July 14, 2011

Mr. Jere H. Jenkins
Director of Radiation Laboratories
Purdue University
Nuclear Engineering Building
400 Central Dr.
West Lafayette, IN 47907-2017

SUBJECT: PURDUE UNIVERSITY - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PURDUE UNIVERSITY REACTOR LICENSE RENEWAL APPLICATION (TAC NO. ME1594)

The U.S. Nuclear Regulatory Commission (NRC) is continuing our review of your application for renewal of Facility Operating License No. R-87, Docket No. 50-182 for the Purdue University Reactor (PUR-1) dated July 7, 2008, as supplemented by letters dated June 3, and June 4, 2010. During our review of the documentation for your renewal request, questions have arisen for which we require additional information and clarification. Enclosed is a partial request for additional information. We are requesting a response to this request within 90 days of the date of this letter. Additional requests for information containing our remaining questions were previously sent under separate cover.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.30(b), 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 security, 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." Following receipt of the additional information, we will continue our evaluation of your renewal request.

If you have any questions regarding this review, please contact me at (301) 415-3724 or by electronic mail at duane.hardesty@nrc.gov.

Sincerely,

/RA/

Duane A. Hardesty, 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 w/encl: See next page

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202 Nuclear Sciences Center
Gainesville, FL 32611

If you have any questions regarding this review, please contact me at (301) 415-3724 or by electronic mail at duane.hardesty@nrc.gov.

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Duane A. Hardesty, Project Manager
Research and Test Reactors Licensing Branch
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Office of Nuclear Reactor Regulation

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

FOR THE RENEWED FACILITY OPERATING LICENSE

PURDUE UNIVERSITY RESEARCH REACTOR

LICENSE NO. R-87

DOCKET NO. 50-182

The purpose of these questions is to assist the U. S. Nuclear Regulatory Commission (NRC) staff in determining that the renewal application from the Purdue University Research Reactor (PUR-1) meets the requirements of the regulations, in particular the regulations in Title 10 of the *Code of Federal Regulations* Part 20 and 10 CFR Part 50. The questions are based on a review of your application using the NRC staff's standard review plan in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 2, "Standard Review Plan and Acceptance Criteria."

We have divided our questions into three groups; the below complex questions have a 90-day requested response time, questions that may require outside resources with a 60-day requested response time, and the remaining questions with a 30-day requested response time. The three groups of questions considered together result in a complete set of consecutively numbered questions. The numbering of the questions below starts where the previous request stopped.

You are requested, where appropriate, to provide specific references in the responses to prior analyses from previous Safety Analysis Reports (SARs) and to provide updates to assumptions and resulting conclusions of these analyses.

For the below questions, we are requesting a response within 90 days of the date of this letter:

62. Inconsistencies are noted throughout the SAR for referenced maximum power and requested maximum licensed power under the PUR-1 license renewal. For example, in Section 1.2 of the SAR, PUR-1 is requesting a license "for a power uprate to continuous operation at 10 kW, and maximum short term power of 12.5 kW." Any reactor power excursions, whether short term or long-term must be within the maximum licensed power for the facility. Please clarify the desired maximum licensed power level requested and ensure this power level, including any uncertainty in reactor power, is consistently applied in the safety analyses for the license. Please provide updates to the safety analysis that correctly reflects the desired maximum licensed power under this renewal of the PUR-1 facility license.

63. Major inconsistencies are noted throughout the SAR related to calculation assumptions for initial and requested maximum licensed power under the PUR-1 license renewal. For example, SAR Section 13.2.2, p. 13-11, references current licensed power of 1 kW for a reactivity insertion with scram. Please clarify the desired maximum licensed power level

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requested and ensure this power level, including any uncertainty in reactor power, is consistently applied in the safety analyses for the license. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested maximum licensed power level.

64. SAR, Section 2.5.2 states “it is highly unlikely that any reactor water would be lost during any severe seismic activity” but no indication is provided that damaging seismic activity to other structures, systems or components (SSC) is unlikely. Please provide an evaluation of a safety analysis indicating that if any seismic activity were to occur, the radiological consequences are bounded, indicate where the analysis occurs in the SAR or justify why an analyses is not required.

65. SAR section 4.6.2, page 4-42 references the margin to incipient boiling shown in Table 4-28 calculated at an operating power of 1 kW for PUR-1. Table 4-28 is not provided in the SAR. Please update the SAR to provide the correct reference for the margin to incipient boiling and ensure that the parameters in the analysis support the maximum reactor power requested, including any uncertainty for power measurement. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested maximum licensed power level.

