Enclosure 2

Safety Evaluation Report

Oregon State University

Docket No. 50-243

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Safety Evaluation Report Related to the Renewal of

the Operating License for the TRIGA Reactor at the

Oregon State University

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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 staff conducted this review in response to a timely application filed by the Oregon State University (the licensee) for a 20-year renewal of Facility Operating License R-106 to continue to operate the Oregon State University TRIGA reactor. In its safety review, the staff considered information submitted by the licensee (including past operating history recorded in the licensee's annual reports to the NRC), as well as inspection reports prepared by NRC personnel and first-hand observations. On the basis of this review, the staff concludes that the Oregon State University can continue to operate the facility for the term of the renewed license, in accordance with the renewed facility license, without endangering the health and safety of the public, facility personnel, or the environment.

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1. INTRODUCTION

1.1 Overview

By letter and supporting documentation dated October 5, 2004, as supplemented, Oregon State University (OSU or the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC or the Commission) an application for a 20-year renewal of the Class 104c Facility Operating License No. R-106 (NRC Docket No. 50-243) for the Oregon State University TRIGA reactor (OSTR or the facility).

The regulations in Title 10, Section 50.51(a) of the Code of Federal Regulations (10 CFR 50.51(a)) state that each license will be issued for a period of time to be specified in the license but in no case to exceed 40 years from date-of-issuance. The OSU facility license was issued on August 15, 1966, for a period of 40 years expiring on August 15, 2006. A renewal would authorize continued operation by issuance of a renewed license for the TRIGA-type research reactor facility. The facility is located in the Radiation Center on the OSU campus in Corvallis, Oregon. Until the staff completes action on the renewal request, the licensee is permitted to continue operation of the OSTR under the terms and conditions of the exiting License in accordance with 10 CFR 2.109, "Effect of Timely Renewal Application."

The NRC staff's (the staff) review, with respect to renewing the OSTR operating license, was conducted on the basis of information contained in the renewal application as well as supporting supplements and licensee responses to requests for additional information (RAIs). Specifically, the renewal application included the safety analysis report (SAR), an environmental report, the operator requalification program, and Technical Specifications (TSs). The licensee also requested that the staff consider as part of the application the OSTR emergency plan and physical security plan previously filed with the NRC. The licensee has since updated these plans as part of the licensee's routine maintenance of the plans under 10 CFR 50.54(p) and 10 CFR 50.54(q). As part of the review, the staff also reviewed annual reports of facility operation submitted by the licensee and inspection reports prepared by NRC personnel. Several site visits were conducted at the facility to observe facility conditions.

With the exception of the physical security plan and emergency plan, this material may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. Documents related to this license renewal may be accessed through the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov. If you do not have access to ADAMS, or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov. The physical security plan is protected from public disclosure under 10 CFR 73.21, "Requirements for the Protection of Safeguards Information." The emergency plan is considered security-related information.

The dates and associated ADAMS accession numbers of the licensee's renewal application and associated supplements are listed in Chapter 18, "References."

In conducting its safety review, the staff evaluated the facility against the requirements of 10 CFR Parts 20, 30, 50, 51, 55, 70, and 73; applicable regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The staff also referred to the guidance contained in

NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996. Because there are no specific accident-related regulations for research reactors, the staff compared calculated dose values for accidents against the requirements in 10 CFR Part 20, "Standards for Protection against Radiation" (i.e., the standards for protecting employees and the public against radiation).

The purpose of this safety evaluation report (SER) is to summarize the findings of the safety review of the OSTR and to delineate the technical details considered in evaluating the radiological safety aspects of continued operation. This SER provides the basis for renewing the license for operation of the OSTR at thermal power levels up to and including 1.1 megawatt (MW(t)), and short duration power pulses with reactivity insertions not to exceed $2.55.

This SER was prepared by Alexander Adams, Jr., Senior Project Manager, and Cynthia Montgomery, Project Manager, from the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Policy and Rulemaking, Research and Test Reactors Branch A. Brookhaven National Laboratory, the NRC's contractor, provided substantial input to this SER.

1.2 Summary and Conclusions Regarding the Principal Safety Considerations

The staff's evaluation considered the information submitted by the licensee, including past operating history recorded in the licensee's annual reports to the NRC, as well as inspection reports prepared by the NRC staff. In addition, as part of its licensing review of several TRIGA reactors, the staff obtained laboratory studies and analyses of several accidents postulated for the TRIGA-type reactor. On the basis of this evaluation and resolution of the principal issues reviewed for the OSTR, the following staff findings were reached:

- The design of the reactor structure and systems and components important to safety during normal operation are inherently safe, and safe operation can reasonably be expected to continue.

- The expected consequences of a broad spectrum of postulated credible accidents and a maximum hypothetical accident (MHA) have been considered, emphasizing those that could lead to a loss of integrity of fuel element cladding. The licensee performed conservative analyses of the most serious credible accidents and the MHA and determined that the calculated potential radiation doses outside the reactor room would not exceed 10 CFR Part 20 doses for unrestricted areas.

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

- The systems provided for the control of radiological effluents can be operated 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 a high degree of assurance that the facility can be operated safely and reliably. There has been no significant degradation of equipment, and the TSs will continue to ensure that there will be no significant degradation of equipment.

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

- The licensee's program for providing for the physical protection of the facility and its special nuclear material complies with the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials."

- The licensee's procedures for training reactor operators and the plan for operator requalification are acceptable. These procedures give reasonable assurance that the reactor facility will be operated with competence.

- The licensee maintains an emergency plan in compliance with 10 CFR 50.54(q) and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which provides reasonable assurance that the licensee is prepared to assess and respond to emergency events.

On the basis of these findings, the staff concludes that OSU can continue to operate the OSTR, in accordance with the renewed license, without endangering the health and safety of the public, facility personnel, or the environment.

1.3 History

On August 16, 1966, the U.S. Atomic Energy Commission (AEC) issued to OSU Construction Permit No. CPRR-93 authorizing the construction of the OSTR. On March 7, 1967, the AEC issued Operating License No. R-106 to OSU for operation of the OSTR, a TRIGA-type research reactor authorized to operate at steady-state power levels up to 250 kilowatts thermal (kW(t)). The reactor reached initial criticality in March 1967. In August 1969, the licensed steady-state power level of the reactor was increased to 1 MW(t). Amendment No. 3 to the license, issued in July 1976, authorized the use of Fuel Lifetime Improvement Program (FLIP) fuel in the reactor. In 1989, the license was amended to allow operation up to its current power level of 1.1 MW(t). On November 6, 2007, the licensee submitted an application to convert the reactor from the use of high-enriched uranium (HEU) FLIP fuel to low-enriched fuel. The conversion is planned to become effective on September 30, 2008.

Modifications made to the facility since issuance of the facility license were technological upgrades to instrumentation or expansion of the radiation experimental facilities. Most licensing actions involved administrative changes and minor changes to the existing design that either enhanced capabilities or improved reactor operations. No substantive changes to the facility are being considered for this license renewal.

OSU also operated an AGN-201 research reactor with a power level of 0.1 watt thermal. This reactor reached initial criticality in January 1959 and was permanently shut down in December 1974. The core of the AGN-201 reactor remains at OSU and is possessed under the OSTR license.

1.4 Reactor Description

The OSTR is located in the OSU Radiation Center complex which consists of four connected buildings including the reactor building. Major nonnuclear experimental facilities in the complex include the advance plant experiment and the Advanced Thermal Hydraulics Research Laboratory. The OSTR is a pool reactor with a steady-state power of 1.1 MW(t) and a peak pulse power over 2000 MW(t). Water is used as the coolant. The neutron reflector in the radial direction is made up of graphite in an aluminum container. Graphite in the top and bottom of the fuel elements also acts as a reflector. Neutron moderation is primarily provided by the zirconium-hydride that is homogeneously combined with the uranium fuel. The reactor has a strong, prompt temperature coefficient of reactivity that limits reactor power in the event of a power excursion. The core is cooled by natural convection of water. The reactor coolant in the pool is circulated through an external heat removal and purification system. The reactor's experimental facilities include a pneumatic transfer system, the in-core irradiation tube (ICIT) and the cadmium-lined in-core irradiation tube (CLICIT) that can be placed in the core, four beam ports, a thermalizing column, a thermal column, and the rotating rack. Four control rods (three standard rods and one transient rod) are used to control the reactivity of the reactor core. The rods can be dropped by gravity into the core for safety purposes. The transient control rod can be quickly removed from the core by air pressure to allow the reactor to operate in a square-wave or pulse mode.

The OSTR is used for student training in reactor engineering theory and operation, nuclear research, and a range of irradiation services.

1.5 Shared Facilities and Equipment

The OSTR is an integral part of the Radiation Center at OSU. Utilities and facilities such as the main electrical supply, sewage, structural walls, and potable water are shared with other parts of the Radiation Center. Offices for reactor program personnel and some laboratories are located in the Radiation Center areas adjoining the reactor building. The reactor building ventilation system, electrical distribution, water distribution and heating are independent.

1.6 Comparison with Similar Facilities

Design of the OSTR is similar to other NRC-licensed TRIGA research reactors. The OSTR is very similar in design to the TRIGA reactors at the University of Texas at Austin and the U.S. Geological Survey Center in Denver, Colorado. Instruments and controls used in the OSTR are similar in principle to most research reactors licensed by the NRC.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 specifies that the NRC may require, as a precondition to issuing or renewing an operating license for a research 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 had determined that universities and other Government agencies operating nonpower reactors had entered into contracts with DOE, which provide that DOE retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing. By entering into contract DE-AC07-76ER01953 with DOE, OSU has satisfied the requirements of the Nuclear Waste Policy Act of 1982, as they apply to the OSTR.

2. SITE CHARACTERISTICS

2.1 Geography and Demography

The follow sections describe the geography of the OSTR site including the location of the OSTR and the demography of the site.

2.1.1 Geography

The OSTR is located in the reactor building of the Radiation Center on the far west end of the OSU campus in the city of Corvallis, Oregon. Corvallis is located approximately 50 miles from the Oregon coast and 35 miles from Salem in Benton County in the Willamette Valley. The OSTR is located approximately 400 yards northeast of Oak Creek and 1.5 miles west of the Willamette River. Good Samaritan Hospital, which can treat patients of radiological incidents, is located approximately 3.5 miles to the northeast. The reactor building is surrounded by a fence on exterior sides. The minimum distance from the outside of the reactor building to the fence (unrestricted environment) is 33 feet.

TSs 1.20 and 5.1a. contain features applicable to the OSU site, as described below:

1.20 Radiation Center Complex: The physical area defined by the Radiation Center Building and the fence surrounding the north, west, and east sides of the Reactor Building.

5.1.a. The restricted area is that area inside the fence surrounding the reactor building and the reactor building. The unrestricted area is that area outside the reactor building and the fence surrounding the reactor building.

Design feature TSs are defined in 10 CFR 50.36(c)(4) as those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety. TS design features need prior review and approval from the NRC to change. The restricted area defined above is a geometric arrangement that is used in calculations that could affect safety such as public doses from radiation. Because the site is clearly defined, TS 1.20 and 5.1.a are acceptable to the staff.

2.1.2 Demography

According to the U. S. Census Bureau 2000 data, Corvallis has a population of 49,332, which represents a 9-percent increase over the preceding 10-year period. This population resides mostly to the north-northeast and south of the OSTR. The total population of Benton County is 78,153 (2000 census data). The next nearest populated area to the OSTR is the city of Philomath, which is 5 miles away and has a population of 3838.

OSU has a total population (students and staff) of approximately 23,000. Sackett Hall and Reed Hall are the nearest OSU dormitories to the OSTR (approximately 1300 feet east), with a combined total capacity of approximately 360 persons. The Orchard Court housing complex, located about 880 feet to the north of the OSTR, provides housing for approximately 300 persons. The typical population for all on-campus residence halls is about 3,500 persons (between the months of September and June). The nearest occupied office building is about 330 feet to the north of the restricted area.

2.1.3 Conclusions

The licensee has provided a sufficiently detailed and accurate description of the geography surrounding the OSTR. The demographic information is sufficient to allow accurate assessments of the potential radiological impact on the public from the continued operation of the OSTR. Based on a review of this information, there is reasonable assurance that no geographic or demographic features render the OSTR unsuitable for continued reactor operation.

