Safety Evaluation Report

Renewal of the Facility Operating License for the Nuclear Science Center Reactor

Docket No. 50-128

Texas Engineering Experiment Station/Texas A&M University System

United States Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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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 the Texas Engineering Experiment Station/Texas A&M University System (the licensee) for a 20-year renewal of Facility Operating License No. R-83 to continue operating the Nuclear Science Center Reactor (NSCR). In its safety review, the NRC staff considered information submitted by the licensee, past operating history recorded in the licensee's annual operating reports to the NRC, inspection reports prepared by NRC staff, as well as direct observations obtained during site visits. On the basis of this review, the NRC staff concludes that the licensee can continue to operate the NSCR for the term of the renewed facility license, in accordance with the license, without endangering the public health and safety, facility personnel, or the environment.

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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
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
Ar-41	argon-41
ARM	area radiation monitor
B_4C	boron carbide
C	Celsius
cm	centimeter
CEDE	committed effective dose equivalent
Ci	curie
Ci/yr	curies per year
cm/s	centimeters per second
DAC	derived air concentration
DDE	deep dose equivalent
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DPR	Division of Policy and Rulemaking
EP	emergency plan
F	Fahrenheit
FAM	facility air monitor
FLIP	Fuel Life Improvement Program
ft	feet
FY	fiscal year
GA	General Atomics
H	hydrogen
HEU	highly enriched uranium
I-125	iodine-125
IFE	instrumented fuel element
in.	inch
IR	inspection report
ISG	interim staff guidance

kW	kilowatt
kWt	kilowatt thermal
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
mR/h	milli-Roentgen per hour
mrem	millirem
mrem/hr	millirem per hour
MW	megawatt
MWD	megawatt-days
MWt	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
NSC	Nuclear Science Center
NSCR	Nuclear Science Center Reactor
PDR	Public Document Room
PRNC	Puerto Rico Nuclear Center
Pu	plutonium
RAI	request for additional information
RO	reactor operator
RSB	Reactor Safety Board
RSO	Radiation Safety Officer
SAR	safety analysis report
SDM	shutdown margin
SER	safety evaluation report
SL	safety limit
Sm	samarium
SNM	special nuclear material

SOI	statement of intent
SRM	staff requirements memorandum
SRO	senior reactor operator
SS	stainless steel
S.S.	steady state
TAMU	Texas A&M University
TAMUS	Texas A&M University System
TEDE	total effective dose equivalent
TEES	Texas Engineering Experiment Station
TRIGA	Training, Research, Isotope Production, General Atomics
TS	technical specification(s)
U	uranium
UZrH	uranium-zirconium hydride
W	watt
wt%	weight percent
Xe	xenon
Zr	zirconium

1 INTRODUCTION

1.1 Overview

In a letter dated February 27, 2003, as supplemented on July 22, 2009; July 28, August 30, August 31, and December 9, 2010; May 27, June 9, and November 21, 2011; January 12, April 11, and November 14, 2012; January 31, 2013; February 3, February 11, and November 13, 2014; and March 2, June 5, June 11, and June 30, 2015; the Texas Engineering Experiment Station/Texas A&M University System (TEES/TAMUS, or “the licensee”) submitted to the U.S. Nuclear Regulatory Commission (NRC, the Commission) a license renewal application (LRA) for a 20-year renewal of the Class 104c Facility Operating License No. R-83, NRC Docket No. 50-128, for the Nuclear Science Center (NSC, or “the facility”) TRIGA (Training, Research, Isotope Production, General Atomics)-type research reactor (NSCR or “the reactor”) (Ref. 1).

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.51(a) states that “Each 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 date-of-issuance.” The TEES/TAMUS holds the Facility Operating License No. R-83 originally issued on December 7, 1961, by the U.S. Atomic Energy Commission. The NRC reissued the TEES/TAMUS NSCR facility operating license on March 30, 1983, for a period of 20 years expiring on March 29, 2003. Because of the timely renewal provision in 10 CFR 2.109(a), the licensee is permitted to continue operating the NSCR under the terms and conditions of the current license until the NRC staff completes action on the license renewal request. A renewal would authorize continued operation of the NSCR for an additional 20 years.

The regulation in 10 CFR 50.64, “Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors,” requires NRC licensees of research and test reactors (RTRs) to convert from the use of HEU fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. In a letter dated December 29, 2005 (Ref. 2), the licensee submitted its application requesting that the NRC approve the HEU to LEU fuel conversion and related changes to the technical specifications (TSs). In its conversion application, the licensee included a safety analysis report (SAR) for the conversion (hereafter called “the Conversion SAR”) (Ref. 2) on which the change from HEU to LEU fuel and the TS changes were based. By letter dated June 1, 2006, the NRC staff issued a request for additional information (RAI) to the licensee (Ref. 3). The licensee responded to the RAI with supplemental information in letters dated July 17 (Ref. 4), August 4 (Ref. 5), and August 21, 2006 (Ref. 6).

The NRC staff issued an Order, as Amendment No. 17 to Facility Operating License No. R-83, for the TEES/TAMUS to convert the NSCR to operation using LEU fuel by letter dated September 1, 2006 (Ref. 7). Included with the Order was the NRC staff’s safety evaluation report (SER) (hereafter called “the Conversion SER”) and a requirement to submit a reactor startup report following the conversion to LEU fuel. The licensee submitted the startup report on August 19, 2010 (hereafter called “the Startup Report”) (Ref. 8). The NRC staff used the information provided in the Conversion SER and the Startup Report as the basis for some of the conclusions provided in this LRA SER (hereafter called “the SER”).

The NRC staff also based its review of the renewal of the TEES/TAMUS NSCR facility operating license on the information contained in the LRA, as well as supporting supplements and the licensee's responses to RAIs. The LRA, by letter dated February 27, 2003, included an SAR (Ref. 1), which was subsequently updated on July 22, 2009 (Ref. 9) and June 9, 2011 (Ref. 10). Throughout this SER, references to the NSCR SAR mean information collectively obtained from the LRA SAR (Ref. 1), the Conversion SAR (Ref. 2), and the updated SARs provided by Ref. 9 and Ref. 10, as described above. Additional LRA information was provided in the licensee's responses to the NRC staff's RAIs described below. The NRC staff forwarded RAIs in letters dated November 13, 2009 (Ref. 46); June 24, (Ref. 11) and October 6, 2010 (Ref. 12); September 29, 2011 (Ref. 47); January 11, 2012 (Ref. 48); December 5, 2013 (Ref. 50); January 7, (Ref. 51), January 8, (Ref. 52), June 19, (Ref. 54), July 15, (Ref. 55), November 5, (Ref. 56), and November 14, 2014 (Ref. 58); and May 7, 2015 (Ref. 64).

The licensee responded to the NRC staff's RAIs by letters dated July 28 (Ref. 13), August 30 (Ref. 39), August 31 (Ref. 14), and December 9, 2010 (Ref. 40); May 27 (Ref. 15), June 9 (Ref. 10), and November 21, 2011 (Ref. 16); January 12 (Ref. 17), April 11 (Ref. 18), and November 14, 2012 (Ref. 41); January 31, 2013 (Ref. 42); February 3 (Ref. 49), February 11 (Ref. 53), and November 13, 2014 (Ref. 57); and March 2 (Ref. 59) and June 5, 2015 (Ref. 65). The licensee also provided updated TSs in letters dated April 11 (Ref. 18) and November 14, 2012 (Ref. 41); and March 2 (Ref. 59) and June 5, 2015 (Ref. 65).

The licensee requested corrections to typographical errors in the proposed TSs by electronic mail, dated June 11 and June 30, 2015 (two requests) (Ref. 63). In addition, by telephone conversation on August 24, 2015 (Ref. 70), the licensee requested additional administrative changes to correct typographical errors and format changes to enhance readability. The NRC staff finds that these proposed changes were administrative in nature and did not change any technical attributes of the proposed TSs. These proposed changes primarily fixed typographical errors, rendered TS wording gender neutral, corrected upper/lower case text for consistency, or enhanced readability. The NRC staff concluded that the proposed changes were acceptable and updated the proposed TSs.

Although the LRA indicated that no changes to the physical security plan, emergency plan (EP), and operator requalification program were needed as a result of the LRA request, the NRC staff reviewed these plans to ensure they were consistent with current NRC regulations and guidance. The results of the NRC staff review of the physical security plan, the emergency plan, and the operator requalification program are discussed below. The NRC staff's LRA review also included information from the TEES/TAMUS NSC annual operating reports from 2005 through 2013 (Ref. 32) and NRC inspection reports (IRs) from 2005 through 2014. The NRC staff conducted site visits on December 15, 2010; March 22 and October 25, 2012; October 24, 2014; and February 3 and 4, and March 5, 2015; to observe facility conditions and to discuss NRC staff RAIs and licensee's RAI responses.

With the exception of the physical security plan and the EP, 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 maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. Documents related to this license renewal may be accessed through the NRC's Public Library on the Internet at http://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 send an e-mail to the PDR at resources@nrc.gov. The physical security plan is protected from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements," and the EP is withheld from public disclosure because it is considered security-related information. Because parts of the SAR and RAI responses from the licensee contain security-related information and are protected from public disclosure, redacted versions are available to the public.

Section 7, "References," of this SER contains the dates and associated ADAMS Accession Numbers of the licensee's renewal application and related supplements.

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," Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," Part 50, "Domestic Licensing of Production and Utilization Facilities," Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Part 70, "Domestic Licensing of Special Nuclear Material," and Part 73, "Physical Protection of Plants and Materials"; the recommendations of applicable 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 referred to the recommendations contained in NUREG-1537, "Guidance for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996 (Ref. 19). 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. 20), the NRC staff provided the Commission with information regarding plans to streamline the review of LRAs for research and test reactors (RTRs). The Commission issued its staff requirements memorandum (SRM)-SECY-08-0161, dated March 26, 2009 (Ref. 21). The SRM directed the NRC staff to streamline the renewal process for such RTRs, using some combination of the options presented in SECY-08-0161. The SRM also directed 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 should consider 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 of Research Reactors," to assist in the review of LRAs. The streamlined review process is a graded approach based on licensed power level. Under the streamlined review process, the facilities are divided into two tiers. Facilities with a licensed thermal power level of 2 megawatts (MWt) and greater, or requesting a power level increase, would undergo a full review using NUREG-1537. Facilities with a licensed thermal power level less than 2 MWt would undergo a focused review that centers on the most safety-significant aspects of the renewal application and relies on past NRC reviews for certain safety findings. The NRC made a draft of the ISG available for public comment, and the NRC staff considered public comments in its development of the final ISG. The NRC staff conducted the TEES/TAMUS LRA review using the guidance in the final ISG, dated October 15, 2009 (Ref. 22). Since the licensed thermal power level for the NSCR is less than 2 MWt, the NRC staff performed a focused review of the licensee's LRA. Specifically, the NRC focused on reactor design and operation, accident analysis, TSs, radiation protection, waste management programs, financial requirements, environmental assessment, and changes to the facility made after submittal of the application.

In the LRA, the licensee indicated no changes were needed to the TEES/TAMUS NSC's physical security plan. However, as part of its review of the LRA, the NRC staff reviewed the physical security plan entitled, "Texas Engineering Experiment Station/Texas A&M University System Nuclear Science Center Physical Security Plan, Revision 1," issued June 1979, as revised by letters dated January 15, 1980, and September 18, 1980, and as changed under 10 CFR 50.54(p). The NRC staff issued RAIs to the licensee in letters dated June 19, and November 14, 2014 (Ref. 54 and Ref. 58), and the licensee responded by letters dated June 24, 2014 and March 11, 2015 (Ref. 66 and Ref. 67), including a revised TEES/TAMUS NSC physical security plan. The NRC staff reviewed the revised TEES/TAMUS NSC physical security plan, found that it met the applicable regulations, and based on that finding concludes that the TEES/TAMUS NSC security plan, dated March 2015, is acceptable. The licensee maintains the program to provide the physical protection of the facility and its special nuclear material (SNM) in accordance with the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials." Changes to the physical security plan can be made, by the licensee, in accordance with 10 CFR 50.54(p), as long as those changes do not decrease the effectiveness of the plan. In addition, the NRC staff performs routine inspections of the licensee's compliance with the requirements of the physical security plan. The NRC staff's review of the TEES/TAMUS NSC IRs for the past several years identified no violations of the security plan requirements.

The licensee is required to maintain the EP, in compliance with 10 CFR 50.54(q) 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. As part of the LRA review, the NRC staff reviewed the existing TEES/TAMUS NSC EP, Second Revision, dated December 1999, and issued RAIs by letter dated January 7, 2014 (Ref. 51). The licensee responded by letter dated February 11, 2014 (Ref. 69), which resolved the NRC staff's questions, without requiring a revision to the EP. The NRC staff completed its review and, by letter dated May 5, 2014 (Ref. 60), acknowledged that the EP, Second Revision, dated December 1999, remains compliant with the regulations and applicable guidance. The NRC staff performs routine inspections of the licensee's compliance with the requirements of the emergency plan, and no violations have been identified for the past several years.

As part of the LRA review, the NRC staff reviewed the TEES/TAMUS NSC's Senior Reactor Operator and Reactor Operator Requalification Program, Revision 4, dated April 16, 1997, as revised by letter dated May 26, 1997. The NRC staff issued RAIs by letter dated January 8, 2014 (Ref. 52). The licensee responded by submitting TEES/TAMUS NSC Senior

Reactor Operator and Reactor Operator Requalification Program, Revision 5, by letter dated February 11, 2014 (Ref. 53). The NRC staff reviewed and approved the TEES/TAMUS NSC Senior Reactor Operator and Reactor Operator Requalification Program, Revision 5, by letter dated April 10, 2014 (Ref. 61).

The purpose of this SER is to summarize the findings resulting from the TEES/TAMUS NSCR safety review 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 TEES/TAMUS NSCR up to a steady-state thermal power level of 1.0 MWt and short-duration power pulses with a maximum reactivity insertion limit calculated not to raise the fuel temperature at the hottest core location above 830 degrees Celsius (C) (1,526 degrees Fahrenheit (F)).

This SER was prepared by A. Francis DiMeglio, Walter Meyer, and Geoffrey Wertz, Project Managers from the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking (DPR), Research and Test Reactors Licensing Branch, and Jo Ann Simpson and Aaron L. Szabo, Financial Analysts from the NRC's NRR, Division of Inspection and Regional Support, Financial and International Projects Branch. Brookhaven National Laboratory, the NRC's contractor, provided substantial input to this SER.

1.2 Summary and Conclusions on Principal Safety Considerations

The NRC staff's evaluation considered the information submitted by the licensee, including past operating history recorded in the licensee's annual operating 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 under consideration for the NSCR, 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 Chapter 4 of the SAR, as supplemented, in accordance with the TSs, are safe, and safe operation can reasonably be expected to continue.

- The licensee considered the expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident, emphasizing those that could lead to a loss of integrity of fuel element cladding and a release of fission products. The licensee performed analyses, using conservative assumptions, of the most serious credible accidents and the maximum hypothetical accident 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.

- 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 licensee's TSs, which provide limits controlling operation of the facility, are such that there is reasonable assurance that the facility will be operated safely and reliably. There has been no significant degradation of the reactor, as discussed in Chapter 4 of the SAR, as supplemented, and the TSs will continue to help ensure that there will be no significant degradation of safety-related equipment.

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

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

- The licensee maintains a physical security plan 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 the physical protection of the facility and its SNM.

- The licensee maintains an EP 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.

On the basis of these findings, the NRC staff concludes that there is reasonable assurance that the licensee can continue to operate the NSCR 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 personnel, or the environment. The issuance of the renewed license will not be inimical to the common defense and security.

1.3 General Description of the Facility

On December 7, 1961, the U.S. Atomic Energy Commission issued Facility Operating License No. R-83 to the TEES/TAMUS for the operation of the NSCR. The NRC subsequently renewed the license on March 30, 1983. The facility operating license authorized the licensee to operate the NSCR at a steady-state power level up to 1,000 kilowatts thermal (kWt) (1.0 MWt) and perform power pulsing with a maximum reactivity insertion limited to pulses that will not raise the fuel temperature at the hottest core location above 830 degrees C (1,526 degrees F). On September 1, 2006, the facility operating license was amended to permit the conversion of the reactor fuel from the use of HEU to LEU fuel.

The NSC is located in the city of College Station, Texas, in a dedicated building on a restricted site apart from the main university campus. Located within the restricted site boundary are the reactor confinement building, reception room, laboratory building, mechanical equipment room, cooling system equipment, holding tanks, and other storage and support buildings. In addition to the NSCR, the site also contains a linear accelerator and its associated laboratory, housed in a separate building about 125 feet (ft) from the NSCR structure. Operation of the linear accelerator is regulated by the State of Texas.

The NSCR was initially licensed with Material Test Reactor type fuel at a maximum thermal power level of 100 kWt. In 1968, the reactor was converted to use TRIGA fuel, the maximum steady-state power level was increased to 1,000 kWt (1.0 MWt), and pulsing operation to a reactivity insertion of $3.00 was authorized. In June 1973, the NRC amended the facility operating license to authorize the TEES/TAMUS to operate the NSCR with cores consisting of standard TRIGA fuel elements, Fuel Life Improvement Program (FLIP) fuel elements (HEU fuel), or a combination of both. The license for the mixed core permitted the TEES/TAMUS to operate the NSCR at a maximum steady-state power of 1.0 MWt with pulsing operation limited to reactivity insertions of $2.00.

In July 1975, the facility operating license was amended to allow for an increase in the maximum permissible pulse reactivity insertion up to $2.70. On September 27, 1976, during a fuel loading operation, four “lead” fuel elements failed to pass through a gauge test for transverse bend. Pulsing operation ceased until an analysis was completed (Ref. 26) in 1981, and NSC Facility Operating License Amendment No. 9 was issued which limited the maximum allowable reactivity insertion for pulsing to the amount that would not cause the peak reactor temperature to exceed a temperature limit of 830 degrees C (1,526 degrees F).

The NSCR is a heterogeneous pool-type nuclear reactor currently fueled with LEU TRIGA 30/20 fuel in four-element fuel bundles. The 30/20 TRIGA fuel is a uranium-zirconium hydride ($U\text{-}ZrH_{1.6}$) with 30 weight percent uranium (U) enriched to 19.75 percent U-235. (Note that the “x” in the UZrHx nomenclature represents the hydrogen to zirconium ratio and the hydrogen content is important because it influences many attributes of fuel behavior). The moderator is the zirconium hydride contained in the fuel and light water that also serves as the coolant, which circulates through the core by natural convection. The core is reflected by a large pool of water and graphite located at the periphery of the reactor core. The maximum allowable steady-state thermal power level is 1.0 MWt. The reactor core is submerged in a steel-lined concrete pool, which has a main section that is 18 ft (5.5 meter (m)) wide, and 33 ft (10 m) deep, and contains a stall section that is 9 ft (2.7 m) wide. A gate can isolate the two sections to allow draining of any one of the sections. Heat generated from the reactor core is directly transferred to the pool water by natural convection. Reactor pool water temperature in normal operation is maintained between 70 to 80 degrees F (21 to 26 degrees C) by a closed loop cooling system with a design flow rate of 1,000 gallons (3,785 liters (l)) per minute. A pump takes water from a pipe connected to the reactor pool, passes it through a plate-type heat exchanger, and returns it through a pipe to the reactor pool.

This system provides the heat removal capability for the water in the reactor pool. Heat is removed from the plate-type heat exchanger by the secondary cooling system loop. The secondary cooling water flows from the basin of the cooling tower through the plate-type heat exchanger and back to the cooling tower. The cooling tower uses evaporative cooling to remove heat from the secondary water to the atmosphere at the cooling tower. The secondary loop has a nominal flow rate of 1,575 gallons (5962 l) per minute. The cooling tower will deliver 28 degrees C (82 degrees F) water at 25 degrees C (77 degrees F) air temperature.

Design features of this system allow transfer of reactor heat from the primary system under all operating conditions, but the system is required only for core heat removal during power operations. These systems are controlled remotely from the control room. The reactor's experimental facilities include space adjacent to the reactor core, a pneumatic transfer system, a through tube, four beam tubes, and a thermal column. Six motor-driven control rods (four shim safety rods, a regulating rod, and a transient rod), all using boron carbide as the neutron absorber, are moved in and out of the reactor core by individual mechanical drives. The shim safety rods and the transient rod can be disengaged to drop by gravity into the core to scram the reactor. The transient control rod operates with a pneumatic-electromechanical drive that may be used either as a control or transient rod to generate a neutron pulse in pulsing mode operation.

1.4 Shared Facilities and Equipment

The NSC shares utilities with an accelerator facility that is part of the Nuclear Engineering Department. This accelerator facility shares a common wall and electrical distribution system with NSC auxiliary shops. This building is external to the NSCR confinement building. The NSC shares no other facilities or equipment. During the NRC staff site visits, no other shared facilities or equipment were identified.

1.5 Comparison with Similar Facilities

In Section 1.5 of the SAR (Ref. 1), the licensee provides general statements about the TRIGA-type nuclear reactors built by General Atomics (GA). The GA TRIGA is one of the most widely used research and training reactors in the United States. TRIGA reactors exist in a variety of configurations and capabilities (Ref. 23). The NSCR is very similar in design to TRIGA reactors at the University of Wisconsin and Washington State University. Instruments and controls used in the NSCR are similar in principle to most non-power reactors licensed by the NRC. The pool size and experimental facility configuration differ among the three reactors, but basic reactor behavior and accident analyses are similar.

The reactor at the University of Wisconsin was converted to the same LEU fuel elements used in the NSCR. The reactor at Washington State University was partially converted to the same LEU fuel elements. The TRIGA Mark F reactor at GA in San Diego was operating with a core partially made up of high-density LEU fuel prior to permanent shut down. The TRIGA fuel typically has no performance-related issues as long as the well-established operating and water quality limits are maintained (Ref. 24).

1.6 Summary of Operations

The NSCR is used for teaching, performing nuclear research, and providing a range of irradiation services. The licensee provides nuclear education for both undergraduate and graduate students. The NSCR provides neutron activation analysis, radioisotope production, and neutron radiography for the TEES/TAMUS and other research and educational facilities. Utilization of the NSCR over the past 20 years has been high. Through 2013, the NSCR has accumulated approximately 4,146 megawatt-days (MWD) of operation. Since the conversion to LEU fuel in 2006, the NSCR had accumulated approximately 583 MWD of operation through 2013. The licensee indicated that the expectations for the upcoming license renewal period were to maintain or improve the present utilization rate (Ref. 1).

This review considered TAMU/TEES annual reports from 2005 through 2013, and NRC IRs from 2005 through 2014. The annual report summaries did not indicate any significant degradation of fuel element integrity, control rod operability issues, or radiological exposure concerns. The fuel temperature and scram circuits required for operation are calibrated routinely. The NRC staff review of the TAMU/TEES NSC IRs identified one Severity Level (SL) IV violation for a failure to comply with the requirements of TS 6.1.3.a.2, to maintain adequate reactor operator staffing in the control room during reactor operation, issued in the Notice of Violation in IR 50-128/2014-201, dated October 16, 2014 (ADAMS Accession No. ML14276A218). The SL IV violations are the least severe of the cited violations. They are considered more regulatory or safety significant than minor violations, but resulted in no or relatively inappreciable potential regulatory or security consequences. Corrective action to preclude recurrence is required to resolve a SL IV non-compliance, which the licensee provided in its response to the Notice of Violation, by letter dated November 12, 2014 (ADAMS Accession No. ML14323A095). The corrective actions were reviewed by the NRC staff and determined to be acceptable.

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 a facility operating license for a research or test reactor, that the applicant shall have entered into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive wastes and spent nuclear fuel. In a letter dated May 3, 1983, R.L. Morgan of DOE informed H. Denton of the NRC that DOE has determined that universities and other government agencies operating non-power reactors have entered into contracts with DOE providing that DOE retains title to the fuel and is obligated to take the spent fuel and high-level waste for storage or reprocessing. An e-mail sent from Kenny Osborne of DOE to Duane Hardesty (NRC) (Ref. 25) reconfirms this contractual obligation with respect to the fuel at the NSCR (DOE Contract No. 78199), valid from August 1, 2008 to December 31, 2017. Additionally, DOE renews these contracts prior to their expiration to ensure that the contracts remain valid. By entering into such an agreement with DOE, the licensee has satisfied the requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility History and Modifications

Facility Operating License No. R-83 was issued on December 7, 1961, authorizing the operation of the NSCR. The NSCR achieved initial criticality on December 18, 1961, as a 100-kWt teaching and research reactor with curved plate fuel elements typical of that used in the Materials Test Reactor. In 1965, the licensee modified the reactor pool by installing a stainless steel liner to eliminate problems of pool water leakage. The licensee also installed a cooling system to allow steady-state operation at power levels up to 1.0 MWt. In 1968, the reactor core was converted to use standard TRIGA fuel elements, and on July 31, 1968, an amended facility operating license allowed the licensee to operate the NSCR at a maximum steady-state power level of 1,000 kWt and to pulse up to a $3.00 reactivity insertion.

NSCR Facility Operating License Amendment No. 4, issued June 26, 1973, authorized the TEES/TAMUS to operate the NSCR with cores consisting of standard TRIGA fuel, FLIP fuel, or mixed TRIGA/FLIP fuel element cores. The facility operating license with these cores permitted operation of the NSCR at a maximum steady-state power level of 1.0 MWt with maximum pulse reactivity insertions of $2.00. A configuration with partial loading of the core with 35-element FLIP fuel (98-element total) achieved criticality in July 1973. With the conversion of the reactor to LEU fuel in 2006, FLIP fuel is no longer in use at NSCR.

The NSCR Facility Operating License Amendment No. 5, issued on July 2, 1975, authorized the increase of the NSCR maximum permissible pulse reactivity insertion from $2.00 to $2.70. On September 27, 1976, during a loading operation, four fuel elements failed to pass through a gauge test for transverse bend. Extensive analyses determined that the damage was caused by steady-state hydrogen migration followed by rapid hydrogen pressurization during reactor pulses. NSCR pulsing operations ceased pending a fuel analysis, which was completed in 1981 (Ref. 26 and Ref. 27). Pulsing operations resumed following the issuance of NSCR Facility Operating License Amendment No. 9, dated March 30, 1983, which renewed the NSCR facility operating license for an additional 20 years. The license renewal revised the NSCR pulse size limit from $2.70 to that reactivity addition that resulted in a peak reactor fuel temperature that would not exceed 830 degrees C (1,526 degrees F).

In a letter dated December 29, 2005, the TEES/TAMUS submitted its proposal to convert the NSCR from HEU to LEU fuel, requesting approval of the fuel conversion and changes in the TSs (Ref. 2). The NRC issued the Order to convert on September 1, 2006 (Ref. 7). The reactor was declared LEU fuel steady-state operational in November 2006 and pulse operational in December 2006.

During the LEU core startup testing, the instrumented fuel element (IFE) indicated higher than expected fuel temperature, but still within operational safety limits. An analysis determined that an anomaly with the fuel-cladding gap closure had caused the increased fuel temperature. The licensee replaced the IFE and has operated the NSCR to date with the installed TRIGA LEU 30/20 fuel without incident.

The NRC staff has reviewed and approved the modifications associated with the NSCR conversion from the use of HEU to LEU fuel in September 2006 (Ref. 7), which mainly involved analyses of the reactor core and various supporting systems. The results of the conversion from HEU to LEU fuel review are reproduced in this SER as bases for acceptability, where applicable.

During this LRA review, most modifications to the NSCR involved technological upgrades to instrumentation and minor changes to the existing design that either enhanced its capability or improved reactor operations. All of these modifications were subject to evaluation under 10 CFR 50.59, "Changes, Tests, and Experiments," to ensure there was no impact on the safety of the NSCR. Furthermore, the NRC staff reviewed the licensee's annual operating reports from 2005 to 2013, and NRC IRs from 2005 to 2014 that documented these changes. The results of these reviews indicated that the changes were performed, as required, in accordance with the requirements of 10 CFR 50.59. The NRC staff concludes that all changes appear to be reasonable and the licensing actions taken over the years seem appropriate. Furthermore, the licensee did not request any substantive facility changes as part of this LRA request.

1.9 Financial Considerations

1.9.1 Financial Ability to Operate the Facility

The regulations in 10 CFR 50.33(f) state the following:

> Except for an electric utility applicant for a license to operate a utilization facility of the type described in § 50.21(b) or § 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 in this chapter, the activities for which the permit or license is sought.

The TEES/TAMUS does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Further, under 10 CFR 50.33(f)(2), "[A]pplicants 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."

The NRC staff has determined that the TEES/TAMUS must meet the financial qualification requirements under 10 CFR 50.33(f) and is subject to a full financial qualification review. The TEES/TAMUS must demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover the estimated operating costs for the period of the renewed facility operating license. Therefore, the TEES/TAMUS 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 these costs.

In a supplement to the application dated February 3, 2014 (Ref. 49), the licensee submitted its projected operating costs for the NSC for each of the fiscal years (FYs) 2015 through 2019. The projected operating costs for the NSC are estimated to range from $1,350k in FY 2015 to $1,415k in FY 2019. The licensee stated that state funding, additional sources of income, such as performing research work for other facilities, and grants will provide funds to cover these operating costs. The NRC staff reviewed the licensee's estimated operating costs and projected sources of funds to cover those costs and finds them to be reasonable.

The NRC staff finds that the licensee has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operation costs for the period of the renewed operating license. Accordingly, the NRC staff finds that the TEES/TAMUS has met the financial qualification requirements under 10 CFR 50.33(f) and is financially qualified to engage in the proposed activities of the NSCR.

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 production or 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(d)(1), 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).

In a supplement to the LRA dated February 3, 2014 (Ref. 49), the TEES/TAMUS provided a decommissioning cost estimate of $10,688,000 in 2014 dollars. The cost estimate summarized costs by labor, radioactive wastes disposal, energy, and a 25-percent contingency factor. According to the licensee, the decommissioning cost estimate was based on information in NUREG-1307, Revision 13, "Report on Waste Burial Charges," issued November 2008; NUREG/CR-1756, "Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors," March 1982; and the Bureau of Labor Statistics Inflation Calculator for 2014. According to the TEES/TAMUS, the same decommissioning cost estimate approach will be taken for future decommissioning cost estimations, which will use regional data, when available, and calculate burial costs from the most recent Atlantic Company value for pressurized-water reactor direct disposal. The NRC staff reviewed the information submitted by the licensee in the application, as supplemented, concerning decommissioning of the reactor facility and the cost estimates provided. The NRC staff concludes that the decommissioning approach and cost estimates submitted by the licensee are reasonable.

The TEES/TAMUS has elected to use a statement of intent (SOI) to provide financial assurance, as allowed by 10 CFR 50.75(e)(1)(iv), for a Federal, State, or local government licensee. The SOI must contain or reference a cost estimate for decommissioning and indicate that funds for decommissioning will be obtained when necessary.

The licensee provided an SOI, dated February 3, 2014 (Ref. 49), stating that the signator "…intends to request that funds be made available when necessary […and] seek to obtain these funds as sufficiently in advance of decommissioning as possible to prevent delay on required activities." The decommissioning cost is estimated at $10.7 million for the DECON option.

To support the SOI and the TEES/TAMUS's qualifications to use an SOI, the licensee also stated in its letter dated February 3, 2014 (Ref. 49), that the TEES/TAMUS is a State of Texas government organization and included documentation to corroborate this statement. The submittal also provided information supporting the TEES/TAMUS's representation that the decommissioning funding obligations of the TEES/TAMUS were backed by the full faith and credit of the State of Texas, and provided documentation verifying that Dr. Dimitris Lagoudas, Deputy Director at the TEES/TAMUS, the signator of the SOI, was authorized to execute contracts on behalf of the TEES/TAMUS.

The NRC staff reviewed the information provided by the licensee on decommissioning funding assurance as described above and finds that the TEES/TAMUS is a State of Texas government licensee, and, in accordance with the requirements of 10 CFR 50.75(e)(1)(iv), the SOI is acceptable; the decommissioning cost estimate as well as the costs for the DECON option are reasonable; and, the TEES/TAMUS's means of adjusting the cost estimate and associated funding level periodically over the life of the facility are reasonable. The NRC staff finds that any adjustment of the decommissioning cost estimate must incorporate, among other things, changes in costs due to the availability of disposal facilities and that the TEES/TAMUS has an obligation under 10 CFR 50.9, "Completeness and accuracy of information," to update any changes in the projected cost, including changes in costs resulting from increased disposal options.

1.9.3 Foreign Ownership, Control, or Domination

Section 104d of the AEA of 1954, as amended, 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 in 10 CFR 50.38, "Ineligibility of certain applicants," contain language to implement this prohibition. According to the licensee, the TEES/TAMUS is a State of Texas government licensee and is not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The NRC staff does not know or have reason to believe otherwise.

1.9.4 Nuclear Indemnity

The NRC staff notes that the TEES/TAMUS currently has an indemnity agreement with the Commission, which does not have a termination date. Therefore, the TEES/TAMUS will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.71, "Scope," the TEES/TAMUS, as a nonprofit educational institution, is not required to provide nuclear liability insurance. The Commission will indemnify the TEES/TAMUS for any claims arising out of a nuclear incident under the Price-Anderson Act, Section 170 of the AEA, and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95, "Appendix E—Form of indemnity agreement with nonprofit educational institutions," up to $500 million. Also, the TEES/TAMUS is not required to purchase property insurance under 10 CFR 50.54(w).

1.9.5 Financial Consideration Conclusions

The NRC staff reviewed the financial status of the TEES/TAMUS and concludes that there is reasonable assurance that the necessary funds will be available to support the continued safe operation of the NSC and, when necessary, to shut down the facility and carry out decommissioning activities. In addition, the NRC staff concludes that there are no problematic foreign ownership or control issues or insurance issues that would preclude the issuance of a renewed license.

1.10 Facility Operating License Possession Limits and License Changes

The renewal of the facility operating license for the TEES/TAMUS NSC authorizes the receipt, possession, and use of special nuclear and byproduct materials. SNM consists of such material as the uranium-235 in the reactor fuel, SNM in fission chambers, fission plates, foils, solutions and SNM produced by operation of the reactor. Byproduct material consists of such material as activation products produced by operation of the reactor in the fuel, experiments, and reactor structure and the antimony-beryllium and americium-beryllium neutron startup sources. The restricted area is defined in TS 5.1.5 and SAR Section 2.2.1.2 (Ref. 9), and all activities performed within this area fall under the jurisdiction of the reactor license. The TEES/TAMUS NSC laboratories are referenced in the SAR Section 1.3, Section 1.8, and Section 2.2.1.2 (Ref. 9) and are located within the restricted area under the jurisdiction of the reactor license. The NRC inspection program has shown that the licensee has procedures and equipment to safely handle licensed material within the restricted area.

