BNL-114531-2017-FORE

Review of the TREAT Conversion Conceptual Design and Fuel Qualification Plan

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December 2017

Nuclear Science and Technology Department

Brookhaven National Laboratory

U.S. Department of Energy

USDOE National Nuclear Security Administration (NNSA),

USDOE National Nuclear Security Administration (NNSA), Office of Nonproliferation and Verification Research and Development (NA-22)

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BNL-114531-2017

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Manuscript Completed:

November 20, 2017

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EXECUTIVE SUMMARY

The National Nuclear Security Administration (NNSA) Office of Material Management and Minimization Reactor Conversion Program is supporting the conversion of the TREAT reactor from using high-enriched uranium (HEU) fuel to using low-enriched uranium (LEU) fuel. This report documents conclusions and recommendations by a team reviewing the conceptual design as documented in two key reports discussing the following subjects:

- the proposed fuel element design and fabrication method
- the fuel qualification plan
- the neutronic calculations to demonstrate the same operational capability as the present HEU core
- the neutronic calculations to demonstrate safety and provide input to thermal analysis
- the thermal analysis to demonstrate safety with respect to temperature and corrosion limits
- the thermal structural analysis to demonstrate that any distortion of the grid plate supporting the fuel would not lead to the inability of control rods to move into the core

The review team found the work done on the conceptual design to be of high quality and clearly documented. The design will provide an LEU core with qualified fuel elements that satisfy operational and safety requirements. The recommendations from the review are meant to assist future work at the preliminary design stage prior to the final design stage. It is recognized that some of these recommendations may already be a part of plans not known to the review team. The most important conclusions and recommendations are summarized below.

Fuel Design, Fabrication, and Qualification

1. The fast neutron fluence in the LEU core should be identified so that requirements for irradiation testing for fuel qualification can be better defined.

2. In a related report, the thermodynamic analyses of the reduction of UO_2 and U_3O_8 by carbon were found to be incorrect and hence, the analysis should be redone as this affects the understanding of the chemical composition of the fuel.

3. The plan for fabricating the fuel elements is considered satisfactory. This includes the use of 4-inch by 4-inch fuel blocks and the use of thin cladding that will collapse to a certain extent. It is also good that pillowing will not be an issue for these fuel elements. If changes are made to this fabrication plan, additional review will be beneficial.

4. The fuel qualification plan is thorough and sufficient for the current state of development. Considerable additional detail will be required to plan and execute the irradiation testing. As proposed by the conversion program, the use of the ATR, MITR, and TREAT reactors is an excellent way to perform different irradiation testing.

5. The multiple transient tests along with post-irradiation examinations being proposed are expected to provide valuable qualification data.

6. Given that it is not feasible to accurately test for the effects of 40 years of transient operations, a surveillance program for the LEU core is recommended. Surveillance will provide an indication of deterioration of fuel block properties prior to potential fuel failure.

Neutronic, Thermal, and Structural Analysis

1. It is recommended that key requirements for energy deposition, maximum power, shutdown margin, etc., be well established before the preliminary design is defined. Currently, key requirements are in a state of flux.

2. If a new neutronics model needs to be developed for radiography applications, a new review of the model should be undertaken.

3. There are uncertainties in the core composition that are inevitable as the fuel specification evolves and hence, it is recommended that the resulting variation in important parameters be quantified and documented.

4. Validation is an essential part of model development and it is recommended that new data, to be collected from TREAT after its restart in 2018, be used to validate the models to be used for the upcoming preliminary design.

5. Decay heat is not taken into account in calculating cooling times and although the fuel has low burnup, it is recommended that the effect be quantified to be assured that the assumption is valid.

6. It is recommended that the conversion design team review the calculational methodology being used by the TREAT restart team in order to see if there are any benefits to changes in the current methodology.

7. The model for calculating the thermal deformation of the grid plate was found to be acceptable based on a review of the discretization of the finite elements, the use of symmetry, and the boundary conditions used.

8. Since only certain locations for control rods were evaluated, it is recommended that some control on placement of rods be set unless a new analysis is undertaken for the unanalyzed worst-case location.

9. The model to analyze binding of a control rod was found to be lacking with respect to several features. Since the validity of the assumptions made in the model were not evident, it is recommended that the technical basis be developed to demonstrate that the modeling and analysis approach is justified, or consideration be given to an alternative methodology.

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ACKNOWLEDGEMENTS

The review committee is indebted to the staff from various national laboratories that participated in the telecoms providing answers to the many questions from the committee as well as additional relevant information. In particular, we appreciate the cooperation of Heather Connaway and her staff at Argonne National Laboratory. The committee also appreciates the support from Douglas Burns, the TREAT Conversion Program Manager/Technical Lead, and his staff at Idaho National Laboratory.

1 INTRODUCTION

1.1 Background[a]

The U.S. Department of Energy (DOE) is preparing to re-establish the capability to conduct transient testing of nuclear fuels at the Idaho National Laboratory (INL) Transient Reactor Test (TREAT) facility. The original TREAT core went critical in February 1959 and operated for more than 6,000 reactor startups before plant operations were suspended in 1994. DOE is now planning to restart the reactor using the plant's original high-enriched uranium (HEU) fuel. At the same time, the National Nuclear Security Administration (NNSA) Office of Material Management and Minimization Reactor Conversion Program is supporting analyses and fuel fabrication studies that will allow for reactor conversion to low-enriched uranium (LEU) fuel (i.e., fuel with less than 20% by weight ^{235}U content) after plant restart. The TREAT Conversion Program's objectives are to perform the design work necessary to generate an LEU replacement core, to restore the capability to fabricate TREAT fuel element assemblies, and to implement the physical and operational changes required to convert the TREAT facility to use LEU fuel.

