April 25, 2016

Dr. Robert Dimeo, Director
NIST Center for Neutron Research
National Institute of Standards and Technology
U.S. Department of Commerce
100 Bureau Drive, Mail Stop 8561
Gaithersburg, MD 20899-8561

SUBJECT: NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY – PRELIMINARY SAFETY ANALYSIS REPORT FOR THE NATIONAL BUREAU OF STANDARDS REACTOR (TAC NO. MF7235)

Dear Dr. Dimeo:

By letter dated December 30, 2014, the National Institute of Standards and Technology National Bureau of Standards Test Reactor (NBSR) staff submitted a Preliminary Safety Analysis Report (PSAR) for conversion of the NBSR from the use of highly-enriched uranium fuel to low-enriched uranium fuel to the U.S. Nuclear Regulatory Commission (NRC). Given the current estimate from the Department of Energy for completion of NBSR's highly-enriched uranium to low-enriched uranium fuel conversion of the mid-2020s, the purpose of submitting the PSAR was to request NRC review of analysis that has been completed.

During our review, questions have arisen for which additional information and clarification is needed. The enclosed request for additional information (RAI) identifies the additional information needed to continue our review. We request that you provide responses to the enclosed RAI within 90 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 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 license amendment request.

If you have any questions, please contact me at (301) 415-1404 or by electronic mail at Xiaosong.Yin@nrc.gov.

Sincerely,

/RA/

Xiaosong Yin, Project Manager
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-184

Enclosure:
As stated

cc: See next page

If you have any questions, please contact me at (301) 415-1404 or by electronic mail at Xiaosong.Yin@nrc.gov.

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

REGARDING PRELIMINARY SAFETY ANALYSIS REPORT

NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY

NATIONAL BUREAU OF STANDARDS TEST REACTOR

LICENSE NO. TR-5; DOCKET NO. 50-184

By letter dated December 30, 2014 (Agencywide Documents Access and Management System Accession No. ML15028A135), the National Institute of Standards and Technology (NIST) National Bureau of Standards Test Reactor (NBSR) (the licensee) submitted a Preliminary Safety Analysis Report (PSAR) for the conversion of the NBSR from highly-enriched uranium (HEU) to low-enriched uranium fuel.

The U.S. Nuclear Regulatory Commission is reviewing your PSAR regarding the technical adequacy of the document which impacts the licensed facility. After reviewing the PSAR, questions have arisen that require additional information and clarification. Please provide responses to the request for additional information within 90 days of the date of the cover letter.

1. The PSAR states that a Monte Carlo Neutron Photon (MCNP) computer model of the NBSR core had been developed, and a version of the MCNP was used as the primary reactor physics modeling tool.

   NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," Section 4.5, "Nuclear Design," states, in part, that "Computer codes that are used should be described in detail as to the name and type of code, the way it is used, and its validity on the basis of experiments or confirmed predictions of operation non-power reactors."

   The PSAR does not indicate such validity and/or comparisons that confirmed predictions have been established to confirm the acceptability of the MCNP model. In addition, in the NIST 2004 version of Safety Analysis Report (SAR), it states, in part, that "The power distribution, excess reactivity, shutdown margin, reactivity coefficients, and core thermal-hydraulic behavior are calculated and compared to measurements." However, such comparisons cannot be found in the SAR. In SAR Tables 4.5.4 and 4.5.2, there are some comparisons of control rods worth and some calculations to indicate critical condition reactivity differences, however, they do not indicate if there are acceptable agreements.

   Provide comparisons of calculated and measured parameters for NBSR that demonstrate that the MCNP model utilized is suitable for the prediction of parameters to assess the validity of the use of the MCNP model. At a minimum, this should include

estimated critical position, control rod worth, and the isothermal-temperature coefficient for sufficient cases to demonstrate validity. Any such comparison should also establish the basis for acceptability.

2. NUREG-1537, Part 1, Section 4.5.1, "Normal Operating Conditions," asks that "the applicant present information on the core geometry and configurations."

   PSAR Section 4.5.1.2 states that 60 unique homogenized fuel element compositions are used to track the burnup of the 1020 fuel plates in the active core. The modelling approach reviewed indicates that there are no burnup gradients established in the fuel depletion calculations along the width of a plate, along the length of a plate, or from the plate-to-plate transition; the burnups are averaged over the upper and the lower plate of each assembly. Explain how your methodology ensures that the core thermal power and neutron flux spatial distributions are established so that reactivity predictions are suitably accurate and thermal-hydraulic (T-H) conditions are suitably conservative.

