Safety Evaluation Report

Renewal of the Facility Operating License for the GE-Hitachi Nuclear Test Reactor

License No. R-33
Docket No. 50-073

United States Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 2023

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation. The NRC staff conducted this review in response to a timely application filed by GE-Hitachi Nuclear Energy Americas LLC (GEH, the licensee) for a 20-year renewal of Facility Operating License No. R-33 to continue to operate the Nuclear Test Reactor (NTR, the facility). In its safety review, the NRC staff considered information submitted by the licensee, past operating history recorded in the licensee's annual reports submitted to the NRC, and inspection reports prepared by the NRC staff. Based on its review, the NRC staff concludes that GEH can continue to operate the NTR for the 20-year term of the renewed facility license, in accordance with the license, without endangering public health and safety, NTR staff, or the environment.

CONTENTS

LIST OF TABLES

LIST OF FIGURES

ABBREVIATIONS AND ACRONYMS

$	dollar of reactivity
%Δk/k	reactivity in percent
10 CFR	Title 10 of the *Code of Federal Regulations*
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954, as amended
AEC	U.S. Atomic Energy Commission
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar	argon
ARM	area radiation monitor
C	Celsius
cm	centimeter
CEDE	committed effective dose equivalent
Ci	curie
Ci/yr	curies per year
cm/s	centimeters per second
CY	calendar year
DAC	derived air concentration
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
ER	environmental report
F	Fahrenheit
ft	feet
FY	fiscal year
GE	General Electric
GEH	GE-Hitachi Nuclear Energy Americas LLC
HEU	highly enriched uranium
in	inch
IR	inspection report
ISG	interim staff guidance
KAPL	Knolls Atomic Power Laboratory
kW	kilowatt
kW(t)	kilowatt thermal
LA	license amendment
LC	license condition

LCO	limiting condition for operation
LEU	low-enriched uranium
l	liter
lb	pound
LOCA	loss-of-coolant accident
LRA	license renewal application
LSSS	limiting safety system setting
m	meter
μCi/ml	microcuries per milliliter
μsec	microsecond
μmhos/cm	micromhos per centimeter
MHA	maximum hypothetical accident
mg	milligram
mhos/cm	mhos per centimeter
mm	millimeter
mrem	milli-roentgen equivalent man (millirem)
mrem/hr	millirem per hour
MPS	manual poison sheet
MW	megawatt
MWD	megawatt-days
MW(t)	megawatt thermal
n/cm^2-s	neutrons per centimeter squared second
N-16	nitrogen-16
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NTR	Nuclear Test Reactor
PDR	public document room
pH	potential hydrogen
PSP	physical security plan
RAI	request for additional information
REP	radiological emergency plan
RO	reactor operator
RSO	radiation safety officer
RTR	research and test reactor
SAR	safety analysis report
SE	safety evaluation
SER	safety evaluation report
SL	safety limit
SNM	special nuclear material
SOI	statement of intent
SRM	staff requirements memorandum
SRO	senior reactor operator
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TS	technical specification

TTR	thermal test reactor
U	uranium
U-Alx	uranium/aluminum
VNC	Vallecitos Nuclear Center
W	watt
wt%	weight percent

1 INTRODUCTION

1.1 Overview

By letter dated November 19, 2020, GE-Hitachi Nuclear Energy Americas LLC (GEH, the licensee) submitted a license renewal application (LRA) to the U.S. Nuclear Regulatory Commission (NRC, the Commission) for a 20-year renewal of the Class 104c Facility Operating License No. R-33, Docket No. 50-073, for the GEH Nuclear Test Reactor (NTR, the facility), which is owned by GEH and located at its Vallecitos Nuclear Center (VNC) in Alameda County, California. The NTR is a heterogeneous, highly enriched uranium (HEU), graphite-moderated and -reflected, light-water cooled reactor, licensed to operate at steady-state power levels not in excess of 100 kilowatts thermal (kW(t)).

The GEH LRA for the NTR includes a cover letter (Reference (Ref.) 1), a safety analysis report (SAR) (Ref. 2 – chapters 1 through 8; Ref. 3 – chapters 9 through 16), an environmental report (ER) (Ref. 4), and proposed technical specifications (TSs) (Ref. 5). The GEH LRA cover letter indicates that updated versions of the licensee’s operator requalification program, radiological emergency plan (REP), and physical security plan (PSP) were submitted to the NRC prior to the LRA.

As described in SAR chapter 1, “The Facility,” the NTR is a multidisciplinary research and training facility providing a broad range of analytic, radiographic, and irradiation services. The NTR is part of the VNC site, which is situated on the north side of Vallecitos Valley in Southern Alameda County located approximately 35 miles east-southeast of San Francisco and 20 miles north of San Jose, California and within five miles of Livermore and Pleasanton, California.

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.51(a) states, in part, that “[e]ach license will be issued for a fixed period of time to be specified in the license but in no case to exceed 40 years from [the] date of issuance.” On October 24, 1957, the U.S. Atomic Energy Commission (AEC) issued Construction Permit No. CPRR-19 to General Electric (GE). This permit authorized GE to construct the NTR at its VNC site in Southern Alameda County, California. On October 31, 1957, the AEC issued Facility Operating License No. R-33, authorizing GE to operate the NTR at steady-state power levels up to 30 kW(t). The NTR first reached criticality on November 15, 1957. On July 22, 1969, the facility operating license was amended authorizing GE to operate the reactor at steady-state power levels not in excess of 100 kW(t). The facility operating license was renewed by License Amendment (LA) No. 18, issued on December 28, 1984 (Ref. 6), and again by LA No. 21, issued on April 20, 2001 (Ref. 7).

Because GEH submitted the LRA, by letter dated November 19, 2020, to the NRC greater than 30 days before the expiration of the facility operating license, which was April 20, 2021, the timely renewal provision provided in 10 CFR 2.109, “Effect of timely renewal application,” paragraph (a) authorizes the licensee to continue operating the NTR under the terms and conditions of the current license until the NRC staff completes its action on the LRA. A renewal would authorize continued operation of the NTR for an additional 20 years from the date of issuance of a renewed license.

A Notice of Opportunity to Request a Hearing was published in the *Federal Register* on January 10, 2023 (88 FR 1433) (Ref. 8). No requests for a hearing were received.

The NRC staff based its review of the LRA on the information contained in the LRA and supplemental information provided during the acceptance review, as well as in supporting supplements provided in response to the NRC staff's regulatory audits and requests for additional information (RAI). The NRC staff conducted two regulatory audits: one focusing on the reactor description, radiation protection, and accident analysis beginning on July 28, 2021, and one focusing on the proposed TSs beginning on August 15, 2022.

The NRC staff requested supplemental information in its acceptance review by letter dated March 11, 2021 (Ref. 17). The licensee provided its responses by letters dated April 22, 2022 (Ref. 18), and September 15, 2022 (Ref. 19). The NRC audit plans were provided to the licensee by letters dated July 26, 2021 (Ref. 9), and August 10, 2022 (Ref. 10), and the audit summary reports were provided by letters dated January 23, 2023 (Ref. 11), and April 13 (Ref. 12), respectively. GEH provided supplemental information following the audits by letters dated September 22, 2021 (Ref. 13); and January 27 (Ref. 36), March 24 (Ref. 14), April 21 (Ref. 38), April 27 (Ref. 41), and June 15, 2023 (Ref. 42). The NRC staff also issued an RAI by letter dated November 2, 2021 (Ref. 15), and the licensee provide an updated operator requalification program by letter dated December 3, 2021 (Ref. 16).

The NRC staff also reviewed the licensee's PSP, REP, and operator requalification program to ensure that they are consistent with NRC regulations and guidance. The results of the NRC staff review of the PSP, REP, and operator requalification program are discussed below. The NRC staff's review of the LRA also included information from the NTR annual operating reports from calendar year (CY) 2014 through CY 2021 (Ref. 20) and NRC staff inspection reports (IRs) from CY 2014 through CY 2022.

During its review, the NRC staff noted that few aspects of the NTR had changed since the issuance of the previous license renewal by LA No. 21, issued on April 20, 2001 (Ref. 7). Therefore, the NRC staff's review included an assessment of the current facility design and any supporting design basis analyses to ascertain if they remained unchanged since the issuance of LA No. 21. When the NRC staff's evaluation determined that no changes had occurred affecting the facility design or any supporting design basis analyses, the NRC staff states that the previous NRC safety evaluation (SE) issued with LA No. 21 remains effective to ensure adequate safety for the operation of the facility.

With the exception of the PSP and portions of the LRA, as supplemented, that contain security-related, proprietary, and/or copyrighted information, the material pertaining to this review may be examined or copied, for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The NRC also maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. Publicly available documents related to this review may be accessed through the NRC's Public Library on the internet at https://www.nrc.gov. If you do not have access to ADAMS or if you experience problems accessing the documents in ADAMS, contact the NRC PDR staff by telephone at 1-800-397-4209 or 301-415-4737, or by email to PDR.resource@nrc.gov. The entire PSP is protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements." Redacted versions of other documents that contain security-related and/or proprietary information are publicly available in ADAMS.

Section 7, "References," of this safety evaluation report (SER) contains the dates and associated ADAMS accession numbers of the LRA, as supplemented, and documents used by the NRC staff to complete its review.

In conducting its review, the NRC staff evaluated the facility against the requirements of the regulations, including 10 CFR Part 20, "Standards for Protection Against Radiation," 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," 10 CFR Part 40, "Domestic Licensing of Source Material," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," and 10 CFR Part 73, "Physical Protection of Plants and Materials," and the guidance in applicable NRC regulatory guides and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS)-15 series. The NRC staff also used the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 22). Because there are no specific accident-related regulations for research reactors, the NRC staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20.

In SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," dated October 24, 2008 (Ref. 23), the NRC staff provided the Commission with information regarding plans to streamline the review of LRAs for research and test reactors (RTRs). The Commission subsequently issued its staff requirements memorandum (SRM)-SECY-08-0161, dated March 26, 2009 (Ref. 24). The SRM directed the NRC staff to streamline the renewal process for RTRs, using some combination of the options presented in SECY-08-0161. The SRM also directs the NRC staff to implement a graded approach with a review scope commensurate with the risk posed by each facility. The graded approach incorporates elements of the alternative safety review approach discussed in Enclosure 1 to SECY-08-0161. In the alternative safety review approach, the NRC staff considers the results of past NRC staff reviews when determining the scope of the review. A basic requirement, as contained in the SRM, is that licensees must be in compliance with applicable regulatory requirements.

The NRC staff developed interim staff guidance (ISG)-2009-001, "Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors" (Ref. 25), to assist in the review of LRAs. The streamlined review process is a graded approach based on licensed power level. Under this process, facilities are divided into two tiers. Facilities with a licensed thermal power level of 2 megawatts (MW(t)) and greater, or requesting a power level increase, undergo a full review using NUREG-1537. Facilities with a licensed thermal power level less than 2 MW(t) undergo a focused review that centers on the most safety-significant aspects of the LRA and relies on past NRC reviews for certain safety findings. The NRC issued a draft of the ISG for public comment and considered public comments in its development of the final ISG. The NRC staff conducted the NTR LRA review using the guidance in the final ISG. Since the licensed thermal power level for the NTR is less than 2 MW(t), and since GEH is not requesting a power level increase for the NTR, the NRC staff performed a focused review of the LRA. Specifically, the review focused on reactor design and operation, accident analysis, TSs, radiation protection, waste management programs, financial requirements, environmental assessment, and changes to the facility made during NRC review of the LRA.

The licensee is required to maintain a program to provide for the physical protection of the facility and its special nuclear material (SNM) in accordance with the requirements of 10 CFR Part 73. Changes to the PSP can be made, by the licensee, in accordance with 10 CFR 50.54, "Conditions of licenses," paragraph (p), as long as those changes do not decrease the effectiveness of the plan. By letter dated April 4, 2022 (Ref. 26), as part of the license renewal, the licensee provided an updated version of the VNC site PSP. By letter dated

May 26, 2022 (Ref. 27), the NRC staff determined that the VNC site PSP is in compliance with the applicable regulations contained in 10 CFR Part 73, referenced in NRC Regulatory Guide (RG) 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance." In addition, the NRC staff performs routine inspections of the licensee's compliance with the requirements of the VNC site PSP. The NRC staff's review of GEH VNC NTR IRs (Ref. 21) for the past several years identified no violations.

The licensee is required to maintain an emergency plan, in compliance with 10 CFR 50.54(q), "Emergency plans," and Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," which provides reasonable assurance that the licensee will continue to be prepared to assess and respond to emergency events. By letter dated February 12, 2021 (Ref. 28), the NRC issued LA No. 24, which approved changes to the GEH VNC NTR REP, Revision 1, dated June 9, 2021 (Ref. 29). The NRC staff review of LA No. 24 included a comprehensive review of the GEH VNC NTR REP, which was submitted for LA by letter dated July 17, 2019, a year prior to the submittal of the LRA (by letter dated November 19, 2020). The NRC staff's review of GEH VNC NTR IRs (Ref. 21) for the past several years identified no violations of its REP. The licensee maintains the REP in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, which provides reasonable assurance that the licensee will be prepared to assess and respond to emergency events.

As part of the LRA review, the NRC staff reviewed the NTR "Requalification Program for the General Electric Nuclear Test Reactor," dated May 2021, submitted to the NRC by letter dated June 21, 2021 (Ref. 30). The NRC staff issued an RAI by letter dated November 2, 2021 (Ref. 15), and the licensee provided its response by letter dated December 3, 2021 (Ref. 16). By letter dated February 8, 2022 (Ref. 31), the NRC staff issued its letter finding the revised operator requalification program to be in accordance with the applicable regulations contained in 10 CFR Part 55, "Operators' Licenses," and consistent with the guidance contained in ANSI/ANS-15.4-2016, "Selection and Training of Personnel for Research Reactors."

The NRC staff also evaluated the environmental impacts of the renewal of the license for NTR in accordance with 10 CFR Part 51. The NRC staff published an Environmental Assessment and Finding of No Significant Impact in the *Federal Register* on March 22, 2023 (88 FR 17274), with a minor correction published on June 1, 2023 (88 FR 35933), which concluded that renewal of the NTR facility operating license will not have a significant effect on the quality of the human environment.

The purpose of this SER is to summarize the findings resulting from the safety review of the LRA for the NTR and to delineate the technical details that the NRC staff considered in evaluating the radiological safety aspects of continued operation. This SER provides the basis for renewing the license for operation of the NTR up to a steady-state thermal power level of 100 kW(t).

This SER was prepared by Duane Hardesty, Senior Project Manager, and Geoffrey Wertz, Project Manager, from the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Advanced Reactors and Non-Power Production and Utilization Facilities, Non-Power Production and Utilization Facility Licensing Branch; Elijah Dickson, Senior Reliability and Risk Analyst, and Zachary Gran, Health Physicist, from the NRC's NRR, Division of Risk Assessment, Radiation Protection and Consequence Branch; Diana Woodyatt, Chief, and Adam Rau, General Engineer, from the NRC's NRR, Division of Safety Systems, Nuclear Systems Performance Branch; and Emil Tabakov, Financial Analyst from the NRC's Office of Nuclear Material Safety and Safeguards, Division of Rulemaking, Environmental and Financial Support, Financial Assessment Branch.

1.2 Summary and Conclusions on Principal Safety Considerations

The NRC staff's evaluation of the LRA considered the information submitted by the licensee, including past operating history recorded in the licensee's annual reports to the NRC, as well as IRs prepared by the NRC staff. On the basis of this evaluation and resolution of the principal issues reviewed for the NTR, the NRC staff concludes the following:

- The design and use of the reactor structures, systems, and components important to safety during normal operation discussed in SAR chapter 3, "Design of Structures, Systems and Components," in accordance with the TSs, are safe, and safe operation can reasonably be expected to continue.

- The facility will continue to be useful in the conduct of nuclear research activities, as described in SAR section 1.6, "Summary of Operations."

- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA), emphasizing those that could lead to a loss of integrity of a fueled experiment encapsulation, and a release of fission products.

- The licensee performed analyses using conservative assumptions of the most serious credible accidents and determined that the calculated potential radiation doses for the facility staff and members of the public would not exceed 10 CFR Part 20 doses for unrestricted areas.

- The licensee's management organization, conduct of training, and research activities, in accordance with the TSs, are adequate to ensure safe operation of the facility.

- The systems provided for the control of radiological effluents, when operated in accordance with the TSs, are adequate to ensure that releases of radioactive materials from the facility are within the limits of the Commission's regulations and are as low as reasonably achievable (ALARA).

- The TSs, which provide limits controlling the operation of the facility, are such that there is reasonable assurance that the facility will be operated safely and reliably. No significant degradation of the reactor has occurred, as discussed in SAR chapter 4, "Reactor Description," and the TSs will continue to help ensure that no significant degradation of safety-related equipment will occur.

- The licensee has reasonable access to sufficient resources to cover operating costs and eventually to decommission the facility.

- The licensee maintains a PSP for the facility and its SNM, in accordance with the requirements of 10 CFR Part 73, which provides reasonable assurance that the licensee will continue to provide for the physical protection of the facility and its SNM.

- The licensee maintains an emergency plan in compliance with 10 CFR 50.54(q) and Appendix E to 10 CFR Part 50, which provides reasonable assurance that the licensee will continue to be prepared to assess and respond to emergency events.

- The licensee's procedures for training its licensed operators and the operator requalification plan provide reasonable assurance that the licensee will continue to have qualified staff that can safely operate the facility.

On the basis of these findings, the NRC staff concludes that there is reasonable assurance that GEH can continue to operate the NTR in accordance with the Atomic Energy Act of 1954, as amended (AEA), NRC regulations, and the renewed facility operating license without endangering public health and safety, facility staff, or the environment. The NRC staff further concludes that the issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Description of the Facility

SAR section 1.1, "Introduction," provides a description of the NTR, which was designed and constructed by GE as an experimental physics tool to advance its nuclear energy programs. GEH operates the NTR for neutron radiography of radioactive and nonradioactive objects, small sample irradiation and activation, sensitive reactivity measurements, personnel training, and calibrations and other tests utilizing its neutron flux.

The AEC issued GE a construction permit for the NTR on October 24, 1957, and an initial facility operating license on October 31, 1957. Renewals to Facility Operating License No. R-33 were issued by LA No. 18 on December 28, 1984, and by LA No. 21 on April 20, 2001.

SAR section 1.3, "General Description of the Facility," states that the NTR is located at the VNC, which is largely undeveloped grasslands within the Livermore area. VNC is situated on the north side of Vallecitos Valley in Southern Alameda County within five miles of Livermore and Pleasanton and approximately 35 miles east-southeast of San Francisco and 20 miles north of San Jose.

The NTR is a heterogeneous, HEU, graphite-moderated and -reflected, light-water-cooled, thermal reactor, licensed to operate at power levels not in excess of 100 kW(t). The fuel consists of HEU-aluminum alloy disks, clad with aluminum. The core is cooled either by natural or forced circulation of deionized light-water circulated in a primary system constructed primarily of aluminum. The reactor operates at low temperature and heat flux. Reactivity is controlled by up to six manually positioned cadmium sheets, four boron-carbide-filled safety rods (spring-actuated for reactor scram), and three electric motor-driven boron-carbide-filled control rods. Conventional instrumentation is provided to indicate, record, and control important variables and shut down the reactor automatically if assigned operating limits are exceeded. The reactor's irradiation facilities include a central sample tube, penetrations through and into the reflector, the reflector faces, and the beams from any of these facilities.

SAR section 1.3 also states that the reactor is located within a thick-walled concrete cell that, along with the control room, north room, setup room, and the south cell, comprises the NTR facility. Principal equipment in the concrete reactor cell includes the reactor, the reactor control mechanisms, the coolant system, and a fuel loading tank that provides radiation shielding and the primary water system reservoir. The control room contains the control console and provides space for experiment equipment, preparation, and an operator work area. The south cell is a concrete-shielded room that provides access to the thermal column, the horizontal facility, and the horizontal facility south beam. The north room provides space for performing experiments utilizing the horizontal facility north beam and the cable held retractable irradiation system. The setup room is used for storage and setup of experiments involving irradiation or testing. There is a wall penetration into the south cell for long trays to utilize the horizontal facility south beam.

1.4 Shared Facilities and Equipment

SAR section 1.4, "Shared Facilities and Equipment," states that the NTR facility shares many facilities and equipment in Building 105 with other laboratory facilities. These include potable water supply, fire protection, emergency supplies and support, heating, ventilation, and air-conditioning (HVAC) system, electrical distribution, compressed air system, and the occupied spaces of Building 105. The NTR shared building spaces are separated by walls to delineate the NTR facility from the other offices and laboratories. Other means of separation have been installed to adequately isolate the shared facilities and equipment, such as the potable water supply to the NTR, which contains an approved reduced pressure backflow preventer. Although there are shared load centers, reactor safety equipment is connected to electrical circuits that are not shared with other facilities and equipment outside of the NTR. Other shared facilities and equipment have been established at the NTR to increase the convenience and the capability of resources available to the facility. These include the fire protection system (building sprinkler system, fire hoses, and portable fire extinguishers), building emergency response teams, an emergency supply cabinet, HVAC system, and a compressed air system, none of which support reactor safety systems.

The NRC staff previously reviewed the licensee's shared facilities, as document in its SE for LA No. 21 for the previous license renewal. The NRC staff stated that the NTR shares Building Number 105 at the VNC with other laboratories. It also shares many facilities and equipment, including potable water, fire protection, emergency supplies and support, HVAC systems, electrical distribution, and compressed air. Walls to delineate the NTR facility from other laboratories separate the shared building spaces. Other separations are installed to isolate the shared facilities and equipment.

The NRC staff reviewed the shared facilities and equipment and determined that the licensee provided a complete listing in the SAR. The NRC staff also determined that a malfunction or a loss of function of these shared facilities would not affect the operation of the NTR, nor would it damage the NTR or affect its capability to be safely shut down. Additionally, a loss of function of the shared facilities would not create the potential or result in an uncontrolled release of radioactive material from the licensed facility to unrestricted areas.

1.5 Comparison with Similar Facilities

SAR section 1.5, "Comparison with Similar Facilities," states that the design of the NTR resulted from the evolution of a series of reactors designed by scientists at the GE Knolls Atomic Power Laboratory (KAPL) in Schenectady, New York. The earlier reactors were known as thermal test reactors (TTRs). Three models were built and operated successfully. The GE TTR operated from 1954 to the mid-1980s at KAPL. The TTR No. 2 operated from 1955 until 1972 at the Battelle Memorial Institute Pacific Northwest Laboratory. The third TTR, the Savannah River National Laboratory Standard Pile, operated from 1953 to 1979.

The GEH NTR includes the following features incorporated into its design from the previous facilities:

- Negative void coefficient of reactivity.
- Small positive coolant temperature coefficient of reactivity which becomes negative at a water temperature slightly above the operating temperature.
- A control system extremely sensitive to changes in reactivity so that minute changes are detectable.
- Safety and control functions that are separate, except for an interlock that requires all safety rods to be fully withdrawn prior to withdrawing any control rod. This ensures that negative reactivity is available if needed for scram before a control rod can be moved.
- Manually positioned cadmium sheets that can be used to limit reactivity controllable from the console and to provide enough negative reactivity to preclude any possible danger or criticality during fuel loading.
- An instrumentation system that includes fail-safe and redundant features as well as proven reliable components.
- A system constructed from materials having properties compatible with their intended service.

Safety measures that have been incorporated into the operation of the facility include:

- Very low heat flux, even at the maximum operating power.
- Temperatures and pressures only a little above ambient.
- Low operating power, resulting in a low fission product inventory.
- Rigid control by operations management of all experiments performed in the reactor facility.
- Performance of all activities that can affect nuclear safety under the direction of an NRC-licensed reactor operator or NRC-licensed senior reactor operator, as required.

The NRC staff finds that the design of the NTR follows from the improvements incorporated from the predecessor reactors, described above. The NRC staff also finds that the NTR operating history has demonstrated consistent safe operation, which should be expected to continue with operation in accordance with its TSs.

1.6 Summary of Operations

SAR section 1.6 states that the NTR was originally built as an experimental tool for diverse applications. In the first 5 years of operation, it was used for pile-oscillator measurements of nuclear cross sections of materials, calibrations of foils and nuclear sensors, neutron activation analysis, studies of radiation damage in semiconductors, nuclear fuel enrichment measurements, and cryo-nuclear investigations. Over the years the reactor has been used for a variety of purposes from neutron absorption measurements of material at a reactor power level of 10 watts to 24 hours per day irradiation of filter tape. More recently, the NTR has been used for sensitivity reactivity measurements, training, and calibrations utilizing a neutron flux. Currently, the NTR is used for neutron radiography of radioactive and nonradioactive objects, and small sample irradiations. The NTR can operate at extremely low power levels not in excess of 100 kW(t) and has operated in recent years at a nominal 800 annual effective full-power hours.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research reactor, that the applicant shall have reached an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel.

SAR section 1.7, “Compliance with the Nuclear Waste Policy Act,” Figure 1-3, “DOE Spent Nuclear Fuel Disposal Agreement,” provides a copy of a letter dated July 13, 1983, from Thomas S. Keefe of the DOE, informing R. W. Darmitzel of General Electric Company that Contract Number DE-CR01-83NE44426 had been executed (Ref. 2). The NRC staff finds that by entering into such a contract with the DOE, the licensee has satisfied the requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility History and Modifications

SAR section 1.8, “Facility Modifications and History,” states that the NTR was constructed under Construction Permit No. CPRR-19, issued October 24, 1957, as requested by General Electric Company’s application dated June 5, 1957. Operation of the reactor up to 30 kW(t) was authorized by Facility Operating License No. R-33, issued on October 31, 1957. Initial loading of the reactor began on November 7, 1957, and criticality was first achieved on November 15, 1957.

The NTR was operated at power levels up to 30 kW(t) for more than 5,000 hours until, on July 22, 1969, the facility operating license was amended and revised in its entirety to authorize operation of the reactor at power levels of up to a maximum of 100 kW(t) steady-state power (later amended to a power level not in excess of 100 kW(t)). Since then, the NTR has operated under Facility Operating License No. R-33, as amended, at power levels up to 100 kW(t) for more than 45,000 hours while performing a wide variety of experiments.

In 1976, the reactor core developed a leak in a weld area, necessitating replacement. The reactor fuel was removed and inspected, and a major portion of the reactor was dismantled. The core can was replaced, as well as some of the graphite in the central area. Some modification of the irradiation facilities occurred at this time. The reactor was reassembled, utilizing the original fuel, and routine operation resumed.

Prior to 1985, many original instruments were replaced, including the pico-ammeters (linear wide range neutron monitors), log N (log wide range neutron monitor), remote area gamma monitors, and the stack effluent gas and particulate monitors.

Facility Licensing Actions

Renewals to Facility Operating License No. R-33 were issued by LA No. 18 on December 28, 1984, and by LA No. 21 on April 20, 2001.

The NRC staff issued LA No. 23 by letter dated October 22, 2007, to reflect a change in ownership of the facility from General Electric Company to GEH (Ref. 32).

The NRC staff issued LA No. 24 by letter dated February 8, 2022, to authorize a revision to the GEH NTR REP (Ref. 28).

The NRC staff issued LA No. 25 by letter dated February 12, 2021, to authorize the unconditional release, pursuant to 10 CFR 50.83, "Release of part of a power reactor facility or site for unrestricted use," of approximately 610 acres of the VNC property (Ref. 33).

Facility Modifications

In 2015, the shop on the south side of the building across from the Control Room was converted into NTR office space. The NTR Setup Room was expanded to enclose the loading dock placing the wall access penetration to the south neutron ray position in the Setup Room.

In March 2020, the entire radiation monitoring system was replaced with a digital system to increase reliability and eliminate installed check sources.

The NRC staff finds that these modifications were subjected to screening and evaluation under 10 CFR 50.59, "Changes, tests, and experiments," to ensure that there was no adverse impact on the safety of the NTR. The NRC staff reviewed NTR annual reports from CY 2014 through CY 2021 (Ref. 20) and NRC IRs from CY 2014 through CY 2022 (Ref. 21) and noted that there were no discrepancies with the facility as described in the SAR. Therefore, the NRC staff concludes that the changes performed under 10 CFR 50.59 appear to be reasonable.

1.9 Financial Considerations

1.9.1 Financial Ability to Operate the Facility

The financial requirements for non-electric utility nuclear reactor licensees are in 10 CFR 50.33, "Contents of applications; general information," paragraph (f):

> Except for an electric utility applicant for a license to operate a utilization facility of the type described in [10 CFR] 50.21(b) or [10 CFR] 50.22, [an application shall state] information sufficient to demonstrate to the Commission the financial qualification of the applicant to carry out, in accordance with regulations of this chapter, the activities for which the permit or license is sought…. Applicants to renew or extend the term of an operating license for a nonpower reactor shall include the financial information that is required in an application for an initial license.

GEH does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions"; therefore, the LRA must include the financial information that is required in an application for an initial license. Accordingly, GEH must provide information to demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for the period of the license. GEH must submit estimates of the total annual operating costs for each of the first 5 years of facility operations from the expected license renewal date and indicate the source(s) of funds to cover those costs. This is consistent with the guidance in NUREG-1537, Part 2, as it pertains to the NRC staff's review of the licensee's financial qualifications.

By letter dated September 15, 2022, GEH supplemented its financial information, including its projected operating costs for the NTR, for each of the fiscal years (fYs) 2021 through 2025 (Ref. 19). According to GEH, NTR expenses are categorized by salaries and operating expenses to conduct contractual neutron radiographic services. The funding for NTR operations is sourced from the GE corporation. The operating revenues from the NTR are generated from contractual services and are part of the overall GE earnings as presented in the U.S. Securities and Exchange Commission GE Company Annual 10-K Filing for 2021. As part of its review, the NRC staff considered guidance in NUREG-1537, Part 2, as well as the projected operating costs and associated funding for similar research reactor facilities. The NRC staff review found the NTR's operating cost estimates, compared to other typical research reactors, and sources of funds to be reasonable.

The NTR is currently licensed under section 104c of the AEA as a facility that is useful in the conduct of research and development activities. Pursuant to 10 CFR 50.21, "Class 104 licenses; for medical therapy and research and development facilities," paragraph (c) and 10 CFR 50.22, "Class 103 licenses; for commercial and industrial facilities," if a facility is to be licensed under section 104c as a non-commercial, non-power reactor facility that is useful in the conduct of research and development activities, then the facility is to be used so that not more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training. Additionally, if a facility is to be licensed under section 104c, then the licensee shall recover not more than 75 percent of the annual cost to the licensee of owning and operating the facility through sales of nonenergy services, energy, or both, other than research and development or education and training, of which not more than 50 percent may be through sales of energy. The NTR facility was originally licensed in the late 1950s. The reactor was built to support commercial neutron radiography, irradiations of specimens, reactivity worth characterizations of reactor fuel cladding material, and training. The NTR provides irradiation services for various private and public researchers, U.S. national laboratories, U.S. Military, and U.S. private industry.

The licensee provided supplemental information that confirmed that the annual cost of conducting the commercial activities at the NTR is less than 50 percent of the annual cost of owning and operating the NTR. The research and development that NTR performs is in the form of neutron radiography experiments for the U.S. Department of Defense. Additionally, the supplemental information also demonstrates that 75 percent or less of the annual costs of owning and operating the NTR are recovered through sales of nonenergy services, energy, or both, other than research and development or education and training, and none of the annual costs of owning and operating the NTR are received from the sale of energy. Because the licensee confirmed in the LRA, as supplemented, that the NTR is used so that it meets the statutory requirements in section 104c of the AEA and the regulatory requirements in 10 CFR 50.21(c), the NRC staff concludes that the renewed license can be issued pursuant to section 104c of the AEA.

Based on the above, the NRC staff finds that the licensee has provided the appropriate information for the NTR operating costs and has also demonstrated reasonable assurance for obtaining the necessary funds to cover these costs for the period of the renewed facility operating license and that the renewed facility operating license can be issued pursuant to section 104c of the AEA. Accordingly, the NRC staff concludes that the licensee meets the financial qualifications requirements in 10 CFR 50.33(f) and is consistent with the guidance in NUREG-1537, Part 2.

1.9.2 Financial Ability to Decommission the Facility

According to 10 CFR 50.33(k)(1), an application for an operating license for a utilization facility must contain information that demonstrates how reasonable assurance will be provided that funds will be available to decommission the facility. Under 10 CFR 50.75, "Reporting and recordkeeping for decommissioning planning," paragraph (d), each non-power reactor applicant for or holder of an operating license shall submit a decommissioning report that contains a cost estimate for decommissioning the facility, an indication of the funding method(s) to be used to provide funding assurance for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing financial assurance for decommissioning are described in 10 CFR 50.75(e)(1).

The NRC staff applied the guidance in NUREG-1537, Part 2 to complete its review of the LRA as it pertains to financial assurance for decommissioning. The NTR decommissioning cost estimate was developed using the methodology of NUREG/CR-1756, "Technology, Safety and Costs of Decommissioning Reference Research and Test Reactors," for a reference research reactor (Ref. 34).

By letter dated April 27, 2023, the licensee provided decommissioning cost information for the NTR (Ref. 41). The cost estimate includes itemized costs for planning/preparation, decontamination/dismantlement, release survey, waste packing/shipping, waste disposal, equipment/supplies, laboratory costs, utilities/shared services, NRC inspection, and a 25-percent contingency. The licensee indicated that the cost estimate is reviewed, and if necessary, adjusted annually. The licensee indicated that this review considers the status of any ongoing or planned facility modifications; operational events that may impact future decommissioning; changes in regulatory requirements and industry guidance; etc. In addition, annual adjustments for inflation are made using site-specific long-term inflation rates related to nuclear facility retirement costs of between 3 and 4 percent. At approximately every five years, a more detailed review is performed, and the cost estimate updated, as necessary. This review validates assumptions used to prepare the estimate including labor rates, labor categories, waste volumes, waste categories, analytical costs, waste disposal options, waste disposal rates, transportation and packaging costs, utility costs, taxes, insurance costs, etc. The current cost estimate as of December 3, 2022, is approximately $6.5 million. Based on the NRC staff's review of the LRA using guidance in NUREG-1537, Part 2 and NUREG/CR-1756, the NRC staff concludes that the decommissioning approach and decommissioning cost estimate submitted for the NTR are reasonable.

By letter dated April 27, 2023 (Ref. 41), GEH stated that it has elected to use a surety bond to provide decommissioning financial assurance, which is allowed by 10 CFR 50.75(e)(1)(iii). GEH indicated that a copy of the surety bond was provided to the NRC by letter dated March 27, 2018 (ML18087A172). A supplemental rider to increase the surety bond amount to account for inflation was provided to the NRC by letter dated March 31, 2023 (ML23104A417). The current surety bond is in the amount of $6,505,919.

The NRC staff reviewed GEH's information on decommissioning funding assurance as described above and finds that the decommissioning cost estimate is reasonable, that the surety method is acceptable, and that the licensee's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable.

