UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 2, 2023

Dr. Thomas H. Newton, Deputy Director
National Institute of Standards and
Technology
NIST Center for Neutron Research
U.S. Department of Commerce
100 Bureau Drive, Mail Stop 6101
Gaithersburg, MD 20899-6101

SUBJECT: NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY – ISSUANCE OF AMENDMENT NO. 15 TO RENEWED FACILITY OPERATING LICENSE NO. TR-5 FOR THE NATIONAL BUREAU OF STANDARDS TEST REACTOR RE: IMPLEMENTATION OF AN ALTERNATIVE FUEL MANAGEMENT SCHEME (EPID L-2023-LLA-0016)

Dear Dr. Newton:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 15 to Renewed Facility Operating License No. TR-5 for the National Institute of Standards and Technology National Bureau of Standards Test Reactor (NBSR). This amendment consists of changes to the NBSR safety analysis report (SAR) in response to the application dated February 1, 2023 (Agencywide Documents Access and Management System Accession No. ML23033A114). Specifically, the amendment modifies the SAR to describe an alternative fuel management scheme and associated analytic methods.

This license amendment will inform the decision of the Commission whether to approve restart under Title 10 of the *Code of Federal Regulations* section 50.36(c)(1) related to the February 3, 2021, event at the NBSR, but the restart decision will not solely rely on this license amendment

A copy of the related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's monthly *Federal Register* notice. If you have any questions, please contact me at (301) 415-3936, or by email to Patrick.Boyle@nrc.gov.

Sincerely,

Patrick G Boyle Signed by Boyle, Patrick on 03/02/23

Patrick Boyle, Project Manager
Non-Power Production and Utilization Facility
Licensing Branch
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Docket No. 50-184
License No. TR-5

Enclosures:
1. Amendment No. 15 to Renewed Facility Operating License No. TR-5
2. Safety Evaluation

cc:

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Test, Research and Training
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SUBJECT: NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY – ISSUANCE OF AMENDMENT NO. 15 TO RENEWED FACILITY OPERATING LICENSE NO. TR-5 FOR THE NATIONAL BUREAU OF STANDARDS TEST REACTOR RE: IMPLEMENTATION OF AN ALTERNATIVE FUEL MANAGEMENT SCHEME (EPID L-2023-LLA-0016) DATED: MARCH 2, 2023

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ADAMS Accession No.: ML23055A300 **NRR-058**

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY

DOCKET NO. 50-184

NATIONAL BUREAU OF STANDARDS TEST REACTOR

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 15
License No. TR-5

1. The U.S. Nuclear Regulatory Commission (NRC, the Commission) has found that:

   A. The application for amendment to Renewed Facility Operating License No. TR-5, filed by the National Institute of Standards and Technology (the licensee) on February 1, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 15, Renewed Facility Operating License No. TR-5 is hereby amended to authorize the revision to the facility safety analysis report as set forth in the application and as evaluated in the NRC staff's safety evaluation issued with this amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Joshua M. Borromeo

Digitally signed by Joshua M. Borromeo

Joshua M. Borromeo, Chief
Non-Power Production and Utilization Facility
Licensing Branch
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Date of Issuance: March 2, 2023

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 15 TO

RENEWED FACILITY OPERATING LICENSE NO. TR-5

NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY

NATIONAL BUREAU OF STANDARDS TEST REACTOR

DOCKET NO. 50-184

1.0 INTRODUCTION

By letter dated February 1, 2023 (Agencywide Documents Access and Management System Accession No. ML23033A114), the National Institute of Standards and Technology (NIST, the licensee) submitted a license amendment request (LAR) to modify the National Bureau of Standards Test Reactor (NBSR) safety analysis report (SAR). Specifically, the amendment would revise the SAR to describe an alternative fuel management scheme (AFMS) and associated analytic methods.