2.2 Nearby Industrial, Transportation, and Military Facilities

The licensee states that there are no refineries, chemical plants, mining facilities, manufacturing facilities, water transportation routes, fuel storage facilities, military facilities or railyards located near the OSTR.

Located on the OSU campus within the vicinity of the OSTR are the Environment Protection Agency's Western Ecology Division, the OSU Forestry Building, and the O.H. Hinsdale Wave Research Laboratory. These three facilities are located within a 1-block radius of the OSTR, and pose no threat to the continued operation of the OSTR by the nature of the facility or the activities performed.

The OSU Hazardous Waste Facility is located 100 feet to the north of the OSTR, and processes all hazardous waste (chemistry products, radioactive and medical wastes, and paint supplies) from the OSU laboratories for transportation to either a permanent storage location or destruction facility. Hazardous materials and radioactive materials are transferred to a waste broker. The facility has three bays connected by sealed fire doors. Each bay has monitored and alarmed air vents, floor drains which empty to a below-grade tank, and interior and exterior fire sprinklers. The mixing station also is equipped with a blow-out wall which faces east, away from the OSTR. The licensee states that the facility does not pose a significant threat of accident to the OSTR because of its design, distance from the reactor, the small quantity of materials stored, and the building security. In addition, the staff notes that the reactor building and biological shield are intervening between the OSU Hazardous Waste Facility and the reactor core providing significant protection. The staff concludes that based on its design and material processed, the OSU Hazardous Waste Facility does not pose a significant risk to continued operation of the OSTR.

There are several regional airfields in the vicinity. The Corvallis Municipal Airport is located approximately 6 miles south of the OSTR. This small airport does not have a control tower and has two runways with a weight limit for double-tandem wheeled aircraft of 75 tons. The licensee states that this airport averages 106 aircraft operations daily, and has 142 aircraft based there, 115 of which are single-engine aircraft. Mahlon Sweet Field Airport is located approximately 35 miles from the OSTR in Eugene, Oregon. This airport averages 263 aircraft operations daily, with a weight limit of 150 tons for double-tandem wheeled aircraft. This is a controlled airport with four runways, and has 173 aircraft based there, of which 131 are single-engine aircraft. The major airport servicing the area is the Portland International Airport located 80 miles to the north of the OSTR. Because of the distance of these airports from the reactor site, the staff concludes that there is no significant risk to continued operation of the OSTR.

There is also a rail line located approximately 1200 feet south of the Radiation Center (the OSTR is located on the north side of the Radiation Center). This rail line is used almost exclusively to transport lumber. Given the distance to the rail line, the cargo carried, the intervening buildings between the rail line and the reactor core, and the reactor biological shield, the staff concludes that there is no significant risk from the rail line to continued operation of the OSTR.

The licensee has discussed all nearby manmade facilities and activities that could pose a hazard to continued operation of the OSTR. Based on a review of these facilities and operations conducted there, the staff concludes that there is reasonable assurance that normal operations at these facilities will not affect continued operation of the OSTR.

2.3 Meteorology

The climatology of the licensee's site is described in the following sections. This includes information on precipitation, winds, and temperature. The sources of meteorological data to be used in case of an emergency is also discussed.

2.3.1 Climatology

The overall climate at the OSTR site is considered to be mild, generally characterized by cool, wet winters and warm, dry summers. A rain shadow, created over the Willamette Valley by the Coastal Range, limits rainfall amounts and coastal winds. Relative humidity is highest in the mornings (80—100 percent). It then typically decreases in the afternoons.

The normal annual precipitation (primarily rain) for the Corvallis area is 43.6 inches with an annual maximum of 73.2 inches in 1996 and a minimum of 27.2 inches in 1985. On a monthly basis, rainfall amounts range from 0.6 inches in July to 7.4 inches in December.

Average wind data from National Weather Service stations in Eugene (45 miles to the south) and Salem (35 miles to the northeast) show that, for the period 1961 to 1990, the average wind speed was 8.5 miles per hour (mph) primarily from the north or south directions. Based on a historical survey of temperatures (1971—2000), monthly normal temperatures range from 82.4 °F in August to 33.6 °F in January. Daily extremes over this time period varied from a high of 108 °F to a low of -7 °F.

Hurricanes and tornadoes are virtually nonexistent in the area surrounding the OSTR. Between 1887 and 1996, there have been two tornadoes in Oregon which measured F2 on the Fujita scale (wind-speeds of 113 to 157 mph). During this same time period, 69 tornadoes were reported for the entire State of Oregon, most classified as F0 on the Fujita scale (wind speeds less than 72 mph). Only eight of these occurred within a 50-mile radius of the OSTR. Historically, hurricanes rarely affect the Oregon coast. In the unlikely event one did, the inland location of the OSTR would minimize any wind effects.

2.3.2 Sources of Meteorological Data for Emergencies

To collect meteorological data in the event of emergency situations, a wind speed and direction anemometer are installed on the reactor building. The data for the anemometer are collected by a data logger and recorded on a computer hard drive. The data are time stamped so for any release, the wind speed and direction data can be downloaded and averaged over a given time period. These data are entered in the licensee's Gaussian plume model calculations used to calculate doses.

2.3.3 Conclusions

Based on a review of the historical data submitted by the licensee, it is concluded that the meteorological information provided for the region around the OSTR is sufficiently documented. The licensee also has sufficient procedures for collecting meteorological information to be used during a facility emergency. The information is sufficient to support dispersion analyses for postulated airborne releases which may occur during operations, including emergency situations. The staff concludes that there are no meteorological-related events of credible frequency or consequences at the OSTR site that would render it unsuitable for continued operation.

2.4 Hydrology

There are two waterways near the OSTR, Oak Creek (1200 feet from the OSTR) and Mary's River (5000 feet from the OSTR). Oak Creek flows into Mary's River which in turn flows into the Willamette River on the east side of Corvallis. The licensee notes that the OSTR is not located in the 100-year flood plain for either waterway.

There are no lakes or dams near the OSTR. Therefore, seismically induced flooding due to a dam failure or seiches is not a risk for the OSTR. With the location of the OSTR 50 miles inland from the Oregon coast, tsunamis are not considered a credible risk. Since the OSTR is not located in any 100-year flood plain or near lakes or dams and it is 50 miles from the coastline, there is very little risk that hydrology issues will significantly impact the safety at OSTR.

Based upon a review of the information, it is concluded that there are no credible hydrologic events which would affect the safe operation of the OSTR during the renewal period.

2.5 Geology, Seismology, and Geotechnical Engineering

Two primary tectonic plates, the Juan de Fuca Plate (located off the Oregon coast) and the North American Plate (located under the state of Oregon), primarily define western Oregon geology. The former is being subducted beneath the North American Plate, and the subduction zone runs south from British Columbia to northern California. No event has occurred in this zone during the 150 years of recorded activity.

The licensee reports that there are two major fault zones within 5 miles of the OSTR. The Corvallis Fault is 1.5 miles northwest and is approximately 34 miles long running in the northeast direction with the most recent detectable movement occurring before 28,500 years ago. The Owl Creek Fault is 3 miles east of Corvallis and is approximately 9 miles long and shows movement between 10,000 and 30,000 years ago.

Since 1956, there have been seven seismic events within a 50-mile radius of Corvallis, with magnitudes ranging from 3.5 to 5.6 on the Richter scale. Other distant earthquakes were felt in Corvallis, none of which resulted in any damage within the city. These were assigned values between I and V on the Mercalli Intensity Scale. The Mercalli Intensity Scale is a subjective measure of intensity ranging from I (corresponding to the event not being felt) to VII (corresponding to buildings nearly destroyed).

The primary geological structure near Corvallis is the Cascade Range, which runs north/south in central Oregon and extends from northern California to Washington. Several volcanoes are included in this general region—Mount Hood, Mount Jefferson, Three Sisters, Newberry, Crater

Lake, and Mt. St. Helens. A review of U.S. Geological Survey volcano and earthquake hazard reports indicate that, with the exception of Mt. St. Helens, the last eruptions occurred hundreds of years ago, and none were considered to have a high probability of eruption in the foreseeable future. As of 1999, the largest earthquake associated with these was at Mt. Hood with a magnitude of 4.0. The OSTR is located a sufficient distance from Mt. St. Helens (60 miles) such that future eruptions would not be expected to affect the safe operation of or the ability to shutdown the OSTR.

The licensee performed a peak bedrock acceleration analysis for the OSTR site in 2003. Estimated bedrock accelerations were given for earthquakes with 10-percent, 5-percent, and 2-percent probability of being exceeded in 50 years, and correspond to approximate return periods of 500, 1000, and 2500 years. The maximum accelerations determined were 0.18g (500-year return), 0.26 g (1000-year return), and 0.38g (2500-year return).

The licensee reports that no known faults exist below the OSTR site. The potential for ground rupture and liquefaction is considered to be very small.

The staff concludes that the licensee has provided sufficient information about geologic features and potential seismic activity at the OSTR site. Seismic activity has not been significant in the vicinity of the OSTR and local faults have not shown movement for at least 10,000 years. A review of the credible accident scenarios (Chapters 3 and 13 of the SAR) confirm that, in the unlikely event of seismic damage, the radiological consequences would not exceed analyzed conditions.

2.6 Conclusions

The staff concludes that the reactor site has experienced no significant geographical, meterological, or geological change since initial licensing, and therefore, the site remains suitable for continued operation of the OSTR. Infrequency of the occurrence of tornadoes and earthquakes and the robustness of the facility continue to make the site suitable for operation of the OSTR. Hazards related to industrial, transportation, and military facilities pose no significant risk to the continued safe operation of the OSTR. The demographics of the area surrounding the reactor have not changed in any way that significantly increases the risk to public health and safety from continued operation of the OSTR.

3. DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The OSTR is housed in a four-story Reactor Building. The reactor is a pool-type reactor with a current maximum operating power level of 1.1 MW(t). The reactor core sits on a support structure in a pool surrounded by a biological shield structure. The reactor instrumentation and control (I&C) system allows operation of the control rod drives and control rods to control the reactivity of the reactor. The I&C system is also a safety system that monitors key reactor parameters and can quickly shut down the reactor by scram if predetermined parameter limits are exceeded. The reactor core is cooled by natural convection of the water through the core. A cooling system removes heat from the reactor pool to the environment. The ventilation system controls the movement of air through the facility to a defined monitored path. The system can automatically shut down if predetermined radiological conditions are met. A number of experimental facilities exist to facilitate the conduct of research, education, and service.

TS 5.1b. contains the design features applicable to the reactor building.

> The reactor building houses the TRIGA reactor and is abutted to the OSU Radiation Center Building.

The purpose of this TS is to require prior NRC review and approval before a change is made to the basic arrangement of the reactor facility. Changing this design feature could impact dose calculations and basic assumptions of the safety analysis. Because these important design features are controlled, TS 5.1.b. is acceptable to the staff.

The original reactor installation in 1966—67 used components manufactured by General Atomics (GA), the reactor designer. GA also provided the specifications to which structures were built. The OSTR was also designed and constructed in accordance with Uniform Building Code (UBC 1964) Zone 3 seismic requirements. Later modifications and equipment additions, such as the installation of upgraded cooling capacity in 1971, were made in conformance with the local building codes in existence at the time.

When the reactor was upgraded to a power level of 1 MW(t) in 1976, the principal design criterion was to ensure that the facility could withstand loss of pool water and any other credible accident with no hazard to the public, without reliance on engineered safety systems. The licensee stated that the criterion was met by the use of stainless-steel clad TRIGA fuel. The fuel has a prompt negative temperature coefficient of reactivity which allows the reactor to withstand large additions of reactivity without fuel damage. The fuel design and reactor operation are such that the TRIGA fuel can be adequately cooled by natural convection of coolant through the reactor core.

The MHA, discussed in Chapter 13, occurs when the fuel cladding is assumed to be removed from a fuel element. The accident assumes that the fuel element is in air, not the reactor pool, and no reactor building exists. Without any reliance on engineered safety systems, the estimated doses to workers in the facility and the general public are within the 10 CFR Part 20 annual limits. The loss-of-coolant accident, also discussed in Chapter 13, shows that an emergency core cooling system is not needed to maintain fuel element cladding integrity.