Based on the above, the NRC staff finds that funds will be made available to decommission the facility and that the financial status of the licensee regarding decommissioning costs meets the requirements of 10 CFR 50.33(k) and 10 CFR 50.75 and is consistent with the guidance in NUREG-1537. Therefore, the NRC staff concludes that there is reasonable assurance that funds will be provided for decommissioning the NTR when needed.

1.9.3 Foreign Ownership, Control, or Domination

Section 104d of the AEA prohibits the NRC from issuing a license under section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The NRC regulations at 10 CFR 50.33(d) and 10 CFR 50.38, "Ineligibility of certain applicants," contain language to implement this prohibition.

The NTR is owned and operated by GEH, an entity of the State of California. According to the LRA, GEH is a private corporate entity and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. Further, license condition (LC) 2.C.(5) of Facility Operating License No. R-33 (unchanged in the proposed renewed license) for the NTR provides a negation action plan that requires that the Manager of the VNC, the Vice-President, Reactor Facility Safety and Security of GEH, and the Manager of GEH shall be U.S. citizens and that these individuals shall have the responsibility and exclusive authority to ensure and shall ensure, that the business and activities of GEH, with respect the license, are at all times conducted in a manner consistent with the protection of the public health and safety and the common defense and security. LC 2.C.(7) also requires that GEH shall cause to be transmitted to the Director, NRR within 30 days of filing with the U.S. Securities and Exchange Commission any schedule 13D or 13G filed pursuant to the Securities Exchange Act of 1934 that discloses beneficial ownership of a registered class of GE stock.

Based on the above, the NRC staff concludes that the NTR is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

1.9.4 Nuclear Indemnity

Pursuant to the requirements of the Price-Anderson Act (section 170 of the AEA) and the NRC's implementing regulations at 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," GEH currently has an indemnity agreement for the NTR with the Commission that will not terminate prior to the expiration of the NTR facility operating license. Therefore, GEH will continue to be a party to the indemnity agreement following issuance of the renewed facility operating license. GEH will be indemnified for any claims arising out of a nuclear incident under the Price-Anderson Act, section 170 of AEA, and in accordance with the provisions of its indemnity agreement of Appendix B, "Form of indemnity agreement with licensees furnishing insurance policies as proof of financial protection," to 10 CFR Part 140, up to $500 million.

1.9.5 Financial Considerations Conclusions

Based on its review as discussed above, the NRC staff concludes that there is reasonable assurance that the necessary funds will be available to support the continued safe operation of the NTR and, when necessary, to shut down the facility and carryout the decommissioning activities consistent with the guidance in NUREG-1537, Part 2 and NUREG/CR-1756. In addition, the NRC staff concludes that there are no foreign ownership, control, or domination issues or indemnity issues that would preclude the issuance of a renewed license.

1.10 Compliance with 10 CFR 50.64

SAR section 1.1 states that the NTR is a heterogeneous, HEU, graphite-moderated and reflected, light-water-cooled reactor, licensed to operate at power levels not in excess of 100 kW(t). SAR section 1.3 states that the fuel consists of HEU aluminum alloy disks, clad with aluminum. Since the NTR uses HEU fuel, it must comply with the requirements of 10 CFR 50.64, “Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors,” which states, in part, that if Federal Government funding for conversion cannot be certified, the proposal’s contents may be limited to a statement of this fact. If a statement of non-availability of Federal Government funding for conversion is submitted by a licensee, then it shall be required to resubmit a proposal for meeting the requirements of 10 CFR 50.64(b)(2) or (3) at 12-month intervals.

By letter dated February 15, 2023 (Ref. 35), GEH indicated that Federal Government funding for the conversion of the NTR to low-enriched uranium fuel is currently not available. The NRC staff’s review finds that this letter conforms with the requirements of 10 CFR 50.64 to provide a statement of non-availability of Federal Government funding for conversion at 12-month intervals.

1.11 Facility Operating License Changes

The NRC staff revised Facility Operating License No. R-33 by reformatting and renumbering the LCs for consistency with current non-power reactor operating licenses. Additionally, the NRC staff, following regulatory audit discussions with the licensee, removed LCs for materials that were no longer needed or used by the licensee and modified other LCs accordingly. The licensee acknowledged all proposed NRC staff changes in its emails dated April 21, (Ref. 38) and June 15, 2023 (Ref. 42). A description of the changes to the LCs is provided below.

Current LC 2.B.(2) states:

(2) Pursuant to the Act and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” to receive, possess and use in connection with the operation of the reactor:

a. 4 kilograms of contained U-235 as in-core reactor fuel;

b. 100 grams of plutonium for use in [including but not limited to] experimental devices, instrument check sources, and encapsulated fission foils;

c. 100 grams of uranium-233 for use in [including but not limited to] ionization chambers and experimental devices;

d. 700 grams of contained uranium-235 or 1500 grams of contained U-235 in uranium enriched to less than 4% U-235. This is not to be used as in-core fuel.

e. The limits in b.-d. above may include the types of materials authorized by Special Nuclear Material License SNM-960, as amended, Docket No. 70-754, and Reactor License TR-1, as amended, Docket No. 50-70, to be used in the reactor cell, south cell, north room, and control room, but not in experimental facilities of the NTR.

f. Such special nuclear material as may be produced by the operation of the reactor. The licensee is not authorized to separate this special nuclear material.

Current LC 2.B.(2) a. was revised to add “but not separate, up to” and “of an enrichment of 20 percent or greater in the isotope uranium-235, in the form of” to enhance the description of the material. Current LC 2.B.(2) b. was no longer needed and replaced with material formerly described by current LCs 2.B.(2) d. and e., as appropriately adjusted, and was revised to add “but not separate, up to.” Current LC 2.B.(2) c. was no longer needed and replaced with current LC 2.B.(2) f. and was revised to add “but not separate” instead of the last sentence.

The proposed LC 2.B.(2) states:

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use in connection with the operation of the reactor:

a. but not separate, up to 4 kilograms of contained uranium-235 of an enrichment of 20 percent or greater in the isotope uranium-235, in the form of in-core reactor fuel;

b. but not separate, up to 350 grams of contained uranium-235, that is not to be used as in-core fuel. This material can be used in the reactor cell, south cell, north room, and control room but not in the experimental facilities of the NTR.

c. but not separate, such special nuclear material as may be produced by the operation of the reactor.

Current LC 2.B.(3) states:

(3) Pursuant to the Act and Title 10, Chapter I Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” (a) to receive, possess and use 2,000 curies of either activated solids as contained in but not limited to such items as encapsulating materials, structural material and irradiated components or as contained materials; (b) any byproduct materials necessary for purposes of instrument calibration and startup sources; (c) 10 curies of tritium for pulsed neutron sources; and (d) to possess, but not to separate (except for byproduct material produced as allowed for experiments), such byproduct material as may be produced by the operation of the reactor.

Current LCs 2.B.(3)(a) through (d) were reformatted to a listing of items versus embedded in a paragraph for greater clarity and improved readability. Current LC 2.B.(3)(a) was revised to remove “but not limited to” to ensure that material specificity was well understood. Current LC 2.B.(3)(b) was revised to include the radium-beryllium sealed startup source, and to add a 0.03 curie limit for other seal sources used for the operation of the NTR. Current LC 2.B.(3)(c) was no longer needed and, therefore, deleted. Current LC 2.B.(3)(d) was renumbered to proposed LC 2.B.(3)(c) and was revised to remove the duplicate phrase “to possess”.

The proposed LC 2.B.(3) states:

(3) Pursuant to the Act and 10 CFR Part 30, to receive, possess, and use in connection with the operation of the facility:

a. up to 2,000 curies of either activated solids as contained in such items as encapsulating materials, structural material and irradiated components;

b. up to 0.2-curie radium-beryllium sealed startup source, and up to 0.03 curies of byproduct materials, in the form of sealed sources, for instrument calibration and source checks; and

c. but not to separate (except for byproduct material produced as allowed for experiments), such byproduct material as may be produced by the operation of the reactor.

Current LC 2.B.(4) states:

(4) Pursuant to the Act and Title 10 CFR Part 40, "Domestic Licensing of Source Material," to receive, possess and use 9.1 kg. of uranium and thorium as source material for experimental devices.

Current LC 2.B.(4) was no longer needed by the licensee and, therefore, was removed by the NRC staff. Additionally, references to "source" material in LC 1.I and to Part "40" in LC 2.C. were deleted as no longer applicable.

Current LC 2.C.(8) states:

(8) Prior to completion of transfer of the license, GE-Hitachi Nuclear Energy Americas LLC, shall provide the Nuclear Regulatory Commission staff satisfactory documentary evidence of a parent company guarantee or another method authorized by and meeting the requirements of 10 CFR 50.75 for decommissioning funding assurance in an amount no less than $3,411,000 for the NTR.

Current LC 2.C.(8) was required by the NRC in support of a pending license transfer from GE Company to GEH (NTR LA No. 23, dated September 6, 2007 (Ref. 40)). The NRC staff finds that this LC is no longer valid since the license transfer has been completed and, therefore, LC 2.C.(8) was removed.

2 REACTOR DESCRIPTION

2.1 Summary Description

Safety analysis report (SAR) section 1, "The Facility," states that the GE-Hitachi Nuclear Energy Americas LLC (GEH, the licensee) Nuclear Test Reactor (NTR, the facility) is a heterogeneous, tank type reactor. The core contains highly enriched uranium fuel that is graphite-moderated and -reflected. The core is cooled either by natural or forced flow of water circulated in a primary system constructed primarily of aluminum. The reactor coolant flows through an external heat removal and purification system. The reactor's experimental systems include a central sample tube, penetrations through and into the reflector, the reactor surfaces, neutron beams, and tubes from any of these facilities.

The NTR fuel is highly enriched uranium-aluminum alloy disks, clad with aluminum. The reactor exhibits a negative void and temperature coefficient of reactivity above 124 degrees Fahrenheit (°F) (51 degrees Celsius (°C)), which is approximately the steady-state operating temperature. Reactivity is controlled by up to six manually positioned cadmium sheets, four safety rods filled with boron carbide, and three control rods, also filled with boron carbide. The nominal thermal power level is 100 kilowatts thermal (kW(t)).

The NRC staff notes that few aspects of the NTR have changed since the issuance of the previous license renewal by License Amendment (LA) No. 21 (Reference (Ref.) 7). Therefore, where applicable, this safety evaluation report (SER) will reference information, including NRC staff findings, previously accepted in its safety evaluation (SE) issued with LA No. 21.

2.2 Reactor Core

SAR section 4.2, "Reactor Core," describes the NTR core as composed of 16 highly enriched-uranium fuel assemblies loaded into the annular fuel container can. The fuel container can was put into service in 1976 after the previous container, which had been in service for approximately 18 years, developed a leak in a weld area. SAR section 4.1, "Summary Description," describes the NTR as a light-water-cooled, highly enriched uranium, graphite-moderated and -reflected, thermal reactor with a nominal power rating of 100 kW(t). SAR section 4.2 states that the annular ends of the core can are aluminum plates. The plates are welded to the rolled aluminum sheets that make up the cylindrical inner and outer skins, respectively.

SAR section 4.2 also states that the fuel assemblies are supported by a core reel assembly. The core reel assembly is supported and guided by circular raceways in the end plates of the core can. A reel drive mechanism rotates the entire reel assembly to a desired position. The fuel loading chute is attached to the outer wall of the core can, inclined at an angle of approximately 30 degrees above horizontal. The aluminum loading chute is rectangular and is roughly 30 inches (in.) (76 centimeters (cm)) long, 20 in. (51 cm) wide, and 3 in. (8 cm) high. Slotted adapters in the chute both provide a guide for the aluminum-clad graphite chute plug and for the fuel loading tool during refueling operations. The fuel container assembly is shown in figure 2-1, "Fuel Container Assembly," below.

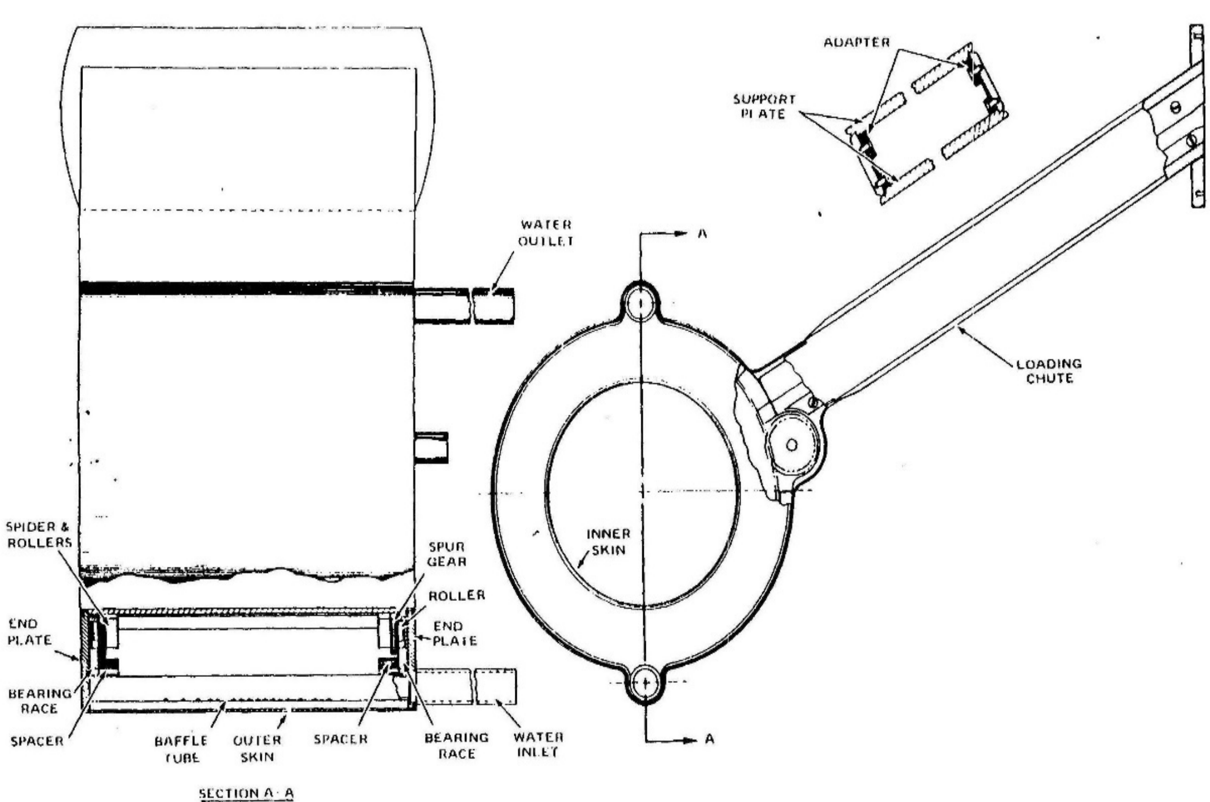

Figure 2-1: Fuel Container Assembly (SAR Figure 4-2)

SAR section 4.2 describes the coolant inlet and outlet lines. The north end plate of the core contains openings for the 1.5 in. (3.8 cm) primary coolant inlet and outlet lines. The inlet pipe is connected to a 1.375 in. (3.49 cm) flow distributor tube located inside the container below the core. Coolant exits the flow distributor tube via 25 holes that are 0.25 in. (0.64 cm) in diameter. The holes are spaced closer together near the core midplane. Moving outward from the center hole, the center-to-center distance between the next five holes is 0.4375 in. (1.11 cm). The next three holes are separated by 0.5 in. (1.3 cm), the next two by 0.75 in. (1.9 cm), and the last two by 1 in. (2.5 cm). An identical tube is connected to the outlet line on the top side of the core.

SAR section 4.2 describes the location of structures that support reactivity control devices. Eight 0.75 in. (1.9 cm) aluminum tubes are installed horizontally outside the outer surface of the core can in order to guide the control, safety, and neutron source rods. These tubes are supported by brackets attached to the end plates. Six slotted graphite ways are attached to the north end plate, parallel to the control, safety, and source rod guide tubes, in order to guide manually positioned cadmium poison sheets.

The NRC staff reviewed the information provided in SAR section 4.2 and finds that it provides an accurate description of the NTR reactor core. The NRC staff finds that SAR section 4.2 is consistent with the guidance in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 2, section 4.2, “Reactor Core,” which states that the SAR should contain the design information of all of the components of the reactor core. As stated in its SE issued with LA No. 21, the NRC staff found the reactor core acceptable. Given that the licensee has proposed no changes to the reactor core design, the NRC staff continues to find the NTR reactor core acceptable.

2.2.1 Reactor Fuel

SAR section 4.2.1, “Reactor Fuel,” describes the NTR reactor fuel, indicating that all the fuel in the core is from the original batch of fuel fabricated in 1957. Each fuel assembly consists of 40 aluminum-clad 2.75 in. (6.98 cm) outer diameter uranium-aluminum fuel disks supported by an aluminum shaft. Each disk is constructed from an outer edge ring, an inner edge ring, and a fuel bearing. The fuel bearing is a 0.142 in. (0.361 cm) thick doughnut-shaped piece consisting of uranium-aluminum alloy sandwiched between two pieces of 0.027 in. (0.069 cm) thick aluminum cladding. The outer edge ring fits around the fuel bearing and covers the outer face of the uranium-aluminum alloy. Similarly, the inner edge ring fits inside the fuel bearing and covers the inner face of the uranium-aluminum alloy. These three pieces are brazed together. The fuel bearing has an inner diameter of 0.58 in. (1.5 cm) and an outer diameter of 2.68 in. (6.81 cm). The assembled fuel element has an inner diameter of 0.516 in. (1.31 cm) and an outer diameter of 2.75 in. (7.0 cm). The inner and outer edge rings are 0.20 in. (0.51 cm) wide. A fuel disk is 0.180 in. (0.457 cm) thick aluminum with spacers installed on the shaft between fuel disks, with an additional 0.031 in. (0.079 cm) thick aluminum washer in every other space. The spacers give the fuel assembly an active length of 15.25 in. (38.7 cm).

SAR section 4.2.1, states that the reactor fuel was fabricated in accordance with GE Specification AP-RG-56-8-1.1, dated October 23, 1956, which included corrosion and helium leak testing. When the fuel container was removed in 1976, the fuel was removed, inspected, and leak checked, and no cleaning, replacement, or repair was necessary.

SAR section 16.1, “Reactor Fuel,” states that fuel cladding thickness is 50 percent of its original value. In its response to audit question 023 (Ref. 36), the licensee clarified that the aluminum fuel cladding is not a structural member; therefore, aging mechanisms that affect its mechanical state such as fatigue, creep, and ductility have limited importance to the residual cladding thickness. The aluminum cladding’s primary purpose is to provide a physical barrier that prevents direct contact between the coolant and the uranium/aluminum fuel and provides a barrier for unrestricted release of fuel and fission products to the coolant. The cladding can serve this physical barrier purpose with only a minimal residual thickness, for example with as little as 0.001-in. (0.00254-cm) residual thickness, which is about 5 percent of the original cladding thickness. Aging mechanisms such as fretting wear and fission product leaching could be delayed to some extent by a thicker residual cladding thickness, but these aging mechanisms, if active, would still be possible whether the thickness is as fabricated or reduced by corrosion to 0.001 in. (0.00254 cm). GEH estimated that 25 percent of the original cladding thickness would remain after 37 more years of operation, and thus concluded that the cladding is deemed adequate over the next licensing period to serve its primary purpose as a physical barrier against unrestricted fuel and fission product release.

The NRC staff reviewed the information provided in SAR section 4.2.1 and finds that it provides an accurate description of the NTR reactor fuel. The NRC staff finds that the licensee has described in detail the dimensions of the fuel elements to be used in the reactor. The NRC staff finds that SAR section 4.2.1 is consistent with the guidance in NUREG-1537, Part 2, section 4.2.1, “Reactor Fuel,” which states that the licensee should describe in detail the fuel elements to be used in the reactor. As stated in its SE issued with LA No. 21, the NRC staff found the reactor fuel acceptable. Given that the licensee has proposed no changes to the reactor fuel, the NRC staff continues to find the NTR reactor fuel acceptable.

2.2.2 Control Rods

SAR section 4.2.2, "Control and Safety Rods," describes means of controlling reactivity in the NTR, including control rods, safety rods, and manual poison sheets (MPS). The control rods are movable neutron absorbers located about the periphery of the fuel container. The NTR uses two coarse control rods, one fine control rod, and four safety rods that provide control and shutdown capability for the NTR. The safety rods are designed to rapidly insert and scram the reactor, and the control rods are designed for precise position control and indication during normal operation. Manual poison sheets are repositioned when the reactor is shut down in order to limit excess reactivity and increase available shutdown margin. Withdrawing the safety rods loads two constant force spiral springs, which drive the rods into the core on a scram signal. The control rods will also drive into the core on a scram signal, provided that electrical power is available.

SAR section 4.2.2 states that the poison section of control and safety rods contains boron carbide. The poison section of the safety rods is 20 in. (51 cm) long and contains 0.5 in. (1.3 cm) diameter boron carbide cylinders. The actual full stroke of the safety rods is 28 in. (71 cm). The poison section of the coarse control rods is 16 in. (41 cm) long and contains 0.5 in. (1.3 cm) diameter boron carbide cylinders. The poison section of the fine control rod is 18 in. (46 cm) long and contains 0.365 in. (0.93 cm) diameter boron carbide cylinders. Manual poison sheets consist of 0.032 in. (0.081 cm) thick by 19 in. (48 cm) long cadmium plates laminated between two 6061-T6 aluminum plates that are 0.08 in. (0.2 cm) thick, 3 in. (7.6 cm) wide, and 40.5 in. (103 cm) long. The licensee possesses seven different widths of cadmium poison sheets ranging from 0.17 in. (0.43 cm) to 2.75 in. (7.0 cm) wide. All cadmium poisons sheets are supported by guides that place the center of the poisons 0.6 in. (1.5 cm) from the outside edge of the active core.

SAR section 4.2.2 states that MPS are only repositioned when the reactor is shut down. The MPS do not have a drive mechanism, or any automatic functions associated with them. The MPS are held in place with a spring-loaded latch handle that provides positive restraint of the sheets with respect to the reactor assembly. Repositioning the MPS requires entering the reactor cell and physically latching or unlatching the sheets with a special tool.

Proposed technical specification (TS) 5.3.1, "Control System," item (3) requires each installed MPS used to satisfy excess reactivity requirements to be restrained in its slot. The NRC staff found proposed TS 5.3.1, item (3) acceptable, as evaluated in SER sections 4.2.9, "Uncontrolled reactivity increase," and 5.5.3, "Reactor Core and Fuel."

In its response to audit question 004 (Ref. 36), the licensee clarified that three of the six slots were updated with a latching mechanism to hold the MPS in place. The other three slots are padlocked in order to prevent their use. The licensee indicated that as reactivity is lost through burnup, the thickness of the MPS needed to limit excess reactivity is reduced. SAR section 4.2.2 states that the core model for the current exposure includes the use of one MPS that is 1/8 of its original thickness.

SAR section 4.2.2 states that there are drive-out, drive-in, safety-rod-in, and separation limit switches to ensure safe control rod operation. The drive-out limit switch ensures that all safety rods are individually withdrawn fully before any control rod is moved. The drive-in limit switch is interlocked to prevent energizing the electromagnets unless all control rods and neutron sources are fully inserted. The safety-rod-in position switch energizes a green light at the operator console to provide a visual confirmation that the safety rods are fully inserted. The separation limit switch operates in series with the carriage-out limit switch to energize a yellow light to indicate that the rods and carriage are out. This interrupts the voltage in the safety system and deenergizes the safety rod electromagnets.

SAR section 4.2.2 states that the control rods are periodically tested to measure flight time. If a rod does not meet an insertion time limit, it will be considered inoperable until it is repaired. The rods have not exhibited any unexpected results. Further, SAR section 13.5.3, “Reactivity Insertions – without Scram,” describes a limiting accident analysis, in which even when no scram occurs, the fuel is not damaged.

Proposed TS 3.2.5, “Scram Time,” requires that the average scram time of the four safety rods shall not exceed 300 milliseconds. The NRC staff found proposed TS 3.2.5 acceptable, as evaluated in SER section 5.3.2.5, “Scram Time.”

SAR section 4.2.2 states that the reactivity worth of the safety rods varies due to the lack of symmetry in the arrangement of nuclear poisons around the core. The reactivity worth of the most effective safety rod is about $1. SAR section 4.4.3, “Operating Limits,” states that the total worth of all safety rods is $3.86, although a conservatively assumed value of $2 is noted in SAR section 4.2.2 and SAR table 4-1, “Nuclear Parameters.” The control rods were calibrated, and the results indicate a total worth of all the control rod of approximately $2.3.

SAR section 4.2.2 states that the reactivity worths of the individual MPS were obtained with a pulsed neutron source. The worth of the widest sheet was historically determined to be approximately $1, and the worth of a sheet roughly half this width was determined to be approximately $0.5. For NTR operation in July 2020, the worth from changing MPS size from the single installed 1/8 sheet in slot #2 to a 1/16 sheet in slot #2 was determined via measurement to be approximately $0.13. This MPS size change reactivity worth was determined with control rods in the critical position, safety rods fully withdrawn, the neutron radiography source log inserted, and a 1.0°F reduction in primary coolant temperature (from 77°F to 76°F (25°C to 24.4°C)) between the pre-change and post-change conditions. Reactor modeling predicted this MPS change to result in a $0.17 reactivity change.

SAR section 4.2.2 states that the withdrawal speed of the coarse control rod drive is 0.140 inches per second (in/s) (0.356 centimeters per second (cm/s)) and that of the fine control rod drive is 0.145 in/s (0.368 cm/s). Based on these speeds and the total worth of all control rods, the licensee estimated the average reactivity insertion rate associated with simultaneous withdrawal of all three control rods to be roughly $.02/s. The licensee indicated that this is a value that is easily controlled manually by the reactor operator.

Proposed TS 3.2.4, “Control Rod Withdrawal Rate,” limits the rate of withdrawal of each control rod to less than 1/6 in/s (0.423 cm/s) during normal operation. The NRC staff found proposed TS 3.2.4 acceptable, as evaluated in SER section 5.3.2.4, “Control Rod Withdrawal Rate.”

SAR section 4.2.2 also estimates the reactivity addition rate from withdrawing safety rods. Since interlocks prevent withdrawing a safety rod until the previous rod has been fully withdrawn, the licensee considered the reactivity addition rate from withdrawing one safety rod at a time. The licensee estimated the average reactivity addition rate from withdrawal of a safety rod to be $.083/s, conservatively assuming that the safety rod is worth $1.5.

Proposed TS 3.2.3, “Safety Rod Withdrawal Rate,” limits the rate of withdrawal of each safety rod to less than 1.25 in/s (3.18 cm/s) during normal operation. The NRC staff found proposed TS 3.2.3 acceptable, as evaluated in SER section 5.3.2.3, “Safety Rod Withdrawal Rate.”

SAR section 4.2.2 states that reactivity is also managed by cadmium sheets installed around the reactor periphery. Installation of these MPS limits excess reactivity and increases available shutdown margin. These MPS are only repositioned when the reactor is shut down and are held in place by a spring-loaded latch handle that provides positive restraint with respect to the reactor assembly.

The NRC staff finds that the licensee has discussed the reactivity control systems and the relationship between the system design and the operational characteristics of the NTR. The NRC staff concludes that the functional and safety-related design bases can be achieved by the reactivity control system designs. The NRC staff finds that the changes in reactivity caused by the control rod dynamic characteristics are acceptable.

The NRC staff reviewed the information provided in SAR section 4.2.2 and finds that it provides an accurate description of the NTR control rods. The NRC staff finds that SAR section 4.2.2 is consistent with the guidance in NUREG-1537, Part 2, section 4.2.2, “Control Rods,” which states that the licensee should describe the control and safety rods to be used in the reactor. As stated in its SE issued with LA No. 21, the NRC staff found the control rods acceptable. Given that the licensee has proposed no changes to the control rods, the NRC staff continues to find the design of the NTR reactivity control devices including safety rods, control rods, and MPS acceptable.

2.2.3 Neutron Moderator and Reflector

SAR section 4.2.3, “Neutron Moderator and Reflector,” describes the neutron moderator and reflector. The reflector-moderator is a 5-foot (ft) (1.5 meter (m)) cube of reactor-grade graphite that also provides physical support of the fuel container. The fuel container is centered in the reflector with the core axis horizontal. The cube is composed of many small pieces (primarily 4 in. by 4 in. (10 cm by 10 cm) of varying lengths) stacked to form a cube. The graphite cube is supported by an aluminum box and base, as described in SAR section 4.2.5, “Core Support Structure.”

SAR section 4.2.3 states that the graphite cube has penetrations for the fuel loading chute, control rod, safety rod, neutron source guide tubes, manually positioned poison sheet slots, cable held retractable irradiation system, and the core reel drive shaft. The graphite cube has removable sections to enable inspection of the fuel container.

SAR section 5.2.3, “System Disruptions,” states that loss of coolant would result in reactor shutdown due to moderator voiding. During a complete instantaneous loss of primary coolant flow without a reactor scram, fuel damage does not occur. Natural convection cooling is sufficient and, therefore, forced coolant flow is only conservatively required above 0.1 kW. A complete loss of coolant in the core tank with a simultaneous failure to scram the reactor at full power would result in a reactor shutdown because of moderator voiding. Peak fuel temperature would reach a maximum of 626°F (330°C) in about 30 minutes after coolant loss. No damage to the fuel would result, so the consequences of this accident are minimal.

The NRC staff reviewed the information provided in SAR section 4.2.3 and finds that it provides an accurate description of the NTR neutron moderator and reflector. The NRC staff also reviewed the information provided in SAR section 5.2.3 and finds that it provides an accurate description of the NTR response to a loss of coolant and accurately concludes that no damage to the fuel would result. The NRC staff finds that SAR section 4.2.3 is consistent with the guidance in NUREG-1537, Part 2, section 4.2.3, “Neutron Moderator and Reflector,” which states that the licensee should describe how the neutron moderator and reflector are integral parts of the reactor core. As stated in its SE issued with LA No. 21, the NRC staff found the neutron moderator and reflector acceptable. Given that the licensee has proposed no changes to the neutron moderator and reflector, the NRC staff continues to find the design of the NTR neutron moderator and reflector acceptable.

2.2.4 Neutron Startup Source

SAR section 4.2.4, “Neutron Startup Source,” describes the neutron startup source. The source-detector arrangement provides the minimum neutron flux signal required for the nuclear instrumentation for startup and also gives a good indication of subcritical multiplication. As described in SAR section 7.1, “Summary Description,” without the neutron source, the compensated ion chamber signal will not meet the rod withdrawal permissive, which will prevent the control or safety rods from being withdrawn.

SAR section 4.2.4 states that the source is installed on an electric motor drive mechanism in a configuration like that of the control rod drives. The source has the same controls and indications as a control rod drive, except that continuous position indication is not provided. The source is in an R-monel encapsulation approximately 0.5 in. (1.3 cm) in diameter and 3.5 in. (8.9 cm) long, and it consists of a 0.2 curie (0.7×10^{10} becquerel) radium-beryllium source emitting 10^6 neutrons per second.

SAR section 4.2.4 states that a scram signal automatically drives the source to the fully inserted position. Subcritical multiplication from the neutron source brings the power range detectors on-scale, as described in SAR section 7.3.3, “Nuclear Scrams.”

The NRC staff reviewed the information provided in SAR sections 4.2.4, 7.1, and 7.3.3 and finds that it provides an accurate description of the NTR neutron startup source. The NRC staff finds that SAR section 4.2.4 is consistent with the guidance in NUREG-1537, Part 2, section 4.2.4, “Neutron Startup Source,” which states that the neutron source is acceptable for this type of reactor. As stated in its SE issued with LA No. 21, the NRC staff found the neutron startup source acceptable. Given that the licensee has proposed no changes to the neutron startup source, the NRC staff continues to find the design of the NTR neutron startup source acceptable.

2.2.5 Core Support Structure

SAR section 4.2.5, “Core Support Structure,” states that the core support structure consists of an aluminum box which contains the graphite cube. The fuel container rests on the sections of the graphite cube beneath it. A cadmium liner is provided for the north and east sides of the box. The box containing the graphite cube rests on an aluminum plate fastened to a framework of aluminum I-beams. The I-beam base is clamped to steel support plates anchored to the reactor cell floor.

The NRC staff reviewed the information provided in SAR section 4.2.5 and finds that it provides an accurate description of the NTR core support structure. The NRC staff finds that SAR section 4.2.5 is consistent with the guidance in NUREG-1537, Part 2, section 4.2.5, “Core Support Structure,” which states that the licensee has provided a description of the core support structure, derived from the planned operational characteristics of the reactor. As stated in its SE issued with LA No. 21, the NRC staff found the core support structure acceptable. Given that the licensee has proposed no changes to the core support structure, the NRC staff continues to find the design of the NTR core support structure acceptable.

2.3 Reactor Tank or Pool

SAR section 4.1 states that the reactor tank uses a light-water coolant. SAR section 5.2.2, “System Operation,” states that maintaining water in the core tank ensures that there will be no reactivity insertions due to removal of voids or sudden addition of water into the core tank during reactor operation. The reactor is not permitted to operate above 0.1 kW unless the tank is filled with water.

The NRC staff finds that this requirement is reflected in proposed TS 3.2.6, “Reactor Safety System and Safety-Related Items,” table 3-2, “Reactor Safety-Related Items,” proposed TS 3.3.2, “Core Tank Full,” and proposed TS 3.8.5, “Experimental Objects in the Core Tank.” The NRC staff found proposed TSs 3.2.6, table 3-2, 3.3.2, and 3.8.5 acceptable, as evaluated in SER sections 5.3.2.6, “Reactor Safety System and Safety-Related Items,” 5.3.3.2, “Core Tank Full,” and 5.3.8.6, “Experimental Objects in the Fuel Loading Chute,” respectively.

SAR section 7.4, “Safety-Related Items,” and SAR table 7-2, “Safety-Related Items,” indicate that an alarm provides visible and audible indication when the water level is too low to ensure that the core tank is filled with water. SAR section 5.2.2 states that primary system leakage is maintained below 10 gallons (38 liters) per day as an operational practice that ensures that there are 10 days between high- and low- level alarms. (Loss-of-coolant accidents are discussed in SER section 4.2.12, “Reactor Loss-of-Coolant Accident.”)

The NRC staff’s review finds that the licensee considered the possibility of primary coolant leakage on core reactivity, and tracks core leakage as an operational practice and has justified limiting conditions for operation (LCOs) in proposed TSs 3.2.6, table 3-2, 3.3.2, and 3.8.5. The NRC staff also finds that SAR section 4.1 is consistent with the guidance in NUREG-1537, Part 2, section 4.3, “Reactor Tank or Pool,” which states that the licensee has considered the loss of coolant from the tank and has TS LCOs that help ensure that the operators are alerted to any loss of coolant. As stated in its SE issued with LA No. 21, the NRC staff found the primary cooling system (including the reactor tank) acceptable. Given that the licensee has proposed no changes to the design of the reactor tank or the applicable TSs (as listed above), the NRC staff continues to find the NTR reactor tank acceptable.