66. SAR, Section 4.6.3.1 and Table 4-22 provides a summary for the onset of nucleate boiling. The margin to incipient boiling provided in the SAR was calculated using reactor power levels of 1 kW and 10 kW. Using the guidance of NUREG-1537, Section 4.6, please provide an updated evaluation of a safety analysis that explains all thermal margin calculations, including the margin to incipient boiling, at the requested maximum licensed power level with uncertainty, including power measurement uncertainty.

67. SAR, Section 4.6.3.3 and Table 4-23 has an associated 50% uncertainty with the PUR-1 maximum reactor power level. SAR, Section 13.1.2 has stated that there is a 50% uncertainty in power measurement (p. 13-1). Using the guidance of NUREG-1537, Section 4.5 and Section 4.6, please provide: (1) an updated evaluation of a safety analysis that explains the power level uncertainty in terms of both the reactor power that is read by the reactor operator from the reactor power instrumentation and the limiting safety system settings of the reactor protective system, and (2) an updated evaluation of a safety analysis that explains and justifies the derivation of the estimate of 50% power measurement uncertainty.

68. SAR, Section 4.6.3.3, p. 4-46 states the Limiting Safety System Setting conditions for PUR-1 thermal-hydraulic calculations were computed using the NATCON code using power levels of 1 kW. Please clarify the desired maximum licensed power level requested and consistently apply this power level, including any uncertainty in reactor power, in the safety analyses for all thermal-hydraulics results. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested maximum licensed power level.

69. The requirements of 10 CFR 20.1201 include limiting the total dose equivalent to facility staff and the public from licensed reactor operations. In Section 5.6 of the SAR, it states that no nitrogen-16 activity has been observed to date in the reactor room. This

referenced observation is known to be at a power level of 1 kW, based on previous licensed power for PUR-1. Please provide an updated evaluation of a bounding safety analysis that explains all analyses, assumptions, and conclusions at the requested licensed power level for the maximum potential release of N-16 from the pool water into the reactor room and any potential dose to the facility staff and members of the public (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.).

70. NUREG-1537, Section 11 provides guidance for radiation protection provisions at the facility. In Section 4.4 of the SAR, it is stated that the radiation level above the reactor pool surface is about 1 mrem/hr and that the radiation level along the outside lateral surface of the concrete biological shield is about 0.1 mrem/hr, when the core is operating at 1 kW. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential radiation levels and the potential radiation effects on facility staff. As part of evaluation, please indicate if the radiation levels bound those that would be encountered during fuel handling and maintenance operations. Additionally, include an evaluation of the safety analysis for potential dose to the facility staff and members of the public (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.).

71. 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) and the total effective dose equivalent to individual members of the public (annual public dose less than 100 mrem). Please provide an evaluation of 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.

72. NUREG-1537, Section 9.3 provides guidance for fire protection systems and programs at the facility. SAR section 9.3 discusses the fire protection system as being “appropriate for the types of fires that could be encountered in the reactor room.” However, Section 9.3 of the SAR does not discuss methods to detect, control, or extinguish fires. Please indicate the means the facility has for preventing, detecting, and combating fires and the facility compliance with the local national fire codes. Please provide an evaluation of the safety analysis for fire protection with sufficient information describing potential radiological consequences of a fire will not prevent safe reactor shutdown or result in a fire-related release of radioactive material or if such a release is possible, that release of radioactive material would not cause radiation exposure that exceeds the requirements of 10 CFR Part 20.

73. NUREG-1537, Section 11.1.7 provides guidance for environmental monitoring at the facility. SAR section 11.1.7.2 generally discusses perimeter monitoring at the PUR-1 facility. Please provide additional information on the number and types of dosimeters that are deployed within and beyond the site boundary and describe the environmental

monitoring program is effectively implemented for the day-to-day operation of the facility, and that any radiological impact on the environment will be accurately assessed.

74. Section 13.3 of NUREG-1537 states “Maximum acceptable non-conservative instrument error may be assumed to exist at accident initiation.” Please provide all accident analysis calculation results performed with the assumption that the power level is the maximum requested power level plus the justified uncertainty resulting from power level measurement and other uncertainty.

75. NUREG-1537, Section 13 provides guidance to the licensee to identify the limiting event for each accident group and to perform quantitative analysis for that event. Please identify and discuss the potential methods whereby excess reactivity could be inserted into the reactor to cause an excursion. Please provide an evaluation of a safety analysis that quantitatively evaluates the most limiting event for addition of excess reactivity.

