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Transient Reactor Test Facility (TREAT) Low Enriched Uranium (LEU) Conversion Progress

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ABSTRACT

The National Nuclear Security Administration Office of Material Management and Minimization (NNSA M^3) is working towards converting the Transient Reactor Test Facility (TREAT) located at the Idaho National Laboratory (INL) from its original high-enriched uranium (HEU) core to a low-enriched uranium (LEU) core. TREAT reactor operations were restarted in November of 2017 and data obtained from the initial testing will be used to validate the existing ANL reactor modeling. Fuel block development has progressed with the completion of 2 and 4 inch crack free surrogate fuel blocks. Alternative fabrication methods for the fuel blocks continue to be explored and are showing promising results. Additionally, cladding oxidation studies have been performed and analyzed to initiate the down selection of cladding material to be used for the TREAT LEU fuel elements. This paper describes the current status of the above mentioned activities, and discusses the anticipated path forward for the conversion program as it progresses toward a final fuel design and fuel fabrication process.

Introduction

The Transient Reactor Test Facility (TREAT) located at the Idaho National Laboratory (INL) is a research facility that first went critical in February 1959 and has been used for more than 6,000 tests, including nearly 3,000 transient irradiation tests. The reactor was placed in standby in 1994 after operating for nearly 40 years on high-enriched uranium (HEU) fuel that achieved approximately 0.7% burnup.

The reactor's original fuel includes 312 regular fuel elements, 266 unfueled graphite reflector dummy elements, and 97 various "special" fuel elements, for a grand total of 675 elements available for insertion into a 19 x 19 element grid. The fuel section of each fuel element is composed of six blocks with 4-in. x 4-in. cross-sections and 8-in. length containing uranium oxide dispersed in graphite. The blocks are stacked to form a 4-ft fuel section encased in an evacuated Zircaloy-3 can. Aluminum-canned graphite reflectors and end-fittings are located above and below the fuel section to form fuel elements that are about 9-ft long and 100 lb in weight (Figure 1). Special purpose fuel elements that include cavities for control/transient rods, gaps for hodoscope viewing holes, and/or integral thermocouple instrumentation are also part of the core [1].

