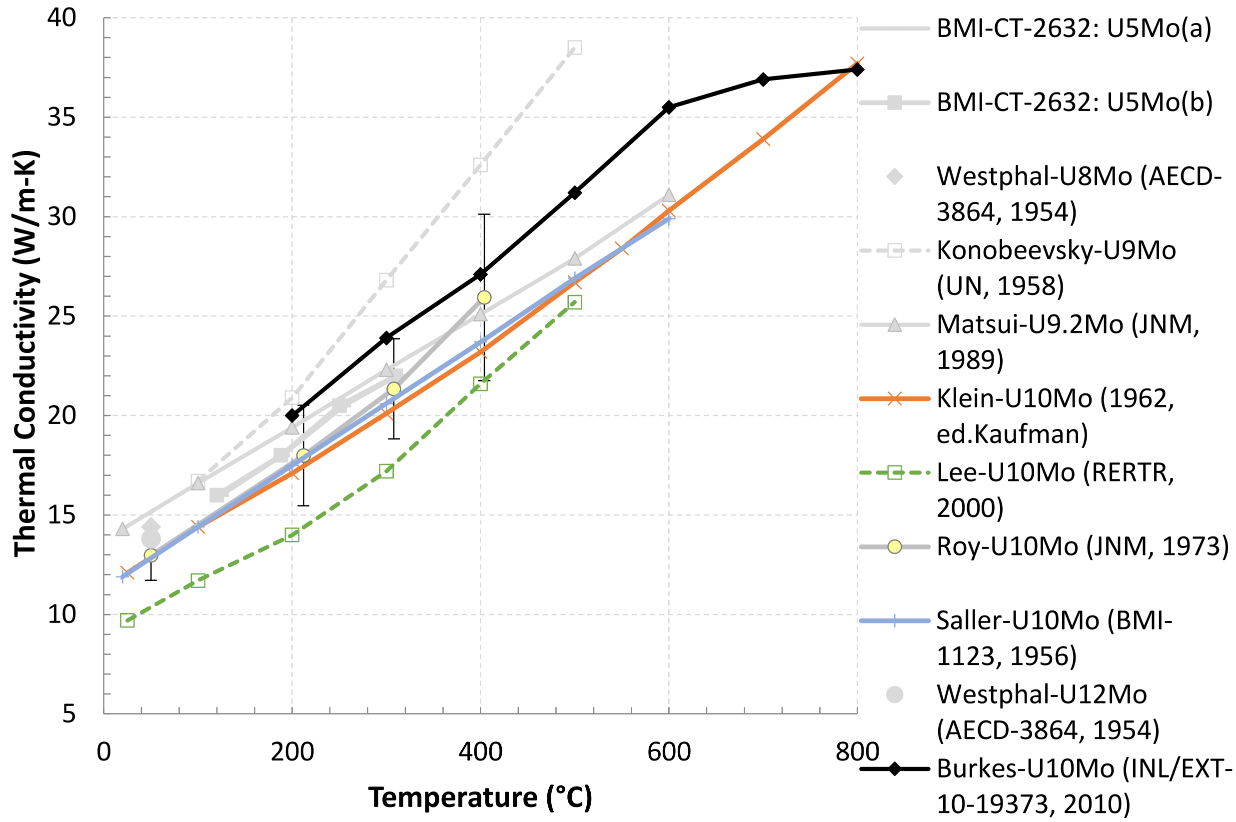

Figure 3: Thermal conductivity of U-Mo alloys temperature dependence

In contrast to the fresh fuel thermal conductivity data, there are very limited data available for irradiated fuel, as shown in Figure 3. This highlights the need for additional data on irradiated samples, preferably with the inclusion of a fresh fuel sample tested on the same equipment so that a correlation from fresh fuel to the final burnup can be collected. Collection on the same equipment will ensure the minimization of errors due to changes in the experimental setup. Although different assumptions are made for analyses of the USHPRR, all are consistent with the limited experimental data available.