2.4 Biological Shield

SAR section 4.3, “Biological Shield,” provides a description of the NTR biological shield and states that the reactor cell and alcove provide biological shielding that ensures continued compliance to established limits and is consistent with as-low-as-reasonably-achievable (ALARA) practices. The shield consists of the Reactor cell, South cell, and Modular Stone Monument. SAR section 4.3 also states that whenever new operating or maintenance conditions are encountered, radiation surveys are made to determine that existing shielding is adequate and consistent with ALARA practices. Either temporary or permanent improvements in the shield are made if the results of the survey indicate that they are necessary.

Further, the SAR provides a list of typical radiation levels in the areas of the NTR facility, while the reactor is operating at 100 kW(t), which is given in SER table 2-1, “Radiation Level in Areas of NTR,” below. Unless shielding changes are made, the listed radiation levels are all proportional to reactor power. The values listed include contributions from fast, intermediate, and slow neutrons and gamma rays.

Table 2-1 Radiation Level in Areas of NTR

Location	Shutters Open (mrem/hr)	Shutters Closed (mrem/hr)
At reactor console	3.0	1
Hallway south of control room	1	1
Building 105 equipment room	1	1
Cell roof directly above reactor (top shield slabs in place)	65	65
Cell sample sink	57	2.5
Setup Room	1	1
North Room (center of room)	5.5	1

As stated in its SE issued with LA No. 21, the NRC staff found the biological shield acceptably designed to reduce external radiation exposures to acceptable levels. Given that the licensee has proposed no changes to the design of the biological shield, the NRC staff continues to find the NTR biological shield acceptable.

2.5 Nuclear Design

The nuclear design encompasses normal operating conditions, core physics parameters, and operating limits.

2.5.1 Normal Operating Conditions

SAR section 4.4.3 states that the NTR operates with a single, fixed core configuration and that the reactor fuel is not reconfigured in any way. Outside the core, experiments and MPS configuration may be altered. SAR section 4.4.1, “Normal Operating Conditions,” states that all fuel assemblies, control rods, and safety rods are in fixed positions that are not changed. SAR section 4.4.1 states that the NTR burns approximately $0.03 positive excess reactivity per year and that the planned core configurations are to remove enough cadmium from the remaining MPS to maintain normal operation while ensuring that the potential excess reactivity is less than or equal to $0.76. During normal operating conditions, the NTR does not exceed 100 kW(t) power with maximum temperature and pressure not to exceed 150°F (66°C) and 20 pounds per square inch (140 kilopascal), respectively.

SAR section 4.4.1 states that since initial criticality, the reactor has accumulated approximately 198.5 megawatt-days of operation. The licensee estimated that over the life of the reactor, $2.1 of reactivity has been lost due to fuel burnup, $1.8 has been lost due to buildup of fission product poisons other than samarium-149, and $0.62 has been lost due to buildup of samarium-149. Plutonium buildup in the NTR is negligibly small.

SAR section 4.4.1 lists administrative and physical controls put in place to prevent inadvertent addition of positive reactivity, including reconfiguration of MPS. MPS are physically latched and cannot move during operation. SAR section 4.2.2 states that MPS can only be inserted or withdrawn by entering the reactor cell. Then, moving the MPS requires removing a shield plug from the shield face and using a special tool to physically latch or unlatch the sheet. SAR section 4.4.1 lists other controls that prevent inadvertent addition of positive reactivity:

- During reactor operation the reactor cell door is locked so that core changes are not possible. The MPS are physically latched and cannot move during operation. During operation, then, the only positive reactivity additions possible are from movable experiments, coolant flow changes, and movement of control and safety rods. Control and safety rods are manipulated by licensed operators in accordance with written procedures. These reactivity additions are limited physically (water coefficient of reactivity) and by design (control and safety rod drive speeds and experiment reactivity worth).

- Entry into the reactor cell, when the reactor is critical, is only authorized by special procedure (Engineering Release) describing the operation to be performed. This procedure must be approved by the Manager, NTR, and reviewed by the Manager, Regulatory Compliance, prior to entry.

- When the reactor is operating, there is a south cell door/shutter interlock to prevent inadvertent entry into the south cell with the south cell shutter open. An electric photocell light mechanism causes an audible alarm to actuate when an entry into the south cell is made.

- Entry into the cell is only permitted when the reactor is critical if the power is stable, entry does not distract the operator, and the task can be performed safely.

- Changes in positive reactivity during shutdown are possible by control rod movement, MPS movement, and horizontal facility changes.

- Minimum staffing requirements exist when the reactor is not secure.

- The senior reactor operator is present during any MPS changes.

The licensee's analysis of other elements contributing to reactivity, including fuel composition changes and control and safety rods, are discussed in additional detail in SER sections 2.2.2, "Control Rods," 2.3, "Reactor Tank or Pool," and 2.5.2, "Core Physics Parameters."

The NRC staff finds that the licensee has described the core configuration for the proposed licensing period. The NRC staff finds that control and safety rods remain in place and are never moved to different guide tubes. The axial position of control and safety rods is changed by insertion and withdrawal during normal startup and shutdown procedures, and thus the reactor fuel is not reconfigured. The NRC staff finds that the MPS are installed and removed to control available excess reactivity. The NRC staff finds that the fuel composition is calculated based on the operating history of the NTR and that the license has analyzed all reactivity changes, including those due to experiment facilities, MPS reconfiguration, burnup, and buildup of plutonium and fission product poisons. The NRC staff finds that the licensee has described controls that ensure appropriate configuration of MPS. Based the information described above, the NRC staff concludes that the normal operating conditions of the NTR are acceptable.

2.5.2 Core Physics Parameters

SAR section 4.4.2, "Reactor Core Physics Parameters," states that the NTR is equipped with features that enable the reactor to be sensitive of reactivity changes. These features include a low critical mass, the fuel-to-sample geometry, and the control system. The thermal neutron flux profiles in the horizontal facility and in three of the MPS slots are expected to correspond very closely to the axial neutron flux and thermal power distribution in the adjacent section of the core. The dominant effect for accident analysis is density change. In the NTR, the reactivity effect from a temperature change in the fuel annulus is observed in a fraction of seconds and is positive up to 124°F (51.1°C) and negative at higher temperatures. The change does not occur fast enough to cause a nuclear excursion. The temperature coefficient occurs primarily as a result of displacement of the coolant from the core region. The coefficient is not affected by fuel burnup and is not expected to vary significantly with core life. In its response to audit question 003, the licensee stated that the temperature coefficient is based on experiments documented between 1959 to 1964. Coefficients of reactivity for the NTR are listed in SAR table 4-2, "Nuclear Parameters."

Proposed TS 5.3.4, "Temperature Coefficient of Reactivity," states that the core is designed to exhibit a negative temperature coefficient of reactivity above 124°F, which is approximately the reactor steady-state operating temperature. The NRC staff found proposed TS 5.3.4 acceptable, as evaluated in SER section 5.5.3.

As described in SAR section 4.4.2, the licensee performed experiments to confirm that the void coefficient of reactivity is negative by removing pieces of aluminum from the core. Since removing the aluminum resulted in a positive reactivity change, the void coefficient is negative. In this experiment, the magnitude of the reactivity coefficient was not calculated, only the sign was confirmed. The licensee analytically obtained the void coefficient of reactivity magnitude by extrapolating the trend in the temperature coefficient of reactivity on the basis that the value of the temperature coefficient primarily reflects changes in water density. Calculation of this coefficient is described in detail in the licensee's supplemental information provided by letter dated September 15, 2022, item No. 7 (Ref. 19).

SAR section 4.4.4.2, "Reactivity Coefficients and Point Kinetics Parameters," discusses the method used by the licensee to calculate temperature reactivity coefficients for graphite reflectors. These reactivity coefficients are calculated using the licensee's core computational model.

SAR section 4.4.4, "NTR Core Computational Model," discusses the computational core model developed for neutronics parameter and reactivity analysis. These calculations were performed using the Monte Carlo N-Particle Transport computer code, version 6 (MCNP6), which solves the linear neutron transport equation using the Monte Carlo process. It can compute the eigenvalue for neutron multiplying systems. SAR section 4.4.4 also states that the MCNP6 model of the NTR uses ENDF/B-VII continuous energy cross section data, and uses $S(\alpha,\beta)$ thermal scattering kernals for thermal neutron scattering in light water and graphite. The NRC staff finds this data acceptable for analysis of the NTR.

SAR section 4.4.4 states that the BURN function in MCNP6 was used to deplete the fuel from beginning of life to the current core exposure. Fuel isotopic results were used as fuel material inputs for core models at different NTR operating statepoints corresponding to MPS configuration changes and critical control rod calibration measurements. Results for critical reactor configurations and measured reactivity worths for MPS changes were compared with NTR measurements. Differences between core model-predicted control rod worths and net reactivity gains from MPS changes are within the overall model uncertainty for the past operation statepoints and for the current exposure statepoint. This validates the core model and depletion analysis fuel isotopic results. In its response to audit question 006, the licensee stated that the uncertainty includes uncertainty of the MCNP6 code, depletion analysis isotopics, as-modeled NTR configuration, changes in control rod position during past operation statepoints, and core configurations modeled. The licensee also stated that while current measurements of safety rod worths are not available, core model predictions have been found to support historical values of safety rod worths.

SAR table 4-3, “Core Model Calculated Point Kinetics Parameters,” shows effective delayed neutron fractions and neutron generation times calculated by the licensee’s core model for beginning of life and the current exposure.

The NRC staff finds that MCNP6 has been widely used for neutronics parameter and reactivity analysis of research and test reactors, and its use in the LRA is appropriate. The NRC staff notes that the effective delayed neutron fraction for both the beginning of life and the current exposures are slightly higher than the effective delayed neutron fraction specified in SAR section 4.4.2, which was used to calculate reactivity parameters in SAR table 4-1. The NRC staff finds that these parameters are within an expected range, and that the licensee’s choice to calculate reactivity parameters using the smaller value of the effective delayed neutron fraction provides a more conservative result for the excess reactivity value. This could have a non-conservative effect on the calculation of the shutdown margin, but the NRC staff expects this effect to be small in relation to the licensee’s margin.

The NRC staff finds that the analyses of neutron lifetime, effective delayed neutron fraction, reactivity, and coefficients of reactivity have been completed using methods validated at similar reactors. The NRC staff also finds that the licensee has considered the effects of fuel burnup and reactor operating characteristics for the life of the reactor in the analyses of the reactor core physics parameters. Therefore, the NRC staff concludes that the licensee’s analysis of core physics parameters is acceptable.

2.5.3 Operating Limits

Operating limits consist of safety limits (SLs), limiting safety system settings (LSSSs), and LCOs related to excess reactivity and shutdown margin.

2.5.3.1 Safety Limits

SAR section 4.4.3 discusses the NTR SLs. The licensee stated that SLs are limits on important process variables which are found to be necessary to reasonably protect the integrity of the NTR fuel. The only accidents that could possibly cause fuel damage and a release of fission products from the NTR fuel are those resulting from large reactivity insertions. Given the $0.76 potential excess reactivity limit, a large reactivity insertion not possible. Therefore, there is no mechanistic way of damaging the fuel. The licensee also stated that a departure from nucleate boiling ratio (DNBR) equal to 1.5 was conservatively selected as a safe operating condition for steady-state and quasi-steady-state operation. The reactor thermal power level with a DNBR of 1.5 is 190 kW. The DNBR safety limit protects against fuel damage due to a rise in surface temperature. Additional SLs for reactor transient conditions were determined not to be needed because the potential excess reactivity limit of $0.76 provides the margin necessary for any other transient conditions. SAR section 13.7.4, “Safety Limits,” provides additional detailed analysis supporting the SL of 190 kW.

Proposed TS 2.1, “Safety Limits,” states that the true value of the reactor thermal power shall not exceed 190 kW.

The NRC staff finds that the requirements reflected in proposed TSs 2.1 and 3.1.1, "Potential Excess Reactivity," help to ensure that the SL of 190 kW will prevent fuel damage and the release of any fission products. The NRC staff found proposed TSs 2.1 and 3.1.1 acceptable, as evaluated in SER sections 5.2.1, "Safety Limits," and 5.3.1.1, "Potential Excess Reactivity." Further, the NRC staff's review finds that the SL of 190 kW remains unchanged since the issuance of the previous license renewal by LA No. 21. Given that the SL of 190 kW and proposed TS 2.1 remain unchanged since the previous license renewal, the NRC staff continues to find the SL and proposed TS 2.1 acceptable.

2.5.3.2 Limiting Safety System Settings

SAR section 4.4.3 discusses the LSSS for the NTR, which is 125 kW. This is an overpower scram limit which indicates that the linear neutron power monitor channel set point shall not exceed the measured value of 125 kW. The trip setpoint is 120 percent of full power, or 120 kW. Accident and transient analysis described in SAR chapter 13, "Accident Analysis," models a scram set point of 150 kW and does not indicate fuel damage.

Proposed TS 2.2, "Limiting Safety System Settings," states that the linear neutron power monitor channel set point shall not exceed the measured value of 125 kW.

The NRC staff finds that the requirements reflected in proposed TS 2.2 help to ensure that the SL of 190 kW will not be reached during any accident or transient conditions. The NRC staff found proposed TS 2.2 acceptable, as evaluated in SER section 5.2.2, "Limiting Safety System Settings." Further, the NRC staff's review finds that the LSSS of 125 kW remains unchanged since the issuance of the previous license renewal by LA No. 21. Given that the LSSS of 125 kW and proposed TS 2.2 remain unchanged since the previous license renewal, the NRC staff continues to find the LSSS and proposed TS 2.2 acceptable.

2.5.3.3 Excess Reactivity

SAR section 4.4.3 states that the potential excess reactivity is limited to $0.76. Further, SAR section 13.5.3 states that a $0.76 step reactivity insertion with failure to scram would not result in fuel failure or the release of fission products. The SAR states that in order to determine the effects of positive reactivity additions from less than full power and temperature, additional transients were run with an initial power level of 1 x 10^{-7} kW. Inlet water temperatures ranged from 55°F to 90°F (12.8°C to 32.2°C) and the initial positive reactivity step of $0.76. Results show that the positive reactivity feedback from the temperature coefficient is more important for the zero-power cases because coolant temperatures are lower. Reactor power and peak fuel temperature versus time for a $0.76 step insertion from 1 x 10^{-7} kW and 55°F (12.8°C) inlet water temperature is given in SAR figure 13-10 (reproduced as figure 2-2, "Reactor Power and Hot Spot Fuel Temperature Versus Time, $0.76 Step from Source Level, 55°F Coolant Inlet Temperature – No Scram," below). As shown in figure 2-2 below, limiting the step reactivity to $0.76 or less ensures that the peak cladding temperature remains below the clad melting temperature, and thus there are no mechanisms available that will cause fuel damage.

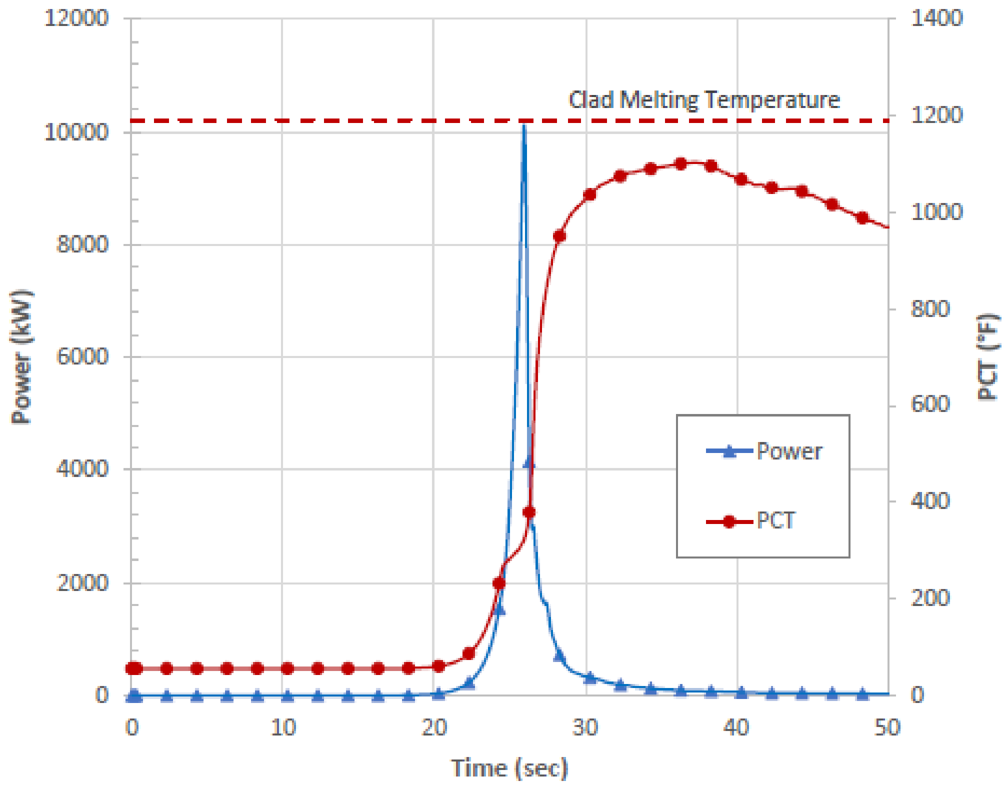

Figure 2-2: Reactor Power and Hot Spot Fuel Temperature Versus Time, $0.76 Step from Source Level, 55°F Coolant Inlet Temperature – No Scram (SAR Figure 13-10)

In its response to audit question 026, the licensee stated that reactivity insertion due to graphite temperature change is not actively factored into excess reactivity calculations. The licensee also stated that graphite temperature increases from approximately 70-80°F (21-27°C) at startup to between 120°F (49°C) and 140°F (60°C) after 3-4 hours of operation at full power. The NRC staff finds that not factoring graphite temperature into the excess reactivity limit is appropriate because graphite temperature change will be gradual and the potential excess reactivity limit is based on the step insertion of reactivity transient. Additionally, analysis of ramp reactivity insertions in SAR section 13.5.2, "Idealized Finite Ramp Reactivity Insertions – with Scram," does not depend on the potential excess reactivity limit, and analyzed reactivity insertion rates far exceed that achievable through graphite heating. Potential excess reactivity is also used to calculate shutdown margin, which is discussed in SER 2.5.3.4, "Shutdown Margin," which was evaluated by the NRC staff and found to be acceptable.

Proposed TS 3.1.1, "Potential Excess Reactivity," limits the potential excess reactivity of the NTR core to $0.76.

The NRC staff finds that the licensee's analysis considers all design features and feedback mechanisms contributing to excess reactivity. The discussion of the limits on excess reactivity shows that credible failures that instantaneously add reactivity would not lead to loss of fuel integrity and, therefore, would not cause unacceptable risk to the public from a transient.

The NRC staff finds that the requirements reflected in proposed TS 3.1.1 help to ensure that fuel failure will not occur if the excess reactivity is limited to $0.76. The NRC staff found proposed TS 3.1.1 acceptable, as evaluated in SER section 5.3.1.1, “Potential Excess Reactivity.” Further, the NRC staff’s review finds the potential excess reactivity limit of $0.76 remains unchanged since the issuance of the previous license renewal by LA No. 21. Given that the potential excess reactivity limit of $0.76 and proposed TS 3.1.1 remain unchanged since the previous license renewal, the NRC staff continues to find the potential excess reactivity limit and proposed TS 3.1.1 acceptable.

2.5.3.4 Shutdown Margin

SAR section 4.4.3 states that the calculated shutdown margin for the NTR is $2.00. The licensee calculated this value as the total safety rod worth minus the worth of the most reactive safety rod and the potential excess reactivity limit.

Proposed TS 3.1.3, “Minimum Shutdown Margin,” limits the shutdown margin with the maximum worth safety rod stuck out to $1.00.

The NRC staff’s review of proposed TS 3.1.3 finds that it is consistent with the guidance in section 4.5.3, “Operating Limits” of NUREG-1537, Part 2, as it requires the most negative reactivity obtainable by control rods from any reactor condition. It assumes that the most reactive safety rod and non-scrammable (control) rods are in their most reactive positions.

As discussed in SER 2.5.3.3, “Excess Reactivity,” the licensee did not explicitly consider graphite temperature in the calculation of potential excess reactivity. However, the NRC staff finds that reactivity insertion due to increasing graphite temperature would be offset by changes in critical control rod position. If graphite temperature were to increase such that excess reactivity from control rods, primary coolant temperature change, and installed experiments would exceed $0.76, then, in order to satisfy the requirements of TS 3.1.3, negative reactivity would need to be added.

The NRC staff finds that the licensee’s analysis and limits on potential excess reactivity and shutdown margin provide reasonable assurance that the NTR can be shut down from any permissible reactor condition, including a loss of electrical power.

The NRC staff finds that the licensee’s analysis considers all design features and feedback mechanisms contributing to the excess reactivity. The discussion of the limits on excess reactivity shows that credible failures that instantaneously add reactivity would not lead to loss of fuel integrity and, therefore, would not cause unacceptable risk to the public from a transient.

The NRC staff finds that the requirements reflected in proposed TS 3.1.3 help to ensure that the reactor can be shut down from any condition by maintaining the minimum shutdown margin of $1.00. The NRC staff finds proposed TS 3.1.3 acceptable, as evaluated in SER section 5.3.1.3, “Shutdown Margin.” Further, the NRC staff’s review finds that the minimum shutdown margin limit of $1.00 remains unchanged since the issuance of the previous license renewal by LA No. 21. Given that the minimum shutdown margin limit of $1.00 and proposed TS 3.1.3 remain unchanged since the previous license renewal, the NRC staff continues to find the minimum shutdown margin limit and proposed TS 3.1.3 acceptable.

2.6 Thermal-Hydraulic Design

SAR section 4.5, “Thermal-Hydraulic Design,” discusses reactor trips based on reactor power, core outlet temperature, and coolant flow rate. The overpower scram limit is 125 kW and the core outlet temperature limit is 222°F (106°C). When reactor power is greater than 0.1 kW, the reactor shall scram when coolant flow rate is less than 15 gallons per minute (57 liters per minute). Analysis provided in SAR chapter 13 shows that adequate core cooling can be achieved with natural circulation when the core power is greater than 0.1 kW.

SAR table 4-4, "Typical NTR Core Thermal and Hydraulic Characteristics," shows thermal and hydraulic characteristics of the NTR at the rated power of 100 kW(t) and recirculation flow of 20 gallons per minute. The licensee noted that the average core exit temperature is 120°F (49°C), and that the hottest channel temperature is 150°F (65.6°C). Compared to the saturation temperature of 228°F (109°C), the coolant has a substantial margin to boiling at the designed operating condition.

SAR section 4.5 states that flow through the core is laminar, and that surface film heat transfer coefficients were calculated from a known laminar correlation. Heat transfer increases substantially when cladding surface temperatures reach a value that will support local boiling.

SAR section 4.5 states that power peaking factors used in the thermal-hydraulic evaluation were the maximum expected values resulting from operation of the reactor with neutron flux peaked on one side of the core. The total peaking factor used is 1.58, composed of a circumferential factor of 1.25, an axial factor of 1.15, and a local factor of 1.1.

SAR section 4.5 states that the thermal-hydraulic analyses show that the burnout heat flux in the hot channel is 227,000 Btu/(h-ft^2) (716 kW/m^2), and that the maximum heat flux for normal operating conditions is 10,300 Btu/(h-ft^2) (32.5 kW/m^2). While the ratio of these quantities provides a DNBR of 22, the licensee conservatively chose a DNBR of 1.5.

The NRC staff finds that the thermal-hydraulic design of the NTR provides significant margins at the steady-state operation power level of 100 kW(t). The hottest channel coolant temperature remains well below the saturation temperature, ensuring that reactor coolant voiding does not occur. The thermal-hydraulic analysis also provides significant margin to the DNBR of 22. The NRC staff also finds that the SL, LSSS, excess reactivity, and shutdown margin values chosen by the licensee, as described in SER section 2.5.3, "Operating Limits," demonstrate that no accident scenarios will result in a failure of fuel cladding or release of any fission products.

2.7 Reactor Description Conclusions

The NRC staff's review finds that the information pertaining to the design, construction, function, and operation of the NTR reactor fuel, neutron moderator/reflector, neutron startup source, core support structure, control and safety rods, reactor tank, and biological shield have not changed since the issuance of the previous license renewal by LA No. 21. On the basis of its review, the NRC staff concludes that the design of these core-related components for the NTR facility are acceptable and continue to permit safe operation and shutdown of the reactor.

The NRC staff reviewed the licensee's limits in proposed TS 2.1, TS 2.2, TS 3.1.1, and TS 3.1.3 and concludes that operation in accordance with the proposed TSs helps to ensure that any accident scenario will not result in fuel failure or the release of any radioactive fission products.

The NRC staff also reviewed the information provided in SAR section 4.5 regarding the thermal-hydraulic design of the NTR. The NRC staff concludes that the licensee has justified the assumptions and methods used. The thermal-hydraulic analysis provides reasonable assurance that the reactor can be operated at its licensed power level without undue risk to the health and safety of the public.

3 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 Radiation Protection

Safety analysis report (SAR) chapter 11, “Radiation Protection Program/Waste Management,” provides a description of how the activities involving radiation at the GE-Hitachi Nuclear Energy Americas LLC (GEH, the licensee) Nuclear Test Reactor (NTR, the facility) are controlled under a radiation protection (RP) program that meets the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for Protection Against Radiation,” and the guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.11, “Radiation Protection at Research Reactor Facilities.” Specifically, the licensee used the guidance in ANSI/ANS-15.11-1993, R2004 (Ref. 37). The regulations in 10 CFR 20.1101, “Radiation protection programs,” specify, in part, that each licensee shall develop, document, and implement an RP program and shall use, to the extent practical, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA). The basic aspects of the RP program include occupational and general public exposure limits, surveys and monitoring, and personnel dosimetry.

The U.S. Nuclear Regulatory Commission (NRC, the Commission) inspection program routinely reviews radiation protection and radioactive waste management at the NTR. The licensee stated that the NTR has operated under the Vallecitos Nuclear Center (VNC) RP program, which is under the responsibility of the regulatory compliance organization and the Radiation Safety Officer (RSO). The RSO is responsible for the ongoing implementation of the RP program and providing reports directly to the regulatory compliance organization. The RP program is reviewed annually by the RSO.

The NRC staff reviewed the NTR annual reports from calendar year (CY) 2014 through CY 2021 and the NRC inspection reports (IRs) from CY 2014 through CY 2022 regarding the RP program. A severity level IV violation was identified in CY 2016 and described in NRC IR 50-73/2016-202 (Ref. 21) for failure to follow RP procedures. Corrective actions planned and taken to correct the violation and prevent recurrence were adequately addressed during a subsequent inspection and documented in IR 50-073/2017-201 (Ref. 21). Otherwise, the NRC staff finds that the licensee’s RP program demonstrated that adequate measures are in place to minimize radiation exposure to personnel and to provide adequate protection against operational releases of radioactivity to the environment. Based on the following discussion, the NRC staff concludes that the RP program at the NTR continues to be acceptable.

3.1.1 Radiation Sources

Radiation sources at the NTR are described in SAR section 11.1.1 “Radiation Sources.” The NRC staff reviewed the descriptions of potential radiation sources, including the inventories of each physical form and their locations. The review of radiation sources included identification of potential radiation hazards as presented in SAR chapter 11, and verification that the hazards were accurately depicted and comprehensively identified. The primary radiation source directly related to reactor operation is thermal neutron activation of naturally occurring argon (Ar) in the air. Liquid sources are limited at the NTR facility and are typically limited to the primary coolant since no routine liquid effluent releases are planned at the facility.

3.1.1.1 Airborne Radiation Sources

SAR section 11.1.1.1, “Airborne Radioactive Sources,” states that during normal operations the primary airborne source of radiation is the thermal neutron activation of Ar-40 in the air to produce Ar-41. In addition, Ar-41 is generated in the cooling water and a small amount of airborne activity is generated from the contamination of the aluminum skin of the reactor fuel element during fabrication of the fuel elements. Ar-41 is the primary airborne radionuclide generated and tracked at the NTR. SAR section 11.1.1.1 provides airborne radiological effluent releases for CY 2018 as an example of the releases tracked at the NTR. The licensee stated that the total noble gases releases from the stack was 190 curies (Ci), the majority of which was Ar-41.

SAR section 11.2.4, “Radioactive Gaseous Waste Management,” states that the release of routine gaseous effluents is dominated by Ar-41. Airborne radioactive waste exiting through the NTR stack is monitored as a radioactive effluent to ensure that it is within the 10 CFR Part 20 requirements. The reactor cell contains any radioactive release while it is exhausted through the ventilation system and out the stack.

Occupational Dose

SAR section 11.1.1.3, “Solid Radioactive Sources,” states that, based on the five years of dosimetry for personnel at the NTR, the estimated maximum annual dose to a single worker is 862 milli-roentgen equivalent man (mrem), and the average dose to all workers is 443 mrem. No single worker at the NTR exceeded a total of 2.5 rem over this period, which is well within the limits in 10 CFR 20.1201, “Occupational dose limits for adults,” for occupational exposure.

The NRC staff’s review of the occupational dose provided in the NTR annual reports for CY 2014 through CY 2021 finds that the licensee’s control of radiation exposure to the workers is consistent with other research reactor occupational doses. The NRC staff’s review finds that the occupational doses are within the limits of 10 CFR 20.1201.

Public Dose

SAR section 11.2.5, “Stack Release Action Levels,” describes the licensee’s assessment and evaluation of the action levels and alarm limits which limit the airborne radioactive releases from the NTR.

The licensee added the alarm setpoints and limited the effluent monitoring to gaseous and particulate activity as described in its LRA supplement dated March 24, 2023. The license stated that the basis for this change was because the real-time NTR effluent monitors are only capable of detecting gaseous and particulate (beta, gamma) releases, making these the only stack release action levels that are actionable by the NTR operators.

Proposed technical specification (TS) 3.7.4, “Effluents – Stack Release Activity,” states:

> The stack discharge rates of gaseous and particulate activity *SHALL* not exceed the limits in Table 3-3, ensuring compliance with the 10 CFR 20.1101(d) limit of 10 mrem/year.

Table 3-3

STACK RELEASE ACTION LEVELS

	Gaseous Activity (Ar-41)	Particulate Activity (Beta)
Weekly release	9 Ci/wk	1.7+03 μCi/wk
Alarm setpoint	9.5E-05 μCi/cc	1.9E-08 μCi/cc

1. If the alarm setpoint is exceeded, then the operator SHALL determine the weekly release rate and take actions to ensure the weekly release rate action level is not exceeded.

2. If the weekly release rate is determined to have been exceeded, then the reactor SHALL be placed in SHUTDOWN until the condition can be evaluated and the release rates determined to be below action levels.

SAR section 11.2.5.1, “Basis for Stack Release Action Levels,” provides the licensee’s methodology for determining that the stack release rate action levels and limits will ensure that doses to members of the public due to airborne releases are at or below the 10 CFR 20.1101(d) limit of 10 mrem per year. Ar-41 has been shown to be the predominant noble gas in the stack effluent. The stack release action levels (weekly total or specific concentrations) for noble gas releases from the NTR stack ensure that the activity released will not exceed an annual average concentration of Ar-41 at the site boundary of 10 percent of the annual effluent concentration limit in 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” Table 2, “Effluent Concentration,” Column 1.

The NRC staff performed confirmatory calculations using the information and assumptions provided in SAR section 11.2.5 and the licensee’s supplemental information to verify the licensee’s stated actions levels in proposed TS 3.7.4, table 3-3. Further, the NRC staff finds that the addition of the alarm setpoints to proposed TS 3.7.4, table 3-3 provides a “real-time” notification to the reactor operators, and requires a determination of the weekly release rate, thus helping to ensure compliance with the annual dose limit of 10 mrem/year to any member of the public. The NRC staff finds that the stack action levels described in SAR section 11.2.5, and limited by proposed TS 3.7.4, table 3-3, are acceptable because these annual average releases correspond to a dose of less than 10 mrem/year which meets the 10 CFR 20.1101(d) limit of 10 mrem/year of radioactive emissions from the NTR.

The NRC staff calculated annual doses of 2 mrem/yr or less for each year using the reported releases of Ar-41 from the licensee’s annual reports from CY 2014 through CY 2021. Based on these dose results, the NRC staff finds that the annual exposure from Ar-41 releases to the unrestricted area is less than 10 mrem per year which is below the applicable limit of 10 mrem/year in 10 CFR 20.1101(d).

The NRC staff finds that the licensee’s airborne release doses are reasonable and satisfy the requirements of 10 CFR Part 20. Based on the information provided above, the NRC staff concludes that the licensee’s airborne dose estimates for the operation of the NTR are acceptable and that the production and release of Ar-41 in accordance with proposed TS 3.7.4 poses little risk to the health and safety of the public and to the NTR staff.

3.1.1.2 Liquid Radiation Sources

SAR section 11.1.1.2, “Liquid Radioactive Sources,” states that the only liquid radiation source for the NTR is from the primary coolant. The primary liquid radioactive material is Nitrogen (N)-16 and Sodium (Na)-24. The N-16 is produced from the neutron activation of oxygen during reactor operation and the Na-24 is produced from the activation of aluminum in the primary coolant piping during reactor operation. The primary coolant is regularly sampled to monitor for the potential of any fuel leakage into the primary coolant. The primary coolant system is vented into a holdup tank prior to reactor startup. The water vented into the holdup tank is small enough that the water in the holdup tank evaporates and the tank does not fill. The licensee performs dose rate measurements of the reactor holdup tank to ensure that no long-lived radionuclides accumulate in the tank. The total amount of liquid waste generated is from the primary coolant sampling, which is approximately one liter per sample. This sample waste is disposed of with the other laboratory waste from the facility.

The NRC staff reviewed the licensee’s annual reports from CY 2014 through CY 2021 and finds that no release of radioactivity in water or to groundwater greater than the limits specified in 10 CFR Part 20, Appendix B, Table 2, Column 2. The NRC staff’s review finds that the liquid radioactive sources generated from reactor operations are consistent with other research reactors. Based on the information described above, the NRC staff concludes that liquid radioactive sources from continued normal operation of the NTR are acceptably controlled, and do not pose a hazard to the health and safety of the public.

3.1.1.3 Solid Radiation Sources

SAR section 11.1.1.3 states the solid radiation sources used at the NTR. The solid radioactive materials consist of the reactor structure; reactor fuel; neutron sources used for pre-startup instrumentation checks; the ion exchange demineralizer and filter system for the primary coolant system; spent power reactor fuel rod sections; activated experiments and neutron-radiographed parts; byproduct material in experiments; check sources; and solid radioactive waste.