2.0 EXIGENT CIRCUMSTANCES

The U.S. Nuclear Regulatory Commission (NRC, the Commission) regulations contain provisions for issuance of amendments when the usual 30-day public comment period cannot be met. These provisions are applicable under exigent circumstances. Consistent with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.91, "Notice for public comment; State consultation," paragraph (a)(6), exigent circumstances exist when: (1) a licensee and the NRC must act quickly; (2) time does not permit the NRC to publish a *Federal Register* (FR) notice allowing 30 days for prior public comment; and (3) the NRC determines that the amendment involves no significant hazards consideration. As discussed in the LAR, the licensee requested that the NRC process the proposed amendment on an exigent basis.

Under the provisions in 10 CFR 50.91(a)(6), the NRC notifies the public in one of two ways: (1) by issuing an FR notice providing notice of an opportunity for hearing and allowing at least 2 weeks from the date of the notice for prior public comments; or (2) by using local media to provide reasonable notice to the public in the area surrounding the licensee's facility. In this case, the NRC notified the public utilizing an FR notice allowing at least 2 weeks from the date of the notice for public comment. The notice was published on February 10, 2023 (88 FR 8918).

In its LAR, the licensee stated that exigent circumstances exist because the amendment is necessary to allow for the completion of refueling and a subsequent timely restart of the NBSR and because the need for the amendment was only recently identified.

Based on the above circumstances, the NRC staff finds that the licensee made a timely application for the proposed amendment following identification of the issue. In addition, the staff finds that the licensee could not avoid the exigency because the need for the amendment was identified during ongoing inspection activities and it is necessary for any restart of the NBSR. Based on these findings, and the determination that the amendment involves no significant hazards consideration as discussed below, the staff has determined that a valid need exists for issuance of the amendment using the exigent provisions of 10 CFR 50.91(a)(6).

3.0 REGULATORY EVALUATION

System Description

Chapter 4, "Reactor Description," of the NBSR SAR (ML041120210) describes the design of the reactor. The NBSR core contains thirty fuel elements. According to section 4.5.1.1.2, "Fuel Management Scheme," of the SAR, the ordinary core design includes a fuel management scheme that, for each operating cycle, introduces four fresh fuel elements, provides a defined shuffle pattern for 26 previously irradiated fuel elements, and discharges four previously irradiated fuel elements.

Under this fuel management scheme, approximately half of the fuel elements used in the core are irradiated for seven cycles and the other half are irradiated for eight cycles. Of the four fuel elements discharged each cycle, two had been irradiated for seven cycles (i.e., seventh-cycle elements) and two had been irradiated for eight cycles (i.e., eighth-cycle elements). NBSR technical specification (TS) 3.1.4, "Fuel Burnup," limits the exposure of fuel elements used in the NBSR, such that eighth-cycle elements approach this limit and, therefore, cannot practically be reinserted into the core without subsequently violating the limit. Therefore, the fuel management scheme in SAR section 4.5.1.1.2 describes that the two discharged, seventh-cycle elements are available for reinsertion and that the two discharged, eighth-cycle elements are not available for reinsertion and would instead be considered spent fuel.

Sections 4.5, "Nuclear Design," and 4.6, "Thermal Hydraulic Design," of the NBSR SAR contain nuclear and thermal-hydraulic analyses, respectively, that are based on a core design that follows the fuel management scheme of SAR section 4.5.1.1.2. Similarly, chapter 13, "Accident Analyses," of the SAR (ML041120237) contains safety analyses that are based on this fuel management scheme. The NRC staff reviewed and approved the analyses in SAR sections 4.5 and 4.6 and in SAR chapter 13 in concert with the renewal of the NBSR Facility Operating License by safety evaluation report (SER) dated June 2009 (ML090990135).

Description of the Proposed Changes

According to the LAR, NIST's current fuel inventory is such that any core that it may design to support a restart and near-term subsequent operations of the NBSR would not be able to adhere to the fuel management scheme described in SAR section 4.5.1.1.2 in that NIST would have to introduce more than four fresh fuel elements in its core designs. Therefore, NIST proposed to update the SAR to describe an AFMS. Specifically, the LAR would add a section titled "Alternative Fuel Management Schemes (AFMS)" to the SAR, which would include the following language:

> Henceforth, the "NBSR-0018-DOC-00 NBSR Alternative Fuel Management Schemes Analysis Procedure" details the bounding (limiting) conditions, analysis of methodology, neutronics and thermal-hydraulics models, relevant codes and scripts, correlations, accident scenarios to be evaluated, quality assurance, version control, and verification

and validation necessary to ensure the safety of any AFMS that may be proposed for NBSR operations.