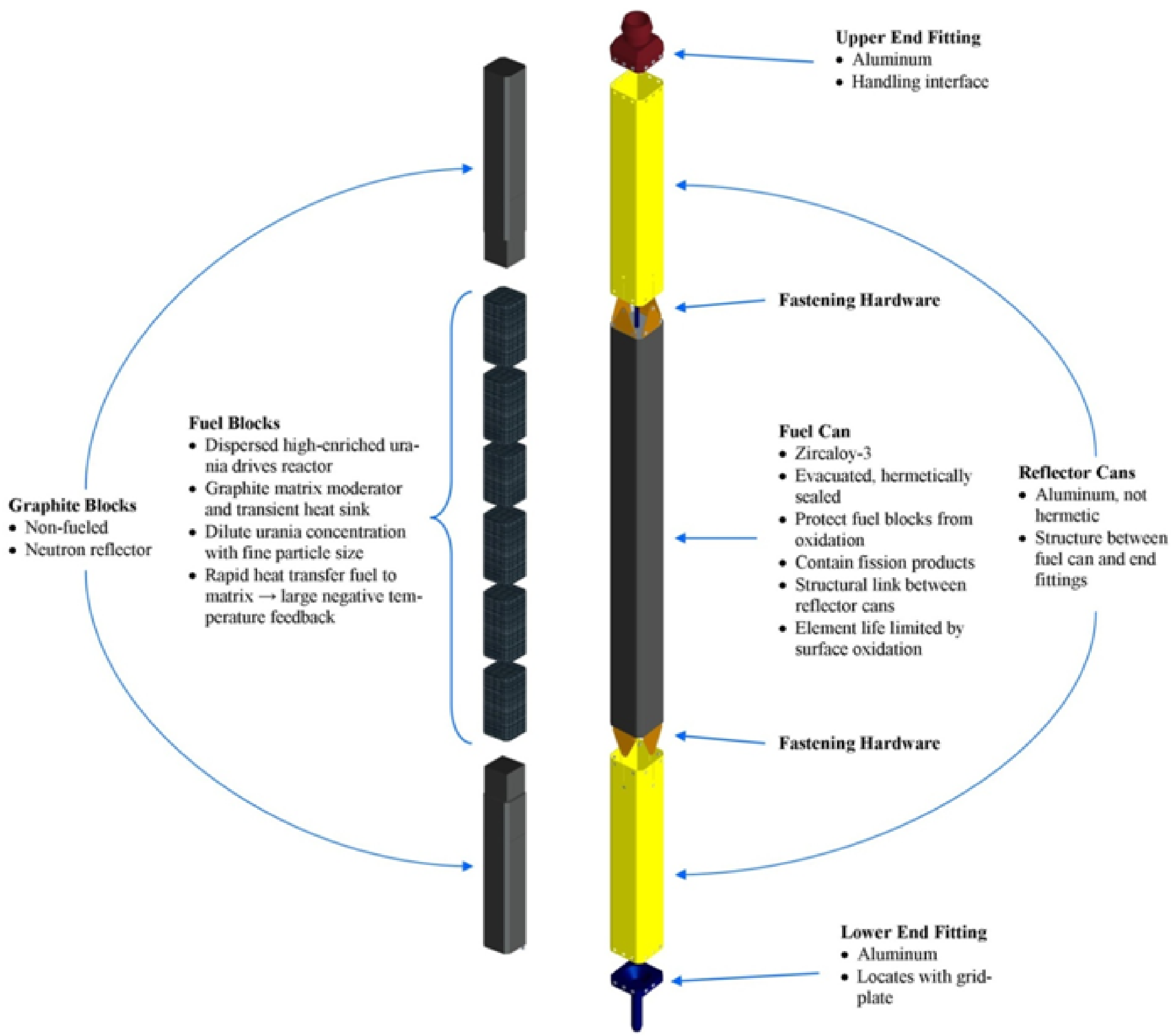

Figure 1. Assembly schematic for existing TREAT fuel elements.

TREAT is capable of safely generating high-powered transients of up to approximately 18 GW. The large mass of graphite present in the reactor's fuel blocks acts as a heat sink for transient energy as well as the primary source of neutron moderation for the reactor. The transient energy generated by fissions in the fuel's U_3O_8 particles is rapidly transferred into the graphite and temperature reactivity feedback safely terminates the transient. The reactor is also capable of 120-kW steady state power operation that is limited by the heat removal capacity of its forced-air cooling system. The reactor's air-cooled design greatly simplifies the configuration of ex-core facilities that penetrate into the core, including the fast-neutron hodoscope, neutron radiography facility, and test specimen video system (not shown), as shown in Figure 2.

