From:	Edward Helvenston
Sent:	Monday, January 8, 2024 11:34 AM
To:	Rusty Towell; Lester Towell; Benjamin Beasley; Tim Head; Jordan Robison; Alexander Adams; Brazos Fitch
Cc:	Richard Rivera; Michael Wentzel; Greg Oberson (He/Him); Mohsin Ghazali; Boyce Travis; Alexander Chereskin; Ryann Bass
Subject:	ACU MSRR PSAR Section 4.3 Audit Questions (Related to Material Degradation)
Attachments:	ACU January 8 2024 Material Degradation Follow Up Audit Questions.pdf

Dear Dr. Towell,

Attached are 2 questions the NRC staff has prepared for Abilene Christian University (ACU) related to the ACU Preliminary Safety Analysis Report, primarily Section 4.3, "Vessel." The NRC staff would like to discuss these questions 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 questions, to public ADAMS. If you have any questions, please let Richard, Mohsin, or I know.

Thank you,

Ed Helvenston, U.S. NRC
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Audit Question 4.3-18

Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) Preliminary Safety Analysis Report (PSAR), Revision 1 (ML23319A094), makes several statements related to functions of reactor system Structures, Systems, and Components (SSCs) and how these SSCs withstand the MSRR environment.

PSAR Table 3.4-1, "Safety, Seismic, and Quality Classification of Structures, Systems, and Components," states that the reactor system is classified as safety related (SR) because it is a fission product barrier under both normal and accident conditions. The reactor thermal management system (RTMS) is also listed as a SR component.

PSAR Section 4.3.2, "Design Bases," states that Design Criteria (DC) 14 requires the reactor system to be "…designed to have an extremely low probability of leakage, rapidly propagating failure, or rupture." This section also states that DC 31 requires the reactor system to "…have sufficient margin to minimalize the probability of rupture." PSAR Section 4.3.4, "Fuel Salt Chemical Attack," states the "…MHA provides the background against which the safety significance of degradation mechanisms is viewed. In most conceivable degradation events, the outcome would be a small leak that would be detected…." Additionally, the Degradation Mechanisms Table posted to the online audit portal on September 29, 2023, in response to Audit Question 4.3-16 provided to ACU on May 2, 2023 (ML23123A044), stated that "[w]ithout a large pressure differential, defects and cracks do not quickly propagate to a large rupture and can be monitored."

The following questions are applicable to SR SSCs and functions of those SSCs that are required to satisfy DCs 14 and 31.

a. The meaning of the statement in PSAR Section 4.3.4 regarding how the MHA impacts safety significance of degradation mechanisms is not clear to the NRC staff. The MSRR PSAR lists SR components that have a safety function to maintain the fuel salt boundary as well as components that are required to meet DCs 14 and 31, which includes guarding against loss of component integrity. Clarify the intent of the statement made in PSAR Section 4.3.4 and describe how the MHA affects treatment of component integrity and related degradation mechanisms.
b. This question is a follow-up to part a. In order to satisfy assumptions made in the MHA, it appears the RTMS needs to maintain boundary integrity. However, the RTMS will be exposed to degradation mechanisms (e.g., thermal aging) throughout its 20-year design life, due to operational leakage. The RTMS would also be exposed to degradation mechanisms during the MHA itself. When considering the safety significance of these degradation mechanisms, how are the effects on RTMS integrity considered?
c. It is not clear what is meant by "…most conceivable degradation events…." This statement could mean that no degradation can occur to compromise reactor system integrity, or it could mean that given available data and other measures, it can be appropriately accounted for and/or mitigated to lessen the probability of consequences more severe than a small leak. Further, use of the word "most" instead of "all" suggests that there could be some conceivable degradation events to which the stated outcome would not apply. Clarify the intent of this statement.
d. The statement made in the Degradation Mechanisms Table posted to the online audit portal, referred to above, that cracks do not quickly propagate without a large pressure differential is not clear to the staff. As discussed during the audit, there are several mechanisms that can cause cracking and embrittlement in the MSRR. The staff notes

that this degradation may cause cracks to propagate rapidly even in the absence of a large pressure differential if the various cracking and embrittling mechanisms are not adequately accounted for via design and fitness-for-service approaches. The statement in the PSAR could be interpreted as meaning that it is not possible for quickly propagating cracks to occur; or, that based on available data, design conservatism, and mitigation and monitoring provisions, cracking can be managed so cracks do not rapidly propagate prior to detection. Clarify the intent of the statement in the Degradation Mechanisms Table.

Audit Question 4.3-19

ACU MSRR PSAR, Revision 1, Section 4.3.8, "Thermal Design Limits," states the "[t]hermal safety limit for the reactor system is defined to ensure that the reactor system structural fission product barrier will not rapidly deteriorate under any condition. The reactor system safety limit is 816°C so that the reactor system remains with code applicability as stated in Section 4.3.3." However, the staff notes that the NRC staff endorsement of ASME Code, Section III, Division 5 limits the use of 316H stainless steel (SS) depending on the time at temperature. During the in-person audit discussion of Audit Question 4-11 (ML23086A014) held on May 18, 2023, ACU stated that it will incorporate a technical specification that will restrict the upper temperature limit to not exceed the temperature for 316H SS endorsed by the NRC in Regulatory Guide (RG) 1.87, Revision 2, "Acceptability of ASME Code, Section III, Division 5, High Temperature Reactor" (ML22101A263). Clarify whether ACU will design and operate the MSRR to remain within the bounds of the staff's endorsement of the ASME Code detailed in Regulatory Guide 1.87, Revision 2.