1.2 Objectives

The objective of the TREAT Fuel Conversion Technical Review Team is to provide an independent assessment of the *conceptual* LEU fuel design and fuel qualification plan developed by the conversion program prior to October 2016.[b] This will be a help in working toward the next step, a *preliminary* design. Specifically, the charter for the review team states the following:

- *Review the Conceptual Design Report [1] for the TREAT Conversion Program and evaluate the adequacy of the program's fuel conceptual design.* The review team should focus on assessment of the fuel design, analysis, and fabrication information presented in Section 5 of the report, and make recommendations on process and design improvements that should be implemented by the conversion project. The Conceptual Design Report refers to a large number of other reports and analyses that form the basis for design decisions, and although the review team should consider those references when relevant, the primary focus of the review should be the Conceptual Design Report. The review team should work with the conversion program's fuel design and assessment lead and the program's fuel fabrication lead to determine if recommended analyses and design improvements have already been implemented prior to finalizing the team's recommendations.

- *Review the TREAT Conversion Fuel Qualification Plan [2] and evaluate its adequacy*. The review team should develop specific recommendations for in-pile and out-of-pile tests that should be performed on materials that may be used in the LEU core. Graphite fuel and cladding material tests that have been performed by other reactor development

[a] Background and other introductory remarks are based on the charter received by the Review Team from INL.
[b] Analyses after this date are likely to only be in draft form.

programs (e.g., the Advanced Gas Reactor Program) and historic TREAT fuel design activities should be considered during development of review team recommendations.

1.3 Methodology

A kickoff meeting was held via telecom in April 2017 with the review team and staff from different national laboratories that are contributing to the conversion program. The review team had access to relevant references by requesting that they be placed on a SharePoint site. Requests for more information on the salient topics were provided in writing and four additional telecoms took place with subject matter experts who provided responses to the comments and questions of the review team. This was felt to be sufficient for this review; no written responses were requested.

Chapter 2 provides the results of the review of the fuel design, qualification, and fabrication and hence, covers both Section 5 of reference [1] and all of reference [2]. Chapter 3 provides the results of the review of the neutronic, thermal, and structural analysis discussed in Section 5 of reference [1]. The comments and questions to the conversion design team that form the basis for the conclusions and recommendations in Chapters 2 and 3 are provided in Appendices A and B, respectively. Chapter 4 contains references for both the body of the report and the appendices.

2 FUEL DESIGN, QUALIFICATION, AND FABRICATION

2.1 Introduction

Fuel conceptual design, qualification, and fabrication are reviewed based primarily on the conceptual design report [1] and the fuel qualification plan [2]. Supporting documents made available to the review team were helpful, especially reference [3]. References [4], [5], and [6] were also considered. Discussions with TREAT program participants resolved many comments and questions from the review committee. Two sets of comments and questions submitted to the TREAT program prior to teleconferences with program participants are provided in Appendix A to document the subject matter discussed. All three of the initiatives are in an early stage of development so the following recommendations are necessarily general in nature.

2.2 Fuel Conceptual Design

1. The review committee discovered that the thermodynamic analyses of the reduction of UO_2 and U_3O_8 by carbon reported in reference [4] are incorrect and recommended that these analyses be redone. We understand this recommendation is being implemented.

2. It is understood that the burnup in the original HEU TREAT core is very low (0.7% U-235) and the burnup required for the LEU core is similarly low: 7.6 MJ lifetime energy production, equivalent to 81 MWd. It is recommended that as the core design matures, burnup also be expressed in units more commonly used for UO_2 fuels in power reactors such as MWd/MTU so that performance parameters such as fission product inventory and fuel swelling can be related to experience in light water reactor fuels. It will also be informative to know the fast neutron fluence anticipated for the LEU core to define requirements for irradiation fuel qualification testing,

2.3 Fuel Qualification

1. The fuel qualification plan, which consists of out-of-pile tests with post-test measurements and steady-state and transient irradiations with post-irradiation examinations, is thorough and sufficient for the current state of development. Important objectives such as measurement of the decrease of thermal conductivity in the fuel block with burnup (and fast neutron fluence), fuel block dimensional changes, and fuel particle microstructure, especially fission gas morphology, are included. Considerable additional detail will be required to plan and execute the irradiation testing; for example, detailed requirements such as test articles and sizes, test objectives, test temperatures, atmosphere (inert, air, stagnant, flowing), burnup, and neutron spectrum, flux and fluence. The review committee is pleased to hear that irradiations are being considered in the MITR reactor in addition to the use of the rabbit facility in the ATR reactor, as experiments in MITR could provide much better control of conditions, such as temperature. The plan to conduct transient testing on LEU fuel assemblies in the HEU TREAT core is also reasonable.

2. The transient testing plan calls for three tests of one LEU Lead Test Assembly in each. Although a detailed transient test plan is not yet completed, it is expected that each test will

consist of multiple transients. We support multiple transients for each test and concur with the post-irradiation examination plans that call for both non-destructive and destructive examinations, including microstructure of the fuel particles within the fuel blocks.

3. Given that it is not feasible to accurately test for the effects of 40 years of transient operations, a surveillance program for the LEU core is recommended. This should be designed and performed to periodically monitor the most highly stressed fuel assemblies. The primary evaluation could be a visual examination for fuel can oxidation and structural integrity. Oxide thickness could also be nondestructively measured to assess the accuracy of the corrosion predictions. At one or more points during the 40-year lifetime, destructive evaluation of a fuel assembly could be performed to evaluate fuel block integrity, thermal conductivity, fuel particle microstructure, and cladding integrity. Surveillance of this nature should provide an indication of deterioration of fuel assembly properties prior to potential fuel failure.

