January 19, 2016

Dr. Robert. Bean, Director
Purdue University Radiation Laboratory
School of Nuclear Engineering
400 Central Drive
West Lafayette, IN 47904-2017

SUBJECT: PURDUE UNIVERSITY - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE RENEWAL OF FACILITY OPERATING LICENSE NO. R-87 FOR THE PURDUE UNIVERSITY REACTOR (TAC NO. ME1594)

Dear Dr. Bean:

The U.S. Nuclear Regulatory Commission (NRC) is reviewing your application for the renewal of Facility Operating License No. R-87, dated July 7, 2008, available on the NRC's public Web site at www.nrc.gov under Agencywide Documents Access and Management System Accession No. ML083040443, as amended, for the Purdue University Reactor. During our review questions have arisen for which additional information is needed. The enclosed request for additional information (RAI) identifies the additional information needed to complete our review. We request that you provide responses to the enclosed RAI within 30 days from the date of this letter.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.30(b), "Oath or affirmation," you must execute your response in a signed original document under oath or affirmation. Your response must be submitted in accordance with 10 CFR 50.4, "Written communications." Information included in your 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 security should be submitted in accordance with 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements." Following receipt of the additional information, we will continue our evaluation of your renewal request.

If you need additional time to complete this request, or have any questions regarding this review, please contact me at (301) 415-3398, or by electronic mail at Cindy.Montgomery@nrc.gov.

Sincerely,

***/RA/***

Cindy K. Montgomery, Project Manager
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-182

Enclosure:
As stated

cc: See next page

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

FOR THE RENEWED LICENSE FOR

PURDUE UNIVERSITY REACTOR

LICENSE NO. R-87; DOCKET NO. 50-182

The U.S. Nuclear Regulatory Commission (NRC) is reviewing your application for the renewal of Facility Operating License No. R-87, dated July 7, 2008, available on the NRC's public Web site at www.nrc.gov under Agencywide Documents Access and Management System Accession (ADAMS) No. ML083040443, as amended, for the Purdue University Reactor (PUR-1). During our review questions have arisen for which additional information is needed. The enclosed request for additional information (RAI) identifies the additional information needed to complete our review. We request that you provide responses to the enclosed RAI within 30 days from the date of this letter.

The NRC staff review of the proposed PUR-1 technical specifications (TSs) was based on the requirements in Title 10 *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications," and the guidance provided in NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," Chapter 14, Appendix 14.1, "Format and Content of Technical Specifications for Non-Power Reactors," and American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1-2007, "The Development of Technical Specifications for Research Reactors."

1. TS 1.49: PUR-1 TS 1.49 provides a definition for an Unsecured Experiment. This definition refers to the definition in PUR-1 TS 1.36 for a secured experiment. The definition for "Secured Experiment," is provided in TS 1.37. Update TS 1.49 to reference the definition for secured experiment in TS 1.37, or explain why it is not necessary.

RAI 2 is based on a review of your responses to RAI 5 in your letter dated July 24, 2015, to NRC's letter dated August 29, 2014 (ADAMS Accession No. ML14115A221).

2. TS 2.2: The basis in PUR-1 TS 2.2, states that a steady-state power level of 94.2 kW is required to initiate the onset of nucleate boiling (ONB) with a maximum fuel temperature of 49 degrees Celsius. The response to RAI 5 in your letter dated July 24, 2015, indicates that the PUR-1 ONB power level is 98.6 kW with a maximum fuel temperature of 43.20 degrees Celsius. Clarify the ONB power level and maximum fuel temperature and provide updates to the PUR-1 TS 2.2 basis that correctly reflects the ONB power and maximum fuel temperature under the renewal of the PUR-1 facility license.

3. TS 3.2: PUR-1 TS 2.2, states that the measured value of the power level scram shall be no higher than 12.0 kW. In the license renewal application, as supplemented (Purdue University Research Reactor, "Application for Relicense of License Number R-87 with Power Uprate, Safety Analysis Report," July 07, 2008, ADAMS ML083040443), PUR-1 requested to operate the PUR-1 facility at 12.0 kW normal power. TS 3.2 Table I identifies the safety channels required for operation that includes the Log N and period

and safety channels. The setpoint for the Log N and period slow scram function is 120% power. The setpoints for the safety channel setback function is 110% and for the fast scram is 120% power. The setpoint values are not consistent with the TS 2.2 power level scram of 12.0 kW, since the requested normal power operating level is 12.0 kW. Describe and explain the relationship between the power level scram and the setpoint values for the Log N and period and safety channels.