The report referenced in the proposed language was included with the LAR and describes the analytic methods for cycle-specific safety analyses that NIST would perform when using an AFMS.

Applicable Regulatory Guidance

The NRC staff reviewed the LAR and evaluated the proposed changes to the SAR based on the following regulations and guidance:

- Part 50, "Domestic Licensing of Production and Utilization Facilities," of 10 CFR, which provides, in part, the regulatory requirements for the licensing of non-power reactors.

- Section 50.34, "Contents of applications; technical information," paragraph (b)(2) of 10 CFR, which requires in SARs a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

- Section 50.34(b)(4) of 10 CFR, which requires in SARs a final analysis and evaluation of the design and performance of SSCs with the objective stated in 10 CFR 50.34(a)(4), which requires analysis and evaluation of the design and performance of SSCs, including determination of the margins of safety, that considers any pertinent information developed since the submittal of the preliminary SAR.

- NUREG-1537, part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," chapter 14, "Technical Specifications" (ML042430048), which provides guidance to the NRC staff on reviewing non-power reactor license applications and SARs.

Since the NRC staff's review concerns proposed analytic methods, it does not address specific design features apart from how those design features are modeled. The staff did consider analyses submitted by NIST, however, insofar as they demonstrated that the NIST analytic methods indicate how acceptance criteria in the NBSR licensing basis (e.g., TS safety limits) are met, and that the results of the analyses are valid and credible.

Consistent with the acceptance criteria in NUREG-1537, part 2, section 4.5, "Nuclear Design," as applicable, the NRC staff reviewed the nuclear design analysis methods described in the LAR to determine whether:

- The information showed a complete, operable reactor core; that control rods remained sufficiently redundant and diverse to control all excess reactivity safely and to safely shut down the reactor and maintain it in a shutdown condition; and that the analyses of reactivities include individual and total control rod effects.

- Analyses showed initial and changing reactivity conditions, including the reactivity worths of control rods, fuel elements, and experimental facilities.

- The reactor kinetic parameters and behavior were shown.

Consistent with the acceptance criteria in NUREG-1537, part 2, section 4.6, "Thermal-Hydraulic Design," as applicable, the NRC staff reviewed the LAR to determine whether:

- The analyses establish the validity of existing safety limits established in the NBSR TS by demonstrating that those limits remain satisfied, with conservative consideration of the effects of uncertainties.

The NRC staff also reviewed whether the analyses similarly confirm the validity of existing limiting safety system settings (LSSSs) established in the NBSR TS by demonstrating that those LSSSs ensure that fuel integrity is maintained when protective action is initiated at the LSSSs.

4.0 TECHNICAL EVALUATION

As noted above, the NRC staff reviewed and approved the NBSR SAR in concert with the issuance of the NBSR Renewed Facility Operating License in 2009. Since then, NIST has updated the SAR consistent with changes to the facility made without prior NRC approval under 10 CFR 50.59, "Changes, tests and experiments." However, the staff bases its evaluation of NIST's proposed analytic methods, in part, on a consideration of the differences between the methods used to perform the safety analyses in the 2009 SAR and associated SER and those proposed in the LAR. This comparative approach facilitates an efficient review because the staff already concluded as part of its license renewal review that the safety analysis methods used in the 2009 SAR were acceptable, and NIST asserted in its LAR that the proposed methods are technically superior to the previously approved methods.

4.1 NUCLEAR DESIGN METHODS

Information Provided by the Licensee

As stated in the LAR, the NBSR reactor analysis is performed using the Monte Carlo N-Particle (MCNP) computer code, version 6.2. This is the same computer code as was used in the license renewal review; however, the licensee notes that the discretization in the core has been refined from 60 compositions to 720 compositions. The NRC staff understands this to mean that the core was previously modeled with a single composition for the upper and lower halves of each of the 30 fuel elements, whereas the current model uses 12 compositions for the upper and lower halves of each of the 30 fuel elements. To justify this change, the licensee compared the effective multiplication factors for a core modeled with the original fuel management scheme with both the 60- and 720-composition models. Given the same material compositions, regulating rod position, and shim arm angles in both models, the licensee indicated that the difference in the effective multiplication factors was less than 0.03 percent.

