From: Richard Rivera
Sent: Friday, September 6, 2024 12:01 PM
To: Rusty Towell
Cc: Benjamin Beasley; Lester Towell; Edward Helvenston; Brian Bettes; Michael Balazik; Ben Adams; Stephen Philpott; Greg Oberson (He/Him)
Subject: Issuance of Audit Reports for Abilene Christian University's Construction Permit Application Review
Attachments: Audit Report for ACU MSRR CP PSAR General Topics.pdf; Audit Report for ACU MSRR CP PSAR Section 9.2 and Chapter 13.pdf; Audit Report for ACU MSRR CP PSAR Technical Topics.pdf; Audit Report for ACU MSRR CP PSAR Chapter 7.pdf; Audit Report for ACU MSRR CP PSAR Chapters 2 and 3.pdf; Audit Report for ACU MSRR CP PSAR Chapters 4 and 6 and Section 9.6.pdf

Dear Dr. Towell,

By letter dated August 12, 2022, (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML22227A201), as supplemented, Abilene Christian University (ACU) submitted a construction permit (CP) application for its proposed Molten Salt Research Reactor (MSRR), pursuant to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Section 104c of the Atomic Energy Act of 1954, as amended. By letter dated November 18, 2022, the NRC staff accepted ACU's MSRR CP application for docketing (ML22313A097).

To assist in the detailed technical review of the application, the NRC staff conducted a total of six audits (see audit plans at ML2301A089; ML23065A052; ML23065A048; ML23065A055; ML23065A051; and ML23065A056), each focusing on certain technical areas or a specific set of chapters of the preliminary safety analysis report (PSAR) submitted with the CP application. The NRC staff conducted all six audits primarily in a virtual setting, via remote meetings with ACU, and utilizing an electronic reading room portal to seek clarification, gain understanding, and verify information contained in the PSAR.

The NRC staff completed the regulatory audit process and held an exit meeting covering all six audits with ACU on August 20, 2024. The audit summary reports documenting the results of these audits are enclosed with this email, and this email and the reports will be made publicly available in ADAMS.

If you have any questions, please contact Richard Rivera at (301) 415-7190 or via email at Richard.Rivera@nrc.gov, Edward Helvenston at (301) 415-4067 or via email at Edward.Helvenston@nrc.gov, or Brian Bettes at (301) 415-3762 or via email at Brian.Bettes@nrc.gov.

Sincerely,

Richard Rivera

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Enclosures: As stated
Cc: GovDelivery Subscribers

Concurrence on Audit Reports

OFFICE	NRR/DANU/UAL2/PM	NRR/DANU/UTB1/BC	NRR/DANU/UAL2/BC(A)
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DATE	09/04/2024	09/05/2024	09/05/2024

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Audit Report for ACU MSRR CP PSAR General Topics.pdf	416634	
Audit Report for ACU MSRR CP PSAR Section 9.2 and Chapter 13.pdf	255268	
Audit Report for ACU MSRR CP PSAR Technical Topics.pdf	209969	
Audit Report for ACU MSRR CP PSAR Chapter 7.pdf	206653	
Audit Report for ACU MSRR CP PSAR Chapters 2 and 3.pdf	469044	
Audit Report for ACU MSRR CP PSAR Chapters 4 and 6 and Section 9.6.pdf	599828	

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SUMMARY REPORT FOR THE REGULATORY AUDIT OF ABILENE CHRISTIAN UNIVERSTY MOLTEN SALT RESEARCH REACTOR CONSTRUCTION PERMIT PRELIMINARY SAFETY ANALYSIS REPORT GENERAL TOPICS

March 2023 – August 2024

1.0 BACKGROUND AND PURPOSE

By letter dated August 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22227A202), as supplemented by letter dated October 14, 2022 (ML22293B816), Abilene Christian University (ACU) submitted to the U.S. Nuclear Regulatory Commission (NRC), an application for a construction permit (CP) for a molten salt research reactor (MSRR), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Section 104c of the Atomic Energy Act of 1954, as amended.

This audit enabled the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) to gain a better understanding of ACU's PSAR Chapters listed below through review and discussion of underlying supporting documentation. Enhanced understanding and communications supported effective and efficient resolution of technical issues, including through development of information needs where needed.

- Chapter 1, "The Facility"
- Chapter 5, "Molten Salt Reactor Cooling Systems"
- Chapter 8, "Electrical Power Systems"
- Chapter 9 (except Sections 9.2 and 9.6), "Auxiliary Systems"
- Chapter 10, "Experimental Facilities and Utilization"
- Chapter 11, "Radiation Protection Program and Waste Management"
- Chapter 12, "Conduct of Operations"
- Chapter 14, "Technical Specifications"
- Chapter 15, "Financial Qualifications"
- Chapter 16, "Other License Considerations"
- Chapter 17, "Decommissioning and Possession-Only License Amendments"
- Chapter 18, "Highly Enriched to Low-Enriched Uranium Conversions"

2.0 AUDIT REGULATORY BASES

The basis for the audit is the regulations in 10 CFR Part 50, Sections 50.34(a), "Preliminary safety analysis report," and 50.35, "Issuance of construction permits."

3.0 AUDIT OBJECTIVES

The primary objective of the audit was to enable a more effective and efficient review of PSAR Chapters 1, 5, 8, 9 (except Sections 9.2 and 9.6), 10, 11, 12, and 14-18 through the staff's review and discussion of supporting documentation with ACU. Gaining access to underlying documentation and engaging in audit discussions about the MSRR design facilitated the staff's understanding of the CP application and aided in assessing the safety of the proposed research reactor. The audit improved communication and provided detailed information for the staff.

4.0 SCOPE OF THE AUDIT AND AUDIT ACTIVITIES

The audit was conducted from March 2023 to August 2024, via teleconference and the online refence portal (Certrec), and also during a site visit conducted May 17-18, 2023. The staff conducted the audit in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction NRR-LIC-111, Revision 1 “Regulatory Audits” (ML19226A274).

Members of the audit team, listed below, were selected based on their detailed knowledge of the audit subject matter. Audit team members included:

1. Boyce Travis, NRR (Senior Nuclear Engineer)
2. Kyle Song, NRR (Electronics Engineer)
3. Shelia Ray, NRR (Senior Electrical Engineer)
4. Nicholas Hansing, NRR, (Mechanical Engineer)
5. Charles Peabody, NRR (Nuclear Engineer)
6. Christopher (Ben) Adams, NRR (Nuclear Engineer)
7. Ryann Bass, NRR (Reactor Systems Engineer)
8. Naeem Iqbal, NRR (Fire Protection Engineer)
9. Alexander Chereskin, NRR (Materials Engineer)
10. Adakou Foli, NRR (Electrical Engineer)
11. Vijay Goel, NRR (Electrical Engineer)
12. Steve Jones, NRR (Senior Safety and Plant System Engineer)
13. Zachary Gran, NRR (Health Physicist)
14. Edward Stutzcage, NRR (Reactor Scientist (Radiation Protection))
15. Kenneth Mott, NSIR (Emergency Preparedness Specialist)
16. Edward Robinson, NSIR (Emergency Preparedness Specialist)
17. Dong Park, NRR (Reactor Operations Engineer (QA Inspector))
18. Amy Beasten, NRR (Reactor Engineer)
19. Glenn Tuttle, NMSS (MC&A Physical Inspection Analyst)
20. Suzanne Ani, NMSS (MC&A Physical Scientist)
21. John Parillo, NRR (Senior Reactor Engineer (Radiation)
22. Shawn Harwell, NMSS (Financial Analyst)
23. Brian Klement
24. Richard Rivera, NRR (Project Manager)
25. Zackary Stone, NRR (Project Manager)
26. Mohsin Ghazali, NRR (Project Manager)
27. Brian Bettes, NRR (Project Manager)
28. Edward Helvenston, NRR (Project Manager)
29. Michael Balazik, NRR (Project Manager)
30. Michelle Hayes, NRR (Branch Chief, Technical)
31. Gregory Oberson, NRR (Branch Chief, Technical)
32. Michael Wentzel, NRR (Branch Chief, Licensing)
33. Stephen Philpot, NRR (Branch Chief, Licensing)

Prior to the audit, the audit team reviewed the PSAR and defined the general range of topics (e.g., PSAR chapters and sections) in the audit plan dated March 2, 2024 (ML23065A052), to be addressed and focused on during the audit. The following table documents dates that the staff transmitted audit questions and when audit meetings were held:

Audit Questions (ADAMS Accession No.)	Audit Meetings
March 21, 2023 (ML23080A191)	May 17, and 18, 2023
April 26, 2023 (ML23116A171)	July 20, and 27, 2023
May 9, 2023 (ML23129A780)	August 3, and 10, 2023
July 3, 2023 (ML23184A146)	September 7, 14, and 21, 2023
July 11, 2023 (ML23192A454)	October 5, and 17, 2023
July 13, 2023 (ML23194A159)	December 7, 2023
October 24, 2023 (ML23297A044)	January 11, 2024
	February 6, 2024
	March 26, 2024
	June 6, 2024
	August 20, 2024 (exit meeting)

The staff reviewed the following documents via the ERR:

- Written responses that ACU prepared for certain questions to address the questions and/or facilitate discussion with NRC staff
- PSAR pages indicating changes proposed by ACU in response to various audit questions
- CAP88 Effluent Synopsis 11.1
- CAP88 Calculation Summary
- CAP88 GUI Inputs
- CAP88 Dose and Risk Summary
- MSRR Effluent Doses to the Public
- Abilene Christian University Waste Management Plan Final 8-2022
- MSRR-GT-SNP-2021-01-R2(MaterialCompositions)
- MSRR-GT-SNP-2021-08-R1(SoilActivationReport)
- MSRR-GT-SNP-2022-01-R0(EpsilonShieldingReport)
- MSRR-TAMU-AESL-2023-02 Rev-1 Drain tank criticality
- 20223.0614 Ltr to ACU re Adv Contracting Req of NWPA
- “MSRR Neutron and Gamma Heating,” dated December 22, 2021.
- ACU’s process flow diagram revision C, “PFD SK-0001_REV_C,” provided July 12, 2023.
- “Noble Gas Fission Product Generation Rate and Air Activation,” dated June 6, 2023.
- “RELAP5-3D model iota design Description,” provided April 13, 2023.
- ACU SERC – Mass Concrete QAQC Plan R1
- ACU SERC – Site Specific QAQC Plan R1
- ACU SERC – eHT QA-QC Letter
- ACU SERC–Parkhill QA-QC Letter
- Linbeck QAQC Manual 20221115

The NRC staff previously issued an interim audit report related to this audit by letter dated June 22, 2023 (ML23157A064).

5.0 SUMMARY OF AUDIT OUTCOME

The staff's audit focused on the review of supporting documents associated with the questions provided to ACU during the audit. The staff reviewed information through the ACU Certrec portal and held discussions with ACU staff to understand and resolve questions. In many cases, ACU updated the PSAR to resolve items discussed in the audit. For some topics, the staff had clarification questions which were discussed with ACU and summarized for this audit report. The tables below replicate specific audit questions transmitted in emails to ACU as listed above, and also summarize the resolution of the audit questions. A table below also lists and summarizes the resolution of additional topics discussed as part of this audit beyond the scope of the specific audit questions.

Resolution of Question on General PSAR Review (General to entire PSAR/application)

Question Number	Question	Resolution
PSAR-1	The NRC staff notes that there are some inconsistencies with different terminology used throughout the PSAR. The NRC staff requests that ACU provides clarification for the following items and, where applicable, apply these consistently throughout the PSAR. I. The NRC staff notes that there appears to be some instances of the term “access tank,” which the NRC staff infers is early terminology for what is elsewhere referred to as the reactor access vessel (RAV) (PSAR Section 1.2.3.3 and 1.2.3.6). II. The NRC staff notes that stronger, more consistent controls are necessary in discussing the 316H materials used in the MSRR design. The PSAR uses “316H stainless steel,” “SS316H,” and “stainless steel 316” in various places. The NRC staff notes that if these components will satisfy the ASME Code requirements, then the materials used to fabricate them must also meet tighter controls on the material. As a technical example, “316 steel” can have wide variability in carbon content, so tighter controls are often indicated to lessen the potential for stress corrosion cracking. The NRC staff notes that loss of normal electric power (LONEP) is used throughout the PSAR and often points to Chapter 13. Upon reviewing PSAR Section 13.1.10, on Loss of Normal Electric Power, the acronym is no longer used.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), which addressed the inconsistencies found by the NRC staff.

Resolution of Question on Chapter 1, "The Facility," Section 1.7, "Compliance with the Nuclear Waste Policy Act of 1982"

Question Number	Question	Resolution
1.7-1	Section 302 of the Nuclear Waste Policy Act of 1982, as amended (the NWPA) (42 USC § 10101 et seq.), specifies that the NRC may require, as a precondition to issuing a facility operating license for a research reactor, that the applicant have entered into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel that may result from the use of such license. Furthermore, the NWPA specifies that the NRC shall not issue a license to any person to use a utilization facility under section 103 or 104 of the AEA unless: (i) such person has entered into a contract with DOE for disposal under section 302 of the NWPA or (ii) DOE affirms in writing that that such person is actively and in good faith negotiating with DOE for such a contract. The staff notes that to be in compliance at the CP stage, an applicant needs to submit documentation showing communications in good faith between the applicant and DOE to enter into a contract for the disposition of high-level waste and nuclear fuel. (See, for example, ADAMS Accession No. ML23019A360.) MSRR PSAR Section 1.7, "Compliance with the Nuclear Waste Policy Act of 1982," states: "Abilene Christian University intends to enter into a contract with the Department of Energy for required fuel cycle services. This will be discussed further in the Operating License application, consistent with Section 302(b)(1) of the Nuclear Waste Policy Act of 1982." MSRR PSAR Appendix 15A provides a letter from DOE to ACU, dated November 15, 2019, indicating that DOE will consider ACU requests for fuel services including fuel disposition once ACU has an NRC-licensed research reactor. However, this letter does not appear to provide specific documentation from DOE of communications between ACU and DOE to enter into a fuel disposal contract.	ACU docketed a letter on August 17, 2023 (ML23230A392) to provide documentation of good faith negotiations between ACU and the DOE. In addition, ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to provide clarification in PSAR section 1.7 regarding the communications between ACU and the DOE.

	Please discuss whether ACU has specific documentation showing communications in good faith between ACU and DOE to enter into a contract for the disposition of high-level waste and nuclear fuel, as necessary to comply with the NWPA for the issuance of a CP for the MSRR.	

Resolution of Questions on Chapter 5, Section 5.2, "Fuel System Boundary and Fuel Salt Heat Transport"

Question Number	Question	Resolution
5.2-1	PSAR Section 4.2.2.2, "Reactivity Control System," states that the control rods and thimble material is 316H SS. ORNL/TM-2020/1478 Section 5.2, "Fuel System Boundary and Fuel Salt Heat Transport," states that the fuel salt should limit corrosion of control element surfaces. What data are available and what testing is planned to determine acceptable environmental conditions to limit control rod cladding degradation? How do these data bound anticipated environmental conditions (e.g. fluence, salt impurities, fission products, etc.)?	ACU's response to Request for Additional Information (RAI) 2 (ML24121A272) provides the resolution to this question.
5.2-2	How will secondary purity limits be set in order to ensure that corrosion of the heat exchanger (i.e. fuel salt boundary) is minimized in order to maintain consistency with PDCs 4, 14, and 31? What data and/or testing will be used to set purity limits for the intermediate salt?	ACU's response to RAI 2 (ML24121A272) provides the resolution to this question.
5.2-3	Has the potential for cooling and precipitation of fissile material or corrosion products in colder portions of the fuel salt boundary been considered for impacts on reactivity and/or fouling?	ACU stated during audit discussions that full flow blockage events were analyzed when developing PSAR Section 13.1.4 and were determined to not result in unsafe conditions.
5.2-4	Design Criteria (DC) 30 commits to "appropriate quality standards". The NRC staff requests that ACU describe what specifications are used to	ACU's response to RAI 1 (ML24094A332) and RAI 2

	design, fabricate, erect, and test the components in the fuel salt boundary as they relate to the corrosion resistance of the fuel salt and the primary cooling salt.	(ML24121A272) provides the resolution to this question.
5.2-5	In PSAR Section 5.3, ACU mentions tritium generated in the reactor may migrate into the coolant loop through the heat exchanger tube walls; PSAR Section 13.1.2 states tritium generated in the fuel salt can diffuse through the heat exchanger and accumulate in the coolant salt in the secondary cooling loop. PSAR Section 5.2.4 only describes the instrumentation and control system that ensures the required range for safe operation which provides information on how fuel temperature and pressure are monitored. How will fission product activity (tritium, noble gas, iodine, particulate, etc.) in the secondary cooling system be detected or managed?	ACU stated during audit discussions that radiation detectors will be used in the coolant loop for monitoring radiation levels. The NRC is not approving the radiation protection program or monitoring system because it is not required at the CP stage. The NRC will review the final design of the radiation monitoring system and the radiation protection program during the MSRR OL application review.
5.2-6	PSAR Section 5.2.3 does not explain how the drain tank removes or manages fission gas volume or gaseous fission products when irradiated fuel salt is drained into the drain tank. Does PSAR Section 3.1.2.6, Criterion 61 which states, the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions, apply to the drain tank?	During audit discussions ACU stated that the drain tank is connected to the primary loop with no valve or seal separating the gas space of the drain tank from the volumes of the primary loop and the gas management system (GMS) of the primary loop. Any gaseous fission products will remain at the surface of the drain tank or be collected by the GMS. ACU stated further that the drain tank is a component of the fuel loop and is designed to the same standard as the core vessel, therefore the drain tank has all the same fission product barriers that exist when the fuel is within the core. The drain tank is designed to hold fuel salt in a non-

		critical configuration in postulated accident scenarios.

Resolution of Questions on Chapter 5, Section 5.4, "Fuel Salt Cleanup System"

Question Number	Question	Resolution
5.4-1	The NRC staff notes that the PSAR does not appear to describe how on-line chemistry control will be performed outside of stating that metallic beryllium (Be) can be added to control the redox potential. How will the composition of the salt (including generation of fission products) be monitored and maintained during operations? Additionally, will ACU provide required action times to correct salt chemistry as part of the OL application?	ACU stated during audit discussions that samples of the fuel salt will be removed remotely from the reactor. These samples will be analyzed in the radiochemistry lab for redox potential and elemental composition. ACU clarified that the redox potential of the fuel salt is influenced by the U(IV/III) ratio. Through the addition of Be to the fuel salt, ACU states that the redox potential can be adjusted to remain within a specified range. The NRC will determine the acceptability of the on-line chemistry control process during the MSRR OL application review.
5.4-2	How will the redox potential be monitored? The PSAR is not clear about how this will be achieved, and the NRC staff are not aware of commercially available electrochemical potential probes for use in fuel salts.	ACU stated during audit discussions that off-line analysis for redox potential will be performed by collecting samples of the fuel salt remotely. The NRC will determine the acceptability of the redox potential monitoring process during the MSRR OL application review.

Resolution of Questions on Chapter 5, Section 5.5, "Salt Makeup System"

Question Number	Question	Resolution
5.5-1	The NRC staff requests that ACU make available the supporting basis (e.g., provide a calculation or analysis) for the statement "Preliminary analyses suggest that radiation heating does not lead to exceeding of safety limits."	During the audit, ACU provided documentation on the ERR of a preliminary heating analysis which concludes that energy deposition from neutron and gamma deposition is conservatively between 2 and 20 mW/cm^3, depending on the location.
5.5-2	The NRC staff requests that ACU provide a supporting basis for how the fuel salt chemistry changes during the course of operation (critically, power density, temperature, thermal hydraulics behavior, etc.). The NRC staff notes fuel salt chemistry changes are directly related to the plant performance which may include cooling performance, and the NRC staff expects at least a set of estimated ranges for fuel characteristics were used to produce preliminary analyses. I. In PSAR Section 5.5.1 ACU stated a small amount of excess volume in the RAV allows the system to accommodate temperature and pressure fluctuations and the resultant volume changes. What is the expected amount of this volume? In PSAR Section 4.2.1.1, ACU stated 100 liters (L) out of 500L are split between the RAV, reactor pump, heat exchanger, and associated piping. 20% of the 500L appears to be a significant amount, considering the potential uncertainty associated with fuel salt parameters. The NRC staff requests ACU provide a preliminary analysis of the salt volume change and its associated impacts on temperature and pressure. II. How does ACU plan to control or ensure there are no significant changes in fuel salt characteristics such as redox chemistry, etc., to control important fuel characteristics within the analytical assumptions?	I. During the audit, ACU provided documentation on the ERR of a preliminary assessment of the relevant thermophysical properties of the MSRR fuel, estimating temperature-dependent correlations of the relevant properties. ACU stated further that the anticipated change in volume over the operating range of the reactor is approximately 14 Liters, and that the RAV is appropriately sized to accommodate this volume change. II. ACU stated during audit discussions that regular monitoring of fuel salt composition, U(IV/III), and impurity content will be performed. ACU states that the reactor will be shutdown if

		the analyses indicate that the fuel salt has fallen out of specification. In addition, ACU stated that experimental real-time redox monitoring methods will be implemented when available but not required as limiting conditions for operations (LCOs).

Resolution of Questions on Chapter 8, "Electrical Power Systems" (General Questions Relevant to Entire Chapter)

Question Number	Question	Resolution
8-1	Criterion 17, "Electric power systems," states, in part, "…If electric power is not needed for anticipated operational occurrences or postulated accidents, the design shall demonstrate that power for important to safety functions is provided." PSAR section 8.2 states, in part, "Considering Design Criterion 17, Electric power systems, electric power systems are provided to permit functioning of structures, systems, and components. Safe shutdown and long-term decay heat removal are passive, and no electric power is required. The Chapter 13 analyses show that with complete loss of electrical power the design limits for fission product barriers are not exceeded. Thus, the bases for Design Criterion 17 are met with no electrical power required for safe shutdown, decay heat removal or accident mitigation." Based on the NRC staff's review, it's not apparent that the PSAR Chapter 8 provides any discussions about the important to safety functions for which electrical power is provided.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to revise DC 17 to remove the term "important to safety."

	a. Please discuss the “important to safety” functions or loads, for which the power will be provided to meet the bases of the Design Criteria 17.	
8-2	PSAR Table 3.4-1, “Structures, Systems, and Components and Associated Quality Level Group,” identified the normal electrical power as safety-related and the backup electrical power as non-safety-related. PSAR section 8.1 states, in part, “The normal electrical power system does not perform any safety-related functions and is not credited for the mitigation of postulated events or performing safe shutdown functions.” PSAR section 8.2 states, in part, “The UPS systems do not perform any safety-related functions and are not credited for mitigation of postulated events. The systems are used for monitoring functions and are not credited with maintaining or performing safe shutdown functions. […] Consistent with Design Criterion 18, Inspection and testing of electric power systems, electric power systems are designed to permit appropriate periodic inspection and testing. However, because loss of power shuts down the reactor and long-term decay heat removal is passive, inspection and testability will be limited to components necessary to ensure trip functions.” In PSAR section 3.1.2.2, Criterion 18, “Inspection and testing of electric power systems,” states, in part, “Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components.” Based on the NRC staff’s review, it appears that the safety classification of the normal power system and the backup power systems is not consistent in the PSAR, and whether the electrical power systems	a. ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to revise the definitions of safety-related and non-safety-related as well as a updating the SSC classification in PSAR Chapter 3. The normal and backup electric power systems are classified non-safety-related in Table 3.4-1. b. ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to revise DC 18 to state that “routine monitoring of operability and inspection of components will assure reliable MSRR operation and availability of electric power.”

	include any components necessary to ensure trip functions. Also, with respect to the design criterion 18, it does not appear that the safety classification of the normal and backup electrical power systems as “important to safety” or “non-safety- related” is clear. a. Provide the safety classification (i.e., safety-related, non-safety-related, important to safety) of the normal electrical power system and the emergency (backup) electrical power system. Table 3.4-1 should be consistent on the safety classification for the normal and backup electrical power systems. b. Clarify if the normal electrical power system and the backup electrical power system will be inspected and tested following the requirements of the design criterion 18. If not, clarify whether the design criteria 18 is applicable to the electrical power systems and how the normal and backup electrical power systems meet the design criterion 18.	

Resolution of Questions on Chapter 8, Section 8.1, “Normal Electrical Power System”

Question Number	Question	Resolution
8.1-1	The NRC staff notes the reactor operations that require normal electrical power, as described in other chapters of the PSAR, are not discussed in PSAR Chapter 8. The NRC staff noted the following examples of equipment/system are expected require the normal electrical power: a. Electrical heaters used to preheat the reactor system and the heat removal system during startup and low power operation. b. Ventilation, and air conditioning systems c. Control rod drives d. Instrumentation and controls systems (Figure 7.2-1) e. Reactor trip valves, reactor pump, and coolant pumps f. Reactor thermal management system	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to provide a list of systems that will be supplied by the normal electric power in PSAR Chapter 8. This list includes safety-related systems/equipment. PSAR Chapter 8 states that safety-related SSCs do not require electric power to perform their safety functions, and electric isolation of safety-related systems and non-safety related systems is not required.

	g. Primary and auxiliary heat removal system (electrical heaters and air blowers) Please confirm that the normal electrical power is provided to the above-mentioned equipment/systems and provide additional equipment/systems that will be supplied by the normal electrical power, if any. Also, specify the equipment/systems that are safety-related and require electrical isolation from the non-safety-related normal electrical power equipment in PSAR Chapter 8.	During audit discussions, the staff discussed with ACU the need for an electrical isolation between the safety-related system/equipment and the non-safety-related electric power system. ACU stated that it will confirm that no fault on the electric power systems will impact the safety-related SSCs. ACU revised PSAR section 8.1 to state that normal electrical protective equipment will be provided for the electric power system and a malfunction of the normal power system will not prevent a reactor trip or impair other safety functions. ACU included in the revised DC 17 this statement "The electric power system shall not be able to impair a safety function," to ensure that a fault on the non-safety-related electric power systems does not impact the safety-related SSCs.

Resolution of Questions on Chapter 8, Section 8.2, "Emergency Electrical Power Systems"

Question Number	Question	Resolution
8.2-1	The PSAR, Table 3.1-1 "Cross Reference to Preliminary Safety Analysis Report Sections," provides a cross reference of the MSRR-specific DC 17 and 18 to the PSAR Section 8.2.3. The NRC staff notes that the PSAR	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to revise Table 3.1-1 to replace Section 8.2.3 with

	does not include a section 8.2.3. Please clarify the section in Chapter 8 that is related to DC 17 and 18.	Section 8.2.1 where DC 17 and DC 18 are discussed.
8.2-2	PSAR Section 8.2.1, “Backup Electrical Power Systems,” states, in part “The UPS are sized to provide sufficient power to those selected loads to maintain functionality for at least 24 hours after loss of power to the facility for operational convenience.” a. Please explain what is meant by “operational convenience,” and provide the technical basis for the 24 hours. Also, explain any impacts if UPS power is not available to the selected loads after 24 hours. b. PSAR Section 13.1.10, “Loss of Normal Electrical Power,” states, “If offsite electrical power is lost, all electrically operated systems, including the auxiliary heat removal system, will stop in the MSRR.” Clarify if power will remain available to selected electrical loads powered by the UPS in the MSRR if offsite power is lost, as stated in PSAR Section 8.2.1. If so, please consider revising the PSAR Section 13.1.10 with respect to the UPS.	In its response to question 8.2-2.a, ACU stated that “operational convenience allows UPS power to provide power to instrumentation so operators can confirm the shutdown of the reactor without using manual methods. There is no safety issue if UPS power is lost at the time of shutdown, so the 24 hours was a discretionary, not safety-related, design decision, as the MSRR is passively safe and no accident analysis credits UPS operation.” During a call, the staff discussed with ACU the impact of not having the uninterruptible power supply (UPS) to power other selected loads (i.e., emergency lighting, the fire alarm system, the security system) for the first 24 hours after loss of normal electrical power. ACU stated that there is no impact of the UPS unavailability on emergency lighting (flashlight can be used) and the fire alarm system (no safety-related fire system is supplied by the UPS) when normal electric power is lost. ACU also stated that the security system is addressed separately.

		In response to question 8.2-2.b, ACU stated that PSAR section 13.1.10 conservatively assumes failure of UPS power during loss of normal electrical power.

Resolution of Questions on Chapter 9, "Auxiliary Systems," Section 9.1, "Heating, Ventilation, and Air Conditioning Systems"

Question Number	Question	Resolution
9.1-1	The NRC staff requests that ACU provide a basis for concluding normal HVAC does not affect post-accident diffusion from confinement (e.g., through development of differential pressure within building). Additionally, the NRC staff would like clarification on whether the ESF actuation of HVAC shutdown is necessary.	ACU modified Sections 6.2.1 and 9.1.3 of the PSAR by letter dated July 30, 2024 (ML24219A258), to state that the heating, ventilation, and air conditioning (HVAC) system does not assist in the isolation of the reactor cell and that the reactor cell leak rate would be confirmed to remain within design limits for all HVAC modes of operation. In addition, ACU clarified that releases during the maximum hypothetical accident (MHA) are assumed to immediately enter the environment, this assumption bounds the performance of the HVAC system.
9.1-2	The NRC staff notes that, based on the information in the PSAR, it is not clear how HVAC for the fuel salt storage enclosure functions. Request ACU provide context in this subject area.	ACU modified Section 9.1.3 of the PSAR by letter dated July 30, 2024 (ML24219A258), to clarify that the HVAC system does not supply air to the fuel salt storage enclosure.

9.1-3	PDC 19, control room, states that adequate habitability will be provided for accident conditions. PSAR Table 19.4-1 identified that up to 15kg of Anhydrous Hydrogen Fluoride (HF) could be present in the facility for treatment of the fuel salt. This quantity could exceed habitability limits in RG 1.78 under accident conditions unless suitable protection against transport of the gas to the control room is available. The NRC staff requests clarification on how PDC 19 would be satisfied.	ACU modified Section 9.1.3 of the PSAR by letter dated July 30, 2024 (ML24219A258), to state that if the control room were to become uninhabitable, the operators would evacuate the control room. In addition, ACU modified PDC 19 to state that the control room design would support egress rather than occupancy under accident conditions considering the facility passive safety features.

Resolution of Questions on Chapter 9, Section 9.4, "Communication Systems"

Question Number	Question	Resolution
9.4-1	The NRC staff requests that ACU provide additional information/details regarding communication systems specifically: I. PSAR Section 9.4.2 mentions a "cell phone in the control room." Is the cell phone a facility phone or personal device? Are other individuals in the facility carrying cell phones as well and are they facility or personal devices? What controls are in place to ensure the device is not carried off-site? II. Is the intent to satisfy the two-way communication criteria with this cell phone? If not the cell phone, please provide information to address the two-way communication criteria.	I. ACU stated during audit discussions that the control cell phone is a facility device and administrative controls will be used to prevent it from leaving the control room. In addition, ACU stated that personal cell phones will be used in other areas of the facility. II. ACU stated during audit discussions that the two-way communication criteria will be satisfied through the use of the control room desk

		phone and individual's cell phones in the facility. In addition, ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to state that hard line phones will be located in the control room, health physics office, and research bay.

Resolution of Questions on Chapter 9, Section 9.5, "Possession and Use of Byproduct, Source, and Special Nuclear Material"

Question Number	Question	Resolution
9.5-1	PSAR Section 9.5 states that special nuclear material (SNM) at the MSRR will be uranium in the fuel salt, and byproduct materials at the MSRR will be those generated by MSRR operation and mixed in the fuel salt. Discuss whether ACU plans to possess other radioactive material (i.e., material not contained in fuel or generated by MSRR operation) under the MSRR license.	ACU modified the PSAR section 9.5.1 by letter dated July 30, 2024 (ML24219A258), to state that small quantities of SNM, source material, and byproduct material may be present at the facility beyond what is in the fuel salt. Examples of this include SNM or byproduct material for a neutron startup source, U-235 in fission chambers, and natural uranium for usage as a surrogate in the radiochemistry lab.
9.5-2	PSAR Section 9.5.1 states that SNM will be located and handled in the "fuel receiving station located in the research bay" and in the "experimental sampling system." However, the staff notes that this terminology does not appear to be used elsewhere in the PSAR.	a. ACU clarified in information provided for audit and during audit discussions that the "fuel receiving station" is part of the fuel handling system (FHS) and noted

	a. Please describe the "fuel receiving station." Is this part of the fuel storage enclosure (see PSAR Figure 3.1-3) located in the systems pit, or is it something different? If the fuel storage enclosure and tanks illustrated in PSAR Figure 3.1-3 are not included in the PSAR Section 9.5.1 description of where SNM will be used and handled, should these be added? b. Please describe the "experimental sampling system." Is this the same as the "sample extraction system" discussed in PSAR Sections 4.3.11 and 9.5.2, or is it something different?	that PSAR section 9.2.3 discusses this station. ACU also modified the PSAR section 9.5.1 by letter dated July 30, 2024 (ML24219A258), to replace the "fuel receiving station" reference with the more inclusive "FHS." b. ACU stated in information provided for audit and during audit discussions that the experimental sampling system, the sample extraction system, and salt sampling system are all referring to the same system. ACU modified the PSAR section 9.5.1 by letter dated July 30, 2024 (ML24219A258), to clarify the wording.
9.5-3	PSAR Section 9.5.2 states that byproduct material will be present in the "reactor system." Please clarify what portions of the MSRR this is intended to include. For example, does it include the fuel storage enclosure and tanks (the staff notes that information in PSAR Chapters 3 and 13 appears to suggest that the tanks could be used to hold irradiated fuel salt)?	ACU modified the PSAR section 9.5.2 by letter dated July 30, 2024 (ML24219A258), to clarify that byproduct material is expected to be present in the reactor system (reactor vessel, RAV, drain tank, fuel side of the heat exchanger, reactor pump, and piping), portions of the gas management system (GMS), FHS, fuel salt sample system, and the

		radiochemistry lab.
9.5-4	PSAR Section 9.5.2 states that byproduct material will be present in the off-gas system. According to PSAR Section 9.6.2, the off-gas system is a subsystem of the gas management system (GMS). Given that gases in the entire GMS interface with fuel salt as discussed in PSAR Section 9.6.2, should PSAR Section 9.5.2 refer to the entire GMS as containing byproduct material? (The staff notes that PSAR Section 9.5.1 indicates the GMS generally as a location where SNM is handled.)	ACU stated in information provided for audit that byproduct material is expected to be found only in the portions of the GMS that are downstream of the salt bearing vessels.

Resolution of Questions on Chapter 9, Section 9.8, "Other Auxiliary Systems"

Question Number	Question	Resolution
9.8-1	Figure 3.1-2, "Cross Section View of Science and Engineering Research Center," of the MSRR PSAR, Rev. 0, depicts a crane that can operate over the research bay and systems pit. Design and operation of the crane does not appear to be addressed in the PSAR, although the NRC staff notes that use of the crane with attached heavy loads could pose a potential challenge to the safety of the MSRR. Please describe how crane operation and any credible malfunction would be precluded from: • creating conditions that would cause an unanalyzed reactor accident; • causing an uncontrolled release of radioactive material beyond those analyzed in Chapter 13 of the MSRR PSAR; or • preventing safe shutdown of the reactor. For any design features that provide this protection, please describe the design bases and the applicable design criteria.	ACU clarified that the crane operation was addressed in Section 13.1.9, "Mishandling or Malfunction of Equipment," of the PSAR, which describes that the MSRR would be protected from any postulated dropped load by a concrete barrier above the systems pit prior to loading fuel salt in the reactor. ACU to include Section 9.8.2, "Cranes," in Rev. 2 of the PSAR (ML24219A258), which is intended to more completely address the reactor bay crane.

Resolution of Questions on Chapter 10, "Experimental Facilities and Utilization," Section 10.2 "Experimental Facilities"

Question Number	Question	Resolution
10.2-1	PSAR Section 10.2.1 references PDC 55 with respect to the salt sampling and measurement experimental systems. PDC 55, "Radionuclide interfacing lines penetrating containment," as provided in PSAR Section 3.1, states: Each line where a single failure could lead to a bypass of functional containment, such as those that interface directly with fuel or fission products and interface with systems outside the functional containment, shall be provided two adequately reliable containment isolation mechanisms, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, or small fuel sampling lines are acceptable on some other defined basis. These mechanisms shall be located to minimize the probability of failure due to environmental or external hazards. [The staff notes that ACU has discussed possible revisions to PDC 55 in response to other audit questions.] PSAR Section 10.2.1.1 states, with respect to the fuel salt sample extraction system, that "[r]emotely controlled isolation values are used to maintain appropriate functional containment." PSAR Section 4.3.11 states that the fuel salt sample extraction system, which penetrates fission product barriers, will utilize a system of interlocks to ensure the intent of PDCs is met. PSAR Section 6.2.2.7 states that the MSRR "enclosure relies upon isolation of penetrations to meet the functional containment design leak rate so that consequences are enveloped by the MHA. Isolation and monitoring of radionuclide bearing penetrations will be implemented by suitably redundant values, physical breaks, or component barriers in other portions of the facility." However, it is not clear to the staff which portions of the salt sampling and measurement experiment systems (i.e., only the fuel salt sample extraction system, or other systems) PDC 55 is meant to apply to. In addition, it is not fully clear to the staff how PDC 55 will be met for any of the applicable	ACU stated in information provided for audit that the entire fuel salt sample system will comply with PDC 55. This will be accomplished by using multiple barriers to prevent release of radioactive material, and integrity of barriers will be maintained via appropriate interlocks or other suitable techniques. ACU stated that additional detail on how PDC 55 will be met will be provided in an OL application. ACU separately clarified in information provided for audit under audit question 9.5-2 that fuel salt sample extraction system and salt sampling system both reference the same system.

	systems, including, for example, whether the fuel salt sample extraction system will be designed with "two adequately reliable containment isolation mechanisms" or other "containment isolation provisions" that "are acceptable on some other defined basis." Please discuss.	
10.2-2	PSAR Section 10.2.2 states that "[t]he gas sampling system is in compliance with [PDC] 55." Please clarify whether PDC 55 applicability includes the entire gas sampling and measurement experimental system discussed in PSAR Section 10.2.2 or only those portions which involve gas sampling or removal from the MSRR off-gas system. In addition, please discuss how PDC 55 is met for any applicable portions of the gas sampling and measurement experimental system.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to add clarification to PSAR Section 10.2.2.1, "Description," on how PDC 55 is satisfied for systems used to sample the off-gas from the reactor access vessel (RAV). In information provided for audit, ACU additionally stated that the means to meet PDC 55 for the gas sampling and measurement experimental system may be constraining volumes to limit available gas for release. In addition, ACU clarified in information provided for audit and during audit discussions that PDC 55 only applies to parts of the gas sampling system within the reactor enclosure. Specifically, it applies to the barriers and penetrations for the system in reactor enclosure and reactor vessel.