The NRC staff reviewed the current TEES/TAMUS facility operating license and noted that some license conditions (LCs) appeared no longer necessary and/or were out of date. The NRC staff confirmed this information during a telephone conference conducted with the TEES/TAMUS NSC staff on January 10, 2013. A summary description of the TEES/TAMUS facility operating licenses LCs that were revised follows:

1) LC II.B.(2), authorizing the possession of Fuel Lifetime Improvement Program-type TRIGA fuel is removed as FLIP fuel is no longer used at NSCR.

2) LC II.B.(6), which requires that TEES/TAMUS maintain all byproduct, source and SNM within the TEES/TAMUS NSC site boundary until transfer to an appropriate materials license for shipment, was associated with Facility Operating License Amendment No. 13, issued October 26, 1993, which involved the transfer of licensed material accountability from the State of Texas material license to the NSCR Facility Operating License, R-83. This LC is no longer needed as this material has either been removed from site or is possessed under the revised LCs of the renewed license.

3) LC II.B.(7), which allowed the possession and storage of byproduct material for decay, which was previously under the State of Texas material license. This license condition applied to byproduct material at the facility at the time of the issuance of Facility Operating License Amendment No. 14, issued July 15, 1997. This LC is no longer needed since this material has been removed and no new material was allowed at the facility.

4) LC II.B.(3) was revised to match the current RTR licensing practice for byproduct material.

5) LC II.C, the reference to “Section 35.92 of 10 CFR Part 35,” was removed as it was not necessary since the “Decay-in-Storage” provision is provided in Section 20.2001(a)(2) of 10 CFR Part 20.

By letters dated July 22, 2009 (Ref. 44), and May 4, 2010 (Ref. 45), the licensee requested an increase in the possession limit for Facility Operating License R-83, LC II.B.(4), from 20 grams of each of the following isotopes: U-233, U-235, plutonium (Pu)-233, Pu-236, Pu-238, Pu-239, Pu 240, and Pu-241, to 40 grams each. The NRC staff noted that the current LC allowed a total of 160 grams of SNM (20 grams for each isotope). The licensee’s request would result in an increase in the potential total amount of SNM to 320 grams. By letter dated January 31, 2013 (Ref. 42), as discussed in a telephone conference call with the NRC staff, the licensee clarified that it needed a total SNM possession limit of 40 grams. The licensee states that a total SNM possession of limit of 40 grams would be sufficient to perform its research needs. Based on the information described above, the NRC staff concludes that a total SNM possession limit of 40 grams is acceptable.

As is current practice, the NRC staff added LCs to prevent the separation of SNM and to clarify the byproduct material possession requirements to prevent the separation of byproduct material except for byproduct material produced in experiments. The NRC staff reformatted the LCs to make them easier to read and understand. Based on its review as discussed above and the acceptable results of the NRC inspection program, the NRC staff concludes that the licensee has procedures and equipment in place to safely receive, possess, and use the materials authorized by the facility operating license.

1.11 Elimination of Iodine Production Technical Specifications

In the licensee’s response to an RAI (Ref. 59), the licensee requested that TS 3.5.3 and TS 3.6.4 be deleted from the current TSs. TS 3.5.3 and TS 3.6.4 were incorporated into the licensee’s TSs with the issuance of License Amendment No. 15, dated November 1, 1999, in order to allow the licensee to produce Iodine-125, from the irradiation of Xenon-124 gas, for application as a diagnostic medical isotope. The results of the licensee’s review of the dose consequences from the release of the amount of Xenon-125 allowed by TS 3.6.4, Specification (a), produced radiation doses that could result in potentially unsafe exposures.

The NRC staff reviewed the results of the licensee’s response to the RAI and deleted TS 3.5.3 and TS 3.6.4, which are provided for reference below:

1.11.1 (Deleted) TS 3.5.3 Xenon and Iodine Monitoring

TS 3.5.3 states the following:

Specification

No experiment that involves the handing of ^{125}Xe and ^{125}I may be performed unless the following radiation monitoring systems are operable. No experiment may be performed, except decay of ^{125}Xe, unless the ^{125}Xe effluent monitoring channel is operable.

Radiation Monitoring Systems*	Number
^{125}Xe Effluent Monitoring Channel	1
^{125}I Air Monitor	1

* For periods of maintenance to the ^{125}Xe effluent monitoring channel, the intent of this specification will be satisfied if it is replaced by building gas samples.

1.11.2 (Deleted) TS 3.6.4 Xenon Irradiation for Iodine Production

TS 3.6.4 states the following:

Specifications

(a) ^{124}Xe activation experiments shall be controlled such that the total single experiment activity produced is limited to no more than 2000 Ci of ^{125}Xe.

(b) The total facility ^{125}Xe inventory of all experiments shall not exceed 3500 Ci.

2 REACTOR DESCRIPTION

2.1 Summary Description

2.1.1 Introduction

The NSCR is a natural convection, water-cooled, moderated, and shielded reactor that was converted to the use of TRIGA fuel. Aspects of its design are similar to those of many other research reactors operating in the U.S. and abroad. The reactor was originally designed for plate-type Materials Test Reactor fuel assemblies but was converted to the use of TRIGA fuel elements contained in bundles that can hold up to four fuel elements each and are designed to fit on a Materials Test Reactor-type grid plate. The reactor operated initially with a combination of HEU and LEU fuel and was converted to the use of LEU fuel and returned to operation using all LEU fuel in December 2006. The LEU TRIGA fuel is in the form of a $U\text{-}ZrH_x$. The “x” represents the hydrogen to zirconium stoichiometry ratio in the $U\text{-}ZrH_x$ nomenclature. The reactor is presently fueled with TRIGA LEU 30/20 $U\text{-}ZrH_{1.6}$ fuel elements where 30 indicates 30 weight percent uranium, 20 indicates up to 20 percent isotopic enrichment of uranium-235, and 1.6 indicated that the hydrogen to zirconium ratio in the zirconium matrix is approximately 1.6. For more information on the fuel characteristics, see NUREG-1282, “Safety Evaluation Report on High-Uranium Content, Low Enriched Uranium Zirconium Hydride Fuels for TRIGA Reactors,” issued August 1987 (Ref. 29).

The NSCR uses graphite moderators/reflectors on two sides of the core and uses the other two sides primarily for sample irradiation. The reactor core is supported on a nine by six grid, with each grid location supporting a fuel bundle consisting of four fuel positions. These positions can be filled with fuel elements, control rods, or other nonfuel elements. The grid is supported by a frame, which is attached to the reactor bridge, which can support the reactor core, which is mounted on rails along the top of the pool. While the reactor is operating, the frame rests on the floor of the pool, which provides support for the reactor core. However, when the reactor is shut down, it is possible to lift the reactor core from the floor and move the reactor core to other locations within the pool by moving the bridge.

The reactor core is located in one section of a water-filled, two-section concrete pool, which has a stainless steel liner. The radial shielding for the reactor consists of water and ordinary concrete. About 20 ft (6.1 m) of water serves as shielding above the core. The NSCR core uses four motor-driven shim-safety rods, a motor-driven regulating rod, and a motor/pneumatic-driven transient rod. The shim-safety rods provide scram capability, the regulating rod is used to maintain reactor power during normal operations, and the transient rod is used in both normal and pulse mode operation. The four shim-safety control rods are fuel follower rods.

The NSCR normally operates at a maximum thermal power level of 1.0 MWt. The reactor can also be pulsed up to a peak power that corresponds to a peak fuel temperature of 830 degrees C (1,526 degrees F). The reactor core is cooled through natural convection of the pool water.

Extensive experience gained from similar designs used throughout the world has demonstrated the safety of TRIGA reactors. The TRIGA fuel is characterized by high fission product retention and the ability to withstand water quenching at temperatures as high as 1,150 degrees C (2,100 degrees F). The safety of the fuel arises from the strongly negative prompt temperature coefficient characteristic of U-ZrH_x fuel-moderator elements. As the fuel temperature rises, this coefficient immediately compensates for reactivity insertions. The NSCR fuel temperature safety limit (SL), in TS 2.1, is specified not to exceed 1,150 degrees C (2,100 degrees F) for 30/20 U-ZrH_x fuel under any conditions of operation. To ensure that this SL is not exceeded, the fuel temperature limiting safety system setting (LSSS) in TS 2.2 is established for steady-state or pulse operation to be less than or equal to 525 degrees C (975 degrees F) as measured in the IFE located in specific locations of the core (see Section 2.6.3 of this SER for a discussion of the SL and LSSS).

On December 29, 2005, TEES/TAMUS submitted an application requesting an amendment to its facility operating license that would allow conversion to the exclusive use of LEU fuel. The NSCR was the first reactor to convert to the use of TRIGA LEU 30/20 fuel. However, the TRIGA LEU 30/20 fuel had been used and extensively tested in the GA Mark F reactor (Ref. 29). GA provided safety and accident analysis using computational methods described in Section 2.6 of this SER. Tests performed using the Puerto Rico Nuclear Center (PRNC) TRIGA HEU FLIP core provided the data to benchmark the computational technique used to evaluate the TRIGA LEU 30/20 fuel in the NSCR TRIGA core. The HEU FLIP fuel used at the NSCR came from the PRNC facility after closing (Ref. 7). The data in this validation effort included both neutronic and thermal-hydraulic measurements obtained under steady-state conditions. In addition, data obtained from pulse operation was used to validate the kinetic analysis methods. The evaluation was performed consistent with the guidance in NUREG-1537 (Ref. 19).

On the basis of its review, the NRC staff issued an Order on September 1, 2006, for the TEES/TAMUS to convert the NSCR to use LEU fuel. The Order included the conversion SER, which provided the results of the NRC staff's evaluation of the licensee's conversion request. The Order also included changes in the TSs that would be required for operation of the facility with LEU fuel (Ref. 7). The TEES/TAMUS NSCR was declared LEU fuel steady-state operational in November 2006 and pulse operational in December 2006.

2.2 <u>Reactor Core</u>

The NSCR uses solid fuel elements in which the zirconium hydride moderator is homogeneously combined with LEU fuel. Up to four fuel elements may be contained within an assembly that allows the fuel elements to be secured into the reactor core grid plate. This fuel assembly, called a fuel bundle, is supported in a grid box consisting of an aluminum grid plate. The operational NSCR core contains fuel element bundles, four fuel-followed control rods, a water-followed regulating rod, an air-followed transient rod, and graphite blocks used for reflection/moderation.

The core is supported from the top of the pool wall using a movable bridge structure that allows the core to be located along the central axis of the pool. The bridge that spans the reactor pool supports the reactor core, the control rod drives, the nuclear instrumentation detectors, and the nitrogen-16 (N-16) diffuser system. The core is generally located against a graphite coupler box, which acts as a reflector and couples the core to the graphite thermal column. The grid plate also accepts reflector elements, experimental devices, a neutron source holder, and other elements.

The reactor normally operates at a maximum thermal power level of 1.0 MWt. The reactor can also be pulsed up to a peak power that corresponds to a peak fuel temperature of 830 degrees C (1,526 degrees F). The reactor core is cooled through natural convection of pool water.

The NSCR core uses four motor-driven shim-safety rods, a motor-driven regulating rod, and a motor/pneumatic-driven transient rod, to provide control and shutdown capability for the reactor. All six control rods are supported by the bridge structure. Section 2.3.4 of this SER discusses the control rods.

The primary reactor coolant water (pool water) is deionized and routinely monitored for quality and to identify any significant change in radioactivity. The reactor core is cooled by the natural convection of the reactor coolant water, which also serves as reflector and moderator and provides radiation shielding for the NSC staff.

A diffuser system is used to reduce the radiation exposure level on the bridge from N-16. The system pulls water from the main area of the pool and discharges it through a diffuser above the reactor core. This helps to disperse the plume of hot water exiting the core. The net effect is an increase in the time that it takes the N-16 to reach the pool surface, allowing decay time for the 7.13-second half-life of N-16.

2.2.1 TS 5.3 Reactor Core

TS 5.3 states the following:

<u>Specifications</u>

1. The core shall be an arrangement of TRIGA LEU uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

2. The reflector, excluding experiments and experimental facilities, shall be any combination of graphite, water, and heavy water.

3. Fuel shall not be inserted or removed from the core unless the reactor is subcritical by at least 50 cents more than the calculated worth of the most reactive fuel assembly.

TS 5.3, Specification 1, helps ensure that only TRIGA fuel elements are authorized to be used in the NSCR. This design feature information is important to help ensure that the NSCR core consists of TRIGA fuel elements that have been evaluated by the SAR, and approved for use.

TS 5.3, Specification 2, helps ensure that the reflector elements used in the NSCR core operating configuration use a combination of water, heavy water, and graphite as evaluated in the SAR.

TS 5.3, Specification 3, helps ensure that the manipulation or relocation of any fuel is not performed unless the proper reactivity controls have been established to prevent an inadvertent criticality excursion event. This specification requires that the reactor core shall be subcritical by the calculated reactivity worth of the most reactive fuel assembly plus an additional $0.50 of reactivity for margin.

The NRC staff reviewed the SAR, the analysis in the Conversion SER (Ref. 7), and TS 5.3. The NRC staff finds that TS 5.3, Specifications 1, 2 and 3, characterizes the NSCR design features for the reactor core and helps ensure that the reactor core configuration conforms to the analysis provided in the SAR. The NRC staff finds that TS 5.3 is consistent with the guidance in NUREG-1537, and helps ensure the NSCR will be operated consistent with the design criteria for the reactor core. Based on the information provided above, the NRC staff concludes that TS 5.3, Specifications 1, 2 and 3, are acceptable.

2.2.2 TS 3.1.4 Reactor Core Configuration

TS 3.1.4 states the following:

Specifications

1. Control rods shall not be manually removed from the core unless the core has been shown to be subcritical, and shutdown margin requirements met, with those control rods removed.

2. Core lattice positions shall not be vacant except for positions on the periphery of the core assembly while the reactor is operating. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions not on the periphery shall contain experiments or an experimental facility to prevent accidental fuel additions to the core.

3. The instrumented fuel element, if serving as the Limiting Safety System sensor, shall be located adjacent to the central bundle with the exception of the corner positions.

TS 3.1.4, Specification 1, helps ensure that the manipulation or relocation of control rods is not performed unless the proper reactivity controls have been established to prevent an inadvertent criticality excursion event. The specification that the shutdown margin (SDM) reactivity requirements are met help ensure sufficient negative reactivity for reactivity control with the control rods removed.

TS 3.1.4, Specification 2, helps ensure that internal core lattice positions are occupied with fuel elements, with water holes limited to one single vacant hole location. As provided in the SAR, this configuration helps ensure that power peaking is properly controlled and excessive power densities do not exist which could result in fuel cladding failure. Vacant lattice positions are filled with mechanical devices that reduce the probability of an accidental reactivity insertion in that location.

TS 3.1.4, Specification 3, helps ensure that the IFE is located in an area of the NSCR core with the highest power densities which are the fuel elements with the highest fuel temperatures. This specification is consistent with the analysis provided in the SAR. The IFE location is chosen to ensure that the hottest location is protected against exceeding the fuel temperature SL. The licensee has shown that an IFE located in the specified core positions with a scram setting of 525 degrees C (977 degrees F) would limit the maximum steady-state temperature in the hottest fuel element to less than 600 degrees C (1,112 degrees F) (Ref. 10) and would protect it from reaching the 30/20 LEU fuel temperature SL of 1,150 degrees C (2,100 degrees F), providing a 47-percent safety margin. The analysis also indicated that the maximum fuel temperature in the hot rod at 1.3 MWt steady-state operations (a power level above the high power scram setpoint of 1.25 MWt) is 440 degrees C (824 degrees F), while the fuel temperature is 384 degrees C (723 degrees F) at the IFE temperature sensing location. At 1 MWt, the licensee indicated that the corresponding temperatures are 373 degrees C (703 degrees F) in the hottest rod and 329 degrees C (624 degrees F) in the IFE at the thermocouple location, which is 0.3 inches (in.) (0.76 centimeter (cm)) away from the fuel axial centerline. (See Section 2.6.3 of this SER for further discussion).

The NRC staff finds that TS 3.1.4, Specifications 1, 2 and 3, establish core configuration limitations which are based on the SAR. The NRC staff also finds that TS 3.1.4 helps to protect the fuel safety limit and is consistent with the guidance in NUREG-1537 (Ref. 19) and ANSI/ANS-15.1-2007 (Ref. 28). As such, the NRC staff concludes that TS 3.1.4 is acceptable.

2.2.3 TS 3.1.3 Shutdown Margin

TS 3.1.3 states the following:

> Specifications
>
> The reactor shall not be operated unless the shutdown margin provided by control rods is greater than $0.50 with:
>
> 1. Irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state,
>
> 2. The highest worth control rod and the regulating rod fully withdrawn, and
>
> 3. The reactor in reference core condition.

The minimum SDM of greater than $0.50 is a value that can be easily determined by the licensee and is a sufficient amount of shutdown reactivity to give reasonable assurance that when considered with TS 3.1.3, Specification 1, 2, and 3, the licensee will be able to shut down the reactor under anticipated conditions and the reactor will remain shut down.

TS 3.1.3, Specification 1, helps ensure constraints on the core condition by considering non-secured experiments to be in their most reactive state to ensure that the reactor remains subcritical, should a non-secured experiment be moved to its most reactive position.

TS 3.1.3, Specification 2, helps ensure that the reactor can be shut down even if the highest worth control rod became stuck out of the reactor core. The regulating rod is also considered fully withdrawn for the determination of SDM because the TSs do not require this rod to have scram capability.

TS 3.1.3, Specification 3, helps ensure proper and consistent core reference conditions for deriving the SDM. The reactivity state of the reactor can be affected by the SDM. The fission product xenon, which is a neutron poison, and the temperature of the reactor can affect the reactivity state of the reactor. The purpose of defining a reference core condition is so that reactivity measurements can be adjusted to a fixed baseline. The reference core condition is the most limiting for satisfying the SDM requirement.

The NRC reviewed the SDM established in TS 3.1.3, Specifications 1, 2 and 3, and determined that the minimum SDM is sufficient and proper controls are established to ensure that the SDM requirements are satisfied and considerations are provided for the reactivity of experiments when the highest worth control rod and regulating rod are withdrawn and the NSCR is in the reference core condition. These requirements provide sufficient negative reactivity to ensure that the NSCR can be shut down and remain in a subcritical condition from any operating condition described and analyzed in the SAR. The NRC staff finds that TS 3.1.3, Specifications 1, 2, and 3 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.1.3, Specifications 1, 2, and 3, are acceptable.

2.2.4 TS 3.1.6 Maximum Excess Reactivity

TS 3.1.6 states the following:

> Specifications
>
> The maximum reactivity in excess of reference core condition shall not exceed 5.5% $\Delta k/k$ ($7.85).

TS 3.1.6 establishes a limit on excess reactivity allowing operational flexibility while limiting the reactivity available for reactivity addition accidents. The maximum excess reactivity helps establish a basis for ensuring that an adequate SDM is available by control rod insertion. With the core assembly in its most reactive position, adjacent to the thermal column, the licensee states in the SAR, a calculated excess reactivity of $6.50. With the core assembly away from the thermal column, the excess reactivity calculated by the licensee is $5.55. The resulting SDM for these two core configurations provided by the licensee were $2.20 and $3.15, respectively (Ref. 17).

The NRC staff reviewed the maximum excess reactivity established in TS 3.1.6, and finds that the calculated excess reactivity and the SDM values are consistent with the measurements made and reported by the licensee after the initial startup with the LEU conversion core, considering the fuel burnup and samarium (Sm)-149 buildup in the core since the initial critical on the LEU core, as provided in the Startup Report (Ref. 8). The NRC staff also finds that the SDM and excess reactivity values specified in TS 3.1.3 and TS 3.1.6, respectively, are supported by analyses in the SAR. Furthermore, the NRC staff finds that the use of the reference core conditions provided in TS 3.1.6 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 3.1.6 is acceptable.

2.2.5 TS 4.1.3 Shutdown Margin

TS 4.1.3 states the following:

> Specification
>
> The reactivity worth of each control rod and the shutdown margin shall be determined annually and following changes in the core, in-core experiments, or control rods.

TS 4.1.3 helps ensure that the SDM and the reactivity of each control rod are determined annually and after significant core configuration changes or control rod changes. The NRC staff finds that TS 4.1.3 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.1.3 is acceptable.

2.2.6 TS 4.1.6 Maximum Excess Reactivity

TS 4.1.6 states the following:

> Specification
>
> The excess reactivity shall be determined annually and following changes in the core, in-core experiments, or control rods for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

TS 4.1.6 helps ensure that the maximum reactivity is determined by requiring that the excess reactivity be determined annually and/or following core configuration, in-core experiments, or control rod changes. The NRC staff finds that the surveillance specified in TS 4.1.6 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.1.6 is acceptable.

The TSs described above (TS 5.3, TS 3.1.3, TS 3.1.4, TS 3.1.6, TS 4.1.3, and TS 4.1.6) help to ensure that normal operating conditions of the reactor core include limits on allowable core configurations, SDM, and excess reactivity and provide corresponding surveillance requirements for SDM and maximum excess reactivity.

The NRC staff reviewed the analyses presented in the SAR, the Conversion SER (Ref. 7), and the Startup Report (Ref. 8), and finds that the results demonstrate that these TS values provide sufficient safety margin to the NSCR operational core to ensure the integrity of the fuel element cladding and fission product barrier. The NRC staff also finds that these TSs are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that these TSs as described are acceptable.

2.3 Reactor Fuel and Core Components

The NSCR fuel components are defined in TS 1.0 and described in Section 2.2 of this SER. The development and use of TRIGA fuel began in 1957, and over 6,000 fuel elements have been fabricated for use in over 60 research reactors both domestically and abroad. TRIGA fuels have safety features that include a large prompt negative temperature coefficient of reactivity, high fission product retention, chemical stability when quenched from high temperatures in water, and dimensional stability over a wide range of temperatures. Over 25,000 pulses have been successfully performed domestically and abroad with TRIGA fuel elements, with temperatures reaching peaks of about 1,150 degrees C (2,100 degrees F).

The fuel bundles are arranged in three- or four-element configurations, depending on whether a control rod is included in the bundle. Coolant water flows by natural convection through the 2 in. (5 cm) diameter holes in the grid plate adapter, passing through the opening and then through the bundle until it leaves the fuel bundle through the openings in the aluminum handle at the top of the bundle. Three-element bundles are similar to the four-element bundle, with the exception that a space is available for the inclusion of a control rod. In addition, the grid and support structure is designed so that a fuel follower attached to the end of a control rod can pass through the grid and extend into the space below the core. A guide tube mounted in the bundle is provided for the regulating rod and transient rod. The guide tube has holes to allow coolant to flow over the rod. The guide tube provides stability to these rods. Each fuel element contains fuel slugs, with a graphite reflector at each end. A zirconium rod is located in the center of the fuel element. Instrumented elements have three thermocouples embedded in the fuel located symmetrically at, above, and below the middle of the fuel.

Table 2-1 shows the physical characteristics of the NSCR LEU 30/20 fuel elements along with a comparison to the PRNC HEU fuel. This comparison is further explained in Section 2.6 of this SER.

Table 2-1 Descriptions of LEU and HEU Fuel Elements

DESIGN DATA	NSCR (LEU)	PRNC (HEU)
Fuel Type	U-ZrH 30/20	U-ZrH FLIP
Enrichment, weight percent (wt%)	19.75	70.00
Number of Fuel Elements per Bundle	4	4
Erbium, wt%	0.90	1.48
Zirconium Rod Outer Diameter, mm	6.35	6.35
Cladding Thickness, mm	0.508	0.508
Cladding Material	304 SS	304 SS

The NSCR uses two IFEs, which are identical to the 30/20 LEU fuel element with the exception of three thermocouples embedded in the fuel. The sensing tips are located halfway between the outer radius and the vertical centerline on the fuel section and 1 in. (2.54 cm) above and below the horizontal center. The IFE allows the licensee to directly measure the temperature of the fuel. Although the IFEs have been located in position 5E4 and 6D4 in the core, they may be placed in other locations as specified in TS 2.2 (See Section 2.6.3.2 of this SER for further discussion).

The NRC has approved the behavior of LEU TRIGA fuels with the above uranium content generically in NUREG-1282 (Ref. 29), and specifically for the NSCR in the NRC's

September 1, 2006, Order to convert from HEU to LEU fuel, which included the Conversion SER and the TS (Ref. 7).

TS 5.2, TS 3.1.5, and TS 4.1.5 define the requirements of the NSCR fuel, as described in the following sections.

2.3.1 TS 5.2 Reactor Fuel

TS 5.2 states the following:

> <u>Specifications</u>
>
> TRIGA LEU 30/20 Fuel: The individual unirradiated LEU fuel elements shall have the following characteristics:
>
> 1. Uranium content: maximum of 30 Wt% enriched to maximum 19.95% Uranium-235 with nominal enrichment of 19.75% Uranium-235,
>
> 2. Hydrogen-to-zirconium atom ratio (in the ZrHx): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65,
>
> 3. Natural erbium content (homogeneously distributed): nominal 0.90 Wt%, and
>
> 4. Cladding: 304 stainless steel.

TS 5.2, Specification 1, provides the maximum weight percent (wt%) and maximum and nominal enrichment of the TRIGA fuel elements and helps ensure that the fuel requirement is consistent with the SAR and the analysis provided in the Conversion SAR. The Conversion SAR indicates that the enrichment may be higher by about 1 percent than the nominal (design) value of 19.75 percent used in the Conversion SAR and that this small increase would result in a corresponding increase in the power density of about 1 percent.

TS 5.2, Specification 2, provides the nominal and maximum hydrogen to zirconium fuel stoichiometry ratios to help ensure that the fuel used in the NSCR is consistent with the analysis provided in the SAR and used to develop the basis for the SL in TS 2.1 and the LSSS in TS 2.2. The SAR indicates that the hydrogen to zirconium ratio influences the fuel element internal pressure during operation. The internal pressure of the fuel element influences the stress in the clad. As shown in the Conversion SER, at the maximum upper limit ratio of 1.65, and the conservative SL of 1,150 degrees C (2,100 degrees F), the pressure is at least a factor of 5 lower than would be necessary for clad failure. While individual fuel slugs may have a hydrogen to zirconium ratio of up to 1.65, the fuel slugs are chosen for each fuel element during manufacture such that the average ratio in each fuel element is about 1.60.

TS 5.2, Specification 3, provides the natural erbium content, a burnable poison used to control the reactivity of the fuel elements, is homogeneously distributed in the fuel element, with a nominal 0.90 wt percent, to help ensure that the fuel is consistent with the analysis provided in the SAR.

TS 5.2, Specification 4, provides the cladding material for the NSCR fuel elements, which helps to ensure that the fuel used in the NSCR is consistent with the analysis provided in the SAR.

The NRC staff reviewed the information regarding the constituents, materials, and components of the fuel elements provided in the SAR and the analysis provided in the Conversion SAR (Ref. 7). On the basis of its review, the NRC staff finds that the licensee has adequately described the fuel elements used in the NSCR, including design limits, and the technological and safety-related bases for these limits. The NRC staff concludes that compliance with these TS limits will ensure uniform characteristics and compliance with design bases and safety-related requirements.

TS 5.2, Specifications 1, 2, 3 and 4, help to ensure that the NSCR fuel uranium content, enrichment, hydrogen to zirconium ratio limits, burnable poison, and cladding material remain consistent with the analysis provided in the SAR and Conversion SAR. The NRC staff reviewed the fuel characteristics supporting the LEU conversion and found the characteristics of the 30/20 LEU fuel acceptable in the Conversion SAR. Based on the information provided above, the NRC staff concludes that TS 5.2, Specifications 1, 2, 3 and 4, are acceptable.

Fuel growth and deformation can occur during normal operations, as described in NUREG-1537; General Atomics Report E-117-833, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," issued February 1980 (Ref. 24); and "The U-ZrH_X Alloy: Its Properties and Use in TRIGA Fuel," 1981 (Ref. 30). Damage mechanisms include fission recoils and fission gases, both of which are strongly influenced by thermal gradients. Swelling of the fuel depends on the time the fuel spends over a temperature threshold of about 750 degrees C (1,382 degrees F). At 1 MWt, the current steady-state power level, the Conversion SAR reports that a calculated IFE operating temperature of about 329 degrees C (624 degrees F) corresponds to a maximum calculated fuel temperature of about 373 degrees C (703 degrees F) (Ref. 2, Table 4-11). The Conversion SAR states that, at this temperature, swelling would be minimal, if present at all. Although fuel temperatures could rise above 750 degrees C (1,382 degrees F) during pulsing, the time at temperature is short enough that pulsing should not cause fuel swelling by these mechanisms. The NRC staff reviewed the data provided by the licensee and concludes that there is reasonable assurance that fuel swelling at these temperatures, and by the mechanism described above, would be precluded.

The licensee inspects the fuel cladding to detect gross failure or visually observed deterioration. Attributes inspected include the fuel element transverse bend and length, and a visual inspection is performed for bulges and other cladding defects.

2.3.2 TS 3.1.5 Reactor Fuel Parameters

TS 3.1.5 states the following:

Specifications

1. The reactor shall not be operated knowingly with damaged fuel, except for the purposes of locating damaged fuel elements.

2. A fuel element shall be considered damaged and must be removed from the core if:

    a. In measuring the transverse bend, the bend exceeds 0.125 inch over the length of the cladding, or

    b. In measuring the elongation, its length exceeds its original length by 0.125 inch, or

    c. A clad defect exists as indicated by release of fission products, or

    d. A visual inspection reveals bulges, gross pitting or corrosion.

3. The burnup of the uranium-235 in the UZrH fuel matrix shall not exceed 50 percent of the initial concentration.

TS 3.1.5, Specifications 1, 2, and 3, helps ensure that the reactor is only operated with fuel that has an effective cladding barrier to the release of any potential fission products. TS 3.1.5, Specifications 1 and 2, help establish inspection requirements to detect gross failure or visual deterioration of the fuel. There are cases where a cladding defect exists but only is detectable when the reactor is operating. In the unlikely event that a cladding defect is detected (normally by a small increase in the pool water radioactive material concentration or airborne radioactive material concentration), the licensee would only operate the reactor in order to locate the damaged fuel element. All operation would need to meet regulatory requirements in 10 CFR Part 20 for release of radioactive material and occupational and public doses. The fuel attributes inspected include the fuel element transverse bend and length, and a visual inspection is performed for bulges or other cladding defects. Fuel that cannot be removed from or placed into the reactor core because of swelling would be considered to have an unacceptable bulge. TS 3.1.5, Specification 3, helps ensure that the fuel is not utilized past its burnup limit as recommended in the guidance in NUREG-1537.

The NRC staff reviewed the fuel limits provided in TS 3.1.5, Specifications 1, 2 and 3, and finds that the limits on transverse bend and length, which are based on values from the reactor designer (GA) are consistent with the guidance in NUREG-1537. Although the reactor may not operate with damaged fuel, there have been instances of fuel cladding defects in which fission products are detected only during reactor operation. Under these circumstances, the reactor needs to be operated to locate the fuel element with the cladding defect. The burnup limit in TS 3.5, Specification 3, is consistent with the guidance in NUREG-1537. The NRC staff finds that the licensee has used the standard definition of damaged fuel for TRIGA reactors consistent with the guidance in NUREG-1537. Based on the information provided above, the NRC staff concludes that TS 3.1.5. Specifications 1, 2 and 3, are acceptable.

2.3.3 TS 4.1.5 Reactor Fuel Elements

TS 4.1.5 states the following:

Specifications

1. The following fuel elements shall be inspected visually for damage or deterioration and measured for length and bend annually:

    a. At least four elements which occupy the highest pulse temperature positions in the core,

    b. At least one-fifth of the fuel elements used in operation of the reactor over the previous inspection year,

    c. The four elements (a) above may be included in the inspection of the fuel elements of (b) above, and

    d. Over a 5 year period every fuel element used in operation of the reactor shall be inspected.

2. If any element is found to be damaged, the entire core will be inspected.

TS 4.1.5, Specification 1, item a through item d, provide criteria for the annual fuel inspection which help ensure that the fuel continues to operate with effective barriers to prevent the inadvertent release of fission products. In the SAR, the licensee states that fuel temperature is the major contributor to fuel damage. Inspection of the four fuel elements that occupy the highest temperature positions in the core during pulsing provides surveillance for detection of the most probable fuel element damage. Inspection of one-fifth of the fuel elements used in the operation of the reactor provides surveillance of the remaining fuel elements such that, over a 5-year period, all fuel elements will have been inspected. The NRC staff finds that the surveillance frequencies are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

TS 4.1.5, Specification 2, provide a threshold to perform an inspection of all fuel elements if a single fuel element fails to pass inspection. TS 4.1.5, Specification 2, helps ensure that a high degree of confidence exists that the remaining fuel can be safely used.

The NRC staff reviewed the surveillance intervals provided in TS 4.1.5, Specifications 1 and 2, and finds that these specifications are sufficient to help ensure that fuel element integrity is maintained and any deterioration in cladding integrity will be detected. The NRC staff also finds that TS 4.1.5 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review of the information provided above, the NRC staff concludes that TS 4.1.5, Specifications 1 and 2, are acceptable.

2.3.4 Control Rods

The NSCR core uses four motor-driven shim safety rods, a motor-driven regulating rod, and a motor/pneumatic-driven transient rod, which provide control and shutdown capability for the reactor. Section 5.1 of this SER presents the definition of control rods.

2.3.4.1 TS 5.4 Control Rods

TS 5.4 states the following:

Specifications

1. The shim safety control rods shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding. These rods shall incorporate fueled followers that have the same characteristics as the fuel region in which they are used.

2. The regulating control rod may not have scram capability and shall be a stainless rod or contain borated graphite, B_4C powder or boron and its compounds in solid form as poison in aluminum or stainless steel cladding. This rod is water followed in that pool water takes the place of the rod as it is withdrawn. It has no physical follower attachment.

3. The transient control rod shall have scram capability and contain borated graphite or boron and its compounds in solid form as a poison in an aluminum or stainless steel clad. The transient rod shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod shall incorporate an air follower.

TS 5.4, Specifications 1, 2 and 3, help ensure that the design specifications and requirements for the shim safety, regulating, and transient rods are maintained as discussed in the SAR. The NRC staff reviewed TS 5.4, and finds that the material characteristics provided in TS 5.4 are consistent with those of other TRIGA reactors, will help ensure that the important aspects of the design of the control rods are maintained, and will help ensure that the control rods will perform their safety function. Based on the information provided above, the NRC staff concludes that TS 5.4, Specifications 1, 2, and 3, are acceptable.