TS 4.0 controls surveillance requirements and changes to certain system designs as follows:

a. Surveillance requirements may be deferred during reactor shutdown (except Technical Specifications 4.3.a and 4.3.e); 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 practicable after reactor startup. Scheduled surveillance, which cannot be performed with the reactor operating, may be deferred until a planned reactor shutdown.

b. Any additions, modifications, or maintenance to the ventilation 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 reviewed by the Reactor Operations Committee. A system shall not be considered operable until after it is successfully tested.

This TS requires that changes to certain important systems are controlled to their original design and fabrication specifications or, if to new specifications, that those specifications have been reviewed. The TS also governs the conduct of surveillance required to allow operational flexibility that does not impact safety. Because the TS maintains rigor in the design control process, it is acceptable to the staff.

The design criteria are based on applicable standards, codes, and criteria, and provide reasonable assurance that the facility structures, systems, and components have been built and will function as designed and required by the analyses in the SAR. The licensee has a TS-required process to control design aspects of the facility. Hence, the design criteria provide reasonable assurance that the public will be protected from radiological risks resulting from operation of the reactor facility.

3.2 Meteorological Damage

Tornadoes and hurricanes are relatively infrequent for the area due to the mild climate (see Chapter 2 of this SER). The OSTR is located in the reactor building, which was constructed to local codes to withstand local wind, rain, snow and ice loads. The OSTR is also protected by the biological shield which is a thick reinforced concrete structure surrounding the reactor tank. Given the meteorological data for the OSTR site and the location of the reactor within the biological shield and reactor building, the staff concludes that significant meteorological damage is unlikely.

3.3 Water Damage

As discussed in Chapter 2 of this SER, the OSTR facility is located outside the projected 100-year flood plains. There are no lakes or dams nearby. The reactor building is designed for anticipated rain or snow loads. In addition, the OSTR is also protected by the biological shield which is a thick reinforced concrete structure surrounding the reactor tank. Therefore, the staff concludes that there is reasonable assurance that the probability of potential damage to the reactor from water damage is small.

3.4 Seismic Damage

The licensee has concluded that the likelihood of a significant seismic event at the site is low. The licensee also concludes that the potential for ground rupture at the site is considered low due to the lack of known faulting below the site and that soil liquefaction is not a significant concern due to the stiff, cohesive soil profile. The 1964 Uniform Building Code (UBC 1964) designated the area of the OSTR as Seismic Zone 2 (zones range from 0 to 3 with Zone 3 of highest seismic concern). However, to be conservative, the OSTR was designed and constructed in accordance with UBC 1964 Zone 3 seismic intensity requirements. A licensee consultant examined UBC 1964, current estimates of peak ground acceleration, and the 2006 International Building Code and determined that the facility conforms to current seismic design level standards. The AEC staff stated in the original 1966 evaluation of the site for the issuance of the construction permit that the site does not lie in a major earthquake zone; nevertheless, the reactor building has been designed in accordance with recommendations of the Uniform Building Code for areas of major seismic activity. The staff determined that the site was suitable for the proposed reactor. There is no evidence to suggest that the bases for the staff's conclusion have changed.

In the unlikely event of an earthquake that causes damage to the reactor, the staff concludes that the radiological effects would be bounded by the licensee's MHA analysis. As discussed in Chapter 13 of the SER, the estimated MHA doses to workers in the facility and the public in unrestricted areas are within the 10 CFR Part 20 annual dose limits.

3.5 Systems and Components

Although not identified as engineered safety features, the control rod drive assemblies, the OSTR reactor building ventilation system, and the reactor building confinement features have been identified by the licensee as part of the safety system and components important for safe operation of the facility. The control rod drive assembly for all control rods are mounted on the reactor bridge structure, which has been performing its intended function as designed for many years. The key components of the system are accessible for inspection and testing. Important design aspects are controlled in the TSs. The reactor control system is subject to TS-required surveillance and is part of the licensee's maintenance program. The reactor control system is discussed further in Chapters 4 and 7 of this SER.

The reactor building and the ventilation system taken together act as a confinement system. The ventilation system reduces the consequences of releases from postulated accidents and controls the release of routine effluents. The system provides a controlled pathway for release of air from the reactor building and automatically shuts down in emergencies. The reactor building serves as a barrier to release of effluents and allows a controlled pathway for release with the ventilation system. All doors are normally closed during operation which, in conjunction with the ventilation system, maintains proper ventilation airflow. Important design aspects are controlled in the TSs. The ventilation system is subject to TS-required surveillance and is part of the licensee's maintenance program. The ventilation system and reactor building are discussed further in Chapter 9 of this SER.

The effectiveness of the preventive maintenance program and TS-required surveillance is attested to by the small number and types of malfunctions of these systems over the years of operation. Review of inspection reports and reports of reportable occurrences has shown that malfunctions, when they have occurred, have generally been one of a kind and/or involved components that were fail-safe or self annunciating. Therefore, the staff concludes that there appears to be no significant uncompensated deterioration of this equipment with time or operation. Thus, there is reasonable assurance that this equipment will perform as designed and the health and safety of the public will be protected for the period of license renewal.

3.6 Conclusions

On the basis of the above considerations, the staff concludes that the OSTR is adequately designed and built to withstand all credible and probable wind, water, and seismic events associated with the site. The design and performance of structures, systems, and components have been verified through many years of operation. Accordingly, the staff concludes that the reactor systems and components are adequate to provide reasonable assurance that continued operation will not cause significant radiological risk to the health and safety of the public.

4. REACTOR DESCRIPTION

4.1 Summary Description

The OSTR is a natural convection, water-cooled, TRIGA pool reactor that uses uranium-zirconium hydride fuel elements in a circular grid array. The reactor core is surrounded by a graphite reflector. The OSTR has a number of different irradiation facilities including a pneumatic transfer tube, a rotating rack, a thermal column, a thermalizing column, four beam ports (three radial and one tangential), five sample-holding (dummy) fuel elements for special in-core irradiations, an ICIT, and a CLICIT for experiments requiring a high energy neutron flux. The OSTR also has an argon Irradiation Facility for the production of argon-41.

The reactor core is located near the bottom of a water-filled aluminum tank. For personnel shielding, the tank is shielded radially by ordinary concrete, water, lead, and graphite. Approximately 16 feet of water above the reactor core provides shielding at the top of the reactor tank. Four control rod drives (three motor-driven and one pneumatic electro-mechanical driven) are mounted at the top of the tank on a bridge structure spanning the tank.

The OSTR can be operated at a maximum steady-state power of 1.1 MW(t), or pulsed up to a peak power of about 2500 MW(t), or operated in a square-wave mode (in square-wave mode the reactor power is quickly raised using the pulsing transient rod to a power level within the steady-state power limit). The OSTR is fueled with FLIP HEU TRIGA fuel rods. As of the date of this SER, the NRC staff is reviewing an application from the licensee to convert to low-enriched uranium fuel and to eliminate the use of HEU fuel elements in the reactor. The conversion is planned to become effective on September 30, 2008.

The inherent safety of TRIGA reactors has been demonstrated by the extensive experience gained from similar designs used throughout the world. The fuel is characterized by inherent safety, high fission product retention, and the ability to safely withstand water quenching at temperatures as high as 1150 °C. The safety of the fuel arises from the strongly negative prompt temperature coefficient characteristic of uranium-zirconium hydride fuel-moderator elements. As the fuel temperature rises, this coefficient immediately compensates for reactivity insertions. The OSTR maximum fuel temperature safety limit (TS 2.1) is specified not to exceed 1150 °C under any mode of operation. To ensure that this safety limit is not exceeded, TS 2.2, establishes the limiting safety system setting (LSSS) for fuel temperature, to be equal to or less than 510 °C as measured in the instrumented fuel element (IFE).

4.2 Reactor Core

The OSTR utilizes solid fuel elements in which the zirconium-hydride moderator is homogeneously combined with enriched uranium fuel ($U\text{-}ZrH_x$ fuel). The reactor core is a lattice of cylindrical stainless-steel clad $U\text{-}ZrH_{1.6}$ fuel-moderator elements and aluminum clad graphite dummy elements. (The 1.6 in the $U\text{-}ZrH_{1.6}$ indicates that there are 1.6 hydrogen atoms for each zirconium atom. This ratio is important because it influences fuel behavior.) Approximately one-third of the core volume is water. Neutron reflection in the radial direction is provided by 10.2 inches of graphite in an aluminum container that is 22 inches high. At the outer perimeter of this container is a thermal shield, which consists of 2 inches of lead.

Reactor core components are contained between top and bottom aluminum grid plates. The top grid plate has 126 positions for fuel elements and control rods arranged in six concentric rings around a central thimble, which can be used for high flux irradiations. The OSTR uses four control rods to control power level—a motor-driven regulating rod, shim rod, and safety rod and a pneumatic electro-mechanical pulsing transient rod.

Four instrumentation channels monitor reactor neutron flux and power level. The primary reactor water is routinely monitored to identify any significant radioactivity increase. The reactor core is cooled by natural convection of the demineralized water in the reactor pool. A diffuser nozzle on the reactor tank inlet provides high-velocity water discharge above the core. This water circulation pattern reduces the nitrogen-16 dose rate from the coolant water.

TS 1.18 defines an operational core in the following manner:

> An operational core shall be a fuel element core which operates within the licensed power level and satisfies all the requirements of the Technical Specifications.

Various reactor core configurations were analyzed in the SAR, but not all represent cores that were operational in the OSTR. There are six possible operable core configurations discussed in the SAR, ranging from all FLIP fuel elements to a combination of standard and FLIP fuel elements. The current operational core configuration, as limited by the TSs, uses all FLIP fuel and is based on core number 8 in the SAR (see Section 4.5.1 below for a further discussion of fuel usage). The licensee can also use variations on these cores. Depending on the core chosen, the number of fuel elements varies from 55 to 85 (there are fuel followers on three of the control rods). The current core has 77 fuel elements and 3 fuel followers.

4.2.1 Reactor Fuel

The development and use of TRIGA U-ZrH fuel began in 1957, and over 6000 fuel elements of seven distinct types have been fabricated for use in 60 TRIGA research reactors both domestically and abroad. TRIGA fuel has 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 range of temperatures. The calculated core lifetime for FLIP fuel is approximately 3500 MW-days. Over 25,000 pulses have been performed with TRIGA fuel elements with temperatures reaching peaks of about 1150 °C.

The active part of the OSTR fuel element is approximately 1.5 inches in diameter and 15 inches long. The fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy with a maximum of 9 weight-percent of uranium enriched to a nominal 70-percent uranium-235. The hydrogen-to-zirconium atom ratio is between 1.5 to 1.65. A 0.19 inch diameter hole is drilled through the center of the active fuel section to facilitate hydriding, and a zirconium rod is inserted in this hole once hydriding is complete. The OSTR uses 1.58 weight-percent erbium as a burnable absorber to increase the core lifetime, control core reactivity and help ensure the large negative prompt temperature coefficient. The erbium is incorporated (by melting) into the fuel during fabrication.

The fuel element has a 304 stainless-steel clad thickness of 0.020 inch, and all closures are made by heliarc welding. Two 3.5-inch sections of graphite are placed above and below the fuel to serve as top and bottom reflectors for the core. Stainless-steel end plugs are attached to both ends with a lower alignment pin and upper handling pin.

A lower end fixture supports the fuel-moderator element on the bottom grid. The upper end fixture has a knob for attachment to the fuel-handling tool and a triangular spacer, which permits coolant to flow through the upper grid plate.

The OSTR uses an IFE, which is identical to the standard 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 of the fuel section, and 1 inch above and below the horizontal center. The IFE allows the licensee to directly measure the temperature of the fuel. The IFE is currently in core position B-4.