2.4 Fuel Block and Element Fabrication

Concerns on fuel fabrication, as covered in Set 2 of the questions in Appendix A below, were addressed in a teleconference with appropriate staff. In particular:

1. Progress is being made in fabrication of the 4-inch by 4-inch fuel blocks and the team seems confident that blocks of this size meeting the fuel requirements can be achieved and therefore, no alternative designs are currently being considered.

2. The fabrication of full-length thin-walled elements that will collapse to a certain extent on the fuel blocks is the only design being considered at this time.

3. Pillowing in the original HEU core was determined to be due to an in-leakage of air at a thermocouple seal. The design has been changed and pillowing is no longer considered a possibility.

In the event that alternative designs must again be considered, the questions in Appendix A remain.

3 ANALYSIS FOR THE CONCEPTUAL DESIGN

3.1 Introduction

The neutronic, thermal, and structural analyses summarized in Section 5.1.2 of the conceptual design report [1] are primarily based on the work done at Argonne National Laboratory (ANL) reported in reference [3]. The latter report documents the large number of calculations that demonstrate that the LEU core will be able to achieve the same sample irradiations as the HEU core while satisfying the many safety requirements. These requirements include the limits on temperature for normal accident transients, the limit on corrosion as a result of high temperature during an accident and the limit on thermal distortion of the grid plate that supports the fuel elements. The original comments and questions from the review committee, found in Appendix B, were, for the most part, satisfactorily answered during telecom discussions with the appropriate staff—primarily from ANL. The following sections summarize conclusions and recommendations from those discussions.

3.2 Neutronic and Thermal Analysis

1. "Key requirements" for the LEU core were originally documented in reference [7]. A subset, found in the conceptual design report [1] (Section 4.1 and Appendix A), relate to the neutronic analysis. These are important to define the design but were not all taken into account for the conceptual design. In part, this is because some are being changed and new ones are being defined. It is recommended that these functional requirements be well-established before the preliminary design is defined. (Note that this recommendation applies not just to neutronics/thermal issues.) At that time, it is also recommended that the requirements be listed in a table with each next to a statement explaining how the requirement was met.

2. One of the original requirements for the LEU core (but not in play currently) is to produce a radiography beam equivalent to that for the HEU core. If this requirement resurfaces for the preliminary design then the neutronic model for the core may need to be modified. Since that new model would not have received as much attention as the model for sample irradiations in the core, a separate review of that model is recommended.

3. There are uncertainties in the core composition that are inevitable. These include the amount of UO_2 vs U_3O_8, the amount of impurities, and the amount of carbon in the form of graphite. Although these variations may only make small differences in key parameters, it is recommended that the effect be quantified and documented.

4. Validation is always an essential part of model development and the ANL team have benchmarked their MCNP and TREKIN models to the extent possible [8] [9]. However, the existing data from relatively old TREAT measurements have unknown uncertainties (e.g., uncertainty in neutron detector response due to a lack of data on core temperature and control rod position). Hence, it is recommended that new data, to be collected from the existing reactor after its restart, be used to further validate the models to be used for the preliminary conversion design. (Those models may be improvements over the existing models due to the

availability of the data; for example, they may have an increased number of radial and axial zones and more information on the distribution of air flow in the core.)

5. The review team noted that decay heat is not taken into account in calculating cooling times. This assumption is reasonable based on the low burnup of the fuel. However, it is recommended that the effect be quantified to be assured that the assumption is valid.

6. It is recommended that the ANL conversion design team review the neutronic and thermal calculational methodology being used at INL for restart, operation, and safety of the present core in TREAT. As a result of that review, there may be benefit to modifying the current methodology.

7. It is recommended that the design team use applicable experimental data to validate their model of the effects of UO_2 particle size on a) the thermal conductivity degradation due to burnup and b) the overall thermal resistance of the fuel matrix system.

3.3 Thermal Structural Analysis

Section 5.1.2, Conceptual Design Analysis, and Section 5.1.2.5, Ancillary Safety Analysis, in the INL conceptual design report [1] identify studies performed for the LEU conceptual design core and refer to an ANL report [3] for full details of the studies performed. With regard to the grid plate, Section 4.3, Grid Plate Thermal Distortion Analysis, in the ANL report [3] was reviewed by the team. The purpose of the thermal structural analysis was to evaluate whether accident conditions in the TREAT LEU core could result in excessive thermal distortion (deformation) in the grid plate that supports the fuel assemblies. The resulting deformation of the grid plate from the thermal analysis was used to perform a structural analysis of the control rods in order to demonstrate that the control rods will not deflect to the point which would cause them to bind with either the guide tube or seal assembly—a result that could affect their ability to control the reactor. In addition, the analysis calculated the maximum acceptable uniform temperature in the grid plate that would result in a thermal distortion that potentially could cause control rod binding.

1. The non-uniform temperature distribution, discussed in Section 4.2 of the ANL report [3], was used as input to the grid plate thermal structural analysis. The first step in the thermal structural analysis was to determine the horizontal thermal deformation of the grid plate due to the accident temperature condition. The thermal deformation of the plate at various locations was determined based on a finite element model of the grid plate analyzed using the ANSYS computer code. As indicated in Appendix B, Thermal Structural Analysis, in this report, a number of questions and requests for additional information were made. One of these was to request that a figure be provided that shows the finite element model of the grid plate used in the thermal structural analysis. This model was provided to the review team, which concluded that the finite elements in the model were sufficiently discretized to obtain a reasonable set of results at the various regions of interest. Furthermore, the use of a quarter model of the grid plate is acceptable due to symmetry along the two vertical planes 90 degrees apart. The boundary conditions of the grid plate model are also appropriate since they include vertical supports at the locations of the seal assemblies which in the actual

structure are embedded in concrete. In addition, the grid plate model is supported around the periphery in the vertical direction to simulate the actual support provided by the surrounding reinforced concrete ledge. Therefore, the review team concluded that the model and boundary conditions of the grid plate finite element model, used for the grid plate thermal structural analysis, are acceptable.

