April 24, 2018

Dr. David M. Slaughter
President and Reactor Administrator
Aerotest Operations, Inc.
3455 Fostoria Way
San Ramon, CA 94583

SUBJECT: AEROTEST OPERATIONS, INC. - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE RENEWAL OF FACILITY OPERATING LICENSE NO. R-98 (EPID NO. L-2017-RNW-0027)

Dear Dr. Slaughter:

Aerotest Operations, Inc. (Aerotest) holds U.S. Nuclear Regulatory Commission (NRC) Facility Operating License No. R-98 for the operation of the Aerotest Radiography and Research Reactor (ARRR). By application dated February 28, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13120A434), as supplemented by letters dated May 5, 2008 (ADAMS Accession No. ML103370137), March 9, 2009 (ADAMS Accession No. ML120900629), December 20, 2017 (ADAMS Accession Nos. ML17363A303 and ML18045A571), and February 28, 2018 (ADAMS Accession No. ML18066A075), Aerotest applied for renewal of the ARRR operating license.

During the NRC staff's review of the portions of Aerotest's renewal application related to neutronics, thermal-hydraulics, argon-41 dose analyses, and accident analyses (specifically, maximum hypothetical accident, reactivity transient, and loss of coolant accident analyses), questions have arisen for which additional information is needed. The enclosed request for additional information (RAI) identifies the additional information needed for the NRC staff to continue its review.

Aerotest's renewal application includes references to NUREG/CR-2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors" (ADAMS Accession No. ML083660125), and the original 1964 Hazards Summary Report (HSR) for the ARRR, and states that ARRR nuclear design, thermal-hydraulics, and accident consequences are bounded by analyses discussed in those documents. However, the NRC staff notes that, due to the possible changes in core configurations that may be used at the ARRR for operation under a renewed license, and factors such as the combined use of new stainless steel fuel with the older aluminum clad fuel and the high variation in burnup of the ARRR fuel, the assumptions used in the analyses in those documents may not necessarily be valid for future ARRR operation. The NRC staff also notes that, although NUREG/CR-2387 provides generic analyses of Training, Research, Isotopes, General Atomics (TRIGA) reactors, the NRC staff has generally required detailed facility-specific analyses to allow it make a finding that the proposed operation of a TRIGA facility is acceptable or is bounded by NUREG/CR-2387. This expectation is consistent with other reviews conducted for similar TRIGA reactors for which similar NRC review guidance (i.e., the guidance listed on page 1 of the enclosed RAI) is applicable.

Therefore, the enclosed RAI includes requests for Aerotest to provide detailed nuclear design, thermal-hydraulic, and accident analysis calculations that are specific to the ARRR and its proposed future operation, and that consider the most limiting conditions under which a renewed license would allow the ARRR to be operated, in order to demonstrate that the ARRR can be operated safely under routine and accident conditions.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.30(b), "Oath or affirmation," Aerotest 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 you seek to have withheld from the public, must be marked in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Any information related to safeguards should be submitted in accordance with 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements." Following receipt of the additional information, the NRC staff will continue its evaluation of Aerotest's renewal request.

The NRC staff requests that Aerotest provide responses to the enclosed RAIs within 90 days from the date of this letter. The NRC staff notes that the timely receipt of the requested information and analyses will contribute to an efficient and effective review of the renewal application.

A draft copy of the enclosed RAIs was provided to you by electronic mail dated March 29, 2018 (ADAMS Accession No. ML18088A753), and the response time frame in this letter was discussed with you following your review of the draft RAIs.

If you have any questions, or need additional time to respond to this request, please contact me at 301-415-4067, or by electronic mail at Edward.Helvenston@nrc.gov.

Sincerely,

/RA/

Edward Helvenston, Project Manager
Research and Test Reactors Licensing Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No. 50-228
License No. R-98

Enclosure:
As stated

cc: See next page

Sandra Warren, General Manager
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SUBJECT: AEROTEST OPERATIONS, INC. - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE RENEWAL OF FACILITY OPERATING LICENSE NO. R-98 (CAC/DOCKET/EPID NO. 000955/05000228/L-2017-RNW-0027) DATE: APRIL 24, 2018

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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) staff is continuing its review of Aerotest's 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 under Agencywide Documents Access and Management System (ADAMS) Accession No. ML13120A434), as supplemented by letters dated May 5, 2008 (ADAMS Accession No. ML103370137), March 9, 2009 (ADAMS Accession No. ML120900629), December 20, 2017 (ADAMS Accession Nos. ML17363A303 and ML18045A571), and February 28, 2018 (ADAMS Accession No. ML18066A075). 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 Aerotest's safety analysis report (SAR) in support of the development of the NRC staff's safety evaluation report. The ARRR renewal application is being evaluated against the applicable regulations in Title 10 of the *Code of Federal Regulations* (10 CFR). Additionally, the renewal application is evaluated using applicable standards and NRC guidance documents, primarily the following:

- 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"

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.

The guidance in 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 its currently 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 April 3, 2018 (ADAMS Accession No. ML18096A689), 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. This information is necessary to show that assumptions used in ARRR thermal-hydraulic and accident analyses are valid and bounding, and that the ARRR can be operated in compliance with TSs.

Provide descriptions of operating core configurations, and an LCC (i.e., a core configuration that 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 LCC, describe how the models are validated for the ARRR. Additionally, provide 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. Include a discussion of the methodologies and assumptions used for any calculations performed. 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.