76. NUREG-1537, Section 13.1.2 provides guidance for analysis of insertion of excess reactivity. In Section 13.2.2 of the SAR the licensee reports two reactivity insertion scenarios: the first with scram, the second without scram.

    A. For the scram assumption scenario, the accident is assumed by PUR-1 to be initiated at a power of 10 kW and scram is assumed to occur on a power trip at 12 kW. This assumption, however, does not appear to account for the power measurement error and uncertainty. Please revise the assumption for this calculation and provide an evaluation of a safety analysis at the requested maximum licensed power level including power level measurement uncertainty.

    B. For the no-scram assumption scenario, the initial power assumed by PUR-1 before the accident is 10 kW and does not appear to account for the power measurement error and uncertainty. Please revise the assumption for this calculation and provide an evaluation of a safety analysis at the requested maximum licensed power level including power level measurement uncertainty.

77. NUREG 1537, Part 2, Chapter 13 states credible accidents should be categorized and the most limiting accident in each group should be analyzed in detail including the potential consequences of the various accident scenarios including loss-of-coolant accident (LOCA) events.

    A. Please provide an evaluation of a safety analysis of the LOCA accident sequence assuming the maximum licensed power level including uncertainty resulting from power level measurement uncertainty.

    B. Please provide an evaluation of a safety analysis for safe cooling of the fuel during complete loss of coolant event at the peak fuel power densities for the maximum requested licensed power level.

    C. Please provide an evaluation of a safety analysis for the slow draining process, which may result in a partially uncovered core (partial LOCA), that may not be cooled by assuming a continuous circulation of air. Please discuss a partial LOCA scenario and indicate whether the fuel temperature in a partially uncovered core is still bounded by the SAR LOCA analysis.

78. NUREG-1537, Part 1, Section 13.1.4 provides guidance for analysis of loss-of-coolant flow resulting from blocked fuel cooling channels.

    A. Please provide an evaluation of a safety analysis that provides a complete assessment of the potential for fuel channel blockages and how adequate heat transfer during such blockages is maintained.

    B. Please discuss facility procedures or any other blockage-mitigating PUR-1 design features for foreign material exclusion from entry to the reactor pool in order to prevent blockage of coolant channels.

79. NUREG-1537, Section 13 provides guidance to identify the limiting event for each accident group and to perform quantitative analysis for that event. Please identify the categories of PUR-1 experiments that are performed and provide an evaluation of a safety analysis using the guidance of NUREG-1537, Section 13.1.6 for potential experiment malfunctions and their consequences.

80. NUREG-1537, Section 13 provides guidance to the licensee to consider accident initiators that do not fall into the other categories, such as operator errors, instrument or control malfunctions, electrical faults, and others. Please provide an evaluation of a safety analysis using the guidance of NUREG-1537, Section 13.1.7 that considers the range of PUR-1 operations for possible scenarios involving the potential for equipment malfunction and potential consequences.

81. NUREG-1537, Part 1, Section 13.1.1 provides guidance to identify Maximum Hypothetical Accidents (MHA) for non-power reactors. The MHA is to be selected so potential consequences of the postulated MHA scenario exceed and bound all credible accidents. NUREG-1537, Part 2, Chapter 13, p. 13-5 suggests that for a low-powered MTR fueled reactor, the MHA may be one of the following two events: cladding is stripped from one face of a fuel plate while suspended in air, or a fueled experiment fails in air. SAR, Section 13.1 states that “the failure of a fueled experiment is designated as the maximum hypothetical accident of the PUR-I.” Please provide an evaluation of a safety analysis of an MHA that considers the failure of one fuel plate in air would have lower consequences than the failure of a fueled experiment by justifying the MHA accident involving the fueled experiment capsule is more bounding than the failure of a fuel plate.

82. 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 an evaluation of 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.

83. NUREG-1537, Part 1, Section 13.1.5 provides guidance for analyzing accidents for mishandling or malfunction of fuel for non-power reactors. Please provide an evaluation of a safety analysis that provides the MHA fission product inventories at the requested maximum PUR-1 licensed power, including power level measurement uncertainty.

84. NUREG-1537 states that the format and content of the TS follow ANSI/ANS 15.1. ANSI/ANS-15.1-2007, Section 3.8.2 provides guidance for double encapsulation of experiments involving fissionable, explosive, reactive, or corrosive materials. Please provide an evaluation of a safety analysis for the MHA experiment of 1.1 g of U-235 with single encapsulation is consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 3.8.2.

85. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. Please explain the use of the fission product inventory from Table A-1 in "*Nuclear Power Reactor Safety*" (E.E. Lewis, 1977) as the basis instead of calculating the inventory in the experiment or the fuel. Please provide an evaluation of a safety analysis that substantiates PUR-1's chosen method of establishing the fission product inventory for the singly encapsulated experiment or the fuel is reliable and conservative.

86. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The NRC staff is unable to reproduce the dose rate results for a failed fueled experiment to the facility workers due to a 1.5 minute exposure in the reactor room for the thyroid dose, the skin dose, and the whole body dose of Table 13-2 in the SAR. Please provide more details for these calculations including verifying that these results are applicable to the maximum requested licensed power for PUR-1 operations.

87. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The NRC is unable to reproduce the γ, β and thyroid dose rates based on the dispersion factor of 1.78×10^{-4} s/m^3 tabulated in Table 13-3 of the SAR. Please provide more details for these calculations including verifying that these results are applicable to the maximum requested licensed power for PUR-1 operations

88. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The NRC is unable to reproduce the results for a dispersion factor (DF) of 1.78x10-4 s/m3, using Pasquill Type F conditions, wind speed of 1 m/s, and a downstream receptor distance of x = 100 m provided in Section 13.2.1, p.13-9 of the SAR. Please provide more details for the derivation and calculation to obtain the stated result in the SAR.

89. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1 *Maximum Hypothetical Accident (Failure of a Fueled Experiment)* states on page 13-7 "The calculated saturation activity for each respective radioisotope and its concentration is the Reactor Room after experiment failure is shown in Table 7.5." Table 7.5 does not exist in SAR. Please update the SAR to provide the referenced saturation activities or state where the information exists in the SAR.

90. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1, page 13-8 states "The radioactive material content, including fission products, of any singly encapsulated experiment should be limited…" Please provide an evaluation of a safety analysis that demonstrates

the radioactive material content will be limited to meet annual doses stated in 10 CFR Part 20 and will not be exceeded for release of all gaseous, particulate, or volatile components from any singly encapsulated experiment.

91. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1, page 13-8 states "The radioactive material content, including fission products, of any doubly encapsulated experiment or vented experiment should be limited…". Please provide an evaluation of a safety analysis that demonstrates the radioactive material content will be limited to justify annual doses stated in 10 CFR Part 20 will not be exceeded for release of all gaseous, particulate, or volatile components from any doubly encapsulated experiment.

92. SAR, Section 13.2.1, makes reference to restricted and unrestricted areas. These types of areas are not defined in the SAR or emergency plan. Please update the SAR and/or emergency plan to use consistent designations or provide the definition of these areas and explanation of relationship to defined areas such as the operations boundary, site boundary, reactor building, or nuclear engineering lab.

93. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1, page 13-8 states the dose to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release. Please provide an evaluation of a safety analysis that substantiates these stated doses for members of the public who may be in the uncontrolled areas of the Engineering building during an MHA event.

94. SAR, Section 13.2.1 discusses doses to any person occupying an unrestricted area continuously for a period of two hours. Please provide an evaluation of a safety analysis that justifies the basis for the 2 hour limit and discuss any requirements for evacuation of the unrestricted area, if applicable.

95. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1, page 13-8 states the dose to any person occupying a "restricted area" during the length of time required to evacuate. Please provide an evaluation of a safety analysis that substantiates these stated doses for personnel, including members of the public who may be in the restricted areas of the Engineering building during an MHA event.

96. SAR, Section 13.2.1 discusses calculation of the reactor building. Please explain if the area evacuated within the stated time constitutes the operations or site boundary. Additionally, please provide documentation substantiating the "past experience" for the evacuation time.

97. SAR, Section 7.5.4 designates the *hallway* immediately outside the reactor room as B76A. This room appears to be an interior room within the Nuclear Engineering laboratories. Please update the SAR for consistency of designated areas.

98. SAR, Section 13.2.1, page 13-9 states "*This radiation exposure approaches the limits established in the Technical specifications, Sec 3.5.f for a singly encapsulated experiment.*" The proposed PUR-1 TSs do not contain a TS 3.5.f. Please propose a TS

to establish limits for a singly encapsulated experiments, reference the applicable TS, or justify why a TS is not required.

99. SAR, Section 13.2.1, page 13-9 states "*This experiment corresponds to the irradiation of 1.1 gm of U-235 in the mid-plane of the isotope irradiation tube located in position F6.*" Please provide an evaluation of a safety analysis that establishes the basis of 1.1 gm of U-235 for failure of a fueled experiment.