The SAR, Section 4.2.2, provides a description of the NSCR control rods. There are four shim safety rods, one regulating rod, and one transient rod. The shim safety rods are electromagnetically coupled to the rod drives and will fall into the core whenever power is lost to the electromagnet, providing a scram. The regulating rod is used during steady-state operation to maintain the reactor power and has no scram capability. The transient rod is used during steady-state and pulse mode operation and has scram capability. The four shim safety rods are fuel followed. Each consists of a poison region and a fuel region. The poison region is borated graphite with the same cladding as the fuel elements. The fueled region is identical to a fuel element. When fully inserted, the fueled portion of the control rod extends through the grid plate, below the reactor core with the poison section in the core. The regulating rod is water followed and consists of a boron carbide poison (B_4C) section. The transient rod is air followed.

The SAR, Section 7.2.3.4, as supplemented by the licensee's RAI responses (Ref. 65), provide a description of the servo system for the regulating rod. The servo system provides an automatic power level by comparing the output signal from the wide range linear power monitor to a preset power level to adjust power, by providing a shim-in or shim-out signal to the regulating rod. The reactivity associated with the change in position of the regulating rod to the shim-in or shim-out signal helps to stabilize the reactor at the preset power level. The servo system measures the departure of indicated power on the wide range linear channel from a preset signal and provides shim-in or shim-out signal to the regulating control rod controller, which results in an adjustment to reactor power. The servo system is capable of moving the regulating rod approximately 0.03 percent of normal range of travel, which equates to a reactivity change of approximately $0.0003. The servo system operates within a $\pm$5 percent power range (e.g., if the servo was set to maintain power at 90 percent, the active range would be 86 to 94 percent power). If power changes in excess of 5 percent, the servo system enters a "fault mode" where it becomes inactive. The licensee evaluated a worst case servo failure event which would result in a continuous withdraw of the regulating rod at its normal rate of travel. The regulating rod withdraw scenario was analyzed using the RELAP code for two initial reactor core power conditions: 300 W (below the point of increasing fuel temperature) and 1 MW. The results of both cases were found to be bounded by existing safety systems and posed no risk of creating fuel damage. In the 300 W case, the negative power coefficient of reactivity prevented the reactor from reaching a power level of 1 MW, and in the 1 MW case, the high power scram ($\leq$ 1.25 MW) terminated the transient as designed.

The licensee states in the SAR, that using neutron-absorbing borated graphite, B_4C powder, or boron and its compounds will satisfy the poison requirements for the control rods. Since the regulating rod normally is a low worth rod, using solid stainless steel as cladding material is satisfactory. Some of these poison materials must be contained in a suitable clad material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the poison from the pool water environment. Control rods that are fuel followed provide additional reactivity to the core when the poison section is withdrawn and increase the worth of the control rod. The use of fuel-followers has the additional advantage of reducing flux peaking in the water-filled regions created by the withdrawal of the control rods if there were no fuel-followers. Scram capabilities are provided for rapid insertion of the control rods. The transient control rod is designed to produce a reactor pulse upon rapid withdrawal. The licensee also projected that the lifetime of the control rods will extend beyond the period of renewal and has no plans to replace or store depleted control rods.

The worth of the transient rod, regulating rod, and the four shim safety rods for this core configuration were reported by the licensee in the Startup Report (Ref. 8) as $3.46, $1.02, $3.17, $2.04, $2.90, and $4.61, respectively. The total control rod worth was measured to be $17.20 for the startup of the LEU core. The control rod worth values provided in the Startup Report reflect the post-HEU to LEU conversion testing performed in late 2006. The SAR states that the total rod worth as determined on July 9, 2011, was $14.62. The change in control rod worth reflects the effects of fuel burnup and the buildup of Sm-149 in the fuel since the initial LEU core startup. The current core excess reactivity is $6.50 with the core assembly against the thermal column (most reactive position) and $5.55 away from the thermal column. This corresponds to an SDM against the thermal column of $2.20 and away from the thermal column of $3.15 (Ref. 17).

The NRC staff reviewed the SAR description, as supplemented, and concludes that the control rod worth values provide sufficient negative reactivity to satisfy the core excess reactivity and the SDM requirements of the NSCR TSs.

2.3.4.2 TS 3.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

TS 3.2.3 states the following:

> Specification
>
> During operation, all control rods shall be operable. For scrammable control rods, the scram time measured from the instant a SCRAM signal is initiated to the instant that the slowest scrammable rod reaches its fully inserted position shall not exceed 1.2 seconds. During core manipulations, i.e. core loading and unloading, all installed control rods shall be operable.

TS 3.2.3 helps ensure that, during the normal operation of the NSCR, all control rods are operable, and the scrammable control rods can be fully inserted from the instant that a scram signal is initiated rapidly enough to prevent fuel damage. TS 3.2.3 helps ensure that the reactor will be promptly shut down when a scram signal is initiated. The 1.2-second value is provided as an analytical assumption in the reactivity insertion event described in SAR Section 13.5.3 and evaluated in Section 4.1.3 of this SER. The NRC staff finds that the 1.2-second scram time is adequate to help ensure the safety of the reactor. The NRC staff finds that the requirements of TS 3.2.3 support the basic design requirements to prevent reactor fuel damage. Based on the information provided above, the NRC staff concludes that TS 3.2.3 is acceptable.

2.3.4.3 TS 4.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

TS 4.2.3 states the following:

> Specifications
>
> 1. The control rods shall be visually inspected for deterioration biennially.
>
> 2. Operability tests of the control rod mechanism shall follow modification or repairs.
>
> 3. The Transient Rod drive cylinder and associated air supply system shall be inspected, cleaned and lubricated semiannually.
>
> 4. The scram time shall be measured annually or whenever any work is done on the control rods or the control rod drive system.

TS 4.2.3, Specification 1, helps ensure that a visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation of the reactor.

TS 4.2.3, Specification 2, helps ensure that the control rod mechanism will be tested after modifications or repairs.

TS 4.2.3, Specification 3, helps ensure the inspection and maintenance of the transient rod drive assembly to reduce the probability of failure of the system because of deterioration.

TS 4.2.3, Specification 4, specifies surveillance intervals to help ensure the operability of the control rods and requires verification that all scrammable control rods meet the scram time requirement. Annual measurement of the scram time not only checks the scram system electronics, but also indicates the capability of the control rods to perform properly.

The licensee states in the SAR that these intervals for control rod inspection, scram time determination, rod worth determinations, and transient rod maintenance are sufficient to help ensure operability. The NRC staff reviewed TS 4.2.3, Specifications 1, 2, 3, and 4, and finds that TS 4.2.3 establishes surveillance intervals to help ensure the operability and performance of the control rods. The NRC staff also finds that the surveillance intervals and TS 4.2.3, Specifications 1, 2, 3, and 4, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and therefore, is acceptable.

TS 5.4, TS 3.2.3, and TS 4.2.3, help ensure that the control rods will promptly shut down the reactor upon a scram signal. The NSCR shall not be operated if any damage is found on the control rods or drives. The scram time for the control rods is specified to be less than 1.2 seconds and is measured annually by the licensee. The safety analysis assumes this 1.2-second value.

Based on its review of the information provided in the SAR, as supplemented, the NRC staff concludes that the control rods conform to the applicable design bases and can shut down the reactor from any operating condition. Specifically, the NRC staff concludes that there is reasonable assurance that the scram features will perform as required during the renewal period to ensure fuel integrity and protect public health and safety. The NRC staff reviewed the design and functional description of the transient rod system and concludes that it offers reasonable assurance that pulses will be reproducible and limited to values that maintain fuel integrity. The control rod design for the NSCR includes reactivity worth that can control the excess reactivity planned for the NSCR, including the assurance of an acceptable shutdown reactivity and margin. The NRC staff finds that the licensee has evaluated appropriate design limits, limiting conditions for operation, and surveillance requirements for the control rods. Based on the above discussion, the NRC staff concludes that the TS requirements related to the NSCR control rods are acceptable.

2.3.5 Neutron Moderator and Reflector

The predominant moderator of the NSCR core is the zirconium hydride incorporated into the LEU fuel elements. The pool water between the fuel elements also serves as a moderator.

The top and bottom 3.5 in. (8.9 cm) of a fuel element contain graphite that serves as a reflector. In addition, individual graphite reflector blocks may be used in the grid plate to provide neutron reflection for the core. A graphite coupler box and a thermal column also serve as reflectors. Absent any graphite reflectors, the pool water serves as a reflector.

The licensee performs visual inspections, during core configuration changes, on the graphite blocks to ensure the graphite reflectors maintain their structural integrity, and to ensure that any degradation is observed and evaluated. The NRC staff reviewed the licensee's neutron reflector and moderator, as discussed in the SAR, and finds that they are consistent with those used in other research reactors. In addition, routine inspections are adequate to identify any physical degradation and implement corrective actions, as necessary. Based on its review of the SAR, as supplemented, the NRC staff concludes that there is reasonable assurance that the neutron moderator and reflectors will function safely in the NSCR core for the renewal period without adversely affecting public health and safety.

2.3.6 Neutron Startup Source

The licensee states that the primary function of the neutron source is to provide sufficient neutrons such that the log power instrumentation will function properly during startup. TS 3.2.2, discussed in Section 5.3.2.2 of this SER, requires that the control and safety system have an interlock that prevents control rod withdrawal when the log power-level channel reading is less than 4×10^{-3} watts (W) corresponding to approximately 2 counts per second.

The SAR describes the startup source for the NSCR as an aluminum-clad antimony-beryllium neutron source. It is located in a neutron source holder in a graphite reflector element to provide neutron multiplication data during startup on the log power channel. Because the antimony is activated by core neutrons, the source normally remains in the reactor during operation. A neutron-source clad failure would be detected during the routine analysis of pool water as required by TS 4.8.1, Specification 1 (discussed in Section 2.4.7 of this SER), to periodically measure the radioactivity content of the reactor pool. The NRC staff finds that the surveillance requirements specified in TS 4.8.1, Specification 1, are acceptable for limiting the radioactivity content of the pool water and detecting potential damage to the source cladding.

The NRC staff reviewed the information provided in the SAR, as supplemented. Based on its review, the NRC staff concludes that the neutron startup source is adequate to allow controlled reactor startup, and therefore, is acceptable.

2.3.7 Core Support Structure

In the SAR, the licensee provides a description of the core support structure. An aluminum suspension frame supports the reactor grid plate. The suspension frame is a welded structure of ⅜ in. (0.9 cm) by 2 in. (5 cm) by 2 in. (5 cm) aluminum angle. An aluminum stabilizer frame, bolted to the bottom of the grid plate, provides for vertical support. Stainless steel guides on the bottom of the stabilizer fit between tracks on the pool floor. This allows accurate repositioning of the reactor core. The stabilizer also allows lowering of the core until it rests on the bottom of the pool, thereby preventing swaying that could introduce reactivity variations. Stainless steel pins of ¼ in. (0.6 cm) attach the aluminum frame to the grid plate on all four corners. This grid plate supports the TRIGA fuel elements. The grid plate contains 54 holes arranged in a nine by six array to accommodate fuel bundle assemblies, graphite blocks, instruments, and experiment devices. A safety plate assembly beneath the reactor grid plate stops a fuel followed control rod 2 in. (5 cm) below its normal down position should the control rod become detached from its mounting. This prevents control rods from falling out of the core.

A bridge that spans the reactor pool supports the suspension frame, the control rod drives, the nuclear instrumentation detectors, and the diffuser system. Mounted on four wheels, the bridge travels on rails provided at the sides of the pool; thus, the reactor core can move from one position to another along the centerline of the pool. The bridge is hand operated, and its speed of travel is limited because of the large gear ratios involved. A cable bundle that lies in a covered trough is parallel to the south wall of the reactor pool and has sufficient slack for bridge movement. The bridge provides electric power, control-circuit wiring, and compressed air to support reactor operations.

The licensee also states that it visually inspects the bridge structure and would observe significant structural degradations. The licensee indicates that the bridge structure maintains its structural integrity and that visual inspections during reactor core changes are sufficient to recognize significant degradations.

On the basis of its review of the information provided in the SAR, as supplemented, the NRC staff concludes that there is reasonable assurance that the reactor bridge core support structure will function safely for the renewal period without adversely affecting public health and safety.

2.4 Reactor Pool

The reactor pool is a reinforced, above-ground, concrete, two-section pool with a volume containing 109,000 gallons (412,610 l) of water. The depth of the pool is 33 ft (10 m). The pool is lined with a stainless steel liner, penetrated by a thermal column, pneumatic tubes, beam ports, and an irradiation cell window. The pool and liner also contains piping penetrations for drain lines, water recirculation lines, a pool leakage collection line, and primary coolant lines. The pool water level is approximately 26 ft (8 m) above the top of the core during normal operation. An alarm sounds if the water level drops 3 ft (0.9 m) below the reference operating level. The pool provides storage for irradiated fuel elements. The reactor pool is a component of the reactor coolant system.

2.4.1 TS 5.8 Reactor Pool Water Systems

TS 5.8 states the following:

Specifications

1. The reactor core shall be cooled by natural convective water flow.

2. The pool water inlet and outlet pipe for the demineralizer, diffuser, and skimmer systems shall not extend more than 15 feet below the top of the reactor pool when fuel is in the core.

3. Pool water inlet to the heat exchanger shall have an emergency cover within the reactor pool for manual shut off in case of pool water loss due to external pipe system failure.

4. A pool level alarm with readouts in the control room and at a continuously monitored remote location shall indicate a pool level less than 3 feet below the reference operating level.

TS 5.8, Specification 1, helps ensure that the assumptions used in the thermal-hydraulic calculations, which show that the TRIGA-LEU core can operate continuously in a safe manner at the licensed reactor power level with natural convection water flow and sufficient bulk pool cooling, are maintained.

TS 5.8, Specification 2, helps ensure that in the event of accidental siphoning of pool water through inlet and outlet pipes of the demineralizer, skimmer, or diffuser systems, the pool water level will drop to no more than 15 ft (4.6 m) from the top of the pool, ensuring adequate core cooling and shielding, and allow for long-term shutdown of the reactor.

TS 5.8, Specification 3, helps ensure that the pool water inlet coolant line to the heat exchanger shall have an emergency cover for use in the event of a pipe failure. This cover provides an additional measure to help prevent water drainage from the reactor pool (Ref. 59).

TS 5.8, Specification 4, helps ensure that a pool level alarm is available to signal a substantial coolant water loss in the event of a leakage path developing that could potentially drain the reactor pool and require appropriate corrective action. This alarm is observed in the reactor control room and in a remote continuously monitored location.

TS 5.8, Specifications 1, 2, 3 and 4, help ensure the maintenance of important design aspects of the reactor pool water system to provide cooling for the reactor core, to prevent the inadvertent loss of cooling due to siphoning of the pool water, to detect pool water leakage, and to provide the coolant pressure (water level) assumed in the thermal-hydraulic calculations. In addition, TS 5.8, Specification 4, provides additional monitoring capability, as it requires an alarm to indicate water loss if a leakage path develops that could potentially drain the reactor pool.

The NRC staff reviewed TS 5.8, Specifications 1, 2, 3 and 4, and finds that TS 5.8 specifies important aspects of the reactor pool water system and that these aspects are acceptable. On the basis of its review, the NRC staff concludes that TS 5.8, Specifications 1, 2, 3, and 4 are acceptable.

The SAR, Section 5.2, states that during normal operation, the water temperature in the reactor pool is maintained between 21 degrees C (70 degrees F) and 27 degrees C (80 degrees F) by a closed loop cooling system with a design flow rate of 1,000 gallons (3,785 l) per minute. A pump takes water from the reactor pool, passes it through the plate-type heat exchanger, and returns it through two pipes to the reactor pool. This system is controlled remotely from the control room. The secondary coolant system provides the heat removal capability for the water in the reactor pool. Heat is removed from the plate-type heat exchanger by an evaporative cooling tower. Water from the cooling tower basin is pumped through the heat exchanger at a nominal flow rate of 1,575 gallons (5,962 l) per minute and returned to the tower. The domestic water supply provides makeup to the tower basin, needed because of evaporation losses and blowdown. Level instrumentation located in the basin controls the makeup. Blowdown water is directed to the facility's sanitary sewer system, which is operated in compliance with all state and local regulations for water treatment discharges. Design features of the cooling system allow for transfer of reactor heat from the primary system under all operating conditions, but are required only for core heat removal during power operations. Malfunctions of the secondary cooling system will not lead to fuel failure because the system is only needed for heat removal during reactor operation. And, if a malfunction were to occur during reactor operation, the heat capacity of the reactor pool system provides sufficient time for the operators to shut down the reactor to allow natural convection pool cooling of the fuel. A secondary cooling system leak could result in an uncontrolled release of radioactivity to the environment; however, the limitations of TS 3.8.1, Specification 3, helps ensure that the concentrations of radionuclides in the bulk pool water are no higher than the limits specified in 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 Concentrations," of 10 CFR Part 20. Because the pool water radionuclide concentration is within the 10 CFR Part 20 limits for release to the environment, a leak from the reactor pool or primary coolant system through the secondary cooling system to the environment will not endanger public health and safety. The instrumentation on this system also provides the necessary status indications in the control room.

A diffuser system controls the dose rate above the core due to gaseous N-16. (See Section 3.1.1.1 of this SER for further discussion).

The licensee also states that there is a robust pool-level monitoring system in place to respond to any reactor pool water leakage. TS 4.8, Specification 2, requires the weekly monitoring of the pool water level. However, the licensee states that in practice, the water level is monitored daily. TS 5.8, Specification 4, states that an alarm for a loss of 3 ft (0.9 m) of reactor pool water is received both locally and at the 24-hour manned location. Procedures are in place for a rapid response from NSCR on-call staff when a reactor pool level alarm is received. During reactor operation, a leak of the reactor pool water would be readily noticeable by the reactor operators since the pool water level is directly observable from the control room and also monitored via a closed-circuit camera system.

The licensee states that any pool leakage would be detected due to the pool water level being continuously monitored by the reactor operators. In addition, the frequency and quantity of makeup water added to the pool is recorded. The licensee states that the minimum change in water level that it is able to observe is about 0.15 in. (38 cm), corresponding to 500 gallons (1,893 l). The licensee states that the NSCR staff is well aware of the requirements for pool makeup water to replace water lost to evaporation and would investigate off-normal changes. A small leak would start the makeup water system, which is able to compensate for small leakages. Any further drop in water level to 3 ft (0.9 m) below the normal operating pool level would activate a low-water alarm. Most leakage pathways through pipes, fittings, beam ports, and the pool wall would result in leakage into observable areas. This leakage would collect in the waste water transfer sump and the waste water storage tanks and would be analyzed before release to the waste water processing facility. If leakage is not observable and water leaks to the environment, TS 3.8.1, Specification 3, helps ensure that the concentrations of radionuclides in the bulk pool water are no higher than the limits specified in Appendix B, Table 2. Because the pool water radionuclide concentration is within the 10 CFR Part 20 limits for release to the environment, a leak from the reactor pool or primary coolant system would not endanger public health and safety.

The reactor pool is filled with demineralized water. The conductivity of the pool water is controlled to minimize the corrosion of the fuel element cladding, minimize the corrosion of reactor components, and minimize neutron activation of dissolved materials in the pool water. The reactor pool water quality requirements are specified in TS 3.8.1 and TS 4.8.1.

The NRC staff completed an analysis (Ref. 68) that demonstrated that a conductivity limit no greater than 5×10^{-6} mhos/cm will ensure that the pH range is limited to 5.6 to 5.8, which is consistent with the guidance provided in NUREG-1537 to maintain the pH range of 5.0 to 7.5. Since the licensee chose this conductivity limit (5×10^{-6} mhos/cm), there was no need for a TS requirement to limit the reactor pool water pH.

2.4.2 TS 3.8.1 Primary Coolant Purity

TS 3.8.1 states the following:

Specifications

1. The reactor shall not be operated for a period exceeding two weeks if the two week averaged conductivity of the bulk pool water is higher than 5×10^{-6} mhos/cm.

2. The concentrations of radionuclides in the bulk pool water shall be no higher than the values presented for water in 10 CFR Appendix B to Part 20 Table 2.

TS 3.8.1, Specification 1, helps ensure that the conductivity of the bulk pool water in the tank is maintained at or less than 5 micromhos per centimeter (μmhos/cm) to control the potential corrosion of reactor components. The licensee stated in the SAR, as supplemented, that a small rate of corrosion continuously occurs in a water-metal system. Limiting this rate extends the longevity and integrity of the fuel cladding. The licensee also states that the corrosion limits help ensure that oxide buildup on the cladding will not reduce the heat transfer between the cladding and coolant. The licensee proposed a conductivity limit that is used by the other TRIGA licensees as a longstanding value for research reactors, which has been shown to be effective in controlling corrosion in aluminum and stainless steel systems. TS 3.8.1, Specification 2, helps ensure that the radioactive content of the primary coolant water remains low and known in the event of any pool or primary coolant leakage.

The NRC staff reviewed TS 3.8.1 and finds that the conductivity limit of 5 μmhos/cm, and a pool water radioactivity limit no higher than the values listed in 10 CFR Part 20, Appendix B, Table 2, are consistent with those of other TRIGA reactors and with the guidance in NUREG-1537. Based on the information provided above, the NRC staff concludes that TS 3.8.1, Specifications 1 and 2, are acceptable.

2.4.3 TS 4.8.1 Primary Coolant Purity

TS 4.8.1 states the following:

Specifications

1. A sample of the coolant shall be collected and analyzed for radioactive material content at least weekly during periods of reactor operation and at least quarterly during extended shutdowns.

2. Conductivity of the bulk pool water shall be measured and recorded weekly.

TS 4.8.1, Specification 1, helps ensure that the radionuclide content of the pool water is measured weekly to in order to promptly detect increased levels of radioactivity content. In addition, periodic monitoring of the pool water will help ensure that the water is within 10 CFR Part 20 release limits under any circumstances or condition of operation.

TS 4.8.1, Specification 2, helps ensure periodic monitoring of the primary coolant water conductivity to alert the operators of any changes in the primary coolant water chemistry.

The NRC staff reviewed TS 4.8.1, Specifications 1 and 2, finds that the surveillance intervals are consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.8.1, Specifications 1 and 2, are acceptable.

2.4.4 TS 3.8.2 Primary Coolant Level and Leak Detection

TS 3.8.2 states the following:

Specifications

1. The reactor shall not be operated if the pool level is below 3 feet from the reference operating level.

2. The reactor shall not be operated if the pool level unexpectedly drops one foot from its operating level.

3. The pool level alarm shall initiate an alarm signal in the control room and at a continuously monitored off-site facility if the pool level is lower than 3 feet from its reference operating level.

TS 3.8.2, Specification 1, helps ensure that the reactor will not be operated when the water level is 3 ft (0.9 m) from the reference water level in order to maintain greater than 20 ft (6 m) of water above the core.

TS 3.8.2, Specification 2, helps ensure that the reactor will be shut down after detection of a sudden 1 ft (0.3 m) drop in the pool water level from the normal operating water level.

TS 3.8.2, Specification 3, helps ensure that the reactor is operated only with an operable water-level monitoring and alarm system. The alarms are monitored in the reactor control room and at a monitored remote location. Procedures are in place for responding to the alarms for both the remote location and the NSCR on-call staff.

2.4.5 TS 4.8.2 Primary Coolant Level and Leak Detection

TS 4.8.2 states the following:

Specifications

1. The reactor pool level shall be recorded at least weekly.

2. The pool water level alarm shall be channel tested weekly.

3. The pool water level alarm shall be channel checked prior to reactor operation.

TS 4.8.2, Specification 1, helps ensure a weekly record of the pool water level. The reactor pool water level record can be used by the operators to detect an unusual change in the makeup of the pool water, which could indicate a potential leak.

TS 4.8.2, Specification 2 and Specification 3, help ensure that the pool water level alarm is maintained by performing a channel test weekly and a channel check before reactor operation. The NRC staff finds that these surveillance intervals are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

The NRC staff reviewed the information provided by the licensee and finds that the water level instrumentation is adequate to help ensure that the pool water level exceeds 20 ft (6.1 m) above the core at all times. In addition, the pool water level is continuously monitored, and procedures are in place to investigate any leakage when a pool-level alarm is initiated, even during off-hours. The NRC staff finds that these surveillance intervals are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Therefore, the NRC staff finds that TS 3.8.2, Specifications 1, 2, and 3, and TS 4.8.2, Specifications 1, 2 and 3, are acceptable. On this basis, the NRC staff concludes that TS 3.8.2, Specifications 1, 2, and 3, and TS 4.8.2, Specifications 1, 2, and 3, are acceptable.

The NSCR pool has two compartments. In the event of a significant pool leak, the leak is not likely to occur in both sections at the same time. Accordingly, the reactor core can be moved to the non-leaking section, and by inserting a gate, the two sections can be separated and appropriate repairs to the leaking section can be undertaken. A rapid loss of pool water would result in a loss-of-coolant accident, discussed in Section 4.1.2 of this SER.

2.4.6 TS 3.8.3 Primary Coolant Temperature

TS 3.8.3 states the following:

Specification

The reactor shall not be operated when pool temperature exceeds 60° C.

TS 3.8.3 helps ensure that the pool water temperature is consistent with the assumptions used in the thermal-hydraulic calculations and specifies the limit for pool water temperature as 60 degrees C (140 degrees F) (Ref. 57). See Section 2.7.1 of this SER for additional discussion and NRC review of the 60 degree C (140 degrees F) analysis. There is a continuous temperature indication with an alarm set at 50 degrees C (122 degrees F). The NRC staff finds that the 60 degree C (140 degrees F) limit specified in TS 3.8.3 is consistent with the design assumptions provided in the licensee's RAI response (Ref. 57). Based on the information described above, the NRC staff concludes that TS 3.8.3 is acceptable.

2.4.7 TS 4.8.3 Primary Coolant Temperature

TS 4.8.3 states the following:

Specifications

1. Primary coolant temperature shall be recorded every 30 minutes while the reactor is operating, or immediately following reactor startup if the reactor is to be operated for less than 30 minutes.

2. The primary coolant temperature channel shall be calibrated semiannually.

TS 4.8.3, Specification 1, helps ensure that the pool temperature is maintained during operation or following startup for short duration of operations in order to detect any changes in the water temperature that may require operator actions. Because of the large volume of coolant in the pool, pool primary coolant temperature changes slowly with time even when the reactor is operated at full licensed power.

TS 4.8.3, Specification 2, helps ensure that the pool water temperature measuring channel is accurate through the implementation of a semiannual calibration and that required actions will be taken consistent with the analysis provided in the SAR.

The NRC staff reviewed TS 4.8.3, Specifications 1 and 2, and find that TS 4.8.3 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff also finds that the temperature limit in TS 3.8.3, as monitored by the surveillance requirements in TS 4.8.3, Specifications 1 and 2, in combination with operator training and existing cooling capacity, helps ensure that the pool temperatures will stay below 60 degrees C (140 degrees F). Based on the information provided above, the NRC staff concludes that TS 3.8.3 and TS 4.8.3, Specifications 1 and 2, are acceptable.

The NRC staff reviewed the information provided in the SAR, as supplemented, regarding pool water level and quality. The NRC staff finds that the water-level instrumentation and the water quality program are adequate to help ensure that the pool water level exceeds 20 ft (6 m) above the core at all times and that the water quality is maintained. In addition, pool water level is monitored, and NSCR staff would investigate leakage. Based on the information provided above, the NRC staff concludes that the likelihood of a significant release to the environment resulting from pool leakage is extremely low. If a limited amount of primary coolant was released to the environment either by a pool or heat exchanger leak, the requirement in TS 3.8.1, Specification 2, to maintain primary coolant radioactivity levels below 10 CFR Part 20 limits for release to the environment helps ensure doses to the public will be within acceptable limits.

2.5 Biological Shield

As described in the SAR, Section 4.4, the NSCR biological shield consists of the stainless-steel-lined concrete pool structure and the pool water. The biological shield was designed to limit doses from reactor operations and reactor-related sources to acceptable levels. The NRC inspection program routinely reviews the licensee's radiation protection program, including routine independent verification of radiation levels in the facility, as well as observation of radiation measurements obtained by the licensee's staff. Based on a review of the information provided by the licensee, operational experience, and results from the NRC inspection program, the NRC staff concludes that there is reasonable assurance that during the renewal period, the NSCR biological shield will limit exposures from the reactor and reactor-related sources of radiation so that the limits of 10 CFR Part 20 will not be exceeded.

2.6 Nuclear Design

The information discussed in this section establishes the design bases for the content of other chapters in this SER. The NSCR nominally operates at a steady-state thermal power level of 1 MWt.

The SAR, Section 4.5, states that the NSCR was the first of the fuel bundle (MTR to TRIGA conversion) TRIGA reactors to be converted from the use of HEU fuel to the use of 30/20 LEU fuel. Thus, measured data from a TRIGA reactor operating with 30/20 LEU fuel bundles were unavailable for validating the computational procedures to be used in the design of the 30/20 LEU fuels and core used in the NSCR. The PRNC TRIGA reactor had operated with a core consisting of fuel bundles with HEU FLIP fuel and the neutronic and thermal-hydraulic characteristics had been experimentally determined during commissioning testing and subsequent operation. The known characteristics of the PRNC TRIGA reactor provided an opportunity to benchmark the computational codes and procedures that were used to determine the neutronic and thermal-hydraulic characteristics of the NSCR 30/20 LEU core. Table 2-1 of this SER presents a comparison of the PRNC HEU FLIP fuel element and the NSCR 30/20 LEU fuel element. The 30/20 LEU fuel introduces a large increase in uranium (U)-238 in the fuel element because of the decreased enrichment, but all other parameters are similar to those of the HEU FLIP fuel.

The licensee also states that the computations for the PRNC TRIGA reactor produced operational parameters that compared favorably with the actual measured values from the commissioning tests conducted by GA for the PRNC TRIGA core loaded with FLIP (HEU) fuel. The experimentally measured parameters included the approach to criticality, the reactivity for the fully loaded core, the control rod calibration values, the reactivity loss and peak fuel temperatures as a function of reactor power, and pulsing performance including peak power, peak fuel temperature, and energy production, all as a function of prompt reactivity insertion (Ref. 2).

Furthermore, the licensee states that the steady-state parameters for the NSCR LEU 30/20 core were calculated using the same computational procedures benchmarked for the PRNC TRIGA reactor adapted to the NSCR four-rod configuration. The reactor was analyzed with two computer codes that have been used extensively for nuclear reactor applications: the finite difference three-dimensional diffusion and transport theory code (DIF3D), and the Monte Carlo Neutron Transport code (MCNP5). The Burnup Replacement Package code (BURP), a GA depletion computer code developed for TRIGA reactors, was used for burnup calculations. The GA space-independent reactor kinetics heat transfer code (BLOOST) was used to predict the dynamic response of the core following the ejection of a transient rod.

The NRC staff finds that during the startup testing of the NSCR with LEU 30/20 fuel, it was demonstrated that the neutronic and thermal-hydraulic characteristics were adequately predicted by calculations and were documented in the Startup Report (Ref. 8). Based on the information provided above, the NRC staff concludes that the computational methodology used by the licensee in the nuclear and thermal-hydraulic analysis of the NSCR is acceptable.

2.6.1 Normal Operating Condition

Steady-State Operation

The NSCR is licensed to operate at a steady-state maximum thermal power level of 1.0 MWt. Information provided by the licensee in an updated SAR Section 4.7 (Ref. 57), indicates the following results for steady state operation at 1.0 MWt:

- Maximum Fuel Temperature – 526 degrees C (979 degrees F)
- Maximum Coolant Exit Temperature – 101 degrees C (214 degrees F)
- Average Coolant Exit Temperature – 90.8 degrees C (195 degrees F)
- Minimum Departure from Nuclear Boiling Ration (DNBR) – 2.53

and at a peak power level 1.25 MWt (power level trip setpoint):

- Maximum Fuel Temperature – 562 degrees C (1044 degrees F)
- Maximum Coolant Exit Temperature – 103 degrees C (217 degrees F)
- Average Coolant Exit Temperature – 95.5 degrees C (204 degrees F)
- Minimum DNBR – 2.01

These analytically predicted values of the NSCR Maximum Fuel Temperature and Minimum DNBR provide satisfactory margin to the limits and guidance provided in NUREG-1537 of 1150 degrees C (2100 degrees F), and 2.0, for the Maximum Fuel Temperature and Minimum DNBR, respectively.

2.6.1.1 TS 3.1.1 Steady State Operation

TS 3.1.1 states the following:

> Specification
>
> The reactor power level shall not exceed 1.0 megawatt (MW) during steady state operation.

TS 3.1.1 helps ensure that the licensee operates the NSCR with a maximum steady-state thermal power level consistent with the analysis provided in the SAR and the licensed thermal power level of 1 MWt. TS 3.1.1 specifies a steady-state power-level limit to help ensure that natural convection of pool water provides adequate cooling for the fuel elements. As discussed in the thermal-hydraulic analysis (Section 2.7 of this SER), operation of the NSCR at a higher thermal power level of 1.25 MWt would still provide for sufficient safety margins (Ref. 57).

Additionally, Table 2 of TS 3.2.2 provides a reactor scram at a thermal power level of 1.25 MWt. On the basis of its review of the information provided above, the NRC staff finds that TS 3.1.1 which helps to limit the operation of the NSCR to a power of 1.0 MWt during steady-state operation provides an acceptable margin of safety for operation. Based on the information provided above, the NRC staff concludes that TS 3.1.1 is acceptable.

2.6.1.2 TS 4.1.1 Steady State Operation

TS 4.1.1 states the following:

> Specification
>
> A channel calibration shall be made of the power level monitoring channels by the calorimetric method annually.

TS 4.1.1 helps ensure that the reactor power measuring channels are calibrated annually using the calorimetric method, which is the standard method for calibrating the reactor power measuring channels. The licensee states in the SAR that an annual calibration helps ensure that the power-level measuring channels are accurately indicating the reactor power level. The NRC staff reviewed TS 4.1.1 and finds that this surveillance interval is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

Based on the information provided above, the NRC staff finds that the requirements in TS 4.1.1 helps ensure that the reactor power will be accurately indicated and that reactor operators will be able to maintain the thermal power level not to exceed 1.0 MWt during steady-state operation. Therefore, the NRC staff concludes that TS 4.1.1 is acceptable.

Pulse Mode Operation

In SAR Section 4.5, the licensee states that the NSCR is designed to be pulsed from a low to a high power level by the rapid insertion of reactivity. In this mode of operation, the maximum reactivity insertion is limited to that which will limit the peak fuel temperature to 830 degrees C (1,526 degrees F) and the pulse may not be initiated from a core power level in excess of 1 kWt. Pulsing from a power level greater than 1 kWt is prevented by a required interlock, which prevents the transient rod from activating. In addition, an adjustable timer is set to initiate a transient rod scram 15 seconds or less following the initiation of a pulse. TS 3.1.2 applies to the peak fuel temperature in the reactor resulting from a rapid insertion of reactivity to help ensure that fuel element damage does not occur.