TSs 1.12 and 1.13 contain definitions for fuel elements as follows:

> 1.12 Fuel Element: A fuel element is a single TRIGA® fuel rod.
>
> 1.13 Instrumented Element: An instrumented element is a special fuel element in which one or more thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

The design features of the fuel are given in TS 5.3.3 as follows:

> The individual fuel elements shall have the following characteristics:
>
> 1. Uranium content: maximum of 9 wt% enriched to a nominal 70% ^{235}U;
> 2. Hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.5 and 1.65;
> 3. Natural erbium content (homogeneously distributed): between 1.1 and 1.6 wt%;
> 4. Cladding: 304 stainless steel, nominal 0.020 in thick; and
> 5. Identification: top pieces of fuel elements will have characteristic markings to allow visual identification of elements.

These TSs control the important aspects of the design of the fuel used in the OSTR SAR and as such are acceptable to the staff.

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

The requirements for fuel performance are given in TS 3.1.6 as follows:

> The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and must be removed from the core if:
>
> a. The transverse bend exceeds 0.0625 inches over the length of the cladding;
>
> b. Its length exceeds its original length by 0.125 inches;

c. A cladding defect exists as indicated by release of fission products; or

d. Visual inspection identifies bulges, gross pitting, or corrosion.

The license has proposed a new surveillance requirement given in TS 4.1.e. as follows:

> Twenty percent of the fuel elements comprising the core shall be inspected visually for damage or deterioration and measured for concentric or other swelling annually such that the entire core is inspected over a five year period. Annual inspections shall be of non-repeating representative samples of fuel elements from each ring.

The licensee has proposed inspecting 20 percent of the fuel elements in the core annually such that all of the fuel elements in the core are inspected over a 5-year period. Non-repeating representative samples of fuel from each ring of the core would be inspected to ensure that a cross section of fuel subject to all core conditions would be inspected annually. The inspection interval was $3500 of pulse reactivity. The licensee pulses infrequently such that it is possible that this old surveillance interval would never come due. The inspections are for visual damage or deterioration and concentric or other swelling. Experience has shown that swelling provides a leading indication of fuel condition. Swelling will become a problem before changes in bend or length.

The limits on transverse bend and length are based on values from GA, the reactor designer. There have been instances of fuel-cladding defects where fission products are only detected during reactor operation. Under these circumstances, the reactor needs to be operated to locate the fuel element with the cladding defect. All limits on radiation exposure need to be met during the search for a defective fuel element. The staff concludes that the licensee has used the standard definition of damaged fuel for TRIGA reactors and has a sufficient surveillance interval to help ensure fuel element integrity.

Fuel growth and deformation can occur during normal operations, as described in Simnad (1980). Damage mechanisms include fission recoils and fission gases, both of which are strongly influenced by thermal gradients. Swelling of the fuel is dependent on the amount of time the fuel spends over a temperature threshold. The threshold temperature for the phenomenon is about 750 °C. At the current steady-state-indicated IFE operating temperature of approximately 350 °C, which corresponds to a maximum fuel temperature of about 570 °C, swelling would be minimal, if present at all. While fuel temperatures could go above 750 °C during pulsing, the time at temperature is short such that pulsing should not cause fuel swelling by these mechanisms. Based upon the data presented by the licensee there is reasonable assurance that fuel swelling by the above mechanisms is precluded.

Based upon a review of the information provided by the licensee, the staff concludes that the licensee has adequately described the fuel elements used in the OSTR, including design limits, and the technological and safety-related bases for these limits. The constituents, materials, and components for the fuel elements have been discussed by the licensee in the SAR. Compliance with the TS limits will ensure uniform characteristics and compliance with design bases and safety-related requirements. There is reasonable assurance that the fuel will function safely in the OSTR core for the renewal period without adversely affecting public health and safety.

4.2.2 Control Rods

The OSTR uses three motor-driven control rods (one regulating, one shim, and one safety) and one pneumatic electro-mechanical transient rod to control reactor power. A control rod is defined in TS 1.7 as follows:

> A control rod is a device fabricated from neutron absorbing material or fuel or both which is used to establish neutron flux changes and to compensate for routine reactivity changes. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. Types of control rods shall include:
>
> a. Regulating Rod (Reg Rod): The regulating rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled-follower section. Its position may be varied manually or by the servo-controller.
>
> b. Shim/Safety Rod: A shim safety rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled-follower section.
>
> c. Transient Rod: The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It may have a voided-follower.

The transient rod, used for pulse mode operation, contains a 15 inch-long solid rod of boron-carbide-impregnated graphite as a neutron poison. This assembly is approximately 37 inches long and is clad in a 1.25-inch outside diameter aluminum tube. The transient rod has a 21 inch-long air-filled follower. It is guided laterally in the core by an aluminum guide tube that passes through the upper and lower grid plates, and is screwed into, and supported by, the aluminum safety plate beneath the lower grid.

The transient control rod drive is mounted on a steel frame bolted to the center channel cover plate. A system of limit switches indicates the position of the air cylinder and the transient rod. A connecting rod couples the transient rod to a piston rod assembly. Rotation of a ball-screw nut raises or lowers the cylinder. Compressed air is supplied to a normally closed port of a three-way air solenoid valve. When this valve is energized, air pressure is placed on the bottom of the piston, causing the piston to contact the hydraulic shock absorber in the top of the cylinder. The resulting reactivity insertion is a function of the initial position of the cylinder. With air applied, energizing the motor will cause the cylinder, piston, and control rod to move as a unit. Scram of the transient rod is accomplished by deenergizing the air solenoid valve.

The fuel-followed safety, shim, and regulating rods are guided by 1.5-inch diameter holes in the top and bottom grid plates. The rods are a sealed stainless-steel tube approximately 43 inches long by 1.375 inches in diameter. The upper portion of the rod is graphite and the next 15 inches is a graphite-impregnated-with-boron-carbide neutron absorber. The follower section consists of 15 inches of $U\text{-}ZrH_{1.6}$ fuel, and the bottom section has 6.5 inches of graphite. An aluminum safety plate attached to the shroud beneath the lower grid plate prevents any control rod from dropping out of the core in the event that it disconnects from the drive. Vertical travel of the control rods in the core is approximately 15 inches.

The safety, shim, and regulating rod drives are rack-and-pinion linear actuators. An electromagnet is secured to the bottom of the draw-tube to which the rack is mounted. The magnet is moved (up or down) in response to a rotating pinion shaft. When the magnet is energized, the armature is magnetically coupled to the draw-tube; deenergizing the magnet results in a rod drop. Rod position is sensed by a series of limit switches.

The TS design requirements for the control rods are given in TS 5.3.2 as follows:

a. The shim, safety, and regulating control rods shall have scram capability and contain borated graphite, B4C powder or boron, with its compounds in solid form as a poison, in aluminum or stainless steel cladding. These rods may incorporate fueled followers which have the same characteristics as the fuel region in which they are used.

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

These TSs control the important aspects of the design of the control rods used in the OSTR SAR and as such are acceptable to the staff.

The nominal rod worths of the transient, regulating, shim, and safety rods are $4.00, $2.70, $2.70, and $2.70, respectively. However, the worth of the rods are dependent on the design of the core being used. The total rod worth is approximately $12.10, and the total excess reactivity in the OSTR core does not normally exceed $7.50. TS 3.1.3 requires that the maximum available excess reactivity based on the reference core condition shall not exceed $7.55.

TS 3.1.2 requires that the shutdown margin by the control rods be greater than $0.55. This shutdown margin is determined at the start of daily operations, or upon commencement of continuous operations (i.e., any operational period which exceeds the usual daily operation period). The licensee projects that the lifetime of the control rods will extend beyond the period of renewal. The licensee has no plans to replace or store depleted control rods.

TS 3.2.1 contains the following requirements for control rod operability:

> The reactor shall not be operated unless the control rods are operable. Control rods shall not be considered operable if:
>
> a. Damage is apparent to the rod or rod drive assemblies; or
> b. The scram time exceeds 2 seconds.

The surveillance requirements for the control rods are given in TS 4.2 as follows:

a. The control rods and drives shall be visually inspected for damage or deterioration biennially.

b. The scram time shall be measured annually.

c. The transient rod drive cylinder and associated air supply system shall be inspected, cleaned and lubricated as necessary, semi-annually.

Scram time is defined by TS 1.28 as follows:

> Scram time is the elapsed time between reaching a limiting safety system set point and the instant that the slowest scrammable control rod reaches its fully-inserted position.

These TSs help to ensure that the control rods will promptly shut down the reactor upon a scram signal. The OSTR 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 2 seconds, and is measured annually by the licensee. The 2-second value for scram time is a margin far enough above normal scram times to allow for some variation in performance while bounding acceptable performance. The 2-second value is assumed in the safety analysis. Experience has shown that the surveillance intervals are sufficient to help ensure operability. Based on the discussion above, the staff finds these TSs acceptable.

Based on a review of the information provided by the licensee, the NRC staff concludes that the control rods conform with the applicable design bases and can effect a shutdown of the reactor from any operating condition. Based on this review, there is reasonable assurance that the scram features will perform as required to ensure fuel integrity and protect public health and safety for the renewal period. A review of the design and functional description of the transient rod system offers reasonable assurance that pulses will be reproducible and limited to values that maintain fuel integrity. The control rod design for the OSTR includes reactivity worths that can control the excess reactivity planned for the OSTR, including the assurance of an acceptable shutdown reactivity and margin. The licensee has justified appropriate design limits, limiting conditions for operations (LCOs), and surveillance requirements for the control rods.

4.2.3 Neutron Moderator and Reflector

The OSTR reflector is a ring-shaped block of graphite that radially surrounds the core. The graphite is 10.2 inches thick radially, with an inside diameter of 21.625 inches and a height of 21.8125 inches. A welded aluminum can protects the graphite. In 1987 there was leakage of water into the reflector can, which was reported to the NRC. Flooding of the reflector can decrease the effectiveness of the reflector which decreases the reactivity of the reactor. Flooding does not pose any problem to safely implementing shutdown of the reactor. The licensee plans to replace the reflector at some point in the future. The reflector can and other reactor components are visually inspected as part of daily startup and shutdown procedures. A reflector at another TRIGA reactor experienced leakage and swelling of the reflector due to the generation of gases within the reflector. Any dimensional distortion of the reflector would be noticed during core inspections. In response to a RAI from the NRC staff, the licensee committed to a more thorough inspection of the reflector can at the next opportunity. This is expected to occur in 2008 during the conversion of the reactor to low-enriched uranium fuel.

A well in the top of the reflector can is for the rotary specimen rack (“Lazy Susan”). This rotary specimen rack is self-contained and does not penetrate the sealed reflector (see Chapter 10 of this SER). A 2-inch-thick lead thermal shield is located on the reflector’s periphery. The lead is flame sprayed with a molybdenum coating on the inner surface of the aluminum can. This provides a good bond for the transfer of the heat deposited by the gamma rays in the lead to the surrounding water.

The graphite/lead/outer aluminum surfaces are pierced by an aluminum tube that forms the inner section of the piercing radial beam port. Two other holes penetrate only the graphite and lead; one for the radial beam port and the other for the tangential beam port.

As discussed by the licensee, the reflector is an integral part of the OSTR reactor core. The NRC staff concludes that there is reasonable assurance that the reflector as designed or in its current flooded state will not affect safe reactor operation, prevent a safe shutdown, or cause an uncontrolled release of radioactive material to the unrestricted environment.

4.2.4 Neutron Startup Source

The OSTR uses a 3-curie americium-beryllium or 7-curie polonium-beryllium neutron startup source. The source used is housed in a 0.981-inch-diameter by 3-inch-deep aluminum holder. The source holder can be located in any vacant fuel or graphite element location in the reactor core. A shoulder at the upper end of the source holder supports the assembly on the upper grid plate, with the source extending down into the core region.

One of the primary functions of a neutron source is to provide sufficient counts such that instrumentation will function properly during startup. As discussed in Chapter 7 of this SER, TS 3.2.3, Table 3, contains operational interlocks one of which is an interlock on control rod withdrawal if there is a count rate less than 2 counts per second.

The neutron source used at the OSTR is similar to that used in other TRIGA reactors. Based on a review of the information provided by the licensee in the SAR, the staff concludes that the source has sufficient strength to allow controlled reactor startup.

