From:	Richard Rivera
Sent:	Monday, January 22, 2024 12:06 PM
To:	Rusty Towell; Lester Towell; Benjamin Beasley; Tim Head; Jordan Robison; Alexander Adams; Brazos Fitch
Cc:	Edward Helvenston; Michael Wentzel; Greg Oberson (He/Him); Mohsin Ghazali; Alexander Chereskin; Ryann Bass; Michael Balazik
Subject:	ACU MSRR PSAR Section 4.3 Audit Questions (4.3-23 and 4.3-24 on Material Degradation)
Attachments:	Audit Questions 4.3-23 and 4.3-24.pdf

Dear Dr. Towell,

Attached are two 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 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 to be scheduled. We will add this email, with the questions, to public ADAMS. If you have any questions, please let Edward, Mohsin, or I know.

Thank you,
Richie

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Audit Question 4.3-23 (Fission Product Induced Cracking (e.g., Te embrittlement))

Concerning fission product induced cracking, in the document titled “Degradation Mechanisms Table September 29 Revision.pdf,” ACU states that “ORNL data shows that [stainless steel] SS316 is resistant to [tellurium] Te embrittlement. Use of [beryllium] Be to control redox potential was shown to mitigate Te embrittlement, even in more prone Ni-based alloys,” and further that, “[n]o testing is planned as any effects will be bounded by historical data and inspection.” In the same document, the design is said to be “…addressed by chemical analysis of salt samples for fission products,” and inspection is “…addressed by exterior inspection for cracks near welds with nondestructive test methods.”

The following questions concern this information.

a. The report from Oak Ridge National Laboratory (ORNL)-4829, titled “Intergranular Cracking of INOR-8 in the MSRE,” referenced by ACU, states that “[s]everal alloys, including 300 and 400 series stainless steels, cobalt- and nickel-base alloys containing more than 15% [chromium] Cr, copper, Monel, and some modified compositions of INOR-8 are resistant to cracking in the tests run to date.” However, it also states that “[f]urther work will be necessary to show unequivocally that these materials resist cracking in nuclear environments, including in-reactor capsule test.” Considering the latter statement, how has ACU concluded that effects will be bounded by historical data for both the base metal and weldments?
b. Concerning the use of Be to mitigate Te embrittlement, ORNL report No. ORNL/TM-6002, titled “Status of Tellurium-Hastelloy N Studies in Molten Fluoride Salts,” referenced by ACU, only demonstrates this for Hastelloy N. Has ACU identified any data demonstrating the viability of this technique, specifically for stainless steel and its weldments, or does ACU assume that demonstrated viability for nickel (Ni)-based alloys is, in itself, sufficient to address stainless steel and its weldments, because the Ni-based alloys are more prone to this phenomenon?
c. Clarify what aspects of the design are "...addressed by chemical analysis of salt samples for fission products." Given that the fuel salt will likely contain some quantity of Te, describe what action(s) will ACU take based on Te being present in the salt. Specify what criteria, if any, ACU has identified that will be used to assess the sufficiency of its inspection program, including the methodology, frequency, and the means by which inspection findings (e.g., indications of cracking) will be dispositioned?
d. Will surveillance specimen locations bound all potentially affected components? Does ACU plan to implement corrective actions if surveillance coupons indicate intergranular attack? How will the sufficiency of those actions, if any, be evaluated?

Audit Question 4.3-24 (Stress Relaxation Cracking)

Stress relaxation cracking (SRC) is not a corrosion-related phenomena. SRC is caused by stresses in susceptible material from being unable to be relieved fast enough; susceptible materials are those that experience reduced material ductility because precipitate strengthening within the grain interior limits strain accommodation to the grain exterior (American Petroleum Institute, API Technical Report 942-B, "Material, Fabrication, and Repair Considerations for Austenitic Alloys Subject to Embrittlement and Cracking in High Temperature 565°C to 760°C (1050°F to 1400°F) Refinery Services," 1st Edition, Washington, DC, May 2017). This feedback was provided by the NRC staff to ACU during the in-person audit held on May 18, 2023. With the exception of Alloy 617, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 5-2017 does not address SRC. Regulatory Guide 1.87, Revision 2, "Acceptability of ASME Code, Section III, Division 5, "High Temperature Reactors,"" Section C, item 1.x.(1) states that "applicants and licensees should develop their own plans to address the potential for stress relaxation cracking in their design."

SRC should be evaluated if its effects on systems, structures, and components (SSCs) would challenge the conformance of relevant design criteria such as design criterions 14 and 31. Moreover, reactor thermal management system (RTMS) integrity is an assumption in the maximum hypothetical accident (MHA) scenario described in ACU's Preliminary Safety Analysis Report (PSAR), Revision 1 (ML23319A094), Chapter 13. The integrity of the RTMS needs to be maintained in order to ensure the MHA remains bounding of all potential events. SRC may challenge the integrity of this system.

The following questions concern SRC:

a. In the response to audit question 4.3-11, posted in the Electronic Reading Room (ERR), ACU states that "[s]tress relaxation cracking (SRS) phenomenon for welded thick-walled areas will be assessed." What is the scope of the referenced assessment and what is that assessment intended to demonstrate? Also, the response to the audit question refers to limited available data. How does the data limitation affect the expected outcome of the assessment?
b. Has ACU determined, as stated in the response to audit question 4.3-11, that the assessment should address only thick-walled areas? If so, explain how ACU has determined that other areas may be excluded from the assessment. API Technical Report 942-B, cited by the NRC staff above, identifies additional areas beyond the welded thick-walled areas that are susceptible to SRC.
c. In the response to audit question 4.3-11 in the ERR, ACU states that "[r]esidual stresses due to fabrication and welding will be minimized through selection of appropriate welding technique and heat treatment… Post-weld heat treatment of welded sections will also be explored to avoid stress relaxation cracking." Clarify if ACU has determined, or when does it plan to determine, what welding techniques and/or heat treatments will be used to mitigate the potential for SRC? Explain how ACU will determine that the residual stresses have been sufficiently minimized.
d. Is ACU implementing any design measures to mitigate SRC? If so, describe these design measures.
e. Does ACU have in-service inspection, surveillance, or performance monitoring plans intended to detect SRC after the reactor begins operation? If so, how will ACU determine

that these plans are adequate to detect the progression of SRC before the capacity of potentially affected components to perform their intended functions is lost?

f. The response to audit question 4.3-11 refers to the testing of welded samples in molten salt. Explain how these tests are intended to support the assessment of SRC, considering that SRC is not a corrosion-related phenomenon.