The licensee performed depletion calculations for eight 38-day cycles assumed to follow the original fuel management scheme (OFMS). This allowed the licensee to determine the fuel isotopic inventory entirely from the MCNP code. Then, the licensee calculated core isotopic inventories for a ninth cycle at four exposure points through a cycle: at startup, beginning-of-cycle (BOC), middle-of-cycle, and end-of-cycle (EOC). Section 2.2.1 of NBSR-0018-DOC-00 describes how NIST will determine fuel element compositions, with additional detail provided in appendix B. This involves performing a decay correction of isotopic inventory based on the actinide inventory estimated for seventh-cycle elements from the license renewal SAR. Section 2.13 of NBSR-0018-DOC-00 describes validation and verification requirements. These requirements are based on startup physics testing required by the NBSR TS, and require the measurement of regulating rod worth, bank shim worth at critical, individual shim worth, BOC core excess reactivity, and BOC shutdown margin.

Chapter 3 of NBSR-0018-DOC-00 provides an analysis of a demonstration core using the AFMS. The demonstration core introduces ten fresh fuel elements and twenty seventh-cycle elements. Section 3.1 of NBSR-0018-DOC-00 reviews the nuclear design characteristics of the core, including key parameters like fuel element peaking factors, projected burnup, and shutdown margins using both shim rods (with one of four withdrawn) and the regulating rod. The shutdown margin calculations assume all moveable experiments to be in their most reactive condition. Additionally, a shielding analysis of the irradiation characteristics of the fresh fuel elements is included to determine the irradiation time required for the fresh fuel elements to become self-shielding, and the depletion analyses are checked to ensure that no seventh-cycle elements exceed the TS burnup limits. The results of these analyses demonstrate that the licensing basis and TS limits on the core design are met using the AFMS. These limits are listed in section 2.1 of NBSR-0018-DOC-00 and are summarized based on the analytic results in section 3.1.6 of NBSR 0018-DOC-00.

<u>NRC Staff Review</u>

With a computational method such as MCNP, a model with increased detail is generally considered to be technically superior. The increased modeling detail allows for more precise representation of the characteristics of the core, which can lead to improved characterization of core performance in downstream thermal-hydraulic modeling. The MCNP code is very general, however, and increased detail alone does not adequately justify the model's credibility. The licensee's comparison of effective multiplication factors indicates that the 720-composition model can represent a known core design with adequate accuracy as compared to the previously approved 60-composition model. The existing SAR OFMS methods are additionally qualified using in-reactor measurements; however, the verification and validation requirements in section 2.13 of NBSR-0018-DOC-00, which are required to be performed by NBSR TS, will provide similar assurance that the as-designed core is operating consistent with the nuclear physics predictions. Based on these considerations and on the acceptable results provided for the demonstration core loading as summarized above, the NRC staff determined that the AFMS nuclear design methods are acceptable for use in core design for the NBSR.

<u>Technical Findings</u>

Based on its review that is summarized above, the NRC staff reached the following technical findings based on the applicable acceptance criteria in NUREG-1537, part 2, section 4.5:

- The reactivity analyses include the reactivity values for the core components, such as fuel elements, control rods, and reflector components, and such in-core and in-reflector components as experimental facilities. As noted above, this information was provided for a demonstration core loading in chapter 3 of NBSR-0018-DOC-00.

- The assumptions and methods used have been justified. As noted above, the licensee used the same computer code as with the previously approved license renewal, although with added detail in the core region. Additionally, the licensee compared effective multiplication factors for an OFMS core and existing TS surveillance requirements will verify that the as-loaded core is performing consistently with the as-analyzed core. The NRC staff considered this to be an adequate justification of the assumptions and methods used.

- The analyses and information in the LAR describe a reactor core system that could be designed, built, and operated without unacceptable risk to the health and safety of the public. As noted above, section 3.1.6 of NBSR-0018-DOC-00 describes the core characteristics of a demonstration core loading pattern that meet applicable TS and licensing basis limits.