4. TS 3.5(g): PUR-1 TS 3.5(g) states that "The radioactive material content, including fission products, of any double encapsulated experiment or vented experiment should be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation or confining boundary of the experiment could not result in (1) a dose to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release in excess of 0.5 Rem to the whole body or 1.5 Rem to the thyroid or (2) a dose to any person occupying a restricted area during the length of time required to evacuate the restricted area in excess of 5 Rem to the whole body or 30 Rem to the thyroid."

   These dose limits do not meet the requirements in 10 CFR Part 20. Propose revised dose limits in accordance with 10 CFR 20.1201, "Occupational dose limits for adults," and 20.1301, "Dose limits for individual members of the public," or explain why the above limits are acceptable.

5. TS 5.2.4: NUREG-1537, Part 1, Section 5.1, "Summary Description," and 5.2 "Primary Coolant System," provides guidance for licensees to provide the design bases and the functional requirements of the primary and secondary cooling system. In Section 5.3, "Secondary Coolant System," of the safety analysis report (SAR), the licensee states that when operating the PUR-1 at a power level of 10 kW, the calculated reactor pool temperature rise would be 0.465 degree Celsius per hour (taking no credit for heat loss to the surrounding sand and gravel or loss by evaporation). Further, the licensee states that since the heat-removal capacity of the heat exchanger is 10,550 Watts, the heat exchanger will maintain the pool temperature at an acceptable level.

   PUR-1 TS 5.2.4, describes the Primary Coolant Chiller System and states that this heat removal capacity is sufficient to maintain the pool temperature at 75 degrees Fahrenheit (23.09 degrees Celsius) at continuous operation at 10 kW. However, no technical specification is provided for the maximum allowable pool temperature when operating. In addition, the PUR-1 license renewal application requests a continuous operating power level of 12.0 kW, which is above the temperature at which analyses were performed. The PUR-1 SAR, Section 4.6, Table 4-18, "Model Dimensions for the Thermal Hydraulic Models," provides the conditions for the performed thermal hydraulics safety analysis and showed that when operating the reactor core at 10 kW, with a pool bulk temperature at 30 degrees Celsius, the maximum fuel temperature is well below the fuel temperature safety limit.

   Provide an analysis of the cooling system, including pool temperature and operating temperature such that the reactor operates within analyzed conditions. Provide a technical specification that ensures that the reactor pool bulk temperature is always maintained below the SAR-analyzed temperature, under all operating conditions, with steady-state operating reactor core power of 12 kW.

6. NUREG-1537, Part 1, Section 13.1.2, “Insertion of Excess Reactivity,” provides guidance for the analysis of insertion of excess reactivity. In Section 13.2.2, of the SAR, the licensee presents two reactivity insertion scenarios, one with and one without a scram. The initial power assumed by PUR-1 at the initiation of the reactivity insertion accident is 10.0 kW. Provide a discussion and supporting analyses that shows that the 10 kW initial power level provides the most limiting transient results. If 10 kW is not limiting, provide the limiting analysis. Include consideration for power level measurement uncertainty as well as the power level scram at 12.0 kW. Alternatively, demonstrate that the analysis results are insensitive to initial power.

RAIs 7-9 are based on a review of your responses to RAI 8 in a letter dated July 24, 2015, to NRC’s letter dated August 29, 2014 (ADAMS Accession No. ML14115A221).

7. Guidance in NUREG-1537 states that the format and content of the TSs follow ANSI/ANS 15.1-2007, Section 3.8.2, provides guidance for experiments involving fissionable, explosive, reactive, or corrosive materials.

   RAI 8(c) in NRC letter dated August 29, 2014, stated:

   > NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The PUR-1 MHA accident analysis for “Failure of a Fueled Experiment” is stated to be based upon a 1 W power deposition in the fueled experiment as consequence of the reactor operating at 1 kW. Please provide … a safety analysis that provides the details of the energy deposition determination in the fueled sample with the reactor operating at the maximum requested licensed reactor power including the power level measurement uncertainty of 50% stated in SAR, Section 13.1.2.

   The response to RAI 8(c) in your letter dated July 24, 2015, indicates that a limit of 0.5 Ci of radio-iodine is specified in the PUR-1 TSs that is half the amount analyzed in the MHA. Your analysis indicates that this limit corresponds to 1.19 gram of fissile material at the proposed operating power of 12 kW (plus 50% margin and assuming 3% enriched uranium). The analysis further indicates that a potential failure of an experiment with the maximum allowable fissile material content results in dose rates which are well below the dose rates of a maximum hypothetical accident (MHA) event both for an occupational worker and also a member of the public.

