OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE RENEWAL OF

THE AEROTEST RADIOGRAPHY AND RESEARCH REACTOR

LICENSE NO. R-98; DOCKET NO. 50-228

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review of your application for the renewal of Facility Operating License No. R-98 for the Aerotest Radiography and Research Reactor (ARRR) dated February 28, 2005 (a redacted version of the application is available on the NRC public Web site at www.nrc.gov under Agencywide Documents Access and Management System (ADAMS) Accession No. ML13120A434), as supplemented. In the course of reviewing the ARRR renewal application, the NRC staff has determined that additional information or clarification is required to continue the review of the safety analysis report (SAR) in support of the development of its safety evaluation report. The ARRR facility, as described in the SAR, is primarily evaluated using the appropriate regulations in Title 10 of the *Code of Federal Regulations* (10 CFR), and the following guidance, as applicable:

- NUREG-1537, Part 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content,” issued February 1996 (ADAMS Accession No. ML042430055)
- NUREG-1537, Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria,” issued February 1996 (ADAMS Accession No. ML042430048)
- “Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research and Test Reactors,” dated October 2009 (ADAMS Accession No. ML092240244)
- American National Standards Institute/American Nuclear Society-15.1-2007, “The Development of Technical Specifications for Research Reactors”

Enclosure

Note: “SAR,” as used below, refers to the updated SAR that Aerotest submitted to NRC by letter dated December 20, 2017 (ADAMS Accession Nos. ML17363A303 and ML18045A571).

RAI 4-1

The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

NUREG-1537, Part 1, Section 4.5.1, “Normal Operating Conditions,” states that licensees should present information on core geometry and configurations, including the limiting core configuration (LCC) (the core yielding the highest power density and fuel temperature using the fuel specified for the reactor), and other proposed operating core configurations that are demonstrated to be encompassed by the safety analysis of the LCC.

The SAR, Section 4.5, “Nuclear Design,” states that as of 2010, the ARRR core consisted of 55 aluminum clad elements and 27 stainless steel clad elements (total of 82 fuel elements), and SAR Figure 4.2-2 shows a typical ARRR core arrangement which appears to contain a total of 86 fuel elements. SAR Section 13.1.3, “Number of TRIGA Fuel Elements in the Reactor,” states that at initial criticality, the ARRR core consisted of 63 aluminum clad fuel elements. SAR Section 13.1.2, “TRIGA Fuel Type (Aluminum Clad versus Stainless Steel Clad),” states that as of 2017, Aerotest possesses 55 irradiated aluminum clad elements, 27 irradiated stainless steel elements, and 12 fresh stainless steel elements. As discussed in the NRC staff’s safety evaluation, dated February 28, 2017, approving the license transfer of Aerotest to Nuclear Labyrinth, LLC (ADAMS Accession No. ML16333A449), during the 2014 evidentiary hearing on the Aerotest license transfer, Dr. David M. Slaughter of Nuclear Labyrinth, LLC, testified and provided supporting evidence that the ARRR can be operated at licensed power with a core design consisting of 36 stainless steel elements and 28 aluminum elements (64 elements total). In its startup plan provided to the NRC by letter dated March 6, 2018 (ADAMS Accession No. ML18071A048), Aerotest discusses a proposed core consisting of 39 stainless steel elements and 18 aluminum elements (57 elements total).

The SAR, Section 4.2.5, “Control Rods and Drive Mechanisms,” provides typical control rod worths, but states that the actual reactivity worth of each rod is dependent on the core configuration.

The SAR, Section 4.5, and SAR, Section 13.1, “Application of Historic and Generic Accident Analyses to the ARRR,” discuss how analyses of routine and transient operation for the ARRR are bounded by analyses in NUREG/CR-2387, “Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors (ADAMS Accession No. ML083660125), and in the original 1964 Hazards Summary Report (HSR) for the ARRR. SAR Section 13.1.3 states that it is expected that power distribution following insertion of new fuel elements will remain within the TRIGA reactor norm and accident analysis assumption that power density of the most reactive fuel element is no more than twice the core average power density.