10.2-3	The revision of PSAR Table 3.4-1 provided for audit on June 2, 2023, in response to audit question Gen-7 states that in general, scientific surveillance systems are non-safety related, and experimental systems are safety-related. However, the boundaries between safety- and non-safety portions of experimental facilities discussed in PSAR Ch 10, as well as what portions of the experimental facilities are considered "experimental systems" versus "scientific surveillance systems" for the purposes of safety classification are not fully clear to the staff. The reasons for the classifications (e.g., are systems safety-related because they are fuel-salt-wetted, because of the functions they perform, or both?) are also not fully clear. Please discuss. (See also question 10.2-4.)	ACU stated in audit discussions that all penetrations through layers that are credited for functional containment, for experimental facilities or scientific surveillance layer (SSL) instruments, will be safety-related and subject to PDCs. ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to add clarifying text to PSAR Section 10.2.1.1, "Design Bases," indicating that the aspects of the MSRR experimental systems which pertain to maintaining fission product boundaries are safety-related. Under technical audit question Gen-7, ACU also modified PSAR table 3.4-1 to clarify that the functions of both the experimental systems and SSL are non-safety-related, but that interfaces (e.g., penetrations) of the experimental systems and SSL with safety-related systems are considered to be safety-related.

10.2-4	PSAR Section 10.2.5 states that the scientific surveillance facilities “are designed to gather information and data to support future licensing and development of molten salt reactors” and “form a layer of instrumentation, computer hardware and software, and supporting design features, called the scientific surveillance layer (SSL), which is capable of capturing the MSRR behavior during its operation.” Do the scientific surveillance facilities consist of dedicated instrumentation which is separate from the instrumentation of other experimental facilities discussed in PSAR Sections 10.2.1 through 10.2.3, or are these facilities based on data feeds from those systems? If the scientific surveillance facilities include dedicated instrumentation, should such instrumentation be subject to PDC(s)? Please also discuss the extent to which portion(s) of the scientific surveillance facilities may be safety-related, if applicable.	ACU stated in information provided for audit and during audit discussions that the SSL instrumentation and the functions it performs will be completely separate from safety-related equipment; it acquires non-essential data that is not needed for operation or safety, only for scientific purposes. However, SSL instruments will be subject to PDCs if they interface with safety-related components (e.g., if lines associated with SSL instrumentation penetrate a safety-related barrier). ACU modified the PSAR section 10.2.5.1, “Description,” by letter dated July 30, 2024 (ML24219A258), to clarify this information regarding the SSL, including that the SSL does not perform a safety-related function. ACU also made other PSAR revisions related to the SSL and its safety classification as discussed under audit question 10.2-3.
10.2-5	PSAR Section 10.2.2.3 discusses potential reactivity changes induced by the gas sampling and measurement experimental facility but refers to “salt sampling and measurement.” Please clarify and confirm why reactivity changes from the gas sampling and measurement experimental facility would be small, as appropriate.	ACU modified the PSAR section 10.2.2.3, “Reactivity,” by letter dated July 30, 2024 (ML24219A258), to correct “salt sampling and measurement,” to “gas sampling and measurement.” ACU also stated in information provided for audit that the reactivity changes are small because the gas sample volumes are negligible.

10.2-6	PSAR Section 10.2.1 states that the reactor access vessel has a “port and system for coupon testing of reactor materials.” PSAR Section 4.3.11 states that these coupons “will be periodically introduced and extracted from the fuel salt” and will include but not be limited to 316H stainless steel. PSAR Section 14.3.8 states that these “coupons testing the response to the fuel salt environment are the only experimental materials used with the reactor.” Please clarify whether ACU intends the use of coupons to be limited to evaluating the performance of actual MSRR materials over time, or whether ACU also plans to use the MSRR coupon system to evaluate “novel” materials, i.e., materials not otherwise found in the MSRR.	ACU clarified in information provided for audit that coupons will only be used to evaluate the materials utilized in the MSRR and not “novel” materials, based on its current plans.
10.2-7	It is not clear whether the information regarding salt sampling and measurement experimental systems ports in PSAR Section 10.2.1, Section 10.2.1.4, Table 10.2-2, and Figure 4.3-1 is fully consistent. For example, PSAR Section 10.2.1 describes a port “included for in-line measurement of salt parameters” and PSAR Section 10.2.1.4 and Table 10.2-2 mention a gamma measurement port that do not appear to be mentioned in the other referenced sections. Please clarify.	ACU modified the PSAR sections 4.3.11, “Description of the Reactor Access Vessel,” 10.2.1, “Salt Sampling and Measurement Experimental System,” and 10.2.1.4, “Instrumentation” (including table 10.2-2, “Anticipated Instruments for Salt Sampling and Measurement Experimental System”), by letter dated July 30, 2024 (ML24219A258), to clarify, update, and ensure consistency of information regarding the experimental systems and their associated ports/penetrations. ACU also clarified during audit discussions regarding the RAV salt height sensor(s) that although they are discussed in PSAR chapter 10 and provide data that may be used for experimental purposes, they also provide critical data for operation of the reactor protection system and

		are therefore safety-related. In addition, ACU clarified during audit discussions that radiation monitors that are part of the experimental systems are separate from those of the radiation monitoring system discussed in PSAR chapter 11, "Radiation Protection Program and Waste Management."

Resolution of Questions on Chapter 10, Section 10.3 "Experiment Review"

Question Number	Question	Resolution
10.3-1	PSAR Section 10.3 states that: "Management review of experiments are [sic] conducted prior to review by the MSRR Review and Audit Committee and includes representation from the Radiation Safety Office. The committee reviews and approves all experimental facilities, procedures for experiments, and assess [sic] each experiment within the guidance of 10 CFR 50.59. Review includes the description and purpose of experiment [sic], experimental facilities, experimental procedures, and a safety assessment of the experiments (described in Chapter 12)." a. PSAR Section 12.3.1 states that new experiments and substantive changes to experiments are reviewed by the MSRR Review and Audit Committee, and approved by the Level 2 (i.e., the MSRR Facility Director). Please clarify the apparent discrepancies and what the approval versus review functions of the committee and the Level 2 are. b. Please clarify what is meant by "[m]anagement review of experiments … includes representation from the Radiation Safety Office."	a. ACU clarified in information provided for audit that although the Facility Director approves experiments after their review by the MSRR Review and Audit Committee as discussed in PSAR chapter 12, the Facility Director is also part of the review process (and is part of the committee although will not be the chair). b. ACU clarified in information provided for audit that the Radiation Safety Officer is on the MSRR Review and

	c. It is not clear to the staff what portion of PSAR Chapter 12 is being referred to in PSAR Section 10.3; please clarify. d. The staff notes that 10 CFR 50.59, "Changes, tests, and experiments," is a regulation that applies to nuclear reactors during operation. Should 10 CFR 50.59 be referred to as a requirement rather than guidance?	Audit Committee and also has opportunity to provide input on experiments prior to review by the committee. c. ACU clarified in information provided for audit that the reference is to PSAR section 12.3.1, "Experiment Review and Approval." d. ACU modified the PSAR section 10.3, "Experiment Review," by letter dated July 30, 2024 (ML24219A258), to state that the MSRR Review and Audit Committee assesses experiments within the regulations of 10 CFR 50.59.

Resolution of Questions on Chapter 11, "Radiation Protection Program and Waste Management," Section 11.1, "Radiation Protection"

Question Number	Question	Resolution
11.1-1	The NRC staff notes that PSAR Section 11.1.1, "Radiation Sources," is missing information on the estimated radionuclides released for Noble Gases, Iodine, and any other anticipated radionuclide releases from the MSRR. The NRC staff notes that preliminary estimates on radionuclide releases informs the design for effluent treatment systems and shielding during the early design phases. Early source term estimations provided during the CP phase help to ensure that the design includes as low as reasonably achievable (ALARA) design features to reduce occupational and public exposures ALARA as described 10 CFR 20.1101(b) and	During the audit, ACU provided documentation on the ERR of the estimated effluent doses to the public from MSRR operation. In addition, ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to state that public doses from all effluents are expected to be less than 1 mrem.

	20.1101(d). The NRC staff requests that ACU provide estimates on noble gas generation, collection, decay, and subsequent release of fission product gases. The NRC staff also requests that this include release from the Off-gas system, and any other release from the building ventilation to the environment.	
11.1-2	PSAR Section 4.4.6 states Ar-41 generation is limited as described in Chapter 11. What is the anticipated generation rate of Ar-41? How does the facility design plan to control Ar-41 generation within the bio-shield areas? What are the anticipated release pathways for Ar-41 to be released to the environment?	ACU stated during audit discussions that Ar-41 is primarily generated by the activation of air within the reactor cell, which is then captured by the cover gas system and released through the exhaust stack. In addition, ACU stated that the evaluation of public dose due to operation of the MSRR accounts for Ar-41 release.
11.1-3	The NRC staff requests ACU provide information that establishes the basis for the PSAR's stated tritium generation rates of 1.5 Ci per megawatt day. a. What does the MSRR plan to do to control the tritium releases from the facility? b. How does tritium get released from the fuel and cooling salts to the environment?	a. ACU stated during audit discussions that tritium is generated from three sources: the fuel salt via fission and activation of Li, coolant salt via activation of Li, and in the reactor cell via activation of H. In addition, an analysis was performed to show the effect of releasing all of the proposed tritium. This resulted in an update to PSAR section 11.1.1 by letter dated July 30, 2024 (ML24219A258), to state that the release of all assumed tritium resulted in public doses of less than 1 mrem. b. ACU stated in audit discussions that tritium will likely diffuse

		through the reactor piping, particularly through the heat exchanger. Tritium that diffuses through the pipes into the reactor cell will be vented out of the exhaust stack. ACU stated that the evaluation of public dose due to the operation of the MSRR accounts for tritium release.
11.1-4	The NRC staff notes that corrosion control appears to be a significant part of the MSRR design. Does ACU have any estimates for the activation of corrosion products within the primary system? How will ACU handle sampling and measurement of corrosion products in the coolant?	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to state that there are no current estimates for the activation of corrosion products in the primary system, however estimated concentrations of Cr and Fe in the primary salt are below 100 ppm. The radiological hazard posed by the activation of this material is negligible compared to the fission products.
11.1-5	Does ACU have any initial effluent dose calculations for their facility? Do these estimates assume the release of other radionuclides in addition to tritium?	ACU stated during audit discussions that the initial effluent dose estimate due to all effluents is 0.492 mrem/year to the public. This estimation was performed assuming that the off-gas system is never operated. ACU stated that a calculation of anticipated releases from the off-gas management system will be provided in an MSRR OL application.

11.1-6	The NRC staff notes that PSAR Section 11.1.5, "Radiation Exposure Control and Dosimetry," seems to be missing information or statements for when information will be provided. Does the applicant plan on providing initial radiation zoning for their facility? A figure is provided in PSAR Section 4.4 for the areas within and around the biological shield, but are there any more areas and maps that can be provided for the initial design?	ACU stated during audit discussions that radiation zoning based on the field produced by the reactor and airborne radioactivity will be estimated and that estimate will provide the basis for initial zoning. Zoning will be finalized by measurements taken during start-up testing. In addition, ACU stated that no more maps have been created at this time, however, it is known that radiochemistry labs will require radiation/high radiation areas as appropriate.
11.1-7	The NRC staff notes that PSAR Section 11.1.5 does not address information about personnel badging or if this topic will be provided in the OL. The NRC staff requests that ACU provide additional information/clarification.	ACU stated during audit discussions that additional details will be provided in an MSRR OL application. In addition, ACU stated that as an initial estimate, personnel that are likely to receive more than 10% of the appropriate limit set in 10 CFR 20, subpart C, will be monitored.
11.1-8	The NRC staff notes that radiation monitoring system references can be tied to how systems will have specific monitors to address compliance with the regulations. For example, providing information that there will be radiological effluent monitoring on the plant stack, off-gas system, and area monitoring around the biological shield. The NRC staff requests that ACU provide discussion, at a high level, about the radiation monitoring that will be in place to support monitoring effluent releases and occupational doses. The NRC staff also requests that this include, at a high level, discussion on the radionuclides (tritium, noble gas, iodine, particulate, etc.) that are anticipated to be measured to ensure that the OL will discuss the specific monitoring needs of the facility, and how compliance with the regulations will be achieved using these monitors.	ACU stated during audit discussions that the following systems and areas will have radiation monitors: • Exhaust Stack • Off-gas management system • Research Bay • Reactor Cell and Cell outlet • Fuel handling system enclosure • Primary heat removal system

		enclosure • Helium gas management system • Radiochemistry Lab As part of this discussion ACU stated that more information will be provided in the OL application. The NRC staff will review the details of the radiation monitoring during the OL application.

Resolution of Questions on Chapter 11, Section 11.2, "Radioactive Waste Management"

Question Number	Question	Resolution
11.2-1	PSAR Section 11.2.2 states, "During normal operations, liquid radioactive wastes are packaged and disposed of using a licensed and qualified low-level radioactive waste disposal vendor. Solid radioactive waste at the MSRR facility is primarily generated by reactor operation, either as a byproduct of experiments, such as material coupons, or from maintenance, such as reactor structural components and tools. Additional radioactive waste is produced by laboratory activities, such as contaminated gloves or pipette tips. Solid radioactive waste is packaged to be stored temporarily onsite in a designated cell in the research bay. Appropriate disposal is organized with the licensing status of the material, its chemical form, and its radioactivity (or lack thereof) defined at the time of disposal. Solid radioactive wastes also include absorbing media such as off-gas charcoal, ion exchange resins, and air filters." a. Does ACU plan on providing expected waste generation rates for waste generated as a part of normal operations? Does ACU have any information related to the storage area that this waste will be stored prior to disposal?	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to include PSAR Section 11.2.4, "Estimated Quantities of Waste Generation," which provides an estimation for the amount of waste generated by the MSRR.

11.2-2	Does ACU's solid waste also include the generation of greater than Class C (GTCC) waste or High-Level Waste (HLW)? Does ACU plan to generate GTTC or HLW because of reactor operations? If GTTC or HLW is anticipated where does ACU plan on storing the waste?	ACU stated during audit discussions that the facility will not generate GTCC waste. PSAR section 1.7, "Compliance with the Nuclear Waste Policy Act of 1982," states that ACU "intends to enter into a contract with the [DOE] for required fuel cycle services."
11.2-3	In PSAR Section 11.2.2, as part of the waste pathways being generated, ACU states that ion exchange resin waste is produced because of operations. In looking for system information around how this ion exchange resin is generated, the NRC staff could not find any specific information for how or where ion exchange resin would be used in the PSAR. The NRC staff requests that ACU provide information on how the fuel salt coolant is maintained and how the PSAR Section 11.2.2 stated solid waste streams are generated at the MSRR.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to remove "ion exchange resin" as a solid waste form because it is not currently planned to be a component of the coolant salt maintenance.
11.2-4	Does ACU have any plans to reference additional guidance related to the development of an offsite dose calculation manual (ODCM), radiological environmental monitoring program (REMP), or process control program (PCP)?	ACU stated in audit discussions that these items are typically required for commercial reactors and not for research reactors. ACU stated further that similar information as relevant to the MSRR will be contained in MSRR procedures.

Resolution of Questions on Chapter 11, Section 11.3, "Respiratory Protection Program"

Question Number	Question	Resolution

11.3-1	In PSAR Section 11.3, ACU indicates that the facility is not subject to the requirements of 10 CFR Part 20, Subpart H. The NRC staff notes that 10 CFR Part 20, Subpart H not only includes requirements for respiratory protection but also requirements for process and engineering controls and other controls to control radiation exposure to airborne radioactive material and is applicable to all licensees. Fission products at the facility include noble gases and other radioactive material that could become airborne. In addition, Table 10.2-3 shows that a gas sample release in the Research Bay would exceed 1 DAC (which is the criteria for an airborne radioactivity area, as defined in 10 CFR 20.1003). While respirators may not need to be used, provided that adequate process, engineering controls, or other measures are implemented to control airborne radioactivity, as needed, the requirements of 10 CFR Part 20, Subpart H are applicable. The NRC staff requests that ACU clarify the statement in PSAR Section 11.3.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to state that airborne radioactivity and inhalation doses shall still be kept ALARA through the use of engineering and administrative controls. However, the use of respirators or other respiratory protection is not expected at the MSRR facility, therefore the development of a respiratory protection program is not required.

Resolution of Questions on Chapter 12, "Conduct of Operations," Section 12.7, "Emergency Planning"

Question Number	Question	Resolution
12.7-1	The NRC staff was not able to find adequate or sufficient information in the PSAR Chapter 12, Appendix 12A, emergency response organization information and descriptions and figures, to address the ANSI/ANS-15.16, Section 3.3, "Organization and Responsibilities," and NUREG-0849, Section 3.0, "Organization and Responsibilities," guidance of: I. The reactor's emergency organization, including augmentation of the reactor staff to provide assistance for coping with the emergency situation, recovery from the emergency, and maintaining emergency preparedness. II. The capability of the emergency organization to function around the clock for a protracted period of time following the initiation of	ACU stated during audit discussions that the emergency response organization will be expanded upon in an MSRR OL application. The additions will include a staffing plan for emergency response coverage and a block diagram that illustrates the interrelationship for the facility emergency organization to the total emergency response effort.

	emergencies that have or could have radiological consequences requiring around the clock emergency response. III. A block diagram that illustrates the interrelationship of the facility emergency organization to the total emergency response effort. Interfaces between reactor and other onsite emergency organization groups and offsite local support organizations and agencies should be specified. The NRC staff requests that ACU provide additional information to address these guidance items.	
12.7-2	PSAR Chapter 12, Appendix 12A, "ACU Research Reactor Facility Preliminary Emergency Plan," Section 12A.2.2, "Emergency Organization Structure," and Section 12A.2.2.1, "Emergency Director (ED)," does not discuss or describe why it is acceptable for a "non-qualified" senior person on the emergency plan contact list to respond and perform the emergency response duties, at the onset of an emergency, of an on-shift senior emergency plan response qualified individual. The NRC staff request that ACU provide additional information.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to clarify that the qualified staff will be onsite at all times during reactor operation. However, when the reactor is not operating and is secure, a qualified staff will be on-call to respond in the event of an emergency. ACU clarified that a non-qualified individual, such as emergency response personnel, may be onsite first at which point they will contact the qualified individual on-call.
12.7-3	The NRC staff was not able to identify within the ACU PSAR, Chapter 12, Appendix 12A, "ACU Research Reactor Facility Preliminary Emergency Plan," the line of succession for the listed emergency response personnel of Emergency Director and Radiation Safety Officer. The NRC staff requests that ACU provide this information.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to include Table 12A-2, "Succession Plan for Emergency Director and Radiation Safety Officer."

12.7-4	The NRC staff was not able to identify within the ACU PSAR, Chapter 12, Appendix 12A, "ACU Research Reactor Facility Preliminary Emergency Plan," the identification by title of the individual, with a line of succession, responsible for relating information about the emergency situation to the news media and the public. The NRC staff requests that ACU provide this information.	ACU stated during audit discussions that communication of emergency information to the media is a primary responsibility of the Emergency Director. As discussed in audit question 12.7-3, the Emergency Director line of succession will follow PSAR Table 12A-2.
12.7-5	Please provide the definitions of the emergency plan terms "dedicated replacement" (As stated in PSAR Section 12A.2.2.1, "Emergency Director") and "off-hours" (As stated in PSAR Section 12A.2.4, "Staffing").	ACU stated during audit discussions that "off-hours" is defined as when the reactor is secured and not operating. In addition, ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to remove the term "dedicated replacement" and as discussed in audit question 12.7-3, the Emergency Director line of succession will follow PSAR Table 12A-2.
12.7-6	The NRC staff request definitions and/or descriptions to define or describe the times when qualified emergency response individuals would not be present at the MSRR facility and would need to be called in during the onset of an emergency.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to provide description of when the facility would be staffed by qualified individuals and when individuals will need to be on call.
12.7-7	The NRC staff request clarity of the PSAR Chapter 12, Appendix 12A, "ACU Research Reactor Facility Preliminary Emergency Plan," Section 12A.1.2, "Definitions," definition of "site boundary." Does this term describe the ACU campus site boundary or the MSRR facility site boundary?	ACU stated in audit discussions that the site boundary refers to the MSRR facility boundary as depicted in PSAR Figure 12A-2, "MSRR Site Layout." ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to clarify the terminology.

Resolution of Questions on Chapter 12, Section 12.9, "Quality Assurance"

Question Number	Question	Resolution
12.9-1	The NRC staff notes that the PSAR does not reference the accepted version of the QAPD Topical Report. Does ACU intend to supplement their PSAR to reference the accepted version?	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to reference the accepted version of the QAPD Topical Report.

Resolution of Questions on Chapter 16, "Other License Considerations," Section 16.1, "Prior Use of Reactor Components"

Question Number	Question	Resolution
16.1-1	The guidance in ORNL/TM-2020/1478 (ML20219A771), Appendix A, Part 1, Section 16.1, "Prior Use of Reactor Components," states that "[f]uel provided by the Department of Energy (DOE) for a new facility … could come from DOE storage and have a history of prior use that must be considered." The guidance also states that prior use should be considered for fuel salt. PSAR Sections 4.2.1 and 16.1 do not appear to address potential prior use of fuel and/or salt. Please discuss whether the MSRR may utilize fuel and/or salt that were previously used in other reactors. If so, will ACU consider such prior use in its evaluation of the fuel salt that will be used in the MSRR?	ACU modified the PSAR section 16.1 by letter dated July 30, 2024 (ML24219A258), to add information about possible sources for MSRR fuel and salt, to clarify that purity specifications for fuel and salt will apply regardless of their source, and to state that an MSRR OL application will identify the sources of the fuel and salt and will describe any prior uses of the material.
16.1-2	PSAR Section 16.1 states that the MSRR will be integrated into a portion of the pre-existing ACU Science and Engineering Research Center (SERC) building and that the weight of the MSRR reactor system will be supported by the SERC research bay floor and systems pit; as such, these portions of the SERC will be SSCs for the MSRR. Please clarify if this description of the portions of the pre-existing SERC that will become safety-related SSCs remains accurate in light of information discussed and provided for audit in response to audit question Gen-7 and questions in the	ACU modified the PSAR section 16.1 by letter dated July 30, 2024 (ML24219A258), to clarify and make it consistent with information provided and discussed in audits and in PSAR chapter 3 (as revised in response to technical audit question Gen-7). Specifically, ACU

	PSAR Chapter 2 and 3 audit.	removed the PSAR section 16.1 statement that the weight of the MSRR reactor system would be supported by the research bay floor.
16.1-3	Please clarify whether ACU plans to use any SERC features which may become MSRR SSCs for other purposes prior to installation of the MSRR. If so, discuss whether and how ACU is considering such prior use in its analysis of the ability of those SSCs to perform their applicable safety function for the MSRR.	ACU modified the PSAR section 16.1 by letter dated July 30, 2024 (ML24219A258), to state that ACU does not plan to utilize any SERC safety-related SSCs before their incorporation into the MSRR.

Resolution of Additional Topics Beyond the Scope of the Specific Audit Questions

Topic	Question	Resolution
Technical Specifications (related primarily to PSAR chapter 14)	The staff provided ACU with other additional follow-up questions and feedback on ACU's preliminary subjects of technical specifications (TSs) during audit interactions. This included questions and feedback related to, for example: • Consistency between TS in PSAR chapter 14 and TS references in other portions of the PSAR. • Relevance of some TS to specific characteristics of the MSRR. • Clarification between (and terminology used to describe) items that could be designated as safety limits for the MSRR versus items that could be in other categories of TS (e.g., limiting conditions for operation) or would be MSRR parameters that would be monitored but would not actually be TS. • Clarification of the basis for items selected as preliminary safety limits. • Clarification of terminology used in preliminary subjects of TS. • Clarification on the scope/applicability of certain TS including what they would require, what the relevant variable(s)/parameter(s) are, and/or what SSCs they cover. • Ensuring appropriate clarity and specificity of preliminary subjects of TS.	ACU revised the PSAR including chapter 14 and sections 4.2.1.7, 7.4.3, 9.2.3, 9.6.4, and 9.6.5 by letter dated July 30, 2024 (ML24219A258), to incorporate changes in response to NRC questions and feedback in audit discussions.

6.0 EXIT BRIEFING

The staff conducted an audit closeout meeting on August 20, 2024. At the exit briefing the staff reiterated the purpose of the audit and summarized the audit activities. Additionally, the staff stated that it did not identify areas where further additional information would be necessary to support the review.

There were no deviations from the audit plan.

7.0 ADDITIONAL INFORMATION RESULTING FROM AUDIT

No RAIs were generated as a result of this audit. However, ACU updated the MSRR PSAR on its own initiative as noted above to address several items discussed during the audit.

8.0 OPEN ITEMS AND PROPOSED CLOSURE PATHS

Not applicable.

SUMMARY REPORT FOR THE REGULATORY AUDIT OF ABILENE CHRISTIAN UNIVERSTY MOLTEN SALT RESEARCH REACTOR CONSTRUCTION PERMIT PRELIMINARY SAFETY ANALYSIS REPORT SECTION 9.2 (HANDLING AND STORAGE OF REACTOR FUEL) AND CHAPTER 13 (ACCIDENT ANALYSES)

March 2023 – August 2024

1.0 BACKGROUND AND PURPOSE

By letter dated August 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22227A202), as supplemented by letter dated October 14, 2022 (ML22293B816), Abilene Christian University (ACU) submitted to the U.S. Nuclear Regulatory Commission (NRC), an application for a construction permit (CP) for a molten salt research reactor (MSRR), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Section 104c of the Atomic Energy Act of 1954, as amended. The application included a preliminary safety analysis report (PSAR) (ML22227A203). PSAR section 9.2, "Handling and Storage of Reactor Fuel," describes the fuel handling system (FHS) which is designed to ensure fuel is enclosed in a manner such that radionuclides are functionally contained during handling and manipulation of reactor fuel. PSAR chapter 13, "Accident Analysis," provides information and analyses considering the potential consequences of a diverse array of adverse events and accidents, as well as the capability of the facility to accommodate such disturbances.

This audit enabled the NRC staff (the staff) to gain a better understanding of PSAR section 9.2, and chapter 13 through review and discussion of underlying supporting documentation. Enhanced understanding and communications supported effective and efficient resolution of technical issues, including though development of information needs where needed.

2.0 AUDIT REGULATORY BASES

The basis for the audit is the regulations in 10 CFR Part 50, Sections 50.34(a), "Preliminary safety analysis report," and 50.35, "Issuance of construction permits."

3.0 AUDIT OBJECTIVES

The primary objective of the audit was to enable a more effective and efficient review of PSAR section 9.2, and chapter 13 through the staff's review and discussion of supporting documentation with ACU. Gaining access to underlying documentation and engaging in audit discussions about handling and storage of reactor fuel and accident analysis facilitated the staff's understanding of the MSRR CP application and aided in assessing the safety of the proposed research reactor. The audit improved communication and provided detailed information for the staff.

4.0 SCOPE OF THE AUDIT AND AUDIT ACTIVITIES

The audit was conducted from March 2023 to August 2024, via teleconference and the electronic reading room (Certrec), and also during a site visit conducted on May 17-18, 2023. The staff conducted the audit in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction NRR-LIC-111, Revision 1 "Regulatory Audits" (ML19226A274).

Members of the audit team, listed below, were selected based on their detailed knowledge of the audit subject matter. Audit team members included:

1. Boyce Travis, NRR (Senior Nuclear Engineer)
2. Christopher (Ben) Adams, NRR (General Engineer)
3. Sean Meighan, NRR (Reactor Scientist)
4. Ryann Bass, NRR (Reactor Systems Engineer)
5. Kyle Song, NRR (Electronics Engineer)
6. Chris Van Wert, NRR (Senior Technical Advisor for Reactor Fuels)
7. Alexander Chereskin, NRR (Materials Engineer)
8. Steve Jones, NRR (Senior Safety and Plant System Engineer)
9. Zachary Gran, NRR (Reactor Scientist)
10. Richard Rivera, NRR (Project Manager)
11. Zackary Stone, NRR (Project Manager)
12. Edward Helvenston, NRR (Project Manager)
13. Mohsin Ghazali, NRR (Project Manager)
14. Brian Bettes, NRR (Project Manager)
15. Michael Balazik, NRR (Project Manager)
16. Michelle Hayes, NRR (Branch Chief, Technical)
17. Gregory Oberson, NRR (Branch Chief, Technical)
18. Michael Wentzel, NRR (Branch Chief, Licensing)
19. Stephen Philpott, NRR (Branch Chief, Licensing)

Prior to the audit, the audit team reviewed PSAR section 9.2, and chapter 13 and defined in the audit plan (ML23065A056) the general range of topics to be addressed and focused on during the audit. The following table documents the dates that the staff transmitted audit questions and when audit meetings were held:

Audit Questions (ADAMS Accession No.)	Audit Meetings
March 16, 2023 (ML23076A015)	May 17, and 18, 2023
May 52, 2023 (ML23123A046)	July 7, 10, and 13, 2023
	August 1, 2023
	October 3, and 12, 2023
	November 2, and 16, 2023
	January 25, 2024
	February 13, 16, 20, and 22, 2024
	March 7, 19, and 21, 2024
	August 20, 2024 (exit meeting)

The staff reviewed the following ACU documents via the electronic reading room (ERR):

- Written responses that ACU prepared for certain questions to address the questions and/or facilitate discussion with the staff.
- PSAR pages indicating changes proposed by ACU in response to various audit questions.
- ACU’s analysis of the maximum hypothetical accident (MHA) at the MSRR, “MHA Calculation Methodology,” dated May 6, 2022.
- ACU’s spill prevention, control and countermeasure (SPCC) plan, “Final SPCC Plan ACU R1,” dated July 28, 2022.

- “Evaluation of SCALE, SERPENT, and MCNP for Molten Salt Reactor Applications using the MSRE Benchmark,” dated April 3, 2023.
- “RELAP-3D model of Loss of Off-Site Power v2,” provided April 4, 2023.
- “RELAP5-3D model of MHA v3,” provided April 6, 2023.
- “RELAP5-3D model iota design Description,” provided April 13, 2023.
- “Noble Gas Fission Product Generation Rate and Air Activation,” dated June 6, 2023.
- “MATLAB Readable MHA Inventory Calcs,” provided June 21, 2023.
- “MHA Dose Calculations Workbook,” provided June 21, 2023.
- ACU’s process flow diagram revision C, “PFD SK-0001_REV_C,” provided July 12, 2023.
- “RELAP5-3D physics parameters Rev 1,” provided August 24, 2023.
- SCALE input for molten salt reactor experiment (MSRE) base model with benchmark geometry, “MSRE Shift,” provided September 15, 2023.
- “Use of ASME Section VIII for Fuel Salt Purification and Storage Vessel,” provided November 11, 2023.

The staff previously issued an interim audit report related to this audit by letter dated June 22, 2023 (ML23157A064).

5.0 SUMMARY OF AUDIT OUTCOME

The staff’s audit focused on the review of supporting documents associated with the questions provided to ACU on March 16, 2023 (ML23076A015) and May 5, 2023 (ML23123A046). The staff reviewed information through the ACU Certrec portal and held discussions with ACU staff to understand the supporting information. In many cases, ACU updated the PSAR to resolve items discussed in the audit. The tables below replicate specific audit questions transmitted in emails to ACU as listed above and summarize the resolution of the audit questions.

Resolution of Questions on Section 9.2, “Handling and Storage of Reactor Fuel”

Question Number	Question	Resolution
9.2-1	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.	PSAR section 9.2, states, in part, that the 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. ACU stated in information provided for audit that fuel salt handled outside the core would be significantly less than a keff of 0.9. In addition, ACU provided a preliminary criticality analysis of the reactor drain tank using conservative assumptions (e.g., entire reactor system offloaded, special nuclear material (SNM) separates from salt) for the staff’s review. ACU revised the PSAR by letter dated July 30, 2024 (ML24219A258) to clarify in PSAR section 9.2.3, “Operational Analyses and Safety Function,” that the lack of a neutron moderator during fuel storage and

		movements keeps the fuel subcritical (keff of approximately 0.6). The staff will evaluate subcriticality for the FHS during its review of an operating license (OL) application to ensure a safe margin of subcriticality is maintained for the storage of fuel.
9.2-2	The section does not describe any technical specification [TS] 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.	ACU revised the PSAR by letter dated July 30, 2024 (ML24219A258) to state in PSAR section 9.2.3 that TS will address pressures and temperatures of lines and vessels in the FHS, rate of temperature change when heating up, number of thermal cycles, number of sparging cycles, total time at elevated temperatures, pressure and leak rate of the FHS enclosure, radiation monitoring, pressures and mass flow rates of the gases flowing through the FHS vessels, maintenance procedure, and inspection/testing procedures.

9.2-3	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.	ACU provided additional information for audit on fuel storage durations, fuel storage enclosure functionality, and fuel salt subcriticality. ACU revised PSAR sections 9.2.2, “System Description,” 9.2.3, and 9.3.3, “System Description,” by letter dated July 30, 2024 (ML24219A258) to address heat removal of stored fuel, capability of the fuel storage enclosure, and fuel salt purification/storage tank.
9.2-4	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).	ACU revised PSAR section 9.2.3 by letter dated July 30, 2024 (ML24219A258) to state that fuel salt purification and storage operations occur inside the fuel storage enclosure, a safety-related, leak tight, pressure and fission product boundary. Further information was added to PSAR section 9.2.3 regarding fuel storage enclosure design leak rates. Finally, ACU stated in PSAR section 9.2.3 that design pressures, temperatures, and leak rates for the fuel storage

		enclosure will be determined to ensure that the radiological consequences of a fuel salt leak are bounded by the MHA. For consistency with principal design criteria (PDC) 61, "Fuel storage and handling and radioactivity control," ACU revised PSAR section 9.2.1, "Design Basis," by letter dated July 30, 2024 (ML24219A258) to state that containment of fuel salt during handling and storage is provided by the safety-related vessels in the system, and the fuel storage enclosure.
9.2-5	ACU MSRR PSAR, Revision 1, Section 9.2.3, states "Welding between SS316H and Alloy 201 will make use of a suitable material as defined by the appropriate code." a. It is not clear to the staff what is meant by the term "suitable material." Describe the attributes or properties of the material that would make it "suitable." The staff notes that this could include, for instance, resistance to stress-rupture, creep and creep-fatigue, and environmental degradation. b. It is not clear to the staff what is meant by the term "appropriate code" in the context of this sentence. Describe the judgment or criteria used to determine that the code is "appropriate," or who makes that determination. The staff presumes that, based on typical engineering practice, necessary conditions for the weld material (i.e., to maintain the attributes that make it "suitable")	The staff determined that this question 9.2-5 is resolved based on ACU's response, dated March 28, 2024 (ML24088A324), to request for additional information (RAI) 1. In addition. ACU modified PSAR section 9.2.3 by letter dated July 30, 2024 (ML24219A258) to update the information provided on welding.

	would be identified, then a code would be selected that conforms to the establishment or maintenance of those attributes. It is not clear to the staff what is meant by the term “as defined by” in the context of this sentence. The staff notes that this could be understood as the specification of a particular material. Alternatively, this could mean the specification of attributes or properties that the fabricator would then apply to the material selection. It is not clear to staff how ACU has concluded that an “appropriate code” will necessarily “define” a “suitable material,” given that the presumptions underlying this claim do not appear to be discussed. Please explain.	
Additional Topic: Corrosion Allowance	The staff discussed corrosion allowance for Ni-201 with ACU for use in the FHS salt-purification vessel and associated components.	A corrosion allowance was quantified in a proprietary enclosure included with ACU’s RAI 1 response, dated March 28, 2024 (ML24088A324), for Ni-201 components. The staff did not review the adequacy of this corrosion allowance, however, as it was determined to be unnecessary to sufficiently resolve the RAI. ACU confirmed in its response to the staff’s RCI, items 4.3-5 and PSAR-1, dated June 12, 2024 (ML24164A236) that effects on corrosion of Ni-201 will be evaluated and incorporated into the design of the FHS or shown to be mitigated and that supporting information for determining this corrosion allowance will be provided in an OL application. The staff will

		evaluate the final design of the FHS during an OL application review to ensure the design bases are met.
Additional Topic: Material Selection	The staff discussed the selection of Ni-201 and ERNi-1 filler metal with ACU for use in the FHS salt-purification vessel and associated components.	PSAR section 9.2.3 states that Ni-201 and ERNi-1 filler metal will be used for the FHS salt-purification vessel and associated components. ACU provided supporting data and a preliminary analysis in support of Ni-201 material selection in a proprietary enclosure included with ACU's RAI 1 response, dated March 28, 2024 (ML24088A324). No supporting information was provided for the selection of ERNi-1 filler metal. The staff did not review the accuracy nor adequacy of this data or analysis as it was not requested by ACU, nor necessary for the issuance of a CP. The staff will evaluate the adequacy of Ni-201 and ERNi-1 filler metal for use in the FHS during an OL application review, one important aspect of which is uncertainty with the elevated-temperature structural integrity, especially at weldments. In addition, the staff notes that Ni-201 and

		ERNi-1 filler material may contain cobalt, and therefore the staff will evaluate the radiation protection program to ensure that cobalt activation, transport and deposition is accounted for.
Additional Topic: Code Selection	The staff discussed the use of ASME Codes for constructing the FHS salt-purification vessel and associated components with ACU.	PSAR section 9.2.2 states that the 2021 edition of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” and 2, “Alternative Rules” and ASME B31.3-2020, “Process Piping,” will be used for constructing Ni-201 components. The staff notes that no codes have been endorsed by the NRC for constructing Ni-201 components. As such, ACU is obligated to demonstrate the use of these codes conforms with Principal Design Criteria (PDCs) that apply to safety-related systems, structures, and components (SSCs). ACU provided additional information supporting the selection of these codes in its response dated March 28, 2024 (ML24088A324), to the staff’s

		RAI 1. The staff determined that the response, in itself, lacked sufficient information to provide the necessary assurance for use of the codes, including with respect to the following issues: (1) ACU did not provide data to demonstrate the adequacy of the allowable stress values in the ASME BPVC for use at elevated temperatures and that the base and filler metal are compatible for elevated temperature applications. (2) ACU stated in its response to the staff's RAI 1 that a fatigue screening analysis was performed in accordance with the Paragraph 5.5.2.3 of ASME BPVC Section VIII, Division 2, which determined fatigue is not a degradation mechanism for the low-cycle, thermally consistent vessel. The staff found the

		screening analysis to be inadequate because Section VIII, Division 2, does not address elevated temperature fatigue. The design temperature for the Ni-201 components is governed by time-dependent properties. Therefore, as stated in Paragraph 5.1.1.3 in the 2021 edition of ASME BPVC Section VIII, Division 2, the only applicable fatigue screening criteria is comparative experience which was not provided by ACU. (3) PSAR section 9.2.2 states that "[t]he fuel salt purification vessel will experience load cycles and time at temperature that are a small fraction of the allowable values in Section VIII, Division 1." The staff was unable to verify this statement because the design life is not defined in Section VIII, Division 1 and the

		allowable stresses used in Section VIII, Division 1 are not a function of time. After further discussion with ACU, staff determined that, notwithstanding these issues, some of which NRC staff and ACU did not agree upon, their resolution was not necessary for the issuance of a CP. ACU is taking additional steps to assure the structural integrity of the subject SSCs, such that the staff's evaluation of conformance with the PDCs does not rely solely on a determination about the adequacy of the design code. This includes, for instance, the use of a surrogate vessel that can be regularly monitored and inspected, and which would be a leading indicator of any deterioration in the condition of the SSCs designed using the indicated codes, as discussed in SE section 9.2.3.2.1, "Leading Secondary Cooling System." Nevertheless, the staff will review the application of ASME BPVC for constructing the FHS salt-purification vessel and

		associated components during the review of an OL application.
Additional Topic: Ni-201 associated piping system and its supports	The staff discussed the use of ASME Codes for constructing the associated piping system including its supports for the FHS salt-purification vessel and associated components with ACU.	ACU stated in its, “MSRR Codes and Service Conditions,” submitted by letter dated July 30, 2024 (ML24219A258) that the “V-1002 and associated piping, piping supports, components, and flanges,” will be constructed of Ni-201. Additionally, PSAR section 9.2.2 states that “[t]he associated piping system, including its supports, will be designed to B31.3-2020, which is supplemented by Section III, Division 5, 2017 Edition at high temperatures.” Ni-201 is not qualified in ASME BPVC Section III, Division 5, “High Temperature Reactors,” for Class A nor Class B construction. ACU clarified through audit discussions with the staff that the Ni-201 piping would be supported by 316 stainless steel supports. The staff will confirm information related to the material to be used for the piping supports and how the ASME code will be applied during its review of an OL application.