4.2.5 Core Support Structure

The reflector assembly rests on an aluminum platform at the bottom of the reactor tank, and provides support for the upper and lower grid plates and the safety plate. The top grid plate is supported by a ring welded to the reflector container. The top plate is an aluminum plate 0.625 inches thick which contains 126 holes, 1.505 inches in diameter drilled in six circular bands around a central hole to locate the fuel moderator and graphite dummy elements, control rods and guide tubes, and the pneumatic transfer tube. Provisions are available to remove sections of the top grid plate to account for larger diameter specimens. The differential area between the triangular-shaped spacer blocks at the top of the fuel element and the holes in the grid plate permit passage of cooling water.

The bottom grid plate is a 0.75-inch aluminum plate which supports the entire weight of the core, and provides accurate spacing between the fuel-moderator elements. Support for the bottom grid plate is provided by the reflector container. The holes in this plate are aligned with the fuel element holes in the top grid plate. The holes are countersunk to interface with the adapter end of the fuel-moderator elements and the adaptor end of the pneumatic transfer tube. A 1.505-inch-diameter central hole serves as the clearance hole for the central thimble. Eight additional 1.505-inch-diameter holes are aligned with the upper grid plate holes to provide passage of the fuel-follower control rods. Those holes not used by the control rod followers are plugged with removable fuel element adapters which rest on the safety plate. The adapters are solid aluminum cylinders 1.5 inches in diameter by 17 inches long. With the adapter in place, a position formerly occupied by a control rod with a fuel follower can accept a standard fuel element.

An aluminum safety plate is provided below the bottom grid plate to preclude the possibility of the control rods falling out of the core. This is a 0.5-inch-thick plate welded to the extension of the inner reflector liner and placed 16 inches below the bottom grid plate.

On the basis of its review of the SAR, the staff concludes that the core support assembly accurately positions and aligns the fuel elements for all anticipated operating conditions. The arrangement ensures a stable and reproducible reactivity. Sufficient coolant flow is provided by the core support. The staff concludes that reasonable assurance exists that the core support structure will continue to operate as designed throughout the period of renewal.

4.2.6 Conclusions

The staff reviewed the information on the design, construction, function, and operation of the reactor fuel, neutron reflectors, grid and safety plates, moderator/graphite elements, neutron source, control rods, and reactor core support structure. On the basis of this review, the staff concludes that the design of these core-related components for the OSTR are acceptable and should continue to permit safe operation and shutdown of the reactor. The design features of this reactor are similar to those typical of the TRIGA reactors licensed by the NRC at comparable power levels.

4.3 Reactor Tank and Pool

The OSTR aluminum reactor tank has an outside diameter of 6.5 feet, a depth of 20.5 feet, and a wall thickness of 0.250 inch. The reactor tank has provisions to accommodate the four beam ports, the thermal column, and the thermalizing column. For corrosion protection from the concrete biological shield, the outside of the tank is coated with an adhesive primer followed by several layers of polyethylene tape.

As discussed in Chapter 5 of the SER, the reactor tank is filled with demineralized water to provide approximately 16 feet of shielding water above the top of the reflector. The net water volume in the tank is approximately 4600 gallons. In accordance with TS 3.3, the tank water level is maintained greater than 14 feet above the core to guarantee sufficient water for effective fuel cooling and to ensure that radiation levels at the top of the reactor are within acceptable levels. Bulk water temperature is maintained below 120 °F (49 °C) to ensure that the reactor tank maintains its integrity and is not degraded and the demineralizer resin does not degrade. The conductivity of the tank water is maintained at less than 5 µmhos/cm to control corrosion.

Support for the central thimble, rotating rack irradiation facilities, control rod drives, and tank covers is provided by the central channel assembly. This assembly is placed over the top of the reactor tank and consists of two structural steel channels with steel cover plates. This assembly can support a shielded isotope cask.

In order to detect possible leakage from the reactor pool, water level is continuously monitored by a water level monitoring device; in addition, it is inspected manually during the performance of routine daily startup and shutdown checklists. The water level monitoring device actuates an alarm in the control room and is monitored remotely during non-working hours. The largest amount of primary water that could be lost from the reactor pool without detection is approximately 40 gallons.

The continuous water level monitor works as follows. The water level in the reactor pool is maintained at 2 inches above the low water level setpoint (one inch is equivalent to 20 gallons).

A 2-inch level drop from normal (equivalent to approximately a 40-gallon loss) actuates a low level alarm and a 1-inch above normal level will actuate a low level alarm and a 1-inch above normal level will actuate a high level alarm in the control room. After hours, the alarm is remotely monitored.

The frequency and quantity of makeup water added to the tank is recorded in a log. Any off-normal changes in the analysis of these parameters would be noticed by the operating staff and investigated. A loss of approximately 10 gallons per week would be the minimum detectable leakage rate, based on OSTR historical primary makeup rates that typically run 25 gallons per week for normal operations.

If a small leak is detected, additional water would be added or the fuel removed and the tank drained to facilitate repairs. In the event of a catastrophic leak, the water would run into the pipe trench from the base of the reactor into the heat exchanger room, which drains into the waste holdup tank. This volume of the waste holdup tank is 2950 gallons. The combined volume of the waste holdup tank and the trench can accommodate approximately 4100 gallons of the total 4600 gallons of reactor tank water, which is almost 90 percent of the tank inventory.

In the event of an unknown failure of the primary water tank (which might result from a weld joint failure or pinhole defect), the most likely leakage path would be from the primary joint failure to either the gap between the tank liner and the bioshield, beam port, or thermal column, ultimately collecting on the reactor bay floor. Any water accumulated would be collected and analyzed before disposal.

In the unlikely event of a failure that releases pool water to the soil beneath the pool, the impact on public safety and the environment would not be significant, the amount of water that would be lost before detection would be small, and the radioacitivity level of the water would be low. As described in Chapter 11 of the SAR, the typical radioactivity concentrations in the primary coolant are very low. The equilibrium concentrations of predominant radionuclides in the primary coolant are within the limits found in Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DAC) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” to 10 CFR Part 20 for release to the sewer. With the exception of sodium-24, which is slightly above the limit (sodium-24 has a 15-hour half life which means that the amount of this isotope in the primary coolant quickly drops after reactor shutdown), the equilibrium concentrations of predominant radionuclides are within effluent concentration limits for water given in a 10 CFR Part 20, Appendix B. The staff concludes that potential leakage of primary coolant will not have significant impact on the health and safety of the public or environment.

Based on the information provided by the licensee, and a history of acceptable tank performance, the staff concludes that the reactor tank will continue to perform acceptably during the license renewal period. As discussed in Chapter 13 of this SER, the radiological consequences of loss of primary coolant are acceptable. The licensee has the ability to detect and contain potential leakage from the tank.

4.4 Biological Shield

The steel-reinforced concrete structure at the OSTR extends 21 feet above the reactor room floor. The shielding at the reactor core is thicker in the radial direction compared to that above the core. The structure is pierced radially by the beam ports, thermal column, and the thermalizing column. As discussed in Chapter 9 and 10 of this SER, small vent pipes leading from these components are embedded in the concrete to allow the purging of argon-41 and other radioactive gases. In addition to the biological shield, the OSTR core is shielded radially by approximately 8 inches of graphite reflector, 2 inches of lead inside the reflector can, and 1.5 feet of water. A bulk shielding experimental tank (8 feet wide by 9 feet long and 12 feet deep) is located on the west side of the lower part of the reactor structure. The effectiveness of the biological shield in shielding the reactor core and controlling radiation exposure is supported by years of survey data showing acceptable radiation levels. The NRC inspection program routinely reviews the licensee's radiation protection program and performs independent measurements of radiation levels in the facility.

Based on a review of the information provided by the licensee, operational experience, and results from the NRC inspection program, the staff concludes that there is reasonable assurance that during the renewal period the OSTR biological shield design 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.

4.5 <u>Nuclear Design</u>

In this section, issues associated with the nuclear design are addressed. The information discussed in this section establishes the design bases for other chapters, especially the safety analyses and some of the TSs.

4.5.1 Normal Operating Conditions

The OSTR is a TRIGA-fueled reactor which nominally operates at a steady-state thermal power of 1 MW(t). The maximum, licensed, steady-state (nonpulsing) power level is set in TS 3.1.1, which states the following:

> The reactor power level shall not exceed 1.1 MW except for pulsing operation.

TRIGA reactors with natural convection cooling have been licensed by NRC up to thermal power levels of 2.3 MW(t). As will be discussed below in the thermal-hydraulic analysis, operation at 1.1 MW(t) allows for sufficient safety margins.

The OSTR is designed to be pulsed from a low power to a high power by the insertion of step reactivity. The OSTR has limited this step insertion of reactivity to $2.55 in TS 3.1.4, as described below:

> The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical block and electrical interlock on the transient rod, such that the reactivity insertion shall not exceed $2.55.

The maximum reactivity addition for pulsing will ensure that the reactor can be safely pulsed without fuel damage. The large prompt negative temperature coefficient of reactivity of the uranium-zirconium hydride fuel moderator provides a basis for safe operation of the reactor in the nonpulsing mode and is the essential characteristic supporting the operational capability of the reactor in a pulse mode. Pulse reactivity is limited to ensure that fuel temperature stays below the safety limit as discussed in Chapter 13 of this SER.

Reactor operating is defined in TS 1.21 as follows:

> The reactor is operating whenever it is not secure or shut down.

Reactor secured is defined in TS 1.23 as follows:

> The reactor is secured when:
>
> a. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material or moderator present in the reactor, adjacent experiments or control rods, to attain criticality under optimum available conditions of moderation and reflection; or,
>
> b. All of the following exist:
>
> 1. The four (4) neutron absorbing control rods are fully inserted as required by technical specifications;
>
> 2. The reactor is shut down;
>
> 3. No experiments or irradiation facilities in the core are being moved or serviced that have, on movement or servicing, a reactivity worth exceeding the maximum value of one dollar; and
>
> 4. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods.

Reactor shutdown is defined in TS 1.24 as follows:

> The reactor is shut down when:
>
> a. It is subcritical by at least one dollar both in the reference core condition and for all allowed ambient conditions, with the reactivity worth of all installed experiments and irradiation facilities included; and
>
> b. The console key switch is in the “off” position and the key is removed from the console.

These definitions delineate the operational state of the reactor and are used in part to determine the applicability of other TSs to the reactor.

The core has 122 positions where fuel elements or other items can be inserted, along with three motor-driven control devices (one regulating rod, one shim rod, and one safety rod) and one pneumatic electro-mechanical safety transient rod. The reactivity worths of these four rods are approximately $4.00 for the transient rod and $2.70 for each of the other three rods, for a total of approximately $12.10. The excess reactivity in the core does not normally exceed $7.50, so there is adequate margin to maintain the reactor shutdown while all of the control rods are inserted.

The fuel element positions (core lattice positions) are concentric rings surrounding a central thimble, core location A-1, which is filled with an aluminum plug. The concentric rings are labeled B through G, with 6 to 36 positions in each ring. This reactor has used both the GA "standard" fuel elements (20-percent uranium-235 enrichment) and the FLIP fuel elements (70-percent uranium-235 enrichment). Several core configurations have been developed and nine different core configurations are discussed in the SAR. The first three configurations, that are discussed used only the GA standard fuel elements, as the reactor was originally operated, and those configurations are no longer used. The other configurations considered mixed standard and FLIP fuel elements, and strictly FLIP fuel elements. The current core uses all FLIP fuel elements. TS 5.3.1 contains the design limits for arrangements of fuel elements and experiments in the core stating the following:

a. The core assembly shall consist of TRIGA fuel elements.

b. The fuel shall be arranged in a close-packed configuration except for single element positions occupied by in-core experiments, irradiation facilities, graphite dummies, aluminum dummies, stainless steel dummies, control rods, and start up sources.

c. The reactor shall not be operated at power levels extending 1 kW with a core lattice position water filled, except for positions on the periphery of the core assembly.

d. The reflector, excluding experiments and irradiation facilities, shall be water or a combination of graphite and water.

This TS controls design criteria by requiring the use of only TRIGA fuel elements (TS 5.3.1.a.) with water or a combination of water and graphite as the reflector (TS 5.3.1.d.). TS 5.3.1.b. and 5.3.1.c. control water holes in the reactor core. The purpose of controlling water holes is to control power peaking in fuel elements. The reactor as described in the SAR and observed by the staff during a site visit meets these design criteria.