However, the SAR does not appear to include specific descriptions of historical or currently-planned proposed future ARRR operating core configurations (i.e., descriptions which consider the locations of stainless steel versus aluminum elements, and the relative burnup and reactivity of the individual elements), or an LCC. The SAR also does not appear to include calculations or measurements, specific to the ARRR core configurations and fuel, demonstrating the power distribution or power peaking in the operating or limiting cores; demonstrating the reactivity worth of individual control rods and other core components, as applicable, in operating or limiting cores; or demonstrating that shutdown margin and excess reactivity will remain within technical specification (TS) limits for operating or limiting cores.

Please provide descriptions of operating core configurations, and an LCC (which would produce the greatest power peaking and fuel temperature of any allowed core configuration, considering possible fuel arrangements, fuel types, fuel burnup, control rod positions, experiments, etc.), for the ARRR, and describe how the LCC was determined. If neutronics models are used to determine the limiting core, please describe how the models are validated for the ARRR. Additionally, please provide detailed calculations and/or measurements demonstrating the power distribution and power peaking in the operating cores and LCC; demonstrating the reactivity worth of individual control rods and other core components, as applicable, in the specific analyzed cores; and demonstrating that shutdown margin and excess reactivity will remain within TS limits in the analyzed cores. Alternatively, justify why no additional information is required.

RAI 4-2

The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

NUREG-1537, Part 1, Section 4.5.2, "Reactor Core Physics Parameters," states that licensees should discuss the core physics parameters (including neutron lifetime, effective delayed neutron fraction, and coefficients of reactivity), including the methods and analyses used to determine them, and how they vary with reactor operating characteristics and fuel burnup.

The SAR does not appear to discuss the core physics parameters for the ARRR. Please provide values for these parameters which are specific to the ARRR and consider the types of fuel used in the ARRR, as well as a discussion of how the values were determined, the uncertainties in the values, and how the values vary with reactor operating characteristics (such as temperature) and fuel burnup, or justify why no additional information is required.

RAI 4-3

The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

NUREG-1537, Part 1, Section 4.6, “Thermal-Hydraulic Design,” states that licensees should present the information and analyses necessary to show that sufficient cooling capacity exists to prevent fuel overheating and loss of integrity for all anticipated reactor operating conditions. The licensee should address the coolant flow conditions for which the reactor is designed and licensed (i.e., natural convection flow, in the case of the ARRR), and a detailed description of the methods used in the thermal-hydraulic analysis should be provided.

NUREG-1537, Part 2, Section 4.6, “Thermal-Hydraulic Design,” states that licensees should propose safety limits based on the criteria for safe operation of the reactor, thus ensuring fuel integrity under all analyzed conditions. Criteria for acceptable safe reactor operation could include the determination that the departure from nucleate boiling ratio (DNBR) is no less than 2 at any core location.

The SAR, Section 4.6, “Thermal-Hydraulic Design,” states that TSs do not require cooling of reactor tank water during reactor operation, and, as demonstrated during initial startup physics testing, the ARRR can operate at 250 kilowatts thermal without external cooling of the reactor tank water for more than 6 hours before the water will reach a TS-limit temperature of 130 degrees Fahrenheit (F). The SAR, Section 4.6, further states that automatic reactor scrams prior to the temperature reaching this limit, and on low water level in the reactor tank, ensure loss of cooling during normal operation will never result in the fuel safety limit temperature of 500 degrees Celsius (C) being exceeded.

However, it is not clear to the NRC staff how the 130 degree F limit on pool water temperature will ensure that the maximum fuel temperature will remain below 500 degrees C, for any allowed core configuration, operational condition, or duration of continuous reactor operation.

Please provide a thermal-hydraulic analysis that is based on the LCC, is specific to the ARRR, and demonstrates that for any allowed duration of continuous reactor operation at full licensed power under any allowed operational condition, the maximum temperature at any location within the fuel will remain below the safety limit. As necessary, the analysis should demonstrate that the DNBR is no less than 2 at any core location. The analysis should utilize the results of your neutronics analysis provided in response to RAI 4-1, as applicable. TS-required scram functions (i.e., limiting safety system settings) or operational limits may be credited in the analysis; however, the analysis must demonstrate how these TSs would ensure the safety limit is not exceeded. Alternatively, justify why no additional information is required.