Resolution of Questions on Chapter 13, "Accident Analysis," Section 13.1, "Accident-Initiating Events and Scenarios"

Question Number	Question	Resolution
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?)	ACU provided calculations for the staff's review during the audit. ACU revised PSAR chapter 13 by letter dated July 30, 2024 (ML24219A258) to include table 13.1-3, "Initial Nuclide Inventories Assumed for the MHA," which provides the assumed radionuclide inventory for the MHA.
Question on the 13.1-2	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.	ACU revised the PSAR chapter 13 by letter dated July 30, 2024 (ML24219A258) to include table 13.1-3, which lists the tritium inventory as a source term for the MHA.
13.1-3	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,	ACU provided additional information during the audit on the source term, selection of noble gases, basis for leakage rates for the reactor enclosure, and material interactions. ACU revised the PSAR section 13.1.1, "Maximum Hypothetical Accident," by letter dated July 30, 2024 (ML24219A258) to add information on the source term

	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)?	and basis for the leak rates from the reactor enclosure. In addition, ACU clarified in PSAR section 13.1.1 that there is no assumed holdup in the systems pit and that the radionuclides released from the reactor cell immediately escape to the atmosphere for the MHA.
13.1-4	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.	ACU revised PSAR section 9.2.3 by letter dated July 30, 2024 (ML24219A258) to include information on the design basis accident of the fuel storage enclosure which is a rupture of the fuel salt purification and storage vessel.
13.1-5	Fission product models are described in Chapter 13, but [reactor thermal management system] 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.	ACU provided analyses during the audit for the staff to review regarding the capability of the reactor thermal management system (RTMS) to remove heat. ACU stated that in the case of the MHA (the entire fuel salt inventory relocates to the RTMS) the reactor enclosure is maintained at an acceptable temperature by passively removing heat through other reactor components and ultimately to the systems pit. ACU revised the PSAR section 13.1.10, "Loss of Normal Electrical Power," by letter dated July 30, 2024 (ML24219A258) to clarify

		that the modeling using RELAP5-3D was used to demonstrate RTMS heat removal capability.
13.1-6	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).	ACU stated during the audit that all calculations were performed using RELAP5-3D. ACU provided the staff access to the RELAP5-3D model and results during the audit. ACU revised PSAR table 13.2-1, “Summary of Accident Scenarios Examined,” by letter dated July 30, 2024 (ML24219A258) to clarify safety consequences of the accidents analyzed in chapter 13.
13.1-7	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	No response needed.

	be precluded as an external event).	
13.1-8	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)?	ACU stated during audit discussions that burnups would be sufficiently low such that concentrations of fission products would not challenge solubility limits and cause an increase in vapor pressure via positive deviations from ideal behavior. In addition, PSAR section 14.3.1 states that fuel salt chemistry is monitored by sampling the fuel salt and that the composition attributes to be monitored and the method will be submitted in an OL application. During the review of an OL application, the staff will review predicted fission product concentrations along with administrative controls (e.g., TS limits and normal chemistry control) to ensure assumptions of a dilute solution are maintained.
13.1-9	Is tritium release from graphite considered in accident analyses?	ACU updated PSAR section 13.1.1 by letter dated July 30, 2024 (ML24219A258) to state that the MHA assumes no holdup of tritium in graphite such that all tritium generated is treated as being released.

13.1-10	Is oxidation of SSCs considered during postulated accidents (e.g. air or water ingress)? Is oxidation and precipitation of uranium considered during postulated accidents?	ACU updated PSAR sections 6.2.2, “Containment,” 6.2.4, “Reactor Thermal Management System,” and 9.2.3 by letter dated July 30, 2024 (ML24219A258) to indicate that the reactor enclosure, the RTMS, and the fuel storage enclosure are inert with nitrogen gas. The staff notes that the use of an inert gas helps prevent oxidation.
13.1-11	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.	ACU updated PSAR section 9.2.3 by letter dated July 30, 2024 (ML24219A258) to state that the fuel salt remains deeply subcritical even if all UF_4 precipitates out of the fuel salt and is collected at the bottom of the tank. In addition, ACU provided criticality analyses for both the drain tank and fuel salt purification tank, which the staff reviewed during the audit.
13.1-12	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?	During audit discussions, ACU stated that their calculations indicate that radiolytic degradation of the salt is not a concern because the temperature of the fuel salt would remain above the region in which radiolytic degradation would occur due to the RTMS insulation. The staff will verify

		this during the review of an OL application.
13.1-13	(Follow-up to audit question 13.1-10) The staff reviewed the information ACU provided for audit on 4/13/2023 in response to audit question 13.1-10, and notes that the testing used to provide ASME Code mechanical properties (referenced in the information) is done in air and does not include any environmental effects on the material (e.g., corrosion in Flibe). ASME BPVC Section III, Division 5, Article HHA-1130, "Limits of These Rules," states that rules in Section III Division 5 "...do not cover deterioration that may occur in service as a result of radiation effects, corrosion...," but that "[t]hese effects shall be taken into account with a view to realize the design or the specified life of the components and supports." Article HBB-1110(g) states that Section III, Division 5 does not "provide methods to evaluate deterioration that may occur in service as a result of corrosion, mass transfer phenomena, radiation effects, or other material instabilities." PSAR Section 13.1.2 states that a postulated accident is loss of fuel salt from piping and components of the reactor system. Another postulated accident identified in PSAR Section 13.1.2 is a rupture of a primary heat exchanger tube. If one of these pipes or components fail the staff notes that it could introduce air into the fuel salt boundary as well as the RTMS and reactor enclosure. As stated in ACU PSAR Section 4.2.1.6, air could create a corrosive environment for the fuel salt in the RTMS and/or reactor enclosure. While corrosion could be a concern affecting long-term operation and maintenance of the MSRR, it is not clear to the staff whether potential rapid corrosion related to a postulated event could also affect the assumptions in the analyses of MSRR postulated events. Please discuss the following to allow the NRC staff to evaluate whether air ingress during a postulated accident could cause rapid corrosion of salt wetted components: a. Clarify whether the fuel salt will be relocated to the RTMS or the reactor enclosure during a postulated accident.	a. Through audit discussions and submission of the process flow diagram by letter dated on March 28, 2024, (ML24094A332), staff developed an understanding of where fuel salt relocates depending on the various design basis accidents. b. ACU updated PSAR sections 6.2.2 and 6.2.4 by letter dated July 30, 2024 (ML24219A258) to clarify the RTMS and reactor enclosure environment is inert nitrogen. c. During audit discussions, ACU clarified that the coolant salt and heat management enclosure contains air. d. During audit discussion, ACU clarified that the GMS does not supply nitrogen to the RTMS and reactor enclosure. Nitrogen is separately supplied to the reactor enclosure and the supply of nitrogen is not continuous. The reactor enclosure pressure is maintained by operation of a

	b. Chapter 6 of the PSAR appears to describe an air environment in the reactor enclosure and RTMS. However, the information provided for audit on 4/13/23 in response to audit question 13.1-10 appears to describe a different environment. Clarify what the environment will be in both the RTMS and the reactor enclosure. c. What is the environment of the coolant salt and heat management enclosure? d. PSAR Section 9.6 appears to indicate that pressure equalization is the only safety-related function for the gas management system (GMS). Clarify whether the function to supply inert gas is safety-related and whether it can supply helium to the RTMS/reactor enclosure during an accident or if it gets isolated. e. Based on allowable leak rates for the reactor enclosure, the potential for air ingress prior to isolation (e.g., the staff notes that air may enter a broken pipe prior to all penetrations isolating), and potential air ingress through a broken heat exchanger tube during postulated accidents, discuss whether and how data will bound the effects of air leaks on corrosion rates of the RTMS and/or the reactor enclosure (depending on where the fuel salt spills). f. Confirm there is no potential pathway for bulk water ingress into the fuel salt boundary.	vacuum pump. ACU revised PSAR section 6.2.2 by letter dated July 30, 2024 (ML24219A258) to describe the nitrogen gas supply to the reactor enclosure. e. The staff determined that question 13.1-13 c is resolved by ACU's response, dated April 30, 2024 (ML24121A272), to RAI 2, including information on ACU's Degradation Management Program (DMP) for the MSRR. f. This was confirmed by ACU in their written response to the audit question provided to the staff via the ERR.

6.0 <u>EXIT BRIEFING</u>

The staff conducted an audit closeout meeting on August 20, 2024. At the exit briefing the staff reiterated the purpose of the audit and summarized the audit activities. Additionally, the staff stated that they did not identify areas where further additional information would be necessary to support the review.

There were no deviations from the audit plan.

7.0 <u>ADDITIONAL INFORMATION RESULTING FROM AUDIT</u>

No RAIs were generated as a result of this audit. However, ACU updated the PSAR on its own initiative as noted above to address several items discussed during the audit.

8.0 <u>OPEN ITEMS AND PROPOSED CLOSURE PATHS</u>

Not applicable.

SUMMARY REPORT FOR THE REGULATORY AUDIT OF ABILENE CHRISTIAN UNIVERSTY MOLTEN SALT RESEARCH REACTOR CONSTRUCTION PERMIT PRELIMINARY SAFETY ANALYSIS REPORT TECHNICAL TOPICS

January 2023 – August 2024

1.0 BACKGROUND AND PURPOSE

By letter dated August 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22227A202), as supplemented by letter dated October 14, 2022 (ML22293B816), Abilene Christian University (ACU) submitted to the U.S. Nuclear Regulatory Commission (NRC), an application for a construction permit (CP) for a molten salt research reactor (MSRR), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Section 104c of the Atomic Energy Act of 1954, as amended.

This audit enabled the NRC staff (the staff) to gain a better understanding of several cross-cutting technical topics related to ACU's preliminary safety analysis report (PSAR) through review and discussion of underlying supporting documentation. Enhanced understanding and communications supported effective and efficient resolution of technical issues, including through development of information needs where needed, and also informed questions and discussions in other chapter-specific audits conducted as part of this review.

2.0 AUDIT REGULATORY BASES

The basis for the audit is the regulations in 10 CFR Part 50, Sections 50.34(a), "Preliminary safety analysis report," and 50.35, "Issuance of construction permits."

3.0 AUDIT OBJECTIVES

The primary objective of the audit was for the NRC staff to seek clarification, gain understanding and verify information regarding technical focus topics (relevant to the NRC staff's safety review of the CP application) provided to ACU in the CP application acceptance letter, dated November 18, 2022 (ML22313A097), as well as other cross-cutting topics. Gaining access to underlying documentation and engaging in audit discussions about the MSRR design facilitated the staff's understanding of the CP application and aided in assessing the safety of the proposed research reactor. The audit improved communication and provided detailed information for the staff.

4.0 SCOPE OF THE AUDIT AND AUDIT ACTIVITIES

The audit was conducted from January 2023 to August 2024, via teleconference and the electronic reading room (Certrec), and also during a site visit conducted May 17-18, 2023. The staff conducted the audit in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction NRR-LIC-111, Revision 1, "Regulatory Audits" (ML19226A274).

Members of the audit team, listed below, were selected based on their detailed knowledge of the audit subject matter. Audit team members included:

1. Boyce Travis, NRR (Senior Nuclear Engineer)
2. Chris Van Wert, NRR (Senior Technical Advisor for Reactor Fuels)
3. Alexander Chereskin, NRR (Materials Engineer)

4. Ryann Bass, NRR (Reactor Systems Engineer)
5. Michael Balazik, NRR (Project Manager)
6. Brian Bettes, NRR (Project Manager)
7. Mohsin Ghazali, NRR (Project Manager)
8. Richard Rivera, NRR (Project Manager)
9. Zackary Stone, NRR (Project Manager)
10. Edward Helvenston, NRR (Project Manager)
11. Gregory Oberson, NRR (Branch Chief, Technical)
12. Michelle Hayes, NRR (Branch Chief, Technical)
13. Michael Wentzel, NRR (Branch Chief, Licensing)
14. Stephen Philpott, NRR (Branch Chief, Licensing)

Prior to the audit, the audit team reviewed the PSAR and defined the general range of topics in the audit plan dated January 13, 2023 (ML23013A089), to be addressed and focused on during the audit. The following table documents dates that the staff transmitted audit questions and when audit meetings were held:

Audit Questions (ADAMS Accession No.)	Audit Meetings
February 6, 2023 (ML23038A009)	January 20, 2023 (entrance meeting)
February 21, 2023 (ML23052A067)	February 9, 2023
	March 9, 2023
	April 20, 2023
	May 4, 2023
	May 17-18, 2023
	June 8, 2023
	July 13, 2023
	March 19, 2024
	April 3, 2024
	May 8, 14, and 21, 2024
	August 20, 2024 (exit meeting)

The staff reviewed the following documents via the electronic reading room:

- Written responses that ACU prepared for certain questions to address the questions and/or facilitate discussion with NRC staff
- PSAR pages indicating changes proposed by ACU in response to various audit questions
- Document “Audit Request Response,” provided January 27, 2023, discussing ACU’s initial plans related to the five technical topic areas as outlined in the CP application acceptance letter dated November 18, 2022.

The NRC staff previously issued an interim audit report related to this audit by letter dated June 22, 2023 (ML23157A064).

5.0 SUMMARY OF AUDIT OUTCOME

The staff’s audit focused on the review of supporting documents associated with the questions provided to ACU during the audit. The staff reviewed information through the ACU Certrec portal and held discussions with ACU staff to understand and resolve questions. In many cases, ACU updated the PSAR to resolve items discussed in the audit. The table below replicates the specific audit questions transmitted in emails to ACU as listed above, and summarizes the resolution of the audit questions. The table also lists and summarizes the resolution of an additional topic covered as part of this audit beyond the scope of the specific audit questions.

Question Number	Question	Resolution
Gen-1	It is not clear to the NRC staff on how some of the material ACU is using is qualified to the specifications needed for the MSRR, for example, per the ASME code. Are R&D programs necessary, for example, with respect to qualifications of these materials, or with respect to other additional novel aspects of the MSRR design?	During audit discussions, ACU stated that although it is conducting research and development (R&D) related to the MSRR, there is no currently ongoing or planned R&D related to confirming the adequacy of the design of the MSRR structures, systems, and components (SSCs). ACU stated that this is because the MSRR will be over-designed to account for uncertainties. ACU modified PSAR section 1.2 by letter dated July 30, 2024 (ML24219A258), to state that ACU does not identify a need for R&D programs as described in 10 CFR 50.34(a)(8), and that no MSRR SSCs require R&D to confirm the adequacy of the design.
Gen-2	ACU provided the document, "Initial Audit Response to Technical Topic areas", in the electronic reading room that provided information on the proposed graphite to be used. However, the NRC staff notes that this response does not appear to describe topics such as whether properties for certain commercially available grades of graphite will bound the ACU qualification envelope (temperature, fluence, and oxidation) and be consistent with ASME Code Section III Division 5 requirements for qualifying or designing graphite components. Additionally, there is no information that describes how salt infiltration will be minimized, whether graphite components will operate past turnaround or crossover, or how property variations will be assessed. Please clarify whether ACU intends to meet ASME	During audit discussions and in information provided for audit, ACU stated that it has not made a final choice on a grade of graphite for the MSRR, but provided preliminary information related to graphite and noted that additional details related to the graphite would be provided in an operating license (OL) application. ACU also noted that the MSRR graphite reflector would be non-safety-related. ACU modified PSAR table 3.4-1 by letter dated July 30, 2024 (ML24219A258), to clarify that the graphite core is non-safety-related and cannot impair a safety function, and that this will be demonstrated in subsequent analysis in an OL application. Based on the non-safety-related classification of the graphite, as well as the resolutions of other questions related to graphite in the separate PSAR chapter 4, chapter 6, and section 9.6 audit (see separate audit

	Code requirements (as endorsed by NRC RG 1.87, Revision 2) for graphite and as appropriate, describe how data for the chosen grade of graphite will meet ASME Code requirements and bound the qualification envelope for the ACU MSRR.	report issued with this audit report), and ACU's response, dated June 12, 2024 (ML24164A236), to request for confirmation of information (RCI) 4.2-1, the staff determined that additional information in the scope of question Gen-2 related to graphite and how it will be qualified is not necessary at the CP stage and can be addressed in the staff's review of an OL application, as appropriate.
Gen-3	The ACU document, "Initial Audit Response to Technical Topic areas", provided discussion on potential corrosion and degradation mechanisms. The NRC staff noted that several comparisons are drawn to MSRE experience and experience from other programs. However, the NRC staff notes that the MSRR uses a different salt and a different structural alloy than the MSRE. The NRC staff would like to understand how ACU will demonstrate that MSRE and other data are applicable to the MSRR design including salt compositions (including generation of fission products), acceptable levels of impurities, appropriate quantities of beryllium (Be) to add for redox control, alloys used (including weld filler metals), and operating and accident conditions (temperatures/fluences).	Following review of written information/responses provided for audit in response to this question and initial discussions with ACU, the staff determined that additional MSRR-specific information related to corrosion and other degradation mechanisms for metallic materials, and materials testing, would be necessary for the review of the CP application. Therefore, the staff provided ACU additional detailed questions on these topics in the scope of the separate PSAR chapter 4, chapter 6, and section 9.6 audit and PSAR section 9.2 and chapter 13 audit (see separate audit reports issued with this audit report). Based on the resolution of topics in these two separate audits, as well as the information ACU provided in its response, dated April 30, 2024 (ML24121A272), to request for additional information (RAI) 2, including information on ACU's Degradation Management Program (DMP) for the MSRR, the staff determined that additional information in the scope of this question Gen-3 is not necessary for the review of the CP application.
Gen-4	The NRC staff notes that the discussion of potential corrosion and degradation mechanisms of metallic components in the ACU document, "Initial Audit Response to Technical Topic areas", does not appear to consider degradation mechanisms other than general corrosion.	Following review of written information/responses provided for audit in response to this question and initial discussions with ACU, the staff determined that additional MSRR-specific information related to corrosion and other degradation mechanisms for metallic materials, and materials testing, would be

	How does ACU plan to address other modes of degradation that should be taken into account for design or service life such as environmentally assisted cracking, irradiation effects, thermal fatigue/stress, etc.?	necessary for the review of the CP application. Therefore, the staff provided ACU additional detailed questions on these topics in the scope of the separate PSAR chapter 4, chapter 6, and section 9.6 audit and PSAR section 9.2 and chapter 13 audit (see separate audit reports issued with this audit report). The staff determined that question Gen-4 is resolved based on the resolution of topics in these two separate audits, as well as the information ACU provided in its response, dated April 30, 2024 (ML24121A272), to RAI 2, including information on ACU's Degradation Management Program (DMP) for the MSRR.
Gen-5	Based on its audit review of the ACU document, "Initial Audit Response to Technical Topic areas", the NRC staff would like to understand: where precisely are the two barriers assumed as part of the maximum hypothetical accident (MHA) located, and what radionuclides are present outside one or both of these barriers? For example, are gas management, tritium, spent fuel, or sample lines outside the reactor system boundary? The NRC staff notes that the restrictive assumed barrier leak rates appear to play a large role in the calculated dose, so a relatively small radionuclide source outside these barriers could be capable of producing a comparable dose.	During audit discussions, ACU stated that some systems and components containing radionuclides (e.g., the fuel storage tanks) are located outside the reactor system boundary and other associated radionuclide barriers (the reactor enclosure and reactor cell). However, ACU stated that its intent is that any other radionuclide releases that could occur outside these barriers would be bounded by the MHA. As an example, for the fuel storage tanks, fuel would not be moved from the reactor system into the tanks until after a decay period (so the radionuclide inventory would be lower) and the airtight fuel storage enclosure would also provide a barrier to radionuclide release due a potential fuel storage tank leak. ACU stated that it would revise the PSAR to provide additional information regarding radionuclide release events of the types discussed in questions Gen-5 and Gen-6 and why they are bounded by the MHA. The staff provided ACU with additional detailed questions on these topics in the scope of the separate PSAR section 9.2 and chapter 13 audit (see separate audit report issued with this audit report). Based on the resolution of topics in that audit and ACU's associated

		PSAR revisions, the staff determined that questions Gen-5 and Gen-6 are resolved.
Gen-6	Based on its audit review of the ACU document, “Initial Audit Response to Technical Topic areas”, the NRC staff would like to understand: what mechanism(s) for release of radionuclides from any sources listed in the examples in Audit Question Gen-5 are credible (e.g., small leaks, handling mishaps, or release of accumulated gases)? The NRC staff notes that this helps inform the potential dose, because although the fuel handling system could be the most obvious potential release pathway (and the treatment is not fully clear in the PSAR), this may not be the only release pathway.	See resolution of question Gen-5.
Gen-7	The PSAR uses the terms “safety-related,” “non-safety-related,” and “important to safety” to describe various MSRR structures, systems, and components (SSCs). The principal design criteria (PDCs) listed in PSAR Section 3.1 refer to SSCs as well as design conditions and functions that are “important to safety.” Other portions of the PSAR commonly refer to “safety-related” SSCs, including in discussing meteorological, hydrological, and seismic loading in PSAR Sections 3.2, 3.3, and 3.4. With respect to design of SSCs for seismic loading, PSAR Section 3.4 states that SSCs designated Seismic Category I are “safety-related SSCs and … those SSCs required to support shutdown and maintain the MSRR in a safe shutdown condition.” PSAR Section 3.5 states that MSRR systems and components are “important to safety” if “they perform safety functions during normal operations or are required to prevent or mitigate the consequences of operational transients or accidents.”	(The staff notes that the original transmitted question Gen-7 contains an editorial error in that the “safety-related” definition quoted from section 3.5.2.1 of the PSAR (Revision 0, transmitted by ACU’s letter dated August 12, 2022) was missing the third bullet. The bullet is added in [] on the version of question Gen-7 reproduced in this audit report.) During audit discussions, ACU stated that it would review and revise the PSAR to make this terminology and its application consistent. ACU also stated that it would revise information in PSAR chapter 3, including PSAR table 3.4-1 which describes safety classifications for MSRR SSCs, for clarity and consistency with other PSAR chapters. Accordingly, by letter dated July 30, 2024 (ML24219A258), ACU revised PSAR chapter 3 and other portions of the PSAR to make changes including the following: • Clearly defining the MSRR safety classification as a binary system (i.e., SSCs are either

	PSAR Section 3.5.2.1 states that: "Safety-related" is a classification applied to items relied on to remain functional during and following a postulated accident to ensure the • integrity of the MSRR facility safety-related infrastructure. • capability to shut down the MSRR and maintain it in a safe shutdown condition. • [capability to prevent or mitigate the consequences of postulated accidents identified through accident analyses (Chapter 13) that could result in potential offsite or worker exposures comparable to the applicable guideline exposures set forth in 10 CFR Part 20.] However, it is not clear to the staff if the PSAR provides a consistent definition of the terms "safety-related," "non-safety-related," or "important to safety," or if these terms are used consistently throughout the PSAR. The NRC staff also notes that some codes and standards cited in the PSAR may use and/or contain definitions for these terms. It is not clear whether definitions in these codes and standards are intended to apply to the MSRR. The NRC staff notes that an understanding of these concepts will be important in classifying and understanding the importance of various SSCs at a more detailed level. The staff requests clarification on the definitions of each of these terms and how ACU is applying them to categorize the SSCs in the MSRR design.	"safety-related" or "non-safety-related") and eliminating the use of the term "important to safety." • Replacing the term "important to safety" with "safety-related" in the PDCs and other parts of the PSAR. • Revising PSAR table 3.4-1 to refine the categorization of SSCs for consistency with revised information in other PSAR chapters, including updated information associated with design updates. • Revising the PSAR to more clearly describe how certain codes and standards are applied to the MSRR (also related to other audit questions). • In response to additional NRC staff feedback, removing the "Quality Level Group" column and the "Quality Level" designation from PSAR table 3.4-1 since this terminology is not used elsewhere in the PSAR and quality classifications are separately addressed in PSAR sections 3.5.1 and 3.5.2.4. • Making other revisions to information in PSAR table 3.4-1 for clarity and consistency with other PSAR chapters in response to additional NRC staff feedback provided during audit discussions of draft revisions of PSAR table 3.4-1 which ACU provided for audit. • Adding a definition of the term "anticipated operational occurrence" in PSAR section 3.5.2.1 in response to additional NRC feedback provided during audit discussions that the meaning of this term as used in the PSAR was not fully clear.

Additional Topic: Editorial Items	The NRC staff provided ACU with approximately 60 miscellaneous editorial or consistency items that it noted during its review of the PSAR. The staff asked ACU to consider whether PSAR revisions were appropriate to make clarifications or corrections related to these items. These items included some apparent inconsistencies with other information in the PSAR or provided for audit, typographical/editorial errors, and unclear or erroneous references. These items included observations that PSAR figures including 3.1-3, 3.5-1, 4.1-1, 4.1-2, 4.1-3, 6.1-1, and 6.2-1 did not appear to be consistent with updated PSAR text.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258), to address necessary items as appropriate. ACU also acknowledged in its July 30, 2024, letter which transmitted the modified PSAR that because the PSAR reflects a preliminary design, some PSAR figures were not updated as the design matured; however, ACU clarified that in the case of any discrepancies, the PSAR text is authoritative.

6.0 EXIT BRIEFING

The staff conducted an audit closeout meeting on August 20, 2024. At the exit briefing the staff reiterated the purpose of the audit and summarized the audit activities. Additionally, the staff stated that they did not identify areas where further additional information would be necessary to support the review.

There were no deviations from the audit plan.

7.0 ADDITIONAL INFORMATION RESULTING FROM AUDIT

RAIs and RCIs were generated as a result of topics discussed in this audit and other chapter-specific audits for this review, and ACU's responses supported the resolution of audit questions as noted above. ACU also updated the PSAR on its own initiative as noted above to address several items discussed during the audit.

8.0 OPEN ITEMS AND PROPOSED CLOSURE PATHS

Not applicable.

SUMMARY REPORT FOR THE REGULATORY AUDIT OF ABILENE CHRISTIAN UNIVERSTY MOLTEN SALT RESEARCH REACTOR CONSTRUCTION PERMIT PRELIMINARY SAFETY ANALYSIS REPORT INSTRUMENTATION AND CONTROL SYSTEMS (CHAPTER 7)

March 2023 – August 2024

1.0 BACKGROUND AND PURPOSE

By letter dated August 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22227A202), as supplemented by letter dated October 14, 2022 (ML22293B816), Abilene Christian University (ACU) submitted to the U.S. Nuclear Regulatory Commission (NRC), an application for a construction permit (CP) for a molten salt research reactor (MSRR), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Section 104c of the Atomic Energy Act of 1954, as amended. The application included a preliminary safety analysis report (PSAR) (ML22227A203). PSAR chapter 7, "Instrumentation and Control Systems," describes the instrumentation and control (I&C) systems that monitor plant parameters, controls and components, and provide an interface to the plant operators.

This audit enabled the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) to gain a better understanding of PSAR chapter 7 through review and discussion of underlying supporting documentation. Enhanced understanding and communications supported effective and efficient resolution of technical issues, including through development of information needs where needed.

2.0 AUDIT REGULATORY BASES

The basis for the audit is the regulations in 10 CFR Part 50, Sections 50.34(a), "Preliminary safety analysis report," and 50.35, "Issuance of construction permits."

3.0 AUDIT OBJECTIVES

The primary objective of the audit was to enable a more effective and efficient review of PSAR chapter 7 through the staff's review and discussion of supporting documentation with ACU. Gaining access to underlying documentation and engaging in audit discussions about I&C design facilitated the staff's understanding of the MSRR CP application and aided in assessing the safety of the proposed research reactor. The audit improved communication and provided detailed information for the staff.

4.0 SCOPE OF THE AUDIT AND AUDIT ACTIVITIES

The audit was conducted from March 2023 to August 2024, via teleconference and the electronic reading room (Certrec), and during a site visit conducted May 17-18, 2023. The staff conducted the audit in accordance with the Office of Nuclear ReactorRegulation (NRR) Office Instruction NRR-LIC-111, Revision 1 "Regulatory Audits" (ML19226A274).

Members of the audit team, listed below, were selected based on their detailed knowledge of the audit subject matter. Audit team members included:

1. Boyce Travis, NRR (Senior Nuclear Engineer)

2. Joseph Ashcraft, NRR (Electronics Engineer)
3. Calvin Cheung, NRR (Electronics Engineer)
4. Kyle Song, NRR (Electronics Engineer)
5. Richard Rivera, NRR (Project Manager)
6. Edward Helvenston, NRR (Project Manager)
7. Zackary Stone, NRR (Project Manager)
8. Michael Balazik, NRR (Project Manager)
9. Brian Bettes, NRR (Project Manager)
10. Mohsin Ghazali, NRR (Project Manager)
11. Gregory Oberson, NRR (Branch Chief, Technical)
12. Michael Wentzel, NRR (Branch Chief, Licensing)
13. Stephen Philpott, NRR (Branch Chief, Licensing)

Prior to the audit, the audit team reviewed PSAR chapter 7 and defined in the audit plan (ML23065A051) the general range of topics to be addressed and focused on during the audit. The following table documents dates that the staff transmitted audit questions and when audit meetings were held:

Audit Questions (ADAMS Accession No.)	Audit Meetings
March 14, 2023 (ML23073A302)	May 17, and 18, 2023
	February 1, 2024
	March 26, 2024
	August 20, 2024

The staff reviewed the following documents via the electronic reading room (ERR):

- Written responses that ACU prepared for certain questions to address the questions and/or facilitate discussion with NRC staff
- PSAR pages indicating changes proposed by ACU in response to various audit questions

The NRC staff previously issued an interim audit report related to this audit by letter dated June 22, 2023 (ML23157A064).

5.0 SUMMARY OF AUDIT OUTCOME

The staff's audit focused on the review of supporting documents associated with the questions identified in the e-mail dated March 14, 2023 (ML23073A302). The staff reviewed information through the ACU Certrec portal and held discussions with ACU staff to understand and resolve questions. In many cases, ACU updated the PSAR to resolve items discussed in the audit. The tables below replicate specific audit questions transmitted by email to ACU as listed above and summarize the resolution of the audit questions.

Resolution of Questions on Section 7.2 "Design of Instrumentation and Control Systems"

Question Number	Question	Resolution
7.2-1	ACU provided an overall instrumentation and control (I&C) architecture drawing (Figure 1) in response to NRC's request for supplemental information (RSI). The NRC staff requests the following clarification/information: a. Does the "DCS" envelope the entire upper box as shown? b. RSI Section 1.2.3, Response to RSI 1.c states, "Communication isolation between NSR and SR systems is accomplished by isolating individual subsystems on separate communication buses." The NRC staff requests additional detail to support this statement. Specifically: I. What is the safety classification of the facility data bus? RCS non safety data bus shows bidirectional communication to the facility data bus and without isolation. II. Facility data bus shows bidirectional communication to the Safety Data Bus and without isolation. Please describe what communication is occurring. III. Are there additional isolation devices, as there is a sole data diode depicted? c. The NRC staff requests details, including basic logic, for the bottom two blocks with "Reactor Trip Breakers" and "ESF Breakers," as it is unclear to the NRC staff what these blocks represent. I. Are they headers to the items below? II. Do all components listed in each block initiate simultaneously? III. How does the manual trip/initiation interface with this? IV. What are the initiating signals for a draining the fuel salt (i.e. SCRAM) and how are they logically connected to de-energize the equalization valves?	ACU provided PSAR Revision 2 (ML24219A258) to update Section 7.2 and Figure 7.2-1 with additional clarifying details, including improved identification of safety classification and communication protocol.

	V. Please describe the heater power disconnects. VI. Please describe the gas pressure isolation. VII. Please describe the PSAR, Section 4.2.5.3, Figure 4.2-1 and if any of the components listed in these bottom two blocks are represented.	
7.2-2	RSI Section 2.2.1, Response to RSI 2.a, states, "The initiation of the ESFAS always triggers the RPS." The NRC staff requests additional information/details on how this happens.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to include PSAR Section 7.4.4.1, "ESFAS Initiation," which provides an explanation for why ESFAS activation results in the reactor being tripped.
7.2-3	RSI Section 2.2.1, Response to RSI 2.a states, "A loss of SR RMS detectors will trigger a SCRAM, potentially after a time delay." The NRC staff requests the following clarifications: I. What RMS detectors are safety related? II. What is the purpose of the time delay and what factors are considered that would necessitate a time delay?	I. ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to identify the safety-related sensors in Figure 7.2-1, "Instrumentation and Controls Diagram with Subsystems." In addition, ACU included statements in PSAR Sections 7.5, "Engineered Safety Features Actuation System," and 7.7, "Radiation and Environmental Monitoring System" clarifying which sensors are safety-related. II. ACU stated during audit discussions that the possibility of time delay was included in error and that

		safety-related sensors which are used to shutdown the reactor will not include a time delay. ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to remove discussion of a time delay.
7.2-4	RSI Section 2.2.1, Response to RSI 2.a states, “As the ESFAS can potentially initiate based on readings from the RMS….” The NRC staff notes that based on PSAR Section 7.5, “ESFAS activates upon detection of fission products…by sensors from the RMS.” No other information indicates other initiating inputs besides from the RMS. The NRC staff requests clarification on what else initiates ESFAS and when would it not initiate based on RMS.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to provide Table 7.5-1, “ESFAS Initiation,” which provides information on which signals initiate an ESFAS actuation. In addition, ACU added clarification on ESFAS initiating limits to the PSAR by including Section 7.5.3, “Initiations Limits and Signals.” The staff will evaluate the final design of the ESFAS and its initiating inputs during an OL application review.
7.2-5	RSI Section 2.2.1, Response to RSI 2.a states that the “Triggering Value” on Table 1 for “Loss of RMS or Components” is TBD. The NRC staff requests clarification on what is still being determined.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to provide Table 7.5-1, “ESFAS Initiation,” which provides information on which signals initiate an ESFAS actuation. In addition, ACU added clarification on ESFAS initiating limits to the PSAR by including Section 7.5.3, “Initiations Limits and Signals.” The staff will

		evaluate the final design of the ESFAS and its triggering values during an OL application review.