The guidance in 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. These parameters are necessary to characterize reactor operations, and for the NRC staff to determine that assumptions used in ARRR reactivity transient analyses are bounding.

Provide values for core physics parameters that are specific to the ARRR and consider the types of fuel used in the ARRR. Include 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. Alternatively, 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.

The guidance in 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.

The guidance in 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 degrees 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. A justification of the 130 degrees F limit is necessary to demonstrate that the limit will prevent fuel overheating.

Provide a thermal-hydraulic analysis that is based on the LCC, is specific to the operating characteristics of 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.

The guidance in 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 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. These values for Ar-41 produced are used as input for the COMPLY calculations. 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, including how this rate would be used in a calculation, is not clear. Specifically, it is not described exactly how this rate is determined; whether this is the 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 Aerotest’s proposed TSs for a renewed license). This information is necessary to demonstrate that continued operation of the ARRR would comply with the dose limits in 10 CFR Part 20.

Provide bounding calculations of the annual restricted area, and unrestricted area (at the maximum dose location and nearest residence), dose from Ar-41 assuming continuous full-power reactor operation. Additionally, provide a description of the assumptions and methodologies 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 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. For research reactors, the results of the accident analysis have generally been compared to 10 CFR Part 20 public and occupational limits.

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 to the core configuration since the ARRR was originally licensed (e.g., 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.

A facility-specific MHA analysis is necessary to demonstrate that the consequences of any credible accident at the ARRR would not result in radiation exposures to Aerotest staff or members of the public exceeding 10 CFR Part 20 limits.

Provide an MHA analysis that is specific to the ARRR and its proposed operation and TSs, as described in the SAR, as supplemented. Include a discussion of the methodologies and assumptions used for the analysis. The analysis should use bounding, worst-case assumptions, should utilize the results of Aerotest’s 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 Sections 13.2.1 and 13.3.4, "Damage from Detonation to Control Rod Drive Mechanisms," state that the step insertion of the maximum TS-allowed excess reactivity accident would bound all other reactivity transient accidents (e.g., a detonation of explosive material being radiographed, that causes an ejection of all 3 control rods), and that the maximum fuel temperature associated with this accident would be below the safety limit.

The SAR, Section 13.2.1, further states that for the ramp reactivity insertion accidents discussed in the SAR, redundant and diverse reactor protection systems channels will automatically terminate the reactivity addition when the reactor period becomes too short to allow the reactor operator to control the reactor power level.

Proposed TS Table 3.2-1, "Reactor Safety Channels," states that instrumentation Channels 1 and 2 are required to provide short period scrams, but Channel 1 scrams are bypassed when Channel 2 exceeds a fixed setting of approximately 1×10^{-10} amps.

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 (but when Channel 1 scrams are bypassed), a ramp insertion transient, could, potentially, bound any potential step insertion transient when the single initiating malfunction is a failure of the Channel 2 short period scram.

Facility-specific reactivity transient analyses are necessary to demonstrate that credible reactivity transient accidents could not result in a safety limit for the reactor fuel being exceeded.

Provide ramp and step reactivity insertion analyses that are specific to the ARRR and its proposed operation and TSs, as described in the SAR, as supplemented. Include a discussion of the methodologies and assumptions used for the analyses. The analysis should use bounding, worst-case assumptions, should utilize the results of the 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. (Note: This last assumption is based on the stainless steel clad elements’ ability to handle higher temperatures than aluminum clad elements, and the expectation that the former would have a higher power density than the latter 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. (Note: The NRC staff also recognizes that, given the changes to the ARRR since its original licensing, the HSR analysis may no longer be applicable.)

A facility-specific LOCA analysis is necessary to demonstrate that no loss of coolant event could result in a safety limit for the reactor fuel being exceeded.

Provide a LOCA analysis that is specific to the ARRR and its proposed operation and TSs, as described in the SAR, as supplemented. Include a discussion of the methodologies and assumptions used for the analysis. The analysis should use bounding, worst-case assumptions, should utilize the results of the 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 (i.e., due to a tank rupture), if appropriate. Alternatively, justify why no additional information is required.

RAI 13-4

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 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. For research reactors, the results of the accident analysis have generally been compared to 10 CFR Part 20 public and occupational limits.

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 (Roentgen equivalent man).

However, the methodologies (e.g., equations used) and assumptions (e.g., core operating history, photon energy or energies assumed, and geometry assumptions) used for the analysis discussed in SAR Section 13.2.2 are not entirely clear. Also, given the differences between previous and potential future ARRR cores (e.g., potential higher burnup and greater fission product inventory of future cores) and other changes that may have occurred at the facility since the 1964 analysis (e.g., 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, as described in the SAR, as supplemented. 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.

A facility-specific LOCA radiation shine analysis is necessary to demonstrate that no loss of coolant event would result in radiation exposures to Aerotest staff or members of the public exceeding 10 CFR Part 20 limits.

Provide a LOCA radiation shine analysis that is specific to the ARRR and its proposed operation and TSs, as described in the SAR, as supplemented. Include a discussion of the methodologies and assumptions used for the analyses. This analysis should use bounding, worst-case assumptions, and should consider potential doses to both facility staff and members of the public in the unrestricted area. Alternatively, justify why no additional information is required.