RAI 11-1

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

NUREG-1537, Part 1, Section 11.1.1.1, “Airborne Radiation Sources,” states that licensees should summarize the predicted concentrations and quantities of airborne radionuclides in areas that could be occupied by personnel, during the full range of reactor operation. Licensees should also estimate the release of airborne radionuclides to the environment and should use these releases to determine consequences in the offsite environment. The models and assumptions used for the prediction and calculation of the dose rates and accumulative doses in both the restricted and unrestricted areas should be provided in detail. In the unrestricted area, potential doses should be analyzed for the maximally exposed individual, the location of the nearest permanent residence, and at any other locations of special interest. Licensees should discuss compliance of the restricted and unrestricted area doses with the applicable regulations in 10 CFR Part 20.

The SAR, Section 3.3, “Radioactive Gaseous Effluents,” states that Aerotest uses the COMPLY code for calculating offsite doses from argon-41 (Ar-41), and provides average offsite Ar-41 doses for 1993-1997 and 1998-2010 based on average Ar-41 release rates during those periods. The SAR, Section 11.1.1, “Radiation Sources,” states that the values for Ar-41 produced (the values used for the COMPLY calculations) are based on geometry and observed values and are computed using air monitor counts, reactor operating hours, atmospheric dilution, and the number of irradiations performed. SAR Section 11.1.1 also provides a rate of 480 microcuries per hour for Ar-41 production at full-power reactor operation.

However, the SAR does not appear to fully describe the assumptions (i.e., release height, geometry, receptor location, meteorology, etc.) used for COMPLY calculations. Additionally, the basis for the 480 microcuries per hour Ar-41 production rate, and how this rate would be used in a calculation, is not clear (i.e., exactly how this rate is determined; whether this is total Ar-41 production rate, or Ar-41 release rate to the environment; and whether or how this rate considers experimental facilities whose use may affect Ar-41 production). The SAR does not appear to include bounding calculations of restricted or unrestricted area Ar-41 doses which assume continuous full-power operation of the reactor over the course of a year (as would be permitted by a renewed license).

Please provide bounding calculations of annual restricted area, and unrestricted area (at the maximum dose location and nearest residence), dose from Ar-41 assuming continuous full-power reactor operation; provide a detailed description of the assumptions used for these calculations; and discuss the compliance of the results of these calculations with 10 CFR Part 20 limits. Alternatively, justify why no additional information is required.

RAI 13-1

The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

The guidance in NUREG-1537, Part 1, Chapter 13, “Accident Analyses,” describes the types of research reactor accidents that should be analyzed in the SAR. These accidents include a maximum hypothetical accident (MHA). NUREG-1537, Part 1, Section 13.1.1, “Maximum Hypothetical Accident,” states that the MHA bounds all credible accidents and can be used to illustrate the analysis of events and consequences during an accidental release of radioactive material.

The SAR, Section 13.4, “Maximum Hypothetical Accident,” references accident analyses in NUREG/CR-2387 and the original 1964 HSR for the ARRR. SAR Section 13.4 states that the ARRR MHA is bounded by the analyses in NUREG/CR-2387, because the ARRR operates at a lower power level, is not operated continuously throughout the year, and has a core with more fuel elements than the hypothetical core described in the NUREG/CR-2387 (meaning that the failure of a single ARRR fuel element would result in a smaller fraction of the total core radionuclide inventory being released). SAR Section 13.4 states that the ARRR core is assumed to have a similar power distribution as the hypothetical core described in NUREG/CR-2387.

The NRC staff notes that, given changes since the ARRR was originally licensed, such as the use of new stainless steel clad elements with higher uranium loading and reactivity in addition to the original aluminum clad elements, and the associated variations in power peaking, neutron flux, fission product inventory, etc., the analyses performed for the original licensing of the ARRR and discussed in the 1964 HSR may no longer be valid. Additionally, it is not clear that the ARRR power distribution would be similar to the hypothetical core described in NUREG/CR-2387 (see RAI 4-1), and therefore the generic analyses in NUREG/CR-2387 may also not be valid for the ARRR.

Furthermore, the fuel handling accident analysis in NUREG/CR-2387 assumes that the hypothetical reactor has been operated continuously for 1 year, followed by 48 hours of decay. The NRC staff notes that, although the ARRR has historically not been operated continuously throughout the year, there is no proposed restriction on operation for a renewed ARRR license, and therefore accident analyses should assume that the reactor has been continuously operated (i.e., operated for a long enough period to reach saturated fission product inventories), if this would produce the bounding accident consequences. Additionally, because a potential fission product release could occur due to a fuel failure during operation (rather than a fuel handling accident), it may not be appropriate for a bounding MHA analysis for the ARRR to credit radioactive decay prior to the fission product release.