   The requirement of a maximum allowable limit of 0.5 Ci radio-iodine for experiments containing fissile material does not seem to appear in the PUR-1 TSs. Amend PUR-1 TSs to include this requirement, provide a reference to where this requirement exists either in the TSs or other controlling procedure, or provide an explanation describing your reason(s) for not including the maximum allowable limit of 0.5 Ci radio-iodine for experiments containing fissile material.

8. NUREG-1537, Part 1, Section 13.1.6, “Experiment Malfunction,” and 13.2, “Accident Analysis and Determination of Consequences, “provides guidance for analyzing experiment failures including the evaluation of the potential radiological consequences. The radiological consequences should include external and internal exposure and

provide an estimate of the whole-body Total Effective Dose Equivalent (TEDE) values for an occupational worker and a member of the public. The dose conversion calculations may use the Environmental Protection Agency's Federal Guidance Report (FGR)-11 and FGR-12 dose conversion coefficients or another equivalent methodology to account for inhalation/ingestion and submersion exposures. Your analysis seems to derive dose values by combining whole-body submersion with thyroid inhalation doses. Provide an updated analysis and discuss the potential maximum whole-body TEDE radiological dose estimate due to the failure of an experiment with the maximum allowable fissile material content or provide justification why it is not necessary.

9. NUREG-1537, Part 1, Section 13.1.6 and 13.2 (7) provides guidance for analyzing experiment failures and includes the evaluation of the potential radiological consequences. The radiological consequences should include external and internal exposure whole-body TEDE values for the duration of the accident. The response to RAI 8(c) in your letter dated July 29, 2015, indicates that the dose analysis was based on several exposure periods. Provide the total radiological whole-body TEDE (thyroid dose, if more limiting) to (1) an occupational worker, allowing for a realistic evacuation process, and (2) to the maximally exposed member of the public considering any emergency evacuation plan.

RAI 10 is based on a review of your responses to RAI 7 in a letter dated July 24, 2015, to NRC's letter dated August 29, 2014 (ADAMS Accession No. ML14115A221).

10. RAI 7 in NRC letter dated August 29, 2014, stated:

> The requirements of 10 CFR 20.1101 states that each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities, in order to limit the total effective dose equivalent to facility workers (annual occupational dose less than 5 rem [roentgen equivalent man]) and the total effective dose equivalent to individual members of the public (annual public dose less than 100 mrem). Please provide a safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential estimate of the total annual production of argon-41 from PUR-1 normal operations. In addition, please evaluate and discuss the potential maximum dose to a facility worker and to a member of the public (i.e., classrooms, hallways adjacent rooms, nearest dormitories, offices, etc.) due to this bounding yearly production and release of argon-41 from the facility.

Your response to RAI 7 by letter dated July 24, 2015, provided an analysis on Ar-41 production in the PUR-1 facility during normal operation due to air being absorbed in the reactor coolant and activated in the reactor core. The maximum allowable Ar-41 concentration for occupational workers is $3x10^{-6}$ $\mu Ci/cm^3$ and the public $1x10^{-8}$ $\mu Ci/cm^3$ established in 10 CFR Part 20, Appendix B, Table 2. The PUR-1 analysis estimates substantially higher concentration levels. Provide an updated Ar-41 safety analysis that explains all analyses and assumptions, and how it conforms to the requirements of 10 CFR Part 20 for occupational workers and members of the public.

RAIs 11-14 are based on a review of your responses to RAI 12 in a letter dated July 24, 2015, to NRC's letter dated August 29, 2014 (ADAMS Accession No. ML14115A221).

11. RAI 12(a) in NRC letter dated August 29, 2014, stated:

> Provide an MHA safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential estimate of the total radioactive fission product release after the failure of one side of one fuel plate. Discuss methodological assumptions associated with the following analytical steps:
>
> (a) Derivation of fission product atmospheric dispersion factor, $\chi/Q$ using either the methodology suggested in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments At Nuclear Power Plants," Revision 1, issued February 1983, or another equivalent method.