2. Section 4.3.2.3 of the ANL report [3] indicates that a larger horizontal deflection (than the one being analyzed) occurs at another location further from the axis of symmetry. The report indicates that this larger value is not evaluated because this location is "not expected to hold a control rod." The concern with this statement is that without any specified controls in place, at some point it may hold a control rod and be relied upon to be operational. As a result, it is recommended that either some control be developed to preclude the placement and reliance on a control rod at this location or this unanalyzed condition of a larger thermal deformation be evaluated as part of the study.

3. Another model of only the control rod is used to evaluate potential binding of the control rod inside the guide tube or seal assembly. This model represents the control rod as a beam supported laterally by the three control rod bearings. Based on the maximum horizontal deformation of the grid plate discussed above, the maximum deformation in terms of angular rotation at the middle bearing was determined. The angular deformation was then applied to the control rod beam model to calculate deflections of the control rod, which are then compared to the available clearances in the guide tube and seal assembly. This method was repeated using a uniform thermal load to determine the maximum acceptable temperature for the grid plate that could cause binding of the control rod.

   a) Based on the cross-sectional view presented in Figure 4-5 of the ANL report [3] and discussion with the ANL staff during a telecom, the calculated grid plate deflection of 0.063 inch (due to the thermal load) is actually imposed on the combined thimble, seal assembly, and guide tube, which then transfers the resulting load into the middle bearing of the control rod. Based on this load path, it is not clear why the single beam model of the control rod is appropriate rather than including all of the structural elements in the load path, i.e., including beam models of the thimble, seal assembly, guide tube, and control rod.

   b) The current analysis of the control rod deflection imposes only an angular deformation (rotation) at the middle bearing while the actual load path discussed above would impose a lateral deformation, which results in both a lateral and angular deformation of the control rod at the middle bearing. This difference between the assumed input of angular deformation and the more realistic lateral deformation that is experienced by the control rod does not appear to be addressed in the current analysis. If all four structural elements in the load path (discussed in Item "a" above) are included in the model, then the appropriate lateral and angular deformation imposed on the control rod would automatically be determined by the finite element analysis. Thus, it is not clear whether the assumed angular deformation in the current analysis is appropriate.

   c) The analysis checks the maximum deformation of the control rod along its length against the original known clearance of the guide tube and the seal assembly. However, based on the load path described in Item a) above, the guide tube and seal assembly would also deform in addition to the deformation of the control rod. The current analysis does not appear to consider the deformation of all structural components in determining the true remaining gap. A model of all four structural components, described in Item a) above would automatically capture this effect. In this case, considering the deformation of the guide tube and seal assembly in the same analysis with the control rod would be beneficial to the evaluation to demonstrate that a sufficient clearance exists, and thus, the approach currently described in the ANL report [3] is considered to be conservative with respect to checking clearances with the original known clearances of the guide tube and the seal assembly.

d) No discussion was found regarding possible tolerances of the structural elements which might affect the calculation to demonstrate that the deformations are less than the available clearances.

In response to Items a) through d) above, ANL staff indicated that the approach followed was to reevaluate the design of the grid plate for the changes made from HEU to LEU based on using the same methodology as in the original grid plate analysis. Since the validity in the assumptions made in the model and approach described in Items a) through d) were not evident, the review team recommends that some technical basis be developed to demonstrate that the modeling and analysis approach is justified or consideration be given to use all of the structural elements in the load path as discussed in items a) and b) above.

The above recommendations would also be applicable to Section 4.3.3.2, Control Rod Deflection due to Maximum Uniform Thermal Load, in the ANL report [3].

4 REFERENCES

1. "Conceptual Design Report for the TREAT Low Enriched Uranium Conversion Program," INL/MIS-16-38804 Rev 0, Idaho National Laboratory, June 3, 2016.

2. I. J. van Rooyen et al., "Transient Reactor Test Facility Fuel Conversion Qualification Plan," PLN-4807, Rev 0, Idaho National Laboratory, October 20, 2015.

3. H. M. Connaway et al., "Analysis of the TREAT LEU Conceptual Design", ANL/RTR/TM-16/1, Argonne National Laboratory, March 2016.

4. M. V. Glazoff, I.J. van Rooyen, B.D. Coryell, and C.J. Parga, "Comparison of Nuclear Fuels for TREAT: UO_2 versus U_3O_8", INL/EXT-16-37972, Rev.0, Idaho National Laboratory, May 2016.

5. C. J. Parga, I. J. van Rooyen, M. V. Glazoff, and E. P. Luther, "TREAT Reactor LEU Fuel-Clad Chemical Interaction Empirical-Modeling Analysis, INL-CON-16-37660, Idaho National Laboratory, 2016.

6. I. J. van Rooyen, N. E. Woolstenhulme, R. K. Jamison, E. P. Luther, and L. N. Valenti, "TREAT Conversion LEU Fuel Design Trade Study," INL/LTD-14-31704, Rev. 1, Idaho National Laboratory, November 2015.

7. "Functional and Operational Requirements for the TREAT LEU Conversion Project," FOR-160, Idaho National Laboratory, September 2015.