4.2 THERMAL-HYDRAULIC ANALYSIS METHODS

The thermal-hydraulic analysis establishes that, under conditions of normal operation and postulated transients, the safety limit established by NBSR TS 2.1, "Safety Limit," is met, in that the fuel cladding temperature does not exceed 450 degrees Celsius (°C) for any operating conditions of power and flow. According to the associated TS Basis:

> For all reactor operating conditions that avoid either a departure from nucleate boiling (DNB), or exceeding the Critical Heat Flux (CHF), or the onset of flow instability (OFI), cladding temperatures remain substantially below the fuel blistering temperature.

The thermal-hydraulic analysis is thus performed to establish that the CHF ratio (CHFR) remains high enough to ensure that cladding heat-up due to a localized dry-out on the cladding surface is precluded from occurring. Similarly, the OFI ratio (OFIR) must remain high enough to ensure that there is no coolant flow instability in any fuel channel that could lead to a significant decrease in fuel cooling. Thus, acceptable CHFR and OFIR values confirm that the cladding temperatures remain substantially below the fuel blistering temperature, consistent with the TS 2.1 basis.

Information Provided by the Licensee

Thermal-hydraulic analysis is performed using the RELAP5/MOD3.3 reactor analysis computer code. This is largely consistent with the methods used in the previously approved license renewal, but a newer version of RELAP5 is used and the correlations used to analyze the CHF and OFI are updated. In the updated model, the Sudo-Kaminaga correlations are used to determine CHFR, and the Saha-Zuber correlation is used to determine OFIR. These correlations are implemented alongside the Mirshak correlation for CHFR and the Costa correlation for OFIR. Mirshak and Costa were the primary correlations used in the previously approved license renewal. Additionally, the updated RELAP5 model includes more detailed heat structure modeling to apply the more detailed power distribution information supplied by the 720-composition MCNP neutronics model.

The RELAP5 analysis is performed in a manner similar to that used in the previously approved license renewal, notwithstanding the updates noted above, as well as updates to nodalization, initial conditions, and component-level performance modeling that better reflect the current design of the NBSR. The core is analyzed for the accident sequences identified in table 1 of NBSR-0018-DOC-00, which represent those accidents within the NIST licensing basis that are deemed credible to occur and could have cycle-specific performance characteristics associated with changes in the power distribution such as minimum CHFR or OFIR. The maximum hypothetical and loss-of-coolant accidents are not reanalyzed, as their consequences are a function of the total core power more so than the specific power distribution.

For the demonstration core loading, the results of each accident analysis and the natural circulation at low power cooling analysis are provided in tables 6 through 9 of NBSR-0018-DOC-00, and the results of the fuel element misloading analysis are provided in tables 10 and 11. The results are provided using both the previously approved Mirshak and Costa correlations for CHFR and OFIR, respectively, and the updated Sudo-Kaminaga and Saha-Zuber correlations, respectively. The inclusion of both sets of results establishes a comparative basis demonstrating that using the updated correlations also establishes that the demonstration core loading meets the current licensing basis and TS requirements. Using the Sudo-Kaminaga correlation, the minimum CHFR of 1.27 is observed for the EOC startup accident, and using the Mirshak correlation, the minimum CHFR of 1.18 is observed for the same event, also at EOC conditions. For the OFIR, the limiting event predicted using the Saha-Zuber correlation was the loss of offsite power with a shutdown pump failure with a minimum OFIR of 1.59. The Costa correlation yielded a minimum OFIR of 2.00 for the maximum reactivity insertion event. Both results were observed for the EOC core.

The licensee also provided cladding temperature plots of the startup accident and the loss of offsite power with a shutdown pump failure. These plots, which correspond to the limiting events with respect to CHFR and OFIR margin, respectively, show that the cladding temperature never exceeds 160 °C for either event, which is well below the 450 °C safety limit for which the CHFR and OFIR are used as surrogates.