SAR section 11.2.3, “Radioactive Solid Waste Management,” states that the licensee generated one to three cubic feet of solid radioactive waste annually, with the activity being in the order of millicuries. The RP program is responsible for the control of solid radiation sources.

The NRC staff’s review of licensee annual reports from CY 2014 through CY 2021 and NRC IRs from CY 2014 through CY 2022 indicates that solid radioactive waste handling has not resulted in any violations or significant personnel exposures at the NTR. The NRC staff finds that solid radiation sources and radioactive wastes from the operation of the NTR are properly controlled, have not resulted in any significant personnel exposures, and can be handled safely.

Based on the above, the NRC staff concludes that the control of solid radioactive sources at the NTR is acceptable.

3.1.2 Radiation Protection Program

The regulation at 10 CFR 20.1101(a) requires that each licensee develop, document, and implement an RP program. The NRC inspection program routinely reviews the RP program at the NTR for compliance.

Program Controls, Organization, and Responsibilities

SAR section 11.1.2, "Radiation Protection Program," states that the RP program for the site, including the NTR, is the responsibility of the RC organization. The Manager, RC, has overall responsibility for the RP program. The RSO is responsible as the site radiation safety function leader for the ongoing implementation of the RP program and reports directly to the Manager, RC. The Manager, RC, may alternatively serve as the RSO if applicable radiation protection experience requirements are met. The staffing level for the RC organization is dependent on the level of activity at the site. Staffing for the RC activities for the NTR includes those necessary to perform health physics monitoring and nuclear safety oversight. The RP program at the NTR is implemented by the RSO. The NTR has a structured RP program, which is implemented by qualified health physics staff that is equipped with radiation detection capabilities to determine, control, and document occupational radiation exposures at the NTR.

Proposed TS 6.3, "Radiation Safety," defines the responsibilities of the RP program. Proposed TS 6.3 states:

> The Level 2 manager (or the Level 3 supervisor when assigned), in coordination with the VNC Radiation Safety Officer (RSO), *SHALL* be responsible for implementing the NTR radiation safety function. The RSO *SHALL* report relevant findings to the Level 2 manager, but SHALL report organizationally to the Manager, RC, thereby maintaining independence from the reactor operations organization. The radiation safety function is informed by the guidelines of the ANSI/ANS 15.11-2016, "Radiation Protection at Research Reactor Facilities."

The NRC staff's review finds proposed TS 6.3 acceptable as evaluated in SER section 5.6.3, "Radiation Safety."

Procedures

SAR section 11.1.2.2, "RP Program Implementation," states that the RP program at the NTR is a subset of the broader VNC site-wide RP program. Procedures implement the use of radiation work permits (RWPs) to ensure safe, authorized work in restricted areas. RWPs provide information and instruction to the worker and prescribe necessary precautions and protective equipment when performing tasks in those areas.

The following is a list of program areas covered by implementing procedures:

- Startup, operation, and shutdown of the reactor.
- Defueling, refueling, and fuel transfer operations, when required.
- Preventive or corrective maintenance that could have an effect on the safety of the reactor, including the replacement of components.
- Surveillance checks, tests, calibrations, and inspections required by the TSs.

- NTR-specific RP program implementing procedures for personnel safety consistent with applicable regulations or guidelines. Management commitment and programs to maintain exposures and releases ALARA are a component of the RP program.
- Administrative controls for operation and maintenance and the conduct of experiments that could affect reactor safety or core reactivity.
- NTR-specific implementing procedures for the site-wide emergency and security plans.
- NTR-specific RP program implementing procedures for the use, receipt, and on-site transfer of byproduct material for such activities performed under the NTR license.

SAR section 11.1.2.2 also states that the use of Change Authorizations and Engineering Releases for NTR activities provides documentation of changes and work and includes the determination of whether the proposed change requires prior NRC approval pursuant to 10 CFR 50.59. All procedures at VNC are administratively controlled to ensure that procedures and changes that impact the RP program are reviewed for adequacy, approved by authorized personnel, and distributed to the applicable staff.

Proposed TS 6.4.1, "Written Procedures," item (5) defines RP procedural and program responsibilities, stating that procedures shall be prepared for the following activities:

> NTR-specific radiation protection program implementing procedures for personnel safety consistent with applicable regulations or guidelines. Management commitment and programs to maintain exposures and releases as low as reasonably achievable *SHALL* be a component of the *SITE*-wide radiation protection program.

The NRC staff's review finds proposed TS 6.4.1, item (5) acceptable as evaluated in SER section 5.6.4, "Procedures."

Training

SAR section 11.1.2.3, "RP Training," states, in part, that radiation safety training is defined by procedure, managed by the RSO, and implemented by the responsible managers. Procedures describe radiation safety courses as well as how they are developed and maintained and delineate responsibility for ensuring that training is performed and documented. Radiation Monitoring Technicians (RMTs) are trained and certified in accordance with a comprehensive Health Physics program that covers all site operations, including those at the NTR. NTR personnel receive radiological safety training per the Reactor Operator Initial and Requalification programs. In addition, site radiation workers and personnel assigned to site response team receive annual radiation safety refresher training.

Audit Function

SAR section 11.1.2.4, "RP Program Oversight," states that the Vallecitos Technological Safety Council (VTSC) is the review committee associated with the activities of the site as a whole and the NTR. The RP program is reviewed each year pursuant to 10 CFR 20.1101(c), in a report to the site manager. The VTSC reviews the report annually and determines the effectiveness of the program. The VTSC also receives and reviews incident investigation reports and countable event reports and uses all the information to implement program improvement and to ensure that root causes are determined and effective corrective action is taken.

Proposed TS 6.2.3, "Review Function," item (4) defines RP program procedure review requirements, stating that activities requiring review shall include the following:

> All new procedures and major revisions of existing procedures having safety significance that are required by the administrative control specifications in Administrative Controls Section 6.4.

The NRC staff's review finds proposed TS 6.2.3, item (4) acceptable as evaluated in SER section 5.6.2.3, "Review Function."

The licensee stated that the RP program is reviewed each year pursuant to 10 CFR 20.1101(c). The licensee has an established committee to review activities of the entire site. The committee reviews and determines the effectiveness of the RP program and determines if any improvements need to be made to the program.

The NRC staff reviewed the NTR RP program, as described in the SAR chapter 11, and finds that the program complies with 10 CFR 20.1101(a), is implemented in an acceptable manner, and provides reasonable assurance that, for all facility activities, the program will provide adequate protection for the NTR staff, the environment, and the public. Based on the above, the NRC staff concludes that the NTR RP program is acceptable.

3.1.3 ALARA Program

The regulation at 10 CFR 20.1101(b) requires that licensees use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.

SAR section 11.1.3, "ALARA Program," states that the RP program at NTR includes a commitment to maintain radiation exposure ALARA. SAR section 11.1.3 also provides information on the structure of the NTR ALARA program. The VNC site manager has the overall responsibility for the NTR license, and the RC Manager is responsible for radiation safety, ensures that the NTR is fully committed to the ALARA principles, and oversees the effective implementation of the ALARA program on the site. During the first quarter of each year, the site manager reviews the ALARA program, exposure reports, and exposure records for all NTR personnel. The licensee submits the results of the environmental monitoring program to the NRC in its annual reports.

The NRC staff's review of occupational doses to workers provided in the licensee annual reports from CY 2014 through CY 2021 finds that the ALARA program appears effective for keeping occupational doses below the limits in 10 CFR 20.1201. The NRC inspection program routinely reviews the ALARA program and concludes that the program as implemented meets the requirements of the regulations.

The NRC staff's review finds that the NTR ALARA program provides reasonable assurance that radiation exposure will be maintained ALARA for all facility activities. Based on the above, the NRC staff concludes that the NTR ALARA program is acceptable.

3.1.4 Radiation Monitoring and Surveying

The regulation at 10 CFR 20.1501, “General,” states, in part:

(a) Each licensee shall make or cause to be made, surveys of areas, including the subsurface, that—

(1) May be necessary for the licensee to comply with the regulations in this part; and

(2) Are reasonable under the circumstances to evaluate—
(i) The magnitude and extent of radiation levels; and
(ii) Concentrations or quantities of residual radioactivity; and
(iii) The potential radiological hazards of the radiation levels and residual radioactivity detected.

The regulation at 10 CFR 20.1501(c) requires, in part, that the licensee ensure that instruments and equipment used for quantitative radiation measurements (e.g., dose rate and effluent monitoring) are calibrated periodically for the radiation measured.

SAR section 11.1.4, “Radiation Monitoring and Surveying,” states that the NTR radiation monitoring and surveying includes the following:

- Monitoring and surveying routines performed by an RMT.
- Special monitoring and surveying by an RMT.
- Fixed air sampling system.
- Stack monitoring system.
- Continuous air monitor (CAM).
- Area Radiation monitors.
- Smear surveys documented by NTR operations personnel.
- Personal dosimeters.
- Sampling and counting of industrial wastewater prior to release.

SAR section 11.1.4 also states that the health physics staff at the NTR performs radiation and contamination surveys on a regular basis. Records of surveys are maintained and reviewed by the regulatory compliance personnel. In addition, dosimetry and fixed air filter records are also maintained and reviewed by the regulatory compliance personnel. This includes stack monitoring records and CAM records.

SAR table 11-3, “Radiation Monitoring Equipment at the NTR,” provides a list of radiation monitoring and survey equipment available at the NTR. This table also includes information on the location and detailed functions of each monitor. The SAR also indicates that the NTR uses established procedures to perform instrument calibrations. The calibrations are performed at the site or by approved offsite vendors. The equipment calibration history is tracked by the licensee.

SAR section 11.1.4 states that radiation and contamination surveys are performed on a regular basis by the health physics staff at the NTR, which provides adequate oversight of areas where work with radioactive materials is performed.

The NRC staff's review of NRC IRs from CY 2014 through CY 2022 finds that the licensee's radiation monitoring and surveying provides the proper surveys and equipment for detecting the types and intensities of radiation likely to be encountered within the NTR and that the surveillance frequencies are appropriate to ensure compliance with 10 CFR Part 20 and the NTR ALARA program under all operating conditions. The IRs document that the placement, use, and control of the radiation monitoring and surveying equipment are in accordance with applicable standards, guidance, and regulations. The NRC staff also finds that the radiation monitoring and surveying is consistent with the guidance in NUREG-1537, Part 2, section 11.1.4, "Radiation Monitoring and Surveying," which states that the licensee should document the types of surveys performed and the associated areas of the facility.

Based on the above, the NRC staff concludes that the NTR radiation monitoring and surveying is acceptable.

3.1.5 Radiation Exposure Control and Dosimetry

SAR section 11.1.5, "Radiation Exposure Control and Dosimetry," states that radiation exposure control is achieved at the NTR by shielding, the ventilation system, security, entry control devices, an active ALARA program, the RP program, environmental monitoring, specific equipment and materials, and through the VNC dosimetry program. The NTR includes several design features used to limit radiation exposures to workers and the public.

SAR section 3.5, "Systems and Components," provides design features regarding the reactor cell ventilation system. SAR section 3.5.2, "Ventilation," provides information regarding the NTR ventilation and exhaust system. This system draws air from the reactor cell, south cell, and the north room through a prefilter and a bank of absolute filters before being discharged through the ventilation stack. The radiation air monitoring system provides information on the concentration of radioactive material in the ventilation and alarms if the concentration is more than the stack release action levels. The stack release action levels are developed based on Ar-41 effluent releases and are described in further detail in SAR sections 11.2.4 and 11.2.5. The sample is drawn from the discharge of the ventilation stack and passes through the particulate detector, a charcoal cartridge, and a nonfilterable radioactive gas detector. SAR figure 3-3, "Line Diagram of Ventilation System," provides the layout of the ventilation system and the flow path of the air monitoring system.

SAR section 11.1.5 states that the equipment and materials used in radiation exposure control consist primarily of protective cloth and respiratory equipment. The licensee's respiratory program is established to ensure compliance with 10 CFR Part 20 requirements. From the licensee's past radiation exposures, it is expected that the average radiation exposure to NTR personnel is 443 mrem per person. For non-NTR personnel, the licensee stated that its records indicate that the radiation exposure is below 100 mrem per year per person.

SAR section 11.1.5 states that procedures and other controls are the primary methods for limiting radiation exposures and the intake of radioactive materials during normal work in controlled areas. Exposure limits for occupational workers are maintained below the requirements of 10 CFR 20.1201 by use of administrative limits which are below the regulatory limits to ensure compliance. The facility procedures describe the exposure and dosimetry requirements for NTR personnel.

SAR section 11.1.5 states that the personal dosimetry used at the NTR includes beta-gamma dosimeters, neutron albedo dosimeters, and electronic dosimeters. Special use dosimeters such as thermoluminescent dosimeter (TLD) finger rings are issued for extremity exposure and high dose rate exposure according to the RP program. Air activity is controlled using the ventilation systems and contamination control. Intake of radioactive material is limited by using respiratory equipment when needed. The licensee stated that whole body counts are performed routinely to confirm the lack of intake of radioactive material. Based on the NRC staff's review of the annual exposure results recorded in the licensee annual reports from CY 2014 through CY 2021, the NRC staff confirmed that worker doses were keep below the regulatory limits. The average worker dose during this time was less than 534 mrem total effective dose equivalent (TEDE) per person per year with the greatest individual TEDE for these years being 876 mrem per year.

SAR section 11.1.5 states that survey meters are used to measure does rates from radiation fields and that these measured rates are posted where required. These provisions ensure that external and internal radiation monitoring of all individuals required to be monitored meet the requirements of 10 CFR Part 20 and the goals of the NTR ALARA program. The licensee stated that it also maintains personnel exposure records and effluent and environmental monitoring readings for the life of the facility.

The NRC staff reviewed the licensee annual reports from CY 2014 through 2021 and the NRC IRs from CY 2014 through 2022 and determined that the highest annual dose equivalent incurred by the NTR staff complies with the facility's ALARA program as well as the efficacy of the radiation exposure and control program. The NRC staff finds that all NTR staff received less radiation dose than the 10 CFR 20.1201 limits. The NRC staff finds the radiation doses to the NTR staff and the application of the equipment and procedures used to be acceptable. The personnel exposures at the NTR are controlled through satisfactory radiation protection and ALARA programs. Based on the above, the NRC staff concludes that the licensee's exposure control and dosimetry are acceptable.

3.1.6 Contamination Control

SAR section 11.1.6, "Contamination Control," states that radioactive contamination is controlled using the established ALARA program, written procedures, trained personnel, and a monitoring program designed to detect contamination in a timely manner. For work in areas where contamination is likely, the licensee has procedures established to maintain control over the contamination. Workers are trained in working with radioactive materials, including how to limit its spread when entering and exiting an area containing radioactive material.

The NRC staff reviewed the licensee annual reports from CY 2014 through 2021 and the NRC IRs from CY 2014 through 2022 and determined that the facility surveys have routinely shown no detectable contamination in non-radiological areas of the facility.

Based on its review of the RP program and on a history of satisfactory contamination control, the NRC staff concludes that adequate controls exist to prevent the spread of radiological contamination within the NTR.

3.1.7 Environmental Monitoring

SAR section 11.1.7, "Environmental Monitoring," states that the primary purpose of the environmental surveillance program is to obtain information essential to assessing and controlling the exposure of the neighboring population to industrial chemicals, radiation, and/or radioactive materials. Secondary objectives include identifying the sources of specific contaminants that might be released, predicting trends in pollutant levels, and improving public relations by showing that the operations at the VNC site are not adversely affecting the health and safety of the public and surrounding areas.

SAR section 11.1.7 states that the NTR environmental monitoring program consists of two categories: effluent monitoring and environmental surveillance. The effluent monitoring for the NTR is limited to only ventilation stacks. The environmental monitoring program is designed to measure the amount of radioactivity released into the environment. The environmental surveillance covers all measurements and observations made of the environment on and adjacent to the site. This includes the setup of environmental air samplers and TLDs and the sampling of water, vegetation, soil, and sediment. The surveillance ensures that there are no unmonitored impacts to the environment from effluent releases.

SAR section 11.1.7 states that a complete description of the current VNC site environmental program is contained in the VNC Environmental Monitoring Manual. The program is conducted to measure the integrated radiation exposure in and around the environs of the site. The environmental monitoring program surveys groundwater, stream sediments, vegetation, storm water, gamma monitoring locations, ambient air monitoring, and gaseous effluent monitoring.

SAR section 11.2.2, “Radioactive Liquid Waste Management,” states that no radioactive liquid waste is released from the NTR directly to the unrestricted environment. The only airborne radionuclide with potential to be released to the environment is Ar-41. In SAR sections 11.2.4 and 11.2.5, the licensee stated that it has established stack action levels to determine if Ar-41 concentrations would result in a dose of greater than 10 mrem/yr. If these action levels are reached, then the licensee will take corrective actions to reduce the gaseous effluent releases. The licensee has established onsite 4 air monitoring stations and 20 locations for measuring gamma with a dosimeter. The licensee also obtains water samples to ensure that there are no releases into water pathways. The water samples are analyzed for gross alpha, beta, and tritium. Soil and vegetation samples are analyzed for gross beta and undergo gamma spectroscopy. These samples and dosimetry are analyzed and documented in the licensee annual effluent and environmental reports.

SAR 11.1.7 also states that RC is responsible for reviewing the environmental protection program for adequacy and for recommending changes as necessary. Further, RC prescribes equipment in support of the environmental protection program and shall review periodically the activities of the specialist assigned to environmental protection.

Further, proposed TS 6.7.1, item (5) requires that the licensee include in its annual reports to the NRC the following:

> A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge. The summary *SHALL* include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is <25% of the concentration allowed or recommended, a statement to this effect is sufficient.

The NRC staff reviewed the NTR environmental monitoring program and the results of the CY 2020 effluent and environmental monitoring report to understand the specific details of the established program and finds that the reporting criteria are sufficient to understand the radioactivity released from the facility. Based on the above, the NRC staff concludes that the environmental monitoring program is adequate to assess the radiological impact of the operation of the NTR on the environment.

3.2 Radioactive Waste Management

SAR section 11.2, “Radioactive Waste Management,” states that the purpose of the radioactive waste management program is to minimize radioactive waste and ensure its proper handling, storage, and disposal. The NTR primarily generates radioactive waste in solid and gaseous form.

3.2.1 Radioactive Waste Management Program

SAR section 11.2.1, “Radioactive Waste Management Program,” states that all radioactive waste handling operations at the VNC site are controlled by procedures. The responsibilities for carrying out the program for the NTR are divided among the Area Manager, RC, the NTR Manager; and those individuals performing radioactive waste activities.

SAR section 11.2.4 states that gaseous effluents, such as Ar-41, are monitored and discharged through the facility stack. Liquid wastes are converted to solid form for disposal using an evaporator. Solid wastes, such as routine laboratory wastes, are properly packaged and stored until final disposition. All radioactive waste handling operations are controlled by procedure to ensure compliance with the requirements of 10 CFR Part 20 and other appropriate NRC regulations. Records associated with radioactive waste activities are maintained by the area managers.

The NRC staff reviewed the licensee radioactive waste release practices and finds that these practices demonstrate reasonable assurance that radiological releases from the NTR will not exceed applicable regulatory limits or pose unacceptable radiation risk to the environment and the public, consistent with the guidance in NUREG-1537, Part 2, section 11.2.1, “Radioactive Waste Management Program.” The NRC staff also finds that the licensee has adequate controls in place to prevent uncontrolled personnel exposures from radioactive waste operations and to provide the necessary accountability to prevent any potential unauthorized release of radioactive waste. Based on the above, the NRC staff concludes that the licensee’s radioactive waste management program is acceptable.

3.2.2 Radioactive Waste Control

SAR section 11.2.4 states that gaseous waste generated at the NTR is considered as an effluent release. The primary radionuclide of focus is Ar-41, which is monitored by the plant stack for effluent releases. The licensee has established stack release action levels to ensure that releases do not exceed the 10 CFR 20.1101(d) limit of 10 mrem/yr in proposed TS 3.7.4.

The NRC staff reviewed the stack release action levels in proposed TS 3.7.4 and finds that compliance with the proposed TS 3.7.4 stack release limits will ensure that releases will not exceed the 10 mrem/yr dose limit in 10 CFR 20.1101(d), as evaluated in SER section 5.3.7.4, “Effluents – Stack Release Activity.”

SAR section 11.2.2 states that liquid waste generated at the NTR is only being generated because of the annual sampling performed each year. This small amount of liquid waste is placed in tanks with other laboratory waste and disposed of according to procedures and processed in the waste evaporator. Secondary and industrial wastewater at the facility is sampled prior to release to the environment to ensure that there is no cross contamination or releases of radioactive water to the environment.

SAR section 11.2.3 states that the primary forms of solid radioactive waste are contaminated paper and plastic, filters, and resins. The licensee estimated that the NTR generates one to three cubic feet of solid radioactive waste annually, with the activity being on the order of millicuries. The licensee ensures that all shipments of radioactive waste are characterized, handled, packaged, surveyed, and shipped in accordance with all applicable U.S. Department of Transportation and NRC regulations.

Based on the above, the NRC staff finds that acceptable procedures are in place to monitor the radiation exposure from radioactive waste and to perform required handling operations, consistent with the guidance in NUREG-1537, Part 2, section 11.2.2, “Radioactive Waste Controls.” Furthermore, the NRC staff concludes that the NTR has adequate radioactive waste controls in place to monitor the radiation exposure from radioactive waste, to perform required handling operations, to minimize the amount of solid waste generated, and to prepare the material for transfer to offsite disposal.

3.2.3 Release of Liquid Radioactive Waste

SAR section 11.2.2 states that the only liquid radioactive waste generated is as a result of the annual sampling, totaling approximately one liter. This waste is placed in tanks with other laboratory-generated liquid radioactive waste and subsequently disposed of in accordance with approved practices and procedures. No liquid radioactive waste is released directly to the unrestricted environment. Contaminated wastewaters created from NTR operation are processed in the waste evaporator. Evaporator bottoms are then processed and shipped as radioactive solid wastes. Industrial wastewater from the NTR single pass, non-contact, secondary cooling water heat exchanger is tested for radiological constituents as well as other potentially polluting constituents in accordance with a National Pollutant Discharge Elimination System permit prior to release to the environment.

The NRC staff finds that the licensee has described the radioactive effluents expected to be release, consistent with the guidance in NUREG-1537, Part 2, section 11.2.3, “Release of Radioactive Waste.” Further, proposed TS 6.7.1, item (5) requires that the licensee include in its annual reports to the NRC a summary of the nature and amount of radioactive effluents released or discharged to the environment beyond the control of the licensee, which includes the liquid, gaseous, and solid waste released from the NTR.

The NRC staff finds that the licensee has provided an effective means of describing the release of liquid radioactive waste from the NTR, consistent with the guidance in NUREG-1537, Part 2, section 11.2.3. The NRC staff review of the licensee annual reports from CY 2014 through CY 2021 confirmed that this information was included and is being tracked. As stated previously, the licensee generates liquid radioactive waste on the order of one liter each time the primary coolant is sampled annually and this waste is subsequently disposed of as laboratory wastes. Proposed TS 6.7.1 specifies that the licensee annually report to the NRC regarding effluent releases from the NTR. The NRC staff reviewed the licensee annual reports from CY 2014 through CY 2021 related to radioactive waste releases and finds that the reports summarize radioactive effluents and waste. The NRC staff finds that the reporting requirement and the results documented in the licensee annual reports are acceptable.

Based on the above, the NRC staff concludes that the licensee's controls and techniques for the release of radioactive liquid waste are acceptable. Furthermore, the NRC staff concludes that the NTR has adequate controls in place to minimize releases of radioactive material to the environment.

3.3 Conclusions for Radiation Protection

Based on its review of the SAR, as supplemented, licensee annual reports, and the results of the NRC inspection program, the NRC staff concludes the following:

- The licensee's RP program is a site-wide program that complies with the requirements in 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the NTR staff, the public, and the environment are protected from unacceptable radiation exposures. The radiation protection staff has adequate lines of authority and communication to implement the program.

- The licensee's ALARA program complies with the requirements of 10 CFR 20.1101(b). Review of controls for radioactive material at the NTR provides reasonable assurance that radiation doses to the NTR staff, the public, and the environment will be ALARA.

- The results of radiation surveys carried out at the NTR, doses to the persons issued dosimetry, and the results of the environmental monitoring program confirm that the RP and ALARA programs are effective and in compliance with the requirements of 10 CFR 20.1501(a).

- Potential radiation sources have been adequately identified and described by the licensee and the licensee sufficiently controls radiation sources.

- Facility design and procedures control the potential exposures of Ar-41 to the NTR staff, the public, and the environment to acceptable levels. Review of licensee annual reports as well as NRC IRs confirms that the quantities of these gases released into restricted and unrestricted areas provide reasonable assurance that doses to the NTR staff and the public will be below applicable 10 CFR Part 20 limits.

- The facility radioactive waste management program provides reasonable assurance that radioactive waste released from the facility will neither exceed applicable regulatory limits or pose an unacceptable radiation risk to the public and the environment.

4 Accident Analyses

4.1 Accident Analyses

The accident analyses presented in safety analysis report (SAR) chapter 13, "Accident Analysis," help to establish safety limits and limiting safety system settings that are imposed on the GE-Hitachi Nuclear Energy Americas LLC (GEH, the licensee) Nuclear Test Reactor (NTR, the facility) through the implementation of technical specifications (TSs).

The SAR provides the licensee's analyzed potential reactor transients and other hypothetical accidents. None of the credible accidents postulated would lead to the failure of the cladding of any fuel assembly or the uncontrolled release of fission products. However, the licensee postulated an enveloping event involving a failure in a fueled experiment. This event would lead to the maximum potential radiation hazard to personnel and members of the public, which is considered to bound the potential effects of natural hazards as well as potential credible design basis accidents (DBAs) involving the operation of the NTR. The licensee makes no assumptions as to the cause of the event. The licensee evaluated only the potential consequences of this event and not the likelihood or mechanisms for its occurrence.

The U.S. Nuclear Regulatory Commission (NRC, the Commission) staff finds the failure in a fueled experiment event to be the maximum hypothetical accident (MHA) for the NTR. In addition, because the initiation and planned operation of the confinement system are relied upon to mitigate the consequences of the event, this event may also be considered an experiment DBA. The NRC staff reviewed the licensee's analytical assumptions, methods, and results. In addition, the NRC staff performed confirmatory calculations and compared the results of those calculations with the accidents analyzed by the licensee. As discussed below, none of the potential accidents considered in the SAR would lead to significant occupational or public radiation exposure.

4.2 Accident-Initiating Events and Scenarios

The licensee and the NRC staff considered the MHA and, in addition, considered the following types of potential initiating events:

- MHA (Experiment DBA)
- Loss of normal electrical power
- Loss of facility air supply
- Loss of secondary coolant
- Inadvertent core inlet temperature change
- Fuel handling errors
- External events
- Mishandling/malfunction of equipment
- Uncontrolled reactivity increases
- Rod withdrawal accidents
- Reactor loss of flow accident
- Reactor loss-of-coolant accident
- Consequences of accidental explosions

4.2.1 Maximum Hypothetical Accident

In SAR section 13.2, “Experiment Design Basis Accident,” the licensee analyzed the postulated failure of a fueled experiment in the reactor. This analysis demonstrates the capability of the NTR and the site to accommodate a radioactive material release with no credit taken for filtration of the release by the NTR stack filter system. In SAR section 13.2.2, “Accident Description,” the licensee established limits on the maximum mass of uranium (U)-235 an experiment can contain and the maximum fluence for the experiment. The limits on these parameters were chosen so that the failure would result in acceptable doses in restricted and unrestricted areas. Based on the demonstrated calculation technique, the licensee has proposed TSs on the radioactive material content, including fission products, of all experiments. The limits are dependent on the physical form of the material and the performance of the confinement system at the NTR.

SAR section 13.2 states that the licensee has developed the material quantity limits, clad requirements, operating limits, and required safety equipment for irradiation experiments involving single and double clad fueled capsules at the NTR. This development was based on the radiological criteria in Regulatory Guide (RG) 2.2, “Development of Technical Specifications for Experiments in Research Reactors” (Ref. 39), and the capabilities of the NTR and the site to accommodate a radioactive material release from the experiment. The limits are dependent on the physical form of the material and the performance of the confinement system at the NTR.

<u>MHA Scenario</u>

SAR section 13.2.2 states that per RG 2.2, C.2.a.(3), the release event consists of “a single-mode nonviolent failure of the encapsulation boundary that releases all radioactive material into the immediate environment of the experiment or to the reactor building, as appropriate.” In addition, RG 2.2 states that “[t]he analysis should establish the most probable trajectory of the material, if any, into restricted and unrestricted areas. Credit for natural consequence-limiting features such as solubility, absorption, and dilution and for installed features such as filters may be taken provided each such feature is specifically identified and conservatively justified by specific test or physical data or well-established physical mechanisms.”

The licensee selected as the MHA the fueled experiment accident using 50 milligrams (mg) of an irradiated U-235 powder in a singly encapsulated container. This is a DBA that could result in fuel degradation with subsequent fission product release into the environment for the NTR.

SAR section 13.2.2 describes the accident as follows:

1. Experiment material is 50 mg of U-235 powder in a singly encapsulated container. Dose consequences for doubly cladded or pellet forms of U-235 equal to or less than 50 mg are bounded by the results of this analysis.

2. The most probable trajectory of the released material is from the experiment location to the reactor cell area. Since the event is a single-mode nonviolent failure, the established conditions would presumedly include the ventilation system being in operation; however, ventilation (filtration) is conservatively not credited in the analysis.

3. The release fractions of U-235 fuel and fission products to the environment are assigned as follows:

Release from capsule to reactor cell:	Powder/Pellet %
U-235	100
Noble Gas	100
Iodine	100
All Remaining Fission Products	100
Release from reactor cell to the environment:	
U-235	100
Noble Gas	100
Iodine	100
All Remaining Fission Products	100

4. Radiological consequences are computed for the TEDE. The licensee chose to adopt an accident dose criterion of 0.1 rem TEDE to represent an acceptable figure-of-merit over a 2-hour period at the site fence-post to a member of the public. In addition, the licensee included an additional accident dose criterion of 0.5 rem TEDE to represent an acceptable figure-of-merit within the restricted area of the reactor cell over a 5-minutes evacuation time to an operator.

5. The unrestricted area at the site fence-post exposure will result from the diluted-dispersed cloud of isotopes released from the NTR stack, which reaches the nearest site boundary under type F meteorological conditions at 1 m/sec over a 2-hour period.

6. The restricted area of the reactor cell exposure will result from the submersion in and inhalation of the isotopes released to the reactor cell for a period of 5 minutes during evacuation.

7. It is assumed that a complete release of the experiment capsule contents to the restricted area will occur uniformly over the two hours following the experiment failure.

8. The fission products from this release will cause high-activity alarms on the stack monitors.

9. The NTR operator will respond to the stack alarms and announce an area evacuation over the building public address system.

10. Evacuation to an upwind location will remove personnel from the stack concentration of released isotopes. Onsite exposures can be controlled by use of the alarm system and evacuation procedures.

11. Assuming the operator is in the reactor cell where the accident occurs for the duration of the evacuation ensures that the operator dose for this event bounds the operator doses that would be received at all other locations inside or adjacent to the NTR for a 5-minute evacuation.

The NRC staff finds the MHA scenario, as described, acceptable because it is consistent with the guidance in NUREG-1537, Part 1, section 13.1.1, “Maximum Hypothetical Accident,” which states that the accident could involve the escape of fission products from a fueled experiment. The NRC staff also finds that the assumption used for the source term is limited by proposed TS 3.8.11, “Fissile Material Experimental Limitations,” to 50 mg of U-235. Finally, the NRC staff finds that the assumptions used in the MHA analysis, including that 100 percent of the fission products are released to the atmosphere, provide a maximum dose rate calculation and result in a conservative MHA analysis. Based on the above, the NRC staff finds the MHA scenario acceptable.

Calculation Methods - Source term

SAR section 13.2.3, “Calculation Method,” includes the calculation of the bounding fission product inventories to produce an MHA source term which constitutes the NTR’s safety basis. The MHA source term is intended to be used in the bounding radiological consequence analyses. To derive the MHA source term, the licensee utilized Version 2.1 of the Oak Ridge National Laboratory Isotope GENeration (ORIGEN) code to calculate the fission products created during the irradiation of the U-235 capsule. ORIGEN Version 2.1 simulates nuclide fission, transmutation, and decay as a function of operational characteristics and fuel burnup.

The MHA source term is calculated by modeling the irradiation of a capsule containing 50 mg of U-235 at a capsule operating power level of 60 watts (W) for a period of 1 day. The results are presented in SAR chapter 13, table 13-1, “NTR Experiment DBA Isotopic Release to Reactor Cell,” which includes radionuclides that have typically been found by the NRC staff to be important contributors to dose for the purposes of performing design-basis radiological consequence analyses. In addition to these 55 radionuclides, the remaining unburned U-235 is added to the MHA source term comprising the 56 radionuclides in SAR table 13-1. Further, the licensee indicated that while these isotopes are known to be the main dose contributors for releases involving irradiated U-235, there may be a contribution from the remaining isotopes not being modeled. To account for this potential contribution, a safety factor of 20 percent is added to the final dose consequences.

The NRC staff finds the ORIGEN Version 2.1 computer code to be acceptable for the purposes of developing radionuclide inventories to derive a bounding MHA source term. The NRC staff also finds that the licensee appropriately utilized this code to derive its safety basis for the MHA source term.

Calculation Methods - Radiological Consequences

SAR section 13.2.3 provides a description of the radiological consequences using Version 3.10 of the RADionuclide, Transport, Removal, and Dose Estimation (RADTRAD) computer code to calculate the TEDE dose resulting from exposure to the released radioactive materials. The licensee’s RADTRAD model applied the MHA source term with both siting- and design-specific inputs required for the evaluation of radiological consequences.

The NRC staff finds the RADTRAD Version 3.10 computer code to be acceptable for the purposes of computing radiological consequences. The NRC staff also finds the licensees’ siting- and design-specific input assumptions to be conservative due to their bounding nature and, therefore, acceptable. Finally, the NRC staff finds that the licensee appropriately utilized this code to compute radiological consequences.

Results

SAR section 13.2.4, “Results,” provides the results of the RADTRAD calculations to determine the MHA radiological consequences at the unrestricted area at the site boundary and within the restricted area within the reactor cell. The results are presented in SAR table 13-2, “NTR Experiment Design Basis Accident Doses,” and reproduced in table 4-1, “MHA Doses,” below:

Table 4-1 MHA Doses

Exposure Location and Duration	Dose Consequences, TEDE	Dose Limits, TEDE
Unrestricted Area Boundary with 2-hour Exposure	66.2 mrem	100 mrem
Restricted Area (NTR Reactor Cell) with a 5-minute Exposure	470 mrem	5,000 mrem

The NRC staff finds that these results are within the acceptable siting criteria limit of 0.1 rem TEDE and the acceptable control room operator limit of 5 rem TEDE. Due to the assumptions of this scenario being bounding, the doses calculated will likely not be exceeded by any accident considered credible. As a result of these analyses, the licensee has established limits in proposed TS 3.8, “Experiments,” concerning the radioactive material content for reactor experiments. Operation of the reactor within the limits of proposed TS 3.8 will not result in radiation exposures in excess of 10 CFR 20.1201 or 10 CFR 20.1301, “Dose limits for individual members of the public.” Operation within proposed TS 3.8 will also limit the likelihood and consequences of malfunctions and ensures the health and safety of the onsite personnel and the public, and protection of the environment.