The reactor was upgraded to use FLIP fuel in 1974 and the present configuration uses only FLIP fuel. In-core irradiations occur in the B-1 position. The licensee uses three basic cores, one with a fuel element in the B-1 position (the "Normal" core), an ICIT in the B-1 position (the "ICIT" core), or a CLICIT in the B-1 position (the "CLICIT" core).

TS 3.1.3 states the following:

> The maximum available excess reactivity based on the reference core condition shall not exceed $7.55.

Excess reactivity is defined in TS 1.9 as the following:

> 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.

The reference core condition is defined in TS 1.25 as the following:

> The reference core condition is the reactivity condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<0.30 dollars).

The reactivity state of a reactor can be affected by the fission product xenon, which is a neutron poison, and the temperature of the reactor. The purpose of defining a reference core condition is so that reactivity measurements can be adjusted to a fixed baseline.

A TS regarding excess reactivity is added to the TSs with this license renewal. The license did not have a limit on excess reactivity. Excess reactivity is needed for reactivity effects due to the power defect, fission product poisoning buildup and experiments. Limiting excess reactivity helps to ensure that the shutdown margin is met and controls available reactivity in potential accidents. The licensee has proposed a value for excess reactivity that will allow it to transition between the normal, ICIT and CLICIT cores with minimum fuel movement. The licensee has shown that the proposed excess reactivity limit was developed with consideration of the limit on shutdown margin. The staff concludes that the excess reactivity accounts for accepted reactivity effects and the shutdown margin and is acceptable to the staff. According to the licensee, recent measurements of the normal core showed the excess reactivity to be $5.79, the ICIT core to be $5.57, and the CLICIT core to be $3.59. The licensee calculated an excess reactivity of $7.10 for the FLIP core at beginning of life (BOL), which agreed well with the measured value of $7.17.

Shutdown margin is defined in TS 1.30 as follows:

> Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems and will remain subcritical without further operator action, starting from any permissible operating condition with the most reactive rod in its most reactive position.

The specific requirement for the shutdown margin is given in TS 3.1.2, which states the following:

> The shutdown margin provided by control rods shall be greater than $0.55 with:
>
> a. Irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state;
>
> b. The most reactive control rod fully-withdrawn; and
>
> c. The reactor in the reference core condition.

The purpose of defining a shutdown margin is to help ensure that the reactor can be shutdown by an acceptable margin even if one of the control rods becomes stuck out of the reactor core. In addition to assuming that the most reactive control rod is not available to help shutdown the reactor, TS 3.2.1 places constraints on the core condition and experiments. The reference core condition is the most limiting for determining the shutdown margin. 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 removed from the core.

For the original FLIP core configuration, the calculated shutdown margin for the normal core met the criteria in TS 3.1.2 for all of the rods except the regulating rod, which at that time had the highest control rod worth, calculated to be $3.71 and measured at $3.72. Measurements performed by the licensee during initial start-up testing of the core showed that the measured value of shutdown value did meet the criteria and this was found acceptable to the NRC. The licensee stated that the measurement method has been used throughout the lifetime of the OSTR to demonstrate compliance with the shutdown margin limit. The difference between calculated and measured values is due to evaluation techniques, which are inherent in the measurement methods that cannot exactly duplicate the calculations. Since the original set of measurements were made, the core has been reconfigured and presently the transient rod is the highest worth control rod. Measurements of shutdown margin of the current core used by the licensee demonstrate compliance with the TS limit.

The licensee has discussed and proposed minimum shutdown margin and excess reactivity limits that are acceptable to the staff. The minimum shutdown margin (TS 3.1.2) ensures that the reactor can be shutdown from any operating condition with the highest worth control rod stuck out of the core. The limit on excess reactivity (TS 3.1.3) allows operational flexibility while limiting the reactivity available for reactivity addition accidents.

GA has performed a considerable number of tests for TRIGA reactors and has developed operational bounds that have been accepted by the NRC. The TSs for the OSTR and the mode of operations of the OSTR are well within the bounds that have been accepted and shown to be safe.

The analysis discussed in Chapter 13 of this SER demonstrates that if a situation arises that results in the uncontrolled removal of a control rod, the reactor will shutdown as the fuel heats up. This shutdown will occur before any fuel damage occurs.

The reactor core power distribution was calculated using a reference HEU model that is similar to the presently loaded core containing four less fuel elements. The reactor power distributions were calculated with the aluminum slug in position A-1, and a FLIP fuel element in B-1. The use of the ICIT or CLICIT in the core does not significantly change the maximum fuel element powers. This model was found to have good predictive capability over a wide range of reactor conditions. The associated peaking factors in each fuel ring, as predicted by the analytical model, are acceptable. Power peaking is further discussed in Section 4.6 below.

The surveillance requirements for shutdown margin and excess reactivity are given in TS 4.1.c. and 4.1.d. as follows:

c. The shutdown margin shall be determined prior to each day's operation, prior to each operation extending more than one day, or following any significant change (>$0.25) from a reference core.

d. The core excess reactivity shall be determined annually or following any significant change (>$0.25) from a reference core.

The worth of the control rods must be accurately known to determine compliance with the reactivity limits for the reactor. TS 4.1.b. states the following:

b. The total reactivity worth of each control rod shall be measured annually or following any significant change (>$0.25) from a reference core.

These TSs are consistent with ANSI/ANS-15.1, "Standard for the Development of Technical Specification for Research Reactors," issued in 2007, and operational experience has shown that these surveillance intervals are acceptable to detect changes in core behavior. The staff concludes that these surveillances will help ensure that these core parameters are within their TS limits.

The staff reviewed the licensee's analyses and found that the licensee considered an appropriate variety of core configurations and that these core configurations contained the components required for an operable reactor core. The staff found that the licensee used input parameters justified by analyses presented in the OSTR SAR. The staff found that the licensee adequately analyzed the reactivity effects of individual core components. TSs related to the normal operating conditions of the reactor core include limits on excess reactivity, minimum shutdown margin, allowable core configurations, and surveillance requirements for the core reactivity parameters and reactivity worth of the control rods. These TSs are consistent with ANSI/ANS-15.1. The staff found that the analyses presented in the OSTR SAR adequately justify these TSs and show that normal reactor operation will not lead to the release of fission products from the fuel. Based on these considerations, the staff concludes that the licensee has adequately analyzed expected normal reactor operation during the period of the renewed license. The staff further concludes that the TSs provide reasonable assurance that normal operation of the OSTR core will not pose a significant risk to the health and safety of the public or the environment.

4.5.2 Reactor Core Physics Parameters

The safe operation of a TRIGA reactor during normal operations is accomplished by the control rods and is monitored accurately by the core power level (neutron) detectors. An additional 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_xmatrix 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 ZrH_x are mixed homogeneously, thus the ZrH_x 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 uranium-238 resonances at high temperatures, which increases nonproductive neutron capture in these resonances.

Because of the large, prompt, negative temperature coefficient, a step insertion of reactivity resulting in an increasing fuel temperature will be compensated for by the fuel matrix rapidly and automatically. 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 (GA-E-117-833, 1980; Simnad et. al., 1976). This inherent characteristic of the U-ZrH_x fuel has been the basis for designing TRIGA reactors with a pulsing capability as a normal licensed mode of operation.

For the FLIP fuel, values for the prompt temperature coefficient of reactivity, α_T, range from approximately -0.3×10^{-4} Δk/k-°C at room temperature to -1.8×10^{-4} Δk/k-°C at 700 °C. For normal operating conditions, the temperature coefficient of reactivity for a prompt insertion was calculated to be approximately -1.26×10^{-4} Δk/k-°C and the steady state temperature coefficient of reactivity was measured to be approximately -5.1×10^{-5} Δk/k-°C. The method of calculating α_T for this reactor used the computer codes developed by GA, including GCC-3, GAZE-2, and GAMBLE-5. The OSTR also used the Los Alamos National Laboratory code DTF-IV. The values quoted for this core are similar to values reported by GA and, therefore, a large step insertion of reactivity will be compensated for by the fuel as the temperature increases, causing the transient to end. This large negative value of α_T was designed into the fuel so that TRIGA reactors can be pulsed. In the pulsed mode of operation, TS 3.1.4 limits the step insertion of reactivity to $2.55. GA performed numerous tests with step insertions up to $5.00 before any fuel damage became apparent; therefore, the TS limit of $2.55 is well within the safety envelope established by GA and approved by the NRC. In Chapter 13 of the SAR, conservative calculations were made using the Nordheim-Fuchs model to determine the step reactivity that the core could be subjected to and have the fuel remain below 1100℃. For a BOL FLIP fuel element the value is $3.21 and for an end-of-life (EOL) FLIP fuel element the value is $2.59.

The FLIP fuel contains erbium as a burnable poison. Since the erbium burns faster than the uranium-235, the excess reactivity of a core will increase with time until about midlife of the core. After that, the excess reactivity decreases with time as the uranium-235 is depleted. The FLIP fuel is expected to have a lifetime of more than 8 MW-year. The normal annual operation of the OSTR is 0.125 MW-year, so the FLIP core is expected to have a lifetime of more than 64 years. The projected life of the present OSTR core is long enough that there is no anticipated reloading of the core.

The thermal neutron flux for the OSTR operating at 1 MW(t) (while the licensed power level is 1.1 MW(t), the reactor is normally operated at 1 MW(t)) is approximately 10^{13} n/cm^2/s, which is consistent with other 1 MW(t) research reactors. The neutron lifetime has been calculated at GA to be 20 microsecond. The best-estimate value for the effective delayed neutron fraction for the HEU FLIP core was found to be β_{eff}=.0076. These values are similar to other TRIGA reactors

The power coefficients provided by the licensee show negative values and all of the coefficients are similar to other TRIGA-type nuclear reactors. According to the licensee, the void coefficient has been measured to be between -13 and -51¢ per percent void, and published values (NUREG/CR-2387, “Credible Accident Analysis for TRIGA and TRIGA-Fueled Reactors”) state that this should be on the order of -28¢ per percent void. The fuel coefficient at 1 MW is -1¢ per °C and the tank water temperature coefficient is -0.05¢ per °C. These values are consistent with other TRIGA reactors.

Since the OSTR normally does not operate for more than 8 hours at a time, the xenon-135 does not reach saturation. The licensee estimates that the value of the xenon and samarium override value at saturation would be $8.00.

The staff reviewed the licensee’s analyses and found that the licensee considered appropriate core physics parameters. The methods used to determine values of the core physics parameters are acceptable. The values are acceptable and are similar to those found acceptable at other TRIGA reactors.

4.5.3 Operating Limits

The regulations in 10 CFR 50.36(d)(1) require reactors to specify safety limits and LSSSs. Safety limits are defined in 10 CFR 50.36(d)(1) as 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. 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 a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

TS 2.1 states the following:

> The temperature in a TRIGA fuel element shall not exceed 2100 °F (1150 °C) under any mode of operation.

Calculations performed by GA and confirmed by experiments indicate that no cladding damage occurs at peak fuel temperatures as high as approximately 1175 °C (2150 °F) for high-hydride-type (U-$ZrH_{1.65}$), stainless-steel-clad elements (Coffer et al., 1966; Simnad, 1980; Simnad et al., 1976). The fundamental consideration in limiting fuel temperature to 1150 °C is to limit the fuel element internal pressure caused by the buildup of hydrogen gas (resulting from the diffusion of the hydrogen out of the zirconium-uranium matrix) within the fuel element gap. Limiting the maximum fuel temperature prevents generating excessive internal pressures from the hydrogen gas, which will increase the stress on the clad beyond the yield point, thereby causing a rupture and release of fission products. An additional consideration is the need to provide adequate cooling relative to the maximum heat flux to prevent departure from nucleate boiling (film boiling) and the resulting rapid increase in clad temperature which will lead to failure of the clad (see discussion in Section 4.6 below). A power-level limit is calculated that ensures that fuel temperatures will not be exceeded and that film boiling will not occur. The design-bases analysis shows that operation at 1.1 MW(t) with the proposed core, across a broad range of core and coolant inlet temperatures with natural convection flow, will not lead to the film boiling that results in high fuel and clad temperatures and the attendant loss of clad integrity. The staff concludes that the safety limit for the OSTR is the long-standing safety limit used for TRIGA reactors which is supported by research conducted by GA and is therefore acceptable.