Resolution of Questions on Section 7.4 "Reactor Protection System"

Question Number	Question	Resolution
7.4-1	PSAR Section 7.4.1 identified the design bases and design criteria for the reactor protection function. Please clarify if the intent is to have PDC 13 applicable to Section 7.4. If not, provide justification as to why PDC 13 is not applicable for the RPS to address safety-related sensors and range requirements.	ACU provided PSAR Rev 2, Section 7.4.1 which includes information on how PDC 13 will be met. A complete evaluation of the reactor protection function and how it addresses PDC 13, 21, and 26 will be performed in the operating license application review.
7.4-2	In PSAR Section 7.4, no logic, schematic, and circuit diagrams are provided, and the descriptions provided are not sufficient for the NRC staff to understand how the signals provide safety trips. The NRC staff requests that ACU provide additional information to explain how the signals provide safety trips.	ACU provided PSAR Rev 2, Section 7.4.1 with revised language to clarify how the reactor protection system (RPS) monitors safety parameters to ensure safety limits are not exceeded.
7.4-3	As stated in Response to RSI 1.c, and in consideration of criteria found in IEEE Standard 603-2018 and IEEE 7-4.3.2-2003, the DCS shall be such that communication between NSR and SR systems is regulated appropriately. The NRC staff requests that ACU describes the communication and how it is regulated as shown on RSI Figure 1, for the communication between the facility data bus and safety data bus.	ACU provided PSAR Rev 2, Figure 7.2-1 with updates to depict communication with greater detail and accurately.
7.4-4	The NRC staff requests that ACU identifies valves and lines shown in the PSAR, Section 4.2.5.3, Figure 4.2-1 as it relates to PSAR Section 7.4	ACU provided PSAR Rev 2 and added Figures 7.4-1 and 7.4-2,

		which provides sufficient detail.
7.4-5	It is unclear to the NRC staff on how loss of power is detected for drain valves and louvers in PSAR Section 7.5. Please provide an explanation and a description on what is actually detected (loss of voltage, loss of frequency, or both, time delay). Additionally, explain if loss of power is detected at the same place for RPS and ESFAS.	ACU provided PSAR Rev 2 and added Figures 7.4-1, “Reactor Trip Relay Circuits” and 7.5-1, “ESFAS Actuation”, which provide sufficient detail on the detection of loss of power.

Resolution of Questions on Section 7.5 “Engineered Safety Features Actuation System”

Question Number	Question	Resolution
7.5-1	PSAR Section 7.5.1 identified the design bases and design criteria applicable to ESFAS. ESFAS should be designed to assume a safe state on loss of electrical power. Based on RSI Section 2.2.3, Response to RSI 2.c, the louvers and valves seal on loss of power. Please clarify if the intent to have PDC 23 applicable to Section 7.5. If not, provide justification as to why PDC 23 is not applicable.	ACU provided PSAR Rev. 2, with an update to Section 7.5.1, which includes PDC 23 to design the ESFAS to fail into a safe state in the event of a loss of power or adversarial environmental conditions.
7.5-2	As stated in PSAR Section 7.5.2, the ESFAS triggers the reactor enclosure isolation system to place the enclosure in its passively safe, low-leakage configuration. The NRC staff requests the following clarification/information: a. Details on what components make up the reactor enclosure isolation system. Are the following items part of this system, and are these different components? If the same, consistent language should be used. I. reactor enclosure penetrating gas valves II. gas penetrating line valves - from Figure 1 from the response to RSI	ACU provided PSAR Rev. 2, with updated Section 7.5.2 which includes corrected and consistent language, and contains Figure 7.5-1, “ESFAS Actuation,” which illustrates the electrical supply of enclosure penetrations and auxiliary heat removal system (AHRS) louvers, and the resistive heating electrical power source.

	III. gas management system reactor enclosure penetration isolation valves – from PSAR 7.5.4 RSI Section 2.2.3, Response to RSI 2.c states "[t]he ESFAS brings the MSRR into a configuration to meet the designed leak rate…by sealing valves on all gas penetrations through the reactor enclosure." How many gas penetrations and valves are there and how are they configured?	
7.5-3	As stated in PSAR Section 7.5.3, the ESFAS actuation closes the auxiliary heat removal system intake and exhaust louvers. The NRC staff requests the following clarification/information: Details on what components make up the reactor cell air louvers. Are the following items part of this system, and are these different components? If the same, consistent language should be used. I. cell air louvers II. AHRD intake and exhaust louvers III. Cell louver mechanism - from Figure 1 from the response to RSI IV. air louvers in the auxiliary heat removal system – from PSAR Section 7.5.4	ACU provided PSAR Rev 2, with updated Section 7.5.3 which includes corrected and consistent language.
7.5-4	As stated in PSAR Section 7.5.4, ESFAS isolates based on inputs from RMS. Please clarify if there anything else that provides input and provide additional information on what actuates ESF (inputs, basic logic) and what happens upon actuation.	Resolved by Question 7.2-4

Resolution of Questions on Section 7.6 "Human-Machine Interface"

Question Number	Question	Resolution
7.6-1	PSAR Section 7.6.1 identified the design bases and design criteria applicable to the Control Console and Display Instruments.	Resolved through other questions via clarifications made to PDCs and Figure 7.2-1.

	I. It is unclear to the NRC staff if the intent for the control console, display instruments, and equipment is to be readily testable and capable of being accurately calibrated. Please provide more information or explain where this is discussed in the PSAR. If this is not intended, please provide justification as to why testability and calibration is not necessary. II. The NRC staff notes that the designed range of operation of each device should be sufficient for the expected range of variation of monitored variables under all normal and transient conditions of operation. Please clarify if the intent is to have PDC 13 applicable to Section 7.6. If not, please provide justification as to why PDC 13 is not applicable. III. It is unclear to the NRC staff if the intent for the control console instruments and equipment is to be designed to assume a safe state on loss of electrical power or to have a reliable source of emergency power sufficient to sustain operation of specific devices. Please provide more information or explain where this is discussed in the PSAR. If this is not intended, please provide justification as to why these power requirements are not necessary.	
7.6-2	Does ACU intend to have a TS for control console and display instruments?	Addressed in Chapter 14.
7.6-3	RSI Section 3.2.1, Response to RSI 3.a stated "PDC19 will be deleted." The NRC staff notes that the 3 bullet portions shown in PSAR Section 7.6.1 on PDC 19 still seem to be appliable. Please clarify if PSAR Section 3.1.2.2 listing for PDC 19, 2nd half of 2nd paragraph on "locations outside the control room," are the only items intended to be deleted.	Addressed in Chapters 10 and 13.

7.6-4	PSAR Section 7.6.3 states "secondary and analog systems sufficient to maintain control in the event of a total failure of the DCS also are present." Please clarify if the portions of the control console and display are safety-related and if the secondary and analog backups are safety-related.	Resolved through other questions via clarifications made to PDCs and Figure 7.2-1.

Resolution of Questions on Section 7.7 "Radiation and Environmental Monitoring System"

Question Number	Question	Resolution
7.7-1	The NRC staff notes that no preliminary PDC have been selected to be applicable to the PSAR Section 7.7. Please clarify if there are any applicable PDCs that are intended to be met.	ACU provided PSAR Rev. 2, Section 7.7.1 which includes information on the applicable PDCs for the Radiation and Environmental Monitoring System.
7.7-2	The NRC staff requests additional details on quantity, location, type, basic logic, etc. for monitors in safety-critical areas, specifically details on sensors that feed into RPS and ESFAS.	ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to include Section 7.7.2, "Facility Sensor Stations," and updated Figure 7.2-1

6.0 EXIT BRIEFING

The staff conducted an audit closeout meeting on August 20, 2024. At the exit briefing the staff reiterated the purpose of the audit and summarized the audit activities. Additionally, the staff stated that they did not identify areas where further additional information would be necessary to support the review.

There were no deviations from the audit plan.

7.0 ADDITIONAL INFORMATION RESULTING FROM AUDIT

No RAIs were generated as a result of this audit. However, ACU updated the PSAR on its own initiative as noted above to address several items discussed during the audit.

8.0 OPEN ITEMS AND PROPOSED CLOSURE PATHS

Not applicable.

SUMMARY REPORT FOR THE REGULATORY AUDIT OF ABILENE CHRISTIAN UNIVERSTY MOLTEN SALT RESEARCH REACTOR CONSTRUCTION PERMIT PRELIMINARY SAFETY ANALYSIS REPORT CHAPTERS 2 (SITE CHARACTERISTICS) AND 3 (DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS)

March 2023 – August 2024

1.0 BACKGROUND AND PURPOSE

By letter dated August 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22227A202), as supplemented by letter dated October 14, 2022 (ML22293B816), Abilene Christian University (ACU) submitted to the U.S. Nuclear Regulatory Commission (NRC), an application for a construction permit (CP) for a molten salt research reactor (MSRR), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Section 104c of the Atomic Energy Act of 1954, as amended. The application included preliminary safety analysis report (PSAR) Revision 0 (ML22227A203). PSAR chapter 2, "Site Characteristics," describes the site location, including a discussion of the population in the vicinity, the distribution of infrastructure and natural features, as well as the basis for selection of the MSRR site. PSAR chapter 3 describes the architectural and engineering design criteria for the structures, systems, and components (SSCs) of the MSRR.

This audit enabled the NRC staff (the staff) to gain a better understanding of PSAR chapters 2 and 3 through review and discussion of underlying supporting documentation. Enhanced understanding and communications supported effective and efficient resolution of technical issues, including through development of information needs where needed.

2.0 AUDIT REGULATORY BASES

The basis for the audit is the regulations in 10 CFR Part 50, Sections 50.34(a), "Preliminary safety analysis report," and 50.35, "Issuance of construction permits."

3.0 AUDIT OBJECTIVES

The primary objective of the audit was to enable a more effective and efficient review of PSAR chapters 2 and 3 through the staff's review and discussion of supporting documentation with ACU. Gaining access to underlying documentation and engaging in audit discussions about the site characteristics and design of SSCs facilitated the staff's understanding of the CP application and aided in assessing the safety of the proposed research reactor. The audit improved communication and provided detailed information for the staff.

4.0 SCOPE OF THE AUDIT AND AUDIT ACTIVITIES

The audit was conducted from March 2023 to August 2024, via teleconference and the electronic reading room (Certrec), and also during a site visit conducted May 17-18, 2023. The staff conducted the audit in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction NRR-LIC-111, Revision 1, "Regulatory Audits" (ML19226A274).

Members of the audit team, listed below, were selected based on their detailed knowledge of the audit subject matter. Audit team members included:

1. Boyce Travis, NRR (Senior Nuclear Engineer)
2. Scott Stovall, RES (Geophysicist)
3. Sarah Tabatabai, NRR (Geophysicist)
4. Jenise Thompson, NRR (Geologist)
5. Kenneth See, NRR (Senior Hydrologist)
6. Michael Mazaika, NRR (Meteorologist)
7. Kevin Quinlan, NRR (Senior Meteorologist)
8. Amit Ghosh, NRR (Hazards Analyst)
9. Andrew Prinaris, NRR (Senior Civil Engineer (Structural))
10. Luissette Candelario, NRR (Civil Engineer (Structural))
11. Patrick Koch, NRR (Civil Engineer (Structural))
12. Zuhan Xi, NRR (Geotechnical Engineer)
13. Nicholas Hansing, NRR, (Mechanical Engineer)
14. Matthew McConnell, NRR (Senior Electrical Engineer)
15. Adakou Foli, NRR (Electrical Engineer)
16. Alexander Chereskin, NRR (Materials Engineer)
17. Mohsin Ghazali, NRR (Project Manager)
18. Brian Bettes, NRR (Project Manager)
19. Michael Balazik, NRR (Project Manager)
20. Richard Rivera, NRR (Project Manager)
21. Zackary Stone, NRR (Project Manager)
22. Edward Helvenston, NRR (Project Manager)
23. Michelle Hayes, NRR (Branch Chief, Technical)
24. Gregory Oberson, NRR (Branch Chief, Technical)
25. Michael Wentzel, NRR (Branch Chief, Licensing)
26. Stephen Philpott, NRR (Branch Chief, Licensing)

Prior to the audit, the audit team reviewed PSAR chapters 2 and 3 and defined the general range of topics in the audit plan (ML23065A048) to be addressed and focused on during the audit. The following table documents dates that the staff transmitted audit questions and when audit meetings were held:

Audit Questions (ADAMS Accession No.)	Audit Meetings
March 24, 2023 (ML23086A017)	May 17-18, 2023
April 5, 2023 (ML23095A081)	July 20, 2023
August 2, 2023 (ML23219A213)	August 10, 2023
October 5, 2023 (ML23283A017)	September 21, 2023
December 1, 2023 (ML23335A117)	October 3, 12, 17, and 22, 2023
December 11, 2023 (ML23345A170)	November 2 and 16, 2023
	December 11, 12, and 21, 2023
	January 23, 2024
	February 13, 2024
	March 7, 12, 14, and 22, 2024
	April 2, 16, and 18, 2024
	May 8 and 14, 2024
	June 4, 2024
	August 20, 2024 (exit meeting)

The staff reviewed the following documents via the electronic reading room:

- Written responses that ACU prepared to address certain questions and/or facilitate discussion with NRC staff
- PSAR pages and other documents indicating PSAR changes or other information proposed for docketing by ACU in response to various audit questions
- Enprotec Hibbs & Todd (eHT) Document “eHT Action Items from ACU NEXT Meeting with NRC 3/14/24” (redacted)
- Example eHT boring logs from other sites near the proposed MSRR site
- Parkhill pier loading calculation
- Air crash hazard calculations
- Geotechnical Investigation Report, including report dated September 15, 2020, and supplemental information added to the report dated December 11, 2020
- Detailed Science and Engineering Research Center (SERC) design drawings including as-built drawings
- Linbeck completed checklists for reinforcement and pouring of concrete piers for the SERC systems pit (provided for all 16 piers)
- Document “Sample Pier Report Explanation”
- Terracon “Drilled Pier Construction Report” documents dated April 18-22, 2022 (five documents covering all 16 piers)
- Draft white paper “MSRR Seismic Hazard Response,” dated December 20, 2023
- Parkhill, Linbeck, and Ingram Concrete & Aggregates documents related to mix design for flowable fill for SERC Research Bay backfill
- Terracon “Laboratory Compaction Characteristics of Soil Report” documents (for onsite fill) dated May 2, 2022
- Terracon “Laboratory Compaction Characteristics of Soil Report” documents (for select fill) dated June 29 and July 12, 2022
- Document “02530 – ACU SERC – MASS CONCRETE QA/QC PLAN,” dated March 8, 2022
- Document “Project Specific Quality Assurance/Quality Control Plan,” dated February 15, 2022
- Letter from G. Scott Yungblut (eHT) to Timothy L. Head (ACU), “Re: QA/QC Program,” dated January 10, 2024
- Letter from Zach Lindauer (Parkhill) to Tim Head (ACU), “Re: ACU Science and Engineering Research Center (SERC) – PSAR Question 2.5-9,” dated February 15, 2024
- Linbeck document “Quality Assurance/Quality Control Program,” revised October 2022
- Information related to reactor system loads and calculation methodologies

The NRC staff previously issued an interim audit report related to this audit by letter dated June 22, 2023 (ML23157A064).

5.0 SUMMARY OF AUDIT OUTCOME

The staff’s audit focused on the review of supporting documents associated with the questions provided to ACU during the audit. The staff reviewed information through the ACU Certrec portal and held discussions with ACU staff to understand and resolve questions. In many cases, ACU updated the PSAR to resolve items discussed in the audit. The tables below replicate the specific audit questions transmitted in emails to ACU as listed above, and summarize the resolution of the audit questions. A table also lists and summarizes the resolution of some additional topics covered as part of this audit beyond the scope of the specific audit questions.

Resolution of Questions on Section 2.3, "Meteorology"

Question Number	Question	Resolution
2.3-1	PSAR Section 2.3.1 describes the general and local climate of the MSRR site, including information on temperature and humidity. However, the PSAR does not appear to specify dry-bulb temperature and concurrent or non-concurrent humidity indicator values (typically wet-bulb temperature, though other indicators of ambient moisture may be provided) that could pertain to the design of the various HVAC and other heating and cooling SSCs serving the MSRR facility. The staff notes that the acceptability of such system-specific specifications is typically needs to be determined prior to system construction. Please provide information to allow the NRC staff to understand what safety impacts, if any, these values could have on associated SSCs such as auxiliary cooling and heating (PSAR Chapter 5) and heating, ventilation, air conditioning and cooling systems (PSAR Chapter 9). Specifically: I. Please discuss whether design dry-bulb temperatures and coincident or non-coincident atmospheric moisture indicators are applicable to the design of cooling and heating systems and whether there are any associated safety impacts. If so, please describe the data used for these designs including the corresponding return periods or percent exceedance levels, and the source(s) of such data. II. Discuss whether there is a need to evaluate the persistence of high and low dry-bulb temperature conditions for the HVAC and other heating and cooling systems described in PSAR Chapters 5 and	In information provided for audit, ACU provided relevant temperature values used in the SERC heating, ventilation, and air conditioning (HVAC) design and stated that these values are based on American Society of Heating, Refrigeration, and Air-Conditioning Engineers (ASHRAE) standards, which the staff notes are an accepted source for temperature data based on NRC guidance. However, ACU stated that the performance of the HVAC systems has no safety-related function for the MSRR (and other safety-related SSCs also do not rely on the HVAC systems to function). Because the temperatures were not used in any of the safety analyses, no PSAR changes or other additional docketed information were necessary in response to this question.

	9, and for the SSCs described in PSAR Chapter 3 as applicable. If so, please provide results of such evaluations, as appropriate.	
2.3-2	PSAR Section 2.3.1.1 discusses, and PSAR Table 2.3-1 lists, the fifteen heaviest snowfall events presumably occurring at Abilene, TX, since 1950. The NRC staff notes that the data are attributed to NOAA via a National Centers for Environmental Information (NCEI) website (PSAR Ref. 2.3-1), but there are multiple, official climate observing stations in the Abilene area. It is not clear to the NRC staff whether the data are attributed strictly to Abilene or based on measurements within a 50-mile radius of Abilene as other text indicates. Therefore: I. confirm whether the data on maximum snowfall events occurred only at Abilene and/or at other stations within 50 miles of Abilene; II. in either case, on the basis of the above, confirm that the conditions are representative of those expected at the ACU MSRR site; III. please clarify statements in the text and PSAR Table 2.3-1 including identifying the names and locations of any other stations used relative to the ACU MSRR site.	ACU modified PSAR section 2.3.1.1 by letter dated July 30, 2024 (ML24219A258), to provide clarifying information as appropriate. Also, in information provided for audit, ACU stated for comparison with PSAR information that the highest recorded snowfall within 50 miles of the proposed MSRR site was 12 inches in Albany, Texas, on January 27, 1895.
2.3-3	PSAR Section 2.3.1.1 states that rainfall, temperature, and humidity are measured hourly at Dyess Air Force Base (AFB) and are summarized in PSAR Figures 2.3-1 to 2.3-3. I. The titles of PSAR Figures 2.3-1, 2.3-2, and 2.3-3 indicate the data are for Abilene. The NRC staff	ACU modified PSAR section 2.3.1.1 and PSAR section 2.3 figures and references by letter dated July 30, 2024 (ML24219A258), to provide clarifying information as appropriate. In addition, ACU confirmed during the audit that although PSAR figures 2.3-1 and 2.3-3 indicate that the data cover

	notes that this appears to be inconsistent with the text of PSAR Section 2.3.1.1 as it is attributed to Dyess AFB. Please clarify. II. Please clarify and confirm the accuracy of PSAR Section 2.3 references relevant to PSAR Section 2.3.1.1. For example, PSAR Section 2.3.1.1 appears to indicate that data in PSAR Figures 2.3-1, 2.3-2, and 2.3-3 are from NOAA, but the actual figures cite different references for this data. Additionally, please clarify and confirm that the data available through the "Climate Explorer" website is traceable to NOAA. III. It is not clear to the NRC staff if the maximum daily rainfall amounts by month in PSAR Figure 2.3-1 and the maximum and minimum daily (dry-bulb) temperatures by month in PSAR Figure 2.3-3 are fully representative of the ACU MSRR site area. The NRC staff notes that, for example, a 12-hour total of 26 inches of rain fell at a location about 43 miles northeast of Abilene in Shackleford County as a result of the remnants of Hurricane Amelia in 1978 (this does not appear to be mentioned in the PSAR). The NRC staff notes that data for a single location may not necessarily be representative of a wider geographical area and consider extremes that could occur in this area. Please discuss why the data on rainfall and temperature extremes in PSAR Figures 2.3-1 and 2.3-3 is sufficient to characterize the MSRR site area meteorology.	a period of reference from 1944-2021, the correct period of reference is 1948-2021, consistent with the figure titles in the updated PSAR.

2.3-4	PSAR Section 2.3.1.1 states that humidity data for Dyess AFB are summarized in PSAR Figures 2.3-1, 2.3-2, and 2.3-3. PSAR Section 2.3.1.2 states that monthly average humidity data are summarized in PSAR Figure 2.3-4. (Audit Question 2.3-1 separately addresses possible dry-bulb temperature and/or atmospheric moisture indicators related to HVAC and other cooling system design considerations.) I. The NRC staff notes that PSAR Figures 2.3-1, 2.3-2, and 2.3-3 do not appear to include humidity data as indicated in PSAR Section 2.3.1.1; please clarify. II. PSAR Section 2.3.1.2 and Figure 2.3-4 do not appear to clearly indicate the location for the data in PSAR Figure 2.3-4; please clarify.	ACU modified PSAR sections 2.3.1.1 and 2.3.1.2 and PSAR section 2.3 figures by letter dated July 30, 2024 (ML24219A258), to provide clarifying information as appropriate.
2.3-5	I. The historical wind gust and mean wind speeds for Abilene in PSAR Section 2.3.1.3 are attributed to NOAA's National Climatic Data Center but are otherwise unreferenced. Please provide citation(s) to the appropriate source(s). II. The wind gust speed referenced in PSAR Section 2.3.1.3 is indicated to have occurred in June 1983. It is not clear to the NRC staff whether this value represents a fastest-mile wind speed (which was typically used by the National Weather Service (NWS) and industry prior to about 1988), a 3-second gust speed, or a value measured over some other duration. Also, the measurement height does not appear to be indicated and the NRC staff notes this may have changed over time. Please clarify.	ACU modified PSAR section 2.3.1.3 and PSAR section 2.3 references by letter dated July 30, 2024 (ML24219A258), to provide clarifying information as appropriate.

	III. Confirm the reference cited at the bottom of PSAR Table 2.3-4, which provides various ranges of maximum wind speeds and the number of damage reports within those ranges. The table indicates the source of the data is PSAR Ref. 2.3-1. However, the NRC staff checked this reference and noted that for the indicated city and time series, only precipitation and temperature statistics (not statistics related to wind) appear to be available. Please clarify the source (including the location(s) and time series (including whether the data represent conditions in Abilene only or reports and measurements within a certain radius of the MSRR site)) of the data in PSAR Table 2.3-4. IV. In PSAR Table 2.3-4 a total of 25 events are attributed to an “Unknown” wind speed range that is not further explained in the text or table. The NRC staff notes that it is not clear what limits of the “unknown” range could be (e.g., if they could be greater than the upper limit of the 91-100 mph wind speed range). Also, it is not clear whether the indicated wind speeds in PSAR Table 2.3-4 represent sustained or gust wind speeds. PSAR Section 2.3.1.3 indicates a maximum wind gust speed of 78 mph, and the staff notes that wind gusts are typically greater than sustained winds. Therefore, without further explanation of what the ranges in PSAR Table 2.3-4 represent, including the range labeled as “Unknown,” it is not clear whether the maximum wind gust in Section 2.3.1.3 is valid. Please clarify PSAR Section 2.3.1.3 and/or PSAR Table 2.3-4 accordingly.	

2.3-6	I. Please verify that the data available for each type of event through the “USA.com” website, as summarized in PSAR Figure 2.3-6, is traceable to NOAA. If not, please specify the respective source(s). II. A check of the “USA.com” link indicates that the frequency labeled in PSAR Figure 2.3-6 as “Thunderstorms” appears to be mislabeled and corresponds to the frequency of thunderstorm wind events. Please clarify the information related to the data presented in PSAR Figure 2.3-6. III. The NRC staff notes that the frequencies of thunderstorm wind events and hail events in PSAR Figure 2.3-6 could represent an overestimate of those event types for the ACU MSRR site area. Please clarify if PSAR Figure 2.3-6 is a composite of the number of “events” recorded at various places from the same storms within 50 miles of Abilene as the cited resource suggests. If so, please discuss whether the PSAR should be revised to indicate the frequencies of thunderstorm days and hail events at various identified stations in the ACU MSRR site area.	ACU modified PSAR section 2.3 figures and references by letter dated July 30, 2024 (ML24219A258), to provide clarifying information as appropriate.
2.3-7	PSAR Section 2.3.1.5 indicates that the period of record (POR) for the hurricane tracks illustrated in Figure 2.3-8 is from “1930 to the present.” The NRC staff verified the cited online resource (i.e., the NOAA Office for Coastal Management) and noted that the database includes events prior to 1900. PSAR Figure 2.3-8 includes a track for an unnamed storm designated as occurring in “1886”. Please clarify the range of the POR for the data discussed in PSAR Section 2.3.1.5 and associated PSAR Figure 2.3-8, and if needed, resolve any discrepancies.	ACU modified PSAR section 2.3.1.5 and PSAR section 2.3 figures by letter dated July 30, 2024 (ML24219A258), to provide clarifying information as appropriate.

2.3-8	I. Please verify that the data available through the "USA.com" website, as summarized in PSAR Figure 2.3-9, is traceable to NOAA, the National Severe Storms Laboratory, or other credible resource. If not, please specify the respective source. As necessary, please clarify the description for Reference 2.3-9 in PSAR Section 2.3.3 accordingly to establish that linkage. II. PSAR Figure 2.3-9 on "Tornado Magnitude and Distance from Abilene, 1950 – 2010" presents this data utilizing bubbles to plot various EF-scale magnitudes of tornadoes in the ACU MSRR site area. Please clarify the significance of the color variation in PSAR Figure 2.3-9 for illustrating events with the same magnitude and discuss whether updates to the figure and/or corresponding text in PSAR Section 2.3.1.6 are necessary. III. The NRC staff notes that operational use of the Enhanced Fujita (EF) scale of tornado intensity by the NWS began in February 2007. The EF-scale replaced the original Fujita scale of tornado intensity which was used since 1971. Most, if not all, of the markers (or bubbles) for tornado events in PSAR Figure 2.3-9 are shown to have occurred prior to 2007 and so would have been classified under the original Fujita scale. Please explain if any pre-2007 events had their Fujita-scale numerical ratings revised when transitioning to only the EF scale as presented in PSAR Figure 2.3-9. IV. PSAR Section 2.3.1.6 indicates that the year range of tornadoes considered in developing PSAR Figure 2.3-9 extends "between 1946 and 2014," however, the figure itself indicates a POR from	ACU modified PSAR section 2.3.1.6 and PSAR section 2.3 figures and references by letter dated July 30, 2024 (ML24219A258), to provide clarifying information as appropriate.

	1950 to 2010. Further, PSAR Section 2.3.1.6 highlights two EF3 and one EF2 tornado events that occurred presumably at the city of Abilene between “1950 and 2013,” but does not mention the two EF4 events included in Figure 2.3-9. Please clarify the POR and tornado events considered for the tornado risk analysis and discuss whether updates to Figure 2.3-9 (and/or Figure 2.3-10) and the corresponding text in PSAR Section 2.3.1.6 and Section 3.2.2.1 (which cross-references PSAR Section 2.3.1.6 to state that the largest tornado reported in Abilene was an EF3) are necessary. V. The staff notes that it is not clear if “bubbles” for the same year, but different distances, in PSAR Figure 2.3-9 correspond to individual tornado events or reports of the same tornado segments at different locations in the vicinity of Abilene over the POR, or if they are truly separate events that occurred during the same year. The staff notes that, in the former case, the frequency of occurrences of tornadoes may be overestimated. Please discuss.	
2.3-9	PSAR Section 2.3.1.6 discusses a variety of tornado characteristics around Abilene, Texas, beginning in 1950. PSAR Figure 2.3-10 illustrates tornado tracks in the vicinity of Abilene within a rectangular area extending about 20 miles to the north and south, about 23 miles to the east, and about 27 miles to the west. PSAR Section 2.3.1.6 states that tornado probabilities were calculated considering a 1-degree latitude/longitude square, which the NRC staff notes appears to be inconsistent with the size of the area illustrated in PSAR Figure 2.3-10. Please clarify this apparent discrepancy and confirm the size of	In information provided for audit, ACU clarified that PSAR figure 2.3-10 is not intended to illustrate the probability calculations in PSAR section 2.3.1.6, but only to show recorded tornado paths in and around the city of Abilene, Texas; information related to the probability calculations is given in PSAR section 2.3.1.6. In addition, ACU confirmed that data associated with PSAR reference 2.3-10 were obtained from the NWS Storm Predication Center. No PSAR changes or other additional docketed information were necessary in response

	the area used for probability calculations. Also, please confirm that data for PSAR Table 2.3-10, which are attributed to the source "Homeland Infrastructure Foundation-Level Data" (PSAR Reference 2.3-10), can be traced to the NWS Storm Predication Center or other reputable source.	to this question.
2.3-10	PSAR Section 2.3.1.6 summarizes the calculation of probabilistic areas and probabilities of site impact for EF2 through EF5 tornado events for various percentiles of tornado lengths and widths related to these intensities. The results are listed in PSAR Table 2.3-5. The NRC staff notes that there appears to be an error in the equation presented in PSAR Section 2.3.1.6 for the calculation of probability of tornado impact per century. As written, the terms for the "Area of Abilene Cell" appear to cancel out. Further, PSAR Section 2.3.1.6 states that "a value of 360 tornado days per century is used," and that "the expected number of tornadoes per century (360)" is applied to the calculation for tornadoes rated EF2 or stronger. Please confirm the equation and the application of the conditions described for this equation and discuss whether updates to PSAR Section 2.3.1.6 are necessary.	ACU modified PSAR section 2.3.1.6 by letter dated July 30, 2024 (ML24219A258), to provide clarifying information as appropriate.

Resolution of Questions on Section 2.4, "Hydrology"

Question Number	Question	Resolution
2.4-1	PSAR Section 13.1.8 states, "The site has not historically experienced flooding, is outside the 500-year flood plan, and is outside the inundation area for dam failures, so flooding is not foreseeably a credible external event (see	ACU modified PSAR section 2.4.1 by letter dated July 30, 2024 (ML24219A258), to clarify that any consequences from internal or external flooding

	Section 2.4). Flooding of the reactor pit with water will not pose a criticality concern, as stated in Section 4.5." PSAR Section 4.5.2.6 states "Flooding is a remote concern in the MSRR, but in the event that a tank of fuel salt located in the FHS is submerged in water, k_{eff} remains deeply subcritical." PSAR Section 2.4 states, "The risk of flood near the reactor is deemed negligible, whether from drainage runoff, ground movement, dam failure, or creek and river blockages." Based on the above statements, how is flooding (both internal and external) dispositioned with regards to safety impacts? The NRC staff understands based on PSAR Section 2.4 that ACU's position is that external flooding is precluded, but it appears from PSAR Section 4.5 that the consequences are addressed via bounding event analysis. Please clarify ACU's strategy to address potential MSRR flooding, and discuss and provide for audit relevant analyses if flooding is dispositioned analytically.	are bounded by MSRR event analyses, specifically the loss of normal electric power analysis in discussed in PSAR section 13.1.10 and the criticality analysis discussed in PSAR section 4.5.

Resolution of Questions on Section 2.5, "Geology, Seismology, and Geotechnical Engineering"

Question Number	Question	Resolution
2.5-1	PSAR Section 2.5.5 states that "it is estimated [Reference 2.5-9] that the PGA [peak ground acceleration] for the proposed site is 0.0305 gravity, as shown in [PSAR] Figure 2.5-22. Using the ASCE [American Society of Civil Engineering] 7 hazard tool, this value is within the predicted 0.039 PGA provided and	ACU modified PSAR sections 2.5, 2.5.2, 2.5.5, 3.1.1, 3.4.2.2, and 3.4.2.3, corrected the reference for PSAR figure 2.5-22, and also added PSAR figures 2.5-26, 3.4-4, and 3.4-5 and table 3.4-2 by letter dated July 30, 2024 (ML24219A258), to

	in the Seismic Design Category A [Reference 2.5-9]." PSAR Reference 2.5-9 is also stated as the source for PSAR Figure 2.5-22. PSAR Reference 2.5-9 is the ASCE 7 hazard tool. However, the staff notes that PSAR Figure 2.5-22, which shows seismic hazard curves along with the uniform hazard spectrum, does not appear to come from PSAR Reference 2.5-9. The staff notes that the graphs shown in Figure 2.5-22 of the PSAR appear to come from the USGS's unified hazard tool (earthquake.usgs.gov/hazards/interactive). The staff notes that, for the latitude and longitude of the SERC, the ASCE 7 hazard tool only provides hazard calculations for the boundary between B and C soil site classes (i.e., the B/C boundary), where the average site shear-wave velocity (V_s) equals 760 m/s. The values provided by the USGS's unified hazard tool and reported in the PSAR appear to be for a B/C boundary site classification, consistent with the ASCE 7-10 hazard tool. However, PSAR Section 3.4.2.3 states that "the SERC is supported on a foundation system on stiff competent soils. The site is classified as Site Class C prescribed in ASCE/SEI 7-10, Table 20.3-1. The typical shear wave velocities for the soils present at the site are 1,200 to 2,500 ft/sec." I. The ASCE 7-10 hazard tool reports site specific design ground motions using hazard estimates from the 2008 USGS Long-term Nation Seismic Hazard Mapping project. The staff notes, however, that in 2018 the USGS updated its hazard maps to reflect changes in hazard resulting from significant updates in source characterization and ground motion modeling approaches. Please explain and justify why the ASCE 7 hazard tool that uses the 2008 USGS	provide additional information, clarifications, and updates in response to this question as appropriate. This included clarification that the proposed MSRR site is Site Class C, clarification and update of information defining and justifying the seismic design of the MSRR and its safety-related foundations, and clarification of information related to the safe-shutdown earthquake (including application of RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 2) and use of design response spectra for the MSRR. Regarding item iii) of this question, ACU also indicated during audit discussions that it does not plan to conduct geophysical testing. The staff reviewed the N-values provided in appendix C of the Geotechnical Investigation Report that ACU provided for audit and confirmed that they are indicative of a Site Class C (which corresponds to an average shear-wave velocity range of 1,200 to 2500 feet/second) based on the correlations provided in table 20.3-1 of ASCE/SEI 7-10.

	hazard estimates is appropriate for reporting the seismic hazard at the site. II. Please clarify why ACU used the B/C boundary site classification to define the uniform hazard, given that the site classification is defined as site class C in PSAR Section 3.4.2.3. If the reported uniform hazard is corrected to site class C, will this affect the anchoring of the design spectrum to 0.03 gravity as discussed in PSAR Section 3.4.2.2? III. Was (or will be) any geophysical testing conducted to verify the shear-wave velocities for the soil and rock strata beneath the site for the determination of the site classification? If not, please justify ACU's estimation of the shear-wave velocities reported in the PSAR. If so, will these be provided in the operating license application?	
2.5-2	PSAR Section 2.5 states that "The seismic design basis reflects … data from a detailed geophysical and geotechnical investigation at the site." Please provide for audit the geotechnical site investigation report detailing relevant information, such as locations of all borings relative to the SERC, systems pit, and planned MSRR, respective soil boring profiles, field and laboratory testing performed, and results obtained that provide subsurface materials engineering and materials properties for the design and construction of the systems pit foundation system and MSRR. The staff notes that PSAR Section 2.5.2, states that 9 borings were drilled to depths of 5, 30, 40 and 60 ft. However, it is unclear where the bore holes are located in relation to the systems pit where the reactor would be located.	ACU provided the Geotechnical Investigation Report for audit, which provided relevant information including the locations of the boreholes relative to the systems pit. ACU also modified PSAR sections 2.5.2 and 3.1.1 and added PSAR figures 3.1-11 and 3.1-12 by letter dated July 30, 2024 (ML24219A258), to indicate that test borings B-8 and B-9, the 60 foot deep borings, are beneath the systems pit trench, and to clarify the location of the drilled piers relative to the systems pit. The staff also provided ACU with follow-up questions related to information in the Geotechnical Investigation Report in the scope of separate audit questions 2.5-9 through 2.5-12; those questions and their resolution are discussed below.