The NRC staff concludes that the computational approach and the projected quantities and fluences for U-235 are acceptable for this MHA. In addition, the NRC staff concludes that the licensee could analyze similar experiments containing radioactive materials and fission products other than U-235 to ensure compliance with proposed TS 3.8. Thus, even for the MHA, whose consequences bound all fission product-based credible accidents possible at the NTR, the health and safety of the NTR staff and the public are protected.

4.2.2 Loss of Normal Electrical Power

SAR section 13.4.1, “Loss of Normal Electrical Power,” describes the licensee’s analysis of a total loss of normal electrical power at the NTR, which results in no unacceptable consequences.

The NRC staff reviewed the analysis of a loss of normal electrical power at the NTR and finds the analysis and its results to be acceptable. Therefore, the NRC staff concludes that a loss of normal electrical power will not lead to unacceptable results and will lead to the safe shutdown of the NTR.

4.2.3 Loss of Facility Air Supply

SAR section 13.4.2, “Loss of Facility Air Supply,” states that the NTR compressed air supply is used to operate the air piston for the south cell door and the radiation shield shutter for the horizontal facility in the south cell. One person can manually move the south cell door. The shutter would remain in the position that it was in at the time of the air supply failure. The licensee has analyzed the consequences of a loss of facility air supply and has found no unacceptable results.

The NRC staff has reviewed the analysis of a loss of facility air supply and finds the analysis and its results to be acceptable. Therefore, the NRC staff concludes that a loss of facility air supply will have minimal effect on the NTR.

4.2.4 Loss of Secondary Coolant

SAR section 13.4.3, "Loss of Secondary Coolant," states that secondary coolant at the NTR flows by gravity through the tube side of the primary heat exchanger. Loss of secondary cooling when the reactor power level is high enough to produce an appreciable heating rate will cause the reactor to scram from high primary coolant temperature. If the heating rate is not high enough to cause a scram quickly, the loss of secondary coolant will be evident to the operator from control room indicators including a reactivity change. The licensee has analyzed the consequences of a loss of secondary cooling and has found no unacceptable results.

The NRC staff reviewed the analysis of a loss of secondary coolant and finds the analysis and its results to be acceptable. Therefore, the NRC staff concludes that a loss of secondary coolant will have minimal effect on the reactor and will lead to a reactor shutdown by its effect on the primary coolant temperature or by operator action. Also, it will not lead to fuel temperatures that would cause loss of integrity of the fuel cladding.

4.2.5 Inadvertent Core Inlet Temperature Change

SAR section 13.4.4, "Inadvertent Core Inlet Temperature Change," states that if the primary pump were inadvertently started, the effect would be to change the reactor inlet temperature. A decrease in the temperature will cause the reactor power to drop if it is below 124 degrees Fahrenheit (°F) (51 degrees Celsius (°C) because of the positive temperature coefficient. For this same reason, an increase in the temperature will cause the reactor power level to rise until 124°F (51°C) is reached. The amount of positive reactivity that could be added is less than $0.10 from room temperature to turnover temperature. After this temperature, the power level will begin to drop. The licensee has demonstrated that a step insertion of $0.76 of reactivity would not cause fuel damage even if the reactor failed to scram on high power level. Therefore, the licensee concludes that a transient caused by a small amount of reactivity (<$0.10) from the temperature increase would also not cause fuel damage. The licensee has analyzed the consequences of an inadvertent core inlet temperature change (i.e., start of the primary pump) and has found no unacceptable results.

The NRC staff has reviewed the analysis of an inadvertent core inlet temperature change and finds the analysis and its results to be acceptable. Therefore, the NRC staff concludes that this transient is bound by the reactivity insertion accident and will not lead to fuel temperatures that would cause loss of integrity of the fuel cladding.

4.2.6 Fuel Handling Errors

SAR section 13.4.5, "Fuel Handling Errors," analyzes fuel handling errors, and states that refueling for reactivity increase is not necessary and that fuel handling is very rare. The most recent fuel handling occurrence was in support of core container replacement in 1976. There are 16 fuel assemblies that completely fill the core reel assembly and are used for operation. The only other available space for a fuel assembly is in the core-loading chute. An element in the loading chute results in a less reactive core configuration than the cylinder formed by having all elements in the core support reel. Dropping a fuel element could only cause an accident if the control and safety rods were withdrawn during loading so that the reactor was almost critical before the element fell into the reel assembly. Such a condition is contrary to operating procedures and would require errors by the console operator and fuel loaders. In addition to having all existing fuel assemblies in their most reactive configuration in the core, additional safety features ensure safety during all phases of fuel handling. These are: reactor design, fuel handling equipment, and administrative controls that are such that not more than two assemblies can be handled at once.

SAR section 13.4.5 also states that all fuel movement must be performed in accordance with written procedures. The cell high-gamma-level alarm system will be in operation. By using the manually positioned poison sheets, the core can be made subcritical by $6.1. Removal of the graphite plug from the fuel loading chute provides negative reactivity of about 1.25 percent. Movement of the source and special nuclear material within the facility must have the approval of the licensed operator on duty. Any storage arrangements used will be analyzed to ensure a subcritical configuration. The licensee concluded that a fuel handling error resulting in a large increase in reactivity is unlikely because of administrative procedures, operating procedures, and physical characteristics of the NTR.

The NRC staff has reviewed the analysis of fuel handling errors and finds the analysis and its results to be acceptable. Therefore, the NRC staff concludes that a fuel handling error would not lead to fuel temperatures that would cause loss of integrity of the cladding and that the health and safety of the public would not be endangered by a fuel handling error.

4.2.7 External Events

SAR section 13.4.6, “External Events,” states that the only credible external event for the NTR is a seismic event, which is the initiating event for the reactivity insertion accident in SAR section 13.5.1, “Idealized Step Reactivity Insertions-with Scram.” The licensee stated that because fuel damage does not occur in this scenario, there is no release of radioactive products and no associated dose consequences.

SAR section 13.5.1 is evaluated in SER section 4.2.9, “Uncontrolled Reactivity Increases,” and found acceptable. Given that the only credible event for the external events accident scenario was a seismic event and that the reactivity insertion accident caused by a seismic event was evaluated and found acceptable (i.e., no fuel damage or release of any radioactive fission products), the NRC staff concludes that the external events accident scenario is acceptable.

4.2.8 Mishandling/Malfunction of Equipment

SAR section 13.4.7, “Mishandling/Malfunction of Equipment,” states that mishandling of equipment is precluded by the NTR TSs and operating procedures. Further, the analyses in SAR section 13.4, “Anticipated Operational Occurrences,” address all of the events that could result from improper operation, and none of these events result in fuel damage, release of radioactive products, or dose consequences.

The NRC staff’s review of the accident scenarios (excluding the MHA because it involves an experiment DBA) finds that fuel failure does not result as a consequence of any non-MHA accident. Therefore, the NRC staff concludes that the mishandling/malfunction of equipment accident scenario is acceptable.

4.2.9 Uncontrolled Reactivity Increases

In SAR sections 13.5.1 and 13.5.2, the licensee analyzed a number of transients resulting from reactivity insertions. The first is an idealized step reactivity increase resulting from step reactivity insertions up to $1.4 with a range of different initial reactor power levels and primary coolant flow rates used. The second is a large reactivity insertion over a short period of time. Reactivity insertions of $2 and $4, with durations from 0.2 to 0.6 seconds, were analyzed using a range of initial reactor power levels and primary coolant flow rates.

The NRC staff concludes that if the reactivity addition caused by control rod and experiment movement is sufficiently large, the resulting power excursion, if not terminated by a scram, could result in fuel melting. Therefore, the licensee will operate the NTR in a way that limits the potential excess reactivity to less than that required to cause fuel damage, assuming failure to scram. Accordingly, proposed TS 3.1.1, "Potential Excess Reactivity," states:

> *POTENTIAL EXCESS REACTIVITY SHALL* be ≤ $0.76. If it is determined to be > $0.76, the reactor *SHALL* be placed in *REACTOR SHUTDOWN* immediately.

SAR 13.5.3 discusses the reactivity insertion without a scram accident scenario. The licensee hypothesized that certain structures (used to support the control and safety rod mechanisms as well as experiments) might fail or move during a seismic event in such a manner as to withdraw the control rods and experiments from the core region and prevent operation of the safety rods. If the reactivity addition caused by control rod and experiment movement is sufficiently large, a power excursion not terminated by a scram could occur and result in fuel melting.

Proposed TS 5.3.1, "Control System," item (3) provides a requirement to help ensure that the MPS do not move, stating:

> Each installed *MANUAL POISON SHEET SHALL* be restrained in its respective graphite reflector slot in a manner which will prevent movement by more than ½ inch relative to the reactor core.

In SAR section 13.5.3, the licensee calculated that for a $0.76 reactivity insertion during operation at 100 kW(t) with forced cooling, the reactor power level peaks at 4,000 kW and the fuel temperature peaks at 373°F (190°C), assuming failure to scram. The licensee also calculated the results of a $0.76 reactivity insertion during operation at source level with an initial inlet water temperature as low as 55°F (13°C) and a failure to scram. For these assumed conditions, the reactor power level peaks slightly above 10,000 kW and the fuel temperature peaks around 1,150°F (622°C). For these calculations, the reactivity addition from the initially positive temperature coefficient was included. As can be seen from the results, limiting the step reactivity insertion to $0.76 or less ensures that there are no mechanisms available that will cause fuel damage. SAR section 13.4.3 states that these transient calculations are extremely conservative since no credit is taken for the negative reactivity feedback from subcooled voids during nucleate boiling.

The NRC staff finds that proposed TS 3.1.1 limits the potential excess reactivity for all reactor core configurations to $0.76. Additionally, the failure of any MPS is not considered credible. The NRC staff finds that proposed TS 5.3.1, item (3) helps to ensure that the MPS are securely held in place and will not move and result in a reactivity insertion event during a seismic event. The NRC staff reviewed the uncontrolled reactivity increase accident scenario and finds the methods, results, and analysis to be acceptable. Therefore, the NRC staff concludes that by limiting the potential excess reactivity of the reactor to $0.76, a power excursion will not lead to fuel temperatures that would cause loss of integrity of the cladding and that the health and safety of the public would not be endangered by such an event.

4.2.10 Rod Withdrawal Accidents

SAR section 13.5.4, "Rod Withdrawal Accidents," analyzes the transient caused by the simultaneous withdrawal of all control rods. The licensee stated that the safety system and rod withdrawal procedures are designed to provide adequate control of the reactor at all times. Also, even if the interlocks fail and the operator deviates from normal procedures so that the rate of power increase is not controlled by normal manual control rod movements, the reactor period and neutron flux level monitors would scram the reactor. If the reactor did not scram, the analysis in SAR section 13.5.3 is applicable. The licensee stated that the transient analysis demonstrates that the reactivity can be introduced in either a step or a relatively long ramp without affecting the outcome. The licensee stated that the analysis indicates that the transient that results from the total reactivity addition of the control rods, experiments, and temperature effect without scram (and the potential excess reactivity is less than or equal to $0.76) does not result in fuel failure (i.e., melt). As such, the transient that would be caused by the withdrawal of all of the rods is bounded by the analysis in SAR section 13.5.3.

The NRC staff finds that proposed TS 3.1.1 limits the potential excess reactivity for all reactor core configurations to $0.76 and that the rod withdrawal accident scenario is bounded by the results of the uncontrolled reactivity increase accident scenario and, therefore, is acceptable.

4.2.11 Reactor Loss of Flow Accident

SAR section 13.5.5, "Reactor Loss of Flow Accident," analyzes the sudden loss of primary coolant flow by assuming that the worst loss of flow accident (i.e., instantaneous seizure of the rotor in the single recirculation pump in the system) occurs. The licensee specified that the following initial conditions were assumed:

- The reactor is operating at full power.
- The pump flow will coast down to natural convection in 0.1 seconds.
- The low flow scram fails.
- The initial core average coolant temperature is 106.2°F (41.2°C).
- The initial core excessive reactivity is 0.

SAR section 13.5.5 states that based on the initially positive temperature coefficient, the excess reactivity increases because of the temperature increase. Reactor power level will rise, but will begin to slow as the temperature coefficient goes negative. The final steady-state operating point will correspond to a power and flow combination that gives the same reactivity contribution from temperature as for initial steady state operation. This final coolant temperature is 151°F (66°C). Thus, there is no bulk boiling in the average channel. The licensee concluded that the maximum fuel temperature during the transient is 193°F (90°C) at 54 seconds and then decreases.

The NRC staff reviewed the methods and assumptions of the licensee's analysis and finds that the resulting calculated temperatures are conservative compared to those reasonably expected as a consequence of any loss of coolant flow accident. Therefore, the NRC staff concludes that there is reasonable assurance that such an event would not lead to a loss of integrity of fuel or release of fission products.

4.2.12 Reactor Loss-of-Coolant Accident

SAR section 13.5.6, "Reactor Loss-of-Coolant Accident," analyzes the postulated loss of all coolant from the core as a result of a rupture in the primary system. The licensee specified that the following initial conditions were assumed:

- The reactor is at the licensed thermal operating power of 100 kW(t).
- The primary system ruptures at some point below the core entrance so that gross removal of the core coolant supply occurs.
- All scrams fail.
- Uncovering of the fuel acts to shut down the reactor.
- The rupture is large enough to cause rapid coolant loss.
- The power peaking factor is 1.3.

SAR section 13.5.6 indicates that shutdown decay heat removal can only occur by natural circulation of air currents and by radiation heat transfer from the core to the graphite stack. It is assumed that no heat escapes from the graphite stack to the outside environment and axial heat transfer is neglected. The calculation was performed using a GEH proprietary version of the Transient Reactor Analysis Code computer program. Based on this calculation, the licensee concluded that the fuel temperature reaches a maximum of 626°F (330°C) 30 minutes after coolant loss and then temperature begins to decrease. The rise in the graphite stack temperature is only 15°F (8°C) in about 3 hours.

In addition, SAR section 13.5.6 states that the analysis was repeated using a higher peaking factor. The maximum fuel temperature for a loss-of-coolant accident with a 1.58 peaking factor is 800°F (427°C) at about 20 minutes. Reactor power at the time of the peak fuel temperature is 1.5 kW. It has been shown that this power could be tolerated indefinitely without increasing graphite temperatures to over 150°F (66°C), assuming a natural convection heat transfer coefficient of 0.6 Btu/h-ft^2-°F on the exposed surface of the reactor. Therefore, a second fuel temperature peak greater than 150°F (66°C) is not possible.

The NRC staff has reviewed the analysis of the reactor loss-of-coolant accident and finds the methods, results, and analysis to be acceptable. Therefore, the NRC staff concludes that a sudden loss of all coolant from the NTR, even if accompanying or caused by a core crushing accident, would not lead to fuel temperatures that would cause loss of integrity of the fuel cladding and that the health and safety of the public would not be endangered by such an event.

4.2.13 Consequences of Accidental Explosions

SAR section 13.6.3, "Consequences of Accidental Explosions," describes the consequences of accidental explosions. The licensee stated that the facilities, equipment, and procedures used for experiment programs that involve explosive material are described in SAR chapter 10, "Experimental Facilities and Utilization." To provide safe limits for the amounts of explosives permitted in the NTR handling and radiography areas, separate DBAs were defined for the south cell, the north room, and the setup room. In general, these DBAs assume a highly improbable accidental detonation of all explosive devices in the particular area and the consequences are evaluated in terms of both radiological and mechanical effects.

Radiological Consequences

SAR section 13.6.3.1, "Radiological Consequences," states that the radiological consequences of an accidental detonation of an explosive device are essentially nonexistent. Induced activities in explosive materials, structural materials containing the explosive, or structures used in neutron radiography are extremely small considering thermal neutron fluxes of 2×10^6 n/cm^{2}s and normal exposure times of 10^3 seconds. However, if sufficient other sources of radioactive materials are present in the immediate area and become dispersed or airborne during the accidental detonation, the radiological consequences could be serious. Operations at the NTR include neutron radiography of uranium fuel pins and capsules containing significant amounts of fission products. Evaluation of the DBAs indicates that while it is virtually impossible to involve these materials in the accident, it is prudent to exclude these large sources of radioactive material from any area in which explosive devices are being handled.

SAR section 13.6.3.1 also states that small amounts of radioactive materials (e.g., uranium contained in fission chambers or irradiated samples used in various experimental programs) may be safely stored in the south cell or the north room during the neutron radiography of explosives. By limiting these quantities to 10 curies of radioactive materials and to 50 grams of uranium, the health and safety of the general public will in no way be compromised. Storage locations are at least 5 feet from any explosive handling position and are normally either in concrete block caves or small lead casks. While accidental detonation of explosive devices might cause minor damage to the storage structures, the probability of releasing even a small percentage of the radioactive materials from their contents is negligible. Assuming a 1-percent release and stable atmospheric conditions (inversion), maximum site boundary doses are less than 20 mrem to the thyroid and 1 mrem to the whole body under this most conservative combination of circumstances. No radioactive materials other than those produced by neutron radiography are permitted in the setup room if explosive devices are present.

Mechanical Consequences

SAR section 13.6.3.2, "Mechanical Consequences," states that the primary safety criterion is that complete simultaneous detonation of all explosive devices in a particular area will not increase the probability or consequences of accidents previously analyzed or create the possibility of a different type of accident not previously analyzed. While minor structural damage and possible injury to personnel will occur in the immediate area, damage to the reactor core, graphite pack, or control system is not expected, and injury to personnel is minimized. Damage to the reactor is prevented by limiting the amount of explosive material allowed in the particular areas (south cell, north room, and set up room) and by design and construction of an additional shield structure (south cell). Potential injury to personnel is minimized by strict adherence to safe explosive handling procedures. The mechanical safety analyses show that the neutron radiography of explosives can be accomplished safely in the reactor facility by limiting both the total quantity of explosive materials in pounds of equivalent TNT and the distance of the explosive material from sensitive components and structures.

Reactivity Effects

SAR section 13.6.3.3, “Reactivity Effects,” states that there are no reactivity effects directly associated with neutron radiography of explosive or other materials. Objects undergoing inspection are located at relatively large distances from the reactor and have no effect on core reactivity. Even the large shutter in the south cell may be moved during reactor operation without affecting core reactivity. Some minor reactivity effects are associated with the neutron radiography beam preparation devices. Under normal circumstances, shock waves from accidental detonation of explosives will be attenuated sufficiently to make movement of the beam preparation device highly improbable. It is also noted that the reactivity added during removal or expulsion of the beam preparation device from the core region is included in the total amount that would be available, as discussed in SAR section 13.4.3. Therefore, the consequences would be less severe than those analyzed, which assume $0.76 step insertion both with and without scram.

Section 14.11, “Flammable or Explosive Device,” of the NRC staff SE for License Amendment (LA) No. 21 states that:

> Some experimental programs at the NTR facility utilize neutron radiography involving flammable or explosive materials. The licensee has analyzed potential accidents involving fissile materials (see Section 14.1 of this SE) and flammable or explosive materials and has established criteria for experimental activities involving these materials. These criteria and their bases are discussed below.
>
> The licensee’s TSs permit experimental activities, primarily neutron radiography, involving flammable and explosive material in the form of finished or test sample devices only. An analysis of the consequences of accidental explosions at the NTR facility is presented in the SAR, Section 13.5. This analysis was used to establish limits on distances from specific points and equivalent TNT mass for the south cell, north room and set-up room. An explosive storage magazine for storage of 10 pounds (4.5 kg) of class A and B explosives with a total maximum of 100 lbs. (45 kg), including class C materials, is provided at a location that is separate from the NTR facility. The various weight and distance limits, as well as other limitations on explosive materials irradiated, are incorporated into Section 3.7 of the TSs and summarized in Table 14-6 of this SE. For flammable materials, TSs limit the potential chemical energy and provide controls to ensure no damage to the reactor.
>
> The NRC staff has reviewed the computational procedures and the TS limits and controls used by the licensee and concludes that they are acceptable, thus ensuring that the licensee’s activities involving flammable and explosives do not represent a hazard to the facility, the staff or the public.

Table 14-6
Explosive Material Limitations

Maximum cumulative radiation exposures:

Neutron	3×10^{12} n/cm^2
Gamma	1×10^4 R

Mass and distance limits:*

South Cell	$W \leq (D/2)^2$	$W \leq 9$ lb, $D \geq 3$ ft
North Room		
Without MSM	$W \leq D^2$	$W \leq 16$ lb, $D \geq 1$ ft
With MSM	$W \leq 2$ lb	
Setup Room	$W \leq 25$ lb	

where W = TNT equivalent mass, and
D = distance from south cell blast wall or north room wall

Radioactivity and Fissile Material:

10 Ci maximum and ≤ 50 g uranium may be in storage in the South Cell or the North Room when explosive materials are present provided the storage location is ≥ 5 ft from explosive material. No radioactive materials are allowed in setup room other than those produced by the neutron radiograph exposure when explosive materials are present.

High-frequency Generating Equipment:

Must not be operated ≥ 50 ft from an explosive device.

*An assembly or accumulation of fissile material is one wherein the parts are separated from each other by less than 12 inches.

The licensee also proposed the following TSs for the control of explosive materials (which are generally unchanged from the existing TSs) (see SER section 5.3.8.3, “Explosives Limits for the NTR”):

Proposed TS 3.8.3, “Explosives Limits for The NTR,” states:

> The amounts of explosives (detonating and deflagrating, DOT Hazard Class/Divisions 1.1, 1.2, 1.3 and 1.4) permitted in the NTR facilities are as follows:
>
> i. South Cell, $W \leq (D/2)^2$ with $W \leq 9$ lbs and $D \geq 3$ ft.
> ii. North room (without Modular Stone Monument [MSM]), $W \leq D^2$ with $W \leq 16$ lbs and $D \geq 1$ft.
> iii. Setup Room, $W \leq 25$ lbs.

Proposed TS 3.8.4, “Explosives Limits for The North Room,” states:

> The amounts of explosives allowed in the North room MSM (inclusive in the limit of 3.8.3.ii. above) are as follows:
>
> i. for DOT Hazard Class Divisions 1.1, 1.2, and 1.3 (detonating): $W \leq 2$ pounds
> ii. for DOT Hazard Class Division 1.4 (deflagrating): $W \leq 4$ pounds
> where: W = Total weight of explosives in pounds of equivalent TNT.
> D = Distance in feet from the South Cell blast shield or the North Room wall.

Proposed TS 3.8.7, “Radioactive Material Near Explosives,” states:

> A maximum of 10 Ci of radioactive material and up to 50 g of uranium *SHALL* be in storage in a neutron radiography area where explosive devices are present (i.e., in the South Cell or North Room). The storage locations *SHALL* be at least 1.5 m (5 ft) from any explosive device.
>
> Radioactive materials, other than byproduct irradiated explosive devices and imaging systems, are not permitted in the Setup Room if *EXPLOSIVE MATERIAL* is present.
>
> <u>Exception.</u> Devices containing not more than 10 grams TNT equivalent of explosives with up to 200 mCi of tritium in the form of tritiated metal (hydride) are permitted. However, no more than one device *SHALL* be in a neutron radiography area or the setup room at any one time, and no other *EXPLOSIVE MATERIAL SHALL* be in the same area at that time.

Proposed TS 3.8.8, “Explosive in Radiation Fields,” states:

> No explosive device shall be placed in a radiation field greater than 1×10^4 roentgens or consisting of greater than 3×10^{12} n/cm^2 thermal neutrons.

Proposed TS 3.8.9, “Electromagnetic Wave Near Explosives Restriction,” states:

> With the exception of communication equipment utilizing low-energy electromagnetic waves in radiofrequencies, such as mobile phones and two-way hand-held radios, unshielded high-frequency generating equipment *SHALL* not be operated within 50 feet of any explosive device.

As stated in section 14.11 of its SE for LA No. 21, the NRC staff found the TS limits for the possession and irradiation of explosives acceptable. Further, the licensee has not requested substantive changes to its explosive limits provided in proposed TSs 3.8.3, 3.8.4, 3.8.7, 3.8.8, and 3.8.9. Based on the above, the NRC staff finds the licensee’s control of explosive materials and its proposed TSs 3.8.3, 3.8.4, 3.8.7, 3.8.8, and 3.8.9 acceptable.

4.3 Conclusions

The NRC staff concludes that the licensee has postulated and analyzed sufficient accident-initiating events and scenarios to demonstrate that the NTR is acceptably designed to avoid inadvertent reactor damage that could prevent a safe shutdown. There is reasonable assurance that no credible accident would cause unacceptable radiological risk to the facility staff, the environment, or the public. The NRC staff also concludes that the proposed license and TSs provide reasonable assurance that the assumptions and conditions of the licensee's safety analysis will be met. Facility operation within the limits of the proposed license and TSs will not result in offsite radiation exposures in excess of 10 CFR Part 20 limits. Finally, the proposed license and TSs will acceptably limit the likelihood of a malfunction and mitigate the consequences to the public in regard to accident events.

5 TECHNICAL SPECIFICATIONS

In this section of the safety evaluation report (SER), the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff provides its evaluation of the GE-Hitachi Nuclear Energy Americas LLC (GEH, the licensee) proposed technical specifications (TSs) for the Nuclear Test Reactor (NTR, the facility). The proposed TSs define specific features, characteristics, and conditions governing the safe operation of the NTR. The NRC staff reviewed the format and content of the proposed TSs for consistency with the guidance in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” Part 1, chapter 14, “Technical Specifications,” appendix 14.1, “Format and Content of Technical Specifications,” and American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, “The Development of Technical Specifications for Research Reactors.” The NRC staff specifically evaluated the content of the proposed TSs to determine if it meets the requirements in Title 10 of the *Code of Federal Regulatio*ns (10 CFR) section 50.36, “Technical specifications.”

5.1 Introduction

5.1.1 Scope and Purpose

Proposed TS 1.1, “Scope and Purpose,” states:

> This document constitutes the Technical Specifications for the GEH Nuclear Test Reactor as required by 10 CFR 50.36 and supersedes all prior Technical Specifications. This document includes the “basis” to support the selection and significance of the specifications. The Technical Specifications are based on the guidance provided in American National Standards Institute/ American Nuclear Society (ANSI/ANS) 15.1-2007, “The Development of Technical Specifications for Research Reactors” as modified by NUREG-1537, Part 1, Appendix 14.1, “Format and Content of Technical Specifications for Non-Power Reactors.”
>
> These Technical Specifications provide limits within which operation of the reactor will assure the health and safety of the public, the environment, and on-*SITE* personnel. Areas addressed are Definitions, Safety Limits (SL), Limiting Safety System Settings (LSSS), Limiting Conditions for Operation (LCO), Surveillance Requirements, Design Features and Administrative Controls.

The NRC staff reviewed proposed TS 1.1 and finds that the information in proposed TS 1.1 related to the scope of the TSs is consistent with the guidance in NUREG-1537, Appendix 14.1, and ANSI/ANS-15.1-2007, and with 10 CFR 50.36. The NRC staff finds that the proposed TSs include a safety limit (SL), limiting safety system settings (LSSSs), limiting conditions for operation (LCOs), surveillance requirements (SRs), design features, and administrative controls, consistent with the requirements of 10 CFR 50.36(c). The NRC staff finds that the SL, LSSSs, LCOs, SRs, and design features in the proposed TSs include applicability and objective statements consistent with the guidance in ANSI/ANS-15.1-2007 and NUREG-1537, appendix 14.1. The NRC staff also finds that the bases for the SL, LSSSs, LCOs, SRs, and design features in the TSs summarize the rationale for those TSs, but that the bases are not part of the TSs, as required by 10 CFR 50.36(a)(1). Therefore, the NRC staff concludes that proposed TS 1.1 is acceptable.

5.1.2 Definitions

Proposed TS 1.2, “Definitions,” states:

ADMINISTRATIVE CHANGE(S):

An editorial, non-technical change, which does not affect nuclear safety, personnel safety, security, quality, or change the intent of the document being changed.

CHANNEL(S):

The combination of sensors, lines, amplifiers, and output devices which are connected for the purpose of measuring the value of a parameter.

CHANNEL CALIBRATION:

A comparison and/or an adjustment of the *CHANNEL* so that its output corresponds with acceptable accuracy to known values of the parameter which the *CHANNEL* measures. Calibration *SHALL* encompass the entire *CHANNEL*, including equipment actuation, alarm, or trip test and *SHALL* include the *CHANNEL TEST*.

CHANNEL CHECK:

A qualitative verification of acceptable performance by observation of *CHANNEL* behavior. This verification where possible *SHALL* include comparison of the *CHANNEL* with other independent *CHANNELS* or systems measuring the same parameter.

CHANNEL TEST:

The introduction of a signal into the *CHANNEL* to verify that it is *OPERABLE*.

CONFINEMENT:

The enclosure of the overall *FACILITY* that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

CONTROL ROD(S):

A non-scrammable device having an electric motor drive. The rod contains boron-carbide material used to establish neutron flux changes and to compensate for routine reactivity losses (Refer to Design Feature 5.3.1.).

CORE CONFIGURATION:

The fixed assembly that includes 16 fuel assemblies each containing 40 fuel discs. The assemblies are contained within and evenly distributed around the annular core tank (Refer to Design Feature 5.3.1.). Positioned around the outer edge of the core tank are four *SAFETY RODS*, three *CONTROL RODS*, and installed *MANUAL POISON SHEETS.*

EXPERIMENT(S):

Any operation, hardware or target (excluding devices such as detectors, foils, etc.) which is designed to investigate non-routine reactor characteristics or which is intended for irradiation in an *EXPERIMENTAL FACILITY,* and which is not rigidly secured to a core or shield structure so as to be a part of their design. *EXPERIMENTS* can include:

1. **SECURED EXPERIMENT:** Any *EXPERIMENT* or component of an *EXPERIMENT* that is held in a stationary position relative to the reactor by mechanical means. The restraining forc*es* must be substantially greater than those to which the *EXPERIMENT* might be subjected by hydraulic, pneumatic, or other forces which are normal to the operating environment of the *EXPERIMENT*, or by forces that can arise as a result of credible malfunctions.

2. **MOVABLE EXPERIMENT:** Any *EXPERIMENT* where it is intended all, or part of the *EXPERIMENT MAY* be moved in or near the core or into and out of an *EXPERIMENTAL FACILITY* during *REACTOR OPERATION*.

EXPERIMENTAL FACILITY or EXPERIMENTAL FACILITIES:

Any location for an *EXPERIMENT* which is on or against the external surfaces of the reactor main graphite pack, thermal column, or within any penetration thereof.

EXPLOSIVE MATERIAL:

Any chemical compound or mixture, the primary or common purpose of which is to function by an essentially instantaneous release of gas and heat. *EXPLOSIVE MATERIAL* in the NTR includes:

- Detonating, DOT Type I
- Deflagrating, DOT Type II - IV

FACILITY:

That portion of Building 105 composed of the NTR reactor cell, control room, north room, setup room, and south cell.

FLAMMABLE:

A *FLAMMABLE* liquid is any liquid having a flash point under 100°F. A *FLAMMABLE* solid is any solid material, other than one classified as an explosive, which is liable to cause fires through friction or which can be ignited easily and when ignited burns so vigorously and persistently as to create a serious hazard. *FLAMMABLE* solids include spontaneously combustible and water- reactive materials.

LICENSE, LICENSED, or LICENSEE:

The written authorization (LICENSE R-33), by the responsible authority (The NRC), for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material, or *FACILITY* requiring licensing.

LICENSED REACTOR OPERATOR(S) / REACTOR OPERATOR(S) / SENIOR REACTOR OPERATOR(S):

A person who is *LICENSED* as a *REACTOR OPERATOR (RO)* or *SENIOR REACTOR OPERATOR (SRO)* pursuant to 10 CFR Part 55 to operate the controls of the Nuclear Test Reactor.

MANUAL POISON SHEET(S) (MPS):

Manually positioned devices containing cadmium material used to compensate for fuel burnout and limit the amount of *POTENTIAL EXCESS REACTIVITY* available to the operator (Refer to Design Feature 5.3.1.).

MEASURED VALUE:

The value of a parameter as it appears at the output of a *CHANNEL*.

OPERABLE / INOPERABLE:

A system or component is / is not capable of performing its intended function.

OPERATING:

A component or system is performing its intended function.

POTENTIAL EXCESS REACTIVITY:

That reactivity which can be added by the remote manipulation of *CONTROL RODS* from the point that the reactor is exactly critical plus the maximum credible reactivity addition from primary coolant temperature change plus the *REACTIVITY WORTH* of all installed *EXPERIMENTs*.

PROTECTIVE ACTION(S):

The initiation of a signal or the operation of equipment within the *REACTOR SAFETY SYSTEM* in response to a parameter or condition of the reactor *FACILITY* having reached a specified limit.

REACTIVITY WORTH (EXPERIMENT):

The value of the reactivity change that results from the *EXPERIMENT* being inserted into or removed from its intended position.

REACTOR OPERATING or REACTOR OPERATION(S):

The reactor is *OPERATING* whenever it is not in *REACTOR SECURED* or *REACTOR SHUTDOWN* conditions.

REACTOR THERMAL POWER:

The *REACTOR THERMAL POWER*, as determined by a primary coolant system heat balance.

REACTOR SAFETY SYSTEM(S):

Those systems, including their associated input *CHANNELS,* which are designed to initiate automatic reactor protection or to provide information for initiation of manual *PROTECTIVE ACTION.*

REACTOR SECURED:

The reactor is considered secured when:

1. EITHER there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection.

2. OR the following conditions exist:

   (a) *REACTOR SHUTDOWN*.
   (b) The console keylock switch is OFF and the key is removed from the lock.
   (c) No work is in progress on core components that can directly affect core reactivity, including core fuel, core structure, installed control or *SAFETY RODS*, or *CONTROL ROD* drives unless they are physically decoupled from the *CONTROL RODS*.
   (d) No *EXPERIMENT*s are being moved or serviced that have, on movement, a *REACTIVITY WORTH* exceeding the maximum value allowed for a single *EXPERIMENT*, or one dollar, whichever is smaller.

REACTOR SHUTDOWN:

The reactor is shutdown if it is subcritical by at least one dollar in the REFERENCE CORE CONDITION with the REACTIVITY WORTH of all installed *EXPERIMENTs* included.