TS 2.2 states the following:

> The limiting safety system setting shall be equal to or less than 510 °C (950 °F) as measured in an instrumented fuel element. The instrumented fuel element shall be located in the B-ring.

The LSSS is set equal to or less than 510 °C (950 °F) as measured in the IFE. Exceeding this limit causes a scram of the reactor and protects the fuel from exceeding the safety limit. The IFE is not measuring the hottest fuel location in the reactor core. The relationship between the measured temperature in the IFE and the actual temperature at the fuel hot spot in the core needs to be determined to show that the setting of 510 °C protects the safety limit at the hottest point in the core. The IFE contains three thermocouples which measure the fuel temperature. The IFE is placed in the B ring since fuel in the B ring has the highest power level. However, the location of the IFE in the B ring may not be the same as the location of the hottest fuel element in the B ring. The licensee determined the power needed in an IFE to result in a temperature of 510 °C at the midplane thermocouple. It was assumed that the IFE was located in the lowest power position in the B ring. Power peaking in the highest power position in the B ring is 1.036. The power in the hottest fuel element was determined. The actual peak temperature related to the peak power is 588 °C. Measurement accuracy is ±5 °C. The licensee has performed calculations that show that an indicated temperature of 510 °C on the fuel temperature measuring channel corresponds to a peak fuel temperature in the core of 588 °C. The calculations account for the adjustments needed between indicated and actual temperature discussed above. The calculations show that the LSSS of 510 °C protects fuel from exceeding the safety limit of 1150 °C by a significant margin. The staff concludes that the LSSS of 510 °C is sufficient to protect the safety limit and is acceptable.

The staff concludes that the safety limit and LSSS for the reactor are based on acceptable theoretical and experimental investigations and are consistent with those approved by the NRC and used at other TRIGA-type reactors and are therefore acceptable.

4.6 Thermal-Hydraulic Design

The important parameter in thermal-hydraulic design is the critical heat flux (CHF), which describes the heat flux associated with the departure from nucleate boiling. If the heat transfer process from fuel to coolant departs from nucleate boiling, clad temperatures will rise. (Essentially, heat is not removed from the cladding at the proper rate, causing clad temperatures and, consequently, fuel temperatures to rise.) The parameter that measures the departure from nucleate boiling is the critical heat flux ratio (CHFR), which is the ratio of the CHF to the maximum heat flux at full power. It is essential that this ratio always be greater than unity.

The licensee has supplemented the thermal-hydraulic analysis presented in the SAR with an RAI response that contains a more detailed analysis using a RELAP5-3D model of the OSTR core. The supplemental analysis presents a more comprehensive and up-to-date thermal-hydraulic design of the OSTR. The evaluation of the safety margin that exists during the operation of the OSTR at the licensed power level is based mainly on the supplemental analysis. In the thermal-hydraulic analysis both steady-state and pulse operation were analyzed at different times in core life and operational data were used to benchmark the analytical results.

The OSTR fuel elements are cooled by natural convection. The steady-state analysis was done at the maximum licensed power of 1.1 MW(t) (nominal operating power is 1.0 MW(t) with the high power scram normally set at 1.06 MW(t)) and a water inlet temperature of 49 °C (the maximum pool temperature per TS 3.3.b.). Results of the steady-state analysis include fuel, clad, and coolant temperatures and the minimum departure-from-nucleate-boiling ratio (MDNBR) (similar to CHFR). For the pulse analysis, calculated peak fuel temperature and pulse peak power are presented.

The RELAP5-3D code, a widely used and benchmarked system code for power and research reactors, was used for the thermal-hydraulic analysis. The steady-state analysis considered the maximum power fuel element and the coolant subchannel associated with this single fuel rod. The driving force for the core flow is supplied by the column of water surrounding the core. A natural circulation flow rate is established to balance the driving head against the core entrance and exit pressure losses, and frictional, acceleration, and hydrostatic head losses in the core flow channel. In order to assess the effect of cross-flow between flow channels a parametric study was performed. Cross-flows between adjacent coolant subchannels were incorporated in a two-channel and eight-channel model. Based on the parametric study it was observed that the cross-flow has the effect of lowering the exit coolant temperature while the effect on CHFR is insignificant (less than 1 percent).

Given the inlet water temperature (49 °C), system pressure (absolute pressure at the top of the core), local pressure loss coefficients, and the axial and radial power distribution for the bounding fuel element, RELAP5-3D calculated the natural circulation flow rate, and along the axial length of the flow channel, the coolant temperature, wall heat flux, the clad temperature, and the peak fuel temperature. The flow channels in the OSTR are triangular, square, or irregular, depending on core location. The RELAP5-3D calculation was based on the subchannel flow area for fuel elements in the B ring which has the smallest rod-to-rod pitch and thus the smallest subchannel flow area in the core.

The maximum powered element is determined using MCNP5 with all control rods (including the transient rod) conservatively removed from the core. The location of the hot rod (maximum powered element) varies according to core configuration but it has not been in the location of the IFE (B1). The hot rod peak factor is defined as the power generation in the hottest rod (element) relative to the core-average rod power generation. Two other power peak factors are defined for the OSTR. They are the hot rod fuel axial peak factor and the hot rod fuel radial peak factor. The axial peak factor represents the axial peak-to-average power ratio within the hot fuel rod, and the radial peak factor represents the peak-to-average power density in a radial plane at the hottest axial location of the hot fuel rod. The only peaking factors that have an impact on the steady-state heat flux at the clad surface are the hot rod peaking factor and the axial peak factor. The heat flux peaking factor, a product of the hot rod peak factor and the axial peak factor, is defined as the ratio of the heat flux for the hot node to the core average heat flux. Using the values of peaking factors calculated for the OSTR core, the heat flux peaking factor is shown in Table 1 below. For the FLIP steady-state core the heat flux peaking factor is the maximum at BOL.

Table 1. Summary of Hot Rod Peaking Factors

	Hot Rod Location	Hot Rod Thermal Power* (kw)	Hot Rod Peak Factor, Pmax/Pavg	Hot Rod Fuel Axial Peak Factor, Pmax/Pavg	Heat Flux Peaking Factor
FLIP-BOL NORMAL Core	B3	18.02	1.376	1.236	1.701
FLIP-MOL NORMAL Core	B6	18.37	1.403	1.209	1.696
FLIP-EOL NORMAL Core	B6	16.48	1.258	1.234	1.552

* Hot rod thermal power corresponds to core power of 1.1 MW(t).

Two CHF correlations were used in the thermal analysis of the OSTR, the 2006 AECL Groeneveld look-up tables and the Bernath correlation. RELAP5-3D also has implemented Groeneveld look-up tables for calculating MDNBR but it is based on the 1986 version of the tables. Based on a study by OSTR it was observed that the 1986 look-up tables produce the least conservative CHFR values (for OSTR operating conditions), followed by the 2006 Tables and the Bernath correlation (the most conservative). It is noted that the predicted MDNBR for a reactor power of 1.1 MW(t) is 2.1 and 3.42 by using the Bernath correlation and the Groeneveld look-up table, respectively. At the maximum licensed power of the OSTR there is significant margin to departure from nucleate boiling.

For the type of TRIGA fuel used in the OSTR, the gas gap between the fuel and the cladding is known to vary between 0.05 and 0.4 mils. The effect of gap size on fuel temperature was evaluated by performing a series of RELAP5-3D calculations assuming different gap sizes and comparing calculated to measured temperatures by the IFE. A gap of 0.1 mil was found to give conservative predictions of fuel temperature as compared to measurement (prediction is 17 to 34 °C higher than measurement). The same gap size was used in the steady-state and pulse calculations.

For the pulse analysis the transient power was calculated by the point kinetics model in RELAP5-3D. The reactor was modeled with two separate hydraulic channels, the hot rod channel (same as for the steady-state analysis) and an average channel representing the rest of the fuel elements. The FLIP BOL NORMAL core configuration was analyzed for pulse operation at the licensed limit of reactivity insertion of $2.55. Two initial reactor powers were considered, 1.1 MW(t) (maximum licensed power) and 1 kW(t) (interlock for pulse operation). A comparison of the results for the two cases demonstrates that a lower initial power produces a higher peak core power. However, in both cases, the peak fuel temperature is below the safety limit during the pulse transient.

The staff concludes that the thermal-hydraulic analysis for the OSTR adequately demonstrates that the reactor can operate at its licensed power level with sufficient safety margins in regard to thermal-hydraulic conditions. The analyses were done with qualified calculational methods and acceptable assumptions.

4.7 <u>Conclusions</u>

Based on the above considerations, the staff concludes that the licensee has presented adequate information and analyses to demonstrate the technical ability to configure and operate the OSTR core without undue risk to the health and safety of the public or the environment. The staff 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 staff concludes that the OSTR 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 the research reactors of the TRIGA type operating in many countries of the world. On the basis of this review of the OSTR and experience with these other facilities, the staff concludes that there is reasonable assurance that the OSTR is capable of safe operation, as limited by the TSs, for the period of the requested license renewal.

5. REACTOR COOLANT SYSTEMS

5.1 Reactor Coolant Systems Description

The reactor coolant systems at the OSTR consist of those systems that remove heat from the reactor, maintain water purity and clarity, and provide makeup water. The design of the reactor coolant systems at the OSTR is typical of TRIGA reactors.

5.2 Primary Coolant System

The primary coolant system at the OSTR consists of an open 6.5-foot diameter, 20.5- foot deep aluminum tank in which the reactor core is submerged in the light water coolant. Heat generated from the reactor core is transferred to the tank water that flows through the core by natural convection. Natural convection cooling is a primary design feature of the reactor as given in TS 5.2.a.:

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

The thermal-hydraulic analysis discussed in Chapter 4 of this SER shows that the heat produced by the reactor can be safely dissipated to the primary coolant by natural convective water flow. Because TS 5.2 a. controls a basic design consideration of the reactor cooling system it is acceptable to the staff.

TS 3.3.a contains the following requirement for the minimum amount of water above the reactor core:

> The reactor primary water shall exhibit the following parameters:
>
> > a. The tank water level shall be greater than 14 feet above the top of the core;

Water level in the tank is normally maintained at 16 feet above the reactor core. The water level is controlled to provide a minimum level of radiation shielding of the reactor core. Radiation surveys with the reactor at full power show that a sufficient level of shielding is provided above the core. The height of the water above the core also determines the saturation temperature of coolant at core level. The reactor has a water level alarm required by TS 5.2.c. as described below:

> c. A tank water level alarm shall be provided to indicate loss of coolant if the tank level drops 6 inches below normal level.

TS 4.3 a. contains the following surveillance requirement for the water level alarm:

> a. A channel check of the reactor tank water level monitor shall be performed monthly.

The water level alarm alerts the reactor operator in the event of a drop in water level. During hours when the facility is not staffed, the alarm is remotely monitored. Normal water level in the pool is about 2 inches above the tank water level alarm low setpoint. Upon receiving an alarm, the reactor operator can then take appropriate action such as shutting down the reactor and responding to the cause of the water level drop. The surveillance interval is based on the operating experience with the system. The surveillance is not postponed if the reactor is shut down. As discussed in Chapter 13, a total loss of primary coolant will not damage the reactor core. Having the reactor operator take action instead of having an automatic reactor scram will not impact the results of a loss-of-coolant accident. Based on the discussion above, the staff finds the tank level alarm and associated surveillance to be acceptable.

To remove heat from the reactor pool, a primary coolant pump moves coolant through the primary piping from the reactor tank and delivers 490 gallons per minute (gpm) of coolant to the tube side of a heat exchanger, which has the capacity to transfer about 1 MW of thermal energy. It is not necessary to have the primary cooling system operating to operate the reactor because the coolant inventory in the reactor tank can act as a heat sink. The reactor heat is transferred to the secondary coolant system in the heat exchanger and the coolant is returned to the reactor tank.