8. D. C. Kontogeorgakos, H M. Connaway, D. D. Papadias and A. E. Wright, "Neutronics and Transient Calculations for the Conversion of the Transient Reactor Test Facility (TREAT)," Proceedings of the 36th International Meeting on Reduced Enrichment for Research and Test Reactors, Seoul, Korea, October 2015.

9. D. C. Kontogeorgakos, H M. Connaway, and A. E. Wright, "TREAT Transient Analysis Benchmarking for the heu Core," ANL/GTRI/TM-14/11, Argonne National Laboratory, May 2014.

APPENDIX A Fuel Design, Qualification and Fabrication

Comments and Questions Set 1 (R. R. Hobbins)

Regarding reference [4][c]*:*

There are too many confusing statements, erroneous descriptions about what is in tables and figures, and questionable calculation results for me to provide detailed comments given the hours allotted me for the LEU TREAT Conversion review. However, I will mention just a few items that need attention.

There is confusion about hyper and hypo stoichiometry in UO2. Hypostoichiometry refers to oxygen contents less than 2.0, hyperstoichiometry refers to oxygen contents greater than 2.0.

Considering the reactions: $UO_2 + 2\ C = U + 2\ CO$ and $UO_2 + 4\ C = UC_2 + 2\ CO$ one notices that the primary difference is that the second reaction produces UC_2 as well as the same two moles of CO. Therefore, given that the Gibbs free energy of chemical elements is zero, one would expect the second reaction to have a Gibbs free energy more negative than the first reaction. However, inspection of the results in Tables 4 vs.6 and 5 vs. 7 shows the opposite. I don't think this is correct. Consider also results from S. Gosse et al., J. *Nucl. Mater. Vol. 352*, pp 13-21 (2006), "Critical Review of Carbon Monoxide Pressure Measurements in the Uranium-Carbon-Oxygen Ternary System," which indicate equilibrium of the second reaction for a pressure of CO at 5 Pa (0.05 mbar) at 1444 K, vs. 2300 K in Table 5 and for a CO pressure of 1 Pa (0.01 mbar) a temperature of 1369 K vs. 2100 K in Table 8. In Figure 9, the Gibbs free energy of the second reaction is higher than that of the first reaction, whereas the reverse would be expected.

Regarding reference [5]:

C:U ratio in fuel compact - I assume atom ratio, correct?

Table adjacent to Fig. 10 shows data with O/U much greater than 2, Why?

Are there any post-test results from fuel/cladding interaction tests to share?

Regarding reference [6]:

Table 5. Where do values of reaction rates for Zircaloy-4 (Zry-4) under heading "Actuals" come from?

In future, there needs to be a review of literature data on air oxidation of Zry.

What is effect of oxide buildup on Zry on heat transfer and temperatures of fuel and cladding?

[c] References are found in Chapter 4.

Comments and Questions – Set 2 (R. R. Hobbins and G.L. Copeland)

Comments are presented on three general topics, fuel conceptual design, fuel qualification and fuel block and element fabrication, principally based on review of references [1] and [2].

Fuel Conceptual Design

1. The burnup of the original TREAT HEU core is stated as approximately 0.7%. Is this 0.7% FIMA, or 0.7% of the ^{235}U atoms? (Not much difference for HEU, but %FIMA will be much less than %^{235}U fissioned for LEU.) The burnup requirement for the LEU core is stated as 7.6 MJ (81 MWd) of deposited energy integrated over time, corresponding to about 2.4E23 fissions of ^{235}U. The lifetime burnup required for the LEU core should be stated in the same units as the burnup achieved for the original HEU core. Units of MWd/MTU would be most useful, so one could think in terms of well-documented light water reactor (LWR) UO_2 fuel experience with respect to fission gas production and fuel swelling. Also, what was the neutron fluence experienced for the original HEU core and what is the lifetime neutron fluence requirement for the LEU core? How does the predicted neutron fluence for the LEU core affect the fuel block conductivity? Is the decrease enough to affect the transient behavior? If so, how?

2. The original TREAT core is described as being used for more than 6,000 reactor startups, of which nearly 3,000 were transient irradiations. This suggests that there were about 3,000 steady state irradiations. Correct? What was the total time of the steady state irradiations and approximately what was the temperature in the fuel assembly, especially the cladding temperature? This question relates to how much cladding oxidation might have taken place under steady-state irradiation. For overall understanding of the conceptual design it would be useful to understand the time, temperature, burnup, and fluence under both steady state and transient irradiations experienced by the original core (and expected for the LEU core – probably similar to the original core) to get a feel for the relative importance of each of these conditions on reactor lifetime. While the time duration of transients is very short compared with steady-state operation, the fission rate is very high compared with that during steady-state irradiation.

3. The conceptual design seems to be a hermetically sealed thin (0.025 inch) Zircaloy-4 canister containing a graphite fuel block column with a narrow gap (0.055 inch) evacuated to an internal pressure less than 0.05 mbar between the fuel and the canister. Experience with the original TREAT core indicates that this gap will close due to the pressure differential across the Zircaloy cladding. The thermal analysis reported in Section 5.1.2.2 indicates that cooling times following normal transients are strongly affected by the air coolant flow rate through the reactor. What are the quantitative results for cladding temperature under normal operation, for both steady state and transient conditions, and for accident conditions? Under steady-state irradiation, normal transient conditions, and accident conditions what is the extent of contact between the fuel block and the can, and what are the impacts on cladding temperature and oxidation? What is the quantitative impact of oxide thickness on heat transfer and cladding temperature?

4. What is the status of alternative canister designs, for example, thicker can wall, larger initial fuel to cladding gap size, and thermal insulator between fuel and cladding?