2.5-3	PSAR Section 3.1.1 states that deep drilled concrete pier foundations will be used with the foundation tip (toe) extending into the shale layer at a depth of about 55 feet below grade. The drilled piers are sized to have a maximum bearing pressure of 17 kilopounds per square foot (ksf). PSAR Figure 2.5-17, which summarizes borings at the SERC site, indicates that the site soil is highly plastic (plasticity index generally >17), with liquid limits as high as approximately 50 percent. The staff notes that design guides such as the Texas Department of Transportation Geotechnical Manual recommend characterizing geotechnical properties to a depth of 15 to 20 ft below the toe of drilled shafts. PSAR Section 2.5.2 states that prior to SERC construction, borings were drilled to depths of up to 60 ft, although the depth of most borings is 40 ft or less based on PSAR Figures 2.5-6 through 2.5-16. The staff also notes that the exact locations of the borings are not clear from the PSAR (see also audit question 2.5-2). Justify the adequacy of the limited number of borings used for defining and assessing factors that affect the MSRR deep foundation stability (such as soil bearing capacity and soil settlement), including the basis for characterizing geotechnical properties to a depth of 60 ft, which is only 5 feet below the drilled pier toe.	ACU modified PSAR sections 2.5, 2.5.2, and 3.1.1 by letter dated July 30, 2024 (ML24219A258), to provide additional information and clarification, including revising the 17 ksf bearing pressure to 40 ksf (consistent with information in the Geotechnical Investigation Report). The staff provided ACU with additional audit questions 2.5-9 through 2.5-12 as a follow-up to question 2.5-3 and others; based on the resolution of questions 2.5-9 through 2.5-12 including additional PSAR changes provided by ACU for those questions, the staff determined question 2.5-3 is resolved.
2.5-4	The staff notes that the PSAR does not appear to discuss potential foundation settlement. Please discuss what information was used or what assumptions were made about the geotechnical properties along and below the drilled piers to evaluate settlement of the systems pit and research bay. If assumptions were made about geotechnical properties, what are the bases for the assumptions?	ACU modified PSAR sections 2.5 and 2.5.2 by letter dated July 30, 2024 (ML24219A258), to state that the structural design of the systems pit ensures that the rigidity of slab combined with the rigidity of the piers is sufficient to avoid differential settlement. The staff provided ACU with additional audit questions 2.5-9 through 2.5-12 as a follow-up to question 2.5-4 and others; based on the resolution of questions 2.5-9 through 2.5-12 including additional PSAR changes provided by ACU for those questions, the staff determined question 2.5-4

		is resolved.
2.5-5	PSAR Section 3.1.1 states that expansive soils exist at the site. Please provide information regarding the location, extent, and expansion characteristics of such soil. The staff seeks this information to evaluate the foundation and subsurface materials stabilities under the design loading (static and dynamic) conditions.	ACU provided "Laboratory Compaction Characteristics of Soil Report" documents for onsite fill for audit. However, during audit discussions, ACU stated that most of the expansive native soils at the site were replaced by engineered (i.e., select) fills. ACU modified PSAR section 3.1.1 and added PSAR figure 3.1-10 by letter dated July 30, 2024 (ML24219A258), to state that native soil was removed and replaced by compacted select fill near the research bay slab and systems pit to alleviate concerns about expansive native clay soils, and illustrate the locations of select fill.
2.5-6	PSAR Section 2.5.7 states that the risk of liquefaction is low and that results of liquefaction hazard analyses based on site conditions and seismic design requirements will be provided in the Operating License application. PSAR Section 19.3.4.2, "Soils," states that "[b]ased on soil moisture content, the groundwater table is considered to exist below 35 ft (10.7 m) across the site." However, it is unclear to the staff whether sufficient geotechnical investigations (such as Standard Penetration Test (SPT), Core Penetration Test (CPT), or shear wave velocities) have been conducted to support a deterministic liquefaction analysis. If an evaluation of the soil at the site to liquefaction has been performed, provide results of the analysis (e.g., in an electronic portal) demonstrating adequacy of mitigating measures. Otherwise justify why liquefaction of the soil at the MSRR site is not possible and why an appropriate analysis of liquefaction susceptibility is not necessary as part of the construction permit application. The staff notes that liquefaction susceptibility and triggering evaluations are	ACU modified PSAR section 2.5.7 by letter dated July 30, 2024 (ML24219A258), to state that standard penetration tests were performed to evaluate the possibility of liquefaction, and to remove information stating the further analysis of liquefaction would be provided in an operating license (OL) application (given the low liquefaction susceptibility). The staff provided ACU with additional audit questions 2.5-9 through 2.5-12 as a follow-up to question 2.5-6 and others; based on the resolution of questions 2.5-9 through 2.5-12 including additional PSAR changes provided by ACU for those questions, the staff determined question 2.5-6 is resolved.

	an important factor in siting evaluation/selection prior to construction.	
2.5-7	PSAR Section 3.4.2.3 states that the typical shear wave velocities for the soils present at the site are 1,200 to 2,500 ft/sec. However, the PSAR does not appear to include any information about in-situ geophysical surveys determining the profiles of shear wave velocities. The staff notes that shear wave velocities are important soil parameters for Soil-Structure Interaction (SSI) analysis. It is unclear whether any of the in-situ geophysical surveys that may be necessary for SSI have been carried out. Additionally, the PSAR does not appear to include fundamental information on other soil parameters such as minimum soil unit weight and Poisson's ratio (that the staff notes would be necessary for SSI analysis and foundation stability assessment), nor on how these soil parameters would be determined, if necessary. Please provide information on the soil parameters applicable for foundation design/SSI analysis in the PSAR. Alternatively, justify why such parameters are not needed for the foundation design/SSI analysis of the research bay and MSRR systems pit foundation system.	During audit discussions and in information provided for audit, ACU stated that it used site-measured N-values to determine the MSRR site classification (Site Class C), and the shear wave velocities cited in PSAR section 3.4.2.3 are based on that classification. ACU confirmed that relevant SSI analyses for MSRR SSCs will be provided in an OL application, and the analyses will capture uncertainties in soil properties as appropriate. No question-specific PSAR changes or other additional docketed information were necessary for the resolution of this question. PSAR section 3.4.2.3 (not revised in response to this question) states that “[t]he analysis of the MSRR facility to the SSE [safe shutdown earthquake] includes the effects of [SSI].”
2.5-8	PSAR Figure 3.1-5 shows “Select Fill” placed after placement of the concrete systems pit. However, the PSAR does not appear to include any information regarding the “Select Fill.” Define what soil backfill material constitutes the “Select Fill.” Clarify whether native soils are acceptable as “Select Fill.” Confirm that compaction requirements of PSAR Section 3.1.1 are followed for the provided “Select Fill” material. If not, provide actual compaction of the material.	ACU provided “Laboratory Compaction Characteristics of Soil Report” documents for select fill for audit. In information provided for audit, ACU also noted that the select fill is imported from offsite and confirmed the select fill meets the compaction requirements of PSAR section 3.1.1. In addition, ACU modified PSAR section 3.1.1 and added PSAR figure 3.1-10 by letter dated July 30, 2024 (ML24219A258), to state that native soil was removed and replaced by compacted select fill near the research bay slab and systems pit to alleviate

		concerns about expansive native clay soils, illustrate the locations of select fill, and also indicate that calculations for structural support were based on the piers only and not on support from the soil. The staff reviewed the "Laboratory Compaction Characteristics of Soil Report" documents for select fill that ACU provided for audit and confirmed that samples of the select fill materials were tested for moisture-density relation, Atterberg limits, and moisture content.
2.5-9	On April 11, 2023, in response to Audit Question 2.5-2 (ML23086A017) and others provided by the NRC staff as part of the ACU PSAR Chapter 2 and 3 Audit (ML23065A048), ACU provided its "Geotechnical Investigation Report" for the MSRR for NRC staff audit. Part 1.2 of the geotechnical investigation report indicates that the "report is provided for general information only" and that the "owner or architect neither guarantee nor accept any responsibility for soil investigation data." a. Please discuss how ACU concluded that the geologic and geotechnical conditions of the MSRR site, and associated uncertainties, have been appropriately determined and considered in light of the referenced statements in the "Geotechnical Investigation Report." b. PSAR Section 12.9 states that the Quality Assurance Program Description for the design and construction of the MSRR was submitted as a topical report that was approved by the NRC (ML22293B802). However, PSAR Chapter 2 does not appear to include information about a quality assurance program for the site investigation. Please clarify what quality assurance plan ACU used for the site investigation, and explain and	In information provided for audit, ACU stated that the quoted statement from the Geotechnical Investigation Report is standard language in Parkhill specifications that acknowledges that the report is providing a representative sample of existing soil conditions only in the areas investigated. In information provided for audit and during audit discussions, ACU also provided information on quality assurance and quality control practices during the geotechnical investigation. ACU modified PSAR section 2.5.2 by letter dated July 30, 2024 (ML24219A258), to provide additional information. In addition, the staff sent ACU RCIs 2.5-1 and 2.5-2 related to information provided for audit and discussed in the scope of this question, and ACU confirmed the information by letter dated April 18, 2024 (ML24109A203).

	justify how ACU determined that there was adequate quality assurance for the site investigation.	
2.5-10	On April 11, 2023, in response to Audit Question 2.5-3 (ML23086A017) and others provided by the NRC staff as part of the ACU PSAR Chapter 2 and 3 Audit, ACU provided its "Geotechnical Investigation Report" for NRC staff audit. On June 5 and August 10, 2023, ACU also provided a written response and a draft PSAR revision (including draft revisions to PSAR Section 2.5.5), respectively, for audit in response to Audit Question 2.5-3. However, ACU did not appear to include information justifying the adequacy of the borings used for defining and assessing factors that affect the MSRR pit foundation system stability, specifically, the limited depths of the borings. In addition, in the draft PSAR Section 2.5.5 revision provided for audit, ACU updated the maximum allowable bearing pressure to be 40 kilopounds per square foot (ksf) instead of 17 ksf. but the information provided for audit including the "Geotechnical Investigation Report" does not appear to provide further details or discussion on the parameters and methodology used for determining the maximum available bearing pressure to be 40 ksf. The staff notes that the geotechnical conditions of subsurface materials underlying the site will influence the stability of the MSRR pit structures. a. Justify the adequacy of boring depths for defining and assessing factors that potentially affect the MSRR pit foundation system stability (such as soil bearing capacity and settlement), including the basis for characterizing geotechnical properties to a depth of 60 feet, which is only 5 feet below the drilled pier toe, as described in the "Geotechnical	ACU modified PSAR sections 2.5.2 and 3.1.1 (and made minor conforming changes to PSAR chapter 19), and added PSAR tables 2.5-2 and 2.5-3, by letter dated July 30, 2024 (ML24219A258), to provide additional information and clarifications including: • Addition of relevant information from the Geotechnical Investigation Report. • Discussion of how Texas Department of Transportation (TXDOT) Geotechnical Manual guidance was used. • Discussion of basis for the 40 ksf bearing pressure and skin friction values, associated FOS valves, and how these values are conservative (e.g., compared to allowable values derived from the geotechnical investigation and TXDOT guidance). • Clarification that although seasonal perched groundwater was found in some site borings, groundwater was not a factor during SERC construction and the groundwater table is at least 60 feet below grade. • Information stating that although test boring data does not show concerns from seams of weakness or weathered rock in shales underlying the site, possible uncertainties in rock quality were taken into account in establishing allowable bearing capacities. • Information on estimated loading on the systems pit piers. In addition, ACU provided information for audit including additional information on determination of bearing capacities and application of TXDOT

		Investigation Report" provided for audit. Specifically, discuss how ACU concluded that the borings, particularly B-8 and B-9, were adequate to characterize the soil strata below the MSRR pit foundation system, such that no additional site investigation was necessary to identify if layers of weaker material may exist below the MSRR pit foundation system that potentially could affect the soil bearing capacity, structural integrity, performance, and intended function of the MSRR pit drilled piers. b. The "Geotechnical Investigation Report" provided for audit reports the recommended allowable bearing capacities. However, the NRC staff notes that the "Geotechnical Investigation Report" does not appear to adequately characterize the rock mass conditions needed to evaluate the rock mass bearing capacity. In discussions of Audit Question 2.5-2 and others on May 17-18, 2023, as part of the ACU PSAR Chapter 2 and 3 Audit, ACU stated that it will revise the PSAR to explain how the MSRR pit drilled piers are designed to bear loads considering adequate margin of safety for rock condition variations. However, the NRC staff notes that the draft PSAR Chapter 2 revisions provided for audit in response to Audit Question 2.5-2 and others on August 10 and 14, 2023, do not appear to include the stated revisions. Please provide information to demonstrate that the geotechnical conditions and associated uncertainties of the soil and rock mass conditions encountered through the depth of influence have been adequately investigated, determined, and considered, as appropriate. For example, please discuss as appropriate:	guidance, and on borings at other sites near the proposed MSRR site. The staff confirmed that the information on borings at other sites indicates that stratigraphic information from the proposed MSRR site borings is consistent with other nearby borings; therefore, the information on nearby borings helps demonstrate the adequacy of the borings at the proposed MSRR site. ACU also provided for audit a pier loading calculation that indicates a load on each systems pit pier of about 1,537 kips. ACU noted that this is larger than the value of 1,473 kips cited in PSAR section 3.1.1 (which was a preliminary value), but it is still well below the systems pit pier capacity of about 2,199 kips given in PSAR figure 3.1-12.

	1) Rock mass condition consideration. (The NRC staff notes that in general, bearing capacities must be reduced where the rock is weathered, fractured, or is non-homogeneous and contains seams of weak and decomposed rock.) 2) Sufficient minimum depth of exploration. (The NRC staff notes that common engineering practice provide guidance for minimum depth of exploration. For example, where appropriate, exploration should be conducted at least to a depth where stress increase due to estimated foundation load is less than 10 percent of the effective overburden stress at that depth, and/or a minimum of 10 feet below the pier tip elevation, in order to determine the geotechnical characteristics of rock/soil within the zone of foundation influence.) c. In discussions of Audit Question 2.5-2 and others on May 17-18, 2023, as part of the ACU PSAR Chapter 2 and 3 Audit, ACU stated that it will revise the PSAR to include a discussion of steps of how soil bearing capacities were obtained and why the margin of safety is adequate. However, the NRC staff notes that the draft PSAR Chapter 2 revisions provided for audit in response to Audit Question 2.5-2 and others on August 10 and 14, 2023, do not appear to include the stated revisions (as discussed above, ACU provided a draft PSAR Section 2.5.5 revision for audit which updates the maximum allowable bearing pressure to be 40 ksf, but does not provide a clear basis for	

	this updated value). The NRC staff notes that ACU's As-Built Design Drawing S-001, provided for audit on May 5, 2023, reports the MSRR pit foundation system drilled pier soil allowable bearing capacity to be 40 ksf, with allowable skin frictions ranging between 3 and 5 ksf depending on the depth of piers. 1) Please justify how the reported bearing capacity (40 ksf) in the draft PSAR Section 2.5.5 revision provided for audit was determined. In addition, please discuss how ACU determined the pier skin frictions values for the MSRR site and confirm they are consistent with those reported in As-Built Design Drawing S-001. As appropriate, provide the factor of safety (FOS) applied from the ultimate capacities to the allowable capacities. 2) The NRC staff also notes that the existence of subterranean water, as noted in PSAR Sections 19.3.4.2 and 19.3.5.2, may challenge bearing capacity and skin friction values. Clarify whether the bearing capacity and pier skin friction values determined for the MSRR site took into consideration the existence of below grade water for the piers (see also Audit Question 2.5-12. B).	
2.5-11	In response to NRC staff questions in the PSAR Chapter 2 and 3 Audit, ACU provided a draft PSAR Section 2.5 revision for audit indicating that local building codes, which are based on ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures," were followed for the SERC building that will house the reactor, including site	ACU modified PSAR section 2.5.2 by letter dated July 30, 2024 (ML24219A258), to provide discussion related to lateral earth pressures. ACU also modified the PSAR to include additional information related to pier bearing capacity and skin friction, including conservatisms in the values used, as discussed

	investigation and preparation. In specific response to Audit Question 2.5-4 (ML23086A017), ACU provided a draft PSAR Section 2.5.2 revision for audit which states that the structural design of the SERC provides sufficient slab and pier rigidity to avoid differential settlement. However, based on information in the PSAR and information provided for audit, it is not fully clear to the staff how ACU analyzed potential foundation settlement and other potential geologic hazards consistent with engineering common practice and standards (e.g., ASCE 7-10). Consistent with ASCE 7-10, please discuss, as appropriate, how ACU considered the following in its evaluation of the MSRR site: total and differential settlement (the draft PSAR Section 2.5.2 revision referenced above mentions differential settlement but does not provide details of any analysis), lateral earth pressures (at rest, passive, and active) at the MSRR pit structures, pier skin friction capacity, and drilled pier end bearing capacity. Please include, as appropriate, information summarizing the methodologies and selected geotechnical engineering properties and parameters used for analyses, with the basis of your assumptions; as well as the results of the foundation stability analyses.	under the resolution of question 2.5-10. ACU also modified PSAR section 3.1.1 by letter dated July 30, 2024 (ML24219A258), to state that the TXDOT method takes excessive settlement into account. The staff expects settlement to be small because the drilled piers are in competent shale, and because of the rigidity of the slab combined with the rigidity of the drilled piers.
2.5-12	In response to Audit Question 2.5-6 (ML23086A017) in ACU PSAR Chapter 2 and 3 Audit, during audit discussions on May 17-18, 2023, and in a draft Audit Question 2.5-6 response provided for audit on June 5, 2023, ACU indicated that the liquefaction analysis for the MSRR site is complete and therefore it would remove information from the PSAR deferring a liquefaction analysis to a MSRR operating license application, and add detail to the PSAR referencing the “Geotechnical Investigation Report” (provided for audit on April 11, 2023) with respect to liquefaction analysis. However, the	ACU modified PSAR sections 2.5.2 and 2.5.7 (and made minor conforming changes to PSAR chapter 19) by letter dated July 30, 2024 (ML24219A258), to provide additional information discussing why the proposed MSRR site is not susceptible to liquefaction, and clarifying the depth of the groundwater table beneath the surface. In information provided for audit, ACU clarified that the pump shown on drawing S-501 is associated with the SERC elevator and is not intended to

	NRC staff notes that the "Geotechnical Investigation Report" does not appear to contain information about the liquefaction potential analysis (i.e., method of analysis, calculations, soil properties used, etc.). In a draft revised PSAR Section 2.5.7 provided for audit on August 10, 2023, in response to Audit Question 2.5-6, ACU indicates that ASCE 7-10 requirements were followed to evaluate possibility for liquefaction. ORNL/TM-2020/1478, Part 1, Section 2.5.7 states: The applicant should discuss soil structure. If the foundation materials at the site adjacent to and under safety-related structures are saturated soils or soils that have a potential for becoming saturated, the applicant should prepare an appropriate state-of-the-art analysis of the potential for liquefaction at the site. The applicant should also determine the method of analysis based on actual site conditions, the properties of the reactor facilities, and the earthquake and seismic design requirement for the protection of the public. a. In PSAR Section 2.5.7, ACU indicates that the soil at the MSRR site is not typically saturated, and that the soil at the site is primarily silty clays; therefore, the risk of liquefaction is low (this portion of PSAR Section 2.5.7 is unchanged in the draft revised PSAR Section 2.5.7 provided for audit on August 10, 2023). However, the NRC staff notes that some silts and clays can be subject to liquefaction, and that liquefaction susceptibility criteria are commonly used to determine liquefaction susceptibility for silts and clays based on engineering common practice. Please discuss the liquefaction susceptibility criteria ACU used to determine that there is no need to perform a state-of-the-art analysis for	manage building water infiltration. Regarding part c. of this question, the staff determined that additional information regarding the chemical composition of groundwater is not needed because the groundwater table is below the bottom of the drilled piers.

	potential soil liquefaction, and/or provide information on how the ASCE 7-10 requirements were followed to evaluate the liquefaction potential as indicated in the draft revised PSAR Section 2.5.7 provided for audit on August 10, 2023. b. PSAR Sections 2.5.2, 19.3.4.2, and 19.3.5.2, and the "Geotechnical Investigation Report" provided for audit indicate that groundwater exists at 9 to 11 feet below grade in the majority of the boreholes which extend to a depth below this depth. In addition, the NRC staff notes that ACU construction drawing S-501, provided for audit on May 5, 2023, shows the existence of a sump pump at 6 feet below grade, which suggest potential concerns regarding water infiltration in SERC areas below grade, including the MSRR pit. PSAR Section 2.5.2 also states that "[t]he groundwater table is estimated to exist at depths greater than 35 ft (11 m) below current grades." (The referenced portions of PSAR Section 2.5.2 are unchanged in the draft PSAR Section 2.5 revisions provided for audit on August 10 and 14, 2023). PSAR Section 19.3.4.2 states that based on soil moisture content, the groundwater table is considered to exist below 35 feet across the site. However, given other information in the PSAR and provided for audit, it is not clear to the NRC staff how ACU concluded that the groundwater table is at depths greater than 35 feet below grade. In the draft revised PSAR Section 2.5.7 provided for audit on August 10, 2023, in response to Audit Question 2.5-6, ACU indicated that following drilling for the foundation piers, there was minimal observable water in only one	

	hole, but this draft PSAR revision does not provide additional information on the groundwater table. The response to Audit Question 2.5-6 that ACU provided for audit on June 5, 2023, also did not include information about the groundwater table. Please clarify the location and depth of the groundwater table at the site, and justify how ACU concluded that the soil is not typically saturated (see PSAR Section 2.5.7) above and below the drilled pier foundation level when most of the boreholes show existence of groundwater between 9 to 11 feet. c. PSAR Figure 2.5-17 indicates that the MSRR site soil, in general, has a plasticity index (PI) > 17 and a liquid limit between 32 and 47 percent. PSAR Sections 19.3.5.2 and 19.3.4.2 note the existence of groundwater either below 35 feet below grade and/or at 9 to 11 feet during the rainy season. However, neither the PSAR nor the "Geotechnical Investigation Report" provided for audit appear to discuss the chemical properties or condition of the ground water as they pertain to assessment of the degradation potential for the foundational piers. As appropriate, please provide information regarding ground water conditions and evaluation of chemical properties of the groundwater such as sulphate, chloride and pH that may potentially impact long-term behavior of foundation drilled piers (concrete and reinforcing steel).	

Resolution of Questions on Chapter 3, "Design of Structures, Systems, and Components" (General Questions Relevant to Entire Chapter)

Question Number	Question	Resolution
3-1	The regulation 10 CFR 50.34(a)(4) requires that the preliminary analysis of the design of a facility include "…determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility…". PSAR Section 3.4.2.1 states that "[b]y designing the SSCs in accordance with RG 1.29 to withstand the effects of [a safe-shutdown earthquake], a designed-in safety margin is provided for bringing the reactor to a safe shutdown condition…". However, it is not clear to the staff whether ACU is committing to design the MSRR SSCs fully consistent with RG 1.29 (RG 1.29, Revision 6 (ADAMS Accession No. ML21155A003), or another version). It is also not clear to the staff what the "designed-in safety margin" for seismic hazards is for the MSRR, including whether it relates to performance requirements, such as margins ensuring structural integrity and adequacy of safety-significant or safety-related facility SSCs. Furthermore, it is not clear to the staff what SSC design margins are for hazards other than seismic hazards (e.g., meteorological and flood hazards). Please discuss the implementation of RG 1.29 in the design of MSRR SSCs. In addition, please discuss and provide in an electronic reading room for audit any relevant reports/calculations for review that would allow the staff to better understand information on construction materials and design methodologies used, and resulting margins associated with the design of SSCs against meteorological, flood/water, and seismic damage, and against damage from other abnormal loads (if any).	ACU modified PSAR chapter 3 by letter dated July 30, 2024 (ML24219A258), to clarify the seismic design methodology including clarifying PSAR section 3.4.2.1 to indicate that design of SSCs will be informed by RG 1.29, "Seismic Design Classification for Nuclear Power Plants," Revision 6, clarifying the seismic classification of SSCs in PSAR table 3.4-1, and clarifying other seismic design information in PSAR section 3.4.2.1. The staff expects that additional information and/or analyses related to seismic design of SSCs would need to be provided in an OL application. In information provided for audit, ACU confirmed that consistent with discussion in PSAR section 3.3.1.1.1 (which was not revised in response to this audit question), flooding of the systems pit would not prevent safe shutdown of the reactor or long-term passive decay heat removal. ACU also confirmed that protection from meteorological hazards will be primarily through the subterranean placement of the MSRR beneath precast concrete panels.

3-2	PSAR Figure 3.1-3 shows a rendering of the MSRR systems pit with the shielded reactor, fuel storage, and coolant salt and heat management enclosures. It also shows that these extend above the MSRR pit and into the research bay. PSAR Figure 4.1-2 also shows that the concrete reactor enclosure extends above the pit and into the bay. However, PSAR Figure 3.4-1 shows the pit covered with 16 inch precast panels. It is not clear to the staff whether all of the MSRR hardware, enclosures, and safety-related SSCs are subterranean (i.e., below the precast panels), for protection against abnormal loads, such as heavy lift drops or tornados. Clarify whether all the MSRR SSCs are subterranean, located in the pit, and covered with the aforementioned precast panels sized against all loading conditions to preclude damage. Provide relevant additional construction drawings and calculations for audit, as appropriate.	During audit discussions and in information provided for audit, ACU clarified that some PSAR figures are approximate and not intended to represent an exact configuration, but confirmed that PSAR figure 3.4-1 is accurate. ACU also confirmed that all safety-related SSCs are contained within the reactor pit (below the precast panels). No question-specific PSAR changes or other additional docketed information were necessary for the resolution of this question.
3-3	PSAR Section 2.2.2.7.3 states that the "probability for an airplane crash into the SERC research bay is 3.6×10^{-6} per year. The MSRR and all structures, systems, and components are positioned inside a subterranean concrete vault within the SERC research bay with an above-grade target size of approximately 3 ft by 30 ft by 18 ft (1m by 10m by 6m), which results in a probability for an airplane crash into the MSRR facility of 8.2×10^{-7} per year." The staff notes, however, that DOE-STD-3014-2006, "Accident Analysis for Aircraft Crash into Hazardous Facilities," which is referenced in PSAR Section 2.2.2 and outlines methodologies to evaluate aircraft crash into hazardous (e.g., nuclear) facilities, also discusses secondary effects such as fuel fires that could spread beyond the area directly damaged by the crash. DOE-STD-3014-2006 also states, "[d]ue to the difficulty of demonstrating that active systems can function following a crash, credit should not be allowed for fire suppression systems unless an explicit analysis shows	During audit discussions and in information provided for audit, ACU stated that the reactor cell has louvers that would close in an accident scenario or loss of power, cutting off air supply to a potential fire. ACU also provided air crash hazard calculations (supporting the results provided in the PSAR) for audit, which the staff noted generally applied methodologies in DOE-STD-3014-2006. Given conservatisms in the DOE-STD-3014-2006 methodology (for example, its use of aircraft crash rates that are over 30 years old and do not reflect improvements in aircraft safety since that time), and other factors such as the robust design of the subterranean systems pit including its precast covers, the staff determined that potential fuel fires due to an aircraft impact do not pose a significant hazard to the MSRR.

	that they will remain effective." Although PSAR Figure 3.4-1 shows precast slabs covering the MSRR pit, the staff notes that PSAR Figure 3.4-1 and other information in the PSAR does not appear to describe any fire stops or other mitigating measures that would preclude burning fuel from entering into the MSRR pit. Discuss whether spreading of aircraft fuel fires resulting from crash have been considered in the design of the MSRR and pit, and if they are not considered, justify the absence of such consideration.	No question-specific PSAR changes or other additional docketed information were necessary for the resolution of this question.
3-4	PSAR Section 4.2.5.3 states that the "reactor enclosure support system is rigid and transmits force to the systems pit wall. The systems pit wall has notches or ledges (not shown in the figure) which serve as anchoring points for the reactor enclosure support structure." However, it is not clear to the staff what figure is being referred to in this portion of the PSAR. The staff also notes that PSAR Figure 3.1-3 does not show the reactor enclosure support system being attached and transmitting force to the systems pit wall. It is not clear to the staff whether this interaction between the reactor enclosure support system and systems pit wall has been considered in the structural analysis including potential seismic loading of the systems pit. Please discuss whether this interaction is considered in the design.	In audit discussions and in information provided for audit, ACU stated that the figure reference was an error and should not have been included in the PSAR. ACU also confirmed that the reactor enclosure will be designed to handle seismic loading of the systems pit. However, ACU clarified that the reactor will be mounted on the systems pit floor, not hung on the wall. ACU modified PSAR sections 4.2.5.2 and 4.2.5.3 by letter dated July 30, 2024 (ML24219A258), to remove the erroneous reference and provided updated information on the support for the reactor enclosure and associated SSCs.
3-5	PSAR Section 13.1.9 discusses a potential malfunction of the research bay overhead crane (a 40-ton crane according to PSAR Section 16.1) that potentially could drop a heavy object onto the external concrete shield of the reactor. PSAR Section 13.1.9 states that the "concrete [reactor enclosure] barrier is able to withstand the weight and impact of any object that would foreseeably be moved with the overhead crane." However, the PSAR does not appear to discuss the potential for dropping of a heavy load on the fuel storage or coolant salt and heat	In audit discussions and in information provided for audit, ACU confirmed that the systems pit precast panels will cover not just the reactor, but all safety-related SSCs; information related to concrete panel strength will be specified in an OL application. In addition, ACU modified PSAR section 9.8.2 by letter dated July 30, 2024 (ML24219A258), to add information on administrative procedures for safe crane use as well as information on concrete

	management enclosures. Please discuss how the fuel storage and coolant salt, heat management enclosures, and any safety-related SSCs and non-safety-related SSCs requiring functionality will be protected from potential dropping of heavy loads, and as appropriate, what controls ACU will use for lifting of heavy loads.	panels protecting safety-related SSCs.
3-6	The regulation 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "[i]nformation relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety." The regulation 10 CFR 50.34(a)(4) requires, in part, that the PSAR include "[a] preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents." The regulation 10 CFR 50.34(a)(7) requires, in part, that the PSAR include "[a] description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility." ORNL/TM-2020/1478 (ML20219A771), Part 1, Section 12.9, states that guidance in ANSI/ANS-15.8-1995, "Quality Assurance Program Requirements for Research Reactors," and NRC Regulatory Guide (RG) 2.5, Revision	ACU modified PSAR section 3.1.1 by letter dated July 30, 2024 (ML24219A258), to add information on quality assurance and quality control for the design and construction of the pre-existing SERC structures, including the systems pit that will safety-related. ACU also provided for audit the Linbeck "Quality Assurance/Quality Control Program," October 2022 revision, and the staff confirmed that this document includes provisions that are consistent with the Linbeck QA/QC processes discussed in the modified PSAR section 3.1.1 as well as other general good practices for ensuring quality including practices similar to ANSI/ANS-15.8-1995, Section 2, recommendations. In addition, ACU provided for audit completed checklists used by Linbeck to document the inspection of key items to ensure the quality of the SERC systems pit foundation and supporting piers, such as verification of pier rebar size and layout and inspection and measurement of holes drilled for the piers; and inspection reports prepared by Terracon, which performed third-party inspections of the drilled pier holes. The staff noted that these example documents also help confirm ACU's and its contractors' application of QA/QC practices consistent with information in the PSAR.

	1, “Quality Assurance Program Requirements for Research and Test Reactors” (ML093520099), which references ANSI/ANS-15.8-1995, provide an acceptable method for complying with quality assurance program requirements for research reactors. ACU MSRR PSAR, Revision 1 (ML23319A094), Section 3.1.2, PDC 1, “Quality standards and records,” states: The safety related SSCs shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program consistent with ANS 15.8 shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of safety related SSCs shall be maintained by or under the control of ACU (licensee) throughout the life of the MSRR facility. ANSI/ANS-15.8-1995 (R2018) provides guidance on quality assurance programs for research reactors. It recommends control provisions to ensure that design requirements are correctly incorporated into the facility design and construction. American Concrete Institute (ACI) 349-13, “Code Requirements for Nuclear Safety-Related Concrete	In information provided for audit, ACU also clarified that RG 1.142 only informed the design insofar as it pointed to the use of ACI 349-13 as a standard for the research bay foundations.

	Structures and Commentary," states that it "provides [the] minimum requirements for design and construction of nuclear safety-related concrete structures and structural members for nuclear facilities," where protection against potential radioactive releases is a concern. ACI 349-13 specifically addresses the need to follow these requirements and states that "quality assurance program[s] covering nuclear safety-related structures shall be developed … before starting any work." Although not applicable to research reactors such as the MSRR, NRC RG 1.142, Revision 3, "Safety-Related Concrete Structures for Nuclear Power Plants" (ML20141L613), endorses ACI 349-13 and adds conservatisms to the minimum requirements of ACI 349-13 for the design and construction of safety-related concrete nuclear structures. RG 1.142 addresses standards for quality assurance in its Regulatory Position 1. The PSAR, Revision 1, Section 3.1.1, states that the "[SERC] research bay systems pit is on a deep drilled concrete pier foundation with the drilled piers extending into the shales…". The PSAR, Revision 1, Section 1.8 states that "the [SERC] research bay floor and subterranean systems pit concrete structure was designed to meet ACI 349-2103 [*sic*]." On June 20, 2023, ACU provided a written response for audit in response to Audit Question 3.3-1 (ML23086A017) provided by the NRC staff as part of the ACU PSAR Chapter 2 and 3 Audit (ML23065A048), stating that the systems pit was designed to ACI 349-2013 as "informed by Regulatory Guide 1.142, but the guide is not strictly incorporated in the entirety of the design." The NRC staff notes that ACU's as-built facility design drawings, provided for audit on May 5, 2023, indicate that ACI 349-13 is the code of reference for the design and construction of the	

	SERC research bay concrete structure. However, it is not fully clear to the staff how the ACI 349-13 requirements were implemented in their entirety in the construction of the safety-related portions of the SERC research bay, MSRR systems pit, and its foundation slab structure including its drilled piers (system), and how the design and construction may have been informed by RG 1.142, particularly with regard to quality assurance. The staff notes that quality assurance provisions throughout all phases of building design and construction are mandated by codes and industry accepted standards. The staff notes that this is because, for example, code designed drilled pier capacities can be significantly affected by quality of their construction, existing subsurface conditions, excavation, reinforcing steel installation, and concrete placement. It is also not clear to the staff whether the safety-related portions of the SERC, including the MSRR systems pit and its foundation slab structure including the drilled piers, may have been designed and constructed applying any other standards and/or specifications relevant to quality assurance that would minimize the inherent risks for defects during construction. The PSAR, Revision 1, Section 12.9, states that the Quality Assurance Program Description, based on ANSI/ANS-15.8-1995 (R2018), for the design and construction of the MSRR was submitted as a topical report that was approved by the NRC (ML22293B802). However, PSAR Revision 1, Chapter 3, does not appear to include information about a quality assurance program for structural design and construction of safety-related portions of the SERC research bay, MSRR systems pit, and its foundation slab structure including its drilled piers. a. Discuss how ACU implemented PDC 1 as given in PSAR Revision 1 with respect to structural design and construction of the as built safety-related	

	portions of the SERC research bay, MSRR systems pit, and its foundation slab structure including its drilled piers. b. Discuss how ACU determined that there was adequate quality assurance to meet PSAR Revision 1 referenced codes and industry-accepted standards at each phase of the structural design and construction of the safety-related portions of the SERC research bay, MSRR systems pit, and its foundation slab structure including the drilled piers. Consider in the discussion whether project specifications for the designed drilled piers included procedures, materials, and performance criteria that would potentially minimize inherent risks for defects during their construction; the extent of the implemented quality assurance program(s) to cover pier drilled holes, subsequent placing of concrete and reinforcing steel, and follow-up inspection to ensure that the inherent risks of defects associated with construction of the drilled piers were minimized; and whether any post-construction testing was performed to help provide assurance that the MSRR systems pit foundation slab structure including its drilled piers could adequately perform its intended function through the proposed 20-year operating license period and subsequent decommissioning. c. Clarify how RG 1.142, Revision 3 (or another revision), informed the design and construction of the safety-related portions of the SERC research bay including the MSRR systems pit and its foundation structure including the drilled piers. Specifically, describe what information contained in	

	RG 1.142, Revision 3 (or another revision), was considered in the analysis, design, construction, testing, and evaluation of the ACU multi-use SERC building that includes the research bay, MSRR systems pit, and its foundation structure including its drilled piers.	

Resolution of Questions on Section 3.1, "Design Criteria"

Question Number	Question	Resolution
3.1-1	Principal design criterion (PDC) 32 uses the term "important areas and features" to describe the scope of functions covered by inspection. Please provide additional context regarding what "important" means as applied here (e.g., important to safety or something else).	ACU modified PSAR section 3.1.2.4 by letter dated July 30, 2024 (ML24219A258), to revise the PDC 32 basis to clarify that that scope of the periodic inspection, functional testing, and material surveillance will be evaluated at the design finalization.
3.1-2	PSAR Section 3.1 does not appear to provide any PDC for fuel inventory maintenance. This function appears to be relevant to the design and fuel inventory and distributions throughout the reactor system are assumed quantities in Section 4.2.1.1 of the PSAR (though the PSAR states that values will be provided in further detail in an OL application). Please discuss whether an additional appropriate PDC should be implemented. If such a PDC is not necessary (whether not needed at all or captured under a different PDC), provide a justification why a PDC is not implemented.	In information provided for audit, ACU clarified that other PDC include fuel inventory maintenance. Specifically, the addition of uranium tetrafluoride (UF_4) is addressed in PDC 10,11, and 12. Additionally, PDC 71 addresses fuel salt composition control. No question-specific PSAR changes or other additional docketed information were necessary for the resolution of this question.
3.1-3	The staff notes that PSAR Section 3.1 does not appear to provide any PDC related to the MSRR graphite. It is not clear to the staff based on the content of the PSAR whether the graphite has functions important to safety in the design. Please provide context regarding the function(s) of the graphite and discuss why a PDC is not required or whether additional PDC(s) may be appropriate to cover any functions important to safety.	Because the graphite is non-safety related and cannot impair a safety function (see table 3.4-1 in the revised PSAR submitted by letter dated July 30, 2024 (ML24219A258)), the staff determined that no response to this question was necessary.

3.1-4	PDCs 71 and 73 use the language “ability of the fuel salt to perform its safety functions” and “could prevent accomplishing a safety function.” Because these PDCs are novel, it is not entirely clear to the staff from the PSAR which safety functions are covered by these PDCs. The NRC staff requests that ACU provide additional context regarding what functions are expected to be covered by these PDCs. Additionally, PDC 73 lists a number of possible causes of cover gas line plugging; is this list meant to be exclusive?	ACU modified PSAR sections 4.2.1.2, 4.2.1.6, and 4.2.2.1 by letter dated July 30, 2024 (ML24219A258), to clarify that PDC 71 is part of the design bases for the fuel salt and PDC 73 is part of the design bases for the reactor protection system. In information provided for audit, ACU also clarified that the list of possible causes of plugging of gas lines in PDC 73 is not all inclusive. ACU modified PSAR section 3.2.1.7 by letter dated July 30, 2024 (ML24219A258), to remove the list of possible causes of plugging.

