From: Richard Rivera
Sent: Thursday, March 16, 2023 4:25 PM
To: Rusty Towell; Lester Towell; Jordan Robison; Tim Head; Alexander Adams
Cc: Edward Helvenston; Richard Rivera; Zackary Stone; Michael Wentzel; Greg Oberson (He/Him); Boyce Travis; Alexander Chereskin
Subject: Abilene Christian University - Audit Questions Regarding ACU CP Chapters 9.2 and 13
Attachments: Audit Questions_Chapters 9.2 & 13 (Round 1).pdf

Dear Dr. Towell,

Attached is a list of questions the NRC staff has prepared for Abilene Christian University (ACU) related to the ACU Preliminary Safety Analysis Report, Section 9.2, "Handling and Storage of Reactor Fuel," and Chapter 13, "Accident Analyses." The NRC staff would like to discuss these questions within the scope of the ACU construction permit (CP) application review Audit Plan for Chapters 9.2 and 13 (see audit plan dated 3/2/2023, ML23065A056), and I am providing these in advance to facilitate discussion during an audit meeting. Once ACU is ready to discuss, please let us know and we can set up an audit meeting. We will add this e-mail, with questions, to public ADAMS. If you have any questions, please let Edward, Zack, or I know.

Thank you,
Richard Rivera

Richard Rivera, MEM
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PSAR Review Information Needs for Chapter 9, Section 9.2, "Handling and Storage of Reactor Fuel"

Item #	Reviewer(s)	Date Sent to ACU (Accession No.)	PSAR Chapter or Topic	Question
9.2-1	B. Travis		9.2	ACU should provide the supporting basis for the statement "Fuel salt is maintained in geometries and in proximity to materials that prevent criticality in all conditions during fuel storage and movement in the facility." NRC staff would expect at least preliminary analysis (to be summarized in the PSAR and updated in the FSAR) and provide some detail as to why k_{eff} is less than/is expected to be less than 0.9, but the staff is seeking clarification as part of the audit.
9.2-2	B. Travis		9.2	The section does not describe any technical specification expectations related to fuel handing, nor does Chapter 14. Because this is a PSAR, NRC staff does not expect values, but given the similarities and connection to the reactor system, the staff would expect similar radiation monitoring, leakage control, and temperature maintenance conditions to be associated with the fuel handing system. The NRC staff requests that ACU provide context in this area.
9.2-3	B. Travis		9.2	The NRC staff requests that ACU provide additional information regarding the limiting conditions for inventory and duration of stored fuel in the fuel handling system (e.g., what is the maximum expected spent and new fuel stored on site, and for how long is the fuel expected to be stored). This has implications related to material properties needed to withstand long-term exposure to irradiated fuel, what the potential limiting criticality condition for stored fuel is, what the total radionuclide inventory available is, whether heating or cooling systems may be required to maintain the fuel storage PDC, and other issues related to the review of the fuel handling system. At the CP phase, specifics are not expected, but ACU should have some idea of the system design bases and operational envelope.
9.2-4	B. Travis		9.2	ACU should provide additional context (if more detailed analyses exist) regarding the potential for fission product releases during fuel salt storage or handling (e.g. spill of fuel or fuel-handling accident).

PSAR Review Information Needs for Chapter 13, Section 13.1, "Accident-Initiating Events and Scenarios"

Item #	Reviewer(s)	Date Sent to ACU (Accession No.)	PSAR Chapter or Topic	Question
13.1-1	B. Travis		13.1.1	The NRC staff requests access to the calculations used to produce Figures 13.1-1 and 13.1-2. These include: • A full accounting of source term and inventory, including differences between the assumed values and the actual expected inventory • All initial conditions, assumptions, and inputs for TEDE calculations • All codes and calculations for radiological dose consequence analysis • Information on how the cited equations are implemented, including inhalation rate equations (what values are used for breathing rates?)

13.1-2	B. Travis		13.1.1	ACU should provide detail on how tritium is treated and accounted for, including in the analyses in Chapter 13. This is a cross-cutting issue in a number of different sections, and more specific questions may be present in other sections.
13.1-3	B. Travis		13.1.1	ACU should provide additional context regarding the basis for the assumptions documented in Table 13.1-1. Specifically: • The source term states that only Te, I, Xe, Br, Kr are considered as source term, as gases, and non-noble gases leak at 10% of the rate of noble gases. What is the basis for the use of this as the source term inventory, and what is the basis for use of a different leak rate between the gas species? • The leakage rate is specified as 0.05% per day from the enclosure for noble gases, 0.005%/day for non-noble gases, and then only 1%/day of the cell volume (outside the enclosure). How does ACU plan to confirm these values (and what was the basis for choosing these values) as part of the design basis (e.g., technical specifications, testing)? Further, the PSAR specifies "the pit" as a holdup volume between cell and bay – how is this treated with regards to the above assumptions? • No information is provided regarding material interactions resulting from the release of the fuel to the reactor enclosure. Were these considered when developing the assumptions in Table 13.1-1 (e.g., leakage, source term species released, heating)?
13.1-4	B. Travis		13.1.1	Non-MHA salt spills are specified as potential accidents in Chapter 13 but no further discussion is provided. ACU should provide context regarding other salt spills, especially if it is possible that they happen outside the leakage barriers assumed in the MHA.
13.1-5	B. Travis		13.1.1	Fission product models are described in Chapter 13, but RTMS heat removal is not, nor is it described elsewhere in the PSAR. There is no clear basis for statements related to the heat removal capability of the system. The NRC staff requests further information regarding the capability and function of the RTMS system.
13.1-6	B. Travis		13.1.1	The NRC staff requests access to the calculations used to produce the figures and summaries in these sections and that are captured in brief in Table 13.2-1. Specifically: • What are the initial conditions, assumptions, and inputs associated with each transient calculation? • What values are used to build the preliminary reactor model, and how representative are these values of the potential design envelope (e.g., fuel parameters that are not yet validated)? • What constitutes the sequence of events for each transient (e.g., when does the upset condition occur and what are relevant datapoints such as reaching limiting conditions)? • Although the NRC staff is interested in reviewing all of the analysis, there is particular interest in the analysis related to void collapse (Section 13.1.5.1). Staff is seeking additional information on parameters not documented in the PSAR (e.g. peak temperature in the salt and data on temperatures near the reflector).
13.1-7	B. Travis		13.1.8	Note (No response required): NRC staff is not asking questions on the scoping of the external events as part of questions in Chapter 13; those questions will be addressed in Sections 2 and 3 of the PSAR, as appropriate. NRC staff notes that if the external event profile changes, those events would need to be analyzed here (flooding, specifically, is stated to be addressed in Section 4.5, but is stated to be precluded as an external event).
13.1-8	A. Chereskin		13.1	Will circulating activity limits be set to ensure the salt maintains a dilute enough solution to avoid positive deviations from ideality (i.e. higher vapor pressure)?
13.1-9	A. Chereskin		13.1	Is tritium release from graphite considered in accident analyses?
13.1-10	A. Chereskin		13.1	Is oxidation of SSCs considered during postulated accidents (e.g. air or water ingress)? Is oxidation and precipitation of uranium considered during postulated accidents?
13.1-11	A. Chereskin		13.1	In a salt spill accident, has the potential for overcooling and precipitation of fissile material been considered to ensure no accidental criticality occurs? In general, describe how solubility limits were considered.
13.1-12	A. Chereskin		13.1	Do postulated accidents consider UF_6 and F_2 products from frozen fuel salt? Can generation of F_2 challenge RTMS integrity during the MHA?