2.6.1.3 TS 3.1.2 Pulse Mode Operation

TS 3.1.2 states the following:

> Specification
>
> The reactivity to be inserted for pulse operation shall not exceed that amount which will produce a peak fuel temperature of 1526°F (830°C). In the pulse mode the pulse rod shall be limited by mechanical means or the rod extension physically shortened so that the reactivity insertion will not inadvertently exceed the maximum value.

TS 3.1.2 helps ensure the maximum allowable reactivity insertion for pulsing so that the NSCR can be safely pulsed without fuel damage. The licensee states in the SAR that the pulse capability is limited to help ensure that the fuel temperature stays below the SL of 1,150 degrees C (2,100 degrees F) for the 30/20 fuel, as discussed in Section 2.6.3 of this SER, and the peak fuel temperature remains below 830 degrees C (1,526 degrees F). GA recommends the fuel temperature limit of 830 degrees C (1,526 degrees F) during pulsing to ensure that no fuel damage occurs because of internal pressure caused by hydrogen migration (Ref. 27). The limit of 830 degrees C (1,526 degrees F) was established following minor fuel distortion experienced at the NSCR (Ref. 26). Note that fuel damage is different than fuel failure which the SL helps protects against. Fuel damage would in all likelihood require that the fuel be removed from service.

TS 3.2.2 (discussed in Section 5.3.2.2 of this SER) requires a preset timer to initiate a scram of the transient rod 15 seconds or less after the initiation of a pulse and also an interlock to prevent pulsing when the reactor power level is 1 kWt or above. TS 4.2.2, Specification 1 (discussed in Section 5.4.2.2 of this SER), requires periodic testing of the timer and the interlock.

2.6.1.4 TS 4.1.2 Pulse Mode Operation

TS 4.1.2 states the following:

> Specification
>
> The reactor shall be pulsed semiannually to compare fuel temperature measurements and core pulse energy with those of previous pulses of the same reactivity value. The

reactor shall not be declared operational for pulsing until such pulse measurements are performed and are determined to be acceptable.

TS 4.1.2 helps establish the surveillance requirement for pulse mode operation by verifying the maximum fuel temperature and integrated energy for a pulse. The periodic reactor pulsing allows a comparison to be made with previous similar pulses and a determination of whether fuel or core characteristics are changing.

In the SAR, the licensee states that during the startup testing of the LEU core, calculations were performed to guide the experimental determination of the maximum pulse that would not exceed the temperature limitation. As shown in the Startup Report, the maximum allowable pulse insertion was determined to be $1.91. The NRC staff reviewed the information in the SAR and finds that the licensee's analysis demonstrated that an administrative limit of $1.91 of reactivity for pulses provides adequate protection for limiting the NSCR fuel temperature to 830 degrees C (1,526 degrees F), as required by TS 3.1.2. The NRC staff concludes that the administrative limit of $1.91 provides acceptable safety margins with regard to limiting the maximum fuel temperature to 830 degrees C (1,526 degrees F). Section 2.6.2 of this SER discusses this further.

On the basis of its review, the NRC staff finds that the peak fuel temperature limit and the associated maximum reactivity addition limit for pulsing help ensure that the reactor can be safely pulsed without concern for fuel damage. Additionally, TS 3.1.2 and TS 4.1.2 provide an acceptable peak fuel temperature limit and surveillance requirement for pulse mode operation. Based on the information provided above, the NRC staff concludes that the NSCR pulse mode operation and TS 3.1.2 and TS 4.1.2 are acceptable.

2.6.1.5 Control Rod Worths

Section 2.2 and Section 2.3.4 of this SER describe the reactor core and the control rods. During the startup testing of the LEU core, initial criticality was achieved with 60 fuel elements installed in the core, which was within the predicted range of 58–68 elements.

The SAR states that after criticality was achieved, the reactor was fully loaded to the desired final configuration, and individual rod worths were determined. The total reactivity worth of all the rods was $17.19. In its reactor Startup Report, the licensee states that the total rod worth as determined on July 9, 2011, is $14.62. The change in control rod worth reflects the effects of fuel burnup and the buildup of Sm-149 in the fuel since the initial LEU core startup. The current core excess reactivity is $6.50 with the core assembly against the thermal column and $5.55 away from the thermal column. This corresponds to an SDM against the thermal column of $2.20 and away from the thermal column of $3.15 as provided by the licensee in an RAI response (Ref. 17).

TS 3.1.6 limits the excess reactivity in the core to less than $7.85. The licensee states that the excess reactivity is reduced by the fuel burnup and by the buildup in the core of long-lived fission product poisons, primarily Sm-149. Therefore, the licensee concluded that there is adequate margin to maintain the reactor shutdown when all of the control rods are inserted with the most reactive control rod and the regulating rod fully withdrawn, which satisfies the most reactive rod withdrawn criteria. The NRC staff reviewed the information in the SAR and finds that the NSCR control rod worths meet the SDM requirement. Thus, on the basis of its review, the NRC staff concludes that the values of the NSCR control rod worths are acceptable.

2.6.1.6 Excess Reactivity

Section 2.2.4 of this SER discusses the excess reactivity criteria. The licensee has proposed in TS 3.1.6 that the maximum excess reactivity of the reactor with the core in a cold, xenon-free condition is less than 5.5-percent delta reactivity (%Δk/k) which is equivalent to $7.85. The licensee's analysis, as stated in the Conversion SAR, and its more recent excess reactivity information (Ref. 17), calculated the current core excess reactivity of $6.50 with the core assembly against the thermal column and $5.55 with the core assembly away from the thermal column. Since both these conditions of the core assembly demonstrate that the core excess reactivity remains less than the value specified in TS 3.1.6, the NRC staff finds the TS 3.1.6 value for the excess reactivity acceptable.

2.6.1.7 Shutdown Margin

Section 2.2.3 of this SER describes the criteria for SDM and the basis for an SDM of $0.50. The purpose of defining the SDM is to ensure that the reactor can be shut down by an acceptable margin even if the most reactive control rod fails to insert into the core. In addition to assuming that the most reactive control rod is not available, TS 3.1.4 places constraints on the core condition and experiments. The core conditions are those most limiting for determining SDM. Non-secured experiments are considered to be in their most reactive state to ensure that the reactor is not subcritical because of experiments that could be readily removed from the core. As demonstrated in the Startup Report, the measurements performed by the licensee during the startup testing of the LEU core showed that the measured value of the SDM met the above criteria. TS 4.1.3 and TS 4.1.6 present the surveillance requirements to ensure compliance with SDM and excess reactivity requirements, as discussed in Section 2.2.3 of this SER.

On the basis of its review of the information provided above, the NRC staff concludes that the SDM, as stated in the SAR, and provided in TS 3.1.3, is acceptable.

2.6.1.8 Core Burnup

The core burnup calculations demonstrated that assuming a power level of 1.0 MWt, the lifetime of the initial core (with no fuel shuffling) is about 2,000 MWDs at 1.0 MWt, with full-equilibrium xenon poisoning, and about $0.60 reactivity remaining for burnup or experiments. The initial core loading was expected to provide operation at 1.0 MWt as required for a time period ranging from 18 to 35 years. Typically, experience has shown that a shuffled core with the movement of less burned fuel clusters to the core center can add 20 percent to the life of an un-shuffled core. The impact of core aging on core physics parameters is discussed in Section 2.6.2 of this SER.

2.6.1.9 Conclusion

The NRC staff reviewed the licensee's analysis for the LEU core and finds that the core contains all of the components for an operable reactor core. The NRC staff finds that the licensee used input parameters justified by analysis presented in the Conversion SAR, as supplemented, and justified by the results of startup testing as summarized in the Startup Report. The NRC staff finds that the licensee adequately analyzed the reactivity effects of individual core components and verified the analysis in the startup program as presented in the Startup Report. TSs related to the normal operating conditions of the reactor core include limits on excess reactivity, the minimum SDM, allowable core configurations, and surveillance requirements for the core reactivity parameters and reactivity worth of the control rods. These TSs are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

The NRC staff finds that the analysis presented in the SAR, as supplemented, adequately support the TSs and demonstrate that normal reactor operation should not lead to the release of fission products from the fuel. Based on these considerations, the NRC staff concludes that the licensee has adequately analyzed expected normal reactor operation during the period of the renewed license.

2.6.2 Core Physics Parameters

The core physics parameters of temperature coefficient of reactivity, neutron lifetime, effective delayed neutron fraction, temperature coefficient, and void coefficient, were determined during the analysis for the conversion of the NSCR to LEU fuel (Ref. 2 and Ref. 7).

The licensee states that an important safety feature of a TRIGA reactor is the reactor core's inherent large, prompt, negative temperature coefficient of reactivity, resulting from an intrinsic molecular characteristic of the U-ZrH_x matrix at elevated temperatures. The negative temperature coefficient results principally from the neutron-hardening properties of the fuel matrix at elevated temperatures, which increases the leakage of neutrons from the fuel-bearing material into the water moderator material, where they are absorbed preferentially. This reactivity decrease is a prompt effect because the fuel and zirconium hydride are mixed homogeneously; thus, the zirconium hydride temperature rises essentially simultaneously with fuel temperature, which is directly related to reactor power. An additional contribution to the prompt, negative temperature coefficient is the Doppler broadening of U-238 resonances at high temperatures, which increases nonproductive neutron capture in these resonances.

Because of the large, prompt, negative temperature coefficient, the fuel matrix will rapidly and automatically compensate for a step insertion of reactivity resulting in an increasing fuel temperature. This can terminate the resulting power excursion without any dependence on the electronic or mechanical reactor safety systems or the actions of the reactor operator. Also, changes of reactivity resulting in a change in fuel temperature during steady-state operation can be rapidly compensated for by the fuel matrix, thus limiting the reactor steady-state power level (Ref. 23). This inherent characteristic of the U-ZrHx fuel has been the basis for designing TRIGA reactors with a pulsing capability as a normal licensed mode of operation.

For the NSCR LEU core, the prompt negative temperature coefficient of reactivity was calculated at 23 degrees C (73 degrees F) to be -5.3×10^{-5} Δk/k/degree C. During pulsing, the zirconium hydride dominates the value of the temperature coefficient and was found to be -13.1×10^{-5} Δk/k/degree C at 700 degrees C (1,292 degrees F). The values quoted for this core are similar to values reported by GA and others for TRIGA reactors. Therefore, the fuel will compensate for a large step insertion of reactivity as the temperature increases, causing the transient to end. This large negative value of temperature coefficient, α_T, was designed into the fuel so that TRIGA reactors can be pulsed. For the pulsed mode of operation, TS 3.1.2 specifies a limit for the temperature of the fuel during a pulse. The licensee has calculated and determined experimentally that a step insertion of reactivity to $1.91 would result in peak fuel temperature during the pulse that is within the TS 3.1.2 limit. GA performed many tests with step insertions up to $5.00 before any fuel damage became apparent; therefore, the TS limit of $1.91 is well within the safety envelope established by GA.

A discussion of the NSCR neutron lifetime, delayed neutron fraction, and void coefficient follows.

The effective delayed neutron fraction, β_{eff}, for the NSCR core was calculated resulting in a value of 0.0070 for the NSCR TRIGA LEU 30/20 fuel core.

The licensee states that the prompt neutron lifetime, ℓ, was computed using the 1/v absorber method (where "v" represents the neutron velocity) using the DIF3D code, as the model for the core, and resulted in the prompt neutron lifetime, at beginning of life, in the NSCR core being equal to 26.3 microseconds (μsec). The licensee measured the value for the prompt neutron lifetime of 26.178 μsec, which validated the results of the computation using the DIF3D code.

The kinetic parameters, delayed neutron fraction and prompt neutron lifetime, vary only slightly over the life of the core. The delayed neutron fraction does not change at all, and the prompt neutron lifetime increases by 1.0 μsec.

The void coefficient of reactivity is defined for a TRIGA reactor as the negative reactivity per 1.0 percent void in the reactor core water. The void coefficient for the NSCR is -0.130 % Δk/k/percent void. In the SAR, Section 4.5.6.9, the licensee states that the impact on safety of an experiment flooding is insignificant since the NSCR reactivity insertion accident is analyzed for significantly larger additions of reactivity with no fuel damage. The reactivity insertion for a dry experimental region near the center of the core that is advertently flooded is $0.26, and significantly less than the reactivity needed for prompt criticality, $1.00.

The licensee states that the calculations were performed using the DIF3D code to determine the usable neutron flux in the sample irradiation positions. The calculations indicate that there are significant variations in the magnitude of the flux and that there are steep flux gradients across the irradiation positions. The maximum and minimum neutron flux, located in core position A6, which has the highest flux of the irradiation positions, is 1.12×10^{13} neutrons per centimeter square seconds (n/cm^2-s) and 8.45×10^{12} n/cm^2-s, respectively. The flux at this location was measured using irradiation foils. The predicted and measured fluxes differed by approximately 21 percent to 48 percent, which was attributed to the placement of the foil and the precise detail of the modeling versus measurements.

The NRC staff reviewed the licensee's analyses, as discussed above, and finds that the licensee considered appropriate core physics parameters. These values are also similar to those found acceptable at other TRIGA reactors. Based on the information provided above, the NRC staff concludes that the methods used to determine values of the core physics parameters and the values of the core physics parameters are acceptable.

2.6.3 Operating Limits

The regulations in 10 CFR 50.36(d)(1) require reactors to specify an SL and LSSSs. The SL is defined in 10 CFR 50.36(d)(1) as a limit on important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. LSSSs for nuclear reactors are defined as settings for automatic protective devices related to those variables having significant safety functions. Where an LSSS is specified for a variable on which an SL has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before an SL is exceeded.

The SL is the maximum fuel element temperature that can be permitted without resulting in failure of the fuel element cladding and the release of fission products.

2.6.3.1 TS 2.1 Safety Limit—Fuel Element Temperature

TS 2.1 states the following:

> <u>Specification</u>
>
> The temperature in a stainless steel-clad TRIGA LEU fuel element shall not exceed 2100°F (1150°C) under any conditions of operation.

TS 2.1 specifies the maximum fuel element temperature for the 30/20 LEU TRIGA fuel to prevent failure of the fuel cladding.

The licensee states in the SAR that a key parameter for a TRIGA reactor is the fuel element temperature. For the NSCR, this SL specification can be measured directly with a thermocouple embedded in an IFE. A loss in the integrity of the fuel element cladding could arise from a buildup of pressure in the gap between the fuel and cladding if the fuel temperature were to exceed the SL. The pressure is caused by the presence of air, fission product gases, and hydrogen from the disassociation of the hydrogen and zirconium in the moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The licensee also states in the SAR that the SL for the TRIGA fuel element is based on data that include the large amount of experimental evidence obtained during high-performance reactor tests on this fuel. The SL for the 30/20 LEU fuel is based on data indicating that the stress in the cladding resulting from the hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate yield stress (clad integrity failure), provided that the temperature of the fuel does not exceed 1,150 degrees C (2,100 degrees F) and the fuel cladding is below 500 degrees C (932 degrees F). The NRC staff evaluated the properties and performance of the TRIGA higher uranium wt% LEU fuel, including the 30/20 LEU fuel, in NUREG-1282 (Ref. 29) and approved for use with the provision that case-by-case analysis discusses individual reactor operating conditions when using the fuel. The NSCR SL is set at the temperature established in NUREG-1282.

An additional consideration is the need to provide adequate cooling relative to the maximum heat flux to prevent departure from nucleate boiling (DNB) and the resulting rapid increase in clad temperature, which will lead to failure of the clad (see Section 2.7 of this SER). A power level limit is calculated that ensures that the fuel temperature SL will not be exceeded and that film boiling will not occur. The design-bases analysis has shown that operation at a thermal power level of 1.0 MWt across a broad range of core and coolant inlet temperatures, with natural convection flow, will not lead to film boiling. Section 2.7 of this SER discusses this further.

The licensee states in the SAR that the LSSS is the measured IFE temperature that, if exceeded, will initiate a scram to prevent the fuel element temperature SL from being exceeded. For the NSCR, the LSSS is set equal to or less than 525 degrees C (977 degrees F) as measured in the IFE at specific locations in the core. Exceeding this limit causes a scram of the reactor and prevents the fuel from exceeding the SL. The location of the IFE is important for ensuring that the hottest fuel location in the reactor core is protected from excessive temperature. The relationship between the measured temperature in the IFE and the actual temperature at the fuel hot spot in the core has been determined to show that the setting of 525 degrees C (977 degrees F) protects the SL at the hottest point in the core. The IFE contains three axial thermocouples that measure the fuel temperature in a vertical distribution.

The NRC staff reviewed the licensee's analysis, as discussed above, and finds that TS 2.1 helps ensure that the maximum fuel element temperature SL for the NSCR 30/20 LEU fuel is consistent with the SLs used for other TRIGA reactor fuel elements (supported by research conducted by GA) and has been previously approved by the NRC staff. Based on the information provided above, the fuel element temperature SL is acceptable to the NRC staff.

2.6.3.2 TS 2.2 Limiting Safety System Setting

TS 2.2 states the following:

> Specification
>
> The limiting safety system setting shall be 975°F (525°C) as measured in an instrumented fuel element (IFE). The IFE shall be located adjacent to the central bundle with the exception of the corner positions.

TS 2.2 specifies the LSSS limit for fuel temperature as measured in the IFEs. The allowed locations are chosen based on calculations performed for the conversion of the NSCR to LEU fuel to ensure that the hottest location is protected not only against exceeding the SL but also against DNB. In the SAR, the licensee provided the results of analyses in support of locating an IFE connected to the fuel temperature safety channel in the periphery and central region of the core. The temperature SL for the 30/20 LEU fuel is 1,150 degrees C (2,100 degrees F) as stated in TS 2.1. The licensee determined, for both the hottest and the coldest thermocouples, an IFE located adjacent to the central bundle, with the exception of the corner positions, would protect the fuel and ensure that the maximum fuel temperature would stay below the temperature SL for reactor power levels less than 1.3 MWt. The analysis indicated that an IFE located in the specified core positions with a scram setting of 525 degrees C (977 degrees F) would limit the maximum steady-state temperature in the hottest element to less than

735 degrees C (1,355 degrees F). The setting of 525 degrees C (977 degrees F) provides a margin of safety of at least 415 degrees C (779 degrees F) for the 30/20 LEU fuel.

The licensee indicates in the SAR, as supplemented, that the LSSS is applicable not only in steady-state operation but also in pulse mode. However, the temperature channel will not limit the peak power generated during a pulse because of the response time of the temperature channel as compared with the width of a pulse. The temperature scram would limit the total amount of energy generated in a pulse by cutting off the tail of the energy transient if the fuel temperature limit was reached. Thus, the fuel temperature scram provides an additional degree of safety in the pulse mode of operation.

The NRC staff finds that TS 2.2 helps ensure that an LSSS is established to protect the fuel cladding integrity and prevent the release of fission products. The TS 2.2 LSSS value is supported by the SAR, as supplemented, and provides a substantial margin of safety. Based on the information provided above, the NRC staff concludes that TS 2.2 is acceptable.

The NRC staff reviewed the licensee's analyses, as discussed above. The NRC staff finds that the SL and LSSS for the NSCR are based on acceptable analytical and experimental investigations and are consistent with those approved by the NRC staff and used at other TRIGA reactors. On this basis, the NRC staff concludes that the LSSS of 525 degrees C (977 degrees F) and the accompanying conditions are sufficient to prevent the fuel temperature from exceeding the SL and, therefore, are acceptable.

2.7 Thermal-Hydraulic Design

2.7.1 Steady-State Operations

The licensee presented the thermal-hydraulic design of the NSCR in the Conversion SAR and the accompanying results of the thermal-hydraulic analysis (fuel temperature, coolant temperature, DNBR, etc.). The Conversion SAR describes the validation methods used for the neutronics calculations that were also employed in validating the thermal-hydraulic calculations for the conversion. The commissioning tests carried out on the PRNC reactor were used to validate the two computer codes used in the thermal-hydraulic analysis for the NSCR: STAT (a GA program developed for calculating the natural convection heat transfer-fluid flow in an array of heated cylinders, GEN-44, July 1989) to determine the natural convection flow through the core; and TAC2D (a GA developed two-dimensional heat transfer computer code, GA-A14032, July 1976) (Ref. 38) to determine the fuel temperatures. Results of the thermal-hydraulic analyses include fuel and coolant temperatures and the DNBR at a thermal power level of 1.0 MWt.

The NSCR fuel elements are cooled by natural convection. A natural circulation flow rate is established, balancing the driving head against the core entrance and exit pressure losses and frictional, acceleration, and hydrostatic head losses in the core flow channels. The STAT code calculates both a convective and a boiling heat transfer coefficient and then uses the larger of the two to determine the clad surface to fluid temperature difference. The critical heat flux (heat flux at which DNB occurs) is determined by two correlations (McAdams and Bernath) (Ref. 2) and the lower of the two is used. The STAT code and its associated methodology have been applied to the licensing of TRIGA reactors for the past 40 years.

Use of the finite-difference code TAC2D to calculate the steady-state fuel temperature, supplements the thermal-hydraulic analysis. The TAC2D code has been benchmarked and used for many TRIGA reactors. The code calculates temperatures in two-dimensional problems, with radial and axial power distributions in the fuel given as input. The fuel element model consists of the central zirconium rod, the fuel annulus, the fuel-to-clad gap, and the stainless steel clad. Input to the code consists of geometric data, thermo-physical property data, and radial and axial power distributions within the fuel element. TAC2D requires a boundary condition given by input from the STAT code, a clad wall surface temperature. A constant gap model is used, with the gap between fuel and cladding assumed to be filled with air. Gap closure is expected to occur with increasing fuel burnup as a relative expansion of the fuel and cladding at normal operating temperatures, which leads to reduced fuel temperatures.

TAC2D analyses of the NSCR LEU fuel element predicted peak fuel temperatures at the IFE locations for three operating power levels: 0.5 MWt, 1.0 MWt, and 1.3 MWt. The TAC2D (calculated) and IFE (measured) peak fuel temperatures were provided in Table 4-18 of the SAR, and reproduced in SER Table 2-2 below. The measured peak fuel temperature values (shown below) were obtained at a power level of 1.3 MWt in order to validate the analytical results. Facility operation at 1.3 MWt was allowed by the TAMU/TEES TSs in effect at the time of the validation testing. The TSs authorized as a result of this renewal will limit the operation of the NSCR to no greater than 1.0 MWt.

The maximum power rod was determined through a hot-rod factor using the DIF3D code by combining the rod power factor (power in a fuel element relative to core averaged rod power) with the axial power factor (axial peak-to-average power ratio within a fuel element). DIF3D is a multi-dimensional diffusion theory code that is used to determine hot channel factors for the core. DIF3D has a strong history of use with TRIGA cores and produces results that have been confirmed with measurements. The hot-rod factor is used in the thermal analysis for determination of the maximum fuel temperature. A radial rod peaking factor is used to calculate maximum fuel temperatures in very rapid transients in the time range when the heat transfer is not yet significant.

Table 2-2 Operating Peak Fuel Temperatures with 30 Degrees C Pool Water

Operating Power Level	0.5 MWt	1.0 MWt	1.3 MWt
Calculated Peak Fuel Temperature	289 degrees C (552 degrees F)	373 degrees C (703 degrees F)	440 degrees C (824 degrees F)
Measured Peak Fuel Temperature	260 degrees C (500 degrees F)	329 degrees C (624 degrees F)	384 degrees C (723 degrees F)

The STAT analysis predicted steady-state operating parameters for the NSCR LEU core for a power level of 1.0 MWt for both the average power and maximum power fuel element. The results indicate that for the average and maximum fuel element, the mass flow rates are 0.077 kilograms (0.17 pound (lb)) per second and 0.089 kilograms (0.19 lb) per second, exit coolant temperatures are 65 degrees C (149 degrees F) and 77 degrees C (170 degrees F) (assuming 30 degrees C (86 degrees F) reactor pool inlet water temperature which is a typical operating value), maximum flow velocities are 14.1 centimeters (5.5 in.) per second (cm/s) and 16.5 cm/s (6.5 in./s), and the maximum clad temperatures are 134 degrees C (273 degrees F) and 136 degrees C (277 degrees F), respectively.

The licensee performed a series of analyses to determine the location and magnitude of the power peaking in the NSCR core. The power peaking calculations provide the basis for selecting the acceptable locations of the IFEs and the relationship between the measured (in the IFE) and actual temperature at the fuel hot spot in the core. The peak fuel temperatures vary from 289 degrees C (552 degrees F) for a power level of 0.5 MWt to 440 degrees C (824 degrees F) for a power level of 1.3 MWt. The average power per fuel element in the NSCR core operating at 1.0 MWt with 90 fuel elements is 11.1 kW per element.

The fuel element immediately adjacent to the transient rod (5D3) produces the greatest steady state power per element (17.4 kW per element) and would be the ideal location for an IFE. However, due to mechanical constraints (at location 5D3), the IFE is located at the next nearest location, 5E4, where the power generated is 15.4 kW per element. The analysis indicated that the IFE located at 5E4 with a scram setting of 525 degrees C (977 degrees F) would limit the maximum steady state temperature in the hottest element to less than 735 degrees C (1,355 degrees F) well below the fuel temperature SL of 1,150 degrees C (2,100 degrees F).

For the NSCR core, the pulsing peak power density does not occur in the fuel element with the maximum steady-state rod power factor (maximum rod vs. core average rod power). The fuel element in location 4D4 generates slightly lower steady-state power (15.3 kW/element), but has a pulsing peak power density during pulsing that is closer to the maximum peak power density of the whole core, even larger than in the fuel element immediately adjacent to the transient rod (5D3). Consequently, an IFE in this location gives the highest measured fuel temperature during pulsing. The locations in TS 2.2 are chosen to help ensure that the fuel element in the hottest location is protected from exceeding the fuel SL specified in TS 2.1.

The thermal-hydraulic analysis described above used a nominal reactor pool inlet water temperature of 30 degrees C (86 degrees F). In its response to an NRC RAI (Ref. 57), the licensee submitted an additional thermal-hydraulic analysis with the reactor pool inlet water temperature at the TS 3.2.2 limit of 60 degrees C (140 degrees F), with the reactor thermal power level at 1.0 MWt and 1.25 MWt. The revised analysis was performed using RELAP5 MOD 3.3 with the Bernath correlation to determine the critical heat flux, instead of the STAT code used in the conversion work. Agreement has been demonstrated between STAT and RELAP5. The licensee's revised thermal-hydraulic analysis also used the RELAP5 computer code to determine the natural convention flow rate, the coolant and cladding axial temperature profiles, and the cladding wall heat flux axial profile. The RELAP5 steady state thermal-hydraulic analysis was done for an average flow channel, an IFE channel, and four flow channels containing the maximum powered fuel rod. The average flow channel has a flow area associated with the 90 fuel rods in the NSCR core. The IFE has the same hydraulic characteristics as the average flow channel except its flow area is for only one fuel rod. The results are presented in updated SAR Section 4.7, Tables 4-19 and 4-20 (Ref. 57), and summarized below:

At 1.0 MWt:

- Maximum Fuel Temperature – 526 degrees C (979 degrees F)
- Maximum Coolant Exit Temperature – 101 degrees C (214 degrees F)

- Average Coolant Exit Temperature – 90.8 degrees C (195 degrees F)
- Minimum DNBR – 2.53 (Bernath)

At 1.25 MWt (TS 3.2.2 reactor power level trip setpoint):
- Maximum Fuel Temperature – 562 degrees C (1044 degrees F)
- Maximum Coolant Exit Temperature – 103 degrees C (217 degrees F)
- Average Coolant Exit Temperature – 95.5 degrees C (204 degrees F)
- Minimum DNBR – 2.01 (Bernath)

The NRC staff reviewed the licensee's analyses in the updated SAR, as described above. NUREG-1537 recommends acceptance criteria for the DNBR to be no less than 2.0. The results of the NSCR analysis demonstrate it will satisfy these criteria. The methodology and computer codes used for this analysis have been compared to actual measurements and consistently produce conservative results. The NRC staff considers the use of RELAP5 with the Bernath correlation to determine the critical heat flux (and thus departure from nucleate boiling) to provide the best estimate analysis of DNBR. Therefore, the NRC staff finds that the analysis performed by the licensee used qualified calculation methods and conservative or justifiable assumptions. The NRC staff concludes that the thermal-hydraulic analysis in the NSCR Conversion SAR, and the SAR, as supplemented, demonstrates that the NSCR LEU core results in acceptable safety margins regarding thermal-hydraulic conditions during steady-state operations.

2.7.2 Pulse Mode Operations

In the SAR, the licensee states that the limiting condition for pulsed operation of the NSCR is the peak fuel temperature. The fuel temperature is limited to a maximum of 830 degrees C (1,526 degrees F) on the basis of early experience with TRIGA fuel, which demonstrated that fuel damage could occur as a result of hydrogen gas accumulation and redistribution in the hydride fuel if the reactor is pulsed after an extended period of operation at 1 MWt or greater. As a result of this physical feature of the fuel, the pulsed maximum reactivity insertion is currently limited to \$1.91.

The licensee's analysis of a TRIGA reactor core in a step input of reactivity has been computed using the GA BLOOST computer code. The code performs combined reactor kinetics-heat transfer calculations using a point kinetics model that analyzes reactor core power and fuel temperature reactivity transients with a variable-temperature fuel heat capacity model. The fuel element is reasonably considered adiabatic for the short duration of the neutronic pulse. The BLOOST model can predict either the core average fuel temperature response, or it can use power peaking factors to model the highest power density locations in the reactor core.

The BLOOST analytical calculations for the NSCR were validated by a series of measurements for pulses with an upper reactivity insertion of \$1.56, to determine the peak fuel element temperature from the measured fuel temperature at the IFE. An analysis of this data indicates that the prompt insertion of reactivity should be kept below \$1.91 to ensure that the maximum fuel temperature does not exceed 830 degrees C (1,526 degrees F). These data compare well with the BLOOST prediction of a maximum allowable reactivity pulse of \$2.10. The administrative reactivity limit of \$1.91 for pulse mode operation provides adequate protection for maintaining NSCR fuel temperature below 830 degrees C (1,526 degrees F) as required by TS 3.1.2 (see Section 2.6.1.3 of this SER).

The NRC staff reviewed the licensee's analyses in the SAR as described above. The NRC staff finds that the analysis performed by the licensee used qualified calculation methods and conservative or justifiable assumptions. The NRC staff concludes that the thermal-hydraulic analysis in the SAR, as supplemented, demonstrates that the NSCR LEU core results in acceptable safety margins regarding thermal-hydraulic conditions during pulse operations.

2.8 Conclusions

Based on the above considerations, the NRC staff finds that the licensee has presented adequate information and analyses to demonstrate its technical ability to configure and operate the NSCR core without undue risk to the health and safety of the public or the environment. The NRC staff's review of the facility has included studying its design and installation, its controls and safety instrumentation, its operating procedures, and its operational limitations as identified in the TSs. The NRC staff finds that the thermal-hydraulic analysis in the NSCR Conversion SAR demonstrates that the LEU fuel results in acceptable safety margins for thermal-hydraulic conditions.

The NRC staff also finds that the licensee's analyses use qualified calculation methods and conservative and justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with measurements obtained from the PRNC and during the startup after completing the HEU fuel to LEU fuel conversion. The NRC staff reviewed the analysis of the pulsed operation of the NSCR and finds that, with pulse sizes up to the administrative limit of $1.91, the maximum core fuel temperature will remain below the limit set by the known mechanical and thermal properties of the fuel. The NRC staff finds that the NSCR TSs regarding the reactor design, reactor core components, reactivity limits, and related surveillance requirements provide reasonable assurance that the reactor will be operated safely in accordance with the TSs. The design features of the reactor are similar to those typical of research reactors of the TRIGA type operating in many countries worldwide. On the basis of its review, the NRC staff concludes that there is reasonable assurance that the licensee is capable of safe operation of the NSCR, as limited by the TSs, for the period of license renewal.

3 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

3.1 Radiation Protection

Activities involving radiation at the NSC are controlled under the radiation protection program that meets the requirements of 10 CFR Part 20 and the guidance in ANSI/ANS-15.11, “Radiation Protection at Research Reactor Facilities.” Specifically, the licensee has used the guidance of ANSI/ANS-15.11-2004 (Ref. 31). The regulations in 10 CFR 20.1101, “Radiation protection programs,” specify, in part, that each licensee shall develop, document, and implement a radiation protection 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 radiation protection program include occupational and general public exposure limits, surveys and monitoring, and personnel dosimetry.

The NRC inspection program routinely reviews radiation protection and radioactive waste management at the NSC. The NRC staff reviewed the licensee’s annual operating reports for the NSC and the NRC inspection reports concerning the radiation protection program. The NRC staff finds that the licensee’s radiation protection 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 radiation protection program at the NSC is acceptable.

3.1.1 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 Chapters 11 and 13 of the SAR, and verification that the hazards were accurately depicted and comprehensively identified.

3.1.1.1 Airborne Radiation Sources

In the SAR, the licensee states that during normal operations, the primary airborne sources of radiation are argon-41 (Ar-41) and N-16. Ar-41 results from irradiation of the argon in air in experimental facilities and the dissolved air in the reactor pool water. The primary producers of Ar-41 are the reactor pool and the thermal column. Other production sources may include beam ports, the pneumatic irradiation system, and irradiation tubes. N-16 is produced when oxygen in the pool water is irradiated by the neutrons from the reactor. The N-16 radionuclide has a half-life of 7.13 seconds.

The dose rate at the top of the reactor pool is mainly from N-16, with a small contribution from Ar-41. A diffuser system directs a flow of water downwards and across the top of the core area, which significantly slows the upward flow of heated water containing the N-16 by breaking up the thermal plume of buoyant hot water, thereby reducing the dose rate at the top of the reactor pool. The measured dose rate with the reactor operating at full power is less than 10 millirem per hour (mrem/hr) with the diffuser and less than 100 mrem/hr with the diffuser turned off. TS 3.5.1 requires that radiation monitoring equipment be in operation at specific locations, including the reactor bridge above the pool, and that the equipment provide information about radiation levels to the reactor operator (see Section 3.1.4 of this SER).

Occupational exposure from N-16 is minimal because access to the reactor bridge above the pool is restricted during reactor operation. Because of the short half-life of N-16, exposure to the public is negligible given the long air transport time from the reactor pool surface to the unrestricted environment.

Ar-41 is induced in the air that flows through the reactor thermal column and other experimental devices and is also emitted from the pool water surface. The core is cooled by natural convection of pool water that causes the heated water to rise to the surface of the pool along with the air dissolved in the water; some of the dissolved air containing Ar-41 escapes into the pool room.