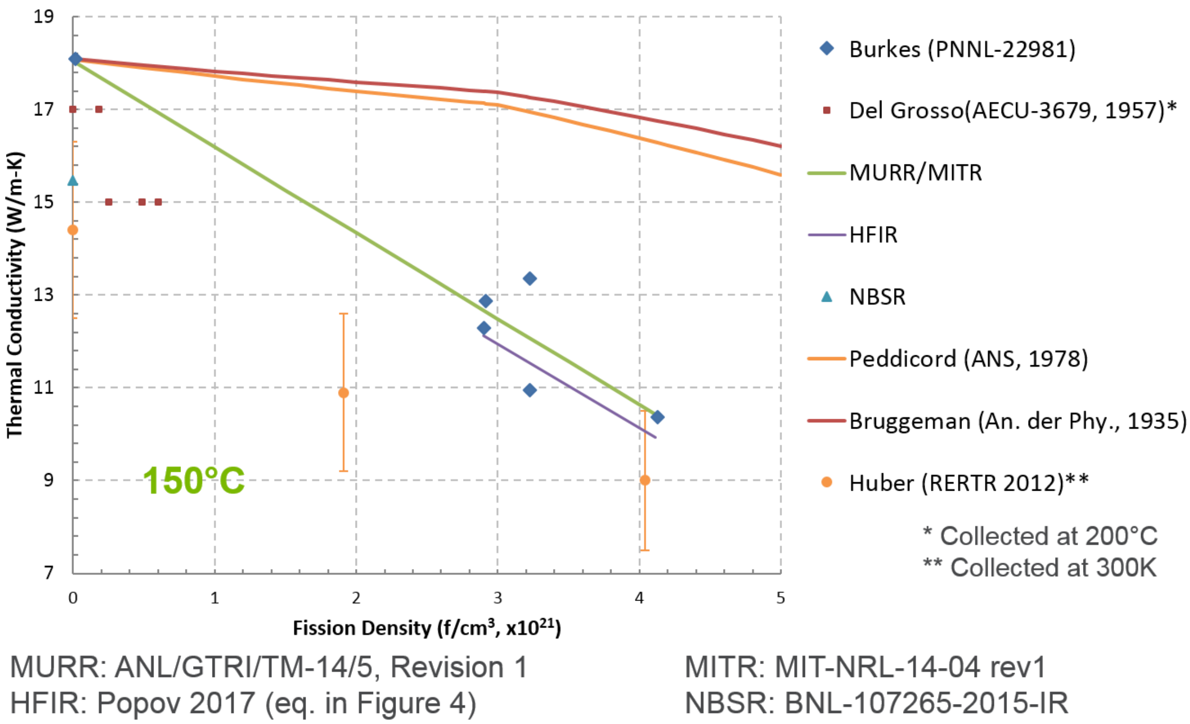

Figure 4: Irradiation dependence of the thermal conductivity of U-10Mo fuel

In the same figure, it can be seen that the simulation correlations traditionally used (Peddicord [7] and Bruggeman [8]) substantially underestimate the reduction in thermal conductivity with irradiation. As shown in Figure 5 below, even with the additional assumption of 8% as-fabricated porosity (an extreme and unlikely situation, well outside of the proposed specifications for the fuel), the thermal conductivity predicted by the models is still too high. This indicates that these models are missing key performance factors influencing the thermal conductivity.