3. The guidance in NUREG-1537, Part 1, Section 4.6, "Thermal-Hydraulic Design," states, in part, that "The calculational methodology should be applicable to the thermal-hydraulic operating conditions, and the applicant should justify its use."

   a) The PSAR Section 4.6.2.1 states that the Sudo-Kaminaga critical heat flux (CHF) correlation was used to characterize the departure from nuclear boiling ratio (DNBR) of the NBSR. That discussion describes the CHF experiments conducted in that research project, but does not refer to or demonstrate the acceptability of the results obtained. Provide the basis for the acceptability of this correlation for the NBSR.

   b) The PSAR Section 4.6.2.2 states that the Saha-Zuber criteria was used to estimate the onset of net vapor generation and that this condition is assumed to be indicative of the onset of flow instability. Provide the basis for the acceptability of this correlation for the NBSR.

4. The guidance in NUREG-1537, Part 1, Section 13.1.3, "Loss of Coolant," states, in part, that "Some initiators of LOCAs [loss-of-coolant accidents] are the following: . . . failure or malfunction of some component in the primary coolant loop."

   The PSAR, at various times, hypothesizes the failure of the DWV-19 valve (PSAR Section 13.3, 13.3.1, 13.5.5, etc.) as part of the LOCA/loss-of-flow accident (LOFA). However, Figure 13.1 in the PSAR illustrates no such valve in the NBSR system. Confirm whether valve DWV-19 is the correct nomenclature for the valve in question, or whether Figure 13.1 under review needs to be updated.

5. NUREG-1537, Part 1, Section 4.6, "Thermal-Hydraulic Design," states, in part, that "for the core geometry and the coolant thermal-hydraulic characteristics, a discussion to establish the fuel heat removal conditions that ensure fuel integrity such as fuel surface saturation temperature, . . ., departure from nucleate boiling and/or flow instability."

   Figure 13.4 in the PSAR indicates that the model defined to describe for the T-H conditions - which has a distinct fuel meat and cladding in the heat structure - is different from the model used in the neutronics analysis, in which the meat and cladding appears

to be homogenized. Confirm that the manner in which the power distribution is determined and translated into the T-H model is suitably conservative for DNBR analysis.

6. NUREG-1537, Part 1, Section 4.5.2, "Reactor Core Physics Parameters," states, in part, that the applicant should provide the information on "the axial and radial distributions of neutron flux densities, justifications for the methods used, and comparisons with applicable measurements."

   PSAR Section 13.2.2 states that the power distributions used are conservative. However, because the upper fuel plates were modeled as homogeneous within each assembly and the same was done for the lower plates, the resulting burnup distribution appears to be flat and may not represent accurately the power distribution that is appropriate for estimated critical position (ECP) calculations, DNBR, or accident analysis. Confirm that the manner in which the fuel material is described in the MCNP model leads to suitably accurate ECPs and is also used in a conservative manner for DNBR and accident analysis, or describe how additional factors or penalties are used to achieve acceptable results.

7. NUREG-1537, Part 1, Section 4.6, "Thermal-Hydraulic Design," states, in part, that "The calculational methodology should be applicable to the thermal-hydraulic operating conditions, and the applicant should justify its use."

   The NBSR RELAP model combines the coolant loops into a single flow path. It is not clear that such a model can model multi-loop effects conservatively (e.g., single pump failure, single shaft seizure, LOCA, LOFA, and the asymmetric effects that they have on reactor performance under accident conditions [e.g., throttling flow to outer plenum] etc.). If there are additional components in the model, and the actual reactor, that contribute to this understanding (e.g., check valves), then provide an updated graphic that shows such components.

   To assist in the staff assessment of the RELAP model used for such purposes, provide the ASCII text input file of the actual DNBR model used. We may also need to obtain access to the NBSR drawings to verify certain aspects of the model.

