From: Edward Helvenston
Sent: Tuesday, May 2, 2023 6:16 PM
To: Rusty Towell; Lester Towell; Jordan Robison; Tim Head; Alexander Adams
Cc: Richard Rivera; Zackary Stone (He/Him/His); Michael Wentzel; Greg Oberson (He/Him); Boyce Travis; Kyle Song; Alexander Chereskin
Subject: ACU MSRR Chapter 4, Chapter 6, and Section 9.6 Audit Question

Dear Dr. Towell,

Below is a question the NRC staff has prepared for Abilene Christian University (ACU) related to the ACU Preliminary Safety Analysis Report, primarily Chapter 4, "Design of Structures, Systems, and Components." The NRC staff would like to discuss this question within the scope of the ACU construction permit (CP) application
review Audit Plan for Chapters 4 and 6 and Section 9.6 (see audit plan dated 3/2/2023, ML23065A055), and I am providing in advance to facilitate discussion during an audit meeting. We will add this email, with the question, to public ADAMS. If you have any questions, please let Richard, Zackary, or I know.

Thank you,

Ed Helvenston, U.S. NRC
Non-Power Production and Utilization Facility Licensing Branch (UNPL)
Division of Advanced Reactors and Non-Power Production and Utilization Facilities (DANU)
Office of Nuclear Reactor Regulation (NRR)
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Item #	Question
4.3-16	In MSRR PSAR Chapter 4, the only degradation mechanisms that appear to be identified for 316H SS and its weld filler metal are oxidation (i.e. general corrosion) and high temperature creep. However, based on the references cited below the staff notes that there appear to be other potential degradation mechanisms that could be applicable to the MSRR design. These degradation mechanisms are as follows: • Effects of fission products on corrosion (i.e., oxidizing fission and decay products); • Fission product induced cracking (e.g., Te embrittlement); • Irradiation assisted corrosion; • Irradiation assisted cracking; • Neutron embrittlement; • Helium embrittlement of structural alloys due to neutron interactions with nickel in metallic alloys; • Phase formation embrittlement (When exposed to beryllium and carbon in Flibe, 316H SS can form intermetallic compounds which decrease the tensile strength of 316H SS) • Stress corrosion cracking; • Environmentally assisted creep; • Corrosion fatigue; • Galvanic corrosion; • Hydride formation and embrittlement; • Thermal aging; and • Erosion/Wear/Flow Effects. Considering the above, please discuss the following: a) Has ACU performed a review to determine degradation mechanisms, including those listed above, applicable to components in the

reactor system?

b) If so, which mechanisms were determined to apply to the MSRR design? If not, how will ACU determine what degradation mechanisms may be applicable to components in the reactor system?

c) How will applicable degradation mechanisms be addressed for both component design and verification of degradation rates (e.g., collecting new data via testing, use of applicable historical data, inspection, surveillance coupons, performance monitoring, etc.)? For methods used to address degradation, describe why these methods are applicable and appropriate, as well as whether methods will be used in conjunction with each other (e.g., inspection to validate test data). If certain mechanisms were determined to not apply or be significant, provide the justification for the determination.

d) If data (new testing or historical) is used to address any of these degradation mechanisms, describe how the data is applicable and/or bounding to the MSRR design including during postulated accident scenarios.

e) For mechanisms that may affect both the 316H SS as well as the weld filler (including the heat affected zone), describe how these effects are considered for both materials.

References:

1. Busby, J., et. al., Oak Ridge National Laboratory, ORNL/SPR-2019/1089, “Technical Gap Assessment for Materials and Component Integrity Issues for Molten Salt Reactors,” March 2019 (ADAMS Accession No. ML19077A137).
2. Gandy, D., et. al., Electric Power Research Institute, 3002010726, “Program on Technology Innovation: Material Property Assessment and Data Gap Analysis for the Prospective Materials for Molten Salt Reactors,” March 2019.
3. Holcomb, D.E., et. al., Oak Ridge National Laboratory, ORNL/TM-2021/2176, “Molten Salt Reactor Fundamental Safety Function PIRT,” September 2021.
4. Keiser, J.R., et. al., Journal of Nuclear Materials, Volume 565, 153698, “Interaction of Beryllium with 316H Stainless Steel in Molten LiF_2BeF_4 (FLiBe),” March 2022.
5. Raiman, S.S., et. al., Oak Ridge National Laboratory, TLR-RES/DE/CIB-CMB-2021-03, “Technical Assessment of Materials Compatibility in Molten Salt Reactors,” March 2021 (ADAMS Accession No. ML21084A039).
6. Singh, P.M, et. al., School of Material Science and Engineering, Georgia Institute of Technology, “Phenomena Identification and Ranking Tables (PIRTs) Report for Material Selection and Possible Material Degradation Mechanisms in FHR,” April 15, 2017.
7. University of Wisconsin, Madison, UCBTH-12-003, “Fluoride-Salt-Cooled High Temperature Reactor (FHR) Materials, Fuels and Components White Paper,” July 2013.
8. US NRC, “Overview of Molten Salt Reactor Technology Training Materials Module 5: Materials,” December 2017 (ADAMS Accession No. ML17331B120).
9. US NRC, DANU-ISG-2023-01, “Material Compatibility for non-Light Water Reactors Draft Interim Staff Guidance,” February 2023.

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