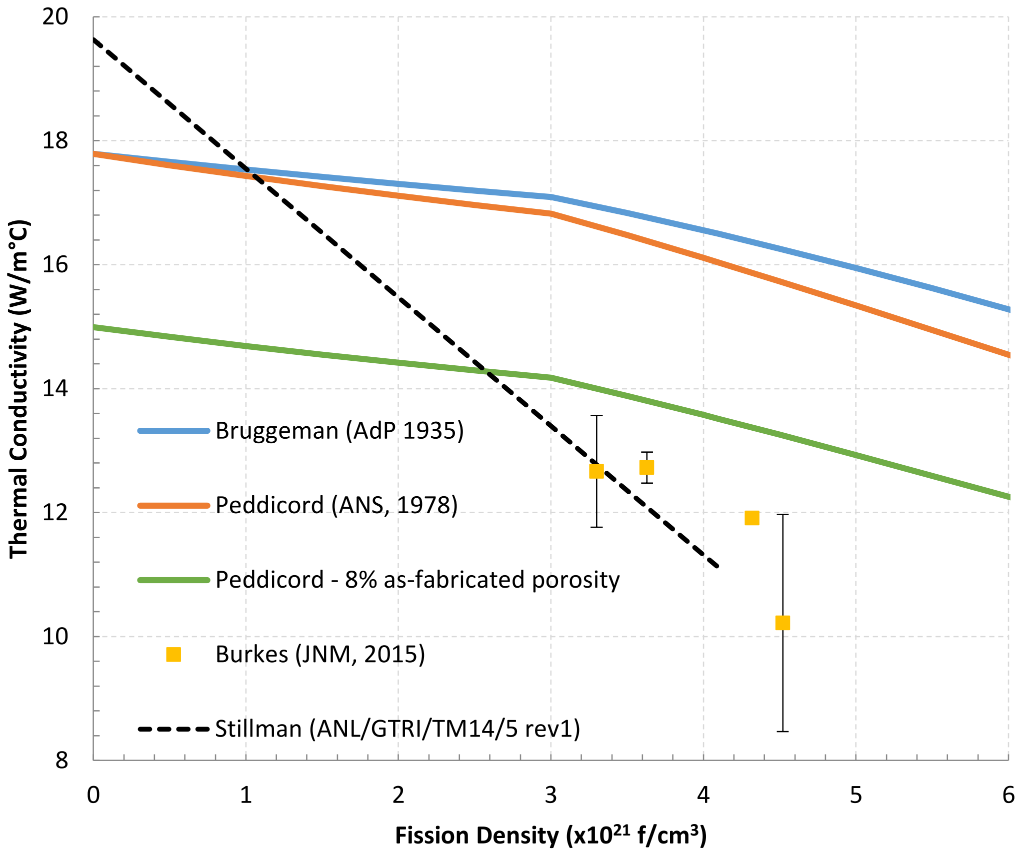

Figure 5: Thermal conductivity of U-10Mo fuel radiation dependence, 8% as-fabricated porosity addition to Peddicord model

Recently, newer irradiated thermal conductivity data has become available from the AFIP-2, AFIP-3, AFIP-6, and AFIP-6kII tests. To assess the effect the new data has on the correlations, the correlation suggested for MURR and MITR (least squares regression of the data) was re-assessed with the data from AFIP-2 and AFIP-3. The data from AFIP-6 and AFIP-6mkII were not utilized, as there was more scatter in the data. The new correlation resulted in a 14% further reduction in thermal conductivity at the highest fission density for which the correlation is valid (5.5×10^{21} fissions/cm^3). This is within the typical uncertainty allowance utilized in safety analyses, so the original fit used for safety analysis is considered acceptable.

4 Blister Temperature of LEU-10Mo Monolithic Fuel Plates

The temperature at which a fuel plate will form a blister, an immediate precursor to failure of the fuel plate integrity, is used as a bounding temperature parameter in safety analyses. Data for the blister temperature of U-Mo plates over a range of fission densities have been collected, as shown in Figure 6. A fit of this data was developed to capture the reduction in blister temperature with increasing fission density. Analyses for MURR [9] have utilized the 95% confidence interval of the fit (dashed line in the figure), while analyses for MITR [10] and NBSR [11] have utilized a single limiting temperature for the lifetime of the fuel. These assumptions are all within reasonable agreement with each other and the available experimental data. More confidence in the data fit and conservative maximum temperature assumptions would be gained by conducting additional blister tests at low and high fission densities.