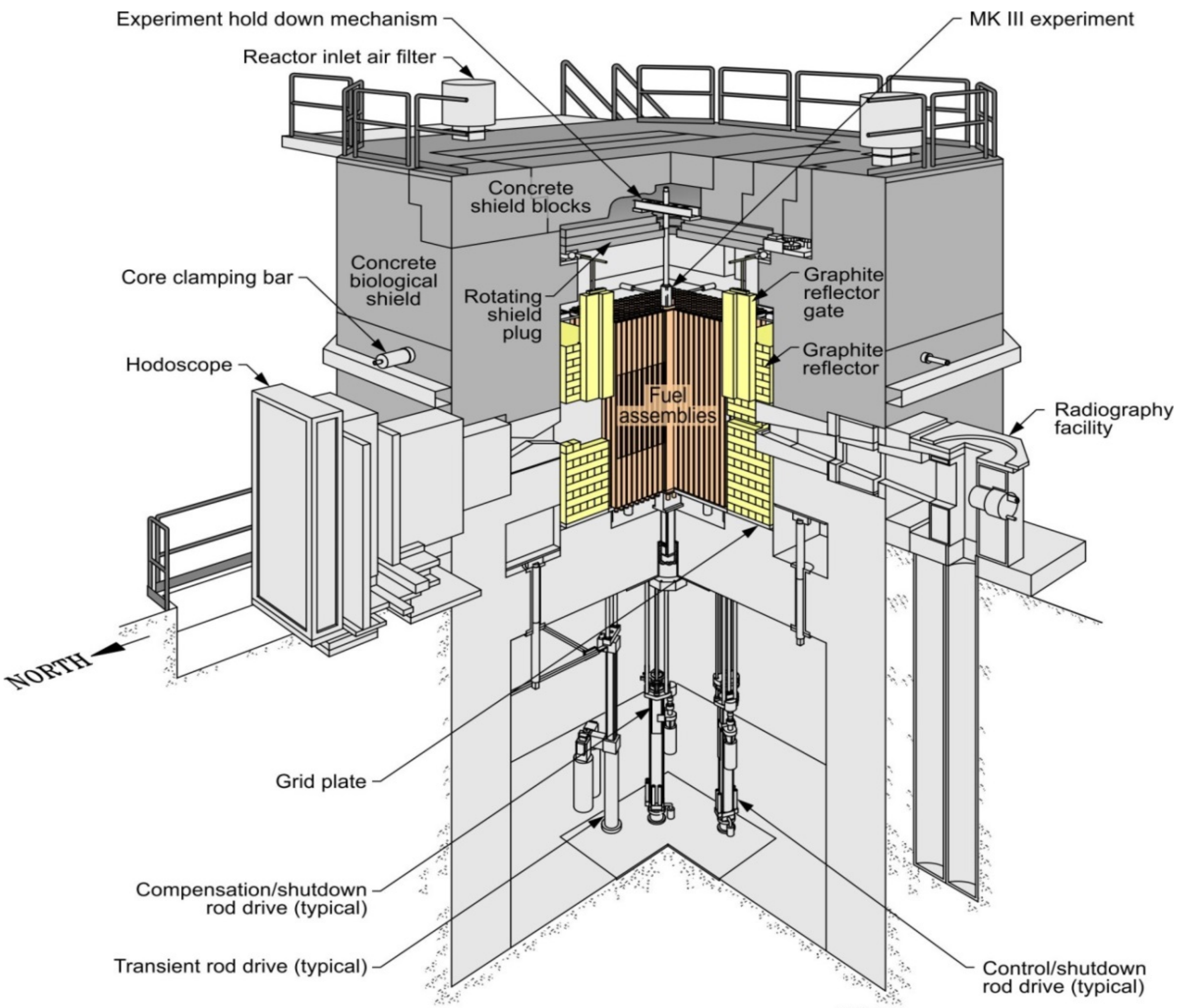

Figure 2. TREAT schematic.

TREAT LEU Conversion Modeling and Analysis

The TREAT LEU conversion program's analyses are focused on ensuring that the LEU core will be able to meet the performance requirements of the reactor, while continuing to operate safely within the existing facility. There have been several phases of TREAT LEU conversion analysis work, including extensive scoping studies to investigate the behavior of different design options [2-3], a detailed evaluation of the TREAT LEU conceptual design [4], sensitivity studies to evaluate the significance of key design parameters [5], and analysis of the more detailed behavior of the fuel-graphite matrix [6-7]. Throughout the analysis process, the calculated LEU core behavior has been evaluated in comparison to the calculated behavior of the existing HEU core.

For the first several years of the TREAT conversion program, the analysis work was constrained by the fact that TREAT was not an operating reactor. Efforts to characterize the core and to validate the analysis models and methods were limited to the work scope possible using historic (pre-1994 standby) measured data. In the fall of 2017, TREAT resumed operation after several decades of non-operational standby. The past year has therefore marked a significant turning point for analysis of TREAT. Moving forward, new measured data will provide a much firmer understanding of the TREAT core's characteristics and the reactor's performance capabilities. This data will also facilitate improved validation of the models and methods used to simulate TREAT behavior.