NRC Staff Review

The SAR as it was previously approved in support of license renewal describes the analysis of the OFMS using the RELAP5 reactor analysis computer code. In its LAR, the licensee noted that it is using an updated version of the same computer codes with the updates described above, namely implementation of new CHFR and OFIR correlations and updates to the system model to reflect the current reactor and coolant system design and to translate the more detailed MCNP model output to a more exact representation of the heat structures in the core. The NRC staff notes that the Sudo-Kaminaga correlations for CHFR were developed for use in plate-fuel arrays, and notes further that the Saha-Zuber correlation is discussed as being used to model OFIR in the license renewal SER and has also been used to model OFIR in other reactors with plate-type fuel. Since Sudo-Kaminaga was developed for plate-fueled reactors, and since Saha-Zuber has been both accepted for use by the staff previously in modeling the NBSR, and is used in other plate-fueled reactors, the staff determined that the use of both of these correlations for modeling thermal-hydraulic behavior at the NBSR is acceptable. Since the other model updates represent changes to better reflect the current design and operation of the NBSR, the staff determined that they too are acceptable.

The limiting results for CHFR and OFIR provided by the licensee for the demonstration core loading are discussed above. Since the licensee compares the two critical safety parameters, CHFR and OFIR, using two separate correlations for each and then shows the estimated fuel cladding temperature to confirm that there is adequate margin to the NBSR TS 2.1 safety limit, the NRC staff determined that the results obtained for the demonstration core loading thermal-hydraulic analysis are acceptable.

Technical Findings

Based on its review that is summarized above, the NRC staff reached the following technical findings based on the applicable acceptance criteria in NUREG-1537, part 2, section 4.6:

- The information in the LAR includes the thermal-hydraulic analysis methods and a demonstration analysis for the reactor. This information is summarized above.

- NIST has justified the assumptions and methods and has validated the results. The assumptions and methods reflect improvements over the SAR as it was previously approved in support of license renewal and were justified as described above. Importantly, the licensee used two sets of correlations to estimate the critical safety parameters of CHFR and OFIR and confirmed the existence of significant margin to the NBSR TS 2.1 safety limit by evaluating the predicted fuel cladding temperature for limiting transients.

- All necessary information on the primary coolant hydraulics and the thermal conditions of the fuel is specific to the NBSR. The analyses give the limiting conditions of these features that ensure fuel integrity. The limiting conditions are provided in section 2.1 and the analytic results are summarized in section 3.1.4 of NBSR-0018-DOC-00. Acceptable margins for CHFR and OFIR are demonstrated, and significant margin to the fuel cladding temperature safety limit is maintained for the limiting transients.

- The thermal-hydraulic analyses establish the continued validity of the safety limit and the LSSSs. As summarized above, the thermal-hydraulic analyses on which these parameters are based ensure that overheating during any operation or credible event will not cause loss of fuel integrity and unacceptable radiological risk to the health and safety of the public.

Technical Evaluation Conclusion

Based on the considerations discussed above, the NRC staff finds that the analytic methods described in NBSR-0018-DOC-00 are acceptable for use in analyzing AFMS that deviate from the OFMS currently described in the NBSR SAR. Therefore, the staff concludes that the proposed addition to the SAR of language that references NBSR-0018-DOC-00 is acceptable.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The regulation in 10 CFR 50.92, “Issuance of amendment,” paragraph (c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

An evaluation of the issue of no significant hazards consideration is presented below:

Per 10 CFR 50.92(c)(1): Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated? Licensee response:

> AFMS loadings deviate from the core loading scheme as described in the [SAR] 4.5.1.1.2 “Fuel Management Scheme”. This amendment introduces a new section to the updated [SAR], “4.5.1.1.3 Alternative Fuel Management Schemes (AFMS)”, which describes bounding conditions and analysis requirements for any AFMS. The amendment also introduces an engineering procedure, namely “NBSR-0018-DOC-00 NBSR Alternative Core Loading Schemes Analysis Procedure” which describes the OFMS and AFMS, a basis for the analysis, providing limitations to evaluate potential AFMS, detailed safety review for a demonstration AFMS, along with a discussion of results and conclusions to be included in subsequent [engineering change notices] dealing with similar AFMS

core loadings. The procedure provides a basis to analyze core loading so that none of the [TS] are exceeded.