READILY AVAILABLE SENIOR REACTOR OPERATOR:

A *Senior Reactor Operator* is readily available on call when the SRO:

1. has been specifically designated and the designation is known to the *REACTOR OPERATOR* on duty, and
2. can be rapidly contacted by phone by the RO on duty, and
3. once contacted, is capable of arriving at the NTR within a reasonable time (½ hour / 30-mile radius) under normal conditions.

REFERENCE CORE CONDITION:

Condition of the core when it is at ambient temperature and the reactivity worth of xenon is negligible (<0.30 dollar).

SAFETY ROD(S):

Spring-actuated scrammable devices containing boron-carbide material used to perform the safety function of ensuring the reactor can be placed in *REACTOR SHUTDOWN* from any *OPERATING* condition. (Refer to Design Feature 5.3.1.).

SCRAM TIME:

The elapsed time between the generation of a safety system scram signal and when the *SAFETY ROD* reaches the full-in position.

SHALL, SHOULD, AND MAY:

The word "*SHALL*" is used to denote a requirement; the word "*SHOULD*" is used to denote a recommendation; and the word "*MAY*" is used to denote permission, neither a requirement nor a recommendation.

SHUTDOWN MARGIN:

The reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible *OPERATING* condition, although the most reactive *SAFETY ROD* is stuck in its most reactive position, and the three *CONTROL RODS* are in their most reactive positions, and that the reactor will remain subcritical without further operator action.

SITE:

The area within the confines of the Vallecitos Nuclear Center (VNC) controlled by the *LICENSEE* (Refer to Safety Analysis Report, Figure 2-3).

SURVEILLANCE INTERVALS:

- Quinquennial – interval not to exceed 70 months.
- Biennial – interval not to exceed 30 months.
- Annual – interval not to exceed 15 months.
- Semi-annual – interval not to exceed 7.5 months.
- Quarterly – interval not to exceed 4 months.
- Monthly – interval not to exceed 6 weeks.
- Weekly – interval not to exceed 10 days.
- Daily - *Must* be done during the calendar day.
- Prior to SU – Prior to the first reactor start-up of the day.

TRUE VALUE:

The *TRUE VALUE* for a parameter is its actual value.

UNSAFE CONDITION:

A condition that can exist related to either nuclear safety or radiological safety. An *UNSAFE CONDITION* relative to nuclear safety exists if the ability to place the reactor in *REACTOR SHUTDOWN* is compromised or the ability to maintain the reactor subcritical is compromised as verified in Chapter 13 analysis. An *UNSAFE CONDITION* relative to radiological safety can only exist if any combination of failures in equipment or administrative radiological work controls results in an individual being assigned an unplanned dose greater-than-or-equal-to 100 mrem. Determination of an *UNSAFE CONDITION SHOULD* consider the single failure of an active component or a single administrative barrier when assessing radiological safety.

UNSCHEDULED SHUTDOWN(S):

Any unplanned shutdown of the reactor caused by actuation of the scram *CHANNELS*, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation excluding shutdowns which occur during planned equipment testing or check-out operations.

The NRC staff reviewed proposed TS 1.2 and finds that the definitions therein are either standard definitions used in research reactor TSs or are facility-specific definitions that are consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Therefore, the NRC staff concludes that proposed TS 1.2 is acceptable.

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 Safety Limits

Proposed TS 2.1, “Safety Limits,” Specification, states:

REACTOR THERMAL POWER

The *TRUE VALUE* of the *REACTOR THERMAL POWER SHALL* not exceed 190 kW.

The NRC staff found the licensee’s proposed SL that the reactor thermal power shall not exceed 190 kW acceptable, as evaluated in SER section 2.5.3.1, “Safety Limits.” Further, the NRC staff finds that this SL will reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity, as required by 10 CFR 50.36(c)(1). The NRC staff also finds that the SL in proposed TS 2.1 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Therefore, the NRC staff concludes that proposed TS 2.1 is acceptable.

5.2.2 Limiting Safety System Settings

Proposed TS 2.2, "Limiting Safety System Settings," Specification, states:

> **Linear Power - *MEASURED VALUE***
>
> > The linear neutron power monitor *CHANNEL* set point *SHALL* not exceed the *MEASURED VALUE* of 125 kW.

The NRC staff found the licensee's proposed LSSS that the linear neutron power monitor channel set point shall not exceed the measured value of 125 kW acceptable, as evaluated in SER sections 2.5.3.2, "Limiting Safety System Settings," and 2.6, "Thermal-Hydraulic Design." Further, the NRC staff finds that this LSSS is a setting for automatic protective devices related to those variables having significant safety functions that is so chosen that automatic protective action will correct the abnormal situation before the SL is exceeded, as required by 10 CFR 50.36(c)(1). The NRC staff also finds that the LSSS in proposed TS 2.2 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Therefore, the NRC staff concludes that proposed TS 2.2 is acceptable.

5.3 Limiting Conditions for Operation

5.3.1 Reactor Core Parameters

5.3.1.1 Potential Excess Reactivity

Proposed TS 3.1, "Reactor Core Parameters," Specification 3.1.1, "Potential Excess Reactivity," states:

> *POTENTIAL EXCESS REACTIVITY SHALL* be ≤ $0.76. If it is determined to be > $0.76, the reactor SHALL be placed in *REACTOR SHUTDOWN* immediately.

The NRC staff found the licensee's proposed potential excess reactivity limit of less than or equal to $0.76 acceptable, as evaluated in SER section 2.5.3.3, "Excess Reactivity." Further, the NRC staff finds that the potential excess reactivity of $0.76 provides sufficient reactivity to compensate for various negative reactivity effects associated with the operation and use of the reactor, as well as allowing some operational flexibility. The NRC staff also finds that proposed TS 3.1, Specification 3.1.1 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Therefore, the NRC staff concludes that proposed TS 3.1, Specification 3.1.1 is acceptable.

5.3.1.2 Subcritical Rod Position

Proposed TS 3.1, Specification 3.1.2, "Subcritical Rod Position," states:

> The reactor *SHALL* be subcritical whenever the four *SAFETY RODS* are withdrawn from the core and the three *CONTROL RODS* are fully inserted. Place reactor in *REACTOR SHUTDOWN* if this condition is not met.

SAR section 4.4.3 states that the reactor shall be subcritical whenever the four safety rods are withdrawn from the core and the three control rods are fully inserted. This ensures that criticality will not be achieved during safety rod withdrawal. Adherence to the $0.76 excess reactivity limit also ensures that the reactor will not go critical during safety rod withdrawal. Further, SAR section 4.2.2 states that the four safety rods were designed for rapid insertion to scram the reactor; whereas the control rods were designed for precision position control and indication required for the analytical work of the reactor.

The NRC staff finds that, although the subcritical rod position described in proposed TS 3.1, Specification 3.1.2 is not a typical or standard term associated with reactivity control, for the NTR it provides an additional margin of safety as the negative reactivity associated with the safety rods will be sufficient to shut down the reactor. Therefore, the NRC staff concludes that proposed TS 3.1, Specification 3.1.2 is acceptable.

5.3.1.3 Shutdown Margin

Proposed TS 3.1, Specification 3.1.3, “Minimum Shutdown Margin,” states:

> The minimum *SHUTDOWN MARGIN* with the maximum worth *SAFETY ROD* stuck out SHALL be $1.0.

The NRC staff found the licensee’s proposed shutdown margin limit of $1.0 acceptable, as evaluated in SER section 2.5.3.4. Further, the NRC staff reviewed proposed TS 3.1, Specification 3.1.3 and finds that it helps ensure that the reactor can be safely shutdown from any operational configuration and remain shut down, even if the maximum worth safety control rod should stick in the fully withdrawn position. The NRC staff also finds that proposed TS 3.1, Specification 3.1.3 is consistent with the guidance in NUREG-1537, Part 1, and ANSI/ANS-15.1-2007. Therefore, the NRC staff concludes that proposed TS 3.1, Specification 3.1.3 is acceptable.

5.3.2 <u>Reactor Control and Safety System</u>

5.3.2.0 *General*

Proposed TS 3.2.0, “General,” states:

> The reactor *SHALL* be placed in *REACTOR SHUTDOWN* immediately if any portion of the *REACTOR SAFETY SYSTEM* malfunctions, except as provided for in Tables 3-1 and 3-2.

The NRC staff finds that proposed TS 3.2.0 requires the systems listed in TS table 3-1, “Reactor Safety System - Scram,” and TS table 3-2, “Reactor Safety-Related Items,” to be operable or the reactor shall be shut down immediately. The NRC staff also finds that proposed TS 3.2.0 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.2, “Reactor Control and Safety Systems,” item (4), “Scram Channels,” which states, in part, that a table should specify all required scram channels, setpoints, minimum number of channels, and other functions performed by the channels. Therefore, the NRC staff concludes that proposed TS 3.2.0 is acceptable.

5.3.2.1 Rods Operable

Proposed TS 3.2, “Reactor Control and Safety System,” Specification 3.2.1, “Rods Operable,” states:

> *REACTOR OPERATION SHALL* be permitted only when all four *SAFETY RODS* and all three *CONTROL RODS* are *OPERABLE*. The reactor *SHALL* be placed in *REACTOR SHUTDOWN* immediately if it is known that a *SAFETY ROD* or *CONTROL ROD* is NOT *OPERABLE*.

The NRC staff finds that proposed TS 3.2, Specification 3.2.1 requires all safety and control rods to be operable or the reactor to be shut down immediately. The NRC staff also finds that proposed TS 3.2, Specification 3.2.1 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.2, item (1), “Operable Control Rods,” which states, in part, that the number and type of operable control and safety rods should be specified. Therefore, the NRC staff concludes that proposed TS 3.2, Specification 3.2.1 is acceptable.

5.3.2.2 Safety Rod Withdrawal

Proposed TS 3.2, Specification 3.2.2, “Safety Rod Withdrawal,” states:

> No more than one *SAFETY ROD SHALL* be simultaneously moved in an outward direction.

The NRC staff found the licensee’s proposed limit of safety rod withdrawal to no more than one rod at a time acceptable, as evaluated in SER section 2.2.2. Further, the NRC staff finds that proposed TS 3.2, Specification 3.2.2 helps ensure that the reactivity insertion rate of the safety rods is limited to a single withdrawn rod. The NRC staff also finds that proposed TS 3.2, Specification 3.2.2 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.2, item (2), “Reactivity Insertion Rates,” which states, in part, that TSs should explicitly state if multiple rod withdrawal is allowed. Therefore, the NRC staff concludes that proposed TS 3.2, Specification 3.2.2 is acceptable.

5.3.2.3 Safety Rod Withdrawal Rate

Proposed TS 3.2, Specification 3.2.3, “Safety Rod Withdrawal Rate,” states:

> The rate of withdrawal of each *SAFETY ROD* during R*EACTOR OPERATION SHALL* be less than 1 ¼ inches per second.

The NRC staff found the licensee’s proposed limit of safety rod withdrawal speed to less than 1.25 inches per second acceptable, as evaluated in SER section 2.2.2. The NRC staff also finds that proposed TS 3.2, Specification 3.2.3 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.2, item (2), which states, in part, that the TSs should specify the reactivity insertion rates for the safety and control rods. Therefore, the NRC staff concludes that proposed TS 3.2, Specification 3.2.3 is acceptable.

5.3.2.4 Control Rod Withdrawal Rate

Proposed TS 3.2, Specification 3.2.4, “Control Rod Withdrawal Rate,” states:

> The rate of withdrawal of *CONTROL RODS* during *REACTOR OPERATION SHALL* be less than 1/6 inch per second. The rods can be inserted or withdrawn singly or multiple rods simultaneously.

The NRC staff found the licensee's proposed limit of control rod withdrawal speed to less than 1/6 inch per second acceptable, as evaluated in SER section 2.2.2. The NRC staff also finds that proposed TS 3.2, Specification 3.2.4 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.2, item (2), which states, in part, that the TSs should specify the reactivity insertion rates for the safety and control rods. Therefore, the NRC staff concludes that proposed TS 3.2, Specification 3.2.4 is acceptable.

5.3.2.5 Scram Time

Proposed TS 3.2, Specification 3.2.5, "Scram Time," states:

> The average *SCRAM TIME* of the four *SAFETY RODS SHALL* not exceed 300 msec.

The NRC staff found the licensee's proposed limit of average scram time of the safety rods to 300 milliseconds or less acceptable, as evaluated in SER section 2.2.2. The NRC staff also finds that proposed TS 3.2, Specification 3.2.5 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.2, item (1), which states, in part, that the TSs should specify the maximum scram time for scrammable rods. Therefore, the NRC staff concludes that proposed TS 3.2, Specification 3.2.5 is acceptable.

5.3.2.6 Reactor Safety System and Safety-Related Items

Proposed TS 3.2, Specification 3.2.6, "Reactor Safety System and Safety-Related Items," states:

> *REACTOR OPERATION SHALL* be permitted only when the *REACTOR SAFETY SYSTEM* is *OPERABLE* in accordance with Table 3-1 and Table 3-2.
>
> - Table 3-1 specifies automatic trip set points, scram system components, and minimum number of *CHANNELS* necessary to ensure *PROTECTIVE ACTIONS* can be taken to place the reactor in *REACTOR SHUTDOWN*. The Trip Points in Table 3-1 reflect the minimum values necessary to avoid approaching the LCOs in Sections 3.1 and 3.2 of these Technical Specifications.
>
> - Table 3-2 specifies alarm set points and rod interlock features that prompt operator actions that ensure the *FACILITY* is maintained within normal *OPERATING* parameters.

Table 3-1
REACTOR SAFETY SYSTEM - SCRAM

Item No.	System	Condition	Trip Point	Function	Min. Number of Channels
1.	Linear Power	High reactor power	≤ 125 kW	Scram (2-out-of-3 or 1-out-of-2)	2
		Loss of positive high voltage to ion chambers (if used)	No less than 90% of *OPERATING* voltage	Scram (2-out-of-3 or 1-out-of-2)	
2.	Log N	Fast reactor period	No less than +5 sec	Scram	1
		Amplifier Mode switch not in operate	N/A	Scram	
		Loss of positive high voltage to ion chambers (if used)	No less than 90% of *OPERATING* voltage	Scram	
3.	Primary Coolant Temperature (Fenwall)	High core outlet temperature	≤ 222 °F	Scram	1
4.	Primary Coolant Flow	Low Flow	No less than 15 gpm when reactor power > 0.1 kW	Scram	1
5.	Manual	Console button depressed	N/A	Scram	1
6.	Electrical Power	Reactor console key in off position (loss of AC power to console)	N/A	Scram	1

Table 3-2
REACTOR SAFETY-RELATED ITEMS

Item No.	System	Condition	Set Point	Function
1.	Reactor Cell Pressure	Low Differential pressure	> 0.5 in. water ΔP	Visible and audible alarm; audible alarm *MAY* be bypassed after recognition.
2	Fuel Loading Tank Water Level	Low Level	< 3 ft. below the overflow	Visible and audible alarm; audible alarm *MAY* be bypassed after recognition.
3.	Primary Coolant Temperatures	High core outlet temperature	<200°F	Visible and audible alarm; audible alarm *MAY* be bypassed after recognition.
4.	Primary Coolant Temperatures	Core Delta temperature	N/A	Provide information for the heat balance determination
5.	Stack Radioactivity	High Level	Complies with TS 3.7.2.1	Visible and audible alarm; audible alarm *MAY* be bypassed after recognition.
6.	Linear Power	Low Power indication	≥2% on any scale	*SAFETY* RODs or *CONTROL ROD*s cannot be withdrawn (2-out-of-3 or 1-out-of-2).
7.	*CONTROL ROD* or *SAFETY ROD*	Rods not in	N/A	*SAFETY ROD* magnets cannot be reenergized
8.	*SAFETY ROD*	Rods not out	N/A	*CONTROL ROD*s cannot be withdrawn; *SAFETY ROD*s *SHALL* be withdrawn in sequence; *MAY* be bypassed to allow withdrawal of one *CONTROL ROD*, or one *SAFETY ROD* (drive) out of sequence for purposes of inspection, maintenance, and testing

SAR section 7.1, “Summary Description,” and SAR table 7-1, “Scram Systems,” describe the scram systems. SAR section 7.4, “Safety-Related Items,” and SAR table 7-2, “Safety-Related Items,” describe the safety-related system The NRC staff finds that proposed TS 3.2, Specification 3.2.6, including TS tables 3-1 and 3-2, is consistent with the scram and safety-related features provided in the SAR and with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.2, item (4) and item (5), “Interlocks,” and remains unchanged (other than minor edits for clarity and consistency with the SAR) since the issuance of the previous license renewal by License Amendment (LA) No. 21. Therefore, the NRC staff concludes that proposed TS 3.2, Specification 3.2.6 is acceptable.

5.3.3 Reactor Coolant System

5.3.3.1 Forced Flow Cooling

Proposed TS 3.3, "Reactor Coolant System," Specification 3.3.1, "Forced Flow Cooling," states:

> For *REACTOR OPERATION* above 0.1 kW, the reactor *SHALL* be cooled by light water forced coolant flow in *REACTOR OPERATING mode*.

The NRC staff found the licensee's proposed limit for forced flow cooling acceptable, as evaluated in SER section 2.3. Further, the NRC staff finds that the licensee's proposed TS remains unchanged since the issuance of the previous license renewal by LA No. 21. The NRC staff also finds that there were no changes to the facility that would affect proposed TS 3.3, Specification 3.3.1. Therefore, the NRC staff concludes that proposed TS 3.3, Specification 3.3.1 is acceptable.

5.3.3.2 Core Tank Full

Proposed TS 3.3, Specification 3.3.2, "Core Tank Full," states:

> *REACTOR OPERATION SHALL* not be permitted unless the fuel loading tank is filled with water which ensures that the core tank is full. If during operation of the reactor it is determined that the fuel loading tank is not filled with water, the reactor *SHALL* be placed in *REACTOR SHUTDOWN* immediately.

The NRC staff found the licensee's proposed limit regarding core tank water acceptable, as evaluated in SER section 2.3. Further, the NRC staff finds that the licensee's proposed TS remains unchanged since the issuance of the previous license renewal by LA No. 21. The NRC staff also finds that there were no changes to the facility that would affect proposed TS 3.3, Specification 3.3.2. Therefore, the NRC staff concludes that proposed TS 3.3, Specification 3.3.2 is acceptable.

5.3.3.3 Primary Coolant Conductivity

Proposed TS 3.3, Specification 3.3.3, "Primary Coolant Conductivity," states:

> The specific conductivity of the primary coolant water *SHALL* be maintained less than 5 μS/cm when averaged over a one-month period.

SAR section 5.4, "Primary Coolant Cleanup System," states that the primary system potential hydrogen (pH) is controlled by controlling the water conductivity. The conductivity is operationally maintained at or below 2 micro-Siemens (S) per centimeter (μS/cm) (note: 1 S equals 1 mho). The pH then will be between 5.6 and 9.0, which is compatible with aluminum/stainless steel systems. The conductivity of the primary coolant is checked prior to the first startup of the day in accordance with NTR standard operating procedures. Both conductivity and pH are checked annually by Analytical Chemistry in accordance with NTR preventive maintenance procedures

The NRC staff finds that the licensee's proposed TS 3.3, Specification 3.3.3 is consistent with the guidance provided in NUREG-1537, Part 1, appendix 14.1, section 3.3, "Coolant Systems," item (9), "Water Chemistry Requirements," which states that the licensee should have an LCO on coolant conductivity, and that acceptable ranges for coolant conductivity have traditionally been ≤ 5 μmhos/cm (5 μS/cm). Therefore, the NRC staff concludes that proposed TS 3.3, Specification 3.3.3 is acceptable.

5.3.4 Confinement

Proposed TS 3.4, "Confinement," states: "This section left intentionally blank." The NRC staff finds that proposed TS 3.4 is acceptable because the operations that require confinement and the equipment required to achieve confinement are specified in TS 3.5, "Reactor Cell, Ventilation, and Confinement System." Further, the NRC staff finds TS 3.4 consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 1.2.2, "Format," to be numbered consistent with ANSI/ANS-15.1-2007, section 1.2.2, "Format," to include the heading and numbering format of major sections and the first level of subheadings to ensure that all items that may be relevant for inclusion in the TSs have been considered. Therefore, the NRC staff concludes that proposed TS 3.4 is acceptable.

5.3.5 Reactor Cell, Ventilation, and Confinement System

5.3.5.1 Reactor Cell Negative Pressure

Proposed TS 3.5, "Reactor Cell, Ventilation, and Confinement System," Specification 3.5.1, "Reactor Cell Negative Pressure," states:

> In *REACTOR OPERATING* mode, reactor power SHALL not be increased above 0.1 kW unless the reactor cell is maintained at a negative pressure of not less than 0.5 in. of water with respect to the control room.
>
> If during operation of the reactor above 0.1 kW, the negative pressure with respect to the control room is not maintained, then the reactor power *SHALL* be lowered to less than 0.1 kW immediately.

The NRC staff finds that proposed TS 3.5, Specification 3.5.1 helps ensure that any airborne radioactive release into the reactor cell will be confined, exhausted through the ventilation system's ventilation filters, and released through the main stack. Further, the NRC staff finds that the licensee's proposed TS remains unchanged since the issuance of the previous license renewal by LA No. 21. The NRC staff also finds that there were no changes to the facility that would affect proposed TS 3.5, Specification 3.5.1. Therefore, the NRC staff concludes that proposed TS 3.5, Specification 3.5.1 is acceptable.

5.3.5.2 Reactor Cell Activity Release

Proposed TS 3.5, Specification 3.5.2, "Reactor Cell Activity Release," states:

> Reactor cell ventilation system *SHALL* be *OPERATING* during performance of activities that could release airborne radioactivity into the reactor cell.

The NRC staff finds that the licensee's proposed TS 3.5, Specification 3.5.2 helps ensure that any airborne radioactive release into the reactor cell will be confined, exhausted through the ventilation system's ventilation filters, and released through the main stack. Further, the NRC staff finds that proposed TS 3.5, Specification 3.5.2 remains unchanged since the issuance of the previous license renewal by LA No. 21. The NRC staff also finds that TS 3.5, Specification 3.5.2 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.5, “Ventilation Systems,” which states that ventilation systems should control the release of any radioactive effluents. Therefore, the NRC staff concludes that proposed TS 3.5, Specification 3.5.2 is acceptable.

5.3.6 Emergency Power

Proposed TS 3.6, “Emergency Power,” states: “This section left intentionally blank.” The NRC staff finds that proposed TS 3.6 is acceptable because SAR section 8.2, “Emergency Electrical Power Systems,” states that the NTR has no emergency power system. As stated in SAR Section 8.1 “Normal Electrical Power Systems,” upon a loss of electrical power to the facility, the four safety rods will scram and the three control rods will fail “as is.” The loss of electrical power is an anticipated operational occurrence that is analyzed in SAR section 13.4.1, “Loss of Normal Electrical Power,” and found acceptable by the NRC staff in section 4.2 of this SER.

5.3.7 Radiation Monitoring Systems and Effluents

5.3.7.1 Monitoring Systems During Reactor Operations

Proposed TS 3.7, “Radiation Monitoring Systems and Effluents,” Specification 3.7.1, “Monitoring Systems During Reactor Operations,” states:

> Functional area radiation monitors* are required in *EXPERIMENTAL FACILITY* spaces while *EXPERIMENTS* are in progress and the control room during REACTOR *OPERATIONS.*

5.3.7.2 Monitoring Systems During Reactor Cell Maintenance

Proposed TS 3.7, Specification 3.7.2, “Monitoring Systems During Reactor Cell Maintenance,” states:

> A functional area radiation monitor* is required in the reactor cell during maintenance activities.
>
> *A functional area radiation monitor *SHALL* include:
> - Instrument readout that is visible in the control room.
> - a gamma-sensitive instrument.
> - A local audible alarm.

The NRC staff finds that proposed TS 3.7, Specifications 3.7.1 and 3.7.2 are consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.7.1, “Monitoring Systems,” item (1), “Air Monitors (Gas and Particulate),” which states that operability of the monitoring system should be described in the TSs. Further, NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.7.1, item (3), “Area Monitors,” states that area radiation monitors located in or near the reactor room, sensitive to gamma radiation, with an audible alarm, should be specified in the TSs. Therefore, the NRC staff concludes that proposed TS 3.7, Specifications 3.7.1 and 3.7.2 are acceptable.

5.3.7.3 Effluents – Environmental Monitoring

Proposed TS 3.7, Specification 3.7.3, "Effluents – Environmental Monitoring," states:

> The VNC *SITE* utilizes environmental air sampling stations and TLD badges in locations specified by the VNC Environmental Monitoring Manual.

The NRC staff finds that proposed TS 3.7, Specification 3.7.3 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.7.1, item (4), "Environmental Monitors," which states that environmental monitoring should be specified in the TSs. Therefore, the NRC staff concludes that proposed TS 3.7, Specification 3.7.3 is acceptable.

5.3.7.4 Effluents – Stack Release Activity

Proposed TS 3.7, Specification 3.7.4, "Effluents – Stack Release Activity," states:

> The stack discharge rates of gaseous and particulate activity *SHALL* not exceed the limits in Table 3-3, ensuring compliance with the 10 CFR 20.1101(d) limit of 10 mrem/year.

Table 3-3

STACK RELEASE ACTION LEVELS

	Gaseous Activity (Ar-41)	Particulate Activity (Beta)
Weekly release	9 Ci/wk	1.7E+03 μCi/wk
Alarm setpoint	9.5E-05 μCi/cc	1.9E-08 μCi/cc

1. If the alarm setpoint is exceeded, then the operator SHALL determine the weekly release rate and take actions to ensure the weekly release rate action level is not exceeded.

2. If the weekly release rate is determined to have been exceeded, then the reactor SHALL be placed in SHUTDOWN until the condition can be evaluated and the release rates determined to be below action levels.

The NRC staff found acceptable proposed TS 3.7, Specification 3.7.4, including the release limits in proposed TS table 3-3, as evaluated in SER section 3.1.1.1, "Airborne Radiation Sources." Therefore, the NRC staff concludes that proposed TS 3.7, Specification 3.7.4 is acceptable.

5.3.7.5 Effluents – Stack Monitor Operability

Proposed TS 3.7, Specification 3.7.5, "Effluents – Stack Monitor Operability," states:

> The stack gaseous and particulate activity monitors *SHALL* be *OPERATING* when the reactor is operated above 0.1 kW or when any activity is performed in the facility that could release airborne radioactivity in the reactor cell. If either monitor is not functional:

1. Reduce power to below 0.1 kW

2. All evolutions that could precipitate airborne releases *SHOULD* be discontinued within the *FACILITY*.

3. The failed monitor *SHOULD* be restored to functionality by the end of the run or at the discretion of management.

4. If these actions cannot be completed, the reactor *SHALL* be placed in *REACTOR SHUTDOWN* and not returned to operation above 0.1 kW until both monitors are functional.

The NRC staff finds that proposed TS 3.7, Specification 3.7.5 is consistent with the guidance in NUREG-1537, Part 1, chapter 14, appendix 14.1, section 3.7.2, “Effluents,” which provides guidance that all radioactive effluents should be monitored for normal operations, and limited by the TSs. Therefore, the NRC staff concludes that proposed TS 3.7, Specification 3.7.5 is acceptable.

5.3.8 Experiments

5.3.8.1 Experiment Reactivity Worth Limit

Proposed TS 3.8, “Experiments,” Specification 3.8.1, “Experiment Reactivity Worth Limit,” states:

> The sum of the *REACTIVITY WORTH* of all *EXPERIMENTS* performed at any one time *SHALL* be limited to comply with the specification on *POTENTIAL EXCESS REACTIVITY* (Refer to LCO 3.1.1.).

The NRC staff finds that proposed TS 3.8, Specification 3.8.1 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.8, “Experiments,” and ANSI/ANS-15.1-2007, section 3.8.1, “Reactivity Limits,” item (2), which states, in part, that reactivity limits shall be established for the maximum reactivity value of all experiments. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.1 is acceptable.

5.3.8.2 Experimental Object Movement

Proposed TS 3.8, Specification 3.8.2 , “Experimental Object Movement,” states:

> No experimental object *SHALL* be moved during *REACTOR OPERATION* unless its potential *REACTIVITY WORTH* is known to be less than $0.50.

The NRC staff finds that proposed TS 3.8, Specification 3.8.2 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.8 and ANSI/ANS-15.1-2007, section 3.8.1, item (1), which states, in part, that reactivity limits shall be established for the maximum reactivity value of individual experiments. Further, the NRC staff finds that the reactivity worth of $0.50 is sufficient to prevent an inadvertent prompt criticality (which occurs at a reactivity value of $1.00). Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.2 is acceptable.

5.3.8.3 Explosives Limits for the NTR

Proposed TS 3.8, Specification 3.8.3, “Explosives Limits for the NTR,” states:

> The amounts of explosives (detonating and deflagrating, DOT Hazard Class/Divisions 1.1, 1.2, 1.3 and 1.4) permitted in the NTR facilities are as follows:
>
> i. South Cell, $W \leq (D/2)^2$ with $W \leq 9$ lbs and $D \geq 3$ ft.
>
> ii. North room (without Modular Stone Monument), $W \leq D^2$ with $W \leq 16$ lbs and $D \geq 1$ft.
>
> iii. Setup Room, $W \leq 25$ lbs.

The NRC staff found acceptable proposed TS 3.8, Specification 3.8.3, as evaluated in SER section 4.2.13, “Consequences of Accidental Explosions.” Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.3 is acceptable.

5.3.8.4 Explosives Limits for the North Room

Proposed TS 3.8, Specification 3.8.4, “Explosives Limits for the North Room,” states:

> The amounts of explosives allowed in the North room MSM (inclusive in the limit of 3.8.3. ii. above) are as follows:
>
> i. for DOT Hazard Class Divisions 1.1, 1.2, and 1.3 (detonating): $W \leq 2$ pounds
>
> ii. for DOT Hazard Class Division 1.4 (deflagrating): $W \leq 4$ pounds
> where: W = Total weight of explosives in pounds of equivalent TNT.
>
> D = Distance in feet from the South Cell blast shield or the North Room wall.

The NRC staff found acceptable proposed TS 3.8, Specification 3.8.4, as evaluated in SER section 4.2.13. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.4 is acceptable.

5.3.8.5 Experimental Objects in the Core Tank

Proposed TS 3.8, Specification 3.8.5, “Experimental Objects in the Core Tank,” states:

> Experimental objects *SHALL* not be allowed inside the core tank when the reactor is at a power greater than 0.1 kW.

The NRC staff finds that proposed TS 3.8, Specification 3.8.5 remains unchanged since the issuance of the previous license renewal by LA No. 21. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.5 is acceptable.

5.3.8.6 Experimental Objects in the Fuel Loading Chute

Proposed TS 3.8, Specification 3.8.6, “Experimental Objects in the Fuel Loading Chute,” states:

> Experimental objects located in the fuel loading chute *SHALL* be secured to prevent their entry into the core region during *REACTOR OPERATION*.

The NRC staff finds that proposed TS 3.8, Specification 3.8.6 remains unchanged since the issuance of the previous license renewal by LA No. 21. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.6 is acceptable.

5.3.8.7 Radioactive Material Near Explosives

Proposed TS 3.8, Specification 3.8.7, “Radioactive Material Near Explosives,” states:

> A maximum of 10 Ci of radioactive material and up to 50 g of uranium *SHALL* be in storage in a neutron radiography area where explosive devices are present (i.e., in the South Cell or North Room). The storage locations *SHALL* be at least 1.5 m (5 ft) from any explosive device.
>
> Radioactive materials, other than byproduct irradiated explosive devices and imaging systems, are not permitted in the Setup Room if *EXPLOSIVE MATERIAL* is present.
>
> <u>Exception.</u> Devices containing not more than 10 grams TNT equivalent of explosives with up to 200 mCi of tritium in the form of tritiated metal (hydride) are permitted. However, no more than one device *SHALL* be in a neutron radiography area or the setup room at any one time, and no other *EXPLOSIVE MATERIAL SHALL* be in the same area at that time.

The NRC staff found acceptable proposed TS 3.8, Specification 3.8.7, as evaluated in SER section 4.2.13. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.7 is acceptable.

5.3.8.8 Explosives in Radiation Fields

Proposed TS 3.8, Specification 3.8.8, “Explosives in Radiation Fields,” states:

> No explosive device *SHALL* be placed in a radiation field greater than 1×10^4 roentgens or consisting of greater than 3×10^{12} n/cm^2 thermal neutrons.

The NRC staff found acceptable proposed TS 3.8, Specification 3.8.8, as evaluated in SER section 4.2.13. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.8 is acceptable.

5.3.8.9 Electromagnetic Wave Near Explosives Restriction

Proposed TS 3.8, Specification 3.8.9, “Electromagnetic Wave Near Explosives Restriction,” states:

> With the exception of communication equipment utilizing low-energy electromagnetic waves in radiofrequencies, such as mobile phones and two-way hand-held radios, unshielded high-frequency generating equipment *SHALL* not be operated within 50 feet of any explosive device.

The NRC staff found acceptable proposed TS 3.8, Specification 3.8.9, as evaluated in SER section 4.2.13. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.9 is acceptable.

5.3.8.10 Experimental Capsule Design

Proposed TS 3.8, Specification 3.8.10, “Experimental Capsule Design,” states:

> Experimental capsules to be utilized in the *EXPERIMENTAL FACILITIES SHALL* be designed or tested to ensure that any pressure transient produced by chemical reaction of their contents and/or leakage of corrosion or *FLAMMABLE* materials will not damage the reactor.

The NRC staff finds that proposed TS 3.8, Specification 3.8.10 remains unchanged since the issuance of the previous license renewal by LA No. 21. The NRC staff also finds that proposed TS 3.8, Specification 3.8.10 is an assumption in the basis of the licensee’s maximum hypothetical accident (MHA), as documented in SE section 14.1, “MHA (Experiment Design Basis Accident),” of LA No. 21. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.10 is acceptable.

5.3.8.11 Fissile Material Experimental Limitations

Proposed TS 3.8, Specification 3.8.11, “Fissile Material Experimental Limitations,” states:

> EXPERIMENTS containing fissile material *SHALL* be encapsulated and limited to a U-235 inventory of 50 mg.

The licensee proposed to modify TS 3.8, Specification 3.8.11 to better align it with the assumptions in SAR section 13.2, “Experiment Design Basis Accident,” by stating, in part, that experimental fuel elements containing uranium must be clad. Although, such experiments have not been performed at the NTR for approximately 40 years and are unlikely to be performed in the future, the uranium fuel capsule scenario was selected as a bounding experiment for the SAR chapter 13 analysis. Proposed TS 3.8, Specification 3.8.11 helps ensure that any such experiment falls within the initiating conditions (assumptions) of the accident according to SAR section 13.2.2.

The NRC staff notes that SAR section 6, “Design Bases and Engineered Safety Features,” states that plutonium capsule fueled experiments are not performed at the NTR and SAR section 13.2.2 states that the assumption used in the DBA for the experimental accident is 50 milligrams of U-235 powder in a singly encapsulated container. Since the assumption for the experimental DBA is encapsulated uranium, the NRC staff finds that the proposed change to TS 3.8, Specification 3.8.11 is consistent with the SAR and acceptable.

