From: Richard Rivera
Sent: Tuesday, January 23, 2024 2:28 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 Question (4.3-25 on Material Degradation)
Attachments: Audit Question 4.3-25.pdf

Dear Dr. Towell,

Attached is a question 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 question, to public ADAMS. If you have any questions, please let Edward, Mohsin, or I know.

Thank you,
Richie

Richard Rivera, MEM
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4.3-25 (Irradiation Effects)

The document titled, “Degradation Mechanisms Table September 29 Revision.pdf,” provided by ACU via electronic reading room (ERR), includes information on its analyses of irradiation assisted cracking, neutron embrittlement, and helium embrittlement of structural alloys due to neutron interactions with nickel in metallic alloys.

Concerning irradiation assisted cracking for stainless steel SS316H, ACU states that it intends to control the redox potential of the salt, and that “…if the corrosion is mitigated then all corrosion related damage mechanisms can be controlled.”

Concerning neutron embrittlement for SS316H, ACU states that published literature data “…will be reviewed and accounted for in the design…”

Concerning helium (He) embrittlement, ACU states that it “…expects a very small amount of helium to be produced over the expected operating time,” and that “…this damage mechanism for the reactor vessel is being considered and will be accounted for based on the expected [displacement per atom] DPA calculations.”

In the document containing the materials degradation matrix, provided by ACU via electronic reading room (ERR), it states that it does not plan additional testing because its radiation dose is low compared to historical data, and that changes to mechanical properties as a result of these phenomena will be accounted for in the design.

Related issues were addressed in audit questions 4.3-7 and 4.3-16. Revision 1 to the response to question 4.3-7 on the portal cites two references (Tavassoli, et. al., 1996 and Xu, et. al., 2016) and states that “[t]he data in these reports indicates that the overall impact of neutron irradiation on SS316 at the operating temperature of the molten salt research reactor (MSRR) will be minor up to 5 dpa.” The response to question 4.3-16, again refers to an accounting for these phenomena in the design.

The following questions concern this information:

1. How has ACU determined that the overall impact of neutron embrittlement is “minor” when considering:
   a. The conclusions one of the references cited by ACU (Tavassoli, et. al., 1996) states that “[l]ow dose irradiation is shown to significantly influence mechanical properties, for example by increasing proof strength and decreasing the creep rupture strength, creep rupture ductility, and creep-fatigue endurance.” Additionally, the other cited data set (Xu, 2016), concludes that austenitic SS “…exhibit[s] hardening and, generally, a reduction of creep resistance under irradiation conditions.” The response to question 4.3-16 also cites a reference (Bloom, 1972) which concludes “[n]eutron irradiation produces significant changes in the physical and mechanical properties of austenitic stainless steels. Ductility is the most adversely affected property, and it is reduced for nearly all irradiation and test conditions.”
   b. Other references not cited by ACU (Reference Nos. 4, 5, and 6 listed below) indicate the impact of irradiation on resistance to creep and creep-fatigue, reduction in fracture toughness, crack initiation and fracture mode, and hardening and loss of ductility. Ward demonstrated that irradiation can result in “…disproportionately large ductility losses in the as-deposited weld metal…,” and that there can be “…reductions in uniform elongations of 11 to 68% and in total elongations of 21 to 33%....” (Ward, 1974). Results

from testing He embrittlement on 316 SS also show an almost 50% reduction in total elongation at 0.5 atomic part per million (appm) helium at 700°C (Horhoianu, 1975). Irradiation can affect these properties in different ways. These include dislocation of atoms, generation of helium (i.e., helium embrittlement), void swelling, etc. (Messner, 2020).

The manner in which irradiation affects materials may also influence what properties are affected and the degree to which they are affected. Changes to material properties may impact factors for design (e.g., creep rupture and subsequent allowable stress), inspection (e.g., time to cracking to determine appropriate inspection interval), and crack growth rate and propagation (e.g., fracture toughness loss and crack mode). In addition to these effects on the base metal (i.e., 316H SS), irradiation may also impact the selected weld filler metal (i.e., ER 316), as well as the heat affected zone (HAZ) near welds.

2. Explain the intended meaning of ACU's statements that the effects of neutron irradiation will be "used," "accounted for," or "addressed" for design? What aspect(s) of the design do these statements refer to (e.g. reduction in ductility)? How will ACU determine that the effects of neutron irradiation, including those cited in point (b) of question (1), have been appropriately "used," "accounted for," or "addressed?"

3. Does ACU intend to account for the effects of neutron irradiation on materials properties in an eventual inspection or fitness-for-service type program (e.g. account for reduction in fracture toughness)? If so, explain how this will be done?

4. Currently, the information provided by ACU in its preliminary safety analysis report (PSAR) and information posted in the ERR does not include estimates or bounding values for He generation. Explain how ACU has determined that the expected He generation is "very small". Describe how historical data cited in the degradation matrix posted to the ERR is adequate for predicted MSRR helium generation in metallic components (e.g., bounding He generation). Has ACU compared the expected He generation for the MSRR to historical data to determine how it should be accounted for in the design?

5. Concerning irradiation assisted cracking, ACU states that "…if the corrosion is mitigated then all corrosion related damage mechanisms can be controlled." ACU also states for this phenomenon that, "…inspection will be performed near welds for cracks…" While corrosion control will help to mitigate irradiation assisted cracking, it may not eliminate it. Does ACU plan to establish and implement an inspection program to address the possibility of irradiation assisted cracking?

6. Describe the safety related (SR) function of the upper and lower grid plates, and how that function may be affected by irradiation. As stated in ACU's response to question 4.3-7, these grid plates will likely be the metallic components most affected by irradiation.

References:
1. Tavassoli, A.A, Picker, C., Wareing, J., Effects of Radiation on Materials 17th International Symposium, ASTM STP 1270, "Data Collection on the Effect of Irradiation on the Mechanical Properties of Austenitic Stainless Steels and Weld Metals," 1996.
2. Xu, S., Zheng, W., Yang, L., "A Review of Irradiation Effects on Mechanical Properties of Candidate SCWR Fuel Cladding Alloys for Design Considerations," CNL Nuclear Review Vol 5, Number 2, December 2016.

3. Bloom, E.E., Weir Jr., J. R., Oak Ridge National Laboratory, “Effect of Neutron Irradiation on the Ductility of Austenitic Stainless Steel,” 1972.
4. Ward, A.L., Hanford Engineering Development Laboratories, Westinghouse Hanford Company, “Thermal and Irradiation Effects on the Tensile and Creep-Rupture Properties of Weld-Deposited Type 316 Stainless Steel,” 1974.
5. Horhoianu, G., Vangermeulen, W., Janssen, C., “Helium Embrittlement in Type 316 Stainless Steel,” November 1975.
6. Messner, M. C., Barua, B., Rovinelli, A., Sham, T.L., Argonne National Laboratory, ANL-19./13, “Environmental creep-fatigue and weld creep cracking: a summary of design and fitness-for-service practices,” January 2020.