The TREAT LEU conversion analysis team's recent work has primarily entailed discussion and review with the reactor operations team to refine the understanding of core performance metrics, safety requirements, and other aspects of reactor behavior. As an outcome of these discussions, the convert team has begun developing more detailed technical plans which capture the complete analysis work scope needed for the LEU core. These plans will be significantly refined over the next several years as TREAT's operation provides further data and improved understanding of reactor capabilities and performance requirements. Other recent efforts included support for the planned future irradiation testing of TREAT LEU fuel material samples (TREAT Irradiation Experiment 1, TIE-1), and a more detailed assessment of historic and planned TREAT tests in order to identify proposed reference experiments for additional core performance analysis. Overviews of each of the recent areas of work are provided here.

TREAT's mission is the transient testing of nuclear reactor fuels and other materials. The performance requirements for TREAT are therefore primarily defined in terms of the core's ability to provide prototypic conditions to the test samples irradiated in the reactor. Key metrics are the achievable energy deposition in the test sample(s) and the timeframe for this energy deposition, which may drive requirements for the achievable widths of power pulses in the reactor.

Through engagement with the reactor operations team, the LEU conversion program will establish a definitive set of TREAT experiments which will form the analysis basis for demonstrating that the LEU core satisfies performance requirements. Initial reviews on this topic were performed over the past year. It is anticipated that the final reference experiment set for analysis purposes may include hypothetical tests which have not been run in the restarted TREAT core yet, but which are based on historic or anticipated future HEU core experiments. Therefore, an initial summary of the anticipated full range of potential TREAT test characteristics was recently developed.

From a safety standpoint, the primary limiting parameter in TREAT operations is the temperature limits of the TREAT fuel cladding, which constrain the allowable energy generation in a given transient. The TREAT LEU core must be capable of achieving the required test sample energy depositions while remaining within the TREAT fuel cladding temperature limits. Additional safety requirements include demonstration that the LEU core can function within the existing facility structure and systems. To support the safety basis, a detailed technical safety analysis plan is in development which will identify the necessary modeling assumptions (including core loading, operating conditions, and more) to be used for each of the safety analyses required for the facility. These assumptions will be identified over the next several years in collaboration with the TREAT facility, as understanding of the reactor grows.

An assessment of the currently-available measured data from the first year of TREAT operations is in progress. Using the data from the first year of restarted TREAT core operations, it is anticipated that the LEU conversion program will be able to validate numerous aspects of TREAT core behavior which were previously very limited with historic data, including (1) the temperature response of the TREAT fuel including cooling to ambient temperature following a transient, (2) the pulse widths achievable in the existing core with pulse clipping, and (3) additional detail on the neutronic relationship between the TREAT core and in-core test samples.

Over the past year the TREAT conversion analysis team has supported the Massachusetts Institute of Technology (MIT) feasibility study on the proposed irradiation of TREAT LEU fuel samples in the MIT Nuclear Reactor (MITR), primarily by providing summary information on the calculated characteristics of the TREAT LEU core. Key parameters of interest have included the peak and core-average results for neutron flux, fuel fission density, and relative fast-to-total flux. This information is relevant in assessing how the relative fission fragment damage vs. fast neutron damage in LEU samples irradiated in MITR compares to the predicted relative damage in the LEU fuel in TREAT. Due to the different energy spectrums in the two reactors (TREAT has a more thermal spectrum), the proportional damage may differ. A decision was made to prioritize matching fission density, as this will provide more conservative experimental results. These analyses will be used in the future to further support TIE-1 planning when the TREAT convert program resumes work on this experiment.

The near-term focus of the analysis team will be continued engagement with the TREAT facility in order to further refine core performance requirements, including further validation of the neutronic and thermal hydraulic analysis models using new measured data from TREAT. In the future, when the capabilities of the HEU core have been further characterized in detail and the performance requirements of the reactor are finalized, a full updated assessment of predicted LEU core behavior will be performed.