The reactor area has a ventilation system that removes the Ar-41 through the building exhaust system, thus minimizing the dose to workers from the Ar-41. The exhaust stack flow rate is nominally 1.19×10^{14} ml/yr (8000 ft^3/min.). Since the Ar-41 generated in the reactor area is continuously removed by the ventilation system (as required by TS 3.3.1 and TS 3.3.2), public occupants in the area of the laboratory building adjacent to the reactor room are not subjected to any Ar-41 dose. The Ar-41 release rate for the NSCR is limited to 30 curies per year (Ci/yr), as required by TS 3.5.2. The licensee states, in the SAR, that all modes of operation at the NSCR including thermal column operations produce air concentrations and total Ar-41 release less than the TS 3.5.2 limit of 30-Ci/yr. Using the 30-Ci/yr value, the stack flow rate provided above, and assuming the release occurs over an entire year, the licensee calculated the average Ar-41 release concentration is 2.5×10^{-7} microcuries per milliliter (µCi/mL), which is 8.3 percent of the derived air concentration (DAC) limiting value of 3×10^{-6} µCi/mL, established in Table 1 of Appendix B of 10 CFR Part 20. Using the building concentration as the average stack flow concentration provided above, the licensee calculated the occupational dose to an NSC worker staying in the confinement for 2,000 hours per year is 416 millirem (mrem) (Ref. 16). In its RAI response (Ref. 59), the license evaluated the dose to a worker given the assumption that the concentration of Ar-41, based on the 30 Ci/yr release limit, was generated during 2000 hours of reactor operation (which is the basis for the occupational exposure). The resulting dose was 1,840 mrem, which is well below the 5,000-mrem limit established in 10 CFR 20.1201, “Occupational dose limits for adults.”

The licensee states, in Section 11.1.1 of the SAR, that the pool room air and air from experimental facilities are combined and exhausted through a stack to the atmosphere after being monitored for Ar-41 content. Only Ar-41 has been found in analysis of effluent samples. The release point is a building stack 85 ft (26 m) high that is part of the ventilation system. The licensee calculated the concentration of Ar-41 leaving the stack assuming a release rate of 30 Ci/yr and a concentration of 2.5×10^{-7} µCi/ml. Using an atmospheric dilution factor of 5×10^{-3} for the distance to the fence line at 328 ft (100 m) (boundary of the restricted area), the potential dose to a member of the public was calculated to be 12.6 mrem for a continuous Ar-41 exposure over the entire year (Ref. 59). The dose is below the limit of 100 mrem/yr given in 10 CFR 20.1301, “Dose limits for individual members of the public.” In order to ensure that the actual dose is also below the annual the 10 mrem ALARA constraint of 10 CFR 20.1101, the licensee indicated in the response to an RAI (Ref. 59), that the Ar-41 releases were monitored and analyzed on both a monthly and quarterly periodicity, and the Radiation Safety Officer

(RSO) would review the results of any abnormal releases to ensure that the Ar-41 doses remain below the 10 mrem ALARA constraint. Additionally, a review of the Ar-41 releases from the licensee's annual reports shows that the annual release of Ar-41 is well below the 30 Ci limit (the 2013 Annual Report indicated that the total release of Ar-41 was 0.44 Ci).

The NRC staff performed a confirmatory analysis using the TS 3.5.2 release rate limit of 30 Ci/yr. The results indicated that the radiation dose to a member of the public exposed to the Ar-41 stack effluents at the site boundary continuously for the year was 12.5 mrem and thus below the limit of 100 mrem/yr given in 10 CFR 20.1301.

3.1.1.1.1 TS 3.5.2 Argon-41 Discharge Limit

TS 3.5.2 states the following:

> Specification
>
> The total annual discharge of ^{41}Ar into the environment shall not exceed 30 Ci per year.

TS 3.5.2 helps limit the total annual amount of Ar-41 that may be discharged from the NSCR such that the resulting dose remains below the limit in 10 CFR Part 20 at the point of a public receptor. TS 3.5.2 helps ensure that the health and safety of the public are not endangered by the discharge of Ar-41. By meeting the TS 3.5.2 discharge limit, the licensee's analysis indicates that the Ar-41 emission will result in a projected annual radiation dose of 12.6 mrem to a member of the public who is at the site boundary whenever the reactor is operating and is immersed in the plume of Ar-41 released from the facility stack. This value is below the limit of 10 CFR Part 20. The licensee's analysis demonstrates that the airborne sources released from the NSCR during normal operations do not present a significant exposure hazard.

The NRC staff reviewed the licensee's analysis demonstrating the NSCR routine gaseous effluent releases. The NRC staff finds that the production and control of Ar-41 are acceptable. The NRC staff also finds that TS 3.5.2 provides reasonable assurance that, during the continued normal operation of the NSCR, the airborne radioactive releases will not pose a significant risk to public health and safety or the environment. Based on the information provided above, the NRC concludes that TS 3.5.2 is acceptable.

3.1.1.1.2 TS 4.5 Radiation Monitoring Systems and Effluents

TS 4.5, Specification 2 and Specification 4, state the following:

> Specifications
>
> (…)
>
> 2. The level of ^{41}Ar in the effluent gas shall be continuously monitored during operation of the reactor.
>
> (…)

4. The annual discharge of ^{41}Ar shall be calculated for each annual report.

(…)

TS 4.5, Specifications 2 and 4, help ensure that the Ar-41 release is monitored and calculated for each annual operating report. The NRC staff reviewed TS 4.5, Specifications 2 and 4, and finds that TS 4.5, Specification 2 and 4 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The NRC staff also finds that Specifications 2 and 4, provide reasonable assurance that, during the continued normal operation of the NSCR, the airborne radioactive releases of Ar-41 will be appropriately monitored and will not pose a significant risk to public health and safety or the environment. Based on the information provided above, the NRC staff concludes that TS 4.5, Specifications 2 and 4, are acceptable.

3.1.1.2 Liquid Radiation Sources

In the SAR, the licensee states that the pool surface exposure rate results mainly from N-16, which is discussed in Section 3.1.1.1 of this SER. The reactor coolant system is another source of exposure. Because the piping carries pool water that has been circulated through the reactor core, irradiated corrosion products generated during normal operation may be capable of producing a whole-body exposure to personnel. This occurs only at high power levels when reactor pool water flows through the core. Potential exposure to the NSCR staff is minimized by the implementation of the ALARA policy, which is discussed in Section 3.1.3 of this SER.

As required by TS 4.8.1 (see Section 2.4.3 of this SER), the licensee samples primary water for radioactive content weekly during periods of operation and at least quarterly when not operating to help detect potential fission product leakage from the reactor fuel, leakage from sealed sources, or activation of materials in the coolant water.

The licensee also states that another potential low-level radiation source would be small quantities of liquid wastes that are accumulated from operations and stored in three holdup tanks before disposal. Because radiation exposures from these liquid radiation sources at the NSC are small, they do not present a significant hazard to either the NSC staff or the public. The liquid waste from these tanks is discharged to the sanitary sewer system through a filter so that particulate matter is not discharged to the sewer. Samples from the discharged liquid are analyzed for radioactivity before discharge to ensure that releases are within the 10 CFR Part 20 limits.

3.1.1.2.1 TS 3.7 As Low As Reasonable Achievable (ALARA) Radioactive Effluents Released

TS 3.7, Specification 4, states the following:

Specifications

(…)

4. The facility liquid effluents collected in the holdup tanks shall be discharged in accordance with 10 CFR 20.2003 “Disposal by release into sanitary sewerage.” The liquid effluent shall also meet local sanitary sewer discharge requirements.

TS 3.7, Specification 4, helps ensure that the radioactivity contents of potential liquid effluents are monitored before discharge and analyzed for the nature and concentration of radioactive effluents. TS 3.7, Specification 4, also requires that all liquid effluent discharges will meet local sanitary codes.

3.1.1.2.2 TS 4.5 Radiation Monitoring Systems and Effluents

TS 4.5, Specification 5, states the following:

Specifications

(…)

5. Before discharge, the facility liquid effluents shall be analyzed for radioactive content.

TS 4.5, Specification 5, helps ensure that all radioactive liquids discharged to the environment via the sanitary sewer are monitored before release. Discharge limits for the isotopes identified will be in accordance with 10 CFR Part 20 criteria for liquid effluents.

Based on a review of the licensee’s annual operating reports, the NRC staff finds that liquid radioactive sources from the continued normal operation of the NSCR are small, and access to the liquid sources and disposal of the liquid sources are controlled. The NRC staff finds that TS 3.7, Specification 4, provides acceptable monitoring, analysis, and release limits and restrictions such that radioactive effluent releases are within the 10 CFR Part 20 limits. The NRC staff also finds that TS 4.5, Specification 5, provides additional analysis of potential liquid effluents prior to discharge. Based on the information provided above, the NRC staff concludes that TS 3.7, Specification 4, and TS 4.5, Specification 5, are acceptable, and the potential liquid discharges to the environment are properly monitored and analyzed before discharge, and do not represent a hazard to the public or environment.

3.1.1.3 Solid Radiation Sources

In the SAR, Section 11.1.1.3, the licensee indicates that the fission products in the reactor fuel and the reactor core are the primary solid radiation sources at the NSC. The fission products in the reactor fuel constitute the most significant solid radiation source. However, the fission products in the fuel are protected by shielding. The nonfuel sources include sealed sources in the reactor pool, activated reactor components, resins from the primary water demineralizer, and irradiated samples. Because final radioactivity is estimated before experimental irradiations are performed, both shielding and storage duration requirements will be known. The radiation protection program controls solid radiation sources. The reactor contains an antimony-beryllium startup source located in a graphite reflector element location within the reactor core. Section 2.3.6 of this SER discusses the startup source.

Based on its review of the information described above, the NRC staff finds that NSC solid radioactive sources from continued operation are properly controlled, have resulted in no significant personnel exposures, and can be handled without endangering the safety of the NSC staff. The NRC staff concludes that the control of solid radioactive sources at the NSC is acceptable.

3.1.2 Radiation Protection Program

The regulations in 10 CFR 20.1101(a) require that each licensee shall develop, document, and implement a radiation protection program. The ultimate responsibility for all activities at NSC is vested in the Director of the Texas Engineering Experiment Station. For nonreactor-related areas, the responsibility for radiation protection is delegated to the RSO and the Reactor Safety Board (RSB). For radiation protection at the reactor, the responsibility is delegated to the Director of the Nuclear Science Center, with the RSB performing the review function for radiation protection activities. The NSC has a structured radiation protection program, which is implemented by a qualified health physicist who reports to the NSC Director. The RSO provides onsite advice concerning personnel and radiological safety. The RSO also provides technical assistance and review in the area of radiation protection. The operating staff performs the day-to-day radiation protection activities at the facility, under the supervision of the Reactor Supervisor. The organizational chart (see Section 5.6.1 of this SER) indicates that there is coordination and cooperation between the RSB and the NSC Director's Office.

3.1.2.1 TS 6.3 Radiation Safety

TS 6.3 states the following:

> The Radiation Safety Officer shall be responsible for implementing the radiation safety program for the TEES/TAMUS NSC TRIGA Research Reactor. The requirements of the radiation safety program are established in 10 CFR 20. The Program should use the guidelines of the ANSI/ANS-15.11-1993; R2004, "Radiation Protection at Research Reactor Facilities."

In Section 11.1 of the SAR, the licensee indicates that the NSC radiation protection program establishes: exposure limits; procedures and record system for surveys and monitoring; and requirements and responsibilities for personnel dosimetry.

The NSC has a structured radiation protection program with the following elements:

- management commitment and worker responsibility
- qualification of personnel and adequacy of resources
- adequacy of authority for responsible positions
- new staff training and continuing education for all personnel
- radiological design as an integral aspect of facility and experiment design
- radiological planning as an integral aspect of operations planning
- performance reviews of designs and operations
- analysis of personnel exposure records
- periodic assessment and trend analysis of the radiological environment
- periodic assessment and audits of the protection program
- surveillance activities
- protective equipment (supply, quality assurance)
- calibration and quality assurance programs
- training

The radiation protection program is implemented using written standard operating procedures. The RSB periodically reviews the program as required in TS 6.2.3 and TS 6.2.4 (see Section 5.6 of this SER). The NRC inspection program routinely reviews the radiation protection program. The NRC staff reviewed the information provided in the SAR, together with the licensee's annual operating reports and NRC inspection reports, and finds that the licensee demonstrated that adequate measures are in place to minimize radiation exposure to personnel and provide adequate protection against operational releases of radioactivity to the environment.

The licensee's radiation protection procedures, as indicated in the SAR, included testing and calibration of the monitors and detection instrumentation; administrative guidelines for receiving, monitoring, handling, transporting, and testing radioactive materials; decontamination; investigation; training; ALARA measures; and personnel access. The health physics staff is equipped with radiation detection capabilities to determine, control, and document occupational radiation exposures at the NSC. All personnel entering the facility are issued the appropriate personnel monitoring devices and any required protective clothing.

General training topics include storage, transfer, and use of radiation and radioactive material in portions of the restricted area; radioactive waste management and disposal; health protection problems and health risks; precautions and procedures to minimize exposure (ALARA); purposes and functions of protective equipment; applicable regulations and license requirements for the protection of personnel from exposure to radiation and radioactive materials; responsibility of reporting potential regulatory and license violations or unnecessary exposure; appropriate response to warnings in events or unusual occurrences; and radiation exposure reports.

All personnel permitted unescorted access to the NSC vital area receive additional training to include access control rules, emergency procedures, dosimetry requirements, key checkout and return, safety in the reactor and control rooms, communication systems, security door requirements, general checkout procedures when exiting the reactor bay, and emergency equipment location and use.

Experiments and reactor equipment areas are surveyed on a regular basis, and radiological conditions are posted for required areas within the facility.

All facility gaseous and liquid effluents are monitored before release according to TS 3.5.2, TS 3.7, and TS 4.5 to comply with 10 CFR Part 20 limits.

The licensee states in the SAR that it audits health physics and radiation protection program procedures annually. This includes all procedures, personnel radiation doses, radioactive material shipments, radiation surveys, and radioactive effluents released to unrestricted areas. The RSO oversees the maintenance of radiation protection program records, including radiological survey data, personnel exposure reports, training records, inventories of radioactive materials, environmental monitoring results, and waste disposal records. Records are kept for the life of the facility.

The licensee states that an environmental monitoring program has been established between the NSC and the Texas Department of State Health Services. At present, more than 20 environmental monitors are evaluated quarterly. The licensee also states that the program is modified as changes are made to the facility operation or equipment.

The NRC staff reviewed the NSC radiation protection program, as described in the SAR, as supplemented, 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 protect the NSC staff, the environment, and the public from unacceptable radiation exposures. Based on the information discussed above, the NRC staff concludes that the NSC radiation protection program, as described, and TS 6.3 are acceptable.

3.1.3 ALARA Program

Section 11.1.3 of the SAR, describes the ALARA program. To comply with the regulations in 10 CFR 20.1101, the licensee established and implemented a policy that all operations are to be planned and conducted in a manner to keep all exposures ALARA. The program to implement this policy is based on the guidelines of ANSI/ANS-15.11-2004 (Ref. 31). The program is applied through written procedures and guidelines described in the SAR. All proposed experiments and operational procedures at the NSC are reviewed for ways to minimize potential exposure to personnel. The NSC health physics staff participates in experiment planning to minimize both personnel exposure and generation of radioactive waste. Additionally, unanticipated or unusual reactor-related exposures are investigated to develop methods of preventing recurrence. The review of controls for limiting access and personnel exposure at the NSC provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA. The NSC staff performs an annual ALARA review and reports the results to the RSB.

The ALARA program is applied not only to operations within the facility but also, through TS 3.5.2, TS 3.7, and TS 4.5, to radioactive effluent releases from the facility (see Sections 3.1.1.1 and 3.1.1.2 of this SER).

In addition to the TS requirements for gaseous and liquid effluent releases from the facility discussed in Section 3.1.1.1 and Section 3.1.1.2 of this SER, TS 3.7 presents additional requirements for effluent releases as described below.

3.1.3.1 TS 3.7 As Low As Reasonably Achievable (ALARA) Radioactive Effluents Released

TS 3.7, Specifications 1, 2, and 3, state the following:

Specifications

1. In addition to the radiation monitoring specified in Section 5.5, an environmental radiation monitoring program shall be conducted to measure the integrated radiation exposure in and around the environs of the facility on a quarterly basis.

2. The annual radiation exposure (dose) to the public due to reactor operation shall not exceed the limits defined in 10 CFR 20.1301. The facility perimeter shall be monitored to ensure this specification is being met.

3. In the event of a fission product leak from a fuel rod or an airborne radioactive release from a sample being irradiated, as detected by the facility air monitor (FAM), the reactor shall be shut down until the source of the leak is located and eliminated. However, the reactor may continue to be operated on a short-term basis, as needed, to assist in determining the source of the leakage.

(…)

TS 3.7, Specification 1, helps establish a periodic measurement of radiation exposure and an environmental radiation monitoring program to measure the integrated radiation exposure in and around the facility.

TS 3.7, Specification 2, helps ensure that the annual radiation exposure to the public does not exceed the limit in 10 CFR Part 20.

TS 3.7, Specification 3, helps identify potential leaks from fuel elements, sources, or samples and to ensure that the reactor is shut down in the event of a leak of radioactive material. This specification also allows reactor operation on a short-term basis because the operating time is used only to identify the source of the leakage.

The NRC staff reviewed TS 3.7, Specifications 1, 2, and 3, and finds that these specifications help to ensure that the annual radiation exposure to the general public resulting from operation of the reactor are maintained to an ALARA level. These specifications provide the basis for a radiological environmental monitoring program and establish requirements that radiation doses be maintained ALARA. Based on the information provided above, the NRC staff concludes that TS 3.7, Specifications 1, 2, and 3, are acceptable.

3.1.3.2 TS 4.5 Radiation Monitoring Systems and Effluents

In addition to the surveillance requirements for gaseous and liquid effluent releases from the facility discussed in Section 3.1.1.1 and Section 3.1.1.2 of this SER, TS 4.5 presents additional requirements for effluent releases.

TS 4.5, Specification 3 states the following:

> Specifications
>
> (…)
>
> 3. The environmental monitoring program required by TS 3.7 shall measure the integrated radiation exposure on a quarterly basis.
>
> (…)

TS 4.5, Specification 3, helps ensure that the environmental monitoring program measures the integrated radiation exposure quarterly. The licensee submits the results of the environmental monitoring program to the NRC in its annual operating reports.

The NRC staff reviewed these annual operating reports and finds that these reports show that the impact on the environment from the gaseous and liquid releases at the NSC is acceptable. In addition, the NRC inspection program routinely reviews the ALARA program and concludes that the program as implemented meets the requirements of the regulations. TS 3.5, TS 3.7, and TS 4.5 require the facility to operate in a manner that minimizes radioactivity exposure to workers and to the public. The NRC staff finds that the NSC ALARA program, as described in SAR Section 11.1.3, and implemented through TS 3.5, TS 3.7, and TS 4.5, complies with 10 CFR 20.1101 and provides reasonable assurance that radiation exposure will be maintained ALARA for all facility activities. On the basis of the information discussed above, the NRC concludes that the NSCR ALARA program is acceptable.

3.1.4 Radiation Monitoring and Surveying

The regulations in 10 CFR 20.1501(a) state that each licensee shall make, or cause to be made, surveys that have the following characteristics:

> (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 radioactive material; and
> (iii) The potential radiological hazards.

The regulations in 10 CFR 20.1501(c) require 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.

In the SAR, as supplemented, the licensee describes a comprehensive set of portable radiation survey instrumentation that has sufficient range to cover the various types of radiation that may be encountered at the NSC. Area radiation monitors (ARMs) and facility air monitors (FAMs) are used for radiation monitoring at the NSC. ARMs are located at strategic points throughout the facility where radiation levels could exceed normal levels. These ARMs have local and remote audible and visual alarms. The remote alarm capability is in both the control room and the reception room, which is also the emergency support center.

Six FAMs (fours are required by TSs) monitor airborne activity within and leaving the NSC in both gaseous and particulate form. All have local audible and remote alarm capability. The remote alarm capability is in both the control room and the reception room (also the emergency support center). Three of these FAMs will shut down the air handling system when the alarm value is reached for radioactive particulates entering the stack, and radioactive particulates above the reactor core.

The licensee also states in the SAR, that the health physics staff at the NSC regularly perform radiation and contamination surveys. The placement, use, and control of the radiation monitoring and surveying equipment are in accordance with applicable national standards, guidance, and regulations. The equipment selected is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to help ensure compliance with 10 CFR Part 20 requirements and the facility ALARA program under all operating conditions.

3.1.4.1 TS 5.5 Radiation Monitoring System

TS 5.5 states the following:

> <u>Specification</u>
>
> The radiation monitoring equipment listed in Table 5 shall have the following characteristics:

Table 5: NSC Radiation Monitoring Equipment

Radiation Monitoring Channel	Detector Type	Function
Area Radiation Monitor (ARM)	Gamma sensitive detector	Monitor radiation fields in key locations. Alarm and readout in the control room and readout in the emergency support center.
Facility Air Monitor (FAM)-Particulates (FAM Ch. 1, 4)	Beta-Gamma sensitive detector	Monitors concentration of airborne radioactive particulates. Alarm and readout in the control room and readout in the emergency support center.
Facility Air Monitor (FAM)-Gases (FAM Ch. 3, 5)	Gamma sensitive detector	Monitors concentration of radioactive gases. Alarm and readout in the control room and readout in the emergency support center.

An alarm signal from the stack particulate or stack gas (xenon) facility air monitor shall automatically isolate the central exhaust system.

In the SAR, as supplemented, the licensee indicates that the radiation monitoring system is intended to inform operating staff of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings. The ARM alarm setpoints (alert and alarm) are typically set at about three times the background radiation reading (5 mrem). The alert and alarm are typically set at 14 mrem and 16 mrem, respectively. The licensee states that it can adjust the alert and alarm setpoints as necessary to account for changes in the background radiation levels. The license indicates, in RAI response (Ref. 65), that the FAM channel setpoints are established using the effluent concentration limits in 10 CFR Part 20, Appendix B, Table 2. By using this method, the licensee provides an alarm setpoint that will notify personnel to evacuate the area prior to exceeding the radiation limits in 10 CFR Part 20.

The NRC staff reviewed the design characteristics and finds that the radiation monitoring system will provide reasonable assurance that airborne radioactivity will be properly detected to comply with 10 CFR Part 20 requirements. The alert and alarm setpoints will provide an advance warning to the operators that radiation levels have changed and an evaluation and assessment is needed to ensure the safety of the workers and the public. Based on the information provided above, the NRC staff concludes that TS 5.5 is acceptable.

The system used at the NSC for monitoring radiation include radiation monitors, as specified in TS 3.5.1 and TS 3.5.3. As described below, TS 3.5.1 specifies the radiation monitoring channels that must be operable during reactor operation and the information that must be available to the reactor operator during reactor operation to ensure safe operation of the reactor.

3.1.4.2 TS 3.5.1 Radiation Monitoring

TS 3.5.1 states the following:

Specification

Reactor operation, movement of irradiated fuel elements or fuel bundles, conduct of core or control rod work that could cause a change in reactivity of more than one dollar, or handling of radioactive materials with the potential for airborne release shall not be conducted unless the radiation monitoring channels listed in Table 3 are operable, displays and alarms are operable in the control room, and displays are operable in the Emergency Support Center.

Table 3: Radiation Monitoring Channels Required for Operation[3]

Radiation Monitoring Channels	Function	Number
Reactor Bridge ARM	Monitor radiation levels within the reactor bay	1
Stack Particulate Monitor (FAM Ch. 1)	Monitor radiation levels in the exhaust air stack	1
Stack Gas Monitor (FAM Ch. 3)	Monitor radiation levels in the exhaust air stack	1
Building Particulate Monitor (FAM Ch. 4)	Monitor radiation levels within the reactor bay	1
Stack Xenon Monitor(FAM Ch. 5)	Monitor radiation levels in the exhaust air stack	1

[3] When a required channel becomes inoperable, operations may continue only if a portable or fixed gamma-sensitive ion chamber is utilized as a temporary substitute, provided that the substitute can be observed by the reactor operator, can be installed within 1 hour of discovery, and is not used longer than one week. If two of the above monitors are not operating, operations shall cease.

In the SAR, as supplemented, the licensee indicates that the radiation monitors inform operating personnel of any impending or existing danger from radiation so corrective action can be taken. All FAMs will alarm in the control room if the radiation levels detected exceed a preset radiation level. The reactor bridge radiation monitor checks the dose rate above the core and will alert the operator if the N-16 diffuser ceases to operate. The stack particulate monitor will alert the operator for radioactive particles in the exhaust system. The stack gas monitor checks for the level of Ar-41 or other radioactive gases entering the NSC exhaust stack. The building particulate monitor will alert the operator of radioactive particles in the confinement area. Each radiation monitoring channel provides indication and an audible facility air monitoring alarm in the control room and in the reception room. The reactor bridge and stack particulate monitors also shut down the air handlers and provide an emergency shutdown air handling system alarm in the control room.

The NRC staff reviewed the licensee's radiation monitoring equipment, as described in the SAR, as supplemented, and TSs, and finds that the licensee has adequate instruments and equipment for quantitative radiation measurements and TS 3.5.1 requires sufficient monitors to evaluate potential radiation hazards. Routine effluent releases are within regulatory limits, and the discussion in Section 4 of this SER shows that the consequences of accidents are acceptable. Based on the information provided above, the NRC staff concludes that TS 3.5.1 is acceptable.

3.1.4.3 TS 4.5 Radiation Monitoring Systems and Effluents

TS 4.5, Specification 1, states the following:

Specifications

1. The area radiation monitoring system (ARM) and the FAM shall be calibrated annually, shall be channel tested weekly, and shall be channel checked prior to reactor operation.

(…)

The licensee states in the SAR, as supplemented, that the combination of weekly testing, annual calibration, and channel checks before reactor operation of the ARM system and the FAM system is sufficient to identify any changes to the operating characteristics of the monitoring systems and to help ensure proper operation. The NRC staff reviewed the information provided in the SAR, as supplemented, and finds that these surveillance intervals are consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and are sufficient to detect any changes to the operating characteristics of the monitoring systems. Based on the information provided above, the NRC staff concludes that the surveillance requirement in TS 4.5, Specification 1, is acceptable.

Based on its review, the NRC staff concludes that the licensee's equipment for detecting the types and intensities of radiation likely to be encountered within the facility and the surveillance frequencies are appropriate to ensure compliance with 10 CFR 20.1501(a) and (c), and the facility ALARA program.

3.1.5 Radiation Exposure Control and Dosimetry

In the SAR, as supplemented, the licensee indicates that the reactor shielding is based on the combination of pool water and the concrete pool structure. Radiation levels on the reactor bridge directly above the reactor are less than 10 milli-Roentgens per hour (mR/h) with the reactor operating at 1.0 MWt, with the N-16 diffuser in operation, while in the lower research level, they are less than 0.5 mR/h. The ventilation system maintains the reactor room at negative pressure with respect to outside areas and helps to lower the concentration of Ar-41 and N-16 to levels that satisfy the occupational dose limits in 10 CFR 20.1201.

The regulations in 10 CFR 20.1502 require monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the specified limits. In the SAR, as supplemented, the licensee states that it uses optically stimulated luminescent dosimeters to monitor personnel whole-body exposure. The dosimeters are assigned to individuals who have the potential to be exposed to radiation. Thermoluminescent dosimeters are provided for extremity monitoring. Portable equipment is used to perform monthly radiation surveys by the Radiation Safety Office. Personnel protective equipment is used as needed. Facilities and equipment to decontaminate persons are available, if needed. Procedures exist that govern the use of this equipment. The licensee states that it uses survey meters to measure dose rates from radiation fields, and these measured rates are posted where required. These provisions help ensure that external and internal radiation monitoring of all individuals required to be monitored meets the requirements of 10 CFR Part 20 and the goals of the facility ALARA program.

The licensee states that it also maintains personnel exposure records and effluent and environmental monitoring readings for the life of the NSC. The NRC staff reviewed the licensee's annual operating reports from 2005 through 2013, and NRC inspection reports from 2005 through 2014. The review showed that the highest annual whole-body exposure received by a single individual was 1,036 mrem deep-dose equivalents (Ref. 32). The highest annual extremity exposure for that period was 540 mrem shallow-dose equivalents, and the highest skin or other shallow dose was 149 mrem shallow-dose equivalents (Ref. 32). The NRC staff finds that all NSC staff received significantly less than the 10 CFR 20.1201 limits.

The NRC staff reviewed the licensee's exposure control and dosimetry program and finds that personnel exposures at the NSC are controlled through satisfactory radiation protection and ALARA programs. On the basis of the information described above, the NRC staff concludes that the licensee's radiation exposure control and dosimetry programs are acceptable.

3.1.6 Contamination Control

In the SAR, the licensee indicates that it performs contamination surveys on a daily, weekly, and monthly basis, depending on the frequency with which the radioactive material is used or handled. Written procedure controls the handling of any radioactive material within the NSC facility. Workers are trained in working with radioactive material, including how to limit its spread when entering and exiting an area containing radioactive material. The facility surveys have routinely shown no detectable contamination in non-radiological areas of the facility.

The NRC staff reviewed the licensee's annual operating reports and the NRC inspection reports for the past 5 years. This review showed that adequate controls exist to prevent the spread of radiological contamination within the facility. Based on its review of the NSC radiation protection program and on a history of satisfactory contamination control, the NRC staff concludes that adequate controls exist to prevent the spread of contamination within the facility.

3.1.7 Environmental Monitoring

TS 3.7, Specification 1 (see Section 3.1.3 of this SER), requires that an environmental radiation program be conducted to measure the integrated radiation exposure in and around the environments of the facility, and that radioactive effluents released from the facility are in accordance with ALARA criteria. TS 4.5, Specification 3, requires that the program be conducted quarterly. The SAR, Section 11.1.2, states that environmental monitoring is performed in conjunction with the Texas Department of State Health Services. In addition to monitoring all radioactive effluents released from the facility, thermoluminescent dosimeters, evaluated quarterly, are also used for environmental dose monitoring and assessment. Twenty dosimeters are placed on the fence surrounding the NSC.

TS 6.2.5 (see Section 5.6.2.5 of this SER) helps ensure that the Reactor Safety Board (RSB) reviews and audits the environmental monitoring program as part of its review and audit of the ALARA program. The RSB review helps ensure that the environmental monitoring program contains enough locations and sufficient frequency of collection such that the analysis of the data has sufficient sensitivity to help ensure that the overall program complies with 10 CFR 20.1301 and will provide an early indication of any environmental impact caused by the operation of the reactor. TS 6.7.1, Specification 8 (see Section 5.6.7 of this SER), helps ensure that the licensee includes in its annual operating report to the NRC a description and summary of any environmental surveys performed outside the facility.

The NRC staff reviewed the licensee's environmental monitoring program and the results of the program as reported in the licensee's annual operating reports and NRC IRs. The reports indicated that the operation of the NSCR had not adversely affected the environment. The NRC staff finds that the environmental monitoring program can properly assess the day-to-day operation of the facility to minimize the radiological impact on the environment. Based on the information provided above, the NRC staff concludes that the environmental monitoring program is sufficient to assess the radiological impact of the operation of the NSCR on the environment.

3.2 Radioactive Waste Management

The purpose of the radioactive waste management program is to minimize radioactive waste and ensure its proper handling, storage, and disposal. The licensee states that all radioactive waste handling operations are controlled by procedure and overseen by the NSC health physics staff.

3.2.1 Radioactive Waste Management Program

The SAR, Section 11.2, describes the licensee's Radioactive Waste Management Program, including the movement, process and release practices for radioactive waste from radioactive controlled to uncontrolled areas. Low-level, solid radioactive waste (gloves, sample containers, paper towels, etc.) is accumulated in waste containers with a plastic liner. When filled, the liners are monitored for radioactivity, sealed and moved to the waste storage building. Short-lived waste is allowed to decay to disposal as a non-radioactive material waste to the local landfill, and longer-lived radioactive waste is compacted, packaged and shipped to a final disposal facility. Liquid waste, collected from various floor drains, laboratories, air-handling units, and the valve pit sump room cooling equipment, is transferred to one of three storage tanks, which is filtered, and analyzed prior to release to ensure that the activity limits are within 10 CFR 20.2003, as indicated in TS 3.7, Specification 4 (see Section 3.1.1.2 of this SER).

TS 6.7.1, Specification 6, helps ensure that the licensee reports radioactive waste in its annual report to the NRC.

3.2.1.1 TS 6.7.1 Annual Operating Report

TS 6.7.1, Specification 6, states the following:

An annual report covering the operation of the reactor facility during the previous calendar year shall be submitted to the U.S. NRC before March 31 of each year providing the following information:

(…)

6. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured 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 less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient.

    a. Liquid Waste (summarized on a monthly basis)

        i. Radioactivity discharged during the reporting period.

            1. Total radioactivity released (in Curies),

            2. The effluent concentration used and the isotopic composition if greater than 1×10^{-7} μCi/cc for fission and activation products,

            3. Total radioactivity (in Curies), released by nuclide during the reporting period based on representative isotopic analysis, and

            4. Average concentration at point of release (in μCi/cc) during the reporting period.

        ii. Total volume (in gallons) of effluent water (including dilution) during periods of release.

    b. Airborne Waste (summarized on a monthly basis)

        i. Radioactivity discharged during the reporting period (in Curies) for:

            1. ^{41}Ar, and

            2. Particulates with half lives greater than eight days.

    c. Solid Waste

        i. The total amount of solid waste transferred (in cubic feet),

        ii. The total activity involved (in Curies), and

        iii. The dates of shipment and disposition (if shipped off site).

(….)

The NRC staff reviewed the specific annual reporting requirements related to radioactive effluent releases and finds that they are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's annual operating reports summarize the radioactive effluent releases. The NRC staff finds that the reporting requirement and the results shown in the licensee's annual operating reports are acceptable. On the basis of the information provided above, the NRC staff concludes that TS 6.7.1, Specification 6, is acceptable.

The NRC staff reviewed the facility radioactive waste release practices as described in the SAR, and supplemented by RAI responses (Ref. 59), and finds that these practices demonstrate reasonable assurance that radiological releases from the facility will not exceed applicable regulatory limits nor pose unacceptable radiation risk to the environment or the public. 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 information provided above, the NRC staff concludes that the NSC radioactive waste program is acceptable.

3.2.2 Radioactive Waste Packaging and Labeling

In SAR Section 11.2, the licensee indicates that low-level solid radioactive waste from laboratory experiments or disposable protective clothing items are accumulated and stored in authorized containers. Activated equipment and activated irradiation samples are stored in the reactor bay area for reuse or to allow decay to low-level activity limits. When filled, the low-level waste containers are sealed and stored in the radioactive waste storage building until the final disposition is determined. The licensee also states that procedures are in place to monitor the radiation exposure from waste storage areas within the facility and to perform required handling operations, such as packaging and transfer, and the preparation of proper documentation associated with shipment.

The NRC staff reviewed the licensee's radioactive waste packaging and labeling techniques and concludes that the radioactive waste packaging and labeling techniques are acceptable.