5. What is the current understanding of the pillowing failure mechanism? What are the potential impacts for the safety of the LEU core?

6. Thermodynamic analyses of the reduction of UO_2 and U_3O_8 by carbon in reference [4] are incorrect and need to be redone. These corrected thermodynamic analyses will likely provide sufficient understanding of these reactions and eliminate the need for the proposed study of electronic density distributions by density functional theory.

Fuel Qualification [2]

1. While the fuel qualification plan is thorough and sufficient for the current state of fuel development, much additional detail will be required prior to submittal to regulators. For example, the plan to test fuel materials and fuel assembly components under both out-of-pile and in-pile (steady state and transient) conditions with appropriate post-test examinations, as shown in Figs. 12 and 27, is reasonable. However, it is not possible to understand exactly what articles are to be tested at each stage due to the proliferation of terms used in describing the tests such as component, subcomponent, subsystem (defined in Fig. 10), sub assembly, mock-up test assembly, and lead test assembly (self-evident). Test designs, conditions, and materials will need to be representative of the final expected conditions in the LEU core.

2. The use of the rabbit hydraulic irradiation system in the ATR reactor is mentioned for steady state irradiations. This is suitable only for small samples and likely has little or no capability for temperature control or immersing the sample in a flowing gaseous atmosphere (air). This irradiation system could be useful for subjecting small samples of fuel block material containing UO2 fuel particles to neutrons to study microstructural changes and fission product behavior if there were a way to control the temperature of the sample to a value representative of steady-state operation in TREAT. The limitations of irradiation in the ATR rabbit system need to be addressed and alternative irradiation facilities for irradiating larger samples and subsystems with enhanced capabilities for temperature control, and possibly flowing air, need to be identified.

3. In order to identify facilities capable of performing steady-state irradiations useful for TREAT fuel qualification tests, the requirements for those tests must first be determined, including test articles and sizes, test objectives, test temperatures, atmosphere (inert or air, stagnant or flowing), neutron spectrum and neutron flux and fluence. The irradiation and temperature conditions experienced by the original TREAT core and required for the LEU core will be important inputs for steady-state irradiation test requirements. If it can be determined from analysis that cladding oxidation is dominated by the time at high temperature during cool-down following normal transients (this seems likely but must be firmly established), then it will not be necessary to measure the effect of neutron irradiation on oxidation.

4. Cladding oxidation has been measured in tests with flowing air. How do the rates used compare with those the cladding would experience in TREAT?

5. Given that it is not feasible to accurately test for the effects of 40 years of transient operations, a surveillance program for the LEU core is very important. What surveillance program is proposed for the LEU core, that is, what data will be obtained at what intervals? Mention has been made of surveillance information obtained during the lifetime of the original TREAT core. What information is available from this effort and what sample material might remain that can still be examined?

6. Under what conditions is thermal shock a concern and for which components? What thermal shock tests will be performed on which components/subsystems and how will they be performed?

7. Section 7.2.2 "Fuel Element Assembly Tests: Transient Testing", contains some curious wording that requires explanation. The sentence starting on line 6 reads "The standard TREAT LEU lead test assemblies will be instrumented with additional instrumentation that is possible *on one or more HEU assemblies* [italics added]." What does the phrase in italics mean? The sentence starting on line 16 reads "The *higher fissile loading of the LEU assembly* [italics added] will cause the lead LEU fuel assembly to generate much-higher power than the HEU assemblies in the mostly HEU core." An LEU assembly will certainly have a greater mass loading of uranium than an HEU assembly, but the fissile loading should be similar. Additionally, this sentence seems to be at odds with the information in Sections 5.1.2.1 and 5.1.2.2 of the Conceptual Design Report, which states "Core energies and temperatures anticipated in the LEU core are similar to those in the HEU core." Table 6 shows that peak temperatures under normal and accident conditions are lower in LEU cores than those of HEU cores. Please explain the apparent discrepancy.

8. The transient testing plan seems to call for three tests of one LEU fuel element assembly in each. Would all three transient tests be identical or would conditions differ, such as increasing the power deposition? Might it be of interest for one of the LEU fuel element assemblies to undergo multiple transients?

Fuel Block and Element Fabrication

The fabrication processes for both fuel blocks and the final element are in the early stages of development according to the documents presented. The preferred design at this point appears to be fuel blocks 4x4x4 inches and the encapsulation of them in thin-walled cladding that collapses onto the fuel blocks up to about 57% of the walls. It is assumed that processes will be successfully developed to qualify this fabrication. This configuration serves a dual function of centering the fuel blocks and holding them in place and a block-to-cladding contact heat transfer surface for a reliable heat transfer surface for temperature calculations. The following questions/comments refer to the possibility that acceptable properties cannot be produced in the 4x4x4 inches configuration and that the blocks have to be an assembly of 1x1x4 inches or 1x2x4 inches.

1. Is a plan in place to join the smaller blocks in some manner to act as a unit to ease assembly?

2. Would this composite 4x4 inches block maintain its integrity in case of loss of cladding contact if pillowing occurs?

3. Would a vertical block-to-block locking alignment be desirable in the case of pillowing? (This question also pertains to the monolithic 4x4 inches block.)

The integration of the graphite reflectors into the single about 9-feet can seems a preferable design to the separate containers in the HEU design. However, it may be difficult to achieve the dimensional and straightness requirements. Are alternate designs being considered?

APPENDIX B Neutronics, Thermal, and Structural Analysis

Comments and Questions Set 1: Neutronics and Thermal Analysis (D.J. Diamond and L.-Y. Cheng)

Unless otherwise specified in the following comments, sections/figures/tables refer to the ANL report, reference [3].