Your response to RAI 12(a) by letter dated July 24, 2015, provided an analysis of the atmospheric dispersion factor, $\chi/Q$ using Eq. (1) from RG 1.145. However, RG 1.145 suggests that the final $\chi/Q$ value should be selected through an evaluation procedure by comparing Eq. (1) with (2) (select higher), and then compare with Eq. (3) (select lower). Please discuss whether your selection of $\chi/Q$ based on Eq. (1) in RG 1.145 followed the RG methodology and is a conservative assumption and provides a bounding value for the atmospheric dispersion factor.

12. RAI 12(b) in NRC letter dated August 29, 2014, stated:

> (b) Dose conversion calculation using the Environmental Protection Agency's Federal Guidance Report (FGR)-11 and FGR-12 dose conversion coefficients or another equivalent methodology to account for inhalation/ingestion and submersion exposures.

Your response to RAI 12(b) by letter dated July 24, 2015, seems to provide occupational and public dose values by combining thyroid inhalation and whole-body submerged dose estimates. The radiological consequences should include external and internal exposure and provide an estimate of the whole-body TEDE values for an occupational worker and a member of the public. The dose conversion calculations may use the Environmental Protection Agency's Federal Guidance Report (FGR)-11 and FGR-12 dose conversion coefficients or another equivalent methodology to account for inhalation/ingestion and submersion exposures. Provide an updated analysis and discuss the potential maximum whole-body TEDE radiological dose estimate due to the MHA.

13. NUREG-1537, Part 1, Section 13.2 provides guidance for analyzing the MHA event including the evaluation of the potential radiological consequences. The radiological consequences should include external and internal exposure whole-body TEDE values for the duration of the accident. The response to RAI 12(b) in your letter dated July 24, 2015, indicates that the dose analysis was based on several exposure periods. Provide the total radiological whole-body TEDE (thyroid dose, if more limiting) dose estimate to

an occupational worker allowing for a realistic evacuation process and also to the maximally exposed member of the public considering any emergency evacuation plan.

14. NUREG-1537, Part 1, Section 13.2 provides guidance to licensees to systematically analyze and discuss credible accidents in each accident category. A postulated MHA event may result in gamma-ray radiation levels in the class room areas above the reactor room shielded by a concrete floor due to the dispersed fission products in the reactor room. Provide an analysis estimating the consequent maximum dose rates in the class room areas including accumulated doses to the maximally exposed member of the public considering procedures required by your Emergency Plan. The results should show compliance with the regulations in 10 CFR Part 20.

RAI 15 is based on a review of your responses to RAI 14 in a letter dated July 24, 2015, to NRC's letter dated August 29, 2014 (ADAMS Accession No. ML14115A221).

15. RAI 14 in NRC letter dated August 29, 2014, stated:

    10 CFR Part 20, "Standards for Protection against Radiation," provides the regulatory framework and NUREG-1537, Part 1, Section 13.1.3 provides the guidance for licensees to systematically analyze and discuss credible accidents in each accident category. Section 13.1.3 of the updated PUR-1 SAR, describes the loss of coolant accident (LOCA) scenario. The updated PUR-1 SAR does not include an estimate for radiation levels in the reactor floor and the roof areas, due to the unshielded reactor core, after a postulated large LOCA event. The SAR should provide the consequent maximum dose rates at various locations on the reactor floor and outside on the reactor building roof. In accordance with 10 CFR Part 20, provide the accumulated doses to reactor building occupants and the maximally exposed member of the public, considering evacuation procedure and potential residence time for staff. In addition, provide an estimate when facility staff may enter the reactor building to start recovery operations**.**

    Your response to RAI 14 by letter dated July 24, 2015, provides a dose rate estimate of 6.6 Rem/hr for a member of the public in the class room area. Provide an analysis and discuss the maximum radiological dose estimate due to the LOCA gamma-ray shine considering any evacuation procedure and potential residence time and demonstrating compliance with 10 CFR Part 20.

16. TS 5.3: TS 5.3, Reactor Core and Fuel," describes normal core configurations. TS 5.3.6 states ""Representative fuel assemblies shall be inspected annually, with no interval to exceed 15 months." However, there is no definition of "representative" within the TS. Further, although it is recommended by ANSI/ANS 15.1, there is no description of conditions for operating the reactor with damaged fuel elements. Provide technical specifications that specifically describe requirements for inspection of the fuel as well and conditions for operating the reactor with damaged fuel elements or provide an explanation why this is not necessary.

17. 10 CFR 50.36 (a)(1), "Technical Specifications," states "A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications." Provide bases for the technical specifications in Section 5.