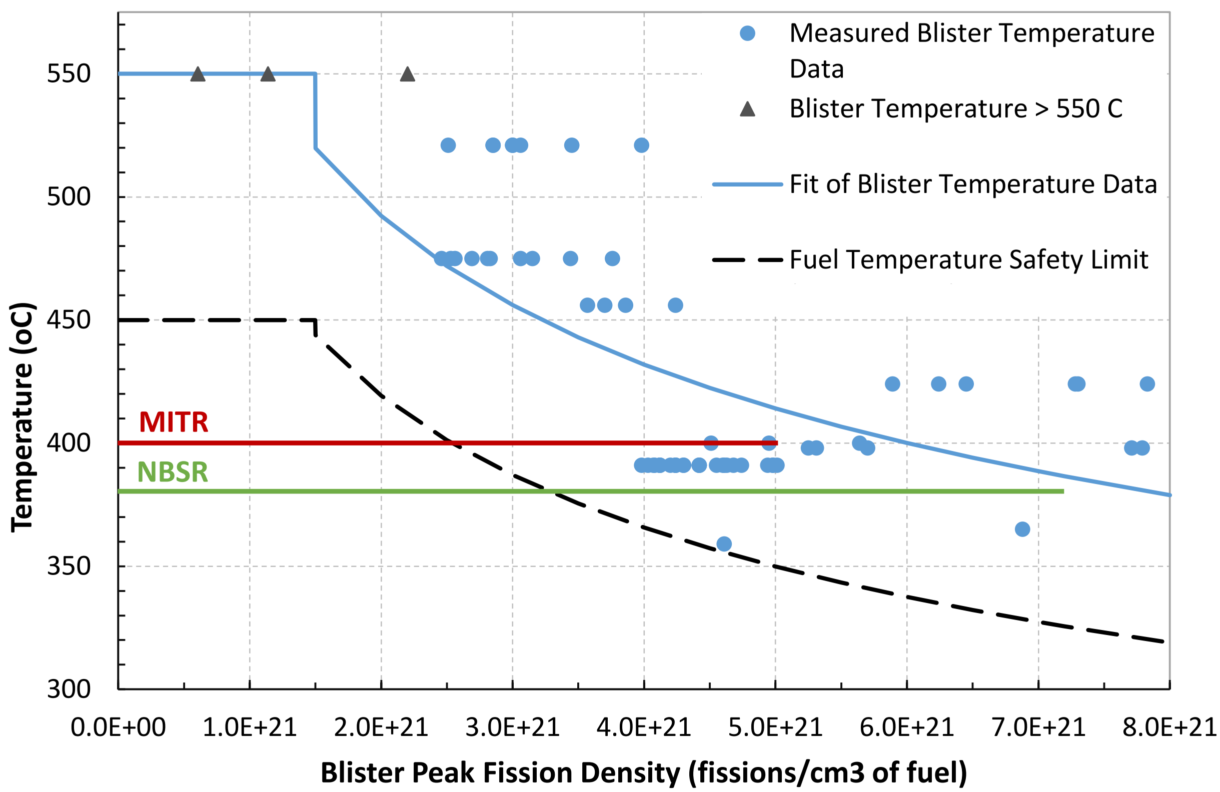

Figure 6: Blister temperature of U-Mo fuel plates with irradiation

5 Conclusions

The USHPRR Conversion program has transitioned from extensive development activities to beginning qualification testing of prototypic and bounding plates and fuel elements with established fabrication processes. This qualification testing envelope is based on the reactor-specific conceptual designs for the USHPRR reactors, which are in turn based on available U-Mo information. Thus, in preparation for later updates and safety analysis, a critical review of available literature data was conducted at this point in the program for comparison to the assumptions made in available conversion reports in order to assess consistency, and to assess gaps requiring additional data.