1. There are several "key requirements" for the design given in Section 4.1 of reference [1] that relate to the pulse analysis described in the ANL report [3].   Please let us know if the following requirements have been met and if so, where they are documented:
   "The TREAT LEU core shall be capable of
   a) depositing 100 to 1670 cal/g of energy into nuclear fuel test samples
   b) generating an energy pulse equivalent to an HEU 2.5 GJ maximum energy pulse [where it is assumed that "equivalent" refers to the test sample TED]
   c) providing a maximum power level of 19 GW"

   Note that Item 3.2.1.1.3 in Appendix A of the INL report refers to these requirements and refers to an Appendix B which does not exist.

2. Other key requirements where documentation was not found are a) a lifetime burnup of 81 MWd (capability with respect to core reactivity); b) producing a neutron radiography beam equivalent to the original; and c) meeting shutdown margin requirements.

3. Fuel Composition: In Section 2.1.1 it states that relative fractions of isotopes in the fuel were adjusted; but the amount of equivalent boron, oxygen-to-uranium ratio and fuel mass density are fixed so why is there a need for any adjustment? Table 2-3 provides mass/atom fractions in a fuel element for all isotopes but confirm that for C there appears to be a typo. In Section 1.4 the fraction of carbon as graphite is given as 0.85. How is this determined and where is it documented? In Section 1.1 it states that the HEU is in UO_2. Confirm what portion of the core is UO_2 and what portion is U_3O_8 and what mix was used in the analysis for the current HEU core. In addition to minor impurities in the graphite/carbon fuel element, are there also significant traces of polymer or other binding material and if so, are they taken into account in the analysis?

4. Please explain whether the LSSS is 600°C or 575°C. Where is temperature measured during operation and how are those measurements used?

5. Although it is understood that the report is meant to show the similarities of the LEU and HEU fueled cores, it is of interest to know what is the range of absolute values of PCFs. PCF calculations are based on steady state power distributions. Does the PCF change when transient rods are withdrawn? This could be ascertained from Tables 2-26 and 2.27 if the cold PCF was calculated with transient rods inserted while the hot PCFs are calculated with them withdrawn.

6. Does the first row in Table 2-20 correspond to the transient rod bank worth for the all rods out (ARO) configuration shown in Table 2-4 (of course assuming results shown in Table 2-

20 are for the reference M8CAL half-slotted LEU core). Transient rod bank worth from Table 2-4 is 8.66% and 8.66% x 0.95 = 8.23% which is different from the 8.27% value shown in Table 2-20. Are the % withdrawn positions for the control/shutdown rod banks shown in Table 2-20 the approximate critical positions for the corresponding transient rod bank withdrawn positions in the table? Is it correct to assume that the compensation/shutdown rod banks are fully withdrawn in the calculations for Table 2-20?

7. Section 2.2.1.3 shows the temperature reactivity effect as a function of either total core energy or core average temperature. Temperature changes in the fuel blocks are calculated by assuming adiabatic heating and constant (over time) radial and axial power profiles for each fuel element.
    a. Has the effect of transient power distribution (spatial and temporal) been evaluated?
    b. Did the calculation assume uniform temperature (fuel and cladding) at each axial elevation of a fuel element?
    c. In Table 2-25, there is a typo for the LEU value at 600°C.
    d. Although the adiabatic model helps to make the temperature calculation conservative with respect to temperature limits, how would it affect transient calculations where it would tend to over-predict negative reactivity feedback?

8. In trying to understand the consistency between Tables 2-26/27 and 2-28 it is noted that the HEU peak core temperatures in Table 2-28 are different from those in Table 2-26 and (without doing the graphing) it is not clear if the LEU peak core temperatures in Table 2-28 are commensurate with the values in Table 2-27. Please comment.

9. Do the peak temperatures in Table 2-29 for operational and accident conditions correspond to peak temperature values in Table 2-28 or Tables 2-26 and 2-27?

10. In Table 2-29 why is the HEU peak temperature greater than the limiting temperature of 820°C? Is this a measure of the conservatism of the ANL analysis and the organization (INL) responsible for safely operating the HEU reactor can show that the temperature would be less than 820°C? If it is a measure of conservatism and the expected ratio of LEU-to-HEU PCF ratio is close to 0.90 there would appear to be "extra" margin available under accident conditions. And if this is the case, is there any interest in adding more reactivity than is currently projected for the LEU core so that the achievable TED can be increased?

11. The reactivity available for an accident is assumed to be from movement of the transient rods. The compensation/shutdown bank has a similar total worth as the transient bank. What are the properties (e.g., position limits, withdrawal speed) that preclude this bank's consideration for a reactivity insertion accident? Can an individual rod get stuck and then drop out of the core like in a boiling water reactor (BWR) rod drop event? If so, what is the maximum worth of an individual rod?

12. How is the transient bank manipulated to obtain the historic transient #2874 (or is period and not reactivity the input to TREKIN)? Why is it considered the limiting transient? Does (essentially) duplicating this transient imply that all other HEU core transients can be duplicated in the LEU core? Does this transient determine the "maximum allowed

reactivity" for an operational transient? Is this maximum then determined by operational constraints (e.g., bank insertion limits)?

13. In Section 2.2.1.1 the measured-to-calculated transient bank worth is given. Where is this documented?

14. Did the calculations of the cooling time histories for normal operation and accident conditions account for heating from residual fission and decay heat?

15. Can we assume that the temperatures shown in Figure 3-5 for the hottest fuel assembly are at the location of the peak axial power?

16. Have the assumptions regarding the flow distribution and magnitude of the cooling air in the core been justified? Is the temperature of the coolant ambient at 25°C?