Fuel Block Development

The TREAT fuel blocks were originally created by a compaction pressing process utilizing graphite, a thermoset resin and U_3O_8 to a density of 1.73 g/cc. The path for manufacturing TREAT fuel blocks was lost over the years due to the nature of the TREAT reactor, the low fuel burnup of 0.7% over 40 years [8], and the lack of need for replacement fuel elements. The first undertaking for trying to create LEU fuel assembly elements was recreating the fabrication process for the fuel blocks. Due to the fact of uranium enrichment being 19.75% for the completed fuel blocks, limited

fuel manufacturing professionals are capable of fabricating the blocks. BWX Technologies, Inc. (BWXT) was ultimately selected to help develop and reestablish the fabrication path for the fuel blocks back in 2014.
The method for manufacturing the blocks was established by jet milling feedstock (graphite, phenolic resin, hardener), adding the oxide via ribbon blender (or surrogate material for developmental purposes), granulating to increase powder tap density in turn decreasing die fill volume, warm pressing at 160°C, and finally heat treat the blocks to carbonize the phenolic binder. [9] Figure 3 below shows the ribbon blender and granulator and Figure 4 shows the warm press and heat treatment furnace.

Figure 3: Left – ribbon blender; center – inside ribbon blender; right – granulator

Figure 4: Left – warm press; right – heat treatment furnace

Early trials started out with one inch diameter pellets with surrogate material and have progressed up to 4-in. x 4-in. x 4-in. surrogate blocks over the years. Figure 5 below shows fuel pellet and blocks progression.

Figure 5: Left – fuel pellet; right – fuel block progression

It was discovered that the generation of volatiles during the heat treatment process and lack of diffusion path were the leading cause for fuel block cracking, so various methods were undertaken to identify methods to allow the volatiles to escape without causing cracking [10]. The best solution was to limit the maximum pressing density to 1.83 g/cc and prolong the heat treatment cycle at a much slower rate (2˚C/hr). Confidence has been established in pressing crack free 2-in. x 2-in. x 2-in. blocks and the progression to 4-in. x 4-in. x 4-in. blocks. 4-in. blocks have been promising, however repeatability has been a little challenging due to varying aspects between the granulation process and the compaction process. Recent pressings need to be expanded to have confidence in repeatability and will be continued until confidence has been established.

In addition to the standard warm pressing method, alternative methods for fabricating the fuel blocks are being explored, spark plasma sintering and vacuum heat pressing. These alternative methods, if proven successful, will streamline the fabrication process and reduce cost by eliminating process steps of jet milling and granulation and combining the pressing and heat treatment steps. Further no organics need be introduced (resin and hardener). Increased performance characteristics are possible due to the increased density from 1.83 g/cc to greater than 2.0 g/cc. Initial one inch cylinders had cracks, however after tweaks to the die design and process parameters, crack free samples up to two-inch diameter have been fabricated [11]. Further development will continue during FY19 and throughout the standby period. Figure 6 below shows initial samples and crack free samples.

Figure 6: Left – cracked SPS sample; right – crack free SPS sample

Cladding Oxidation Studies

The original TREAT fuel element utilized Zircaloy-3 (Zr-3) as the cladding. Unfortunately, Zr-3 is no longer commercially available and would require substantial capital to reestablish a commercial Zr-3 fabrication line, so additional cladding materials are being explored for other viable options. Considering the unique nature of the TREAT reactor being an air-cooled reactor, three zirconium alloys, Zircaloy-3, Zircaloy-4 and Zr-1Nb, have been proposed as the most promising cladding candidates.

Oxidation studies performed at INL and Boise State University of Zircaloy-3, Zircaloy-4 and Zr-1Nb alloys have resulted in these main findings:

(1) Zircaloy-4 experiences breakaway the soonest, while Zr-1Nb is the most resistant to breakaway. Zircaloy-4 also oxidizes faster than both Zircaloy-3 and Zr-1Nb. The latter two have similar oxidation rates to each other, both before and after breakaway.