3.2.3 Release of Liquid Radioactive Waste

In SAR Section 11.2.2, the licensee indicates that normal operation of the NSCR does not produce significant liquid radioactive waste. However, small quantities of liquid waste are periodically generated by minor leakages and sampling of the reactor pool and the primary coolant cooling equipment. These liquid wastes are collected and stored in three holdup tanks until final disposition is determined. Liquid radioactive waste from the holdup tanks is periodically released to the sanitary system in accordance with the approved discharge permit from the Texas Natural Resource Conservation Commission. Written procedures govern all releases to ensure that they are within the limits stated in 10 CFR Part 20, Appendix B, Table 3.

The NRC staff reviewed the licensee's controls and techniques for release of radioactive liquid waste to the sewer system and concludes that these controls and techniques are acceptable.

3.3 Conclusions

On the basis of its review of the information presented in the SAR, as supplemented, observations of the licensee's operations, review of the licensee's annual operating reports, and the results of the NRC inspection program, the NRC staff concludes the following:

- The NSC radiation protection program complies with the requirements in 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the NSC 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 NSC provides reasonable assurance that radiation doses to the NSC staff, the public, and the environment will be ALARA.

- The results of radiation surveys carried out at the NSC, doses to the persons issued dosimetry, and the results of the environmental monitoring program verify that the radiation protection 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. The licensee sufficiently controls radiation sources.

- Facility design and procedures limit the production of Ar-41 and N-16 and control the potential exposures to the NSC staff. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas provide reasonable assurance that doses to NSC staff and 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 regulations nor pose unacceptable radiation risk to the public and the environment.

The NRC staff reviewed the NSC radiation protection program and radioactive waste management program summary as described in the SAR, as supplemented. The NRC staff finds that the licensee implemented adequate and sufficient measures to minimize radiation exposure to workers and the public. Furthermore, the NRC staff concludes that there is reasonable assurance that the NSC radiation protection and radioactive waste management programs will provide acceptable radiation protection to its workers, the public, and the environment.

4 Accident Analysis

4.1 Accident Analysis

The accident analysis presented in the SAR, as supplemented, for the NSCR helped establish safety limits (SLs) and limiting safety system settings (LSSSs) that are imposed on the NSCR through the TSs. The licensee analyzed potential reactor transients and other hypothetical accidents. The licensee also analyzed the potential effects of natural hazards as well as potential accidents involving the operation of the reactor. The NRC staff reviewed the licensee's analytical assumptions, methods, and results. In addition, the NRC staff performed independent calculations and obtained independent analysis of accidents with other TRIGA reactors (Ref. 33 and Ref. 34) and compared those results with accidents analyzed by the licensee. As will be demonstrated below, none of the potential accidents considered in the SAR, as supplemented, would lead to significant occupational or public exposure.

NUREG-1537 suggests that each licensee consider the applicability of each of the following accident scenarios:

- maximum hypothetical accident (MHA)
- loss-of-coolant accident (LOCA)
- accidental insertion of reactivity
- loss-of-coolant flow
- mishandling or malfunction of fuel
- experiment malfunction
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment

4.1.1 Maximum Hypothetical Accident

The licensee states in Section 13.5.1 of the SAR (Ref. 10), that the failure of the cladding of one highly irradiated fuel element with simultaneous loss of pool water is designated as the MHA for the NSCR. The MHA is the bounding accident involving the release of fission products. The MHA scenario assumes the instantaneous release of the noble gases and halogen fission products directly into the reactor room air without radioactive decay. The licensee provided updated information related to its MHA analysis results in its responses to RAIs (Ref. 17 and Ref. 41). Boundary conditions and assumptions include using a conservative fuel element power during operation of 28 kWt with saturated inventories used for all released isotopes, and no credit for iodine absorption in the reactor pool water, or dilution, or filtration removal by the ventilation system. The bounding fuel element assumption is acceptable because the calculated peak fuel element power at 1.0 MW operation is 17.4 kWt, which is less that the assumed value of 28 kWt.

The licensee performed analyses for the MHA with the reactor room ventilation in the isolation mode (fans shut down, dampers closed) without radioactive decay given the following scenarios:

A. Occupational dose inside reactor room to a member of the operating staff for the assumed 5-minute evacuation period. The dose is due to inhalation of and submersion in radioactive isotopes. If a member of the public is in the reactor room for the 5-minute evacuation period assumed for the reactor staff, their dose will be well within the dose limits in 10 CFR Part 20 for members of the public.

B. Dose to a member of the public inside the laboratory building in the restricted area. The dose is due to leakage of radioactive gases from the reactor room into the laboratory building resulting in an inhalation and submersion dose, and direct radiation dose from the radioactive isotopes suspended in the reactor room. The individual is assumed to be evacuated from the building in 5 minutes.

C. Dose to the nearest member of the public in an unrestricted area at the facility fence line. This case was calculated two ways: 1) from a plume model assuming a ground-level release of all radioactive material from the reactor room (leakage) to the outside environment; and 2) from a confinement model using the direct radiation assuming the entire radioactivity was contained in the reactor building (no leakage). The individual is located at the perimeter fence approximately 100 m (328 feet) from the reactor building. In the first case (plume model), the dose is due to inhalation of and submersion in the radioactive effluent. The exposure time is the duration that the individual is within the radioactive plume. In the second case (confinement model), the dose is from direct radiation through the reactor room walls. The dose calculation assumes that there is no building leakage and no radioactive decay.

D. Dose to an individual at nearest permanently occupied residence. The dose is from a ground-level release of radioactive material from the reactor building under conservative weather condition assumptions. The dose is due to inhalation of and submersion in the radioactive effluent. The exposure time is the duration that the individual is within the radioactive plume (i.e., the time for the plume to travel through the individual). Direct shine in this case will be negligible given the distance from the residence to the facility so that only inhalation and submersion contribute to any measurable dose.

The NRC staff reviewed the methodology provided by the licensee in the SAR and RAI responses (Ref. 59). The NRC staff finds that the analysis performed by the licensee is consistent with the TRIGA MHA guidance in NUREG-1537 and adequate to calculate occupational and public radiation doses.

The MHA is based on a single fuel element cladding failure in air in the reactor room. The licensee states that the analysis is based on the following assumptions:

- The fuel element has a power density of 28 kWt. Fission product inventory is based on reactor operation at 1.0 MWt for 200 days. Hence, the radioactive halogen and noble gas fission products (with the exception of krypton-85) are at saturation. The burnup calculation for krypton was extended to achieve saturation.

- The fuel element has a conservative maximum temperature of 535 degrees C (995 degrees F) corresponding to the 28 kWt rod power density. After averaging over the fuel volume and temperature, a release fraction of 2.6×10^{-5} is applied to arrive at the total gap fission product activity. The fission product release fraction is based on experiments performed by GA on TRIGA fuel (Ref. 35).

- All of the radioisotopes in the fuel element gap are released to the reactor room and instantly mixed uniformly with the air.

- All of the noble gases and halogens from the fuel element gap are available for release to the environment.

- The ventilation system shuts down as designed, and the airborne radioactive isotopes are immediately released to the environment through building leakage. This release scenario assumes a rapid ground level release of all active isotopes, and is based on conservative assumptions compared to an elevated release through the stack.

- All of the reactor room air containing the radioactive gases is available for release to the environment.

- A 5-minute evacuation time was assumed for staff (radiation workers) in the reactor room.

- A 5-minute evacuation was assumed for any non-radiation workers (members of the public) in the restricted areas of the laboratory building. In its response to an RAI (Ref. 59), the licensee also states that the laboratory building dose would be less than the dose in the reactor building confinement, for the 5-minute exposure duration assumed prior to the evacuation. This is because the confinement would contain and therefore lessen some of the radioactive material available to be subsequently released to the laboratory building during the 5-minute evacuation time.

- A Gaussian plume dispersion model with ground-level release is used to compute the doses external to the reactor building. The model assumed the most stable atmospheric class (Pasquill F) with a windspeed of 2 m/s (6.6 ft/s), and neglecting deposition and meandering processes.

- The closest unrestricted location is at the fence line 100 meters (328 ft) from the reactor building, and the nearest permanently occupied residence is about 800 meters (2,624 ft) away.

- The calculation of direct radiation shine from the confinement building assumes no decay of radioactive isotopes and no leakage of radioactive material from the confinement building. The radiation dose rate is only decreased for distance through air with no contribution for the shielding of the confinement building. The dose calculation assumes the person at the fence line is exposed to the dose rate for a year. These assumptions are conservative because all of the radioactive material will leak out of the confinement building in significantly less than a year.

The NRC staff reviewed the licensee's analytically generated radionuclide inventory for the MHA, including the assumptions and boundary conditions. The NRC staff finds that the licensee's radionuclide inventory is accurately represented by the fuel used in the NSCR. The NRC staff considers the use of a 5-minute exposure to occupationally exposed workers in the reactor room and to individuals in the laboratory and other buildings inside the restricted area reasonable because evacuation drills conducted by the licensee for the NSC have demonstrated the ability to evacuate personnel from the facility within the 5-minute timeframe.

During normal operation, the reactor room ventilation system is required to operate as stated in TS 3.3.2. For a high reading of the stack particulate monitor of fission products, the ventilation system automatically shuts down (isolate mode). The reactor operator may also isolate the building using the Air Handler Shutdown button in the control room, which simultaneously closes all ventilation dampers and shuts off the air handlers (similar to the automatic actions). As stated in the SAR, the licensee performed the MHA analyses with the ventilation system in the isolate mode.

The licensee determined the committed effective dose equivalent (CEDE) for inhalation and deep dose equivalent (DDE) from immersion for occupational workers, as well as for members of the public, from the MHA for scenarios A through D (Ref. 17). The NRC staff performed confirmatory calculations using the licensee's source term for radioactive isotopic concentration and found acceptable agreements.

Table 4-1 presents the licensee's results for the DDE, CEDE, and total effective dose equivalent (TEDE) as well as the NRC staff's confirmatory TEDE calculations, as follows:

Table 4-1 MHA DDE, CEDE and TEDE Dose Values

Scenario	Description		NSCR Staff Calculations			NRC Staff Confirmatory Calculations	10CFR20 Limit (mrem)
			DDE (mrem)	CEDE (mrem)	TEDE (mrem)	TEDE (mrem)	
A	Maximum dose to operating personnel (5 min. exposure) **Occupational Dose**		6.2	17.8	24	25.9	5000
B	Maximum dose to personnel inside the laboratory building (5 min. exposure)* **Public Dose**		Less than 6.2	Less than 17.8	Less than 24	Less than 25.9	100
C	Maximum dose to nearest person in unrestricted area at fence line **Public Dose**	Plume Model	0.4	0.089	0.489	0.47	100
		Confinement Model	11	n/a	11	n/a	
D	Maximum dose to persons in nearest occupied residence **Public Dose**		0.1	0.025	0.125	0.13	100

*Because of retention of radioactive material in the confinement building and a similar evacuation time, dose is no greater than the Scenario A dose.

The 10 CFR Part 20 dose limits are:

- Occupational 5 rem (5,000 mrem) TEDE
- Members of the Public: 0.1 rem (100 mrem) TEDE

The licensee states, in the SAR, that the calculated exposures are below the occupational and public exposure limits given in 10 CFR 20.1201, "Occupational dose limits for adults," and 10 CFR 20.1301, "Dose limits for individual members of the public."

The NRC staff reviewed the analysis performed by the licensee regarding the consequences of the MHA. The NRC staff finds that the licensee's analyses used qualified methodologies with an acceptable radiation source term and incorporated conservative or justifiable assumptions on other boundary conditions. In conducting the MHA evaluation, the NRC staff used the dose limits in 10 CFR Part 20. The NRC staff finds that the calculated occupational and public radiation exposures for the MHA are within the limits applied for licensing of the NSCR and also within the limits provided in 10 CFR Part 20. Based on the information provided above, the NRC staff concludes that the potential MHA calculated doses are within acceptable limits.

4.1.2 Loss-of-Coolant Accident

The licensee provides the results of the LOCA in the SAR Section 13.5.1 (Ref. 10), as supplemented, which considered two scenarios for a significant LOCA: (1) draining of the reactor pool water because of a failure of a component such as a coolant line or a failure in a pool penetration such as a beam tube; and (2) a partial draining of the reactor pool with portions of the fuel element in water. Pumping or draining all of the coolant water in the pool from installed components is not possible because of the location of the piping and siphon breaks in the piping. Pumping using a portable pump is administratively controlled.

The licensee describes the emergency core cooling system in the SAR Section 6.2.3. The reactor pool level is alarmed (TS 3.8.2, Specification 3), and reactor operators would be notified of a low-level condition and would respond. Two raw water lines are installed adjacent to the reactor pool which can supply a combined 400 gallons (1,514 l) per minute of makeup water.

A LOCA would result in the water draining into liquid radioactive waste storage tanks with a capacity of 37,000 gallons (140,060 l) which would be able to accept a large volume of the reactor pool leakage. The collected water would be analyzed before disposal. The radioactivity level of the water is low and expected to be within the limits found in Appendix B of 10 CFR Part 20 for release to the sewer.

The licensee states that if the reactor pool was completely drained of coolant water, the fuel elements would be cooled by air circulation. Calculations and experiments performed by GA have shown that air circulation would be adequate to prevent fuel damage by removing the decay heat of the fuel (Ref. 34). GA studies indicated that, given continuous 1.0 MWt operation, the TRIGA fuel element power would be 19.7 kWt, and the corresponding peak LOCA fuel surface temperature would reach 585 degrees C (1,085 degrees F). This is well below the temperature limit of 950 degrees C (1,742 degrees F) for fuel with a cladding temperature of greater than 500 degrees C (932 degrees F), which ensures that the cladding integrity is maintained. The 500 degrees C (932 degrees F) is used for the cladding temperature due to the air cooling scenario of the LOCA. The maximum peak NSCR fuel element power is 17.4 kWt, and thus, bounded by the GA analysis (Ref. 15).

For a partial LOCA, GA performed experiments, which concluded that the temperature rise for a partial loss of coolant is less severe than for a complete loss of coolant and that fuel damage would not result (Ref. 34). For a partial LOCA, with a drain time of approximately 15 minutes, the peak fuel temperature would reach 578 degrees C (1,072 degrees F) (Ref. 15).

The licensee also states that calculations of the maximum fuel temperature during the LOCA versus fuel element power density during operation were performed assuming an infinite operating period. The calculations show that with fuel element power density no higher than 21 kWt per element, air cooling of the fuel elements is sufficient to prevent damage to fuel element cladding with essentially no delay between reactor scram and loss of coolant (Ref. 15). For higher power densities up to 23 kWt per element, a 15-minute delay is required between reactor scram and complete loss of coolant before air cooling of the fuel elements is sufficient to prevent damage to fuel element cladding. The licensee states that for reactor operations at 1.0 MWt, the maximum power density is 17.6 kWt per element. Because this is less than 21 kWt per element, a delay between reactor scram and complete loss of coolant is not required to protect the integrity of the fuel element cladding.

A more likely scenario than complete draining of the reactor pool is that the core is only partially uncovered. GA performed experiments that concluded that the temperature rise for a partial loss of coolant is less severe than for a complete loss of coolant (Ref. 34). These conclusions are supported by studies performed for TRIGA fuels which show that, in general, as long as the operating power is less than 1.5 MWt, the fuel cladding should not breach during a partial LOCA (Ref. 33 and Ref. 34).

The licensee states that another consequence of a LOCA event is the gamma-ray dose from the exposed core. The licensee calculated doses from a completely uncovered reactor core. Control room personnel would be exposed to principally scattered radiation from the uncovered core. The exposure rate in the control room was calculated to be 4.27 R/hr immediately after the core was uncovered. A 5-minute evacuation time would lead to a dose of 356 mrem, which is well below the 10 CFR 20.1201 limit. The licensee's calculations also showed that the maximum dose rate at the fence line to members of the general public from radiation arising from a LOCA is less than 1 mrem/hr. However, to ensure that the dose to the public would not exceed the limits of 10 CFR Part 20, the licensee indicates that emergency measures could be enacted as part of the emergency plan, following a LOCA, that would limit members of the public from any unwarranted radiation exposure by extending the exclusion boundary around the facility beyond the fence. Additionally, field measurements of radiation levels would be performed to verify that any resultant radiation levels would be acceptable to members of the public, or additional changes to the exclusion boundary would be implemented. The calculations in the SAR, as supplemented, showed that the reactor bridge above the core will be inaccessible for a long time without some refill of pool water. However, the remainder of the facility, where only scattered radiation occurs, will be accessible for recovery operations following an assessment of radiation levels.

The NRC staff reviewed the information provided in the SAR, as supplemented, with respect to a potential leak from the reactor pool and finds that the licensee has the ability to monitor the pool water level, and respond to low-level alarms. If a catastrophic leak occurred, the licensee's analysis indicates that fuel element integrity would remain intact, and radiation doses to the worker's and members of the public would remain within the limits of 10 CFR Part 20. Based on the information provided above, the NRC staff concludes that a potential total loss of primary coolant is extremely unlikely and, if it occurs, the impact on the health and safety of the public, the environment, or the NSC staff will be minimal.

The NRC staff reviewed the licensee's LOCA analysis for the core documented in the NRC SER for the LEU conversion (Ref. 7) and updated in the licensee's response to RAIs (Ref. 14 and

Ref. 15). The NRC staff finds that the LOCA does not result in damage to the reactor fuel because the maximum fuel temperature is below the air-cooled limit of 950 degrees C (1,742 degrees F). In addition, as discussed above, doses within the NSC facility are within the limits that would allow timely recovery operations to proceed. On the basis of its review of the licensee's analysis and experimental results as discussed above, the NRC staff concludes that neither full, nor partial LOCAs would damage the reactor fuel. Additionally, the NRC staff concludes that the doses resulting from the LOCA analysis are acceptable, and the LOCA does not pose significant risk to the health and safety of the public, the environment, or NSC staff.

4.1.3 Reactivity Insertion Event

In the SAR, Section 13.5.3 (Ref. 10), the licensee provides a summary of its analysis of two events that could result in a sudden insertion of reactivity in the LEU core:

(1) the rapid insertion of a large amount of reactivity into the reactor operating at full power by the ejection of the inserted transient rod worth of $2.95, and

(2) the insertion of a large amount of reactivity into the reactor operating at full power by the unplanned removal of a secured experiment whose reactivity worth is $2.00.

The licensee states that based on the analysis of pulsed operation, the maximum reactivity insertion of the transient rod is mechanically limited to $2.95 in order to limit peak fuel temperatures to less than the TS SL of 1,150 degrees C (2,100 degrees F). The licensee also states that accidental pulsing from full power is an unlikely event because there are existing design features and procedural steps that prevent it from happening, such as the following:

- An interlock is in place to prevent the initiation of pulsing from an initial reactor power greater than 1 kWt

- An interlock is in place to prevent application of air to the transient rod unless the cylinder is fully inserted to prevent pulsing the reactor in the steady-state mode.

- Procedures require that during a startup to steady-state power, the transient rod is raised to fully withdrawn before other rod movement and remains out during steady-state operation.

Despite the design and procedural safeguards, the licensee postulated that the transient rod is ejected during full-power operation at 1.0 MWt. The reactor scrams on high power with all five control rods fully inserting into the core in less than 1.2 seconds (TS 3.2.3). The licensee calculated that the maximum fuel temperature in the hottest fuel element at beginning of core life will be 864 degrees C (1,587 degrees F) and the TS SL of 1,150 degrees C (2,100 degrees F) would not be reached. At the end of life, the licensee calculated that the maximum fuel temperature in the hottest fuel element will be 888 degrees C (1,630 degrees F), also below the TS SL of 1,150 degrees C (2,100 degrees F).

The accidental removal of a secured, $2.00 worth experiment requires a failure to follow written procedures as the reactor operator unfastens the experiment and pulls it rapidly out of the core. The licensee states that this accident is less severe than the accidental pulsing accident described above primarily because the removal of the experiment (and the introduction of the reactivity) occurs over a much longer period than the accidental pulsing of the transient rod.

The much slower reactivity addition time will result in activating the 125-percent power scrams at a lower portion of the reactivity insertion curve, resulting in lower peak power and lower peak fuel temperatures than those for the accidental pulsing of the transient rod.

The licensee states that both of these scenarios are extremely unlikely because they require either the failure of an operator to follow written procedures together with the failure of an interlock, or deliberate violation of written procedures with rapid removal of an experiment. Even in these unlikely scenarios with extremely conservative assumptions, the analytical results indicate that the TS SL of 1,150 degrees C (2,100 degrees F) would not be exceeded.

The NRC staff reviewed the licensee's evaluation of a postulated reactivity insertion event, as described in the SAR, and finds that the licensee's analysis is comprehensive, and the results demonstrate that the maximum fuel temperature reached during the positive reactivity addition event would remain well below the TS SL. The NRC staff finds that the licensee's two postulated scenarios that could result in a potential positive reactivity insertion event to be realistic examples for a pulsing TRIGA research reactor. The NRC staff also finds that the licensee's stated design features and administrative controls would render the postulated positive reactivity insertion event to be unlikely; are consistent with the controls and design features found at other pulsing TRIGA reactors; and, can be maintained, tested and/or inspected by the licensee to help ensure operability and TS compliance. Based on the information described above, the NRC staff concludes that the licensee has sufficient design features and administrative restrictions in place to make accidental pulsing or removal of secured experiment during reactor operation to be unlikely, and that the TS SL would not be exceeded if either event did happen.

4.1.4 Loss-of-Coolant Flow

In the SAR, the licensee provides a description of the NSCR core. The reactor core is cooled by natural convection of pool water between the fuel elements, which transfers the heat produced by the reactor to the water in the reactor pool. The pool water is cooled by a primary cooling system which circulates pool water through a heat exchanger with a secondary cooling system where heat is rejected to the environment in a cooling tower. In SAR Section 4.2.1, the licensee provides a description of the coolant flow, by natural circulation, around the fuel elements, through 2 in. (5 cm) diameter holes provided in the fuel grid plate, and by additional 0.5 in. (1.3 cm) diameter holes located at the corner of each of the four fuel element bundles. In the SAR, Section 4.2.5, the licensee describes the core support structure which details that the west side of the suspension frame is open to the pool to allow an unrestricted flow of cooling water to the fuel. In the event of a possible blockage of a coolant channel created by a foreign object lodged in the grid plate, the additional cooling paths in the grid plate would allow continued cooling of the fuel elements. In addition to the multiple flow paths provided by the reactor design, the driving force provided by natural circulation is limited and would minimize the potential for any foreign object to be transported to the grid plate and block cooling flow.

In Section 5.3 of the SAR, the licensee states that a loss of either primary or secondary cooling would provide an alarm indicator in the control room, thereby alerting the reactor operator, and resulting in a reactor shut down. The reactor would continue to be safely cooled by natural circulation, while the pool water temperature would slowly increase at the decay heat rate. The reactor pool contains a significant inventory of water (106,000 gallons (401,254 l)). Reactor operators would correct any primary coolant pool lost by evaporation by adding makeup water.

The NRC staff reviewed the licensee's description associated with a potential reactor core loss-of-coolant flow event, as described in the SAR, and finds that the reactor design provides multiple flow paths for coolant, which would minimize the potential for a cooling flow blockage to occur that could affect fuel cooling. The NRC staff also finds that the licensee's primary and secondary cooling systems have alarms to alert the operators to a loss of coolant flow condition in those systems so that the operators can take proper response actions. The NRC staff finds that a failure of the primary or secondary cooling systems has no impact on the ability to cool the reactor core because of the large inventory of primary coolant and the ability to add coolant to the pool if needed. Based on the information described above, the NRC staff concludes that, should a loss-of-coolant flow condition occur, the reactor core would retain adequate cooling, and a loss-of-flow condition poses no adverse risk to the health and safety of the public or NSC staff.

4.1.5 Mishandling or Malfunction of Fuel

The SAR Section 13.5.5 (Ref. 10), describes a postulated accident scenario that states that if fuel elements were dropped while submerged, or if the fuel element suffered damage because of a manufacturing defect or corrosion, the MHA would bound the consequences of the incident. Moreover, any fission product release would be significantly mitigated beyond the assumptions of the MHA analysis as a result of scrubbing by the pool water overlying the fuel element. There would be no significant effect on the building occupants or members of the public.

The NRC staff reviewed the information provided in the SAR, as supplemented, and finds that the licensee has established procedures for handling fuel and has no recorded incidents of mishandling its fuel elements and no incidents of cladding failure. TS 3.1.5 and TS 4.1.5 require surveillance of fuel elements by visual inspection and annual measurement to verify the continuing integrity of the fuel element cladding (see Section 2.3.2 of this SER). Based on the information provided above, the NRC staff concludes that any potential fission product release resulting from mishandling or malfunction of the fuel would be less than computed for the MHA analysis.

4.1.6 Experiment Malfunction

The SAR Section 13.5.6 (Ref. 10), provides a description of the licensee's controls for the prevention of the occurrence of a postulated experimental malfunction. TS 3.6 and TS 4.6 place limits on experiments installed in the reactor and associated experimental facilities (see Section 5.3.6 and Section 5.4.6 of this SER). The objectives of these limits are to help prevent damage to the reactor and to limit any potential releases of radioactive materials and resulting exposures to personnel in the event of an experimental failure. Limits are placed on reactivity worths, mass of explosive materials, and other experiment materials to limit accidental reactivity insertions, damage to reactor components, and release of radioactivity.

In addition, TS 6.5 requires the Reactor Safety Board to review and approve all new or previous experiments with substantial modification. Included in this review is the determination if these new or modified experiments can be done without NRC approval under the requirements of the regulations in 10 CFR 50.59. The limits on experiments are also analyzed and approved to help ensure that releases are within 10 CFR Part 20 limits.

The NRC staff reviewed the design of the NSCR's experimental irradiation facilities and found them acceptable as part of the original evaluation for operation of the facility. Additionally, the experimental irradiation facilities were reviewed by the NRC staff and found acceptable as part of the license renewal review, completed on March 30, 1983. A detailed description of this review can be found in NUREG-0947 (Ref. 71).

The NRC staff reviewed the information provided in the SAR and TSs, and finds that the licensee's limitations, controls, and procedures for experiments are in place, and are adequate to help to minimize the potential occurrence of an accidental experiment malfunction. The design of experimental irradiation facilities has also been reviewed and found acceptable. If an experiment were to malfunction, the TS controls, which limit the reactivity worths, mass of explosive materials, and other experiment materials, would limit the accidental reactivity insertions, damage to reactor components, and release of radioactivity. Based on the information provided above, the NRC staff concludes that the licensee has proper controls established to minimize the potential occurrence of an accidental experiment malfunction, and to help ensure that the radiation dose consequences would not be more severe than the limits provided in 10 CFR Part 20.

4.1.7 Loss of Normal Electrical Power

The licensee states in Section 13.5.7 of the SAR, that the electrical power system is not necessary to safely shut down the reactor, or to maintain the reactor in a safe shutdown condition, and is not required to ensure public health and safety. In the event of a loss of electrical power, all control rods would insert into the core automatically by gravity because of the loss of power to the control rod drive electromagnets and, for the transient rod, to the three-way solenoid valve that holds it in position. The NSCR operators would be able to confirm control rod insertion through direct visual observation of the position of the control rods in the core. Upon loss of electrical power, the primary and secondary coolant pumps would stop. Reactor decay heat would be dissipated through natural circulation of the primary coolant through the core into the pool. Battery powered instrumentation is available to perform radiological surveillances.

The NRC staff reviewed the information provided in the SAR, as supplemented, and finds that, upon a loss of electrical power, the reactor will shut down and there is sufficient coolant in the reactor pool to absorb the decay heat from the reactor without the need for the primary or secondary cooling system operation during the reactor shutdown. Based on the information provided above, the NRC staff concludes that loss of normal electrical power poses little risk to the health and safety of the public or to NSC staff.

4.1.8 External Events

The licensee states in Section 13.5.8 of the SAR, that the likelihood of external events such as hurricanes, floods, and tornadoes is considered extremely low. Any such event, however unlikely, would lead to a reactor shut down. The licensee also states that the structural integrity of the reactor building and the reactor pool is unlikely to be affected by hurricanes or tornadoes. Earthquakes are extremely unlikely and would lead the operators to shut down the reactor. Calculations show that the fuel can be cooled in air as long as the reactor is shut down (see SER Section 4.1.2). The license also states that the potential consequences of any external events would be bounded by the MHA analysis.

The licensee analyzed the potential impact of an aircraft hitting the reactor building given the proximity of the Easterwood Airport. The NSCR is located 450 m (1500 ft) west of the north-south runway of Easterwood Airport. The analysis indicated that the probability of an airplane collision is low due to the trajectory of the runway. Additionally, the potential consequences of such an event are also low because of the design and construction of the confinement building and the location of the reactor core below grade elevation. The NSCR is located within a steel and concrete confinement building, below ground level, and is protected by thick stainless steel-lined concrete pool walls. As such, an aircraft collision would be unlikely to breach the thick, reinforced concrete pool walls at the core level. Because of the design of the confinement building and pool walls, the probability of an airplane damaging the pool and the reactor is low.

The NRC staff reviewed the licensees' analysis of an accident initiated by an external event, and finds that the probability of this type of potential accident to be extremely low. The NRC staff also finds that the fuel is protected by the facility design, structural barriers or its location below grade, from any accident initiated by an external event. The NRC staff also finds that the potential for an accident resulting from an aircraft collision to be unlikely to affect the fuel due to the thick, reinforced steel and concrete confinement building and pool walls, and location of the reactor below ground. Based on the information provided above, the NRC staff concludes that members of the public are not subject to undue radiological risk as a result of an external event, and that external events do not pose a risk to the health and safety of the public and to the NSC staff that would exceed the results of the MHA.

4.1.9 Mishandling or Malfunction of Equipment

The licensee provides in its response to an RAI (Ref. 59), six categories of accident initiators associated with the mishandling or malfunction of equipment, and provided an analysis for each. The areas involved:

- Operator error at the controls,
- Other operator errors,
- Malfunction or loss of safety-related instruments or controls, such as amplifiers or power supplies,
- Electrical fault in control/safety rod systems,
- Malfunction of confinement or containment system, and,
- Rapid leak of contaminated liquid, such as waste or primary coolant.

For each category, the licensee states that there are processes, procedures, or designs in place to help to ensure that any potential consequence is minimal and in no case exceeds the consequence of the MHA. Some of these potential accidents may occur as a result of human errors which could constitute a violation of the TSs, and which would be reported as required.

The NRC staff reviewed the information provided in the RAI response, and finds that the physical limitations of the NSCR design are such that mishandling or malfunction of equipment would lead to consequences that are bounded by the MHA. The NRC staff concludes, therefore, that the consequences of mishandling or malfunction of equipment pose negligible risk to the health and safety of the public or to the NSC staff.

4.2 Conclusions

The NRC staff reviewed the licensee's analyses of potential accidents at the reactor facility. The NRC staff concludes that the licensee has postulated and analyzed sufficient accident-initiating events and scenarios. On the basis of its review of the information provided in the licensee's SAR, as supplemented, the NRC staff concludes the following:

- The licensee considered the expected consequences of a sufficiently broad spectrum of postulated credible accidents and an MHA, emphasizing those that could lead to a loss of integrity of fuel element clad and a release of fission products.

- The licensee analyzed the most significant credible accidents and the MHA and determined that, under conservative assumptions, the most significant credible accidents and the MHA will not result in occupational radiation exposure of the NSC staff or radiation exposure to a member of the public in excess of the applicable 10 CFR Part 20 limits.

- The licensee has employed appropriate methods in performing the accident and consequence analysis.

- For accidents involving insertions of excess reactivity, the licensee has demonstrated that a pulse reactivity limit of $2.95 will result in peak fuel temperatures below the TS SL of 1,150 degrees C (2,100 degrees F). The licensee has established an administrative limit for pulse reactivity insertions of $1.91.

- The review of the calculations, including assumptions, demonstrated that a LOCA would not result in unacceptable fuel element temperatures. The reactor can be safely cooled with all fuel elements in an air environment. Doses to individual evacuating the reactor room and at the site boundary are calculated to be below the 10 CFR Part 20 limits.

- External events that would lead to fuel disruption are unlikely.

- The accident analysis confirms the acceptability of the licensed power of 1.0 MWt, including the response to anticipated transients and accidents.

- The accident analysis confirms the acceptability of the assumptions stated in the individual analyses provided in the SAR, as supplemented.

The NRC staff reviewed the radiation source term and MHA calculations for the NSCR. The NRC staff finds the calculations, including the assumptions, demonstrated that the source term assumed and other boundary conditions used in the analysis are acceptable. The radiological consequences to the public and occupational workers at the NSCR are in conformance with the requirements in 10 CFR Part 20. The licensee reviewed the postulated accident scenarios provided in NUREG-1537 and did not identify any other accidents with fission product release consequences not bounded by the MHA. The NSCR design features and administrative restrictions found in the TSs help to prevent the initiation of accidents and mitigate associated consequences. Therefore, on the basis of its review, the NRC staff concludes that there is reasonable assurance that no credible accident would cause significant radiological risk and the continued operation of the NSCR poses no undue risk to the NSC staff, the public or the environment.

5 TECHNICAL SPECIFICATIONS

In this section of the SER, the NRC staff provides its evaluation of the licensee's proposed TSs. The NSCR TSs define specific features, characteristics, and conditions governing the safe operation of the NSCR facility. TSs are explicitly included in the renewal license as Appendix A. The NRC staff reviewed the format and content of the TSs for consistency with the guidance in NUREG-1537, Part 1, Chapter 14, and Appendix 14.1, and ANSI/ANS-15.1-2007. The NRC staff specifically evaluated the content of the proposed TSs to determine if it meets the requirements in 10 CFR 50.36, "Technical Specifications." The NRC staff also relied on the references provided in NUREG-1537 and the ISG (Ref. 22) to perform this review.

5.1 Definitions

5.1.1 TS 1.3 Definitions

The following definitions are listed to provide uniform interpretation of terms and phrases used in the TSs. The licensee has proposed TS definitions to be consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

ALARA

The ALARA program (As Low as Reasonably Achievable) is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environs as low as reasonably achievable.

Audit

An audit is a quantitative examination of records, procedures, or other documents after implementation from which appropriate recommendations are made.

Channel

A channel is the combination of sensors, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a parameter.

Channel Test

A channel test is the introduction of a signal into the channel to verify that it is operable.

Channel Calibration

A channel calibration is an adjustment of the channel such that its output corresponds, with acceptable accuracy, to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.

Channel Check

A channel check is 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 variable.

Confinement

Confinement is an 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

A control rod is a device fabricated from neutron-absorbing material or fuel, or both, that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

Regulating Control Rod

The regulating rod is a low-worth control rod used primarily to maintain an intended power level that need not have scram capability. Its axial position may be varied manually or by the servo controller.

