UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 21, 2023

Dr. Rusty Towell
Director of NEXT Lab
Abilene Christian University
ACU Box 27963
Abilene, TX 79699

SUBJECT: ABILENE CHRISTIAN UNIVERSITY – TRANSMITTAL OF REQUESTS FOR ADDITIONAL INFORMATION (EPID NO. L-2022-NFW-0002)

Dear Dr. Towell:

The purpose of this letter is to transmit requests for additional information (RAIs) related to Abilene Christian University's (ACU's) construction permit application for its proposed Molten Salt Research Reactor (MSRR), dated August 12, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22227A201), as supplemented.

The U.S. Nuclear Regulatory Commission (NRC) staff continues to engage with ACU in regulatory audits supporting the review of the construction permit application. Since issuance of the September 14, 2023, status letter (ML23249A095) noting the need for a potential schedule change, the NRC staff has conducted additional audit meetings with ACU and has been able to reach resolution on several audit topics and has made progress towards the resolution of others. However, based on the information discussed with ACU during regulatory audit meetings, the NRC staff understands that ACU needs additional time to further consider and refine certain design aspects of its MSRR. Therefore, the NRC staff has determined that additional information is needed in the following technical areas for the NRC staff to complete its review of the ACU application and identify a revised schedule and resources to document the regulatory findings:

- Application of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section VIII, for the design, fabrication, inspection, testing, and certification of pressure vessels operating at pressures exceeding 15 pounds per square inch gauge.

- Incorporation of surveillance procedures in the MSRR design to monitor and test for potential material degradation mechanisms.

The NRC staff provided ACU with a draft of the associated RAIs on December 14, 2023. As discussed by teleconference on December 19, 2023, ACU is requested to provide a response to the RAIs, or a written request for additional time to respond, including the proposed response date and a brief explanation of the reason by February 1, 2024. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.30(b), "Oath or affirmation," ACU must execute its response in a signed original document under oath or affirmation. The response must be submitted in accordance with 10 CFR 50.4, "Written communications." Information

included in the response that is considered sensitive or proprietary, that ACU seeks to have withheld from the public, must be marked in accordance with 10 CFR 2.390, “Public inspections, exemptions, requests for withholding.”

Once ACU provides the requested information on these topics, the NRC staff will provide an update to the schedule and level of effort communicated in its December 16, 2022, letter (ML22341A615). The NRC staff remains committed to working with ACU to resolve the outstanding technical topics including those addressed in this letter in a timely manner.

If you have any questions, or if you intend to request additional time to respond to this request, please contact me at (301) 415-7190 or via electronic mail at Richard.Rivera@nrc.gov, or contact Ed Helvenston at (301) 415-4067 or via electronic mail at Edward.Helvenston@nrc.gov.

Sincerely,


Signed by Rivera, Richard
on 12/21/23

Richard Rivera, Project Manager
Advanced Reactor Licensing Branch 2
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Docket No. 50-610

Enclosure:
As stated

cc: Abilene Christian University MSRR via GovDelivery

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ADAMS Accession No: ML23348A196 **NRR-106**

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DATE	12/21/2023		

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE CONSTRUCTION PERMIT APPLICATION FOR

THE ABILENE CHRISTIAN UNIVERSITY MOLTEN SALT RESEARCH REACTOR

DOCKET NO. 50-610

Regulatory Basis for Requests for Additional Information (RAIs) 1 and 2

Section 50.34 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.34), "Contents of applications; technical information," provides requirements for information to be provided in a construction permit (CP). Regulation10 CFR 50.34(a)(3)(ii) requires in part that CP applicants provide:

> *The design bases and the relation of the design bases to the principal design criteria*.

Regulation 10 CFR 50.34(a)(3)(iii) requires that CP applicants provide:

> *Information 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.*

Regulation 10 CFR 50.34(a)(4) requires, in part, that CP applicants provide:

> *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.*

Regulation 10 CFR 50.34(a)(8) requires, in part, that CP applicants provide:

> *An identification of those structures, systems, or components of the facility, if any, which require research and development to confirm the adequacy of their design; and identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems or components…*

The U.S. Nuclear Regulatory Commission (NRC) staff recognizes that Abilene Christian University (ACU) plans to apply for an operating license under Section 104c of the Atomic Energy Act of 1954, as amended, and, as such, will apply the minimum amount of regulation necessary to promote the common defense and security and to protect the health and safety of the public. The NRC staff will evaluate ACU's demonstration of how structures, systems, and components (SSCs) will perform as assumed in the preliminary analysis and evaluation presented in the Preliminary Safety Analysis Report (PSAR) as required by 10 CFR 50.34(a)(4). The NRC staff will also confirm the adequacy of ACU's design assumptions regarding

information relative to materials of construction and conformance to the design bases provided as required by 10 CFR 50.34(a)(3)(iii).