(2) Zircaloy-4 is most affected by rapid heating and experiences accelerated oxidation at the chamfers after two thermal cycles of 600 °C transient with maximum thermal resistance. Zircaloy-3 and Zr-1Nb maintain their protective oxide after two 600 °C transient cycles.

(3) The overall oxidation kinetics of Zircaloy-4 are significantly affected by plastic deformation whereas Zircarloy-3 and Zr-1Nb are much less affected. However, visually, all of the zirconium alloys experience accelerated oxidation in the plastically deformed and weld regions.

It is preliminarily concluded that Zr-1Nb would outperform Zircaloy-4 and would perform similarly to Zircaloy-3 in prototypic oxidation and transient conditions of the TREAT reactor. Oxidation validation studies will need to be conducted to confirm Zr-1Nb as the most viable, cost effective candidate once standby mode has concluded. Figure 7 below shows the effects of oxidation on different welded sections as well as humid air (Electron beam welding (EBW) and Tungsten Inert Gas (TIG)). Figure 8 shows weight gain from oxidation on the weld affected areas and humid air. [12]

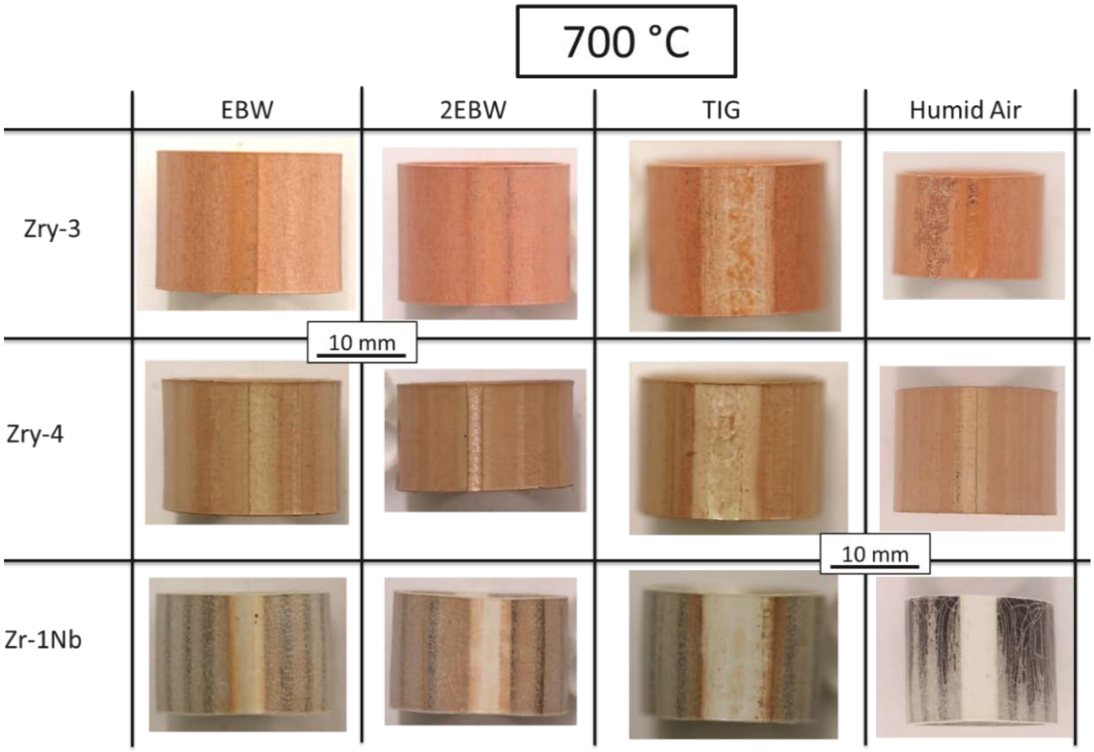

Figure 7: Oxidation on weld affected areas and humid air

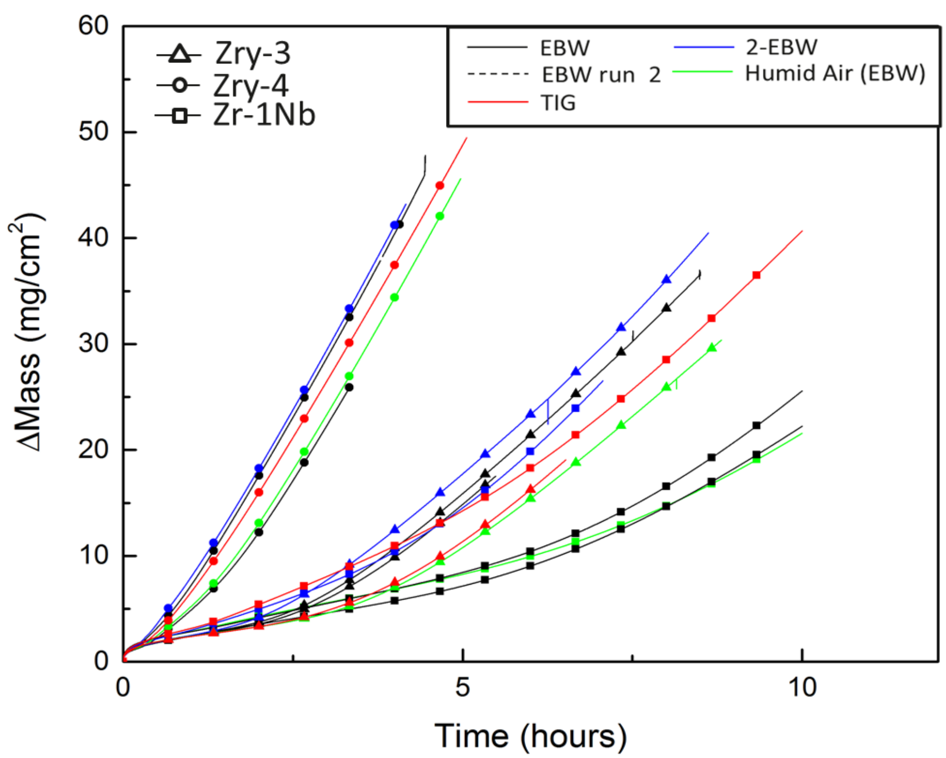

Figure 8: Mass gain due to oxidation on weld affected areas and humid air

Conclusion

Reactor operational data is extremely valuable in reactor validation modeling. Since the restart of the TREAT reactor in November of 2017, valuable data has been shared with the LEU conversion program for modeling and analysis validation. Additional data will be shared throughout the standby mode and beyond for the various experiments that are anticipated to be the focus for the TREAT reactor allowing the LEU conversion program to finalize operational and performance requirements of the LEU fuel elements.

Reestablishing the fabrication lines for the TREAT fuel elements is obviously imperative to converting the TREAT reactor from HEU to LEU especially fuel block fabrication. Inroads have been made for the fuel block fabrication with newer technologies possibly providing more cost-effective streamline processes, however fabrication repeatability for the desired specifications and dimensions still need to be verified. Over the standby mode, continued development with a goal of repeatability will continue until confidence is established for crack free block fabrication. It is anticipated confidence of repeatability will come sometime after the standby mode once the program has ramped up again.

Oxidation of the cladding has been one of the primary concerns with converting the TREAT reactor from HEU to LEU due to possible higher operational temperatures. With Zr-3 no longer being a commercially available cladding material and without substantial capital cost to reestablish a commercial line, other cladding candidates were examined as possible alternates for the original Zr-3 cladding. Studies were performed on Zr-3, Zr-4, and Zr-1Nb and results were compared demonstrating that Zr-1Nb was a better candidate than Zr-4 and performed similarly to Zr-3. Validation of these results will be performed at a later date most likely after standby mode, however Zr-1Nb is believed to be the candidate of choice and most likely the cladding of the LEU fuel elements.

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