5.3.8.12 Chemical Energy from Flammable Materials

Proposed TS 3.8, Specification 3.8.12, "Chemical Energy from Flammable Materials," states:

> The potential *REACTIVITY WORTH* of any component which could be ejected from the reactor by a chemical reaction *SHALL* be less than $0.50.
>
> The maximum possible chemical energy release from the combustion of *FLAMMABLE* materials contained in any *EXPERIMENTAL FACILITY SHALL* not exceed 1000 kW-sec. The total possible energy release from chemical combination or decomposition of substances contained in any experimental capsule *SHALL* be limited to 5 kW-sec, if the rate of the reaction in the capsule could exceed 1 W. *EXPERIMENTAL FACILITIES* containing *FLAMMABLE* materials *SHALL* be vented external to the reactor graphite pack.

The NRC staff determined that the licensee's limit that the potential reactivity worth of any component which could be ejected from the reactor by a chemical reaction shall be less than $0.50 was reviewed and found acceptable, as documented in section 5.6.4, "Experiments," and section 14.11, "Flammable or Explosive Device," of the SE for LA No. 21. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.12 is acceptable.

5.3.8.13 Experiment Approval

Proposed TS 3.8, Specification 3.8.13, "Experiment Approval," states:

> A written description and analysis of the possible hazards involved for each type of *EXPERIMENT SHALL* be evaluated and approved by the area manager, or his designated alternate, before the *EXPERIMENT* is conducted.

The NRC staff finds that proposed TS 3.8, Specification 3.8.13 remains unchanged since the issuance of the previous license renewal by LA No. 21. The NRC staff's review of this issue is documented in section 11.2, "Experiment Review," of the SE for LA No. 21. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.13 is acceptable.

5.3.8.14 Experiment Interference in Reactor Shutdown

Proposed TS 3.8, Specification 3.8.14, "Experiment Interference in Reactor Shutdown," states:

> No irradiation *SHALL* be performed which could credibly interfere with the scram action of the *SAFETY RODS* at any time during *REACTOR OPERATION*.

The NRC staff finds that proposed TS 3.8, Specification 3.8.14 remains unchanged since the issuance of the previous license renewal by LA No. 21. The NRC staff's review of the requirements to help ensure that experiments are performed safely is documented in section 5.6.4 of the SE for LA No. 21, which included limits on the reactivity worth of all experiments, including non-secured and secured experiments. Additionally, NUREG-1537, Part 1, appendix 14.1, section 3.8.1, "Reactivity Limits," provides guidance that reactivity limits should be specified on secured, unsecured, and movable experiments. The NRC staff's review of proposed TS 3.8, Specification 3.8.14 finds that ensuring the scram operation of the safety rods during any irradiation experiment helps maintain the ability to safely shut down the reactor. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.14 is acceptable.

5.3.8.15 Experiment Radiation Limits

Proposed TS 3.8, Specification 3.8.15, “Experiment Radiation Limits,” states:

> The radioactive material content, including fission products, of any singly encapsulated *EXPERIMENT* to be utilized in the *EXPERIMENTAL FACILITIES SHALL* be limited, so that the complete release of all gaseous, particulate, or volatile components from the encapsulation could not result in doses in excess of 10% of the equivalent annual doses stated in 10 CFR Part 20. This dose limit applies to persons occupying unrestricted areas continuously for 2 hours starting at time of release or restricted areas during the length of time required to evacuate the restricted area.

The NRC staff finds that proposed TS 3.8, Specification 3.8.15 remains unchanged since the issuance of the previous license renewal by LA No. 21. The NRC staff’s review of this issue is documented in section 14.1.3, “Results,” of the SE for LA No. 21. Therefore, the NRC staff concludes that proposed TS 3.8, Specification 3.8.15 is acceptable.

5.4 Surveillance Requirements

5.4.0 General Surveillance Intervals

Proposed TS 4.0, “General Surveillance Intervals,” states:

> Surveillances SHALL not exceed their defined *SURVEILLANCE INTERVALS* (Refer to Definitions, 1.2.) unless deferred according to Surveillance Requirements 4.0.1 or 4.0.2.

Proposed TS 4.0.1, “Deferred Operating Surveillances,” states:

> Surveillance (except those required for safety while in *REACTOR SHUTDOWN*) *MAY* be deferred during a period which the reactor is shutdown, except, for Table 4-2 Items 2, 4, and 5 (Test and Calibration), and Surveillance Requirement 4.7.1 (Test and Calibration). Deferred surveillances *SHALL* be completed prior to reactor startup unless *REACTOR OPERATION* is required for performance of the surveillance. These surveillances *SHALL* be performed as soon as practical after startup.

Proposed TS 4.0.2, “Deferred Shutdown Surveillances,” states:

> Scheduled surveillances which cannot be performed with the *REACTOR OPERATING*, *MAY* be deferred until the subsequent scheduled *REACTOR SHUTDOWN*.

The NRC staff reviewed proposed TS 4.0, TS 4.0.1, and TS 4.0.2 and finds that the proposed TSs are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4, “Surveillance Requirements,” and ANSI/ANS-15.1-2007, section 4, “Surveillance requirements,” which states that surveillances should be performed to ensure the operability of the LCOs, surveillance periods should be specified, surveillances may be deferred, if necessary, based on the operation of the facility, but must be performed prior to the reactor being considered operable, and surveillances should be performed following modification or repairs as part of the operability for that component or system. The NRC staff also finds that the surveillances in proposed TS table 4-2, “Surveillance Requirements of Reactor Safety-Related Items (Information Instruments),” item 2, “Fuel Loading Tank Water Level,” item 4, “Primary Coolant Temperatures (TC-2 & TC-5),” and item 5, “Stack Radioactivity (Gas and particulate *CHANNELS*),” and proposed TS 4.7, “Radiation Monitoring Systems and Effluents,” Specification 4.7.1, “Monitoring Systems During Reactor Operations,” should not be deferred as a result of an extended reactor shutdown and are necessary to ensure that the reactor coolant system and radiation monitoring systems remain operable to support continued monitoring of the reactor and radioactivity. Based on the above, the NRC staff concludes that proposed TS 4.0, TS 4.0.1, and TS 4.0.2 are acceptable.

5.4.1 Reactor Core Parameters

5.4.1.1. Potential Excess Reactivity

Proposed TS 4.1, “Reactor Core Parameters,” Specification 4.1.1, “Potential Excess Reactivity,” states:

> *POTENTIAL EXCESS REACTIVITY SHALL* be calculated before each startup. Actual critical rod position *SHALL* then be used to verify that the *MEASURED VALUE* is ≤ $0.76.

The NRC staff reviewed proposed TS 4.1, Specification 4.1.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.1, “Reactor Core Parameters,” item (1), “Excess Reactivity,” which states that excess reactivity should be determined at least annually and after changes in the core, in-core experiments, or control rods when a predicted reactivity exceeds the absolute value of the shutdown margin. The NRC staff also finds that the licensee has proposed a more conservative surveillance test period of prior to each startup. Based on the above, the NRC staff concludes that proposed TS 4.1, Specification 4.1.1 is acceptable.

5.4.1.2 Subcritical Rod Position

Proposed TS 4.1, Specification 4.1.2, “Subcritical Rod Position,” states:

> The reactor *SHALL* be placed in *REACTOR SHUTDOWN* if it is not in a subcritical condition with all four *SAFETY RODS* withdrawn and all *CONTROL RODS* inserted during every reactor startup. *SAFETY ROD* withdrawal *SHALL* be stopped if it appears criticality will be reached before all *SAFETY RODS* are withdrawn.

The NRC staff finds that although the subcritical rod position described in proposed TS 4.1, Specification 4.1.2 is not a typical or standard term associated with reactivity control, for the NTR it provides an additional margin of safety as its application confirms that the negative reactivity associated with the safety rods will be sufficient to shut down the reactor. The NRC staff also finds that performing the surveillance test during every startup helps ensure that proposed TS 4.1, Specification 4.1.2 is maintained effective. Based on the above, the NRC staff concludes that proposed TS 4.1, Specification 4.1.2 is acceptable.

5.4.1.3 Minimum Shutdown Margin

Proposed TS 4.1, Specification 4.1.3, “Minimum Shutdown Margin,” states:

> The minimum *SHUTDOWN MARGIN SHALL* be determined by calculation or measurement biennially or whenever a decrease in the reactivity worth of a *SAFETY ROD* is suspected.

The NRC staff reviewed proposed TS 4.1, Specification TS 4.1.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.1, item (2), “Shutdown Margin,” which states that the shutdown margin should be determined at least annually and after changes in either the core, in-core experiments, or control rods. The NRC staff also finds that the licensee’s shutdown margin determination, which is performed biennially or whenever a decrease in the reactivity worth of a safety rod is suspected, is appropriate since the NTR does not perform any core alterations or fuel movements. Finally, the NRC staff finds that the operation of the NTR results in a very minimal loss of safety rod reactivity since the safety rods are usually removed (pulled out) from the core. Based on the above, the NRC staff concludes that proposed TS 4.1, Specification 4.1.3 is acceptable.

5.4.2 Reactor Control and Safety System

5.4.2.1 Rods Operable

Proposed TS 4.2, “Reactor Control and Safety System,” Specification 4.2.1, “Rods Operable,” states:

> Each *SAFETY ROD* and *CONTROL ROD* drive *SHALL* be tested for operability annually.

The NRC staff finds that proposed TS 4.2, Specification 4.2.1 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (4), “Scram Times of Control and Safety Rods,” and ANSI/ANS-15.1-2007, section 4.2, “Reactor control and safety systems,” item (4) to perform annual operability tests. Therefore, the NRC staff concludes that proposed TS 4.2, Specification 4.2.1 is acceptable.

5.4.2.2 Safety Rod Withdrawal

Proposed TS 4.2, Specification 4.2.2, “Safety Rod Withdrawal,” states:

> The interlock which restricts *SAFETY ROD* withdrawal to one rod at a time, in the pre-determined sequence, *SHALL* be tested annually.

The NRC staff reviewed proposed TS 4.2, Specification 4.2.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2 and ANSI/ANS-15.1-2007, section 4.2, item (9) to annually test interlocks. Therefore, the NRC staff concludes that proposed TS 4.2, Specification 4.2.2 is acceptable.

5.4.2.3 Safety Rod Withdrawal Rate

Proposed TS 4.2, Specification 4.2.3, "Safety Rod Withdrawal Rate," states:

> The rate of withdrawal of each *SAFETY ROD SHALL* be measured annually.

The NRC staff finds that proposed TS 4.2, Specification 4.2.3 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (2), "Rod Withdrawal and Insertion Speeds," and ANSI/ANS-15.1-2007, section 4.2, item (2) to measure rod withdrawal times annually. Therefore, the NRC staff concludes that proposed TS 4.2, Specification 4.2.3 is acceptable.

5.4.2.4 Control Rod Withdrawal Rate

Proposed TS 4.2, Specification 4.2.4, "Control Rod Withdrawal Rate," states:

> The rate of withdrawal of each *CONTROL ROD SHALL* be measured annually.

The NRC staff finds that proposed TS 4.2, Specification 4.2.4 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (2) and ANSI/ANS 15.1-2007, section 4.2, item (2) to measure rod withdrawal times annually. Therefore, the NRC staff concludes that proposed TS 4.2, Specification 4.2.4 is acceptable.

5.4.2.5 Scram Time

Proposed TS 4.2, Specification 4.2.5, "Scram Time," states:

> The *SAFETY ROD SCRAM TIME SHALL* be measured semi-annually. The *SCRAM TIME SHALL* also be measured after any work is performed which could affect it.

The NRC staff finds that proposed TS 4.2, Specification 4.2.5 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (4) and ANSI/ANS-15.1-2007, section 4.2, item (4) to measure scram times semi-annually. Therefore, the NRC staff concludes that proposed TS 4.2, Specification 4.2.5 is acceptable.

5.4.2.6 Reactor Safety System and Safety-Related Items

Proposed TS 4.2, Specification 4.2.6, "Reactor Safety System and Safety-Related Items," states:

> Checks, tests and calibrations of the *REACTOR SAFETY SYSTEM* and safety-related items *SHALL* be performed as specified in Tables 4-1 and 4-2 of these Technical Specifications.

Table 4-1

SURVEILLANCE REQUIREMENTS OF REACTOR

SAFETY SYSTEM SCRAM INSTRUMENTS

Item No.	System	Surveillance	Frequency*
1.	Linear Power	*CHANNEL CHECK* (neutron source check)	Prior to SU
		CHANNEL TEST (high level trip test)	Prior to SU
		CHANNEL TEST (lack of high voltage)	Monthly
		CHANNEL CHECK (comparison against a heat balance)	Monthly
		CHANNEL CALIBRATION	Annual
2.	Log N	*CHANNEL CHECK*	Prior to SU
		CHANNEL TEST	Monthly
		CHANNEL CALIBRATION	Annually
3.	Primary Coolant Temperature (Fenwall)	*CHANNEL TEST*	Prior
		CHANNEL CALIBRATION	Annually
4.	Primary Coolant Flow	*CHANNEL CHECK*	Prior to SU
		CHANNEL TEST	Prior to SU
		CHANNEL CALIBRATION	Annually
5.	Manual	*CHANNEL TEST*	Prior to SU
6.	Electrical Power	*CHANNEL TEST*	Prior to SU

*Prior to placing into service an instrument which has been repaired or declared *INOPERABLE*, the instrument check, or test or calibration, as appropriate will be performed to demonstrate operability.

Table 4-2
SURVEILLANCE REQUIREMENTS OF REACTOR SAFETY-RELATED ITEMS (INFORMATION INSTRUMENTS)

Item No.	System	Surveillance	Frequency*
1.	Reactor Cell Pressure	*CHANNEL CHECK*	Prior to SU
		CHANNEL TEST	Quarterly
		CHANNEL CALIBRATION	Annually
2.	Fuel Loading Tank Water Level	*CHANNEL TEST*	Quarterly
3.	Primary Coolant Temperature (TC-7)	*CHANNEL TEST*	Quarterly
		CHANNEL CALIBRATION	Annually
4.	Primary Coolant Temperatures (TC2 & TC5)	*CHANNEL CHECK*	Monthly
		CHANNEL CALIBRATION	Annually
5.	Stack Radioactivity (Gas and particulate *CHANNELS*)	*CHANNEL CHECK*	Prior to SU
		CHANNEL TEST	Monthly
		CHANNEL CALIBRATION	Annually
6.	Linear Power – Low Power Rod Block Setpoint	*CHANNEL TEST*	Monthly
7.	*CONTROL ROD* or *SAFETY ROD* not IN	*CHANNEL TEST*	Annually
8.	*SAFETY ROD* Sequence	*CHANNEL TEST*	Annually
9.	Primary Coolant Conductivity	*CHANNEL CHECK*	Quarterly
		CHANNEL CALIBRATION	Biennially

*Prior to placing into service an instrument which has been repaired or declared *INOPERABLE*, the instrument check, or test, or calibration, as appropriate will be performed to demonstrate operability.

The NRC staff reviewed proposed TS 4.2, Specification 4.2.6, including proposed TS table 4-1 and table 4-2, and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.2, item (5), which states that scram channels, including scram actions with safety rod release and interlocks should be tested at least annually and prior to reactor startup.

Therefore, the NRC staff concludes that proposed TS 4.2, Specification 4.2.6, including proposed TS table 4-1 and table 4-2, is acceptable.

5.4.3 Reactor Coolant System

Proposed TS 4.3, "Reactor Coolant System," states:

> Specifications regarding surveillance requirements of the reactor coolant system for flow, fuel loading tank level, and conductivity are included in the *REACTOR SAFETY SYSTEM,* Surveillance Requirements Section 4.2, Tables 4-1 and 4-2.

The NRC staff reviewed proposed TS 4.3 and finds that it is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, section 4.2, item (5), which states that scram channels (for flow and temperature) should be tested annually and prior to reactor startup and section 4.3, item (6), which states that the primary coolant (for conductivity) should be measured quarterly. Therefore, the NRC staff concludes that proposed TS 4.3 is acceptable.

5.4.4 Confinement

Proposed TS 4.4, "Confinement," states: "This section left intentionally blank." The NRC staff concludes that proposed TS 4.4 is acceptable because the applicable confinement system SRs are included under TS 4.5, "Reactor Cell Ventilation and Confinement System."

5.4.5 Reactor Cell Ventilation and Confinement System

5.4.5.1 Reactor Cell Negative Pressure

Proposed TS 4.5, "Reactor Cell Ventilation and Confinement System," Specification 4.5.1, "Reactor Cell Negative Pressure," states:

> Surveillance requirements for the instrumentation and equipment required to comply with LCO 3.5.1 *SHALL* be tested as listed in Surveillance Requirements Section 4.2, Table 4-2, Item No. 1 & 5.

The NRC staff reviewed proposed TS 4.5, Specification 4.5.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.5, "Ventilation Systems," and ANSI/ANS-15.1-2007, section 4.5, "Ventilation systems," item (1) to perform a channel check prior to startup, a channel test quarterly, and a channel calibration annually of the reactor cell pressure and stack radioactivity systems. The NRC staff also finds that TS 4.5, Specification 4.5.1 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.4.2, "Confinement," to perform a functional test quarterly. Based on the above, the NRC staff concludes that proposed TS 4.5, Specification 4.5.1 is acceptable.

5.4.5.2 Reactor Cell Activity Release

Proposed TS 4.5, Specification 4.5.2, "Reactor Cell Activity Release," states:

> A *CHANNEL CHECK SHALL* be performed *DAILY* during activities that could release airborne radioactivity into the reactor cell.

The NRC staff reviewed proposed TS 4.5, Specification 4.5.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.5 and ANSI/ANS-15.1-2007, section 4.5, item (1) to perform a quarterly functional test. The NRC staff finds that a daily channel check during activities that could release airborne radioactivity into the reactor cell provides assurance that the reactor cell ventilation system will be available if needed. Based on the above, the NRC staff concludes that proposed TS 4.5, Specification 4.5.2 is acceptable.

5.4.6 Emergency Power

Proposed TS 4.6, "Emergency Power," states: "This section left intentionally blank." The NRC staff finds that proposed TS 4.6 is acceptable because the NTR does not require emergency power to perform any functions related to reactor safety given a loss of normal electric service, as reviewed and found acceptable in SER section 5.3.6. The NRC staff also finds that proposed TS 4.6 is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 1.2.2, "Format," to be numbered consistent with ANSI/ANS-15.1-2007, section 1.2.2, "Format," to include the heading and numbering format of major sections and the first level of subheadings to ensure that all items that may be relevant for inclusion in the TSs have been considered. Based on the above, the NRC staff concludes that proposed TS 4.6 is acceptable.

5.4.7 Radiation Monitoring Systems and Effluents

5.4.7.1 Monitoring Systems During Reactor Operations

Proposed TS 4.7, "Radiation Monitoring Systems and Effluents," Specification 4.7.1, "Monitoring Systems During Reactor Operations," states:

> Surveillances for the Area Radiation Monitors during *REACTOR OPERATIONS* include a *PRIOR to SU CHANNEL CHECK*, a *MONTHLY CHANNEL TEST*, and an *ANNUAL CHANNEL CALIBRATION*. Prior to placing into service an Area Radiation Monitor which has been repaired or declared *INOPERABLE*, the applicable surveillance will be performed to demonstrate it is *OPERABLE*.

The NRC staff reviewed proposed TS 4.7, Specification 4.7.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.7.1, "Monitoring Systems," and ANSI/ANS-15.1-2007, section 4.7.1, "Monitoring systems," to perform an operability check or channel test monthly and perform a channel calibration annually. Therefore, the NRC staff concludes that proposed TS 4.7, Specification 4.7.1 is acceptable.

5.4.7.2 Monitoring Systems During Reactor Cell Maintenance

Proposed TS 4.7, Specification 4.7.2, "Monitoring Systems During Reactor Cell Maintenance," states:

> A *CHANNEL CHECK SHALL* be performed *DAILY* during reactor cell maintenance.

The NRC staff reviewed proposed TS 4.7, Specification 4.7.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.7.1 to perform a channel check daily before reactor startup. Therefore, the NRC staff concludes that proposed TS 4.7, Specification 4.7.2 is acceptable.

5.4.7.3 Effluents – Environmental Monitoring

Proposed TS 4.7, Specification 4.7.3, "Effluents – Environmental Monitoring," states:

a. Monitoring of dose on *SITE* using thermoluminescent dosimeters or other equivalent devices *SHALL* be performed and documented annually.

b. Environmental monitoring (e.g., sampling of soil and vegetation) *SHALL* be performed and documented annually.

The NRC staff reviewed proposed TS 4.7, Specification 4.7.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.7.2, "Effluents," and ANSI/ANS-15.1-2007, section 4.7.2, "Effluents," to monitor dose using thermoluminescent dosimeters annually and to perform environmental monitoring (i.e., soil and vegetation sampling) annually. Therefore, the NRC staff concludes that proposed TS 4.7, Specification 4.7.3 is acceptable.

5.4.7.4 Effluents – Stack Release Activity

Proposed TS 4.7, Specification 4.7.4, "Effluents – Stack Release Activity," states:

The stack alarm *SHALL* be verified *MONTHLY*.

The NRC staff reviewed proposed TS 4.7, Specification 4.7.4 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.7.2 to confirm airborne radioactive effluents annually. Therefore, the NRC staff concludes that proposed TS 4.7, Specification 4.7.4 is acceptable.

5.4.7.5 Effluents – Stack Monitor Operability

Proposed TS 4.7, Specification 4.7.5, "Effluents – Stack Monitor Operability," states:

Stack activity monitors *SHALL* be performed according to Table 4-2, Item No. 5.

The NRC staff reviewed proposed TS 4.7, Specification 4.7.5 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.5 and ANSI/ANS-15.1-2007, section 4.5, item (1) to perform a channel check prior to startup, a channel test quarterly, and a channel calibration annually of the stack radioactivity monitoring systems. Therefore, the NRC staff concludes that proposed TS 4.7, Specification 4.7.5 is acceptable.

5.4.8 Experiments

Proposed TS 4.8, "Experiments," states:

Specific surveillance activities *SHALL* be established during the review and approval process as specified in Administrative Control 6.2.3, "Review Function" and are not part of the Technical Specifications.

The NRC staff reviewed proposed TS 4.8 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 4.8, "Experiments," and ANSI/ANS-15.1-2007, section 4.8, "Experiments," which states that the surveillance requirements for experiments need not be included explicitly in the TSs. Therefore, the NRC staff concludes that proposed TS 4.8 is acceptable.

5.5 <u>Design Features</u>

5.5.1 Site and Facility Description

Proposed TS 5.1.1, "Facility Location," states:

> The Nuclear Test Reactor (NTR) *FACILITY SHALL* be located on the *SITE* of the Vallecitos Nuclear Center (VNC).

Proposed TS 5.1.2, "Controlled Area and Restricted Area Terminology," states:

> The controlled area, as defined in 10 CFR Part 20 of the Commission's regulations, is the area within the VNC *SITE* boundary. The restricted area, as defined in 10 CFR Part 20 of the Commission's Regulations, is the NTR *FACILITY*.

Proposed TS 5.1.3, "Effluent Discharge," states:

> The discharge of all gaseous radioactive effluents *SHALL* be from the effluent stack at a minimum height of 45 feet (14 meters) above the grade level of Building 105.

The NRC staff reviewed proposed TS 5.1.1, TS 5.1.2, and TS 5.1.3 and finds that they are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 5, "Design Features," and ANSI/ANS-15.1-2007, section 5.1, "Site and facility description," which states that a general description of the site and facility, including the controlled area, and features such as the ventilation system release point should be described. The NRC staff also finds that the descriptions provided in proposed TS 5.1.1, TS 5.1.2, and TS 5.1.3 are consistent with the descriptions provided in the SAR. Based on the above, the NRC staff concludes that proposed TS 5.1.1, TS 5.1.2, and TS 5.1.3 are acceptable.

5.5.2 Reactor Primary Coolant System

5.5.2.1 Primary System Pressure

Proposed TS 5.2.1, "Primary System Pressure," states:

> The reactor coolant system is maintained at atmospheric pressure by a vent line to the holdup tank and the top of the fuel tank being open to the reactor cell.

The NRC staff reviewed proposed TS 5.2.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 5 and ANSI/ANS-15.1-2007, section 5.2, "Reactor coolant system," which states that the reactor coolant system should be described. The NRC staff also finds that the description provided in proposed TS 5.2.1 is consistent with the description provided in the SAR. Based on the above, the NRC staff concludes that proposed TS 5.2.1 is acceptable.

5.5.3 Reactor Core and Fuel

Proposed TS 5.3.1, “Control System,” states:

> The control system *SHALL* consist of four scrammable, spring-actuated *SAFETY RODS*, three nonscrammable *CONTROL RODS*, and *MANUAL POISON SHEETS*. Up to three *MANUAL POISON SHEETS* MAY be added or removed as needed to limit positive excess reactivity and compensate for reactivity loss from fuel burnup.
>
> (1) The *SAFETY RODS* and *CONTROL RODS SHALL* be boron carbide clad in stainless steel.
>
> (2) The *MANUAL POISON SHEETS SHALL* contain metallic cadmium.
>
> (3) Each installed *MANUAL POISON SHEET SHALL* be restrained in its respective graphite reflector slot in a manner which will prevent movement by more than ½ inch relative to the reactor core.
>
> (4) When the *CONTROL RODS, SAFETY RODS*, and *MANUAL POISON SHEETS* are inserted, they *SHALL* be located in the graphite reflector at the outer periphery of the core tank.

Proposed TS 5.3.2, “Reactor Fuel,” states:

> The core SHALL consist of 16 fuel element assemblies. Each fuel element assembly *SHALL* consist of 40 disks separated by spacers of varying widths on an aluminum support shaft. Other nominal specifications of the assemblies *SHALL* include the following:

Fuel	23.5% (by weight uranium) / 76.5% aluminum (by weight aluminum)
Enrichment	Approximately 93% U-235 (unburned)
Cladding	Aluminum, 0.027-inch thickness
Fuel disk active diameter	2.75 inch (OD)
Fuel disk spacing on shaft	0.24 to 0.27-inch, face-to-face

Proposed TS 5.3.3, “Core Reel Assembly,” states:

> The fuel assemblies *SHALL* be positioned in a reel assembly inside the core tank. The core reel assembly *SHALL* be rotated only when in *REACTOR SHUTDOWN* and by manual operation of a crank inside the NTR cell.

Proposed TS 5.3.4, “Temperature Coefficient of Reactivity,” states:

> The core is designed to exhibit a negative temperature coefficient of reactivity above 124°F, which is approximately the reactor steady-state operating temperature.

The NRC staff reviewed proposed TS 5.3.1, TS 5.3.2, TS 5.3.3, and TS 5.3.4 and finds that they are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 3.1, “Reactor

Core Parameters," item (5), "Reactivity Coefficients (Add by NRC)," and ANSI/ANS-15.1-2007, section 5.3, "Reactor core and fuel," which states, in part, that non-power reactors should specify reactivity coefficients. The NRC staff also finds that the descriptions provided in proposed TS 5.3.1, TS 5.3.2, TS 5.3.3, and TS 5.3.4 are consistent with the descriptions provided in the SAR. Based on the above, the NRC staff concludes that proposed TS 5.3.1, TS 5.3.2, TS 5.3.3, and TS 5.3.4 are acceptable.

5.5.4 Fissionable Material Storage

5.5.4.1 Fuel Storage

Proposed TS 5.4.1, "Fuel Storage," states:

> Fuel including fueled *EXPERIMENTS* and fuel devices not in the reactor *SHALL* be stored in a geometrical array where k_{eff} is no greater than 0.9 for all conditions of moderation and reflection using light water.

Proposed TS 5.4.1 establishes criticality limits for fissionable material storage as $k_{eff} \leq 0.9$ for all conditions of optimum moderation and full reflection using light water. These limits apply to all special nuclear material (SNM) at the NTR with the exception of SNM in the reactor core.

The NRC staff reviewed proposed TS 5.4.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 5 that accepts the guidance in ANSI/ANS-15.1-2007, section 5.4, "Fissionable material storage," which states that "[f]uel, including fueled experiments and fuel devices not in the reactor, shall be stored in a geometric array where k_{eff} is no greater than 0.90 for all conditions of moderation and reflection using light water…." Based on the above, the NRC staff concludes that proposed TS 5.4.1 is acceptable.

5.6 Administrative Controls

5.6.1 Organization

Proposed TS 6.1, "Organization," states:

> The NTR *SHALL* be owned and operated by the *LICENSEE* with management and operations organization as shown in Figure 6-1.

The NRC staff reviewed proposed TS 6.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1, "Organization," and ANSI/ANS-15.1-2007, section 6.1, "Organization," which states that the responsibilities should be specified. Based on the above, the NRC staff concludes that proposed TS 6.1 is acceptable.

5.6.1.1 Structure

Proposed TS 6.1.1, “Structure,” is Figure 6-1, “Facility Organization,” which is reproduced below:

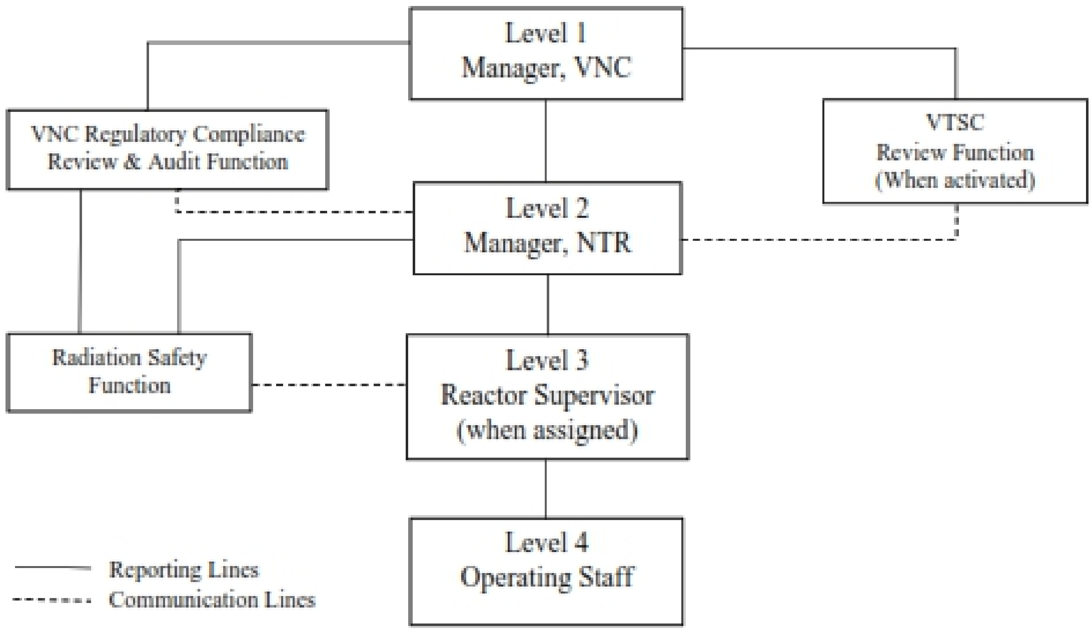

Figure 6-1 Facility Organization

The NRC staff reviewed proposed TS 6.1.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1.1, “Structure,” and ANSI/ANS-15.1-2007, section 6.1.1, “Structure*,”* which states that the structure should follow the guidance in the Figure 1 organization chart. Based on the above, the NRC staff concludes that proposed TS 6.1.1 is acceptable.

5.6.1.2 Responsibilities

Proposed TS 6.1.2, “Responsibilities,” states:

(1) The Level 1 manager *SHALL* be responsible for the NTR *FACILITY LICENSE*.

(2) The Level 2 manager is designated the area manager for the NTR and *SHALL* be responsible for the overall safe operation and maintenance of the *FACILITY*.

(3) The Level 3 Reactor supervisor (if utilized) is the individual responsible for supervising daily operations. In the absence of this position, the Level 2 manager is responsible for supervising daily operations.

(4) The Level 4 Operations staff includes *SENIOR REACTOR OPERATORS, REACTOR OPERATORS,* and trainees.

(5) Responsibilities of one level *MAY* be assumed by alternates when designated in writing.

(6) Functions performed by one level *MAY* be performed by a higher level, provided the minimum qualifications are met (e.g., *SENIOR REACTOR OPERATOR LICENSE*).

The NRC staff reviewed proposed TS 6.1.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1.2, "Responsibility," and ANSI/ANS-15.1-2007, section 6.1.2, "Responsibility," which states that the review and audit function reports to the Level 1, radiation safety personnel report to the Level 2 or higher, and the Levels 1-4 have the following responsibilities:

- Level 1, individual responsible for the license, which is the VNC Manager;
- Level 2, individual responsible for the reactor facility operation, which is the NTR Manager;
- Level 3, individual responsible for day-to-day operation, which is the Reactor Supervisor; and
- Level 4, operating staff, which are the NTR Reactor Operators and Senior Reactor Operators.

Based on the above, the NRC staff concludes that proposed TS 6.1.2 is acceptable.

5.6.1.3 Staffing

Proposed TS 6.1.3, "Staffing," states:

(1) The minimum staffing when the REACTOR IS NOT SECURED (Refer to *REACTOR SECURED*.) *SHALL* be composed of:

- A *LICENSED REACTOR OPERATOR* in the control room.
- A second person present at the *SITE* who is familiar with the VNC Radiological Emergency Plan and Emergency Procedures relevant to the NTR and is capable of carrying out *FACILITY* written procedures.
- A *LICENSED SENIOR REACTOR OPERATOR SHALL* be present at the NTR *FACILITY*, or a *READILY AVAILABLE SENIOR REACTOR OPERATOR* designated.

(2) A list of reactor *FACILITY* personnel by name and telephone number *SHALL* be available in the control room for use by the operator and includes:

- Management personnel
- Radiation safety personnel

- Other operations personnel

(3) A *LICENSED SENIOR REACTOR OPERATOR SHALL* be present at the NTR *FACILITY* during the following events:

- first daily startup and approach to power
- recovery from an *UNSCHEDULED SHUTDOWN*
- all reactor fuel, *SAFETY ROD*, and *CONTROL ROD* relocations within the reactor core region
- *MANUAL POISON SHEET* changes
- relocation of any *EXPERIMENT* or *FACILITY* changes with a *REACTIVITY WORTH* greater than one dollar.

The NRC staff reviewed proposed TS 6.1.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1.3, "Staffing," and ANSI/ANS-15.1-2007, section 6.1.3, "Staffing," which provides the minimum staffing levels for the reactor when it is not secured, the requirements of the second person at the facility and the on-call person who is not at the facility, the on-call list of management, radiation, and reactor personnel, and the list of events that require the presence of the senior reactor operator. Based on the above, the NRC staff concludes that proposed TS 6.1.3 is acceptable.