As a part of that broader effort, in this work, three key properties of U-Mo fuel are presented to assess the data currently available in literature and compare it to the assumptions made in the safety analyses for conversion of USHPRR. The conversion report assumptions are generally in agreement with each other and are within the range of experimental data, with some reactor-specific application of the data. However, the data available for density and thermal conductivity are limited, particularly data on irradiated fuel. Although theoretical models for the reduction in thermal conductivity due to irradiation have been proposed, they do not accurately capture the measured behavior of the fuel, so it is key to utilize empirical relations, even if they are based on limited data. In the development of new empirical fits, it is imperative that measurements of density, thermal conductivity, and blister temperature are made on both fresh and irradiated prototypic samples from the same production batch. The composition and phase of the selected batch must be within the proposed specifications. To ensure measurement consistency and to reduce uncertainties, cross-calibration measurements of fresh fuel samples in PIE facilities would be highly beneficial. Due to the manner utilized for reactor core design and safety analysis, the data collected should encompass the temperature and burnup range pertinent to the USHPRR.

6 Acknowledgement

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7 References

[1] Wilson E, Ravenhill S, Rabin B, Wight J, Dunn K., *HEU to LEU Fuel Conversion of U.S. High Performance Research Reactors: An Overview of the Fuel Development Effort*. IAEA Conf. Nuclear Security, Vienna, Austria, December 5-9, 2016.

[2] Meyer M, Rabin B, Cole J, Glagolenko I, Jones W, Jue J-F, Keiser Jr. D, Miller C, Moore G, Ozaltun H, Rice F, Robinson A, Smith J, Wachs D, Williams W, Woolstenhulme N, *Preliminary Report on U-Mo Monolithic Fuel for Research Reactors*, INL/EXT-17-40975 Idaho National Laboratory, November 2017.

[3] McGeary (ed) R, Bostrom W, Burkart M, Halteman E, Leggett R et al., *Development and Properties of Uranium-Base Alloys Corrosion Resistant in High Temperature Water*, WAPD-127, Westinghouse Electric Corporation, 1955.

[4] Meyer M, Woolstenhulme N. *Density of U-Mo Alloys*, TEV-1338, Idaho National Laboratory, 2011.

[5] Lee S-H, Kim J-C, Park J-M, Ryu H-J, Kim C-K, *An Investigation on Thermophysical Properties of U-Mo Dispersion Fuel Meats*, 2000 International Meeting on Reduced Enrichment for Research and Test Reactors, Las Vegas, NV, October 1-6, 2000. p. 261-272.

[6] Rest J, Kim YS, Hofman GL, Meyer MK, Hayes SL, *U-Mo Fuels Handbook*, ANL-09/31, Argonne National Laboratory, November 2009.

[7] Peddicord KL, Cunningham ME, Tripathi A, *Porosity correction to thermal conductivity based on analytical temperature solutions*, ANS Annual Meeting Transactions, San Diego, CA, June 18, 1978. p. 548-549.

[8] Bruggeman DAG, *Berechnung verschiedener physikalischer Konstanten von heterogenen Substanzen: I-Dielektrizitatskonstanten und Leitfahigkeiten der Mischkorprt aus isotropen Substanzen*, Annalen der Physik 1935, vol. 5(24) p. 636-664.

[9] Stillman J, Feldman E, Jaluvka D, Wilson E, Foyto L 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, rev.1, Argonne National Laboratory, 2017.

[10] Dunn F, Olson A, Wilson E, Sun K, Newton T et al. *Preliminary Accident Analysis for Conversion of the Massachusetts Institute of Technology Reactor (MITR) from Highly Enriched to Low Enriched Uranium*, ANL/GTRI/TM-13/5 Argonne National Laboratory, 2017.

[11] Diamond D, Baek J-S, Hanson A, *Conversion preliminary safety analysis report for the NIST*

research reactor, BNL-107265-2015-IR Brookhaven National Laboratory, December 2014.