The NRC staff also notes that the accident analysis in NUREG/CR-2387 uses a generic fuel element release fraction which, depending on the maximum fuel temperatures in a limiting ARRR core, may not necessarily be bounding for ARRR operation.

Additionally, the NRC staff notes that the accident analysis discussed in NUREG/CR-2387 and referenced in SAR Section 13.4 considers doses to members of the public outside the reactor building, but does not appear to evaluate potential accident doses to occupational workers located in the restricted area.

Please provide a detailed MHA analysis (including a discussion of the methodologies and assumptions for the analysis) that is specific to the ARRR and its proposed operation and TSs following the issuance of a renewed license. The analysis should use bounding, worst-case assumptions, should utilize the results of your neutronics analysis provided in response to RAI 4-1, as applicable, and should determine maximum doses to both occupational workers in the restricted area and members of the public in the unrestricted area. Alternatively, justify why no additional information is required.

RAI 13-2

The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Chapter 13, describes the types of research reactor accidents that should be analyzed in the SAR. These accidents include insertion of excess reactivity accidents. NUREG-1537, Part 1, Section 13.1.2, “Insertion of Excess Reactivity,” states that insertion of reactivity events include the rapid (or step) insertion of reactivity (i.e., due to rapid removal of a control rod or rods, rapid insertion of a fuel element into the core, or experiment malfunction), and the ramp insertion of reactivity (i.e., due to drive motion of a control rod or rods). NUREG-1537, Part 1, Section 13.2, “Accident Analysis and Determination of Consequences,” describes how accidents should be analyzed in the SAR, and states that the SAR should base accident scenarios on a single initiating malfunction.

The SAR, Section 13.2.1, “Insertion of Excess Reactivity,” summarizes analyses discussed in the 1964 ARRR HSR and in letters from Aerotest to NRC dated April 26, 1968; July 11, 1968; and May 8, 1970. These analyses include a startup rod withdrawal transient (i.e., a ramp insertion of reactivity due to sequential withdrawal of all 3 control rods, starting from shutdown conditions); an uncontrolled rod withdrawal (i.e., a ramp insertion of reactivity due to simultaneous withdrawal of all 3 control rods); a step insertion of the maximum TS-allowed excess reactivity; and step insertions of reactivity due to a fuel loading or experiment removal accident. SAR Section 13.2.1 states that the step insertion of the maximum TS-allowed excess reactivity accident would bound all other reactivity transient accidents, and that the maximum fuel temperature associated with this accident would be below the safety limit.

However, given the differences between previous and potential future ARRR cores, it is not clear that the reactivity transient analyses discussed in the SAR would be applicable for proposed operation of the ARRR under a renewed license. Additionally, the NRC staff notes that given the additional time it could take the reactor to scram (on high power) under a ramp reactivity insertion transient starting from low power when the single initiating malfunction is a failure of a short period scram, a ramp insertion transient, could, potentially, bound any potential step insertion transient.

Please provide detailed ramp and step reactivity insertion analyses (including a discussion of the methodologies and assumptions for the analyses) that are specific to the ARRR and its proposed operation and TSs following the issuance of a renewed license. The analysis should use bounding, worst-case assumptions, should utilize the results of your analyses provided in response to RAIs 4-1, 4-2, and 4-3, as applicable, and should demonstrate that potential reactivity transients will not cause the fuel temperature at any location in the fuel to exceed the safety limit. As necessary, the analysis should demonstrate that the DNBR is no less than 2 at any core location. TS-required scram functions (i.e., high power and/or short period scrams) or operational limits may be credited in the analysis, as appropriate; however, the analysis must demonstrate how these TSs would ensure the safety limit is not exceeded. Alternatively, justify why no additional information is required.

RAI 13-3

The regulations in 10 CFR 50.34(b)(2) require that the SAR include the evaluations required to show that safety functions of the structures, systems, and components listed in 10 CFR 50.34(b)(2)(i) will be accomplished.