Resolution of Questions on Section 3.2, “Meteorological Damage”

Question Number	Question	Resolution
3.2-1	PSAR Section 19.4.12.1.7.2 states that “[a]lthough damage to the reactor building is likely if it is in the direct path of a strong tornado, a reactor shutdown is standard protocol in the event of severe weather warnings, and no safety critical components are foreseeably affected by this occurrence.” PSAR Section 3.2.2.2 states that “[t]he systems pit is covered with precast concrete panels … to protect required safety-related SSCs from tornado missiles and debris from failure of the research bay structure.” However, the PSAR does not appear to clearly indicate whether the pit precast panels are designed to resist tornado-generated high wind speeds, atmospheric pressure changes, and missile impacts (see PSAR Section 3.2.2), to protect against and mitigate subterranean MSRR SSC damage. Clarify whether and how the panels are designed to resist such loads, and if	In information provided for audit, ACU confirmed that the precast panels will be flush with the floor of the SERC bay area, and designed to stay in place under all tornado loads, missile impacts, and atmospheric pressure changes; additional analysis will be included in an OL application. PSAR section 3.2.2.2 (which was not revised in response to this audit question) states that the systems pit is covered with precast concrete panels to protect required safety-related SSCs from tornado missiles and debris from potential failure of the research bay structure, and that final design details will be provided in an OL application.

	they are not, discuss whether ACU has analyzed the application of such loads internal to the systems pit and its safety-related SSCs.	
3.2-2	The regulation 10 CFR 50.34(a)(1)(i) requires, in part, that the PSAR include "[a] description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design." The regulation 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "[i]nformation relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety." The regulation 10 CFR 50.34(a)(4) requires, in part, that the PSAR include "[a] preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents." ACU MSRR PSAR, Revision 1 (ML23319A094), Section 3.1.2, PDC 2, "Design bases for protection against natural phenomena," states: The safety related SSCs shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding	Based on ACU's commitments to provide additional analysis of tornado loading on the systems pit and precast panels in an OL application, the staff determined that no response to this question was necessary.

	area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed. ACU MSRR PSAR, Revision 1, Section 3.1.2, PDC 4, “Environmental and dynamic effects design bases,” states, in part: The safety related SSCs … shall be appropriately protected against dynamic effects of events and conditions outside the MSRR facility. The PSAR, Revision 1, Section 3.2.1.1, states that the “MSRR facility is designed to withstand the basic wind velocity of 120 mph for Risk Category IV structures.” The PSAR, Revision 1, Section 3.2.2.2, confirms that the 120 mph wind load is applicable to “[t]he above-grade structure of the research bay.” PSAR Revision 1, Section 3.2.2.1, also defines the design basis tornado for the safety-related portions of the MSRR facility to have a maximum wind speed of 230 mph and an 83 millibar pressure drop in accordance with guidance in NRC Regulatory Guide (RG) 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1 (ML070360253). However, it is not clear how PDCs 2 or 4 are met for the safety-related portions of the SERC, including consideration of lateral and dynamic loads due to the fact that non-safety portions of the SERC are designed for a lower (non-tornadic) wind load and failure of the non-safety portions of the SERC could have impacts on the safety-related portions, and including consideration of design basis tornado generated lateral and dynamic loads	

	(which are separate from those of wind). a. How do safety-related portions of the as-built MSRR facility, including safety-related portions of the SERC research bay, the MSRR systems pit, and its foundation slab structure including the drilled piers comply with PDCs 2 and 4 with regard to the design basis tornado? Specifically, how are the safety-related structural portions of the facility designed to resist the effects of lateral and dynamic loads resulting from the design basis tornado? Summarize analyses performed and discuss results demonstrating adequacy of the design and resulting safety margins. b. Provide information such as a figure comparable to As-Built Drawing S-001 provided for audit on May 5, 2023, which includes design information (e.g., materials of construction) relevant to the constructed safety-related portions of the SERC research bay, MSRR pit and foundation slab structure including the drilled piers needed to demonstrate compliance with 10 CFR 50.34(a)(1)(i), 10 CFR 50.34(a)(3)(iii), 10 CFR 50.34(a)(4), and PDCs 2 and 4; and which clarifies what is meant by "other ACI applicable standards" used in the design and construction of MSRR safety-related facility SSCs as referred to in As-Built Drawing S-001.	

Resolution of Questions on Section 3.3, "Water Damage"

Question Number	Question	Resolution
3.3-1	PSAR Section 3.3.1.1.1 states that the "systems pit is a reinforced concrete structure designed to meet ACI 349-2013...". The staff notes that NRC endorses ACI 349-13, "Code Requirements for Nuclear Safety-Related Structures and Commentary" subject to the staff regulatory positions and regulatory guidance in RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Components)," Revision 3 (ADAMS Accession No. ML20141L613). Although RG 1.142 is guidance for nuclear power reactors, it is not clear to the staff to what extent ACU incorporated the RG's staff regulatory positions and guidance augmenting ACI 349-13 to the design of the MSRR, as applicable. Clarify whether and to what extent the design of the MSRR facility will be consistent with RG 1.142 endorsing the ACI 349-13.	In information provided for audit, ACU clarified that RG 1.142 only informed the design insofar as it pointed to the use of ACI 349-13 as a standard for the research bay foundations. No question-specific PSAR changes or other additional docketed information were necessary for the resolution of this question.

Resolution of Questions on Section 3.4, "Seismic Damage"

Question Number	Question	Resolution
3.4-1	PSAR Section 3.4.2.1 describes seismic categories, and includes a reference to NRC RG 1.29, "Seismic Design Classification for Nuclear Power Plants." Specifically, PSAR Section 3.4.2.1 states "By designing the SSCs in accordance with RG 1.29 to withstand the effects of [a safe-shutdown earthquake], a designed-in safety margin is provided for bringing the reactor to a safe shutdown condition, while also reducing potential offsite doses from seismic events." However, it is not fully clear if ACU is committing to designing SSCs in accordance with RG 1.29	ACU modified PSAR section 3.4.2.1 by letter dated July 30, 2024 (ML24219A258), to indicate that design of SSCs will be informed by RG 1.29, Revision 6 (i.e., ACU is not fully committing to it). In response to additional staff follow-up questions about how ACU would use RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," and RG 1.180, "Guidelines for Evaluating

	(RG 1.29, Revision 6 (ADAMS Accession No. ML21155A003), or another version; see audit question 3-1). The staff notes that there appear to be some differences between the terminology used in RG 1.29 and the PSAR that could require clarification if ACU is committing to RG 1.29, such as the difference between “collapse” and “failure” for Seismic Category II. Please clarify as necessary.	Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems” (which were also cited in the PSAR), in the design, ACU also modified PSAR chapter 3 including sections 3.5.2.3.1 and 3.5.2.3.2 by letter dated July 30, 2024 (ML24219A258), to clarify the seismic classification and qualification methodology for MSRR SSCs including how ASME QME-1-2017, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities” (which is referenced in RG 1.100, Revision 4), will be for used seismic qualification by testing and analysis, and to delete the RG 1.180 references. In information provided for audit in response to other additional staff follow-up questions, ACU further clarified that it will consider guidance in Nonmandatory Appendix QR-B to ASME QME-1-2017 when it is applying guidance from Nonmandatory Appendix QR-A to ASME QME-1-2017 for seismic qualification; and provided further clarification regarding how it plans to use American Society of Civil Engineers (ASCE) 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities” (which is referenced in PSAR chapter 3), alongside ASME QME-1-2017.
3.4-2	PSAR Section 3.5.2.4 states that “Safety-related SSCs are classified as Quality-Related, while non-safety related SSCs are classified as Not Quality Related.” PSAR Section 3.4.2.1 states that non-safety-related SSCs are	ACU modified PSAR section 3.4.2.1 by letter dated July 30, 2024 (ML24219A258), to clarify the classification scheme for MSRR SSCs.

	those not classified as Seismic Category I or II, which suggests that all Seismic Category II SSCs are safety-related (and therefore also quality-related). Please clarify if this is correct. (See also audit question 3.5-6.)	
3.4-3	PSAR Section 3.1.1 discusses the MSRR research bay systems pit. PSAR Figures 3.1-5 through 3.1-9 show the pit to be attached to drilled piers (which extend to 55 feet below grade according to PSAR Section 3.1.1) and the research bay slab to be on more slender deep foundations, which the staff presumes to be piles. The staff also notes that Section 3.1.1 of the PSAR states that "[t]he floor slab [of the research bay systems pit] is supported by the piers and structurally isolated from the subgrade to prevent interaction with the expansive surficial soils." The PSAR does not appear to discuss the slender drilled piles that appear to support the research bay and its slab, including whether they are safety-related, what material they are made of (e.g., steel or concrete) and what their depth of penetration is. The staff notes that any feedback from the research bay to its supporting slender drilled piles would create an interaction of these piles with the soil that could affect the response of the systems pit and its drilled pier foundation system. The staff notes that drilled piers or piles should typically be designed and constructed to withstand maximum imposed curvatures from earthquake ground motions and structure response. The staff also notes that curvatures should typically include free-field soil strains modified for soil/pile/pier/structure interaction and potential coupling of curvatures with pier or pile deformations induced by lateral pier or pile resistance to structure seismic forces. It is not clear whether the design of the SERC/slender pile/systems pit/drilled pier foundation system described in PSAR Section 3.4.2.3 has addressed such soil-structure	In information provided for audit and during audit discussions, ACU clarified and confirmed that all of the deep foundations for both the research bay slab and the systems pit are straight-shaft drilled piers of steel-reinforced concrete, extending to 55 feet below the surface. ACU provided information for audit related to the systems pit piers, including design drawings and construction checklists and reports. In addition, ACU modified PSAR chapter 2 by letter dated July 30, 2024 (ML24219A258), to add information on bearing capacities and other information related to the systems pit piers, as discussed under the resolution of audit question 2.5-10. In addition, ACU modified PSAR table 3.4-1 by letter dated July 30, 2024 (ML24219A258), to clarify that all portions of the SERC which provide structural support for safety-related SSCs are themselves considered safety-related. In information provided for audit and during audit discussions, ACU also confirmed that relevant SSI analyses for MSRR SSCs will be provided in an OL application.

	interactions. Please discuss the piles that appear to support the research bay and its slab, including whether they are safety-related, materials, and penetration depth. In addition, please describe any analyses that ACU performed of the SERC/slender pile/systems pit/drilled pier foundation system, and whether these analyses considered the potential aforementioned couplings of curvature and deformations of the drilled piers, slender piles, and the systems pit, subject to all applicable loads. Provide information regarding these analyses for audit.	
3.4-4	PSAR Section 3.4.2.5 references codes and standards for seismic analysis of SSCs, including ASCE 4 and ASCE 43. The staff notes that ASCE 4-16, "Seismic Analysis of Safety-Related Nuclear Structures," states that it provides "criteria for determining the response of structural elements in new facilities when subjected to earthquake ground motion [however] the analysis of caisson [i.e., drilled pier] and pile-supported foundations are not covered by the standard." It is not clear to the staff to what standard the SERC/pile/systems pit/drilled pier foundation system is designed to. For the SERC/pile/systems pit/drilled pier foundation system, specify the standard to be used for analysis and the acceptance criteria for "caisson and pile-supported foundations" that are applicable for the SERC/pile/systems pit/drilled pier foundation system.	ACU modified PSAR section 3.4.2.5 by letter dated July 30, 2024 (ML24219A258), to remove references to ASCE 4 and ASCE 43. In information provided for audit and during audit discussions, ACU clarified that the SERC was built to ASCE 7-10, and that ASCE 4 and ASCE 43 were not applied to the piers during construction (although the staff notes they will be relevant to parts of the MSRR facility including portions of the pre-existing SERC as discussed elsewhere in PSAR section 3.4 and in PSAR section 3.5).
3.4-5	PSAR Section 2.1.1.2 states that the "MSRR operations area, or the exclusion area boundary, which is anticipated to be the area directly under the U.S. Nuclear Regulatory Commission (NRC) facility operating license […] consists of the research bay, reactor control room, radiochemistry lab, and dress-out room."	During audit discussions and in information provided for audit, ACU clarified that Seismic Category I systems will not extend beyond the exclusion area boundary into other portions of the pre-existing SERC. No question-specific PSAR changes or other

	PSAR Section 3.4.1 states that: Where portions of an MSRR system are classified as Seismic Category I, the boundary limits of that portion of the SSCs designed to Seismic Category I provisions are reviewed against the design of the existing SERC facility. … [F]or fluid systems that are partially Seismic Category I or are Seismic Category II because of location in the existing MSRR facility, the Seismic Category I portion of the system extends to the first seismic restraint beyond the isolation valves that isolate the part that is Seismic Category I. At the physical interface between seismic and non-seismic Category I piping systems, the Seismic Category I dynamic analysis is extended to either the first anchor point in the non-seismic system or to a sufficient distance into the non-seismic system so as not to degrade the validity of the Seismic Category I analysis. Those interfaces and seismic classifications are clearly identified on the final arrangement drawings of the MSRR facility." It is not clear to the staff whether the aforementioned Seismic Category I systems extend beyond the exclusion area boundary into other portions of the existing SERC, and whether they have been designed to same seismic design standards and acceptance criteria as other Seismic Category I systems. Please clarify, and as appropriate and available, provide drawings indicating interfaces and seismic classifications of SSCs in an electronic reading room for audit. Discuss the standards used for seismic classification of each system extending beyond the	additional docketed information were necessary for the resolution of this question.

	exclusion area boundary; if different from other systems, discuss what acceptance criteria were used for systems extending beyond the exclusion area boundary and why such criteria were used.	
3.4-6	PSAR Section 2.5.5 states that it is estimated based on the ASCE 7 hazard tool that the peak ground acceleration (PGA) for the proposed site is 0.0305 gravity (g), "within the predicted 0.039 PGA denoting ASCE 7 Seismic Design Category A." PSAR 3.4.2.2 states that the MSRR facility design basis uses a response spectrum anchored to a PGA of 0.03 g. PSAR Section 3.4.2.2 states that NRC RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," is used for the development of the design response spectra. The staff notes that RG 1.60, Revision 2 (ADAMS Accession No. ML13210A432), states that a "response spectrum, anchored at 0.1 g, is an appropriately shaped response spectrum to define the minimum seismic input requirement at the foundation ...". The staff also notes that the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," and ORNL/TM-2020/1478, "Proposed Guidance for Preparing and Reviewing a Molten Salt Non-Power Reactor Application," references IAEA TECDOC-403, "Siting of Research Reactors, which similarly assigns a value of 0.1 g (i.e., 0.08 g multiplied by 1.25; see TECDOC-403 Tables 3.2 and 3.3) for the type of soil described in the MSRR application. It is not clear to the staff why PSAR Section 3.4.2.2 states that RG 1.60 is used for the development of the design response spectra, but PSAR Section 2.5.5 appears to indicate that ACU developed the response spectra consistent with ASCE 7-10. Clarify what guidance ACU follows to determine the response spectrum for the design of the MSRR and its foundation system. (See also	ACU modified PSAR section 3.4.2.2 by letter dated July 30, 2024 (ML24219A258), to clarify that the SERC was designed to ASCE 7-10 using two-point design response spectra, but the MSRR and its safety-related foundations will be demonstrated to withstand a safe-shutdown earthquake based on RG 1.60, Revision 2. (See also resolution of question 2.5-1.)

	audit questions 2.5-1 and 3.4-7.)	
3.4-7	PSAR Section 3.4.2.5 references ASCE 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," and states that this standard, along with ASCE 4-16, "Seismic Analysis of Safety-Related Nuclear Structures," is used for the MSRR seismic analysis. However, the PSAR does not appear to clearly describe how this framework (or a different framework) is followed for evaluating loads for nuclear safety significant (or safety-related) SSCs, particularly for seismic design. The staff notes that ASCE 43-19 indicates that the seismic design basis for nuclear facilities is a combination of a qualitative description of the acceptable level of damage, denoted by its limit state (LS), and a seismic design category (SDC). However, it is not clear to the NRC staff how the proposed design response spectrum (DRS) for the MSRR conforms to ASCE 43-19. It is not clear to the staff whether ACU assigns SDCs for its SSCs consistent with ASCE 43-19 (and ANSI/ANS-2.26-2004, "Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design," which is referenced in ASCE 43-19). In addition, it is not clear to staff whether (and if not, why not) ACU's design satisfies the ASCE 43-19 provision of elevating the DRS to at least 0.04 g for certain SDCs. Please discuss, and as appropriate, provide information on SDC and corresponding LS assignment for each of the MSRR safety significant (or safety related) SSCs.	ACU modified PSAR section 3.4.2.5 by letter dated July 30, 2024 (ML24219A258), to remove references to ASCE 4 and ASCE 43. In information provided for audit and during audit discussions, ACU clarified that the SERC was built to ASCE 7-10, and that ASCE 4 and ASCE 43 were not applied during construction (although the staff notes they will be relevant to parts of the MSRR facility including portions of the pre-existing SERC as discussed elsewhere in PSAR section 3.4 and in PSAR section 3.5). (See also resolution of questions 2.5-1, 3.4-1, and 3.4-6.)

Resolution of Questions on Section 3.5, "Systems and Components"

Question Number	Question	Resolution
3.5-1	The staff requests that ACU discuss (or provide relevant information for audit, as appropriate) the following to allow the staff to better understand how the MSRR design, specifically ACU's proposed equipment qualification program, will comply with PDCs 1, 2, 4, and 23: I. PSAR Sections 3.5.2.2 and 3.5.2.3.2 reference IEEE/IEC 60780-323, "International Standard – Nuclear Facilities – Electrical Equipment Important to Safety - Qualification," with respect to qualification of MSRR equipment. Provide a list of electric equipment that must be qualified and discuss the criteria used to identify this equipment as needing to be qualified (i.e., scoping criteria). II. The staff notes that NRC has yet to officially endorse IEEE/IEC 60780-323. It is not clear to the staff from the PSAR whether ACU is committing to this standard in its entirety, or may take certain exceptions from the standard. Explain how ACU plans to use this standard as well as any other equipment-specific standards that will be used to qualify electric equipment, as appropriate.	During audit discussions and in information provided for audit, ACU clarified that electrical power does not perform a safety function for the MSRR and there is no safety-related electrical equipment required or relied upon to mitigate postulated accidents. ACU modified PSAR section 3.4.2.5 by letter dated July 30, 2024 (ML24219A258), to remove the reference to Institute of Electrical and Electronics Engineers/International Electrotechnical Commission (IEEE/IEC) 60780-323, and to clarify that safety-related SSCs susceptible to electro-magnetic interference and/or radio frequency interference are confined to the scope of the instrumentation and control systems.
3.5-2	PSAR Section 3.5.2.1 provides safety classification criteria, specifically, a definition of "safety-related." However, the definition appears to contain a circular reference. Please clarify the meaning of the definition and discuss whether revision of the definition may be necessary.	ACU modified PSAR section 3.5.2.1 by letter dated July 30, 2024 (ML24219A258), to clarify the definition of safety-related SSCs.
3.5-3	ORNL/TM-2020/1478, Appendix A, Part 1, Section 3.5, states that applicants should give the design bases for the systems and components required to function for safe	ACU provided for audit information related to reactor system loads and calculation methodologies, including information on preliminary design

	reactor operation and shutdown. However, it is not clear to the staff if the PSAR contains such design bases based on a comprehensive consideration of conditions that may be important for reliable operation of MSRR systems and components. Please discuss or provide for audit preliminary design specifications for MSRR electromechanical systems and components, to allow the NRC staff to confirm that appropriate considerations discussed in the ORNL/TM-2020/1478, Appendix A, Parts 1 and 2, Section 3.5, guidance for systems and components (for example, dynamic and static loads, number of cycles, vibration, wear, friction, and effects of the operating environment) are properly incorporated into the MSRR design.	methodologies and considerations for satisfying the ASME Boiler and Pressure Vessel Code (BPVC), Section III, Division 5, “High Temperature Reactors,” requirements to which systems would be designed. ACU also modified PSAR section 3.5.1 by letter dated July 30, 2024 (ML24219A258), to provide information on load combinations for MSRR systems and components. In addition, in its response, dated March 28, 2024, to request for additional information (RAI) 1 (ML24088A324), ACU provided relevant information on service conditions and codes and standards used for the design of the MSRR SSCs.
3.5-4	PSAR Section 3.5.2.2 discusses qualification of safety-related systems and components using IEEE/IEC 60780-323. However, the staff notes that this standard covers electrical equipment. Please provide additional details regarding equipment qualification for other SSCs, including mechanical equipment and any equipment containing non-metallic materials (e.g., O-rings, gaskets, or seals), as applicable.	ACU revised the PSAR including adding PSAR section 3.1.4, and modifying PSAR sections 3.5.1, 3.5.2.3.1, and 3.5.2.3.2, by letter dated July 30, 2024 (ML24219A258), to provide information on codes and standards used for the MSRR, load combinations for MSRR systems and components, and seismic qualification. In addition, in its response, dated March 28, 2024, to request for additional information (RAI) 1 (ML24088A324), ACU provided relevant information on service conditions and codes and standards used for the design of the MSRR SSCs. ACU’s PSAR revisions and response to RAI 1 placed special emphasis on the safety-related valves and the ASME QME-1 qualification process used for SSCs.
3.5-5	PSAR Section 3.5.2.2 states that safety-related systems and components are qualified using applicable guidance in IEEE/IEC 60780-323. PSAR Section 3.5.2.2 also states that “nonsafety-related components and systems are	In audit discussions, ACU stated that it would redefine its safety classifications to remove the “important to safety” designation, and because any SSC that could prevent a safety-related SSC from

	qualified to withstand stress caused by environmental and dynamic service conditions under which their failure could prevent satisfactory accomplishment of the safety-related functions." Please discuss and provide details on the qualification methodology used for non-safety-related components and systems.	performing its function will itself be a safety-related SSC, there would be no non-safety-related components or systems that could impair a safety function. ACU revised the PSAR by letter dated July 30, 2024 (ML24219A258), to remove the "important to safety" designation. ACU also revised PSAR section 1.3.3.2 by letter dated July 30, 2024 (ML24219A258), to clarify in the context of stress from service conditions that although ACU intends to submit an OL application for 20 years of MSRR operation, the MSRR would only be operated for up to 5 effective full-power years.
3.5-6	PSAR Section 3.5.2.4 states that "safety-related" SSCs are quality-related, while non-safety-related SSCs are not quality related. However, the PSAR, including in Sections 3.1 and 3.5, also refers to SSCs that are "important to safety," and it is not clear to the staff what quality controls such SSCs may be subject to. Describe the controls for non-safety-related SSCs that are "important to safety." (See also audit question 3.4-2).	ACU revised the PSAR by letter dated July 30, 2024 (ML24219A258), to remove the "important to safety" designation.
3.5-7	PSAR Sections 3.5.2.3.1 and 3.5.2.3.2 discuss seismic qualification of MSRR SSCs, and reference IEEE 344, "IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations," and NRC RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems." The staff notes that the NRC endorsed IEEE 344-2013 with certain regulatory (staff) positions in NRC RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 4 (ADAMS Accession No. ML19312C677). Please clarify how ACU will use the	ACU revised PSAR sections 3.5.2.3.1 and 3.5.2.3.2 by letter dated July 30, 2024 (ML24219A258), to clarify the seismic qualification process for MSRR SSCs (see also resolution of question 3.4-1). In addition, ACU revised the PSAR by letter dated July 30, 2024 (ML24219A258), to clarify that the cited version of IEEE 344 is IEEE 344-2013, and to include revisions/versions of various RGs.

	IEEE 344 standard, including what specific edition ACU will use, and if it will incorporate the regulatory (staff) positions in NRC RG 1.100, Revision 4, when using the standard. In addition, please clarify which revision of NRC RG 1.180 ACU intended to reference in the PSAR.	

Resolution of Additional Topics Beyond the Scope of the Specific Audit Questions

Topic	Question	Resolution
Aircraft Impact Assessment (related to PSAR section 2.2)	The staff requested that ACU provide for audit details of its aircraft impact analyses discussed in PSAR section 2.2.	ACU provided air crash hazard calculations (supporting the results provided in the PSAR) for airport hazards for audit. The staff confirmed that these calculations generally applied airport hazard methodologies in DOE-STD-3014-2006.
PDC (related to PSAR section 3.1)	In addition to audit questions 3.1-1 through 3.1-4, the staff provided ACU with other additional follow-up questions and feedback on PDC during audit interactions. This included questions and feedback related to, for example, consistency between PDC in PSAR section 3.1 and PDC references in other portions of the PSAR; clarification of terminology used in PDC; clarification on the scope/applicability of certain PDC including what they require and what SSCs they cover; and ensuring appropriate structure, clarity, and specificity of PDC.	ACU revised the PSAR including the PDC and bases in section 3.1.2 by letter dated July 30, 2024 (ML24219A258), to incorporate changes in response to NRC questions and feedback in audit discussions, and other changes.
SERC Design (related to PSAR sections 3.2 and 3.4)	The staff requested that ACU provide for audit detailed SERC design drawings including as-built drawings.	ACU provided detailed SERC design drawings including as-built drawings for audit. The staff reviewed these drawings to verify certain information, including that the SERC research bay wind load design is consistent with ASCE/SEI 7-10 requirements, and that seismic design loads for the SERC, including the research bay and systems pit, are in accordance with (1) the building code of record (International Building Code (IBC) (2012)) as adopted by the City of Abilene,

		Texas; (2) the spectral responses for short and 1-second periods as reported in PSAR section 2.5.5; and (3) site characteristics associated with seismicity, earthquake, and other potential vibratory ground motion.

6.0 EXIT BRIEFING

The staff conducted an audit closeout meeting on August 20, 2024. At the exit briefing the staff reiterated the purpose of the audit and summarized the audit activities. Additionally, the staff stated that they did not identify areas where further additional information would be necessary to support the review.

There were no deviations from the audit plan.

7.0 ADDITIONAL INFORMATION RESULTING FROM AUDIT

RAIs and RCIs were generated as a result of topics discussed in this audit and other chapter-specific audits for this review, and ACU's responses supported the resolution of audit questions as noted above. ACU also updated the PSAR on its own initiative as noted above to address several items discussed during the audit.

8.0 OPEN ITEMS AND PROPOSED CLOSURE PATHS

Not applicable.

SUMMARY REPORT FOR THE REGULATORY AUDIT OF ABILENE CHRISTIAN UNIVERSTY MOLTEN SALT RESEARCH REACTOR CONSTRUCTION PERMIT PRELIMINARY SAFETY ANALYSIS REPORT CHAPTERS 4 (MOLTEN SALT RESEARCH REACTOR DESCRIPTION), AND 6 (ENGINEERED SAFETY FEATURES), AND SECTION 9.6 (GAS MANAGEMENT SYSTEM)

March 2023 – August 2024

1.0 BACKGROUND AND PURPOSE

By letter dated August 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22227A202), as supplemented by letter dated October 14, 2022 (ML22293B816), Abilene Christian University (ACU) submitted to the U.S. Nuclear Regulatory Commission (NRC), an application for a construction permit (CP) for a molten salt research reactor (MSRR), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and Section 104c of the Atomic Energy Act of 1954, as amended. The application included a preliminary safety analysis report (PSAR) (ML22227A203). PSAR chapter 4, "Molten Salt Research Reactor Description," describes the principal features, operating characteristics, and parameters of the MSRR non-power test reactor. PSAR chapter 6 describes the engineered safety features (ESFs) which are features designed to mitigate the consequences of accidents and to keep radiological exposures within acceptable values. PSAR section 9.6 "Gas Management System," describes the gas management system (GMS) which handles those gases which directly interface with salts across multiple subsystems.

This audit enabled the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) to gain a better understanding of PSAR chapters 4, 6 and section 9.6 through review and discussion of underlying supporting documentation. Enhanced understanding and communications supported effective and efficient resolution of technical issues, including though development of information needs where needed.

2.0 AUDIT REGULATORY BASES

The basis for the audit is the regulations in 10 CFR Part 50, Sections 50.34(a), "Preliminary safety analysis report," and 50.35, "Issuance of construction permits."

3.0 AUDIT OBJECTIVES

The primary objective of the audit was to enable a more effective and efficient review of PSAR chapters 4, 6, and section 9.6 through the staff's review and discussion of supporting documentation with ACU. Gaining access to underlying documentation and engaging in audit discussions about reactor design, ESF, and the GMS, facilitated the staff's understanding of the MSRR CP application and aided in assessing the safety of the proposed research reactor. The audit improved communication and provided detailed information for the staff.

4.0 SCOPE OF THE AUDIT AND AUDIT ACTIVITIES

The audit was conducted from March 2023 to August 2024, via teleconference and the online refence portal (Certrec), and also during a site visit conducted May 17-18, 2023. The staff conducted the audit in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction NRR-LIC-111, Revision 1 "Regulatory Audits" (ML19226A274).

Members of the audit team, listed below, were selected based on their detailed knowledge of the audit subject matter. Audit team members included:

1. Boyce Travis, NRR (Senior Nuclear Engineer)
2. Kyle Song, NRR (Electronics Engineer)
3. Chris Van Wert, NRR (Senior Technical Advisor for Reactor Fuels)
4. Nicholas Hansing, NRR, (Mechanical Engineer)
5. Charles Peabody, NRR (Nuclear Engineer)
6. Benjamin Parks, NRR (Senior Nuclear Engineer)
7. Alexander Chereskin, NRR (Materials Engineer)
8. Chakrapani Basavaraju, NRR (Mechanical Engineer)
9. Zachary Gran, NRR (Health Physicist)
10. Steve Jones, NRR (Senior Safety and Plant Systems Engineer)
11. Ting Sham, RES (Senior Technical Advisor for Advanced Reactors)
12. Joseph Bass, RES (Reactor Engineer)
13. Ryann Bass, NRR (Reactor Systems Engineer)
14. Edward Stutzcage, NRR (Reactor Scientist (Radiation Protection))
15. Mohsin Ghazali, NRR (Project Manager)
16. Brian Bettes, NRR (Project Manager)
17. Michael Balazik, NRR (Project Manager)
18. Richard Rivera, NRR (Project Manager)
19. Zackary Stone, NRR (Project Manager)
20. Edward Helvenston, NRR (Project Manager)
21. Michelle Hayes, NRR (Branch Chief, Technical)
22. Gregory Oberson, NRR (Branch Chief, Technical)
23. Michael Wentzel, NRR (Branch Chief, Licensing)
24. Stephen Philpott, NRR (Branch Chief, Licensing)

Prior to the audit, the audit team reviewed PSAR chapters 4, 6, and section 9.6 and defined the general range of topics in the audit plan (ML23065A055) to be addressed and focused on during the audit. The following table documents dates that the staff transmitted audit questions and when audit meetings were held:

Audit Questions (ADAMS Accession No.)	Audit Meetings
March 24, 2023 (ML23086A014)	May 17, and 18, 2023
May 2, 2023 (ML23123A044)	October 3, 5, 24, and 26, 2023
December 20, 2023 (ML23354A285)	November 2, and 9, 2023
January 8, 2024 (ML24008A092)	January 4, 10, 16, and 25, 2024
January 11, 2024 (ML24015A010)	February 6, 16, and 22, 2024
January 17, 2024 (ML24017A303)	March 7 and 21, 2024
January 22, 2024 (ML24022A212)	April 11, 18, 25, and 26, 2024
January 23, 2024 (ML24023A624)	May 8, 10, and 24 2024
	August 20, 2024 (exit meeting)

The staff reviewed the following documents via the ERR:

- Written responses that ACU prepared for certain questions to address the questions and/or facilitate discussion with staff

- PSAR pages indicating changes proposed by ACU in response to various audit questions
- Draft responses to the staff's requests for additional information (RAIs)
- "MSRR Neutron and Gamma Heating," dated December 22, 2021.
- Report on radiation damage for reactor components, "MSRR-GT-SNP-2021-09-R1(Material Damage Report)," dated January 21, 2022.
- ACU chemical hygiene plan, "HRP Chemical Hygiene Plan 2022," dated February 15, 2022.
- Report on thermophysical properties of MSRR salts, "MSRR Thermophysical Properties Rev 2-7," dated March 28, 2023.
- Gibbs free energy diagrams for various fluorides and oxides, "4.2-4 Oxide Thermodynamics," provided May 11, 2023.
- Report on MSRR graphite properties, "Graphite – POCO Entegris AXF-5Q," dated May 25, 2023.
- "Noble Gas Fission Product Generation Rate and Air Activation," dated June 6, 2023.
- ACU's proposed experiments investigating 316H stainless steel (SS) compatibility with FLiBe, "MSRR Corrosion Test Matrix 06-2023," provided June 20, 2023.
- "Confirmatory analysis of dpa in MSRR reactor vessel," dated July 18, 2023.
- ACU's process flow diagram revision C, "PFD SK-0001_REV_C," provided July 12, 2023.
- "Confirmatory analysis of dpa in MSRR Kappa design using SCALE," dated August 1, 2023.
- "Degradation Mechanisms Table September 29 Revision," dated September 29, 2023.
- ACU proposed in-service inspection and testing of components, "In-Service Non-Destructive Testing of the MSRR," provided October 23, 2023.
- "Loss of Off-Site Power modeling in RELAP5-3D," dated May 6, 2022, as edited March 30, 2023.

The staff previously issued an interim audit report related to this audit by letter dated June 22, 2023 (ML23157A064).

5.0 SUMMARY OF AUDIT OUTCOME

The staff's audit focused on the review of supporting documents associated with the questions provided to ACU during the audit. The staff reviewed information through the ACU Certrec portal and held discussions with ACU staff to understand and resolve questions. In many cases, ACU updated the PSAR to resolve items discussed in the audit. The tables below replicate specific audit questions transmitted in emails to ACU as listed above and summarize the resolution of the audit questions.

Resolution of Questions on Chapter 4, "Molten Sat Research Reactor Description" (General Questions Relevant to Entire Chapter)

Question Number	Question	Resolution
4-1	Fail-open helium gas valves are relied on to allow salts to drain into drain tanks for several scenarios in the safety analyses. This action is described as shutting down the nuclear reaction. Additional details are necessary supporting the design, qualification, testing, and inspection of these valves. NRC staff will need to confirm the reliability, redundancy, and independence of the valves to support findings around DC 21 and 22.	This question was addressed through the resolution of audit question 3.5-4.
4-2	Please describe plans to monitor, test, and analyze for vibration and other dynamic effects which may be encountered during preoperational testing, startup and/or full power operation.	ACU stated in information provided for audit that resonance frequency ranges of the reactor system will be identified during design and compared to frequencies associated with fluid flow and mechanical operations. In addition, ACU stated that acoustic monitors will be used to identify vibrations during pre-operational testing.
4-3	Define the reactor system boundary, materials of construction, and which codes and standards are applicable to which SSCs. For example, are the fresh fuel and effluent tanks designed to ASME Code Division 5? Provide a description of all salt-wetted containment structures (e.g., preliminary arrangement/dimensions, fabrication, welds), and provide context for any reactor system boundary components for which the safety treatment is not consistent with other portions of the reactor system boundary.	ACU provided a proprietary enclosure in its response, dated, March 28, 2024 (ML24088A324), to the staff's request for additional information (RAI) 1. This enclosure provides the materials of construction for all safety-related components, and the quality standards that apply to the safety-related structures, systems, and components (SSCs). A redacted version of

		this enclosure was provided by ACU by letter dated July 30, 2024 (ML24219A258). In addition, ACU added PSAR section 3.1.4, “Codes and Standards Used by the MSRR” by letter dated July 30, 2024 (ML24219A258) which lists the applicable codes and standards for the MSRR.
4-4	Reference 4.8-1 appears to show that ACU is using the 2021 edition of ASME Code Section III Division 5. The NRC staff has endorsed the 2017 edition of Section III Division 5 in RG 1.87, Revision 2, subject to certain limitations and conditions. Does ACU intend to use the 2021 edition? If so, provide the justification for why differences in the Code editions are acceptable.	ACU modified the PSAR section 4.8, “References,” by letter dated July 30, 2024 (ML24219A258) to include the NRC endorsed 2017 edition of ASME Code Section III Division 5 as the code that will be followed.
4-5	Design Criterion 32 states the fuel salt boundary shall be designed to be inspectable. However, it doesn’t appear that a description of how the boundary will be inspected is provided in either Chapter 4 or Chapter 5. Describe how inspections will be performed, and what portions of the boundary are subject to inspection in order to meet DC 32.	ACU modified the PSAR section 4.3.2, “Design Bases,” by letter dated July 30, 2024 (ML24219A258) to state that the reactor system is designed to permit visual inspection of the exterior and limited ultrasonic testing of certain welds. ACU stated in its response dated April 30, 2024 (ML24121A272) to the staff’s RAI 2 that the need for inspection will be determined by their Degradation Management Program (DMP). ACU confirmed by letter dated June 12, 2024 (ML24164A236)

		in response to the staff's request for confirmation of information (RCI), item 4.3-5, that all safety-related parts of the fuel handling system (FHS) that may experience degradation which could challenge FHS barrier integrity will be physically inspectable and details of any inspection plans, as determined to be needed by the DMP, will be provided in an operating license (OL) application.
4-6	Provide clarification on what is meant by "stainless steel 316H or equivalent (as determined by the carbon content)."	ACU clarified during audit discussions that stainless steel (SS) 316H (UNS S31609) will be used. ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to clarify when SS 316H will be used versus SS 316L.
4-7	Does ACU plan to use coatings on any salt-wetted materials? If so, provide context/a comprehensive description of preliminary information for relevant SSCs.	ACU stated in information provided for audit that coatings are not anticipated for the MSRR, and that further information may be provided in an OL application.
4-8	Confirm or provide context to assure that no galvanically dissimilar metals are used in the fuel salt boundary (this includes the base metal and weld fillers).	ACU stated during audit discussions with the staff that there will be no galvanically dissimilar metals used within the reactor system. ACU modified the PSAR

		section 4.3.3, “Reactor System Structural Material,” by letter dated July 30, 2024 (ML24219A258) to clarify that a weld filler of similar composition to the reactor structural material will be used. In addition, ACU clarified that a reduction-oxidation probe may be used in the reactor system which will be electrically insulated from the reactor system structure to prevent galvanic corrosion. Additionally, ACU confirmed by letter dated June 12, 2024 (ML24164A236) in response to the staff’s RCI, item 4.3-2 (2), that the effect of graphite on corrosion rates of SS 316H will be considered when determining a corrosion allowance. ACU stated during audit discussions with the staff that the only location where galvanically dissimilar metals will be used is in the FHS. ACU modified the PSAR section 9.2.3, “Operational Analyses and Safety Function” by letter dated July 30, 2024 (ML24219A258) to clarify that the MSRR will not have Ni-201 to SS 316H welds, instead the Ni-201 vessels will be bolted to

		the SS 316H enclosure with an electrically insulating material between the two metals. ACU stated in its response dated April 30, 2024 (ML24121A272) to the staff's RAI 2 that the proposed DMP will also cover all potential material degradation mechanisms. Therefore, the DMP will ensure that any potential dissimilar metal interactions are identified and mitigated.
4-9	Chapter 14 of the ACU PSAR states that composition, level and leakage of the fuel salt will be used as preliminary technical specifications. Describe or provide context on how technical specification limits will be determined. For example, are novel means being used to measure these parameters?	ACU stated during audit discussions that composition of the fuel salt will be determined by off-line analytical sampling. In addition, ACU modified PSAR section 4.2.1.3, "Location and Composition of the Fuel Salt," by letter dated July 30, 2024 (ML24219A258) to clarify the fuel salt sampling process. The staff will review the adequacy of off-line fuel sampling during an OL application review.
4-10	Given that the fuel salt boundary interfaces with a pressurized gas system, describe how, or provide a commitment that the fuel salt boundary will be protected from dynamic effects associated with the failure of a pressurized piping system (e.g., pipe whip).	ACU modified PSAR section 3.5.1, "General Design Basis Information," by letter dated July 30, 2024 (ML24219A258) to clarify that dynamic effects associated

		with the failure of a pressurized gas piping system will be accounted for in the MSRR design.
4-11	Provide the expected preliminary temperature profile for each of the fuel-salt-bearing SSCs to provide assurance that these components will remain within their qualified parameters in the proposed operating envelope.	During the in-person audit on May 18, 2023, ACU committed to incorporating a Safety Limit that will limit the upper temperature so that components will remain within the time and temperature limits for SS 316H that are endorsed by the NRC in Regulatory Guide (RG) 1.87, Revision 2, “Acceptability of ASME Code Section III, Division 5, ‘High Temperature Reactors.’” The staff will review final temperature profiles during an OL application review to ensure the limits found in NRC RG 1.87, Revision 2, are followed. Additionally, the staff will verify the proposed testing is consistent with the final temperature profiles.