Several accident scenarios and therefore consequences may be affected by AFMS core loading deviations. Particularly, all accidents shown in Table 1 are required to be reevaluated for any AFMS core loading. Other accident scenarios given in the [SAR], including “Loss of Primary Coolant” (a major rupture in the cold leg of the primary system is assumed, which leads to draining the reactor core), “Maximum Hypothetical Accident (MHA)”, “Experiment Malfunction” and “External Event” are independent of core loading changes and therefore remain unchanged. Additionally, Natural Circulation Cooling at Low Power Operation must be analyzed for each AFMS to show compliance with [TS] 2.2. Natural Circulation Cooling at Low Power Operation is not an accident scenario but an analysis to show natural circulation at low power operations.

Note that all of the accident scenarios and Natural Circulation Cooling at Low Power Operation conditions are analyzed using the RELAP5 model as described in the “NBSR-0018-DOC-00 NBSR Alternative Core Loading Schemes Analysis Procedure”. The misloading accident is the only one that will require unique power distributions from corresponding MCNP simulations with the misloaded fuel configuration. Some scenario conditions are updated based on facility changes and available new information. “NBSR-0018-DOC-00 Appendix C” provides descriptions and modifications for accident scenarios.

	Accident Sequence	**Section in the SAR**
#1	Startup Accident	13.1.2.2.2.1
#2	Maximum Reactivity Insertion Accident	13.2.2
#3	Loss of Offsite Power	13.1.4.1
#4	Loss of Offsite Power with Shutdown Pump failure	13.1.4.5
#5	Seizure of One Primary Pump	13.1.4.2
#6	Throttling of Coolant Flow to the Outer Plenum	13.1.4.4
#7	Throttling of Coolant Flow to the Inner Plenum	13.1.4.3
#8	Misloading of Fuel	13.1.5

Table 1. The accident sequences to be re-analyzed, and their references in the SAR.

Based on detailed analysis provided in the technical report, and because these AFMS accident scenarios are specifically analyzed for probability and consequences, there are, by definition, no changes in the probability of occurrences or the consequences of previously analyzed accidents. Therefore, the proposed [SAR] amendment allowing analysis of AFMS does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Per 10 CFR 50.92(c)(2): Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated? Licensee response:

The requested amendment to the facility license involves a SAR change to describe engineering analysis procedures for any [AFMS] wherein which the NBSR core is loaded with a different core loading pattern than as described in the updated [SAR]. An AFMS is any core loading pattern that deviates from the [OFMS] in a manner such that the number of the specific type of fuel elements, such as fresh, or used is different than usual and/or their locations in the core are

> modified. As there are no other changes besides that of fuel loading, changes in the core loading pattern do not initiate a different kind of accident. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Per 10 CFR 50.92(c)(3): Does the proposed amendment involve a significant reduction in a margin of safety? Licensee response:

> The requested amendment to the facility license involves a SAR change to describe engineering analysis procedures for any [AFMS] wherein which the NBSR core is loaded with a different core loading pattern than as described in the updated [SAR]. An AFMS is any core loading pattern that deviates from the [OFMS] in a manner such that the number of the specific type of fuel elements, such as fresh, or used is different than usual and/or their locations in the core are modified. The AFMS can be deemed acceptable as long as the proposed AFMS is analyzed according to the “NBSR-0018-DOC-00” and found to be within the updated [SAR], [TS] limitations and boundary conditions listed therein. The boundary conditions are based on the [TS] and updated [SAR] requirements. Because these alternate fuel management schemes are specifically analyzed for a reduction in margin of safety, there is, by definition, no significant reduction in margin of safety. The proposed amendment contains no changes in the [TS] or other safety limitations as described in the updated [SAR]. Therefore, the proposed amendment of the SAR in allowing this operation does not involve a significant reduction in a margin of safety.

Based on its review of the above evaluation, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

6.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Maryland State official was notified of the proposed issuance of the amendment on February 24, 2023. The State official did not provide any comments (ML23058A060).

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR part 20, “Standards for Protection against Radiation.” The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant changes in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, as published in the FR on February 10, 2023 (88 FR 8918), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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