17. Figure 3-7 shows the axial variation of cladding loss as a function of peak temperature and contact resistance. What was the length of time used in calculating the cumulative loss of metal in the figure and how is that time determined?

18. What is the definition of effective axial fuel conductivity? Thermal conductivity for the TREAT fuel is on the order of 15 W/m-K. The label and units of the y-axis in Figure 3-11 appear to be incorrect. What is the reason for the thermal conductivity of the TREAT LEU fuel to be about a factor of six lower than that for the reflector block?

19. This review will not be considering the analysis done at INL as part of the restart project. Does ANL have any comments on how their methodology might be significantly different from the INL methodology?

Comments and Questions Set 2: Structural Analysis (J. Braverman)

The request for additional information and questions listed below relate to the ANL report, reference [3]. The thermal structural analysis in this report focuses on the heating of the core support grid and using the resulting deformation of the grid plate to perform a structural analysis of the control rods in order to demonstrate that the control rods will not deflect to the point which would cause them to bind with either the guide tube or seal assembly.

1. Section 4.3 – Grid Plate Thermal Distortion Analysis, in the ANL report, indicates that the structural analysis for the grid plate utilized a finite element model in ANSYS. Please provide a figure of the ANSYS model that includes the discretization of the finite elements.

2. Section 4.3.1 – Assumptions, refers to Table 4-1 that presents the material properties for the A36 steel which is used in the grid plate thermal distortion analysis. The table indicates that Poisson's ratio is equal to 2.6. Explain why this value is presented because Poisson's ratio for A36 steel is 0.26 at room temperature. If this is a typo, please confirm that the correct value corresponding to the grid plate temperature was used in the various analyses and identify that value.

3. Section 4.3.2.1 – Geometry of Grid Plate and Control Rods, shows the cross-sectional view of the control rod assembly. The figure identifies the bearings in red, guide tube in orange, and seal assembly in green. Explain what the blue color region below the grid plate elevation represents and what the blue color region above the grid plate represents.

4. Section 4.3.2.3 – Grid Plate Deflection Due to Non-uniform Thermal Load, indicates that a larger horizontal deflection (than the one being analyzed) occurs at another location further from the axis of symmetry. This section of the report also states that this larger value is not evaluated because this location is "not expected to hold a control rod." Even though this location is not expected to hold a control rod, is it possible that it may at some point hold a control rod and be relied upon to be operational? If not, then how is this assured because the location with a larger deflection is not evaluated?

5. Section 4.3.2.4 – Control Rod Deflection due to Non-uniform Thermal Load, describes the use of another ANSYS model of only the control rod represented by a beam, which is supported laterally at the three bearing locations. An angular displacement of 0.30° at the middle bearing, due to this thermal deflection, was imposed on the beam and the deflection of the control rod was determined to be 0.0832 inch within the guide tube and 0.0423 inch within the seal assembly. Since the clearance in the guide tube is 0.125 inch, and the clearance in the seal assembly is 0.063 inch, the ANL report concluded that the total deflection in the control rods is not expected to result in loss of control rod functionality.

   a) Based on the cross-sectional view presented in Figure 4-5, it appears that the grid plate deflection of 0.063 inch (due to the thermal load) is actually imposed on the guide tube which then transfers the resulting load into the seal assembly, which then transfers the load into the middle bearing. Please confirm if this is the correct load path, and if not explain the correct load utilizing the information in Figure 4-5.

b) If the load path described in Item “a” above is correct, then explain why the angular distortion calculated in Section 4.3.2.3 of the ANL report is based on the dimension of 12 inches which corresponds to the distance between the grid plate and the intersection point of the seal assembly with concrete. It appears that the point of rotation would not be at the intersection point of the seal assembly with the concrete but at the bottom bearing in the seal assembly that is buried in the concrete. Therefore, explain why the 12-inch dimension was used.

c) If the load path described in item “a” above is correct, then explain why the single beam representing the control rod is adequate rather than including all of the structural elements in the load path, that is, including beam models of the guide tube, seal assembly, and control rod.

d) It should be noted that the current analysis imposes only an angular distortion while the actual load path would impose both a lateral deformation and angular deformation. How is this addressed in the current analysis? If all three structural elements in the load path are included in the model, then this would be automatically incorporated in the analysis.

e) The current analysis checks the maximum deformation of the control rod along its length against the original known clearance of 0.125 inch in the guide tube and 0.63 inch in the seal assembly. However, based on the load path in Item “a”, the guide tube and seal assembly would also deform in addition to the deformation of the control rod, and so explain how the current analysis would capture the true remaining gap. A model of all three structural components, described in Item “d” above would automatically capture this effect.

f) No discussion was found regarding possible tolerances of the structural elements which might affect the calculation to demonstrate that the deformations are less than the available gaps. How is this addressed?

The above items would also be applicable to Section 4.3.3.2 – Control Rod Deflection due to Maximum Uniform Thermal Load.

6. In view of the relatively thin material of the guide tube, was it confirmed that the resulting localized load on the guide tube from the grid plate deflection (i.e., localized pressure on one side of the guide tube, over the length corresponding to the thickness of the grid plate) would not crush, overstress, or reduce the available gap between the control rod and the guide tube?

7. Section 4.3 of the report evaluates the control rods to demonstrate that they will not deflect to the point that would cause them to bind with either the guide tube or seal assembly. It does not include an evaluation to demonstrate the structural integrity of the control rod assembly, including its various structural elements (e.g., control rod, seal assembly, guide tube, outer housing, etc.) subjected to design basis and accident loads (e.g., dead load, thermal, seismic, etc.). Does some structural integrity evaluation based on analysis or testing exist or why not?