4-12	To the extent possible, describe how inspection, monitoring, and testing programs (e.g., salt chemistry, radiation damage, chemical damages, erosion, pressure pulses, deterioration during the projected lifetime) will be planned and implemented. For example, consideration of surveillance specimens measuring critical degradation mechanisms (e.g., stress needed for creep, welds). If a description of these programs is premature, provide context on what commitments will be made in the PSAR to implement these programs.	The implementation of inspection, monitoring, and testing programs is satisfactorily addressed by the DMP which is described in ACU's response to RAI 2 (ML24121A272). In addition, ACU confirmed by letter dated June 12, 2024 (ML24164A236) in response to the staff's RCI, items 4.3-3 and 4.3-4, that the MSRR will have an appropriate materials surveillance program. The staff will review the results of the DMP including inspection, monitoring, and testing programs, during an OL application review.

Resolution of Questions on Section 4.1, “Summary Description”

Question Number	Question	Resolution
4.1-1	References to the refueling cycle require clarification between loading and unloading the reactor and adding UF_4 through the reactor access vessel (RAV). Once the UF_4 is in solution, given it is not possible to remove it, will that impact the fuel handling system (FHS) if it is utilized during the operational period?	ACU modified PSAR section 4.3.11, “Description of the Reactor Access Vessel by letter dated July 30, 20,”24 (ML24219A258) to include information on how UF_4 will be added to the flowing fuel salt during operation. In addition, ACU clarified in information provided for audit that fuel loading and unloading refer to the bulk transfer of fuel salt between the reactor drain tank (RDT) and the fuel handling system (FHS).

Resolution of Questions on Section 4.2, "Active Reactor Core"

Question Number	Question	Resolution
4.2-1	Section 4.2.1 indicates that more specific ranges of salt composition and volume will be provided in the OL, however this information is needed to verify the flow dynamics in the vessel and the RAV, as well as the flow and heat transfer in the heat exchanger. Salt composition ranges and volume changes will need to address fission product contamination ranges since the non-gaseous fission products are expected to remain in the salt solution for the remainder of the operating cycle. Provide the nominal and bounding compositions of the fuel salt. Provide fuel salt thermophysical and thermochemical properties as a function of temperature and composition, including off-normal temperatures and compositions. This should include any properties needed to model salt behavior in an accident and normal operations. Allowable compositions should include consideration of fission products, corrosion products, and other impurities, as build-up will affect thermophysical (e.g. viscosity) and thermochemical (e.g. CsI formation) properties.	ACU provided preliminary information during the audit supporting the ranges of salt composition and volume provided in the PSAR. The staff determined this information was sufficient to perform the required safety analysis at the CP stage from a reactor physics standpoint throughout the operating cycle, although ACU stated additional information would be provided in an OL application. ACU provided written information during the audit to describe the normal and bounding compositions of the fuel salt. ACU provided the total anticipated concentration of impurities, corrosion products, fission products, and plutonium in the fuel salt at end of life. PSAR section 15.1.2, "Sources of Funds," states that ACU is developing a fuel qualification plan to support an OL application.

4.2-2	What are the solubility limits for actinides in the molten salt and what measures are in place to ensure these are not exceeded?	ACU modified PSAR section 4.5.2.3, “Reactivity induced by fuel salt composition changes,” by letter dated July 30, 2024 (ML24219A258) to include a basic description of the UF_4 slug addition which details acceptable safeguards for reactivity management and control that ensure UF_4 does not exceed solubility limits. In addition, PSAR section 4.5.2.3 states that burnup calculations will be used to determine fission product actinide levels which will inform the fuel addition plan over the reactor lifetime. ACU stated in information provided for audit that validation of these calculations will be performed through periodic sampling and analysis of the fuel salt. ACU stated in audit discussions that further information on actinide solubility limits will be provided for review in a future licensing submittal.
4.2-3	Will technical specifications (TS) be provided to limit fuel salt composition?	ACU stated in information provided for audit that TS will be proposed to limit fuel salt composition as part of an OL application. In addition, ACU modified PSAR section 14.3,

		"Limiting Conditions for Operations," by letter dated July 30, 2024 (ML24219A258) to include two limiting conditions for operation related to fuel salt composition. These conditions are oxygen and chromium concentration and the ratio of UF_3 to UF_4 in the fuel salt.
4.2-4	What is the allowable O_2 ingress/concentration to avoid precipitation of UO_2? Will any circulating buffer like ZrF4 be used?	ACU modified PSAR section 6.2.2, "Containment" by letter dated July 30, 2024 (ML24219A258) to state that the reactor enclosure atmosphere will be inert with nitrogen gas. The staff determined that O_2 ingress into the fuel salt would be limited by the inert atmosphere, thus significant precipitation of UO_2 is not credible.
4.2-5	4.2.1.2 describes how the fuel salt is consistent with DC 10 and 16. Provide the technical basis for these statements.	During audit discussions, ACU noted that draft ANSI/ANS-20.2, "Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten Salt Reactor Nuclear Power Plants," provides additional context related to PDC 10 and 16 and their bases for molten salt reactors. In addition, ACU modified PSAR section 3.1.2.2 by letter

		dated July 30, 2024 (ML24219A258) to clarify the bases for PDC 10 and 16. Based on its review of ANSI/ANS-20.2 (now published in final form as ANSI/ANS-20.2-2023) and the PSAR changes, the staff determined that no further information was necessary to resolve this question.
4.2-6	Provide the technical basis for the conclusion stated in section 4.2.1.3, “These changes are managed such that they do not impact reactor safety.”	ACU modified PSAR section 4.2.1.3 by letter dated July 30, 2024 (ML24219A258) to state that, “These changes to fuel salt composition will be managed such that the fuel salt remains within operating bounds.” The staff finds that this statement is supported by the discussion in PSAR section 4.2.1.3.
4.2-7	What is the expected fuel salt vapor phase in the RAV and drain tank head spaces and what chemical and radiological migration impacts are expected. As none are described, is there a technical basis for zero or negligible vapor phase?	ACU stated in information provided for audit that the expected fuel salt vapor in the reactor access vessel (RAV) and drain tank head spaces will be composed of LiF, BeF_2, and gaseous fission products. The staff determined that the GMS, described in PSAR section 9.6 is sufficient for handling the expected vapor phase.

4.2-8	Provide the basis for the Helium remaining soluble except in the RAV bubbler. Specifically, address homogenous mixing of Helium, preclusion of void formation, and consideration of 2-phase flow in and around the Helium Bubbler Off-gas system.	ACU provided clarification and supporting basis on the ERR that the MSRR can run for the operating cycle without any off-gas function credited to remove helium and other gases. ACU clarified that they plan to run off-gas to improve operational performance.
4.2-9	What are the allowable pressure limit boundaries which correspond to the temperature limit boundaries provided in 4.2.1.7?	ACU modified PSAR table 4.1-1, “Reactor Parameters” by letter dated July 30, 2024 (ML24219A258) to include a maximum reactor operating pressure.
4.2-10	Do the RPS and RCS shutdown margin calculations consider any amount of fuel salts seeping in between the graphite moderator blocks? Provide either the basis for no seepage or, if seepage can occur, describe the reactivity and timing effects on the reactor drainage sequence.	ACU provided calculations on the ERR that showed the volume of salt drainage from the reactor to ensure subcriticality. ACU demonstrated that the drainage volume is many orders of magnitude larger than the seepage area between the moderator blocks, and therefore the seepage effects on criticality are negligible.
4.2-11	Section 4.2.2 states the RCS is credited as a separate and diverse means of bringing the reactor subcritical but doesn’t play a role in the shutdown. What is the safety function of the RCS (versus the RPS)?	ACU provided clarification in information provided for audit that the RCS is an operational system only and does not have a safety function. The RPS is the sole reactivity

		control safety system. In addition, ACU modified PSAR table 3.4-1, "Safety and Seismic Classification of Structures, Systems, and Components" by letter dated July 30, 2024 (ML24219A258) to include the RCS as a non-safety-related system and the RPS as a safety-related system.
4.2-12	What is the MHA SCRAM time acceptance criteria and how is it measured (e.g., the bottom of the graphite at 1 min; a k_{eff} level corresponding to subcritical (~15s))? a. More specifically what is the shutdown margin acceptance criteria for the RPS? b. What is the SCRAM time acceptance criteria for the RPS? These would likely be in the form of a calculation.	ACU clarified in information provided for audit that MHA SCRAM time is measured by a k_{eff} corresponding to subcritical. a. ACU provided calculations on the ERR of the subcriticality in the reactor drain tank. ACU estimates k_{eff} is approximately 0.6 in the drain tank. k_{eff} value in the drain tank is significantly below the limits provided in 10 CFR 50.68. b. ACU indicated to the staff that SCRAM time testing and acceptance criteria will be established during an OL application.
4.2-13	Section 4.2.2.2 discusses the control rod (CR) assemblage as structurally attached to the top of the reactor vessel and supported by it. Please provide additional details on the means of attachment and how its interaction with the reactor vessel is analyzed.	ACU provided information for audit describing how the CR assemblies are structurally attached to the reactor vessel.

		ACU stated that final design information will be provided in an OL application. Audit question 3.5-4 addresses related staff questions regarding use of consensus codes and standards and the anticipated service conditions for the MSRR SSCs.
4.2-14	4.2.5.1 states “the consequent movement of graphite is accounted for in the mechanical design of both the grid plate and the graphite.” Provide additional details to explain this statement, potentially in a preliminary design specification.	ACU provided information for audit describing how graphite movement is accounted for. ACU stated that final design information will be provided in an OL application. Audit question 3.5-4 addresses related staff questions regarding use of consensus codes and standards and the anticipated service conditions for the MSRR SSCs.
4.2-15	Additional details are requested to better understand the core support structure and how it interfaces with other elements of the facility. For example, the eye hooks and notches/ledges mentioned in Section 4.2.5.3.	ACU provided information for audit describing the reactor enclosure support structure. ACU stated that final design information will be provided in an OL application. In addition, ACU modified PSAR section 4.2.5.3, “Reactor Enclosure Support Structure” by letter dated July 30, 2024 (ML24219A258) to state that the reactor enclosure is structurally supported from

		below, transmitting the load to the floor of the systems pit. ACU removed the mention of eye hooks from the design description as the design was reevaluated. Audit question 3.5-4 addresses related staff questions regarding use of consensus codes and standards and the anticipated service conditions for the MSRR SSCs.
4.2-16	Are there penetrations and supports for detectors?	ACU stated in information provided for audit that the final design will include penetrations and supports detectors. Audit question 3.5-4 addresses related staff questions regarding use of consensus codes and standards and the anticipated service conditions for the MSRR SSCs.

Resolution of Questions on Section 4.3, “Vessel”

Question Number	Question	Resolution
4.3-1	More definitive statements are needed for the codes and standards used for this design. Please specify the Code Edition and Addenda and indicate whether the Codes and Standards used will incorporate the conditions imposed on them by the NRC for acceptable use. This supports the staff’s findings regarding DC 1.	This question was addressed by ACU’s response dated March 28, 2024 (ML24088A324), to the staff’s RAI 1. In addition, ACU added PSAR

		section 3.1.4, "Codes and Standards Used by the MSRR" by letter dated July 30, 2024 (ML24219A258) which lists the applicable codes and standards for the MSRR.
4.3-2	Section 1.2.3.3 states "Reactor loop consists of the reactor vessel, access tank, reactor (fuel salt) pump, heat exchanger, drain tank and associated 2.5-in. (nominal) diameter piping." Clarify if the reactor loop contains any valves, orifices, or other SSCs. Further granularity on Table 3.4-1 would be helpful for understanding how DC 1 will be satisfied. Clarity on the boundaries between classifications and which specific SSCs fall into which category would further support satisfaction of DC 1.	ACU provided a proprietary process flow diagram in its response dated, March 28, 2024 (ML24088A324) to the staff's RAI 1 which lists the applicable codes for each component. In addition, a proprietary table was included in ACU's response to the staff's RAI 1 which lists the safety-related valves in the MSRR. A redacted version of this table was provided by ACU by letter dated July 30, 2024 (ML24219A258). Finally, ACU modified PSAR Table 3.4-1, "Safety and Seismic Classification of Structures, Systems, and Components," to provide further detail.
4.3-3	Although several ACU MSRR DC are listed as applicable to Section 4.3, it is not clear how these are met. Provide a description that demonstrates how all relevant DC are or will be satisfied, and that the selected DC are appropriate for the SSCs in this Section.	ACU modified PSAR section 4.3.2 by letter dated July 30, 2024 (ML24219A258) to include further explanation as to how PDC 1, 30, and 32 will be

		met. In addition, ACU submitted a redacted enclosure, “MSRR Codes and Services Conditions,” (ML24219A258) which lists the applicable codes to each MSRR component and provides a justification for its safety classification.
4.3-4	Section 4.3.1 states that this section describes design features common to all fuel-salt-bearing components within the reactor enclosure. As part of the audit, describe all components that are covered by the analysis in this section and provide preliminary descriptions of these components (e.g. preliminary dimensions, fabrication methods, penetrations, consequence of loss of integrity etc.). Additionally, provide the lifetimes for all fuel-salt-wetted SSCs, and whether any components are anticipated to need replacement during the MSRR life, based on anticipated degradation.	ACU provided a written response to the audit question and provided clarifications during audit discussions. ACU updated the PSAR by letter dated July 30, 2024 (ML24219A258) to clarify that the function of the RTMS is to act as a catch basin for salt from the reactor loop, and also to indicate that the RTMS will be capable of withstanding (without failure) relocation of the entire fuel salt inventory from the reactor system to the RTMS. During audit discussions, ACU stated that the design of the RTMS does not include penetrations nor welds below the salt level in the scenario that all fuel salt is relocated to the RTMS. Finally, ACU’s DMP, which is described in ACU’s response to the staff’s RAI 2

		(ML24121A272), will evaluate anticipated degradation which will inform fabrication methods and component lifetimes.
4.3-5	PSAR Section 4.3.10 states that small leaks from the gas management system (GMS) will be detected by the radiation monitoring system (RMS). ACU MSRR DC 30 requires that means be provided to detect and identify the location/source of fuel salt leakage. Clarify whether this is meant to be leaks from the GMS and/or from the fuel salt boundary. Additionally, describe how the RMS will detect leaks and if leaks can be detected throughout the entirety of the fuel-salt boundary.	ACU modified PSAR section 4.3.2 by letter dated July 30, 2024 (ML24219A258) to clarify for PDC 30 that the reactor enclosure gas will be periodically sampled and that a substantial quantity of fission products detected in the enclosure will indicate the presence of a fuel salt rupture. In addition, PSAR section 4.3.2 states that limits and actions will be defined in TS provided in an OL application.
4.3-6	Describe whether the gas composition and gas purity can affect the structural materials. Additionally, can the differences in heat transfer properties of Helium and the fuel salt cause thermal stresses in surrounding SSCs at the salt-gas interfaces?	ACU stated in its response dated April 30, 2024 (ML24121A271) to the staff's RAI 2 that the degradation mechanism assessment, will consider operating, transient, and environmental conditions, such as temperatures and service environments. The degradation mechanism assessment will inform the DMP as to which degradation mechanisms are

		unable to be precluded by the design of the MSRR.
4.3-7	Section 4.3.5, “Radiation Damage to Reactor System,” states that 0.1 dpa is below levels of mechanical property degradation in SS. Are all SR metallic components expected to see <= 0.1 dpa? Fluence at the vessel wall may be less than other components as it is shielded by the graphite components. Provide the data used to determine that mechanical property degradation at 0.1 dpa is negligible.	ACU stated in information provided for audit that the wording in PSAR section 4.3.5 is incorrect, specifically the sentence was meant to refer to “SS316H components,” rather than “solid components.” ACU also stated that some SS 316H components will exceed 0.1 dpa over 5 MWth power years. ACU revised PSAR section 4.3.5 by letter dated July 30, 2024 (ML24219A258) to state that the maximum damage to SS 316H components after 5 effective full power years will be bounded by 1 dpa. In addition, ACU provided their fluence calculations used to determine the bounding value of 1 dpa to the staff for audit.
4.3-8	Provide context on whether the effects of potential coolant salt ingress have been assessed (e.g. in the event of heat exchanger failures).	During audit discussions, ACU stated that the same FLiBe will be used for the fueled and coolant salt. ACU provided their plans for the purification of the fuel and coolant salt in a proprietary enclosure by letter dated on March 28, 2024, (ML24088A324) in response to the staff’s RAI 1. This

		mitigates the effects of potential coolant salt ingress. Additionally, fuel salt chemistry and rate of chemical attack will be monitored by fuel salt sampling. Negative impacts of a coolant salt ingress into the fuel salt will be detected by fuel salt sampling.
4.3-9	Provide context on whether the effects of salt freezing or precipitation (e.g. thermal expansion, gas generation) have been assessed to determine its effects on component integrity.	In information provided for audit, ACU clarified that the MSRR is designed and operated in such a manner that the fuel and coolant salts are only allowed to freeze in their respective drain tanks which are designed for salt freezing. Audit questions 4.2-7 and 9.6-18 address the staff's related questions regarding salt vapors.
4.3-10	Provide preliminary information about how welds will be performed on salt-wetted components. This includes the filler metal to be used, welding method, etc. Different weld filler materials are qualified to different allowable temperatures in ASME Code Section III Division 5. Describe how the chosen weld filler materials meet Code requirements for the maximum allowable temperature or provide appropriate commitments to ensure that there are appropriate materials available for the proposed use case and such materials will be procured for use in the MSRR.	ACU updated PSAR section 4.3.3 by letter dated July 30, 2024 (ML24219A258) to state that the weld filler will be ER 316. Audit question 4.3-21 was issued as a follow-up to ACU's response to this audit question.
4.3-11	Provide context on how the effects of stress relaxation cracking will be assessed, and what data will be used to determine the adequacy of degradation of welds in the fuel salt environment of the MSRR?	This audit question was superseded by audit question 4.3-24.

4.3-12	Provide context regarding chemistry control measures in place for the drain tank and/or RTMS (e.g., will these measures be needed)?	ACU stated in information provided for audit that active chemistry control of the drain tank is not planned as the salt will solidify when the reactor is not operational, therefore corrosion of the reactor drain tank is not expected to be an issue. In addition, ACU stated that salt is not expected to be in the RTMS during normal operation, thus no active chemistry control of the RTMS is planned.
4.3-13	Provide context on how heat transfer is considered and assessed when setting salt purity limits (e.g., effects from fouling, composition changes). If conservative values are assumed, provide context on how these values will be confirmed during operation (e.g., technical specifications).	ACU stated in information provided for audit that the results of ongoing experiments will be used to determine anticipated corrosion rates and fission product deposition and the effect that their presence has on fouling over time. This data will be incorporated into operational parameters to ensure that corrosion and fouling are minimized. In addition, ACU stated that PSAR section 13.1.4, “Reduction in Cooling,” analyses a full blockage of flow through either the reactor or coolant loop which found that the negative temperature

		feedback of the reactor causes reactor power to drop from just under 1 MWth to only a few tens of kWth over the course of a few minutes.
4.3-14	MSRE experience showed that when adding metallic Be for redox control, dendrites formed in the basket used to lower the Be into the salt. Provide context on how the effect of dendrite formation and potential flow blockages are considered.	ACU stated in information provided for audit that a Be rod will be dipped into the fuel salt in the main reactor vessel, instead of a secondary inlet, as needed to control the redox of the salt. Dendrites, if formed on the Be rod surface, are not expected to block any flow path of molten salt.
4.3-15	It is not clear to the NRC staff how the proposed redox probe arrangement accounts for potential local effects throughout the reactor system. As part of the audit, describe how redox probes mounted from the top of the RAV will able to adequately measure the redox potential throughout the fuel salt system.	ACU modified PSAR section 4.2.1.3 by letter dated July 30, 2024 (ML24219A258) to clarify that chemistry control of the system will not rely on redox probes. The staff will review the adequacy of off-line fuel sampling during an OL application review.
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);	This audit question was superseded by the staff's RAI 2 (ML23348A196) and audit questions 4.3-20, 4.3-22, 4.3-23, 4.3-24, and 4.3-25.

	• 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.	
4.3-17	Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) Preliminary Safety Analysis Report (PSAR), Revision 1 (ML23319A094), Section 4.3.5, states, "Considering that MSRR is a low-pressure system, and that the fuel salt chemistry will be tightly controlled to minimize corrosion susceptibility, we expect that up to 5 [displacements per atom (dpa)] for SS316 components will be acceptable. Further justification for this assumption as well as a more detailed assessment of maximum dpa to any component will be provided in the application for the Operating License." a. It is not clear to the staff what the term "…will be acceptable" means. For instance, the staff notes that this could mean that ACU does not believe that the cited level of irradiation will have any effect to degrade the material properties. Alternatively, it could mean that ACU understands that the material properties will be degraded, but that ACU considers the design margin to be sufficient to compensate for any loss capacity to accommodate operating conditions. (The staff notes that other explanations may also be possible.) Please clarify the intent of this term. b. The PSAR states, as quoted above, that this expectation is an "assumption" for which a "more detailed assessment" will be provided later. However, it is not clear to the staff how the word "assumption" should be interpreted, as used here; for instance, if	ACU modified PSAR section 4.3.5 by letter dated July 30, 2024 (ML24219A258) to state that SS 316H components in the reactor system will be exposed to less than 1 dpa and will use 1 dpa as a bounding fluence value when assessing the impacts of irradiation on safety-related components. ACU stated in its response dated April 30, 2024 (ML24121A272) to the staff's RAI 2 that the DMP will identify and assess degradation mechanisms applicable to safety-related components in the MSRR. This would include irradiation effects, as appropriate. Additionally, the DMP will be used to establish any necessary mitigation or monitoring measures needed

	this is based on any evaluation or analysis. Please clarify. The staff also notes that the phrase “more detailed assessment” could imply that some “less detailed” or preliminary assessment has been already performed to support the statement that 5 dpa will be acceptable. Please clarify whether any assessment has been performed, and if so, describe the preliminary assessment and what it entails (e.g., reduction of allowable stress values). c. The PSAR does not appear to address the effect of irradiation on the ER316 weld metal (see PSAR Section 4.3.3). In addition, references previously identified as part of the ongoing MSRR construction permit application audits (e.g., E. E. Bloom and J. R. Weir Jr., “Effect of Neutron Irradiation on the Ductility of Austenitic Stainless Steel,” Nuclear Technology, 16:1, 45-54 (1972); D. Kramer, K.R. Garr, A.G. Pard, and C.G. Rhodes, "Survey of Helium Embrittlement of Various Alloy Types” (1972); and A-A. Tavassoli, C. Picker, and J. Wareing, "Data Collection on the Effect of Irradiation on the Mechanical Properties of Austenitic Stainless Steels and Weld Metals," Effects of Radiation on Materials: 17th International Symposium, ASTM STP 1270, David S. Gelles, Randy K. Nanstad, Arvind S. Kumar, and Edward A. Little, Eds., American Society for Testing and Materials (1996)) do not appear to contain data for effects of irradiation on the weld metal. Clarify whether the PSAR statement relating to SS316 components above also pertains to the weld metal, and describe how any preliminary assessment has accounted for the impact of irradiation on degradation of the weld metal.	to ensure safety-related components can withstand the effects of degradation (including irradiation) in order to perform functions needed to meet safety functions, as well as applicable PDC. ACU confirmed by letter dated June 12, 2024 (ML24164A236) in response to the staff’s RCI, item 4.3-2 (5), that ACU will evaluate the effects of degradation on the weld metal (ER 316) as well. ACU cited chemistry control as a potential mitigation measure for irradiation damage. The staff notes that while chemistry control will likely help minimize certain effects of irradiation (e.g. potential for irradiation-assisted cracking), it likely will not mitigate other effects of irradiation (e.g. reduction in fracture toughness, helium embrittlement, effects on creep behavior). The staff will evaluate ACU’s mitigation strategy of all irradiation impacts, as appropriate, on safety-related components during review of an OL application.

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	a. ACU clarified in information provided for audit that the MHA aids in understandings the safety significance of SSCs. In particular, the degradation of the reactor vessel and/or RTMS would not result in an accident that has larger consequences than the MHA. b. ACU stated in information provided for audit that the RTMS will be designed to the same code and standard of the reactor vessel. In addition, ACU stated that the RTMS is included in the DMP which is described in ACU's response to the staff's RAI 2 (ML24121A272). ACU submitted a redacted enclosure, "MSRR Codes and Services Conditions," (ML24219A258) which lists the applicable codes to the RTMS. c. ACU clarified in information provided for audit that the intent of the statement is that available data will be used to inform design and operational controls to minimize the probability of a degradation consequence that is more

	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	severe than a small leak. In addition, ACU clarified that the usage of the word “most” is not intended to exclude the MSRR from being protected against all relevant degradation mechanisms. d. ACU clarified in information provided for audit that the intent of the statement was to explain that the probability for a vessel to fail catastrophically and/or energetically is reduced in the absence of a large pressure differential. In addition, ACU stated that the possibility of brittle failure from rapidly propagating cracks will be minimized or precluded based on available data and by design. Finally, ACU stated that the DMP, as described in its response to the staff’s RAI 2 (ML24121A272), will help ensure the MSRR design maintains a high margin against brittle failure.

	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.	
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.	ACU updated the PSAR by letter dated July 30, 2024 (ML24219A258) to state that construction will be in accordance with ASME BPVC Section III, Division 5 as endorsed by RG 1.87, Revision 2 and as adjusted for quality requirements. ACU stated during audit discussions with the staff that the operation of the MSRR will be within the bounds of RG 1.87, Revision 2.
4.3-20	The NRC staff notes that during elevated-temperature nuclear service, thermal aging will occur in which the microstructure of the structural alloy evolves. This microstructural evolution may involve changes in grain size, elements in solution, and precipitates. Precipitate evolution includes the type, location, and size. Thermal aging is dependent on time and temperature. Thermal aging is also applicable to welds. Thermal embrittlement may result from a number of processes that may occur during thermal aging. These may include the formation of hardening phases, accumulation of adverse elements on grain boundaries, segregation of impurities to dislocations, and changes in solid solution. The staff notes that a number of structures, systems, and components (SSCs) in the Abilene Christian University (ACU) Molten Salt Research	a. ACU stated in information provided for audit that PDC 10, 14, 30, 31, and 32 address the need to ensure adequate structural and mechanical integrity of the RTMS. b and c. ACU provided a preliminary example of how SS 316H components and welds would be analyzed for the effect of thermal embrittlement. The staff did not verify the adequacy of

	Reactor (MSRR) could be impacted by thermal aging. According to the document titled "MSRR Codes and Standards, service conditions and safety classification Version 2" which ACU provided for audit on August 10, 2023, SSCs that comprise the reactor fuel salt boundary including the drain tank are designed to be at 1202°F for 20 years, and the reactor thermal management system (RTMS) is also designed for such conditions. The staff notes that long term exposure of 316H stainless steel (SS) for the conditions described by ACU may precipitate embrittling intermetallics such as σ (W. Ren and L. Lin, "Consideration of Thermal Embrittlement in Alloy 316H for Advanced Non-Light Water Reactor Applications," Proceedings of the ASME 2019 Pressure Vessels & Piping Conference, PVP2019-93431 (2019)). Thermal embrittlement is not addressed in Section III, Division 5 of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, as stated in Subsubarticle HAA-1130. Nevertheless, the staff notes that this phenomenon should be evaluated if its effects on SSCs would challenge the conformance to relevant design criteria. Moreover, the staff notes that RTMS integrity is an assumption in the maximum hypothetical accident (MHA) scenario described in the MSRR Preliminary Safety Analysis Report (PSAR), Revision 1 (ML23319A094), Chapter 13. Therefore, the staff notes that the RTMS integrity needs to be maintained in order to ensure the MHA remains bounding of all potential events. It is not clear to the NRC staff if or how thermal embrittlement is accounted for in the analyses or evaluations that ACU has performed to demonstrate conformance with design criteria that relate to the integrity of the RTMS. During audit discussions of material degradation topics including Audit Question 4.3-16 (ML23123A044) on November 9, 2023, ACU indicated that it had not considered whether embrittling intermetallics such as σ could form in the SSCs during the SSCs 20-year design life. Additionally, the information provided for audit by ACU on September 29, 2023, in the document titled "Degradation Mechanisms Table September 29 Revision," does not appear to include sufficient information on this degradation mechanism and how ACU intends to address it. ACU stated verbally during the November 9, 2023, audit discussions that it would	ACU's response, as it was not necessary for the issuance of a CP. However, the staff understands that the mitigation of thermal embrittlement falls under the DMP which is described in ACU's response to the staff's RAI 2 (ML24121A272). Therefore, the staff determined the evaluation of ACU's mitigation strategy for thermal embrittlement can reasonably be conducted during an OL application review after the final design has been completed.

	evaluate this degradation mechanism and revise that table, if necessary. In order to assess this degradation mechanism, provide the following information to address the following questions: a. Identify which design criteria address the need to ensure adequate structural and mechanical integrity of the RTMS. b. Describe how the degradation mechanism discussed above will be mitigated for the MSRR (e.g., testing, design, inspection, surveillance coupons, performance monitoring). Have the results of any analyses or evaluations performed by ACU indicated that certain component attributes (e.g., wall thickness) should be chosen to ensure that there is sufficient design margin to accommodate the effects of thermal embrittlement? Have the results of any analyses or evaluations performed by ACU indicted that the potential for thermal embrittlement should be considered when ACU establishes plans for in-service inspection or monitoring of SSCs (this may include locations, frequencies, or methodologies)? If yes, what criteria will be used to determine that these analyses or evaluations are appropriate for their intended purpose? If surveillance coupons are utilized, what information will they provide, and what action will ACU take based on that information? Where will the surveillance coupons be located, if applicable? Describe how the surveillance coupons compare to the SSCs, if applicable (e.g., time and temperature profiles, and whether the coupons lead the SSCs). c. In addition to the considerations for the base metal described in question b., describe how these considerations apply to any affected welds.	
4.3-21	Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) Preliminary Safety Analysis Report (PSAR), Revision 1 (ML23319A094), Section 4.3.3, “Reactor System Structural Material,”	a. ACU stated in information provided for audit that it will use gas tungsten arc welding for any weldments with Type

	states the following: "Weld filler ER 316 is very similar in composition to the base metal and will be used as the weld filler material, as outlined in ASME BPVC.III.5-2017 Table HBB-I-14.1(b) 'Permissible Weld Materials.' Welding procedures will comply with appropriate sections of the codes and standards." a. Regulatory Guide (RG) 1.87, Revision 2, "Acceptability of ASME Code, Section III, Division 5, "High Temperature Reactors," does not endorse the material properties in Table HBB-I-14.10B-3 for Type 316 stainless steel (SS) base metal welded with Type 316 SS filler using processes other than gas tungsten arc welding; see RG 1.87 C.1.u.(1)(f). RG 1.87 states that "[a]pplicants wishing to use these base metal/weld metal combinations for welds made with processes other than gas tungsten arc welding may be able to demonstrate the adequacy of these [stress rupture factors] R-factors by submitting additional data." Has ACU determined that it will use gas tungsten arc welding or whether it intends to use a different process for which additional data may need to be provided? If the latter, please describe ACU's plans to submit the additional data. b. Has ACU determined if it will use a post weld heat treatment? If so, has ACU determined how the potential impact of the post weld heat treatment on materials' properties would be accounted for in the component design? c. It is not clear to the NRC staff what ACU means when it refers to the "appropriate sections of the codes and standards" at the end of the quoted section in the introduction of this question. For instance, the staff notes that this could be the codes and standards that are cited in ASME BPVC.III.5-2017 for ER 316. Alternatively, it could mean codes and standards for ER 316 that are not utilized by ASME BPVC.III.5-2017. The staff notes that ACU's response to audit question 5.2-4, provided via Electronic Reading Room (ERR), where it states that "[s]pecial processes like welds are	316 SS base metal and SFA 5.9 ER316 filler metal. b. ACU stated during audit discussions that a post-weld heat treatment is not anticipated for the MSRR. ACU's DMP will determine if a post-weld heat treatment is needed. The DMP is described in ACU's response dated April 30, 2024, to the staff's RAI 2 (ML24121A272). ACU confirmed by letter dated June 12, 2024 (ML24164A236) aspects of the DMP in response to the staff's RCI, item 4.3-2. The results of the DMP will be reviewed, as appropriate, by staff during review of an OL application. c. ACU modified PSAR section 4.3.3 by letter dated July 30, 2024 (ML24219A258) to state that construction will be in accordance with ASME BPVC Section III, Division 5 as endorsed by RG 1.87, Revision 2 and as adjusted for quality requirements.

	intended to be performed in accordance with the AWS D1.1. structural welding code." Other explanations could be possible. Please clarify the intent of this statement and clarify which code(s) and standard(s) will be utilized.	
4.3-22	The ACU document in the ERR titled "Degradation Mechanisms Table September 29 Revision.pdf" states that "During operation, the aim is to keep the salt under reducing conditions to mitigate corrosion… Fuel salt will be exposed to beryllium [Be] until the chemical analysis gives the desired ratios of uranium ions in salt. Beryllium will be removed once the desired ratio is achieved." In the same report, ACU further states that "when exposed to excess beryllium and carbon in FLiBe, S316H may form intermetallic compounds like Fe-Be or Ni-Be or metal carbides over time which may degrade the mechanical behavior of SS316H," but, according to the same document, this would affect a thin surface layer and the bulk properties would not be changed. The effect of excess Be additions on mechanical behavior of 316H SS has been demonstrated and documented (J. R. Keiser et al., "Interaction of beryllium with 316H stainless steel in molten Li2BeF4 (FLiBe)," Journal of Nuclear Materials, 565, 153698 (2022)). The same document provided by ACU also states that "SS316H does have high carbon content and that is expected to lead to carbide formation at the grain boundaries. Any extra carbon from graphite may form a carburized layer at the surface… However, the bulk properties will be more affected by the carbide formation due to the carbon alloyed into SS316H. Design calculations will account for these changes." Concerning the management of this potential material degradation mode, ACU states in its document titled "Degradation Mechanisms Table September 29," that "Inspection is not necessary since this is addressed in the design," and that "[surveillance c]oupons will be characterized by using [X-Ray Diffraction] XRD and metallography on sectioned samples to detect any phase changes in SS316H." The following questions address this information and, as relevant, apply to both base metals and welds:	The mitigation of embrittling mechanisms falls under the DMP which is described in ACU's response to the staff's RAI 2 (ML24121A272). ACU confirmed aspects of the DMP in its response to RCI 4.3-2 by letter dated June 12, 2024 (ML24164A236). Based on ACU's commitments with respect to the DMP, the staff determined that its evaluation of ACU's mitigation strategy for embrittling mechanisms can reasonably be conducted during an OL application review after the final design has been completed.

	a. Describe how ACU determined that the amount of Be needed to achieve the desired redox conditions is less than the amount which would affect the bulk properties of the material. b. ACU states in its document titled “Degradation Mechanisms Table September 29,” that beryllium addition technical specifications will be defined to preclude excess beryllium in the fuel salt. Describe how ACU will determine what amount of beryllium is excessive. c. With respect to phase formation embrittlement (Be), ACU states that "Inspection is not necessary because this is addressed by design." Please clarify the intent of the phrase “addressed by design” in this statement. Does this refer, for instance, to component dimensions or some other means to mitigate the potential propagation of cracks that initiate in an embrittled surface layer (e.g., synergism with other surface sensitive degradation mechanisms such as fatigue)? What criteria will be used to determine whether this potential degradation mechanism is satisfactorily addressed? d. Describe how ACU will account for the effect of additional carbon from graphite, including the potential for synergism with other surface sensitive degradation mechanisms, such as fatigue? e. Describe how ACU determined or will it determine whether XRD and metallography are adequate methods to detect phase formation embrittlement such that potentially affected components satisfy the relevant design criteria. Will surveillance specimen locations bound all potentially affected components? Does ACU plan to implement corrective actions if surveillance coupons indicate phase formation embrittlement?	
4.3-23	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	The mitigation of fission product induced cracking falls under the DMP which is described in ACU’s response to the staff’s RAI 2

	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	(ML24121A272). ACU confirmed aspects of the DMP in its response to RCI 4.3-2 by letter dated June 12, 2024 (ML24164A236). Based on ACU's commitments with respect to the DMP, the staff determined that its evaluation of ACU's mitigation strategy for fission product induced cracking can reasonably be conducted during an OL application review after the final design has been completed.