5.6.1.4 Selection and Training of Personnel

Proposed TS 6.1.4, "Selection and Training of Personnel," states:

> The selection, training and requalification of operations personnel *SHALL* meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS 15.4-2016, and the latest revision of the *FACILITY* Operator Requalification Program.

The NRC staff reviewed proposed TS 6.1.4 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.1.4, "Selection and Training of Personnel," and ANSI/ANS-15.1-2007, section 6.1.4, "Selection and training of personnel," which states that the selection, training, and requalification of operations personnel should meet or exceed the requirements of ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors." Based on the above, the NRC staff concludes that proposed TS 6.1.4 is acceptable.

5.6.2 Review and Audit

5.6.2.1 Composition and Qualifications

Proposed TS 6.2.1, "Composition and Qualifications," states:

(1) The RC organization *SHALL* conduct routine audits and perform periodic reviews of the implementation of these Technical Specifications.

(2) The Vallecitos Technological Safety Council (VTSC), at the direction of the Level 1 manager, *SHALL* perform independent reviews to ensure proper ongoing operation of the NTR.

(3) The VTSC *SHALL* not have more than half of its members from either Operations or RC Organizations.

(4) The VTSC *SHALL* be composed of a minimum of three members.

(5) VTSC members and alternates *SHALL* be appointed by the Level 1 manager.

(6) VTSC members *SHALL* collectively represent a broad spectrum of expertise in the appropriate reactor technology.

(7) Qualified and approved alternates *MAY* serve in the absence of regular members.

The NRC staff reviewed proposed TS 6.2.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2.1, "Composition and Qualifications," and ANSI/ANS-15.1-2007, section 6.2.1, "Composition and qualifications," which states that the review and audit group should be composed of a minimum of three members if a single group is used; the members should collectively represent a broad spectrum of expertise in the appropriate reactor technology; members and alternates should be appointed by and report to Level I management; and individuals may be either from within or outside the operating organization. Based on the above, the NRC staff concludes that proposed TS 6.2.1 is acceptable.

5.6.2.2 Charter and Rules

Proposed TS 6.2.2, "Charter and Rules," states:

The VTSC functions *SHALL* be conducted under a written charter including provision for:

(1) A meeting frequency of not less than once per calendar year.

(2) Allowing only one vote for each member or alternate for each issue reviewed.

(3) Quorum rules whereby a quorum is at least one-half of the voting members, and the NTR operations staff doesn't constitute a majority of the quorum.

(4) The use of support organizations.

(5) Maintenance of records; including the dissemination, review, and approval of minutes.

The NRC staff reviewed proposed TS 6.2.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2.2, "Charter and Rules," and ANSI/ANS-15.1-2007, section 6.2.2, "Charter and rules," which states that the meeting frequency will not be less than once per year; a quorum will consist of not less than one half of the voting membership, where the operating staff does not constitute a majority; and the meeting minutes should be reviewed and approved. Based on the above, the NRC staff concludes that proposed TS 6.2.2 is acceptable.

5.6.2.3 Review Function

Proposed TS 6.2.3, “Review Function,” states:

> Activities requiring review *SHAL*L include the following:
>
> (1) Determinations that proposed changes in equipment, systems, tests, *EXPERIMENTS*, or procedures are allowed without prior NRC approval as determined by 50.59 evaluation.
>
> (2) Determinations that new *EXPERIMENTs* or classes of *EXPERIMENTs* that could affect reactivity or result in the release of radioactivity do not require prior NRC approval as determined by 50.59 evaluation.
>
> (3) Determinations that proposed changes to the Fire Protection program as described in the Safety Analysis Report that do not require prior NRC approval, would not adversely affect the ability to achieve and maintain safe *REACTOR SHUTDOWN* of the NTR in the event of a fire as determined by 50.59 evaluation.
>
> (4) All new procedures and major revisions of existing procedures having safety significance that are required by the administrative control specifications in Administrative Controls Section 6.4.
>
> (5) Proposed changes to the Technical Specifications or the *FACILITY* operating *LICENSE.*
>
> (6) Violations of Technical Specifications, and *FACILITY LICENSE* requirements.
>
> (7) Unusual or abnormal occurrences which are reportable to the NRC under provisions of the Federal Regulations or Administrative Control 6.7.2.
>
> (8) Significant operating abnormalities or deviations from normal and expected performance of *FACILITY* equipment that affect, or could affect, nuclear safety.
>
> (9) Audit Reports.

The NRC staff reviewed proposed TS 6.2.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2.3, “Review Function,” and ANSI/ANS-15.1-2007, section 6.2.3, “Review function,” which states that the review function required by 10 CFR 50.59 is explicitly stated for: all new procedures; all new experiments; proposed changes to the TSs; violations of TSs or the license; operating abnormalities; reportable occurrences (proposed TS 6.7.2); and audit reports. Based on the above, the NRC staff concludes that proposed TS 6.2.3 is acceptable.

5.6.2.4 Audit Function

Proposed TS 6.2.4, "Audit Function," states:

> Audits *SHALL* include examination of operations records, logs, and documents as well as discussions with staff and observations as appropriate. Deficiencies *SHALL* be reported to the Level 1 manager as soon as identified and a written report of the findings of the audit submitted to the Level 1 manager within 3 months after the audit has been completed. The following *SHALL* be audited:
>
> (1) *FACILITY* operation for conformance to these Technical Specifications and applicable *LICENSE* conditions: at least once per calendar year not to exceed 15 months between audits.
>
> (2) Retraining and requalification program for the *LICENSED* operations staff: at least once every other calendar year not to exceed 30 months between audits.
>
> (3) The results of condition reports initiated relative to the NTR and operation of the NTR: once per calendar year not to exceed 15 months between audits.
>
> (4) The VNC Radiological Emergency Plan and implementing procedures: once every other year not to exceed 30 months between audits.

The NRC staff reviewed proposed TS 6.2.4 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.2.4, "Audit Function," and ANSI/ANS-15.1-2007, section 6.2.4, "Audit function," which states that the audit function will include an examination of facility operations for conformance to the TSs; retraining and requalification of operating staff; results of past corrective actions; the emergency plan; and deficiencies are immediately reported to the Level 1. Based on the above, the NRC staff concludes that proposed TS 6.2.4 is acceptable.

5.6.3 Radiation Safety

Proposed TS 6.3, "Radiation Safety," states:

> The Level 2 manager (or the Level 3 supervisor when assigned), in coordination with the VNC Radiation Safety Officer (RSO), *SHALL* be responsible for implementing the NTR radiation safety function. The RSO *SHALL* report relevant findings to the Level 2 manager, but SHALL report organizationally to the Manager, RC, thereby maintaining independence from the reactor operations organization. The radiation safety function is informed by the guidelines of the ANSI/ANS-15.11-2016, "Radiation Protection at Research Reactor Facilities."

The NRC staff reviewed proposed TS 6.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.3, "Radiation Safety," and ANSI/ANS-15.1-2007, section 6.3, "Radiation safety," which states that the individual responsible for implementation of the radiation safety program should use the guidance in ANSI/ANS-15.11, and should report to the Level 1 or 2. Based on the above, the NRC staff concludes that proposed TS 6.3 is acceptable.

5.6.4 Procedures

Proposed TS 6.4, “Procedures,” states:

> Written procedures *SHALL* be prepared, reviewed, and authorized prior to initiating any of the activities listed in this section. Because the VNC is a multi-license *FACILITY*, procedures implementing elements of *SITE*-wide programs (i.e., radiation protection, emergency planning, security) are authorized by the *SITE* Manager, RC. NTR-specific implementing procedures as components of those larger programs *SHALL* be authorized by the Level 2 manager according to Administrative Control 6.4.2. Procedures exclusive to the implementation of administrative and operational requirements of the NTR Licensing basis and their revisions *SHALL* be authorized by the Level 2 manager or his designated alternate(s) according to this section. Several of the activities in Administrative Control 6.4.1 *MAY* be included in a single manual or set of procedures or divided among various manuals or procedures.

Proposed TS 6.4 also includes proposed TSs 6.4.1 through 6.4.4.

5.6.4.1 Written Procedures

Proposed TS 6.4.1, “Written Procedures,” states:

> Written procedures *SHALL* be prepared for the following activities as required:
>
> (1) Startup, operation, and shutdown of the reactor.
>
> (2) Defueling, refueling, and fuel transfer operations, when required.
>
> (3) Preventive or corrective maintenance which could have an effect on the safety of the reactor, including the replacement of components.
>
> (4) Surveillance checks, tests, calibrations, and inspections required by the Technical Specifications.
>
> (5) NTR-specific radiation protection program implementing procedures for personnel safety consistent with applicable regulations or guidelines. Management commitment and programs to maintain exposures and releases as low as reasonably achievable *SHALL* be a component of the *SITE*-wide radiation protection program.
>
> (6) Administrative controls for operation and maintenance and the conduct of *EXPERIMENTS* that could affect reactor safety or core reactivity.
>
> (7) NTR-specific implementing procedures for the *SITE*-wide emergency and security plans.
>
> (8) NTR-specific radiation protection program implementing procedures for the use, receipt, and on-*SITE* transfer of by-product material for such activities performed under the R-33 *LICENSE*.

5.6.4.2 Level 2 Approval

Proposed TS 6.4.2, "Level 2 Approval," states:

(1) The Level 2 manager *SHALL* authorize all new procedures required by Administrative Control 6.4.1 before implementation.

(2) The Level 2 manager *SHALL* authorize all non-*ADMINISTRATIVE CHANGES* to procedures required according to Administrative Control 6.4.1.

5.6.4.3 Administrative Changes to Procedures

Proposed TS 6.4.3, "Administrative Changes to Procedures," states:

(1) *ADMINISTRATIVE CHANGES* to procedures required by Administrative Control 6.4.1 *MAY* be made by the Level 3 reactor supervisor or Level 2 manager before implementation.

(2) *ADMINISTRATIVE CHANGES* made by authorization of the Level 3 reactor supervisor *SHALL* be subsequently approved by the Level 2 manager.

5.6.4.4 Temporary Deviations

Proposed TS 6.4.4, "Temporary Deviations," states:

> Temporary deviations from established procedures *MAY* be made by a *LICENSED SENIOR REACTOR OPERATOR* in order to deal with special or unusual circumstances. These deviations *SHALL* be documented and reported to the Level 2 manager by the end of the next working day.

The NRC staff reviewed proposed TS 6.4 and finds that it follows the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.4, "Procedures," and ANSI/ANS-15.1-2007, section 6.4, "Procedures," which states that written procedures should be prepared and approved prior to use by the Level 2 for: operation of the reactor; fuel movement; maintenance of major components; surveillances required by the TSs; personnel radiation protection; administrative controls for operations and experiments; implementation of the security and emergency plans; and use of byproduct material. Based on the above, the NRC staff concludes that proposed TS 6.4 is acceptable.

5.6.5 Experiments Review and Approval

Proposed TS 6.5, "Experiments Review and Approval," consists of proposed TSs 6.5.1 through 6.5.3.

Proposed TS 6.5.1, "New Experiment Approval," states:

> All new *EXPERIMENT*s or class of *EXPERIMENT*s *SHALL* undergo review according to Administrative Control 6.2.3 and be approved in writing by the Level 2 manager or designee.

Proposed TS 6.5.2, "Changes to Experiments," states:

> Changes, except for *ADMINISTRATIVE CHANGES,* to *EXPERIMENT* implementing documents or to previously approved *EXPERIMENTS SHALL* undergo review according to Administrative Control 6.2.3 and be approved in writing by the Level 2 manager or designee.

Proposed TS 6.5.3, "Administrative Changes to Experiments," states:

> *ADMINISTRATIVE CHANGES* made to previously approved *EXPERIMENT* implementing procedures (e.g., ERs and EAFs) do not require independent review and *MAY* be approved by an *SRO*.

The NRC staff reviewed proposed TS 6.5 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.5, "Experiments Review and Approval," and ANSI/ANS-15.1-2007, section 6.5, "Experiments review and approval," which states that experiments should be conducted in accordance with approved procedures; all new experiments should be reviewed by the Level 2; substantive changes to previously approved procedures should be reviewed by the review group and approved by the Level 2; and minor changes may be made by the Level 3. Based on the above, the NRC staff concludes that proposed TS 6.5 is acceptable.

5.6.6 Required Actions

5.6.6.1 Actions to be Taken in Case of Safety Limit Violation

Proposed TS 6.6.1, "Actions to be Taken in Case of Safety Limit Violation," states:

(1) The reactor *SHALL* be placed in *REACTOR SHUTDOWN*, and *REACTOR OPERATION*s *SHALL* not be resumed until authorized by Level 1 management and the NRC.

(2) The safety limit violation *SHALL* be promptly reported to the Level 2 manager or designated alternates.

(3) The safety limit violation *SHALL* be reported to the NRC.

(4) A safety limit violation report *SHALL* be prepared. The report *SHALL* describe the following:

    (a) Applicable circumstances leading to the violation including, when known, the cause and contributing factors.

    (b) Effect of the violation upon reactor *FACILITY* components, systems, or structures and on the health and safety of personnel and the public.

    (c) Corrective action to be taken to prevent recurrence.

(5) The report *SHALL* be reviewed by the Manager, Regulatory Compliance (RC) or designee and any follow-up report *SHALL* be submitted to the NRC when authorization is sought to resume operation of the reactor.

The NRC staff reviewed proposed TS 6.6.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.6.1, "Action To Be Taken in Case of Safety Limit Violation," and ANSI/ANS-15.1-2007, section 6.6.1, "Action to be taken in case of a safety limit violation," which states that in the event of a SL violation, the reactor shall be shut down and not operated until approved by the NRC; the SL violation shall be promptly reported to the Level 2 and NRC; a report shall be prepared describing the violation and corrective actions; and SL violation will be reviewed by the review group and submitted to the NRC when authorization for operation is requested. Based on the above, the NRC staff concludes that proposed TS 6.6.1 is acceptable.

5.6.6.2 Action to be taken in the event of an occurrence of the type identified in Section 6.7.2(1)b and 6.7.2(1)c

Proposed TS 6.6.2, "Action to be taken in the event of an occurrence of the type identified in Section 6.7.2(1)b and 6.7.2(1)c," states:

> (1) Reactor conditions *SHALL* be returned to normal or the reactor *SHALL* be placed in *REACTOR SHUTDOWN*. If *REACTOR SHUTDOWN* is necessary to correct the occurrence, operations *SHALL* not be resumed unless authorized by the Level 2 manager or the Level 1 manager.
>
> (2) Occurrence *SHALL* be reported to the area manager and to the NRC addressed in accordance with 10 CFR 50.4.
>
> (3) Occurrence *SHALL* be reviewed by the Manager, RC, or designee, or the VTSC at its next scheduled meeting.

The NRC staff reviewed proposed TS 6.6.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.6.2, "Action To Be Taken in the Event of an Occurrence of the Type Identified in Sections 6.7.2(1)(b) and 6.7.2(1)(c)," and ANSI/ANS-15.1-2007, section 6.6.2, "Action to be taken in the event of an occurrence of the type identified in Secs. 6.7.2(1)(b) and 6.7.2(1)(c)." Based on the above, the NRC staff concludes that proposed TS 6.6.2 is acceptable.

5.6.7 Reports

5.6.7.1 Operating Reports

Proposed TS 6.7.1, "Operating Reports," states:

> Annual operating report(s) *SHALL* be submitted to the NRC Document Control Desk. The report(s) *SHALL* include the following:
>
> (1) A narrative summary of reactor operating experience including the hours the reactor was critical and total energy produced.
>
> (2) The *UNSCHEDULED SHUTDOWNS* including, where applicable, corrective action taken to preclude recurrence.
>
> (3) Tabulation of major preventive and corrective maintenance operations having safety significance.

(4) A summary report in accordance with 10 CFR 50.59(d)(2).

(5) A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge. The summary *SHALL* include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is <25% of the concentration allowed or recommended, a statement to this effect is sufficient.

(6) Summarized results of environmental surveys performed outside the *FACILITY*.

(7) A summary of exposures received by *FACILITY* personnel and visitors where such exposures are greater than 25% of that allowed or recommended.

The NRC staff reviewed proposed TS 6.7.1 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.7.1, "Operating Reports," and ANSI/ANS-15.1-2007, section 6.7.1, "Operating reports." Based on the above, the NRC staff concludes that proposed TS 6.7.1 is acceptable.

5.6.7.2 Special Reports

Proposed TS 6.7.2, "Special Reports," states:

Special reports are used to report unplanned events as well as planned major *FACILITY* and administrative changes. The following special reports *SHALL* be forwarded to the NRC addressed in accordance with 10 CFR 50.4:

(1) There *SHALL* be a report not later than the following working day by telephone and confirmed in writing by telegraph or similar conveyance to the NRC, to be followed by a written report within 14 days, that describes the circumstances of any of the following events:

a. Violation of safety limit

b. Release of radioactivity from the *SITE* above allowed limits.

c. Any of the following:

i. Operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in the Technical Specifications.

ii. Operation in violation of limiting conditions for operation established in the Technical Specifications unless prompt remedial action is taken.

iii. A *REACTOR SAFETY SYSTEM* component malfunction which renders or could render the *REACTOR SAFETY SYSTEM* incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or *REACTOR SHUTDOWN* periods.

NOTE: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems are not considered reportable provided that the minimum numbers of components or systems specified or required perform their intended reactor safety function.

iv. An unanticipated or uncontrolled change in reactivity greater than $0.50.

v. Abnormal and significant degradation in reactor fuel, cladding, or coolant boundary, which could result in exceeding prescribed radiation limits for personnel or the environment.

vi. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an *UNSAFE CONDITION* with regard to *REACTOR OPERATION*s.

(2) There *SHALL* be a written report within 30 days to the NRC for:

a. Permanent changes in the *FACILITY* organization involving Level 1 or Level 2 management.

b. Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

The NRC staff reviewed proposed TS 6.7.2 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.7.2, “Special Reports,” and ANSI/ANS-15.1-2007, section 6.7.2, “Special reports.” Based on the above, the NRC staff concludes that proposed TS 6.7.2 is acceptable.

5.6.8 Records

Proposed TS 6.8, “Records,” states:

Records *MAY* be in the form of logs, data sheets, or other suitable forms. The required information *MAY* be contained in single, or multiple records, or a combination thereof.

Proposed TS 6.8 also includes proposed TSs 6.8.1 through 6.8.3.

Proposed TS 6.8.1, “Records to be retained for a period of at least five years or for the life of the component, whichever is less:,” states:

(1) Normal reactor *FACILITY* operation (supporting documents such as checklists, log sheets, etc., *SHALL* be maintained for a period of at least one year).

(2) Principal maintenance operations.

(3) Reportable occurrences.

(4) Surveillance activities required by the Technical Specifications.

(5) Reactor *FACILITY* radiation and contamination surveys where required by applicable regulations.

(6) EXPERIMENTS performed with the reactor.

(7) Fuel inventories, receipts, and shipments.

(8) Approved changes in operating procedures.

(9) Records of meeting and audit reports of the review and audit groups.

The NRC staff reviewed proposed TSs 6.8 and 6.8.1 and finds that they are consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.8, “Records,” and ANSI/ANS-15.1-2007, section 6.8.1, “Records to be retained for a period of at least 5 years or for the life of the component involved if less than 5 years.” Based on the above, the NRC staff concludes that proposed TSs 6.8 and 6.8.1 are acceptable.

Proposed TS 6.8.2, “Records of the requalification programs,” states:

> Records of the requalification programs *SHALL* be maintained in accordance with 10 CFR 55.59(c)(5).

The NRC staff reviewed proposed TS 6.8.2 and finds that it requires records of requalification programs to be maintained in accordance with the regulation at 10 CFR 55.59(c)(5). Based on the above, the NRC staff concludes that proposed TS 6.8.2 is acceptable.

Proposed TS 6.8.3, “Records to be Retained for the Lifetime of the Reactor *FACILITY*,” states:

> Note: Applicable annual reports, if they contain all the required information, *MAY* be used as records in this section.
>
> (1) Gaseous and liquid radioactive effluents released to the environs.
>
> (2) Off-*SITE* environmental-monitoring surveys required by the Technical Specifications.
>
> (3) Radiation exposure for all personnel monitored.
>
> (4) Drawings of the reactor *FACILITY*.

The NRC staff reviewed proposed TS 6.8.3 and finds that it is consistent with the guidance in NUREG-1537, Part 1, appendix 14.1, section 6.8 and ANSI/ANS-15.1-2007, section 6.8.3, “Records to be retained for the lifetime of the reactor facility.” Based on the above, the NRC staff concludes that proposed TS 6.8.3 is acceptable.

5.7 Conclusions

The NRC staff evaluated the proposed TSs as part of its review of the license renewal application for Facility Operating License No. R-33 for the NTR. The proposed TSs define certain features, characteristics, organizational and reporting requirements, and conditions governing the operation of the NTR. The proposed TSs are included in the renewed license as appendix A. The NRC staff evaluated the content of the proposed TSs to determine whether they met the requirements in 10 CFR 50.36. Based on its review, the NRC staff concludes that the proposed TSs meet the requirements of the regulations. The NRC staff also reviewed the format and content of the proposed TSs for consistency with the guidance in NUREG-1537, Part 1 and ANSI/ANS-15.1-2007. The NRC staff concludes that the proposed TSs are consistent with the guidance. The NRC staff based these conclusions on the following findings:

- As required by 10 CFR 50.36(a)(1), the licensee provided proposed TSs with its license renewal application and included appropriate summary statements of the bases or reasons for the TSs. The bases are included for reference on applicable TS pages, but are not part of the TSs as required by 10 CFR 50.36(a)(1).

- As required by 10 CFR 50.36(b), each license for a research reactor must include TSs and the TSs must be derived from the analyses and evaluation included in the SAR, and amendments thereto, and the Commission may include such additional specifications as it finds appropriate. The licensee provided proposed TSs derived from analyses in the SAR, as supplemented.

- The proposed TSs acceptably implement the recommendations of NUREG-1537, Part 1 and ANSI/ANS-15.1-2007 by using definitions that are acceptable and appropriate.

- The proposed TSs specify an SL on the fuel temperature and an LSSS for the reactor protection system to preclude reaching the SL that satisfy 10 CFR 50.36(c)(1) requirements.

- The proposed TSs contain LCOs on each item that meets one or more of the criteria specified in 10 CFR 50.36(c)(2)(ii).

- The proposed TSs contain SRs that satisfy the requirements of 10 CFR 50.36(c)(3).

- The proposed TSs contain design features that satisfy the requirements of 10 CFR 50.36(c)(4).

- The proposed TSs contain administrative controls that satisfy the requirements for 10 CFR 50.36(c)(5). The proposed administrative controls contain requirements for initial notification, written reports, and records that satisfy 10 CFR 50.36(c)(1), (2), and (7); and for special reports that the NRC staff deemed necessary in accordance with 10 CFR 50.36(c)(8).

The NRC staff reviewed the proposed TSs and finds that they are acceptable. The NRC staff concludes that normal operation of the NTR within the limits of the proposed TSs will not result in radiation exposures in excess of the limits in 10 CFR Part 20 for members of the public or for the NTR staff. The NRC staff also concludes that the proposed TSs provide reasonable assurance that the NTR will be operated as analyzed in the SAR, as supplemented; that adherence to the proposed TSs during the license renewal period will limit the likelihood of malfunctions and the potential accident scenarios analyzed in the SAR and discussed in SER section 4; that facility operation will be in accordance with the applicable regulations; and that the conduct of activities by the licensee will not endanger the facility staff or members of the public.

6 CONCLUSIONS

On the basis of its evaluation of the GE-Hitachi Nuclear Energy Americas LLC (GEH, the licensee) application for renewal of the Nuclear Test Reactor (NTR, the facility) license as discussed in the previous chapters of this safety evaluation report, the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff concludes the following:

- The application for license renewal by letter dated November 19, 2020, as supplemented by letters dated September 22, December 3, and December 16, 2021; April 4, April 22, and September 15, 2022; and January 27, March 24, April 21, April 27, and June 15, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (AEA), and the Commission’s rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR).

- The facility will operate in conformity with the application, as supplemented, and with the provisions of the AEA and the rules and regulations of the Commission.

- There is reasonable assurance that (1) the activities authorized by the renewed license can be conducted at the designated location without endangering the health and safety of the public and (2) such activities will be conducted in compliance with the rules and regulations of the Commission.

- The facility will continue to be useful in the conduct of research and development activities.

- The licensee is technically and financially qualified to engage in the activities authorized by the renewed facility operating license, in accordance with the rules and regulations of the Commission.

- The applicable provisions of 10 CFR Part 140, “Financial Protection Requirements and Indemnity Agreements,” have been satisfied.

- The issuance of this license is in accordance with 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,” and all applicable requirements have been satisfied, as documented in the Environmental Assessment and Finding of No Significant Impact published in the *Federal Register* on March 22, 2023 (88 FR 17274), with a minor correction published on June 1, 2023 (88 FR 35933), which concluded that renewal of the NTR license will not have a significant effect on the quality of the human environment.

- The receipt, possession, and use of byproduct and special nuclear materials as authorized by this renewed facility operating license will be in accordance with the Commission’s regulations in 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.”

- The issuance of the renewed facility operating license will not be inimical to the common defense and security or to the health and safety of the public.

7 REFERENCES

1. GE-Hitachi Nuclear Energy, “Nuclear Test Reactor License Renewal (R-33),” November 19, 2020, Agencywide Documents Access and Management System Accession No. ML21053A071 (Redacted).

2. GE-Hitachi Nuclear Energy, “General Electric Nuclear Test Reactor Safety Analysis Report,” NEDO 32740P, Rev. 3, chapters 1 through 8, November 19, 2020, ML20325A205 (Redacted).

3. GE-Hitachi Nuclear Energy, “General Electric Nuclear Test Reactor Safety Analysis Report,” NEDO 32740P, Rev. 3, chapters 9 through 16, November 19, 2020, ML20325A206 (Redacted).

4. GE-Hitachi Nuclear Energy, “Vallecitos Nuclear Center Environmental Report 2020,” July 2020, ML20325A195.

5. GE-Hitachi Nuclear Energy, “Technical Specifications for the Nuclear Test Reactor Facility License R-33,” NEDO 32765, Rev. 3, November 2020, ML20325A196.

6. U.S. Nuclear Regulatory Commission, Issuance of Amendment No. 18 to Facility License No. R-33, General Electric Nuclear Test Reactor, December 28, 1984, ML20111A480.

7. U.S. Nuclear Regulatory Commission, “Issuance of Amendment No. 21 to Facility License No. R-33, General Electric Nuclear Test Reactor,” April 20, 2001, ML003775776.

8. U.S. Nuclear Regulatory Commission, *Federal Register* Notice, GE-Hitachi Nuclear Test Reactor, “License renewal application; docketing; opportunity to request a hearing and to petition for leave to intervene; order imposing procedures,” January 10, 2023, ML22339A139.

9. U.S. Nuclear Regulatory Commission, “GE-Hitachi Nuclear Energy Americas, LLC – Regulatory Audit RE: The Reactor Description, Radiation Protection, and Accident Analysis in the Operating License Application,” July 26, 2021, and Enclosure 1, Regulatory Audit Plan, ML21197A139; Enclosure 2, Regulatory Audit Topics, ML21197A137.

10 U.S. Nuclear Regulatory Commission, “GE-Hitachi Nuclear Energy Americas, LLC – Regulatory Audit Regarding the Nuclear Test Reactor Technical Specifications in the License Renewal Application,” August 10, 2022, ML22221A199.

11. U.S. Nuclear Regulatory Commission, “GE-Hitachi Nuclear Energy Americas, LLC – Report on the Regulatory Audit RE: Nuclear Test Reactor Technical Specifications in Support of the License Renewal Application Review,” January 27, 2023, ML23026A325.

12. U.S. Nuclear Regulatory Commission, “GE-Hitachi Nuclear Energy Americas, LLC – Report on the Regulatory Audit Regarding the Renewal of the Nuclear Test Reactor Facility Operating License No. R-33,” April 13, 2023, ML23101A185.

13. GE Hitachi Nuclear Energy, “GEH Supplemental Information Supporting GE Nuclear Test Reactor License Renewal Audit – Audit Questions and Responses,” dated September 22, 2021, ML21265A247; Audit Information, ML21265A249.

14. GE Hitachi Nuclear Energy, “GE Nuclear Test Reactor Safety Analysis Report and Technical Specifications,” dated March 24, 2023, ML23086C024; Enclosure 1, Audit Information, ML23086C025; Enclosure 2, Proposed Technical Specifications, ML23086C026; and Enclosure 4, Updated Safety Analysis Report, ML23086C028 (Redacted).

15. U.S. Nuclear Regulatory Commission, “GE-Hitachi Nuclear Energy Americas, LLC – Request for Additional Information Regarding Renewal of Facility Operating License No. R-33,” November 2, 2021, ML21301A218.

16. GE Hitachi Nuclear Energy, “Requalification Program for the General Electric Nuclear Test Reactor,” dated December 3, 2021, ML21337A257.

17. U.S. Nuclear Regulatory Commission, “GE-Hitachi Nuclear Energy Americas LLC - Receipt and Supplemental Information Needed for Renewal Review of GE-Hitachi Nuclear Energy Americas LLC Nuclear Test Reactor Facility Operating License No. R-33,” March 11, 2021, ML21062A250.

18. GE Hitachi Nuclear Energy, “Response to Request for Additional Information for GE Nuclear Test Reactor License Renewal Application,” dated April 22, 2022, ML22112A237; Enclosure 3, Proposed Technical Specifications, ML22112A240.

19. GE Hitachi Nuclear Energy, “Response to Request for Public Docketing of Information Relating to GE Nuclear Test Reactor License Renewal,” dated September 15, 2022, ML22258A118; updated by GE Hitachi Nuclear Energy, “Public Docketing of Information Relating to GE Nuclear Test Reactor License Renewal,” dated March 27, 2023, ML23086C061; Enclosure 1, GEH Supplemental Information for Nuclear Test Reactor License Renewal Application, ML23086C062 (Redacted).

20. GE Hitachi Nuclear Energy, Nuclear Test Reactor Annual Operating Reports for calendar years (CYs):

    2021 – ML22118B146
    2020 – ML21088A323
    2019 - ML20234A326
    2018 – ML19081A042
    2017 – ML18108A251
    2016 – ML17095A289
    2015 – ML16176A110
    2014 – ML15089A343

21. U.S. Nuclear Regulatory Commission, Nuclear Test Reactor Inspection Reports for CYs:

2022 – ML22102A227
2021 – ML21118B022
2020 – ML20054A241
2019 – Not Available
2018 – ML19074A201
2017 – ML17208B061
2016 – ML16334A383
2015 – ML15232A653
2014 - ML14171A622

22. U.S. Nuclear Regulatory Commission, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," NUREG-1537, Parts 1 and 2, February 1996, ML042430055 and ML042430048, respectively.

23. U.S. Nuclear Regulatory Commission, SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," October 24, 2008, ML082550140.

24. U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum SECY-08-0161, "Review of Research and Test Reactor License Renewal Applications," March 26, 2009, ML090850159.

25. U.S. Nuclear Regulatory Commission, "Interim Staff Guidance for the Streamlined Review Process for License Renewal for Research Reactors," October 2009, ML092240244.

26. GE Hitachi Nuclear Energy, "GEH Revised Vallecitos Nuclear Center (VNC) Site Physical Security Plan (PSP)," April 4, 2022, ML22132A272 (cover letter only).

27. U.S. Nuclear Regulatory Commission, "GE-Hitachi Nuclear Energy Americas LLC – Approval of the Revised Physical Security Plan for the Vallecitos Nuclear Center," May 26, 2022, ML22144A196.

28. U.S. Nuclear Regulatory Commission, "General Electric-Hitachi Nuclear Energy Americas, LLC – Issuance of Amendment No. 24 to Facility License No. R-33 for Vallecitos Nuclear Center Radiological Emergency Plan," February 12, 2021, ML20125A077.

29 GE Hitachi Nuclear Energy, "Vallecitos Nuclear Center Reactor Facilities Radiological Emergency Plan," Rev. 1, June 9, 2021, ML23086C063 (Redacted).

30. GE Hitachi Nuclear Energy, "Requalification Program for the General Electric Nuclear Test Reactor (NTR)," June 21, 2021, ML21172A185.

31. U.S. Nuclear Regulatory Commission, "GE-Hitachi Nuclear Energy Americas LLC – Approval of the Revised Operator Requalification Program for the Nuclear Test Reactor," February 8, 2022, ML22020A389.

32. U.S. Nuclear Regulatory Commission, “The Vallecitos Boiling Water Reactor, the General Electric Test Reactor, the Nuclear Test Reactor, and the ESADA Vallecitos Experimental Superheat Reactor — Issuance of Conforming Amendments Regarding the Transfer of Ownership to GE-Hitachi Nuclear Energy Americas, LLC,” October 22, 2007, ML072400217.

33. U.S. Nuclear Regulatory Commission, “General Electric-Hitachi Nuclear Energy Americas, LLC–Issuance of Amendment No. 25 to Facility License No. R-33 for the Nuclear Test Reactor Regarding Release Unrestricted Land,” June 29, 2022, ML18243A462 (Package).

34. U.S. Nuclear Regulatory Commission, NUREG/CR-1756, “Technology, Safety and Costs of Decommissioning Reference Research and Test Reactors,” March 1982, ML080500542.

35. GE Hitachi Nuclear Energy, “Annual Statement of Non-availability of Federal Government Funding for Conversion from HEU to LEU for VNC Nuclear Test Reactor (NTR),” February 15, 2023, ML23046A142.

36. GE Hitachi Nuclear Energy, “GEH Supplemental Information Supporting GE Nuclear Test Reactor License Renewal Audit – Audit Questions and Responses,” January 27, 2023, ML23027A210; Enclosure 1, GE NTR License Renewal Questions and Responses, ML23027A211.

37. American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.11-1993; R2004, “Radiation Protection at Research Reactor Facilities.”

38. GE Hitachi Nuclear Energy, email to U.S. Nuclear Regulatory Commission, “RE: Review of proposed GEH LR license - Summary of Changes/Oath & Affirmation,” April 21, 2023, ML23111A233.

39. U.S. Nuclear Regulatory Commission, Regulatory Guide 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” November 1973, ML003740125.

40. U.S. Nuclear Regulatory Commission, “Order Approving Transfer of Licenses and Conforming Amendments Related to the Vallecitos Boiling Water Reactor, the General Electric Test Reactor, the Nuclear Test Reactor and the ESADA Vallecitos Experimental Superheat Reactor,” September 6, 2007, ML071500598 (Package).

41. GE Hitachi Nuclear Energy, “GEH Response for Request for Additional Information for NRC License No. R-33 Decommissioning Cost Estimate,” April 27, 2023, ML23117A331.

42. GE Hitachi Nuclear Energy, email to U.S. Nuclear Regulatory Commission, “RE: Action Request on ‘GEH NTR LR Issuance Package,’” June 15, 2023, ML23166B147