Shim Safety Control Rod

A shim safety rod is a control rod having an electric motor drive and scram capabilities. It shall have a fueled follower section.

Transient Control Rod

The transient rod is a pneumatically driven control rod with scram capabilities that is capable of providing rapid reactivity insertion to produce a pulse.

Core Configuration

The core configuration includes the number, type, or arrangement of fuel elements, reflector elements, and regulating/shim-safety/transient rods occupying the core grid.

Core Lattice Position

The core lattice position is that region in the core (approximately 3" x 3") over a grid-plug hole. A fuel bundle, an experiment, or a reflector element may occupy the position.

Excess Reactivity

Excess reactivity is that amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical (k_{eff} = 1) at reference core conditions.

Experiment

An operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics, or which is intended for irradiation within the pool, or in a beam port or irradiation facility. Hardware rigidly secured to a core or shield structure so as to be part of its design to carry out experiments is not normally considered an experiment.

Secured Experiment

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

Unsecured Experiment

An unsecured experiment is any experiment or component of an experiment that does not meet the definition of a secured experiment.

Movable Experiment

A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

Experimental Facilities

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems, and in-pool irradiation facilities.

Experiment Safety Systems

Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information for operator intervention.

Fuel Bundle

A fuel bundle is a cluster of two, three, or four fuel elements and/or non-fueled elements secured in a square array by a top handle and a bottom grid plate adapter. Non-fueled elements shall be fabricated from stainless steel, aluminum, boron, or graphite materials.

Fuel Element

A fuel element is a single TRIGA fuel rod of LEU 30/20 type.

Instrumented Fuel Element (IFE)

An instrumented fuel element is a special fuel element in which one or more thermocouples are embedded for the purpose of measuring the fuel temperatures during operation.

License

The written authorization, by the U.S. NRC, for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

Licensee

A licensee is an individual or organization holding a license.

LEU Core

A LEU core is an arrangement of TRIGA-LEU fuel in a reactor grid plate.

Limiting Safety System Setting (LSSS)

The limiting safety system setting is the fuel element temperature, which if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from being exceeded.

Measured Value

A measured value is the value of a parameter as it appears on the output of a measuring channel.

Operable

Operable means a component or system is capable of performing its intended function.

Operating

Operating means a component or system is performing its required function.

Operational Core—Steady State

A steady-state operational core shall be an LEU core which meets the requirements of the Technical Specifications.

Operational Core—Pulse

A pulse operational core is a steady-state operational core for which the maximum allowable pulse reactivity insertion has been determined.

Pool Water Reference Operating Level

The pool water reference operating level is 10 inches below the top of the pool wall. This level is designed to prevent pool water from rising above the top of the liner.

Protective Action

Protective action is 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.

Pulse Mode

Pulse mode operation shall mean any operation of the reactor with the mode selector switch in the pulse position.

Reactivity Worth of an Experiment

The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

Reactor Console Secured

The reactor console is secured whenever all control rods have been verified to be fully inserted and the console key has been removed from the console.

Reactor Operating

The reactor is operating whenever it is not secured or shutdown.

Reactor Operator

A Reactor Operator is an individual who is licensed to manipulate the controls of a reactor.

Reactor Safety Systems

Reactor safety systems are 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 secured when:

Either

(1) 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;

Or

(2) All of the following conditions exist:

   (a) All control rods are fully inserted;

   (b) The console key switch is in the “off” position and the key is removed from the console lock;

   (c) The reactor is shutdown;

   (d) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless the control rod drives are physically decoupled from the control rods;

   (e) No experiments are moved or serviced that have, on movement, a reactivity worth exceeding $1.00.

Reactor Shutdown

The reactor is shutdown if it is subcritical by at least $1.00 in the reference core condition with the reactivity worth of all installed experiments included.

Reference Core Condition

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is less than $0.01.

Reportable Occurrence

Any of the following events is a reportable occurrence:

(1) Operation with actual safety system settings for required systems less conservative than the LSSS specified in the Technical Specifications;

(2) Operation in violation of a Limiting Condition of Operation listed in Section 3 unless prompt remedial action is taken as permitted in Section 3;

(3) Operation with a required reactor or experiment safety system component in an inoperative or failed condition which renders or could render the system incapable of performing its intended safety function. If the malfunction or condition is caused during maintenance, then no report is required;

(4) An unanticipated or uncontrolled change in reactivity greater than $1.00. Reactor trips resulting from a known cause are excluded;

(5) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary;

(6) An observed inadequacy in the implementation of either 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 operations.

Review

A review is a qualitative examination of records, procedures, or other documents prior to implementation from which appropriate recommendations are made.

Safety Channel

A safety channel is a channel in the reactor safety system.

Safety Limit

Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. For the Texas A&M NSC TRIGA reactor the safety limit is the maximum fuel element temperature that can be permitted with confidence that no damage to any fuel element cladding will result.

Scram Time

Scram time is the elapsed time between the initiation of a scram signal and the instant that the slowest scrammable control rod reaches its fully inserted position.

Senior Reactor Operator

A Senior Reactor Operator is an individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

Shall, Should and May

The word “shall” is used to denote a requirement; the word “should” to denote a recommendation; and the word “may” to denote permission, neither a requirement nor a recommendation.

Shutdown Margin

Shutdown margin is the minimum shutdown 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. This margin is determined assuming that the most reactive scrammable rod and any non-scrammable rods are fully withdrawn, and that the reactor will remain subcritical by this calculated margin without any further operator action.

Steady State Mode

Steady state mode of operation shall mean operation of the reactor with the mode selector switch in the steady state position.

Surveillance Intervals

Annually - an interval not to exceed 15 months.

Biennially - an interval not to exceed 30 months.

Monthly - an interval not to exceed 6 weeks.

Quarterly - an interval not to exceed 4 months.

Semiannually - an interval not to exceed 7.5 months.

Weekly - an interval not to exceed 10 days.

True Value

The true value is the actual value of a parameter.

Unscheduled Shutdown

An unscheduled shutdown is any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation. It does not include shutdowns that occur during testing or check out operations.

The definitions above are either standard definitions used in research reactor TSs or are facility-specific definitions that NRC staff finds to be consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that the licensee's TS definitions are acceptable.

5.2 Safety Limits and Limiting Safety System Settings

5.2.1 TS 2.1 Safety Limit—Fuel Element Temperature

See Section 2.6.3.1 of this SER.

5.2.2 TS 2.2 Limiting Safety System Setting

See Section 2.6.3.2 of this SER.

5.3 Limiting Conditions of Operation

5.3.1 TS 3.1 Reactor Core Parameters

5.3.1.1 TS 3.1.1 Steady State Operation

See Section 2.6.1.1 of this SER.

5.3.1.2 TS 3.1.2 Pulse Mode Operation

See Section 2.6.1.3 of this SER.

5.3.1.3 TS 3.1.3 Shutdown Margin

See Section 2.2.3 of this SER.

5.3.1.4 TS 3.1.4 Core Configuration Limitation

See Section 2.2.2 of this SER.

5.3.1.5 TS 3.1.5 Reactor Fuel Parameters

See Section 2.3.2 of this SER.

5.3.1.6 TS 3.1.6 Maximum Excess Reactivity

See Section 2.2.4 of this SER.

5.3.2 TS 3.2 Reactor Control and Safety Systems

5.3.2.1 TS 3.2.1 Reactor Measuring Channels

TS 3.2.1 states the following:

> Specification
>
> The reactor shall not be operated in the specific mode of operation unless the channels listed in Table 1 are operable.

Table 1: Channels Required for Operation

Channel	Minimum Number Operable	Operating Mode	
		S.S.	Pulse
Fuel Element Temperature	1	x	x
Linear Power Level	1	x	-
Log Power Level	1	x	-
Integrated Pulse Power	1	-	x
Pool Water Temperature	1	x	x

TS 3.2.1 helps ensure that during the normal operation of the NSCR, in the specified mode of operation (i.e., steady-state or pulse), sufficient information is available to the reactor operator to help ensure safe operation of the reactor. The minimum number of operable measuring channels shown in Table 1 of the TS will provide the reactor operator with the following safety measures: fuel temperature displayed at the control console giving continuous information on this parameter, which has a specified safety limit, and linear, log, and integrated pulse power-level monitors to ensure that the reactor power level is adequately monitored for both steady-state and pulsing modes of operation.

The licensee states that the reactor pool water temperature is monitored to ensure that the reactor thermal-hydraulic conditions are bounded by conditions analyzed in the SAR. TS 3.8.3 provides that the pool water temperature limit shall be less than 60 degrees C (140 degrees F). There is a continuous temperature indication with an alarm at 50 degrees C (122 degrees F). All operators are well trained to respond to an increase in reactor pool temperature. The administrative limit, in combination with operator training and existing cooling capacity, helps ensure that the pool temperatures will stay below 60 degrees C (140 degrees F).

The NRC staff reviewed the information provided in the SAR, as supplemented, and finds that TS 3.2.1 helps ensure that the reactor will not be operated unless the required minimum number of measuring channels is available to the operator to ensure safe operation of the reactor. On this basis of the information provided above, the NRC staff concludes that TS 3.2.1 is acceptable.

5.3.2.2 TS 3.2.2 Reactor Safety Systems and Interlocks

TS 3.2.2 states the following:

> Specification
>
> The reactor shall not be operated unless the safety circuits and interlocks described in Tables 2a and 2b are operable. However, any single safety channel or interlock may be inoperable when the reactor is operating for the purpose of performing a channel check, channel test, or channel calibration. If any required safety channel or interlock becomes inoperable while the reactor is operating, for reasons other than identified in these TS, then the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shutdown.

Table 2a: Safety Channels Required for Operation

Safety Channel	Number Operable	Function	Operating Mode	
			S.S.	Pulse
Fuel Element Temperature	1	Scram ≤ 975^{0}F	X	X
High Power Level	2	Scram ≤ 1.25MW	X	-
Console Scram Button	1	Manual Scram	X	X
High Power Level Detector Power Supply	2	Scram on loss of supply voltage	X	-
Preset Timer	1	Transient Rod Scram 15 seconds or less after pulse	-	X
Pool Water Temperature	1	Manual scram if temperature reaches 60 C	X	X

Table 2b: Interlocks Required for Operation

Safety Channel	Number Operable	Function	Operating Mode	
			S.S.	Pulse
Log Power	1	Prevents withdrawal of the Shim Safety Control Rods at an indicated log power of less than $4x10^{-3}$ W.	X	-
Log Power	1	Prevent pulsing of the Transient Rod when log power is above 1 kW.	-	X
Transient Rod Position	1	Prevents application of air to the Transient Rod unless the Transient Rod is fully inserted.	X	-
Shim Safety and Regulating Rod Position	1	Prevents Shim Safety and Regulating Control Rod withdrawal during a pulse.	-	X
Pulse Stop Electro-Mechanical Interlock	1	Prevents application of air to the Transient Rod unless the mechanical pulse stop is installed.	-	X

TS 3.2.2, Table 2a and Table 2b, help ensure that during the normal operation of the NSCR in the specified mode of operation (i.e., steady-state or pulse), the minimum number of reactor safety channels and interlocks required for safe operation of the reactor are operable. The minimum number of operable reactor safety channels and interlocks shown in TS 3.2.2, Table 2a and Table 2b, will provide the following safety measures:

- Fuel element temperature scram, as measured by an IFE thermocouple, and high power level scram provide protection to ensure that the reactor can be shut down before exceeding the SL on the fuel element temperature.

- Manual scram allows the reactor operator to shut down the system if an unsafe or abnormal condition occurs.

- The reactor scrams if the power supply for the power level safety channels fails to help prevent the operation of the reactor without adequate instrumentation.

- The preset timer scram helps ensure that the reactor power level will reduce to a low level after pulsing.

- The pool water temperature indication will alert the reactor operator to scram the reactor if the pool temperature reaches its limit of 60 degrees C (140 degrees F).

- Interlock to prevent startup of the reactor at power levels less than 4×10^{-3} W, which corresponds to approximately 2 counts per second, ensures that sufficient neutrons are available for proper startup.

- Interlock to prevent the initiation of a pulse above 1 kWt is to ensure that the magnitude of the pulse will not cause the fuel element temperature to exceed the safety limit.

- Interlock to prevent application of air to the transient rod unless the cylinder is fully inserted prevents pulsing of the reactor in steady-state mode.

- Interlock to prevent withdrawal of the shim safety or regulating control rods in the pulse mode to prevent the reactor from being pulsed while on a positive period.

- Interlock to prevent application of air to the transient rod unless the mechanical pulse stop is installed to prevent a reactor pulse of sufficient worth to exceed the safety limit on fuel element temperature.

The NRC staff finds that an important parameter in ensuring fuel element integrity is the fuel element temperature. TS 3.2.1 and TS 3.2.2 require a fuel element temperature measuring channel and a fuel element temperature safety channel. To help ensure that the fuel element temperature is properly monitored, TS 4.2.2 (see SER Section 5.4.2.2) defines the surveillance requirements of the fuel element temperature measuring channel and fuel element temperature safety channel.

The NRC staff reviewed TS 3.2.2 and finds the safety channels and interlocks provided in TS 3.2.2, Table 2a and Table 2b, consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. These safety channels and interlocks provide a comprehensive and diverse method to help ensure that the NSCR will be operated safely. Based on the information provided above, the NRC staff concludes that TS 3.2.2 is acceptable.

5.3.2.3 TS 3.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

See Section 2.3.4.2 of this SER.

5.3.3 TS 3.3 Confinement

5.3.3.1 TS 3.3.1 Operations that Require Confinement

TS 3.3.1 states the following:

> <u>Specifications</u>[1]
>
> Confinement of the reactor building shall be required during the following operations:
>
> 1. Reactor operating;
> 2. Movement of irradiated fuel elements or fuel bundles;
> 3. Core or control rod work that could cause a change in reactivity of more than one dollar; or
> 4. Handling of radioactive materials with the potential for airborne release.
>
> [1] For periods of maintenance to the central exhaust system, entry doors to the reactor building shall remain closed except for momentary opening for personnel entry or exit. The central exhaust system shall not remain inoperable during periods of maintenance for more than one hour.

TS 3.3.1 helps ensure that the confinement of the reactor building is in effect during reactor operation or handling of radioactive materials with the potential for airborne release, so that if a potential radioactivity release occurred, the consequences would be minimized. The handling of radioactive materials can result in the accidental or controlled release of airborne radioactivity to the reactor building or directly into the building exhaust system. In these cases, the control of air into and out of the reactor building is necessary.

The NRC staff reviewed TS 3.3.1 and finds that TS 3.3.1 helps provide an additional barrier to limit the spread of airborne radioactive material and helps ensure that the potential radiological consequences are below the limits of 10 CFR Part 20. Based on the information described above, the NRC staff concludes that TS 3.3.1 is acceptable.

5.3.3.2 TS 3.3.2 Equipment to Achieve Confinement

TS 3.3.2 states the following:

Specifications[2]

1. The minimum equipment required to be in operation to achieve confinement of the reactor building shall be the central exhaust system, which consists of the central exhaust fan, isolation louvers, and associated duct work.

2. The central exhaust system shall be considered operating when it creates a minimum of 0.1 inch of water negative pressure at the sample point in the central exhaust system duct work.

3. Controls for establishing the operation of the central exhaust system during normal and emergency conditions shall be available in the Emergency Support Center.

4. The central exhaust system shall be isolated automatically by alarm level signals from the stack particulate, or stack gas (xenon) facility air monitor.

[2] During periods of maintenance to the central exhaust system, entry doors to the reactor building shall remain closed except for momentary opening for personnel entry or exit. The central exhaust fan system shall not remain inoperative during periods of maintenance for more than one hour.

In the SAR, as supplemented, the licensee states that the purpose of TS 3.3.2 is to help ensure that the ventilation system is operable and in operation to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation.

TS 3.3.2, Specification 1, helps ensure that the equipment required to establish confinement is operable, which keeps the consequences of a potential release below 10 CFR Part 20 limits.

TS 3.3.2, Specification 2, helps ensure that the exhaust system has the capability to maintain negative pressure in the confinement building. Operation of the central exhaust fan will achieve confinement of the reactor building during normal and emergency conditions when the controls for air input are set such that the central exhaust fan capacity remains greater than the amount of air being delivered to the reactor building.

TS 3.3.2, Specification 3, helps ensure the availability of ventilation control in the reception room for remote accessibility during a declared emergency condition. The control panel for the ventilation system provides for manual selection of air input to the reactor building and the automatic or manual selection of air removal.

TS 3.3.2, Specification 4, helps ensure the isolation of the central exhaust system when radioactive fission products are present in the confinement or exhaust duct allowing the operator to secure the confinement and limit the release of airborne radioactive material.

TS 3.3.2 helps ensure that, during the normal operation of the NSCR, the concentration of gaseous radioactive isotopes Ar-41 and N-16 are kept below the limits in Appendix B, “Annual

Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20 for occupational and public dose. TS 3.3.2 also helps ensure that occupational and public exposures during accident conditions at the NSCR will be kept below 10 CFR Part 20 limits.

The NRC staff reviewed TS 3.3.2 and finds that TS 3.3.2 helps ensure that the confinement and central exhaust system is properly operating when a potential for radiological release is present and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 3.3.2 is acceptable.

5.3.4 TS 3.4 Ventilation System

TS 3.4 states the following:

> The LCO for Ventilation System is covered by TS 3.3.2 Equipment to Achieve Confinement.

See Section 5.3.3.2 of this SER.

5.3.5 TS 3.5 Radiation Monitoring Systems and Effluents

5.3.5.1 TS 3.5.1 Radiation Monitoring

See Section 3.1.4.2 of this SER.

5.3.5.2 TS 3.5.2 Argon-41 Discharge Limit

See Section 3.1.1.1.1 of this SER.

5.3.6 TS 3.6 Limitations on Experiments

5.3.6.1 TS 3.6.1 Reactivity Limits

TS 3.6.1 states the following:

> <u>Specifications</u>
>
> The reactor shall not be operated unless the following conditions governing experiments exist:
>
> 1. The absolute reactivity worth of any single, movable or unsecured experiment shall be less than $1,
>
> 2. The reactivity worth of any secured experiment shall be less than $2, and
>
> 3. The sum of the absolute reactivity of all experiments shall be less than $5.

TS 3.6.1, Specification 1, helps ensure that the reactivity limit of a single unsecured experiment is less than $1.00 in order to prevent an unexpected prompt criticality. The NRC staff finds that the $1.00 limit is substantially below the analyzed maximum allowable pulse size of $2.95.

TS 3.6.1, Specification 2, helps to establish the reactivity limit of $2.00 for secured experiments. Because the experiment is held stationary in the reactor, the likelihood that it would fall away from the core to produce an undesirable step increase in reactivity is minimized. The NRC staff finds that in TS 3.6.1, Specification 2, the reactivity limit of $2.00 for secured experiments is designed to be below the analyzed maximum allowable pulse size of $2.95.

TS 3.6.1, Specification 3, helps ensure that the proposed limit of the sum of the absolute value reactivity worth of all individual experiments is less than $5.00. The purpose of TS 3.6.1, Specification 3, is to have total experiment reactivity worth be consistent with the limit on excess reactivity and shutdown margin. The NRC staff finds that this value is permissible if the licensee demonstrates that it does not violate the TS limit on excess reactivity and shutdown margin. See Sections 2.2.4 and 2.2.5 in this SER for the NRC staff's review of excess reactivity and shutdown margin, respectively.

The NRC staff reviewed TS 3.6.1, and finds that the experimental reactivity limits in TS 3.6.1, Specifications 1 through 3, help ensure that the reactivity limits for various experiments remain within the bounds that have been analyzed and shown to be safe. Based on the information provided above, the NRC staff concludes that TS 3.6.1, Specifications 1 through 3, are acceptable.

5.3.6.2 TS 3.6.2 Material Limitations

TS 3.6.2 states the following:

<u>Specifications</u>

1. Explosive materials in quantities inclusively between 25 milligrams and 5 pounds (TNT-equivalent) shall not be allowed within the reactor building except as noted below in TS. Explosive materials in quantities greater than 5 pounds (TNT-equivalent) shall not be allowed within the reactor building. Irradiation of explosive materials shall be restricted as follows:

    a. Explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall not be irradiated in the reactor pool. Explosive materials in quantities up to 25 milligrams (TNT-equivalent) may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the container;

    b. Explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall only be allowed in the lower research level and laboratory building, excluding the heat exchanger room and demineralizer room;

c. Irradiation of explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall be permitted only in the neutron radiograph facility;

d. Explosive materials in quantities greater than or equal to 5 pounds (TNT-equivalent) shall not be irradiated in experimental facilities; and

e. Cumulative exposures for explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall not exceed 10^{12} n/cm^2 for neutron or 25 Roentgen for gamma exposures.

2. Corrosive materials used in a reactor experiment shall be double encapsulated. Exceptions may only be made if a detailed analysis and/or prototype testing with small amounts of materials demonstrates that the experiment presents negligible risk.

TS 3.6.2, Specification 1, item a through item e, help ensure that explosive materials are limited and evaluated before irradiation to prevent damage to the reactor or reactor safety systems resulting from failure of an experiment involving explosive materials.

TS 3.6.2, Specification 1, item a, helps ensure that explosive material to be irradiated in the reactor is limited to less than 25 milligrams (mg). It also helps to ensure that for explosive material up to 25 mg that may be irradiated, the pressure produced on detonation of the explosive has been calculated or experimentally demonstrated to be less than half the design pressure of the irradiation container.

TS 3.6.2, Specification 1, item b, helps ensure that explosive materials within the reactor building in a quantity equal or exceeding 25 mg are restricted from areas containing the reactor pool, the upper research level, the demineralizer room, the cooling equipment room, and the interior of the pool containment structure.

TS 3.6.2, Specification 1, item c, helps ensure that the irradiation of explosive materials in quantity equal or exceeding 25 mg is restricted to the neutron radiograph facility.

TS 3.6.2, Specification 1, item d, helps ensure that the total amount of explosive materials that may be irradiated in the neutron radiograph facility is limited to less than 5 pounds.

TS 3.6.2, Specification 1, item e, helps ensure that the maximum radiation exposure of explosive materials is limited to prevent any change in explosive stability (Ref. 16).

TS 3.6.2, Specification 1, items a through e, involve limits for the irradiation of explosive materials, which were evaluated by the NRC staff and described in the SER for License Amendment No. 7, dated October 31, 1979 (Ref. 43). As part of the license renewal review, the NRC staff reviewed License Amendment No. 7, and concluded that the analytical results remained acceptable. Amendment No. 7 remains as part of the staff's basis for approval of this license renewal. Based on the information provided above, the NRC staff concludes that TS 3.6.2, Specification 1, items a through e, are acceptable.

TS 3.6.2, Specification 2, helps ensure that double encapsulation is required for potentially corrosive materials to reduce the likelihood that encapsulation failure could occur and cause corrosive materials to damage the fuel element cladding or other reactor components. The NRC staff finds that TS 3.6.2, Specification 2, is consistent with the guidance in Regulatory Guide 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” issued November 1973 (Ref. 36). Based on its review of the information described above, the NRC staff concludes that TS 3.6.2, Specification 2, is acceptable.

5.3.6.3 TS 3.6.3 Failures and Malfunctions

TS 3.6.3 states the following:

<u>Specifications</u>

1. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (a) normal operating conditions of the experiment or reactor, (b) credible accident conditions in the reactor, or (c) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor building or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10 CFR 20.

2. In calculations pursuant to 1) above, the following assumptions shall be used:

    a. If the effluent from an experimental facility exhausts through a holdup tank that closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape;

    b. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles at least 10% of these vapors can escape; or

    c. For materials whose boiling point is above 130°F and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape.

3. If a capsule fails and releases material that could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the NSC Director or designated alternate and determined to be satisfactory before operation of the reactor is resumed.

TS 3.6.3, Specification 1 and Specification 2, set limits on the radioactive products produced in experiment materials that may release airborne radioactive particles and help provide conditions to be used in the safety analysis of the experiment. The purpose of TS 3.6.3, Specification 1 and Specification 2, is to help ensure that potential releases of radioactive material from experiments are bounded by the exposure limits in 10 CFR Part 20 for NSC staff and members of the public. This includes experiment failures under normal reactor operations, credible reactor accident conditions, and accident conditions in the experiment. The assumptions in TS 3.6.3, Specification 1 and Specification 2, are consistent with those made for other research reactors and help ensure that source term calculations are conservative.

The NRC staff reviewed TS 3.6.3, Specifications 1 and 2, and finds that these specification help to limit doses from potential experiment failure or malfunction from exceeding 10 CFR Part 20 limits. Based on the information provided above, the NRC staff concludes that TS 3.6.3, Specifications 1 and 2 are acceptable.

TS 3.6.3, Specification 3, helps ensure that any capsule failures are inspected to determine the consequences and need for corrective action. The NRC staff finds that the requirements of TS 3.6.3, Specification 3, help ensure that the reactor does not operate without the NSC staff checking for possible damage to reactor fuel or structure from a failed experiment. Potential damage to reactor fuel or structure must be brought to the attention of the NSC Director or designated alternate for review to determine if corrective actions were adequate to help ensure continued safe operation of the reactor. Based on the information provided above, the NRC staff concludes that TS 3.6.3, Specification 3, is acceptable.

5.3.7 TS 3.7 As Low As Reasonably Achievable (ALARA) Radioactive Effluents Released

See Sections 3.1.3.1 and 3.1.1.2.1 of this SER.

5.3.8 TS 3.8 Primary Coolant Conditions

5.3.8.1 TS 3.8.1 Primary Coolant Purity

See Section 2.4.2 of this SER.

5.3.8.2 TS 3.8.2 Primary Coolant Level and Leak Detection

See Section 2.4.4 of this SER.

5.3.8.3 TS 3.8.3 Primary Coolant Temperature

See Section 2.4.6 of this SER.

5.4 Surveillance Requirements

TS 4 states the following:

> Specification
>
> Surveillance requirements may be deferred during reactor shutdown (except TS 4.1.5, 4.2.3, 4.5, 4.8.1, and 4.8.2); however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practical after reactor startup. Scheduled surveillance, which cannot be performed with the reactor operating, may be deferred until a planned reactor shutdown.

	TS	Possible to Defer During Shutdowns?	Required prior to operations?
1.	4.1.1 Steady State Operation	Yes	Yes
2.	4.1.2 Pulse Mode Operation	Yes	Yes
3.	4.1.3 Shutdown Margin	Yes	Yes
4.	4.1.4 Core Configuration Limitation	Yes	Yes
5.	4.1.5 Reactor Fuel Elements	No	N/A
6.	4.1.6 Maximum Excess Reactivity	Yes	Yes
7.	4.2.1 Reactor Channels	Yes	Yes
8.	4.2.2 Reactor Safety Systems and Interlocks	Yes	Yes
9.	4.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time	No	N/A
10.	4.3 Confinement	Yes	Yes
11.	4.5 Radiation Monitoring Systems and Effluents	No	N/A
12.	4.6 Experiments	Yes	Yes
13.	4.8.1 Primary Coolant Purity	No	N/A
14.	4.8.2 Primary Coolant Level and Leak Detection	No	N/A
15.	4.8.3 Primary Coolant Temperature	Yes	Yes

Any additions, modifications, or maintenance to the central exhaust system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the rod drive mechanism, or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications approved by the Reactor Safety Board. A system shall not be considered operable until it is successfully tested.

TS 4 helps ensure that the quality of systems and components will be maintained to their original design and fabrication specifications, or, if to new specifications, that those specifications have been reviewed. TS 4 also governs the conduct of surveillance requirements required to allow operational flexibility that does not impact safety. TS 4 follows the guidance in NUREG-1537, Appendix 14.1, Section 4.0.

The NRC staff finds that TS 4 provides appropriate surveillance practices and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis its review of the information provided above, the NRC staff concludes that TS 4 is acceptable.

5.4.1 TS 4.1 Reactor Core Parameters

5.4.1.1 TS 4.1.1 Steady State Operation

See Section 2.6.1.2 of this SER.

5.4.1.2 TS 4.1.2 Pulse Mode Operation

See Section 2.6.1.4 of this SER.

5.4.1.3 TS 4.1.3 Shutdown Margin

See Section 2.2.5 of this SER.

5.4.1.4 TS 4.1.4 Core Configuration Limitations

TS 4.1.4 states the following:

> Specification
>
> Each core configuration change shall be determined to meet the requirements of TS 3.1.4 prior to the core loading.

TS 4.1.4 provides core configuration requirements of TS 3.1.4 and specifications assumed in the SAR, as supplemented. The NRC staff finds that TS 4.1.4 helps ensure that the core configuration is maintained consistent with the SAR assumptions and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on its review of the information provided above, the NRC staff concludes that TS 4.1.4 is acceptable.

5.4.1.5 TS 4.1.5 Reactor Fuel Elements

See Section 2.3.3 of this SER.

5.4.1.6 TS 4.1.6 Maximum Excess Reactivity

See Section 2.2.6 of this SER.

5.4.2 TS 4.2 Reactor Control and Safety Systems

5.4.2.1 TS 4.2.1 Reactor Channels

TS 4.2.1 states the following:

> Specification
>
> A channel test of each of the reactor channels for the intended mode of operation, as identified in Table 1, shall be performed before each day's operation or before each operation extending more than one day.

TS 4.2.1, helps ensure that the reactor safety system and interlocks, listed in TS 3.2.1, Table 1, are operational before each day's operation or before each operation extending more than one day. The licensee states that a channel test of the two power-level safety channels and a test of interlocks before reactor startup, when combined with the channel calibrations required by TS 4.2.2, provide assurance that the power-level measuring channels are providing accurate power-level indications that will help ensure that the reactor thermal power level will not exceed 1 MWt. A channel test of the fuel element temperature, integrated pulse power and pool water temperature channels ensures that the channels will be available to provide accurate indications for the operators.

The NRC staff reviewed the information in the SAR, as supplemented, and finds that the scope and frequency of performing these channel tests on the reactor channels are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 4.2.1 is acceptable.

5.4.2.2 TS 4.2.2 Reactor Safety Systems and Interlocks

TS 4.2.2 states the following:

Specifications

1. A channel test of each of the reactor safety system channels and interlocks for the intended mode of operation, as identified in Table 2, shall be performed before each day's operation or before each operation extending more than one day.

2. A channel calibration of the fuel element temperature measuring channels shall be performed semiannually.

TS 4.2.2, Specification 1, helps ensure that the reactor safety system and interlocks, listed in TS 3.2.2, Table 2a and 2b, are operational before each day's operation or before each operation extending more than one day. The licensee states in the SAR that a channel test of the safety channels and interlocks, in TS 3.2.2, Table 2a and 2b, prior to reactor startup provide assurance that the safety channels and interlocks are available to help ensure reactor operation is performed in accordance with the assumptions of the SAR, as supplemented, and that the reactor thermal power level will not exceed 1 MWt. The NRC staff finds that the scope and frequency of performing these channel checks on the reactor safety systems and interlocks are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

TS 4.2.2, Specification 2, helps ensure that the fuel element temperature measuring channels are calibrated semiannually. This calibration helps ensure that the scram function provided by the temperature measuring channels is accurate and provides the scram function at the power level provided in the SAR. Based on its review of the information provided above, the NRC staff concludes that TS 4.2.2 is acceptable.

5.4.2.3 TS 4.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

See Section 2.3.4.3 of this SER.

5.4.3 TS 4.3 Confinement

TS 4.3 states the following:

Specifications

1. The central exhaust system shall be channel checked prior to reactor operation or radioactive material handling.

2. During periods of operation, or radioactive material handling, the central exhaust system shall be verified operable weekly including automatic isolation on receipt of a high radiation signal. This specification is not required during periods of non-operation, e.g., holidays, extended maintenance outages.

TS 4.3, Specification 1 and Specification 2, help ensure that the central exhaust system is channel checked before reactor operation or radioactive material handling and tested weekly during operation or radioactive material handling.

The NRC staff reviewed TS 4.3, and finds that TS 4.3, Specification 1 and Specification 2, help to ensure that the system will operate in accordance with the design features and will provide confinement should a radioactive airborne release occur. The NRC staff finds that TS 4.3 surveillance intervals are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007 and help ensure that the system is operating properly. On the basis of the information provided above, the NRC staff concludes that TS 4.3 is acceptable.

5.4.4 TS 4.4 Ventilation System

TS 4.4 states the following:

Specification

The Ventilation Systems surveillance requirements are specified in TS 4.3.

TS 4.4 indicates that TS 4.3 provides the surveillance requirement for the ventilation system.

The NRC staff reviewed TS 4.3 and finds that the surveillance requirements in TS 4.3 provide the necessary testing to help ensure that the ventilation system is maintained and available. Based on the information provided above, the NRC staff concludes that TS 4.4 is acceptable.

5.4.5 TS 4.5 Radiation Monitoring Systems and Effluents

See Sections 3.1.1.1.2, 3.1.1.2.2, 3.1.3.2, and 3.1.4.3 of this SER.

5.4.6 TS 4.6 Experiments

TS 4.6 states the following:

Specifications

1. A new experiment shall not be installed in the reactor or its experimental facilities until a hazard analysis has been performed and reviewed for compliance with Section 3.6 and Section 6.5 of the Technical Specifications. Minor modifications to a reviewed and approved experiment may be made at the discretion of the Director, or designee, with concurrence from the Radiation Safety Officer, or designee. The Director, or designee, and the Radiation Safety Officer, or designee, shall review the hazards associated with the modifications and determine that the modifications do not create a significantly different, a new, or a greater safety risk than the original approved experiment, and does not require a review under 10 CFR 50.59.

2. The performance of an experiment classified as an approved experiment shall not be performed until a licensed senior operator and the Radiation Safety Officer, or designee has reviewed it for compliance with these TS.

3. The reactivity worth of the experiment shall be estimated or measured, as appropriate, before reactor operation.

TS 4.6, Specification 1, helps ensure that, before installation of a new experiment in the reactor, a hazard analysis is performed and reviewed for compliance in accordance with TS 3.6 and TS 6.5. Minor modifications to an approved experiment may be made at the discretion of the Director with concurrence from the Radiation Safety Officer, or designee. The NRC staff reviewed TS 4.6, Specification 1, and finds that the review of the hazard analysis for new experiments or modification to an experiment helps ensure that hazards associated with any modifications do not create a significantly different, a new, or a greater safety risk than the approved experiment. The NRC staff also finds that new experiments and modifications to experiments will be reviewed in accordance with 10 CFR 50.59. The NRC staff finds that TS 4.6, Specification 1, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

TS 4.6, Specification 2, helps ensure that an approved experiment is reviewed by a licensed senior reactor operator and the Radiation Safety Officer, or designee, before performance. This review helps ensure that an adequate safety analysis and reactivity measurement or estimate has been performed prior to the implementation of the experiment. The NRC staff reviewed TS 4.6, Specification 2, and finds that TS 4.6, Specification 2, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007, and that the review of an approved experiment for TS compliance prior to review is an acceptable means to confirm that the TS requirements are satisfied prior to performance.