8. NUREG-1537, Part 1, Section 13.1.2, "Insertion of Excess Reactivity," asks for the analysis of insertion of excess reactivity that includes a ramp insertion of reactivity by drive-initiated motion of the most reactive control rod or shim rod, or ganged rods, if possible. PSAR Section 13.4.2.1 describes an operator induced rod withdrawal error. It is unclear if this analysis meets the intent of the guidance. Identify the maximum blade withdrawal rate and the worth of the maximum worth blade, including any applicable uncertainties and ensure that the supplied analysis models this accident in a suitably conservative manner.

9. The guidance in NUREG-1537, Part 1, Section 13.1.4, "Loss of Coolant Flow," asks for the analysis of loss of coolant flow accident. PSAR Section 13.5.2.1 included the sequence of events (SOE) for the seizure of one primary pump. It is unclear whether this SOE is conservatively modeled, given the fact that the behavior of an idle loop,

including backflow, cannot be modeled using the RELAP model described. Provide a justification for the SOE specified using the supplied RELAP model, and show that the resulting analysis is justified and suitably conservative.

10. NUREG-1537, Part 1, Section 13.1.4, "Loss of Coolant Flow," asks for the analysis of a loss of coolant flow accident. PSAR Sections 13.5.3.1 and 13.5.4.1 discuss the SOE for throttling coolant flow to the outer/inner plenum. It is unclear whether this SOE is conservatively modeled, given the fact that the behavior of an idle loop cannot be modeled using the RELAP model described. Provide a justification for the SOE specified using the supplied RELAP model and show that the resulting analysis is justified and suitably conservative. Describe in details the location of piping and valves so as to understand whether there is any significant potential for asymmetric flow.

11. NUREG-1537, Part 1, Section 13.1.1, "Maximum Hypothetical Accident," asks for the analysis of maximum hypothetical accident (MHA).

    - Since NBSR is a closed system that contains the fission products, there is virtually no dose to occupational workers or public from an in-reactor failed fuel assembly, which is the MHA analyzed. However, the NBSR technical specifications (TSs) mention fueled experiments. Consider a failed experiment, a failed fueled experiment, a fuel handling accident, and a fuel disassembly accident occurring under conditions and at various locations (e.g., spent fuel storage locations, hot cells, fume hoods, reactor bay, etc.) that maximize their potential impact, and then determine whether one of these events, or the existing failed fuel assembly accident is the MHA for NBSR. Analyze the impact of the maximum source term in terms of the resulting dose to occupational workers and the public. Provide assumptions and analysis for these events for staff review.

    - For the fuel disassembly accident, consider events such as debris impingement, cladding longitudinal tears, cutting blade fracture, or any other credible failure that could lead to breaching the cladding-fuel interface.

    - The PSAR Table 13.21 indicated that there are smaller HEU inventories than those in the 2004 SAR. Explain the differences and explain the nomenclature used.

    - Confirm which HOTSPOT version was used in the licensee analysis. Provide HOTSPOT output for the limiting case.

    - Explain why the filters cited on page 148 are not controlled by the TSs.

    - Address the following possible inconsistencies:
      Should the reference to Table 13.24 be to 13.21 on page 146?
      Should the reference to Table 13.25 be to 13.22 on page 147?
      Should the reference to Table 13.26 be to 13.23 on page 148?
      Should the reference to Table 13.27 be to 13.24 on page 148?

12. NUREG-1537, Part 1, Section 13.1.5, "Mishandling or Malfunction of Fuel," asks for the analysis of mishandling or malfunction of fuel. PSAR Section 13.9 is titled "Mishandling, malfunction, or misloading of fuel." However, only misloading is analyzed. Provide a comprehensive analysis of both mishandling and malfunction events, or provide a justification for why such analysis is not required.

13. NUREG-1537, Part 1, Section 13.1.6, "Experiment Malfunction," asks for the analysis of experimental malfunction. PSAR Section 13.10 describes experiment malfunction, but did not characterize dose coming from fueled experiments such as those allowed under the TSs 3.9.1(1) and the basis. Provide your comprehensive analysis of the experimental malfunction.

14. NUREG-1537, Part 1, Section 13.1.6, "Experiment Malfunction," asks for the analysis of experimental malfunction. Confirm whether there are hot cells, fume hoods, or glove boxes at NBSR connected to a ventilation system that can exhaust to public receptors.

15. Provide reference copies of Brown 2013, as cited in PSAR 13.2.2, and NBS 1980, as cited on page 149.