The guidance in NUREG-1537, Part 1, Chapter 13, describes the types of research reactor accidents that should be analyzed in the SAR. These accidents include loss of coolant accidents (LOCAs).

The SAR, Section 13.2.2, “Loss of Coolant,” states that LOCAs are bounded by the instantaneous loss of all cooling water, and references NUREG/CR-2387, which summarizes a loss of coolant analysis for the Reed College TRIGA reactor that calculated a maximum fuel temperature of less than 150 degrees C after infinite operation at 250 kilowatts thermal power with all aluminum clad fuel elements. SAR Section 13.2.2 states any ARRR LOCA would be bounded by the Reed College reactor LOCA analysis, given the operational history of the ARRR, the larger number of fuel elements in the ARRR compared to the Reed College reactor, and the use of some stainless steel clad fuel elements in the ARRR (the stainless steel elements can handle higher temperatures, and would be expected to have the higher power density at the time of a LOCA).

However, given the possible power densities and peaking for potential future ARRR cores, it is not clear to the NRC staff whether the Reed College analysis discussed in NUREG/CR-2387 and referenced in the ARRR SAR would necessarily bound an ARRR LOCA. Additionally, the NRC staff notes that the ARRR loss of coolant analysis contained in the 1964 ARRR HSR provides a maximum fuel temperature of 650 degrees C, which is significantly greater than the 150 degrees C from the Reed College analysis (although the NRC staff also recognizes that, given the changes to the ARRR since its original licensing, the HSR analysis may no longer be applicable).

Please provide a detailed loss of coolant analysis (including a discussion of the methodologies and assumptions for the analysis) that is specific to the ARRR and its proposed operation and TSs following the issuance of a renewed license. The analysis should use bounding, worst-case assumptions, should utilize the results of your analyses provided in response to RAIs 4-1 and 4-3, as applicable, and should demonstrate that potential LOCAs will not cause the fuel temperature at any location in the fuel to exceed the safety limit. As appropriate, the analysis should determine the amount of time following shutdown that the fuel needs to remain covered with water to prevent the safety limit from being exceeded. The analysis may credit the time it would take to drain the reactor pool during a worst-case pool draining scenario, if appropriate. Alternatively, justify why no additional information is required.

RAI 13-4

The regulations in 10 CFR 50.34(b)(3) require that the SAR contain information regarding the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling a limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR Part 20.

The guidance in NUREG-1537, Part 1, Chapter 13, describes the types of research reactor accidents that should be analyzed in the SAR. These accidents include LOCAs. NUREG-1537, Part 1, Section 13.2, describes how accidents should be analyzed in the SAR, and states that potential radiological consequences should be evaluated using realistic methods. NUREG-1537, Part 2, Chapter 13, “Accident Analyses,” states that licensees should analyze potential doses to facility staff and the public in unrestricted areas from direct and scattered gamma radiation from the unshielded core during a LOCA.

The SAR, Section 13.2.2, states that based on calculations described in the original 1964 HSR, radiation levels above the reactor water tank after a complete loss of coolant would allow sufficient time for personnel to view the interior of the tank with a mirror and to make emergency repairs. Additionally, SAR Section 13.2.2 states if an individual were not directly exposed to the core, the individual could work for approximately 90 minutes at the top of the reactor tank after one day without being exposed to radiation in excess of approximately 1.25 rem.

However, given the differences between previous and potential future ARRR cores, and other changes that may have occurred at the facility since the 1964 analysis (for example, changes in the layout or equipment in the reactor room that could affect radiation scattering), it is not clear that the LOCA radiation shine analyses discussed in the SAR would necessarily be applicable for proposed operation of the ARRR under a renewed license. Additionally, the SAR does not appear to address LOCA radiation shine doses to members of the public in unrestricted areas. The NRC staff notes that members of the public outside the restricted area could, potentially, be exposed to scattered radiation emitted by the exposed core following a LOCA.

Please provide a detailed LOCA radiation shine analysis (including a discussion of the methodologies and assumptions for the analysis) that is specific to the ARRR and its proposed operation and TSs following the issuance of a renewed license; that uses bounding, worst case assumptions; and that considers potential doses to both facility staff and members of the public in the unrestricted area. Alternatively, justify why no additional information is required.