TS 4.6, Specification 3, helps ensure that the reactivity worth of an experiment shall be estimated or measured, as appropriate, before the reactor is operated with the experiment. The NRC staff reviewed TS 4.6, Specification 3, and finds that TS 4.6, Specification 3, helps ensure that the reactivity of an experiment is reviewed prior to the operation of the reactor and will meet the requirement of TS 3.6.1.

The NRC staff finds that TS 4.6, Specification1 through 3, help ensure that experiments, including their reactivity worths, are reviewed and approved before implementation. TS 4.6, Specifications 1 through 3, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1--2007. On the basis of its review of the information described above, the NRC staff concludes that TS 4.6 is acceptable.

5.4.7 TS 4.7 ALARA Radioactive Effluents Released

TS 4.7 states the following:

> <u>Specification</u>
>
> Surveillance for the LCO 3.7 ALARA Radioactive Effluents Released is incorporated into TS 4.5.

See Section 5.4.5 (Sections 3.1.1.1.2, 3.1.1.2.2, 3.1.3.2, and 3.1.4.3) of this SER.

5.4.8 TS 4.8 Primary Coolant Conditions

5.4.8.1 TS 4.8.1 Primary Coolant Purity

See Section 2.4.3 of this SER.

5.4.8.2 TS 4.8.2 Primary Coolant Level and Leak Detection

See Section 2.4.5 of this SER.

5.4.8.3 TS 4.8.3 Primary Coolant Temperature

See Section 2.4.7 of this SER.

5.5 <u>Design Features</u>

5.5.1 TS 5.1 Site and Facility Description

TS 5.1 states the following:

> <u>Specification</u>
>
> The licensed area of the facility is the area inside the site boundary. The boundary is defined by a fence surrounding the site. This description coincides with that of the restricted area.

TS 5.1 describes the NSC facility site boundary, including the licensed and restricted areas. The NRC staff reviewed the facility boundary as described in the SAR, and toured the facility during license renewal reviews, and finds that TS 5.1 helps ensure that important features associated with the licensed and restricted areas of the NSC facility, used to house the NSCR, are defined. The NRC staff finds that TS 5.1 accurately describes the facility site’s licensed and restricted areas consistent with the SAR, as supplemented, and the NSC emergency plan. The

NRC staff finds that TS 5.1 is consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1-2007. On the basis of the information described above, the NRC staff concludes that TS 5.1 is acceptable.

5.5.2 TS 5.2 Reactor Fuel

See Section 2.3.1 of this SER.

5.5.3 TS 5.3 Reactor Core

See Section 2.2.1 of this SER.

5.5.4 TS 5.4 Control Rods

See Section 2.3.4.1 of this SER.

5.5.5 TS 5.5 Radiation Monitoring System

See Section 3.1.4.1 of this SER.

5.5.6 TS 5.6 Fuel Storage

TS 5.6 states the following:

Specifications

1. All fuel elements and fueled devices shall be stored in a geometrical array for which the k-effective is less than 0.8 for all conditions of moderation and reflection.

2. Irradiated fuel elements and fueled devices shall be stored in an array, which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed design values.

TS 5.6, Specification 1, helps ensure that the k_{eff} value is limited to 0.8, which is less than the k_{eff} value of 0.9 that is provided as guidance in NUREG-1537 and ANSI/ANS-15.1-2007. The licensee provided k_{eff} values for fuel elements in storage in Section 9.2.2 of the SAR. The k_{eff} values for fuel elements stored in the vault or reactor pool, in the most reactive condition (fully submerged in water), were 0.45 and 0.47, respectively. The NRC staff finds that the licensee's calculated k_{eff} values to be much lower than the proposed TS value of 0.8.

TS 5.6, Specification 2, helps ensure that adequate cooling by natural convection, either by water or air, of stored irradiated fuel elements and fueled devices will be incorporated into the design of any storage array. The NRC staff finds that this design feature is acceptable to protect the fuel element cladding and fission product barrier.

The NRC staff reviewed TS 5.6 and finds that the k_{eff} value required in TS 5.6 to be less than the values in guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Additionally, the NRC staff finds that TS 5.6 cooling requirements are adequate to ensure protection of the fuel cladding integrity. Based on the information provided above, the NRC staff finds that TS 5.6, Specifications 1 and 2, are acceptable.

5.5.7 TS 5.7 Reactor Building and Central Exhaust System

TS 5.7 states the following:

Specifications

1. The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 180,000 cubic feet.

2. The reactor building shall be equipped with a central exhaust system designed to exhaust air or other gases from the reactor building and release them from a stack at minimum of 85 feet from ground level.

3. Emergency isolation controls for the central exhaust system shall be located in the emergency support center and the system shall be designed to shut down in the event of an alarm on the stack particulate monitor (FAM Ch.1) or stack gas xenon (FAM Ch. 5) radiation monitoring channels.

TS 5.7, Specification 1, helps ensure that the NSCR is housed in a facility designed to restrict leakage and with a minimum free volume sufficient to capture and control any radioactive effluents before release. The NRC staff finds that the minimum free volume assumption is used to calculate the radioactive concentration used in the MHA accident dose calculation for an exposed worker in the NSC.

TS 5.7, Specification 2, helps ensure that the central exhaust system is designed to exhaust air from the reactor building and release it from a stack elevated a minimum of 85 ft (25.9 m) above the ground. The NRC staff finds that the proposed height of the exhaust stack helps ensure dispersion and dilution of effluents released from the stack before they reach the ground. The NRC staff finds that TS 5.7, Specification 2, helps ensure a ventilation system with a controlled air pathway release point.

TS 5.7, Specification 3, helps ensure that the emergency isolation controls for the central exhaust system are located in the emergency support center, which is outside the control room, should the control room become inaccessible because of a radiological or other emergency condition. TS 5.7, Specification 3, also helps ensure that the central exhaust system can be automatically shut down on an alarm signal from the stack particulate or stack gas monitors. The NRC staff finds that TS 5.7, Specification 3, helps ensure the ability of the exhaust system to isolate and to provide confinement to limit the potential release of radioactive material.

The NRC staff reviewed TS 5.7, Specification 1 through Specification 3, and finds that these specifications help ensure the design features support the assumptions used in the SAR, as supplemented, and to provide for the confinement and controlled release of radioactive materials by the central exhaust system. The NRC staff reviewed the ventilation system design as presented in the SAR, as supplemented, and finds that these design aspects are acceptable. On the basis of the information provided above, the NRC staff concludes that TS 5.7, Specification 1 through Specification 3, are acceptable.

5.5.8 TS 5.8 Reactor Pool Water Systems

See Section 2.4.1 of this SER.

5.6 TS 6 Administrative Control

TS 6 includes requirements for the conduct of operations for the NSC. The administrative controls presented in TS 6 include responsibilities, facility organization, staff qualifications, training, the safety committee, operational review and audits, procedures, required actions, and reports and records.

The primary guidance for the development of administrative controls for research reactor operation is NUREG-1537 and ANSI/ANS-15.1-2007. The licensee's proposed TS 6 is based on these guidance documents. In some cases, the wording proposed was not identical to that in NUREG-1537 and ANSI/ANS-15.1-2007. However, this review considered these cases and determined that the licensee's proposed administrative controls meet the intent of the guidance and are acceptable.

5.6.1 TS 6.1 Organization

TS 6.1 states the following:

> The Nuclear Science Center is operated by the Texas A&M University System's Texas Engineering Experiment Station (TEES), with responsibility within TEES resting with the Director or designated alternate. The Director of the Nuclear Science Center is responsible to the TEES director, or designated alternate, for the administration and the proper and safe operation of the facility. Figure 1 shows the administration chart for the Nuclear Science Center. The Reactor Safety Board advises the director of the NSC on all matters or policy pertaining to safety. The NSC Radiological Safety Officer provides onsite advice concerning safety and provides technical assistance and review in the area of radiation protection.

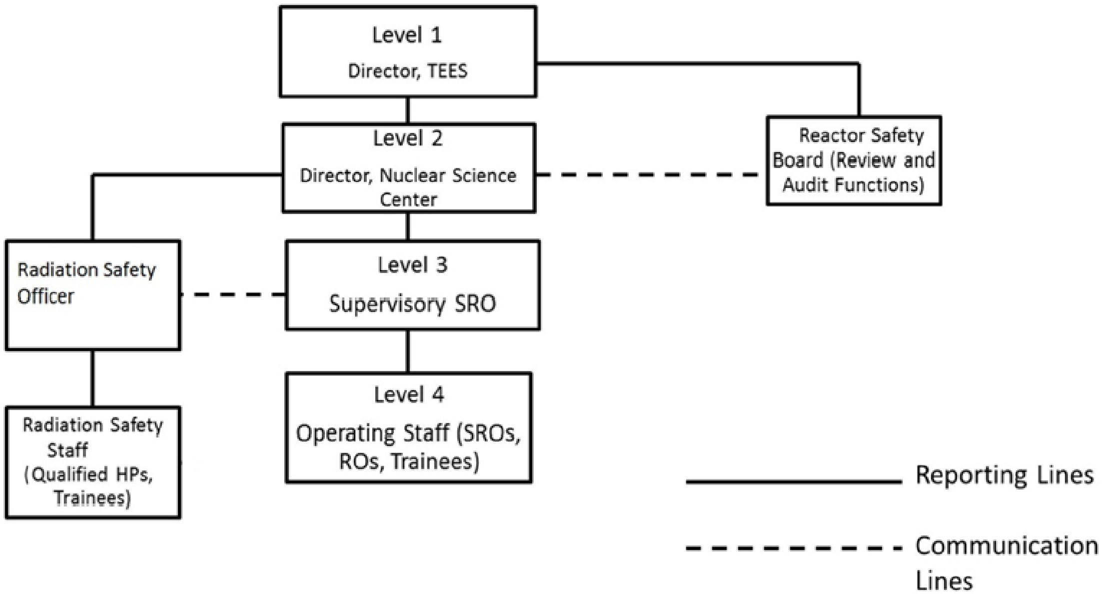

Figure 1: Organization Chart for Reactor Administration

TS 6.1 helps ensure that NSC organizational responsibilities are delineated. The NRC staff finds that the organization described in TS 6.1 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.1 is acceptable.

5.6.1.1 TS 6.1.1 Structure

TS 6.1.1 states the following:

1. A line management organizational structure provides for personnel who will administrate and operate the reactor facility.

2. The Director of TEES and the Director of the NSC have line management responsibility for adhering to the terms and conditions of the Nuclear Science Center Reactor (NSC) license and Technical Specifications and for safeguarding the public and facility personnel from undue radiation exposure. The facility shall be under the direct control of the NSC Director or a licensed senior reactor operator.

3. Management Levels:

   a. Level 1: TEES Licensee (Director of TEES): Responsible for the NSC facility license.

   b. Level 2: NSC Director: Responsible for reactor facility operation and shall report to Level 1.

c. Level 3: Supervisory SRO (One of the following—Associate Director, Reactor Manager, Reactor Supervisor): Responsible for the day-to-day operation of the NSC or shift operation and shall report to Level 2.

d. Level 4: Reactor Operating Staff: Licensed reactor operators and senior reactor operators and trainees. These individuals shall report to Level 3.

4. Reactor Safety Board (RSB):

The RSB is responsible to the licensee for providing an independent review and audit of the safety aspects of the NSC.

TS 6.1.1 helps ensure that the NSC organization structure is delineated. The NRC staff reviewed TS 6.1.1 and finds that the NSC organizational structure described in TS 6.1.1, and shown in Figure 1, is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.1.1 is acceptable.

5.6.1.2 TS 6.1.2 Responsibility

TS 6.1.2 states the following:

> Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established in Section 6.1.1. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional on appropriate qualifications.
>
> The reactor facility shall be under the direct control of the Supervisory SRO. The Supervisory SRO shall be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, procedures and requirements of the Radiation Safety Officer and the Reactor Safety Board.

TS 6.1.2 helps ensure that the NSC specific organization levels and responsibilities are delineated. The NRC staff reviewed TS 6.1.2 and finds that the organizational responsibilities stated in TS 6.1.2 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.1.2 is acceptable.

5.6.1.3 TS 6.1.3 Staffing

TS 6.1.3 states the following:

1. The minimum staffing when the reactor is not secured shall be as follows:

   a. At least two individuals shall be present at the facility complex and shall consist of at least a licensed senior reactor operator and either a licensed reactor operator or operator trainee;

   b. During periods of reactor maintenance the two individuals who shall be present at the facility complex may consist of a licensed senior reactor operator and a member of the maintenance staff who is able to carry out prescribed written instructions. During periods when the reactor is unsecured, it shall be under the direct control of the senior reactor operator;

c. A licensed reactor operator or senior reactor operator shall be in the control room;

d. The NSC Director or designated management alternate is readily available for emergencies or on call (the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility); and

e. At least one member of the Radiation Safety Staff shall be readily available at the facility or on call (the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility), to provide advice and technical assistance in the area of radiation protection.

2. A list of reactor facility personnel by name and telephone number shall be readily available for use in the control room. The list shall include:

   a. Management personnel;

   b. Radiation safety personnel; and

   c. Other operations personnel.

3. The following designated individuals shall direct the events listed:

   a. The NSC Director or designated alternate who shall be SROs shall direct any loading or unloading of fuel or control rods within the reactor core region,

   b. The NSC Director or designated alternate who shall be SROs shall direct any loading or unloading of an in-core experiment with a reactivity worth greater than $1,

   c. The senior reactor operator on duty shall be present at the facility and shall direct the recovery from an unplanned or unscheduled shutdown,

   d. The senior reactor operator on duty shall be present at the facility and shall direct each reactor startup and approach to power, and

   e. The senior reactor operator on duty shall be present at the facility and shall direct all significant reactor power changes after initial startup. A significant reactor power change is defined as one that would disable the automatic servo control, i.e. equal to or greater than 5% of reactor power.

TS 6.1.3, Specification 1, helps ensure that minimum staffing requirements are implemented when the reactor is not secured, such that at least two individuals, one a licensed senior reactor operator, are at the facility; that a licensed reactor operator or senior reactor operator is in the control room; and that the NSC Director and at least one member of the radiation safety staff are readily available. The NRC staff finds that the TS 6.1.3, Specification 1, is in conformance with the regulations in 10 CFR 50.54(m)(1), which state the following:

> A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

The NRC staff finds that TS 6.1.3, Specification 1, is also consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

TS 6.1.3, Specification 2, helps ensure that support personnel are readily available to the operating staff. The NRC staff reviewed TS 6.1.3, Specification 2, and finds that this specification is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

TS 6.1.3, Specification 3, helps ensure that the individuals responsible for directing certain operations within the facility and recovery from an unscheduled shutdown are designated. The NRC staff finds that TS 6.1.3, Specification 3, provides the necessary oversight of certain operations that could affect reactor safety and is in conformance with the regulatory requirements of 10 CFR 50.54(m)(1), and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007.

The NRC staff reviewed TS 6.1.3, and finds that TS 6.1.3, Specifications 1 through 3, satisfy the requirements of 10 CFR 50.54(m)(1), and are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.1.3, Specifications 1 through 3, are acceptable.

5.6.1.4 TS 6.1.4 Selection and Training of Personnel

TS 6.1.4 states the following:

> The selection and training of operations personnel shall be in accordance with the following:
>
> 1. Responsibility: The NSC Director or designated alternate is responsible for the selection, training and requalification of the facility reactor operators and senior reactor operators.
>
> 2. Selection: The selection of operations personnel shall be consistent with the standards related to selection in ANSI/ANS-15.4-2007.
>
> 3. Training Program: The Training Program shall be consistent with the standards related to training in ANSI/ANS-15.4-2007.
>
> 4. Requalification Program: The Requalification Program shall be consistent with the standards related to requalification in ANSI/ANS-15.4-2007.

TS 6.1.4, Specification 1 through Specification 4, help ensure that the criteria for the training and requalification programs for operations personnel are consistent with the guidance in

ANSI/ANS-15.4-2007, “Selection and Training of Personnel for Research Reactors” (Ref. 37). Requalification of licensed senior reactor operators and reactor operators also must follow the NRC approved Senior Reactor Operator and Reactor Operator Requalification Program. The NRC staff reviewed TS 6.1.4, and finds that TS 6.1.4 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.4-2007. Based on the information described above, the NRC staff concludes that TS 6.1.4, Specification 1 through Specification 4, are acceptable.

5.6.2 TS 6.2 Review and Audit

5.6.2.1 TS 6.2.1 Reactor Safety Board (RSB)

TS 6.2.1 states the following:

> The Reactor Safety Board shall be comprised of at least 3 voting members knowledgeable in fields which relate to Nuclear Safety. One of these members, the Director of TEES (Level 1 Management), will serve as the Chairman. If the Chairman is unable to attend one or a number of committee meetings he may designate a committee member as Chairman pro tem. The members are appointed by the Director of TEES (Level 1 Management) to serve one year terms. It is expected that the members will be reappointed each year as long as they are willing to serve so that their experience and familiarity with the past history of the NSC will not be lost to the committee. The Director of the NSC, TAMU Radiological Safety Officer, Head of the Department of Nuclear Engineering, and a senior member of the NSC Radiation Safety Staff shall be ex-officio members of the RSB.

TS 6.2.1 helps ensure that the RSB composition, qualifications, and operation are properly delineated. TS 6.2.1 describes the composition of the RSB and requires the RSB to include experts who are not directly involved with the operation of the NSCR. TS 6.2.1 requires that the Director of the NSC, TAMU Radiological Safety Officer, Head of the Department of Nuclear Engineering, and a senior member of the NSC Radiation Protection Office be ex-officio members of the RSB.

The NRC staff reviewed TS 6.2.1, and finds that the requirements in TS 6.2.1 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.2.1 is acceptable.

5.6.2.2 TS 6.2.2 RSB Charter and Rules

TS 6.2.2 states the following:

> The operations of the RSB shall be in accordance with a written charter, including provisions for:
>
> 1. Meeting Frequency: The RSB shall meet annually at intervals not to exceed 15 months. (Note: The facility license requires a meeting at least once per year and as frequently as circumstances warrant consistent with effective monitoring of facility activities);
>
> 2. Quorum: A quorum is comprised of not less than one-half of the voting membership where the operating staff does not constitute a majority;

3. Voting Rules: On matters requiring a vote, if only a quorum is present a unanimous vote of the quorum is required; otherwise a majority vote is required;

4. Subcommittees: The Chairman may appoint subcommittees comprised of members of the RSB including ex-officio members to perform certain tasks. Subcommittees or members of the RSB may be authorized to act for the board; and,

5. Meeting Minutes: The Chairman will designate one individual to act as recording secretary. It will be the responsibility of the secretary to prepare the minutes which will be distributed to the RSB, including the Director of TEES (Level 1 Management), within three months. The RSB will review and approve the minutes of the previous meetings. A complete file of the meeting minutes will be maintained by the Chairman of the RSB and by the Director of the NSC.

TS 6.2.2, Specifications 1 through 5, help establish the RSB charter and rules. TS 6.2.2 describes the operation of the RSB, which is responsible for an independent audit of the NSCR activities and conducts its review and audit functions in accordance with a written charter. The charter includes provisions for meeting frequency, voting rules, quorums, method of submission and content of presentations to the RSB, use of subcommittees, and minutes. NUREG-1537 and ANSI/ANS-15.1-2007 specify that the purpose of the review committee is to provide independent oversight and that the operating staff should not constitute the majority of a quorum. The RSB charter establishes a quorum of three excluding the ex-officio members, which guarantees that operations personnel will not be a majority.

The NRC staff reviewed TS 6.2.2, and finds that the composition and qualifications for the RSB, as stated in TS 6.2.2, Specifications 1 through 5, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information described above, the NRC staff concludes that TS 6.2.2 is acceptable.

5.6.2.3   TS 6.2.3 RSB Review Function

TS 6.2.3 states the following:

> The review responsibilities of the Reactor Safety Board or a designated subcommittee shall include, but are not limited to the following:
>
> 1. Review and evaluation of determinations of whether proposed changes to equipment, systems, tests, experiments, or procedures can be made under 10 CFR 50.59 or would require a change in Technical Specifications or license conditions;
>
> 2. Review of new procedures, major revisions of procedures, and proposed changes in reactor facility equipment or systems which have significant safety impact to reactor operations;
>
> 3. Review of new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;

4. Review of proposed changes to the Technical Specifications and U.S NRC issued license;

5. Review of the NSC radiation protection program;

6. Review of violations of Technical Specifications, U.S. NRC issued license, and violations of internal procedures or instructions having safety significance;

7. Review of operating abnormalities having safety significance;

8. Review of reportable occurrences listed in Section 6.6.1 and 6.6.2 of these TS; and

9. Review of audit reports.

TS 6.2.3, Specification 1 through Specification 9, help ensure that the RSB review functions are properly delineated.

The NRC staff reviewed TS 6.2.3, and finds that the RSB review functions specified in TS 6.2.3 are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.2.3, Specification 1 through Specification 9, are acceptable.

5.6.2.4 TS 6.2.4 RSB Audit Function

TS 6.2.4 states the following:

> The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. Audits shall include but are not limited to the:
>
> 1. Facility operations, including radiation protection, for conformance to the Technical Specifications, applicable license conditions, and standard operating procedures, at least once per calendar year (interval between audits not to exceed 15 months);
>
> 2. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment systems, structures, or methods of operations that affect reactor safety (at least once per calendar year (interval between audits not to exceed 15 months);
>
> 3. The retraining and requalification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months);
>
> 4. The facility emergency plan and implementing procedures (at least once every other calendar year (interval between audits not to exceed 30 months); and

5. The reactor facility security plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months).

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Director of TEES (Level 1 Management). A written report of the findings of the audit shall be submitted to the Director of TEES (Level 1 Management) and the review and audit group members within 3 months after the audit has been completed.

TS 6.2.4, Specification 1 through Specification 5, help ensure that the RSB audit functions are properly delineated, and reported to TEES Level 1 Management for corrective actions, if necessary. The NRC staff reviewed TS 6.2.4, and finds that the RSB audit functions, as stated in TS 6.2.4, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.2.4, Specification 1 through Specification 5, are acceptable.

5.6.2.5 TS 6.2.5 Audit of ALARA Program

TS 6.2.5 states the following:

> The Chairman of the RSB or designated alternate (excluding anyone whose normal job function is within the NSC) shall conduct an audit of the reactor facility ALARA program annually. The auditor shall transmit the results of the audit to the RSB at the next scheduled meeting for its review and approval.

TS 6.2.5 helps ensure that the Chairman of the RSB or designated alternate performs an annual audit of the ALARA program. The NRC staff reviewed TS 6.2.5, and finds that TS 6.2.5 is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.2.5 is acceptable.

5.6.3 TS 6.3 Radiation Safety

See Section 3.1.2.1 of this SER.

5.6.4 TS 6.4 Procedures

TS 6.4 states the following:

> Written operating procedures shall be prepared, reviewed, and approved before initiating any of the activities listed in this section. The procedures shall be reviewed and approved by the NSC Director or designated alternate, the Reactor Safety Board, and shall be documented in a timely manner. Procedures shall be adequate to ensure the safe operation of the reactor but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be used for the following items:

1. Startup, operation, and shutdown of the reactor;

2. Fuel loading, unloading, and movement within the reactor;

3. Control rod removal or replacement;

4. Routine maintenance of the control rod, drives and reactor safety and interlock systems or other routine maintenance of major components of systems that could have an effect on reactor safety;

5. Surveillance checks, calibrations, and inspections of reactor instrumentation and controls, control rod drives, area radiation monitors, facility air monitors, the central exhaust system and other systems as required by the Technical Specifications;

6. Administrative controls for operations, maintenance, and conduct of irradiations and experiments, that could affect reactor safety or core reactivity;

7. Implementation of required plans such as emergency or security plans;

8. Radiation protection program to maintain exposures and releases as low as reasonably achievable (ALARA);

9. Use, receipt, and transfer of by-product material, if appropriate; and

10. Surveillance procedures for shipping radioactive materials.

TS 6.4, Specification 1 through Specification 10, help ensure that procedures are written, reviewed, and approved before performance of the group of important activities listed in the specifications. The NRC staff reviewed TS 6.4, and finds that TS 6.4, Specification 1 through Specification 10, are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. Based on the information provided above, the NRC staff concludes that TS 6.4 is acceptable.

5.6.5 TS 6.5 Experiment Review and Approval

TS 6.5 states the following:

> Approved experiments shall be carried out in accordance with established and approved procedures.
>
> 1. All new experiments or class of experiments shall be reviewed by the RSB as required by TS 6.2.3 and implementation approved in writing by the NSC Director or designated alternate.
>
> 2. Substantive changes to previously approved experiments shall be made only after review by the RSB and implementation approved in writing by the NSC Director or designated alternate.

TS 6.5, Specification 1 and Specification 2, help ensure acceptable management control over NSCR experiments. TS 6.5 provides requirements for the review and approval of different types of experiments before they are performed at the NSCR and specifies the extent of the analysis needed to be submitted for RSB review. The NRC staff reviewed TS 6.5, and finds that TS 6.5

is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.5, Specifications 1 and 2, are acceptable.

5.6.6 TS 6.6 Required Actions

5.6.6.1 TS 6.6.1 Action to be Taken in the Event of a Safety Limit Violation

TS 6.6.1 states the following:

> In the event a safety limit is violated:
>
> 1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. NRC;
>
> 2. An immediate notification of the occurrence shall be made to the RSB Chairman and the NSC Director, and reports shall be made to the U.S. NRC in accordance with Section 6.7.2 of these specifications; and
>
> 3. A report shall be prepared which shall include:
>
>   a. Applicable circumstances leading to the violation, 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.
>
> This report shall be submitted to the RSB for review and then submitted to the U.S. NRC when authorization is sought to resume operation of the reactor.

TS 6.6.1, Specification 1 through Specification 3, help ensure that the proper actions are taken if an SL violation occurs. TS 6.6.1 requires the facility to shut down if a SL is exceeded. The facility may not resume operation without authorization from the NRC. The violation also must be reported to the RSB and to the NRC. TS 6.7.2 details the reporting requirement, specifying that the NRC must be notified within 24 hours by telephone and a report must be submitted to the NRC within 14 days. TS 6.6.1, Specification 3.c, specifies that the report should include the corrective actions to be taken to prevent recurrence. The NRC staff finds that TS 6.6.1, Specification 1 through Specification 3, are in conformance with the requirements of 10 CFR 50.36, and are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.6.1 is acceptable.

5.6.6.2 TS 6.6.2 Action to be Taken in the Event of a Reportable Occurrence Other Than a Safety Limit Violation

TS 6.6.2 states the following:

Action to be taken in the event of a reportable occurrence other than a safety limit violation:

1. NSC staff shall return the reactor to normal operating via the approved NSC procedure or shut down conditions. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the NSC Director or designated alternate;

2. The NSC Director or designated alternate shall be notified and corrective action taken with respect to the operations involved;

3. The NSC Director or designated alternate shall notify the RSB Chairman who shall arrange for a review by the RSB;

4. A report shall be made to the RSB which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence; and

5. A report shall be made to the U.S. NRC in accordance with Section 6.7.2 of these specifications.

TS 6.2.2, Specification 1 through Specification 5, help ensure that actions are taken if a reportable occurrence other than a SL violation occurs. TS 6.6.2 helps ensure that the reactor is returned to normal operation or shut down, the NSC Director is notified and corrective actions are taken, the RSB Chairman is notified to arrange a review by the RSB, and the NRC is notified within 24 hours by telephone and by written report within 14 days. The NRC staff finds that TS 6.6.2 provides acceptable notification of the NSC Director and the NRC, provides for a review by the RSB, provides corrective actions, and is consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.6.2, Specification 1 through Specification 5, are acceptable.

5.6.7 TS 6.7 Reports

5.6.7.1 TS 6.7.1 Annual Operating Report

TS 6.7.1 states the following:

An annual report covering the operation of the reactor facility during the previous calendar year shall be submitted to the NRC before March 31 of each year providing the following information:

1. A brief narrative summary of (1) reactor operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

2. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

3. The number of unscheduled shutdowns and inadvertent scrams, including where applicable, corrective action to preclude recurrence;

4. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

5. A brief description, including a summary of the safety evaluations of changes in the facility or in procedures and of tests and experiments carried out pursuant to Section 50.59 of 10 CFR Part 50;

(…)

7. A summary of radiation exposures received by facility personnel and visitors, including dates and time where such exposures are greater than 25% of that allowed or recommended; and

8. A description and summary of any environmental surveys performed outside the facility.

TS 6.7.1, Specifications 1 through 5, and Specifications 7 and 8, help ensure that adequate annual reporting information is maintained. TS 6.7.1, Specifications 1 through 5, and Specifications 7 and 8, provide requirements for the status of the facility, major changes, radiation exposures, and other pertinent information to be provided to the NRC. The NRC staff finds that TS 6.7.1, Specifications 1 through 5, and Specifications 7 and 8, provide NSC annual operating report requirements that are consistent with guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.7.1, Specifications 1 through 5, and Specifications 7 and 8, are acceptable. TS 6.7.1, Specification 6, is evaluated and found to be acceptable in Section 3.2.1.1 of this SER.

5.6.7.2 TS 6.7.2 Special Reports

TS 6.7.2 states the following:

> In addition to the requirements of applicable regulations, reports shall be made to the NRC Document Control Desk and special telephone reports of events should be made to the Operations Center as follows:
>
> 1. There shall be a report not later than the following working day by telephone and confirmed in writing by fax or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report that describes the circumstances of the event and sent within 14 days to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555, of any of the following:
>
>   a. Violation of safety limits (see TS 6.6.1);
>
>   b. Any release of radioactivity from the site above allowed limits; and

    c. Any reportable occurrences as defined in TS 1.3.

2. A written report within 30 days to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC, 20555, of:

    a. Permanent changes in the facility organization involving Level 1 and Level 2; and

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

TS 6.7.2, Specification 1 and Specification 2, help ensure that special reporting requirements are adequately delineated.

The NRC staff reviewed TS 6.7.2, Specification 1 and Specification 2, and finds that the special report requirements are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.7.2 is acceptable.

5.6.8 TS 6.8 Records

TS 6.8 states the following:

> Records of facility operations in the form of logs, data sheets, or other suitable forms shall be retained for the period indicated as follows:

TS 6.8.1 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less Than Five Years

1. Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which 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, and

9. Records of meeting and audit reports of the RSB.

TS 6.8.2 Records to be Retained for at Least One Certification Cycle

Records of retraining and requalification of licensed operations personnel shall be maintained at all times the individual is employed or until the license is renewed.

TS 6.8.3 Records to be Retained for the Lifetime of the Reactor Facility

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.

5. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

TS 6.8.1, Specification 1 through Specification 9, help ensure that record retention requirements for records to be retained for at least 5 years or for the life of the component involved if less than 5 years, are properly delineated in the TSs. The NRC staff modified the title of TS 6.8.1, with the addition of “if less than 5 years,” to clarify the record requirements for components. By telephone conversation with the licensee, the licensee did not object to the change. The NRC staff finds that the record requirements in TS 6.8.1, Specifications 1 through 9, are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1 2007. On the basis of its review of the information provided above, the NRC staff concludes that TS 6.8.1 is acceptable.

TS 6.8.2 helps ensure that the licensee maintains training records of certified operators while employed or until the certification is renewed. The NRC staff reviewed TS 6.8.2, and finds that the record retention requirements stated in TS 6.8.2 are consistent with the guidance in NUREG-1537 and ANSI/ANS 15.1 2007. On the basis of the information provided above, the NRC staff concludes that TS 6.8.2 is acceptable.

TS 6.8.3, Specifications 1 through 5, help ensure that NSC facility records retention requirements for records that need to be retained for the lifetime of the NSC facility are appropriately delineated. The NRC staff reviewed TS 6.8.3, and finds that TS 6.8.3, Specifications 1 through 5, for record retention requirements are consistent with the guidance in NUREG-1537 and ANSI/ANS-15.1-2007. On the basis of the information provided above, the NRC staff concludes that TS 6.8.3 is acceptable.

5.7 Conclusions

The NRC staff reviewed and evaluated the proposed TSs as part of its review of the application for license renewal of Facility Operating License No. R-83, NRC Docket No. 50-128. The TSs define certain features, characteristics, and conditions governing the operation of the NSCR. The TSs are explicitly included in the renewed license as Appendix A. The NRC staff reviewed and evaluated the content of the TSs to determine whether the TSs meet the requirements in 10 CFR 50.36. Based on its review, the NRC staff concludes that the proposed TSs do 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 and ANSI/ANS-15.1-2007 and finds that the proposed TSs are consistent with these guidance. The NRC staff based this conclusion on the following findings:

- To satisfy the requirements of 10 CFR 50.36(a), the licensee provided proposed TS with the application for license renewal. As required by the regulations, the proposed TSs included appropriate summary bases for the TS. Those summary bases included in the TSs but are not specifications required by the regulations.

- The NSCR is a facility of the type described in 10 CFR 50.21(c), and therefore, as required by 10 CFR 50.36(b), the facility operating license will include TSs. To satisfy the requirements of 10 CFR 50.36(b), the licensee provided proposed TSs derived from analyses in the SAR, as supplemented.

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

- To satisfy the requirements of 10 CFR 50.36(c)(1), the licensee provided proposed TSs specifying a SL on the fuel temperature and a LSSS for the reactor protection system to preclude reaching the SL.

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

- The proposed TSs contain surveillance requirements 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 of 10 CFR 50.36(c)(5). The NSC's administrative controls contain requirements for initial notification, written reports, and records that meet the requirements specified in 10 CFR 50.36(c)(1), (2), (7), and (8).

The NRC staff reviewed the proposed TSs and finds the proposed TSs acceptable and concludes that normal operation of the NSCR within the limits of the proposed TSs will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or for the worker's occupational exposures. The NRC staff concludes that the proposed TSs provide reasonable assurance that the NSCR will be operated as analyzed in the SAR, as supplemented, and that adherence to the proposed TSs will limit the likelihood of malfunctions and the potential accident scenarios discussed in Section 4 of this SER.

6 CONCLUSIONS

On the basis of its evaluation of the application for license renewal as discussed in the previous chapters of this SER, the NRC staff concludes the following:

- The application for license renewal dated February 27, 2003, as supplemented by letters dated July 22, 2009; July 28, August 30, August 31, and December 9, 2010; May 27, June 9, and November 21, 2011; January 12, April 11, and November 14, 2012; January 31, 2013; February 3, February 11, and November 13, 2014; and March 2, June 5, June 11, and June 30, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations*.

- The facility will operate in conformity with the application, as supplemented, and with the provisions of the Atomic Energy Act of 1954, as amended, 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 licensee is technically and financially qualified to engage in the activities authorized by the renewed license in accordance with the rules and regulations of the Commission.

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

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