RAI 1

The ACU Molten Salt Research Reactor (MSRR) PSAR, Revision 1 (ML23319A094), Section 9.2.3, "Operational Analyses and Safety Function," states that "Fuel salt purification occurs in the fuel salt purification and storage tank. [….] The vessel is constructed of a nickel alloy that is resistant to HF [hydrofluoric acid] exposure. [….] The fuel salt purification and storage tank is constructed of Alloy 201. Proprietary service conditions and ASME [American Society of Mechanical Engineers] code assignments are transmitted by separate letter. Welding between SS316H and Alloy 201 will make use of a suitable material as defined by the appropriate code." The NRC staff requests ACU to respond to the following:

a. The "separate letter" is not referenced in the PSAR. Identify (and docket, as appropriate) the "separate letter."

b. PSAR, Revision 1, Table 3.1-1, "Cross Reference to Preliminary Safety Analysis Report Sections," identifies design criteria applicable to the fuel handling system (FHS). Identify which design criteria ensure adequate structural and mechanical integrity of the safety-related components in the FHS.

c. In addition to the PSAR statement that the vessel is resistant to hydrofluoric acid exposure, the NRC staff requests further information regarding the construction of the fuel salt purification tank and associated piping to ensure that it will provide adequate structural and mechanical integrity during the operating life. Describe the bases that will be used to demonstrate that the materials from which FHS components are fabricated will appropriately withstand operational conditions (e.g., load cycles, time at temperature).

d. In addition to identifying the ASME codes, including the version of the code, that will be assigned to FHS components, the NRC staff requests further information regarding the codes' selection to ensure that components will not be affected by degradation mechanisms that go undetected prior to a loss of capacity to perform their safety functions. Describe the bases that will be used to demonstrate that those codes are appropriate for the components and the conditions to which they are applied. State if and how any supplemental analyses, in addition to those directed by the selected code, would be used to address potential degradation mechanisms, including creep, fatigue, and other time-dependent or high-temperature phenomena.

e. PSAR Revision 1 states, as referenced above, that "[w]elding between SS316H and Alloy 201 will make use of a suitable material as defined by the appropriate code." However, the NRC staff notes that neither ASME Boiler and Pressure Vessel Code (BPVC) Section VIII, "Rules for Construction of Pressure Vessels," nor ASME B31.3, "Process Piping," include specific combinations of weld and base materials and they do not address high temperature degradation of weld metals. Therefore, the NRC staff requests further information about the welds, including the Alloy 201 to Alloy 201 welds, to ensure that they will provide adequate structural and mechanical integrity during the operating life. Describe the bases that will be used to demonstrate the weld material suitability, and the bases (e.g., allowable stress rupture values, tensile strength, etc.)

that will be used to ensure that the weld materials will not be affected by degradation mechanisms, such as stress-rupture due to creep, that go undetected prior to a loss of components' capacities to perform their safety functions.

f. If ACU is relying on pre-existing data to demonstrate the adequacy of the FHS design, state what those data are and provide the bases for the determination that these adequately represent, as relevant, the materials and operating conditions of the FHS.

g. If it is necessary for ACU to acquire additional data to demonstrate the adequacy of the FHS design, describe the scope of such testing and state the purpose and schedule for the acquisition of those data (e.g., tensile strength, stress rupture values, etc.), as well as what bases will be used to determine that these adequately represent, as relevant, the materials and operating conditions of the FHS.

h. If ACU will rely on in-service methods (e.g., periodic surveillance or inspection) to confirm that the FHS continues to conform with the design criteria during the operating life, describe the scope of this in-service approach and state the bases for determining that those methods, including the frequency and techniques, will detect degradation prior to the loss of capacity for the affected components to perform their intended functions.

RAI 2

The ACU MSRR PSAR, Revision 1, Section 4.3.2, "Design Bases," states that "the reactor system is designed to accommodate the effects of and to be compatible with the expected environmental conditions during operation, maintenance, testing, and postulated accidents …. to have an extremely low probability of leakage, rapidly propagating failure, or rupture …. [and] to permit periodic inspection, testing, and surveillance." Further, PSAR Revision 1, Section 4.3.5, "Radiation Damage to Reactor System," states that "[c]onsidering 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 dpa [displacements per atom] for SS316 components will be acceptable." The NRC staff requests ACU to respond to the following:

a. Provide additional information regarding how ACU is utilizing pre-existing data to demonstrate conformance with these design bases referenced above, for instance, to preclude the possibility that certain degradation phenomena will occur or to assess the extent or rate of progression for certain degradation phenomena when its potential occurrence cannot be excluded. This includes, for instance, the conclusion that 5 dpa for SS316 components will be acceptable, and determinations about the potential for specific degradation mechanisms that may be caused by interactions with the MSRR environment (e.g., thermal aging, environmentally assisted cracking, helium embrittlement, etc.). Please identify what pre-existing data will be used, and how the data will be applied. Also, please identify the criteria that have been or will be used to determine that the data adequately represent or bound the materials and operating conditions of the MSRR.

b. If it is necessary for ACU to acquire additional data to demonstrate the adequacy of the reactor system design, describe the scope of such data acquisition activities and state the purpose of any testing to be undertaken, how the data will be applied, and the schedule for acquisition of those data. Also, identify the criteria that will be used to

determine that the data adequately represent or bound the materials and operating conditions of the MSRR.

c. If ACU will rely on in-service methods (e.g., periodic surveillance or inspection) to confirm that the reactor boundary system and RTMS integrity will be maintained over the operating life, please describe the approach including stating the criteria that ACU will use to determine the scope of the in-service methods, how pre-existing or to-be-acquired data will be used to inform the application of methods (e.g., techniques and frequency), and the criteria that ACU will use to demonstrate that the methods can adequately detect degradation prior to the loss of capacity for the affected components to perform their intended functions.