	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?	
4.3-24	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	The mitigation of SRC falls under the DMP which is described in ACU’s response to the staff’s RAI 2 (ML24121A272). ACU confirmed aspects of the DMP in its response to RCI 4.3-2 by letter dated June 12, 2024 (ML24164A236). Based on ACU’s commitments with respect to the DMP, the staff determined the evaluation of ACU’s mitigation strategy for SRC can reasonably be conducted during an OL application review after the final design has been completed.

	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.	
4.3-25	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	1, 2, and 3. These audit questions were addressed by the establishment of the ACU DMP as described in ACU's response to the staff's RAI 2 (ML24121A272). The DMP will allow ACU to identify the effect of irradiation on MSRR structural materials (including weldments). Because the ACU DMP will be used to determine effects of irradiation, and take appropriate action, the staff determined that its evaluation of these areas can reasonably be conducted during an OL application review. 4. ACU confirmed in response to the staff's RCI, 4.3-2(9) (ML24164A236) that it will calculate helium generation in structural alloys and provide this information in an OL application. 5. This audit question was

	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 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%	addressed by the establishment of the ACU DMP as described in its response to the staff's RAI 2 (ML24121A272). The DMP will be used to determine appropriate in-service monitoring and inspection techniques. 6. ACU modified PSAR table 3.4-1, "Safety and Seismic Classification of Structures, Systems, and Components" by letter dated July 30, 2024 (ML24219A258) to state that the core support structures are safety-related and that their safety function is to prevent graphite movement that would damage the reactor vessel. The DMP will be used to determine how irradiation may impact the core support structures' safety function.

	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- creep-fatigue and weld creep cracking: a summary of design and fitness-for-service practices," January 2020.	
Additional Topics: Graphite Materials Properties	The staff discussed graphite moderator materials properties with ACU during the audit.	ACU stated during audit discussions that the preliminary graphite grade had been selected for use in the MSRR. ACU provided select data to the staff on the ERR related to irradiation response of a similar graphite grade from the same manufacturer. The staff noted that not all properties that are typically required by ASME BPVC, Section III, Division 5, were included in the provided data but, because the MSRR is a research reactor, which will likely experience a very low fluence to the graphite, the parameters provided for the irradiation data (fluence and temperature) will likely bound the parameters that the graphite in the MSRR will experience. During the audit, ACU stated that the impact of neutron irradiation will be quantified. ACU stated that an OL application will contain acceptable ranges for a

		variety of graphite properties and ACU noted that, where applicable, the information will be provided as a function of temperature. Additionally, ACU provided information related to unirradiated properties of the candidate grade, as well as the standards used to determine these properties. ACU stated that these data will be assessed against quality assurance program requirements in the OL application. The staff will assess the available data during an OL application review, as appropriate, considering ACU stated the graphite moderator will be non-safety related and is not needed to satisfy any PDC.
Additional Topics: Fuel Salt Infiltration	The staff discussed the potential for fuel salt infiltration into the graphite pores.	ACU provided references that discuss the potential for fuel salt infiltration into graphite. The references provided were ORNL/TM-2020/1621, “Progress Report on Graphite-Salt Intrusion Studies,” ORNL/TM-2021/2247, “FY21 Progress Report on Graphite-Salt Interaction Studies,” and Vergari, L., et. al., “Infiltration

		of molten fluoride salts in graphite: Phenomenology and engineering considerations for reactor operations and waste disposal," Journal of Nuclear Materials 572 (2022). The staff and ACU also discussed the parameters that affect salt infiltration. The staff notes that the equation to determine the pressure differential needed to promote salt intrusion into graphite is relatively well established, and the factors that determine salt intrusion are discussed in ORNL/TM-2020/1621. This report from ORNL also contains intrusion tests on a similar grade of graphite to the MSRR candidate grade. The staff will confirm during an OL application review that ACU sets appropriate controls (e.g. TS to limit system pressure) to ensure conditions for fuel salt infiltration into graphite pores will not be met. Additionally, the relatively low pressure of the MSRR reactor system helps to mitigate the potential for fuel salt intrusion into graphite.
Additional Topics:	The staff discussed the potential for decarbonization of the graphite.	The staff notes that if the oxidation-reduction (redox)

Graphite Decarbonization		potential of the fuel salt is overly reducing, the potential for decarburization of the graphite exists. However, as described in PSAR section 4.2.1.6, “Stainless Steel SS316H Corrosion Monitoring and Control,” ACU will provide a detailed description of the MSRR chemistry control program with the OL application. The staff determined it is reasonable to defer this portion of the review until an OL application review.
Additional Topics: Graphite Failure	The staff discussed the potential for graphite failure.	PSAR table 3.4-1 lists the graphite core as non-safety-related. The staff discussed with ACU several potential ways the MSRR graphite moderator could fail including, but not limited to, the potential for graphite to impact reactor vessel integrity, whether a broken piece of graphite could impact draining, and how graphite restraints are configured. ACU confirmed in its response dated June 12, 2024 (ML24164A236) to the staff’s RCI, item 4.2-1(1), that it will analyze postulated failures of graphite components and confirm that these would not

		impact the ability of another component to perform its safety function or cause an unanalyzed accident, as would be appropriate in order to support the non-safety-related classification of the graphite moderator.
Additional Topics: Use of Codes for Graphite	The staff discussed the application of codes and standards for the graphite moderator.	PSAR section 4.2.3, does not identify the use of ASME BPVC, Section III, Division 5, for graphite components, which is not mandatory for research reactors. During audit discussions with the staff, ACU indicated the potential to use ASME BPVC, Section III, Division 5, rules to inform the design of the MSRR moderator. ACU confirmed in its response dated June 12, 2024 (ML24164A236) to the staff's RCI, item 4.2-1(2) that a description of which, if any, ASME Code Section III, Division 5 rules for graphite will be used for the MSRR, and the basis for which rules were used, will be provided in an OL application. The use of an NRC-endorsed standards (or parts of it) will help to ensure the correct information is considered when designing graphite components. During

		the review of the OL application, the staff will review final design information of graphite components, as well as programmatic elements of the MSRR such as performance monitoring programs, and technical specifications.

Resolution of Questions on Section 4.4, "Biological Shield"

Question Number	Question	Resolution
4.4-1	PSAR Section 4.4.4, "Design Methods," the applicant states: *"In general, Monte Carlo methods are used for shielding analyses to allow accurate representation of the MSRR geometry and, in particular, to enable accounting for penetrations directly."* Based on the staff's understanding of the statement, please confirm what codes are used to perform these shielding calculations.	ACU modified PSAR section 4.4.4 by letter dated July 30, 2024 (ML24219A258) to state the computer codes used in their analysis.

4.4-2	The staff requests to audit/review the shielding calculation files from the previous question, the Monte Carlo Neutral Particle Transport Program (MCNP) input and output files, along with any other supporting documentation. This review is conducted so that the staff may verify the shielding specifications and the targeted dose rates specified for the areas around the reactor. a. Seeking information on the source term used for the bioshield analysis, the materials used in the analysis, and the geometry used in the analysis. The staff also requests available information on how air activation and soil activation are determined.	ACU provided the requested files to the staff for review on the ERR. The provided calculation packages considered multiple sources of radioactivity. The dose rate assessment considers the reactor core source term, as well as the activation source term from multiple components around the reactor. The result of ACU's calculation shows that the specified 2.5 mrem/hr dose rate in the research bay and covered systems pit is achievable based on the available shielding from radiation sources within the reactor cell. The staff reviewed the information provided in response and found that the audit discussions and reference material made available allowed the staff to understand and verify the information used for their bioshield analysis, air activation analysis, and soil activation calculation.

Resolution of Questions on Section 4.5, "Nuclear Design"

Question Number	Question	Resolution
4.5-1	The staff requests the Reactor Physics Model of the MSRE stated to be available upon request in Section 4.5.1.3 to be made available.	ACU provided the Reactor Physics Model of the MSRE to the staff for audit.
4.5-2	The limiting parameters that are of safety interest (highest power density, largest source term, and highest excess reactivity are listed but not quantified with acceptance criteria. Will acceptance criteria be defined for these parameters? If so, when (e.g. PSAR, OL, UFSAR)?	ACU modified PSAR section 4.5.2.1, "Limiting Parameters That Are of Safety Interest," by letter dated July 30, 2024 (ML24219A258) to state that power density, source term, or excess reactivity will not be directly limited during operation of the MSRR. Instead, the MSRR will be designed and licensed where those parameters are limited through control of reactor power, control rod height, and reactor temperature. ACU indicated in information provided for audit that acceptance criteria for these operational and limiting parameters will be provided in an OL application.
4.5-3	A 0.04 percent void fraction was chosen based on the MSRE. Is this consistent with the values assumed in calculations (e.g., those in Chapter 13, which appears to use a He entrainment fraction of 0.018, consistent with the discussion in Section 4.5.4)? The void fraction needs to be realistic, because overestimating voids can assume a more negative reactivity effect than would be present in the core	ACU modified PSAR section 4.5.2.2, "Impact of Gas Management System on Reactor Operation and Safety," by letter dated July 30, 2024 (ML24219A258) to specify that the void fraction will be established and

	during operating conditions.	validated by testing in TS which will be provided in an OL application. Uncertainty is eliminated by directly measuring the void fraction by testing.
4.5-4	Regarding the Off Gas system impact on reactivity, the system is assumed not to run for the source term and power density for conservatism. However, the system running has a non-conservative impact on the void reactivity coefficient. What is the treatment and effects of the off gas system for transients where the void reactivity coefficient would be the most limiting reactivity parameter.	ACU provided discussion and supporting information on the ERR that indicated that the off-gas management system bubbler would not affect the overall void fraction in the RCS loop outside the RAV. In addition, ACU modified PSAR section 4.5.2.2 by letter dated July 30, 2024 (ML24219A258) to add discussion of the off-gas management system bubbler impact on void fraction and to state the TS will be provided in an OL application which will require monitoring of reactivity during bubbler operation.
4.5-5	When evaluating the reactivity coefficient in the last paragraph of 4.5.2.2 a temperature of 600 degrees C was used. This is the average system temperature not the minimum system temperature of 550 degrees C. Shouldn't the minimum temperature be used, as that is the most limiting per the last paragraph of section 4.5.2.1?	ACU modified PSAR section 4.5.2.2 by letter dated July 30, 2024 (ML24219A258) to include the conclusion that the 600 degrees Celsius (°C) reactivity coefficient encompasses the entire operating range. In addition, ACU provided supporting information for audit that the anticipated density changes across the operating range are

		not significant, making the 600 °C reactivity coefficient value indicative of the entire range.
4.5-6	What safeguards are in place to ensure that the correct amount of UF4 is added through the RAV without effectively increasing or decreasing the mole fraction of UF4 in the fuel salts? Over-addition of UF4 would invalidate many of the assumptions in section 4.5.2, and if this is a realistic possibility then these scenarios should be analyzed as reactivity control accidents.	ACU modified PSAR section 4.5.2.3, "Reactivity induced by fuel salt composition changes," by letter dated July 30, 2024 (ML24219A258) to include a basic description of the UF_4 slug addition which details acceptable safeguards for reactivity management and control that preclude inadvertent over-fueling of the reactor.
4.5-7	Has any evaluation been done to ensure that plutonium buildup from U-238 through core life is managed and that the fast fission cross section does not have an adverse impact on fuel salt subcriticality margin outside of the reactor vessel?	ACU stated in information provided for audit that Plutonium buildup has been modeled over the lifetime of the reactor and that the fuel remains subcritical outside the reactor vessel under all conditions. In addition, ACU modified PSAR section 4.5.2.6, "Analysis Showing the Fuel Salt Outside the Reactor Vessel Remains Subcritical at All Times," by letter dated July 30, 2024 (ML24219A258) to include a statement that spontaneous fission from Pu-239 at end of core life will not affect the k_{eff}

		acceptance criteria of < 0.6 outside the reactor vessel.
4.5-8	The thermal expansion described between the vessel and the graphite portends uncertainty about the molten salt flow through the core. What is the expected vessel flow fraction between: a. Fuel Salt Channels b. Annular region between moderator blocks and reactor vessel wall c. Flow between the adjacent surfaces of the graphite moderator block	ACU modified PSAR section 4.5.3.2, "Power Profile in Reactor Vessel," by letter dated July 30, 2024 (ML24219A258) to add a description of a flow distributor device at the bottom of the reactor vessel that will ensure proportional even flow across the core.
4.5-9	Will the core end of life (EOL) prompt neutron lifetime calculation in the OL account for plutonium generation and fast fission cross section?	This question was addressed through the resolution of audit question 4.5-7.
4.5-10	Will the variance in axial or radial power density across the core cause any deformation to the flow path, particularly in the graphite channels?	ACU provided information for audit about the expected graphite thermal expansion coefficient that indicated that flow through the fuel channels in the core would not be affected. ACU plans to provide the exact graphite thermal expansion during in an OL application.

4.5-11	Is the graphite moderator expected to react chemically to any of the fuel salts, fission, or activation products?	ACU modified PSAR section 4.2.3, "Neutron Moderator and Reflector," by letter dated July 30, 2024 (ML24219A258) to state that the graphite moderator is not expected to react chemically with the fuel salt and its constituents. Fission products in the salt may or may not interact with the graphite and ongoing research is being performed to determine those effects which will be used to determine appropriate TS values for the fuel salt and will be included in an OL application.
4.5-12	NRC staff is seeking further information to support the assertion in 4.5.4.6 that "it is concluded that the MSRR does not have a limiting core configuration" in conjunction with the statement that "[hot spots] are monitored during operation and can be accurately modeled with appropriate tools." Staff recognizes that the term "limiting core configuration" may not be applicable in this instance, and will be seeking to ensure that limiting conditions are assessed as appropriate for the events analyzed in Chapter 13.	Through audit discussions with ACU the staff gained an understanding of the statement that the MSRR does not have a limiting core configuration. Specifically, the staff understands that there are no variable core SSCs and therefore the normal operating configuration of the core is the only core configuration.
4.5-13	Why is the maximum reactivity insertion pressure initial condition 150kPa in 4.5.4.4 so different from the maximum operating pressure of 500kPa? Also, what is the anticipated normal operating pressure, assuming that the maximum of 500kPa is a safety limit?	This question was addressed through the resolution of audit question 4.2-9.

Resolution of Questions on Section 4.6, "Thermal Hydraulic Design"

Question Number	Question	Resolution
4.6-1	According to PSAR 4.6, Thermal-hydraulic (TH) design cooling capacity is consistent with PDC 34 and 35, which is sufficient to maintain the salt at temperatures that will not damage the reactor systems. It is not clear how pressure is regulated in the heat removal system during normal or postulated accidents without a pressure relief valve(s). Does the MSRR TH design or RTMS require any pressure relief valve(s)?	ACU stated in information provided for audit that pressures throughout the reactor and heat removal systems will be controlled by the GMS which will be regulated via pressure valves. Further details on the GMS pressure relief functions are provided in the resolution to audit question 9.6-4.
4.6-2	PSAR 4.6.1.1 states the radiator is cooled by atmospheric air and Figure 4.6-1 labels this as "PHR" air, but PHR is not defined in the list of acronyms or described in the text. What is the PHR?	ACU modified the PSAR acronym list by letter dated July 30, 2024 (ML24219A258) to clarify that PHR refers to the primary heat removal system.

Resolution of Questions on Section 6.2, "Detailed Descriptions"

Question Number	Question	Resolution
6.2-1	NRC staff is seeking additional information regarding the confinement function for fuel salt storage tanks and off-gas system components, which represent potential pathways for radionuclide release.	ACU modified PSAR sections 9.2.3 and 9.6.3, "Operational Analysis and Safety Function," by letter dated July 30, 2024 (ML24219A258) to add descriptions of low-leakage enclosures around the fuel salt storage and off-gas

		components, respectively. Related issues regarding enclosure penetration isolation valve performance were addressed by audit question 3.5-4.
6.2-2	It is not clear based on the information provided in the PSAR what the applicability of PDC to the reactor cell is. NRC staff requests that ACU provide context on how the reactor cell function is addressed within the proposed PDC.	ACU added PSAR section 6.2.1.8, “Reactor Cell Compliance with Design Criteria,” by letter dated July 30, 2024 (ML24219A258) to provide discussion of the applicable PDC to the reactor cell.
6.2-3	NRC staff is seeking additional context on the cited 1 percent per day leakage rate. Specifically, the staff requests ACU provide context on how this value is planned to be confirmed or achieved (e.g., construction codes, methods, planned tests). This question also pertains to some of the information needs submitted regarding Chapter 13.	ACU modified PSAR section 6.2.1, “Confinement,” by letter dated July 30, 2024 (ML24219A258) to describe the extent of the pre-operational test program to verify reactor cell leak rate is within design for a variety of plant conditions and the construction basis supporting the low leak rate.
6.2-4	ACU should provide context (make the analysis available and potentially summarize relevant findings) regarding the preliminary analysis to support the following PSAR statement: “Sufficient thermal mass exists within the reactor system, reactor enclosure, reactor cell, and research bay to ensure completely passive cooling mechanisms (conduction, natural convection, and thermal radiation) safely remove decay heat without violating a thermal limit.”	ACU provided analysis of a loss of off-site power event for the staff to review during the audit. The analysis showed that the reactor cell absorbs the excess heat from the reactor system.

6.2-5	Provide the supporting basis (e.g., analysis, preliminary assumptions, and material properties) for the following statements in 6.2.4: "The stainless steel is capable of withstanding, without failure, the direct contact of approximately 1.5 tons of hot fuel salt falling on it and collecting at the bottom of the RTMS," and "In the event heater power is lost, the RTMS and the components inside will slowly cool. This cooldown period is long enough for the fuel salt to drain."	ACU provided supporting analyses on the ERR for the staff to review during the audit. In addition, ACU modified PSAR section 6.2.4.2, "Design Bases of the Reactor Thermal Management System," by letter dated July 30, 2024 (ML24219A258) to clarify the design bases of the RTMS. ACU also corrected "air" to "nitrogen," in PSAR section 6.2.4.3, "Design Parameters Needed for Analysis," and provided additional information in PSAR section 6.2.2.1, "Safety Functions of the Reactor Enclosure," clarifying how the reactor enclosure and other safety-related vessels are protected from failure of the nitrogen supply.
6.2-6	NRC staff requests ACU provide preliminary analysis as part of the audit for the statement "The utility of the RTMS is demonstrated in the Chapter 13 MHA analysis." Although the system is utilized in Chapter 13, information in the PSAR is insufficient for the staff to understand how the RTMS heat removal capability is assumed to function when relied upon (e.g., system configuration, assumed heat transfer characteristics).	ACU provided supporting analyses on the ERR for the staff to review during the audit. In addition, ACU modified PSAR section 6.2.4, "Reactor Thermal Management System," by letter dated July 30, 2024 (ML24219A258) to clarify the design of the RTMS.

6.2-7	It is not clear from the information presented in PSAR 6.2.4 how the RTMS meets the DC 72 function (heat retention) under accident conditions (which accidents, capability of insulation). ACU should provide additional context regarding the conditions under which the RTMS is credited to retain heat (e.g., keep the salt liquid) and which the RTMS removes heat. A preliminary analysis, including any assumptions, should be provided as part of the audit.	ACU provided supporting analyses regarding a loss of off-site power event on the ERR for the staff to review during the audit. In addition, ACU stated in information provided for audit that the RTMS is surrounded by insulation, therefore the RTMS will cool down gradually upon loss of power.
6.2-8	PSAR Section 6.2.4.3 states the RTMS is independently supported by the reactor loop. Provide clarification if this statement refers to the thermal management aspect (e.g. salt temperature control) or mechanical function.	ACU clarified in information provided for audit that this refers to structural support and that details will be provided in an OL application.

Resolution of Questions on Section 6.3, “Compliance with Design Criteria”

Question Number	Question	Resolution
6.3-1	No information was provided regarding technical specifications that may be needed for ESF or manual isolation of fuel salt storage tanks. ACU should provide context on why these are not present, or plan to add these to the list of variables for preliminary TS in Chapter 14.	ACU modified PSAR section 9.2.3 by letter dated July 30, 2024 (ML24219A258) to add discussion of proposed TS addressing fuel salt enclosure leak rate, these TS will be provided in an OL application.

Resolution of Questions on Section 9.6, "Gas Management System"

Question Number	Question	Resolution
9.6-1	ACU should describe how the design of the GMS is planned to meet each of the applicable ACU MSRR DC.	ACU modified PSAR section 9.6.1, "Design Basis," by letter dated July 30, 2024 (ML24219A258) to clarify the design basis of the GMS. ACU provided additional detail regarding the GMS design basis in a proprietary enclosure included in its response dated March 28, 2024 (ML24088A324) to the staff's RAI 1. A redacted version of this enclosure was provided by ACU by letter dated July 30, 2024 (ML24219A258).
9.6-2	Clarify where portions of the GMS and cover gas system are located. What subsystems and components are located inside the reactor enclosure, reactor cell, and outside of these buildings? Clarify what components in the GMS are safety related.	During audit discussions between ACU and the staff, ACU provided information that clarifies where different components in the GMS and cover gas systems are located. In addition, ACU provided a proprietary process flow diagram in its response dated, March 28, 2024 (ML24088A324) to the staff's RAI 1 which provides additional detail on GMS component locations, materials of construction, safety classifications, as well as Code classification.

9.6-3	PSAR Section 9.6.1 states that one of the design bases for the GMS is to maintain integrity to limit dose releases. ORNL/TM-2020/1478 Section 9.6 states that the GMS should be able to withstand any pressure transients in the reactor system. Provide the design basis and preliminary postulated accident temperatures and pressures, or bounding assumptions, if transient analysis details are not available to demonstrate that the GMS can maintain boundary integrity under all postulated conditions. Additionally, describe the pressure relief functions mentioned in PSAR Section 9.6.2.	During audit discussions between ACU and the staff, ACU provided information to describe how the GMS will be able to accommodate pressure transients. ACU modified PSAR section 9.6.2, "System Description," by letter dated July 30, 2024 (ML24219A258) to state that the GMS is designed to the same standards as the reactor system, those standards are ASME BPVC Section III Division 5. In addition, ACU modified PSAR section 3.5.1 to include load combinations for safety-related components (including potential gas pressures).

9.6-4	PSAR Section 9.6.1 states that one of the design bases is to support reactor protection system (RPS) functions. Provide context as to which RPS functions the GMS supports. Explain how the GMS performs the pressure equalization safety function.	ACU provided information for audit that explains the GMS is needed to shut down the reactor. The GMS supports shutdown by equalizing the pressure in the gas space in the RAV and the reactor drain tank. The GMS will have three interconnected paths with two valves in series, so that a reactor shutdown only requires two of six valves to open. Additionally, as discussed in audit question 3.5-4, the valves will be appropriately qualified to ensure they can perform their safety functions. ACU also noted that valves will be cycled during normal shutdowns to ensure operability and that changes in gas pressure and vessel level will be monitored in accordance with proposed TS.
9.6-5	It appears that PSAR Section 9.6, “Gas Management System,” does not specifically discuss storage and removal of tritium and section 9.6.3 states that only iodine, xenon, and krypton will enter the scrubber beds. How does the GMS handle tritium storage, removal, and processing so that dose limits are not exceeded during normal operations or a postulated accident? Additionally, how does the GMS handle decay products from iodine, xenon, or krypton?	During audit discussions between ACU and the staff, ACU stated that the MSRR design will not offer substantial hold-up of tritium and that dose calculations assume all tritium is eventually released to the atmosphere. ACU also stated that tritium hold-up in the off-gas scrubber bed is not accounted for in licensing. The staff notes that the evaluation of tritium releases during normal

		operations is discussed in chapter 11 of the staff's SE, and releases during postulated accidents are described in chapter 13 of the staff's SE. ACU stated in information provided for audit that the scrubber beds will be designed to hold-up decay products of fission products like xenon or krypton.
9.6-6	PSAR Section 4.3 references certain construction codes that will be used for components in the reactor system but doesn't specify which codes are used for what components/subsystems. ACU should specify the construction codes that will be used for the GMS and why the chosen codes are appropriate given the safety significance of the GMS system and components.	ACU provided a proprietary enclosure in its response, dated, March 28, 2024 (ML24088A324), to the staff's RAI 1. This enclosure provides the materials of construction for all safety-related components, and the quality standards that apply to the safety-related structures, systems, and components (SSCs). A redacted version of this enclosure was provided by ACU by letter dated July 30, 2024 (ML24219A258). In addition, ACU added PSAR section 3.1.4, "Codes and Standards Used by the MSRR" by letter dated July 30, 2024 (ML24219A258) which lists the applicable codes and standards for the MSRR.

9.6-7	ORNL/TM-2020/1478 Section 9.6 states that the GMS should be designed to ensure there are adequate decay heat removal mechanisms to ensure a credible failure would not lead to loss of fuel system boundary integrity, and Section 9.6.1 of the ACU MSRR PSAR, “Design Basis,” states the GMS is designed to remove decay heat. Section 9.6.2 states decay heat will be “appropriately managed.” However, there is no description of how this will be achieved. Describe how decay heat is removed from the GMS to ensure that there is no loss of the fuel system integrity boundary.	During audit discussions between ACU and the staff, ACU stated that it has preliminary models for decay heat removal in the reactor enclosure via natural convection, which is contained in RELAP5-3D models simulating loss of off-site power scenarios. ACU also stated that the detailed analysis of decay heat removal from the GMS has not been completed. The staff will review the detailed analysis of decay heat removal in the GMS during an OL application review to ensure integrity of the GMS is maintained.
9.6-8	ACU MSRR PSAR Section 9.6.2 states that hydrogen will be used in the GMS. ORNL/TM-2020/1478 Section 9.6 states that the GMS should be designed so that acceptable concentrations of constituents (includes processing, storing, and recombination of reactive gases) are maintained. Describe how the design of the GMS allows for appropriate processing, storage, and recombination, if necessary, of reactive gases.	ACU modified PSAR section 9.6.3, by letter dated July 30, 2024 (ML24219A258) to state the GMS has the ability to scrub hazardous materials from the gas stream, including anhydrous hydrogen fluoride (HF) and H_2. The staff will review the limits regarding the release of hazardous materials and handling of reactive gases as part of its review of an OL application

9.6-9	ACU PSAR Section 9.6 states that actinides are not gaseous and so criticality in the GMS is prevented, although there was evidence from the MSRE that a fuel salt mist may be deposited within the off-gas system. ORNL/TM-2020/1478 Section 4.7 specifies that an analysis should be provided to demonstrate that no single failure can result in criticality outside the active reactor core. Given these considerations, how is inadvertent criticality in the GMS prevented?	ACU modified PSAR section 9.2.3 by letter dated July 30, 2024 (ML24219A258) to include discussion of how criticality outside the core is precluded (including the GMS). The staff notes that PSAR section 9.6.4, "Instrumentation and Control Requirements," states that bulk transport of actinides to the GMS will not occur. The staff did not make any findings or conclusions related to whether this can be entirely precluded. The staff also notes that transport of fissionable material to the GMS via helium sparging may occur. However, the MSRR will have measures in place to monitor radioactivity in the GMS as well as the analysis discussed in audit question 4.5-7 to demonstrate criticality cannot occur outside of the core.
9.6-10	ORNL/TM-2020/1478 4.7 states that the maximum release of hazardous chemicals should not exceed applicable regulatory criteria including effects on workers in the facility. Describe how hazardous materials such as F_2 or HF are scrubbed from the GMS in order to ensure releases don't exceed applicable regulatory criteria.	ACU modified PSAR section 9.6.3, by letter dated July 30, 2024 (ML24219A258) to state the GMS has the ability to scrub hazardous materials from the gas stream. ACU confirmed in its response to the staff's RCI, item 9.6-1(2), that HF gas will be appropriately handled and there is sufficient basis for release limits. The

		staff will review the limits regarding the release of hazardous materials and handling of reactive gases as part of its review of an OL application.
9.6-11	ORNL/TM-2020/1478 Section 9.6 states that the GMS should have systems to assess the required purity or concentrations of contained gases. Provide context regarding how corrosive impurities (e.g. air, moisture) in the gas system will be monitored to ensure the appropriate fuel salt redox potential is maintained.	During audit discussions between ACU and the staff, ACU noted that the sampling ports described in section 9.6.4 of the PSAR can be used to monitor gas impurities. This can include impurities that would degrade the GMS boundary. In addition, ACU modified PSAR section 14.3, “Limiting Conditions for Operation,” by letter dated July 30, 2024 (ML24219A258) to include GMS hydrogen concentration and GMS activity as limiting conditions for operation. The staff will evaluate the GMS sampling plans as part of its review of an OL application.
9.6-12	ORNL/TM-2020/1478 Section 9.6 states that the GMS should provide periodic monitoring for long-term accumulation of fissionable material in the system. Section 9.6.4 includes monitoring for scrubber bed activity, but it isn’t clear if this can be used to monitor accumulation of fissionable material throughout the GMS as this only appears to monitor one component and doesn’t seem to discriminate between fission products and fissionable material. Describe how fissionable material in the GMS is planned to be monitored.	During audit discussions between ACU and the staff, ACU stated the MSRR will provide for sampling of the gas space, which would allow for measurement of radionuclide contents. The GMS will also have radiation monitors which could help to detect fissionable material in the GMS. In addition, ACU stated it intends

		to account for the phenomena of helium sparging entraining salt via design features. These design features would be located in the RAV.
9.6-13	The GMS uses He which is much less dense than ambient air. What are the consequences of air ingress in sub-systems such as the primary salt tanks which can't drain to the RTMS?	ACU modified PSAR section 6.2.2.1 by letter dated July 30, 2024 (ML24219A258) to state that the reactor enclosure atmosphere will be inert with nitrogen gas, which helps to minimize the potential of air incursion into the GMS. In information provided for audit, ACU stated that the type of gas in the GMS does not affect the RPS function of pressure equalization in the GMS. In addition, ACU stated that air ingress into the sub-systems would result in contamination of the fuel salt with oxygen and moisture, requiring the fuel salt to be purified.
9.6-14	ACU MSRR DC 73 requires provisions to prevent or mitigate plugging of gas lines due to salt solidification which could prevent the safety related (SR) function of pressure equalization. Additionally, ORNL/TM-2020/1478 Section 9.6 states that monitoring should be provided for hazardous chemicals and fission products to detect build-up, clogging, and leaks. PSAR Section 9.6.3 states hazardous chemicals and fission products will be monitored for plugging and leaks, but doesn't describe how this will be accomplished. Describe how plugging of gas lines, or leaks are prevented or mitigated, and demonstrate that the GMS can provide pressure equalization during a postulated accident and perform any other required safety functions.	ACU modified PSAR sections 4.2.2.1, "Reactor Protection System," and 9.6.1 by letter dated July 30, 2024 (ML24219A258) to describe monitoring measures that can be used to ensure that the valves needed for pressure equalization have not become plugged or leak to the degree that prevents the GMS from performing its safety function.

		Additionally, during the audit discussions ACU stated that corrective actions if plugging is detected have not yet been finalized. However, ACU described potential actions such as insulating certain piping, or providing heat tracing.
9.6-15	DC 74, NUREG-1537, and ORNL/TM-2020/1478 stipulate that an analysis should be provided that demonstrates if the gas characteristics are changed, that it will not impact safe shutdown. The GMS performs the SR function of pressure equalization. If the characteristics of the gas are changed, can pressure equalization be achieved? Demonstrate that events such as air leaks do not inhibit the ability of the GMS to perform its SR functions.	ACU committed to valve qualification under ASME QME-1-2017, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” in a proprietary enclosure included in its response dated March 28, 2024 (ML24088A324) to the staff’s RAI 1. A redacted version of this enclosure, “Safety Related Valve Requirements,” was submitted by letter dated July 30, 2024 (ML24219A258). The commitment to ASME QME-1-2017 helps to ensure the safety-related valves can perform under the conditions that will be experienced. Additionally, ACU confirmed in its response dated June 12, 2024 (ML24164A236) to the staff’s RCI, item 9.6-1, that changing characteristics of the gas will not impact the pressure equalization function.
9.6-16	If the fuel salt is drained to the drain tank or RTMS in a postulated accident, is the GMS still credited to perform its function to retain radionuclide gases from	ACU modified PSAR section 9.6.3 by letter dated

	the drain tank or RTMS? Provide context as to the preliminary expected GMS function during these events.	July 30, 2024 (ML24219A258) to add descriptions of GMS isolation. The PSAR states that penetrations of the off-gas enclosure are protected by isolation valves. The PSAR also states that these valves isolate the off-gas system in the event the system deviates from operating limits such as stack activity or off-gas pressure. Audit question 3.5-4 addresses questions regarding enclosure penetration isolation valve performance.
9.6-17	How is the GMS piping isolatable as described in Section 9.6.3?	This question was addressed through the resolution of audit question 9.6-16.
9.6-18	Describe how fission product partitioning between the fuel salt and the gas space is determined. The thermochemistry of the salt will determine whether certain fission products exist in solution or as a vapor and could increase the quantity of radionuclides available for release from the GMS.	During audit discussions between ACU and the staff, ACU stated that a thermochemical analysis would be performed to determine the partitioning of fission products between the reactor system and the GMS and be provided for an OL application. The analysis will consider noble metal transport and contribution to dose. The staff notes that MSRE operating experience may inform fission product partitioning and transport.

9.6-19	Integrity of the GMS boundary: a. ACU MSRR DC 42 requires inspection of containment atmosphere cleanup systems and DC 43 requires testing of containment atmosphere cleanup systems. PSAR Section 9.6.5, “Technical Specifications, Testing, and Inspection,” only states that a surveillance program is established to ensure barrier integrity. Does the program referenced in PSAR Section 9.6.5 include periodic inspection of important components and functional testing? b. It appears the ACU MSRR DC 51, “Fracture prevention of containment boundary,” is missing from the evaluation in Section 9.6. It appears the GMS forms part of the containment boundary as it is needed to retain gaseous fission products generated in the fuel salt. This DC requires the containment be designed with margin to ensure probability of rupture is minimized and to account for material properties, stresses, and flaws. Clarify whether ACU MSRR DC 51 applies to the GMS and, if so, describe how it is met. c. ORNL/TM-202/1478 Section 4.7 contains an evaluation finding that the applicant designed the system to be compatible with the chemical environment to which it will be exposed. ACU should demonstrate that the 316H and associated weld filler material is compatible with the gases to which it can be exposed (e.g. HF at different concentrations/temperatures, H_2 which can cause cracking/embrittlement, Helium embrittlement, temperature, dose) as well as the volatile fission products and fuel salt mist that can deposit in the GMS, or describe what measures will be in place to preclude adverse interactions. d. ACU should provide context regarding the interaction between gaseous fission products and the GMS boundary (e.g., by increasing temperature, cracking, or deposition causing blockage of gas pathways) e. ACU should describe preliminary measures regarding how the GMS will	a. ACU modified the PSAR by letter dated July 30, 2024 (ML24219A258) to remove PDC 42 and 43 from the MSRR PDC. ACU stated in information provided for audit that the GMS will be designed to be testable and inspectable. ACU added PDC 61 to the applicable PDC for the GMS which requires the GMS to be designed to permit appropriate periodic inspection and functional testing of safety related components. b. ACU clarified that PDC 51 does not apply to the GMS. PSAR section 6.2.2.7, “Compliance with Design Criteria (See Section 3.1.2),” states that the PDC 55 requires that radionuclide interfacing lines that penetrate functional containment rely upon isolation of penetrations to meet the functional containment design leak rate so that radiological consequences are enveloped by the MHA. The lines of the GMS that penetrate the reactor enclosure are designed with isolation valves. c, d, and e. These audit

	account for thermal stresses and thermal fatigue (e.g., thermal cycling) and what data will be used to design for these effects.	questions are addressed by the DMP, which is described in ACU’s response to the staff’s RAI 2 (ML24121A272). ACU confirmed by letter dated June 12, 2024 (ML24164A236) RCI 9.6-1 that changes in gas properties will not affect the ability of the GMS to perform its safety functions. During review of an OL application, the staff will review the results of the DMP including inspection, monitoring, and testing programs. This is acceptable because inspection, monitoring, and testing programs are addressed in operational programs.

6.0 EXIT BRIEFING

The staff conducted an audit closeout meeting on August 20, 2024. At the exit briefing the staff reiterated the purpose of the audit and summarized the audit activities. Additionally, the staff stated that they did not identify areas where further additional information would be necessary to support the review.

There were no deviations from the audit plan.

7.0 ADDITIONAL INFORMATION RESULTING FROM AUDIT

RAIs and RCIs were generated as a result of topics discussed in this audit and other chapter-specific audits for this review, and ACU's RAI and RCI responses supported the resolution of related audit questions as noted above. ACU also updated the PSAR on its own initiative as noted above to address several items discussed during the audit.

8.0 OPEN ITEMS AND PROPOSED CLOSURE PATHS

Not applicable.