All piping (primary and demineralizer) with the potential to siphon coolant from the reactor pool if a break were to occur have siphon breaks with holes located 22 inches below normal water level. The siphon breaks are a required design feature as discussed in TS 5.2 b.:

> b. The tank water inlet and outlet pipes to the heat exchanger and to the demineralizer shall be equipped with siphon breaks not less than 14 feet above the top of the core.

Because this design feature helps to ensure that the pool water level is maintained at least 14 feet above the top of the core as required by TS 3.3.a., it is acceptable to the staff.

About 50 gpm is returned to the pool by a diffuser nozzle above the core whose purpose is to control radiation levels at the top of the pool from nitrogen-16 (see Section 5.6 below).

Any break or leakage of the primary coolant system or reactor pool is bounded by the catastrophic reactor tank failure addressed in Section 13.2.3.2.2 of the SAR. Most of the primary coolant would be contained in the pipe trench and the minimum available volume of the waste holdup tank, which is the collection point for the reactor and equipment room drains.

There is a limit of less than 120 °F (49 °C) on the bulk coolant temperature in the reactor tank. The limit is in TS 3.3.b., as stated in the following:

> The reactor primary water shall exhibit the following parameters:
>
> b. The bulk tank water temperature shall be less than 120 °F (49°C).

The purpose of the bulk temperature limit is to ensure that the mixed-bed demineralizer does not experience thermal breakdown, and to help ensure that the aluminum reactor tank maintains its integrity and is not degraded by temperature-induced stresses. The temperature value is given by the reactor manufacturer. The temperature of the coolant is also an input to the thermal-hydraulic analysis. This TS is acceptable to the staff because it helps to protect the demineralizer resin and the reactor tank from degradation.

The design feature related to the temperature limit is given in TS 5.2.d. as follows:

> d. A bulk tank water temperature alarm shall be provided to indicate high bulk water temperature if the temperature exceeds 120 °F (49 °C).

This alarm will allow the reactor operator to take actions as needed to control the temperature. Because this alarm helps to protect the demineralizer resin and the reactor tank from temperature-related degradation it is acceptable to the staff.

The surveillance requirements in TS 4.3 associated with the bulk coolant temperature are as follows:

b. A channel check of the reactor tank water temperature system, including a verification of the alarm setpoint, shall be performed prior to each day's operation or prior to each operation extending more than one day.

c. An operability check of the reactor tank temperature alarm shall be performed monthly.

d. A channel calibration of the reactor tank water temperature system shall be performed annually.

Operating experience has shown that an annual calibration and daily channel check give reasonable assurance that the tank water temperature system will operate with acceptable accuracy. The surveillance requirements are acceptable to the staff.

The staff has reviewed the primary coolant system and associated TSs. The staff concludes that the primary coolant system is designed in accordance with the design bases derived from the thermal-hydraulic analysis in the SAR. The system is designed to remove sufficient heat from the reactor pool to allow all licensed operations without exceeding the established bulk coolant temperature limits that are included in the TSs. The staff also concludes that sufficient shielding is provided by the coolant above the reactor when coolant levels are controlled in accordance with the TSs. The staff concludes that the surveillance requirements given in the TSs provide reasonable assurance of necessary primary coolant system operability for reactor operations as analyzed in the SAR. Therefore, the staff concludes that the design and operation of the primary coolant system is acceptable.

5.3 Secondary Coolant System

The secondary coolant system, along with the primary coolant system, transfers the heat produced by the reactor to the environment. Heat is removed from the shell side of a heat exchanger, with an operating capacity of about 1 MW of thermal energy, by the water passing through it to an evaporative cooling tower where the heat is transferred to the atmosphere. Water from the cooling tower basin is pumped by the cooling tower circulation pump through the heat exchanger at a nominal flow rate of 700 gpm and returned to the tower. Design features of this system allow transfer of reactor heat from the primary system under all operating conditions. However, it is only needed to maintain primary coolant temperature during full-power operations.

The secondary system pressure is higher than the primary system pressure when both cooling system pumps are in operation or shut down. Therefore, a tube leak or failure in the heat exchanger will result in secondary coolant entering the primary coolant and will not result in uncontrolled release of radioactivity to the environment. Changes in primary coolant chemistry will alert the operations staff to the heat exchanger failure.

Makeup to the tower basin due to evaporation losses is provided by the city water supply. Level instrumentation, located in the basin, controls a float valve that opens only if makeup water is needed. Backflow is precluded, since the city water system is at a higher pressure than the cooling tower basin, which is at atmospheric pressure.

Secondary chemistry is maintained by an automatic water conditioning system that controls corrosion, biological growths, and water conductivity.

Secondary coolant system instrumentation provides the necessary status indications in the control room.

Because the secondary coolant system has an operating capacity of approximately 1 MW of thermal energy, the staff concludes that the design of the OSTR secondary system is adequate to remove heat from the primary coolant system under licensed operating conditions.

Because the secondary coolant system is maintained at a higher pressure than the primary coolant system, the staff concludes that there is reasonable assurance that in the event of an internal failure of the heat exchanger primary coolant would be contained within the OSTR facility.

5.4 <u>Primary Coolant Cleanup System</u>

The primary cleanup system extracts 10 gpm of primary coolant using a 0.75 horsepower (hp) pump and passes it through two 25 micron cartridge filters and a mixed-bed demineralizer to keep the primary water chemistry within operational limits. This system is designed to provide reasonable assurance that the required water quality and inventory for all modes of operation is maintained. The requirements for primary coolant conductivity are given in TS 3.3.c. as follows:

> 3.3 The reactor primary water shall exhibit the following parameters:
>
> > c. The conductivity of the tank water shall be less than 5 μmhos/cm.

The surveillance requirement for primary coolant conductivity is given in TS 4.3.e as follows:

> > e. The reactor tank water conductivity shall be measured monthly.

The conductivity limit used by the licensee is a longstanding value for research reactors accepted by the NRC which has been shown to be effective in controlling corrosion in aluminum and stainless-steel systems. Experience has shown the surveillance interval to be acceptable for ensuring acceptable water conductivity. The surveillance is not postponed if the reactor is shut down.

The system is designed such that no malfunction or leakage will lead to a large uncontrolled release of reactor coolant. The demineralizer piping has siphon breaks to prevent a failure in the demineralizer piping from siphoning a large amount of primary coolant from the reactor tank.

The system also contains a water monitor, a vessel that includes a temperature probe (inlet water temperature), a conductivity probe (inlet water conductivity), and a Geiger-Muller tube for detecting radioactivity. A skimmer system keeps the reactor tank surface free of debris.

The staff has reviewed the design of the primary coolant cleanup system and associated TSs and concludes that the system will acceptably control quality of the primary coolant to limit corrosion of the reactor fuel and other systems that contact primary coolant.

5.5 Primary Coolant Makeup Water System

Makeup water to the primary coolant system is supplied by an onsite reverse-osmosis distilling unit. The makeup water is added to the primary coolant system through the primary coolant cleanup system demineralizer. This system replaces normal water losses from evaporation at the reactor tank surface or due to sampling operations.

There is no direct physical connection between the city water system and the reactor primary water system. Water is added to the primary only after it has passed through a reverse osmosis unit designed to purify the water. Water is passed from one side of the reverse osmosis unit to the other by dripping into a flask, thereby precluding the possibility of siphoning. Additionally, all water entering the facility is protected against siphoning with a reverse pressure backflow prevention device which is inspected annually. This eliminates the possibility of siphoning primary water into the city water system.

The staff has reviewed the design of the primary coolant makeup water system and concludes that the system is sufficient to replace normal primary water loss and will protect against the entry of primary coolant into the city water system.

5.6 Nitrogen-16 Control System

This system, which is common in pool-type research reactors, reduces the gamma dose rate at the pool surface from radioactive nitrogen-16 (7 second half-life) which is produced when oxygen in the water passing through the core is irradiated. The warm coolant plume exiting the top of the core rises to the pool surface carrying the nitrogen-16 which can result in a radiation field at the top of the reactor tank. Approximately 50 gpm of the 490 gpm primary coolant discharge flow is returned to the tank to a diffuser nozzle located just above the core. The water flow from the diffuser disrupts and breaks up the warm coolant plume from the reactor core which significantly increases the amount of time it takes nitrogen-16 to reach the top of the tank, which allows for additional radioactive decay and limits the dose from this radiation source. The dose rate at the tank surface during full-power operation is approximately 100 millirem per hour (mrem/hr). This dose rate is from nitrogen-16, argon-41 and radiation from the reactor core. The dose rate nearly doubles with the diffuser turned off.

As discussed in Chapter 12 of this SER, TS 3.7.1 specifies that the reactor top area radiation monitor will operate and TS 4.7a specifies that a channel check will be performed daily. These radiation monitors help ensure that the Nitrogen-16 control system is functioning as designed.

The staff concludes that the design of the nitrogen-16 control system along with the licensee's radiation protection program and ALARA program provide sufficient reduction of radiation fields at the top of the reactor tank from nitrogen-16 to maintain personnel exposures below the limits in 10 CFR Part 20.

5.7 Conclusions

On the basis of its evaluation of the information presented in the licensee's SAR, the staff concludes as follows:

- The OSTR reactor coolant systems are adequate to remove heat from the fuel and prevent loss of fuel integrity under normal full-power operating conditions.

- There is reasonable assurance that any accidental leakage from the primary coolant system would be contained within the OSTR facility. As a result, there would not be significant radiation exposure to the public in the event of such leakage.

- The TSs provide reasonable assurance that the cooling system will operate as designed and be adequate for reactor operations as described in the SAR.

- The water purification system will control chemical quality of the primary coolant to limit corrosion of the reactor fuel and other systems that contact primary coolant.

- The design of the nitrogen-16 control system along with the licensee's radiation protection program and ALARA program provides sufficient reduction of radiation fields at the top of the reactor tank from nitrogen-16 to maintain personnel exposures below the limits in 10 CFR Part 20.

6. ENGINEERED SAFETY FEATURES

6.1 Introduction

As discussed in Chapter 13 of this SER, the licensee has demonstrated in Section 13.2.1 of the OSTR SAR that the consequences from an MHA involving the failure of the cladding of one fuel element in air does not require active engineered safety features. The licensee performed dose calculations for members of the public assuming no ventilation system or confinement building. This analysis demonstrated that doses from a fission product release are within regulatory limits. The analysis of the loss-of-coolant accident assumed no emergency core cooling system, just natural convection cooling from the reactor room ambient air.

6.2 Engineered Safety Features

The most common engineered safety features found in research reactors are confinement or containment, including an associated ventilation system, and an emergency core cooling system. These engineered safety features and their applicability to the OSTR are discussed below.

6.2.1 Confinement and Ventilation Systems

The OSTR reactor bay ventilation system and reactor building, which act as a confinement, are discussed in Section 9.1 of this SER and are described in Chapter 9 of the SAR. ANSI/ANS-15.1 defines a confinement as 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. As shown in Chapter 13 of this SER, the confinement and ventilation systems are not required to keep doses during potential accident conditions within regulatory limits. The system is independent of the Radiation Center and laboratory ventilation systems. The reactor bay is kept at a negative pressure in relation to the outside ambient air. If an abnormal situation arises, the ventilation system can be automatically shut down, which effectively isolates the reactor bay free air volume from other areas of the Radiation Center complex and the environment. The capability to remotely monitor the reactor bay atmosphere when the ventilation system is isolated also exists. The confinement and ventilation systems help to control releases from the reactor building and serve to reduce doses to members of the public during normal operation and potential accident conditions. The staff concludes that the confinement and ventilation systems as used at the OSTR are not engineered safety features.

6.2.2 Containment

Most research reactors can be designed, sited, and operated such that a containment is not required for normal operation or accident mitigation. ANSI/ANS-15.1 defines a containment as an enclosure of the facility designed to be at a negative internal pressure to ensure in-leakage, control the release of effluents in the environment, and mitigate the consequences of certain analyzed accidents or events. Containments are much more robust than confinements. If a facility does not need a confinement engineered safety feature it also does not need a containment. Staff review of the accidents analyzed in Chapter 13 of the SAR demonstrated that a containment is not necessary for the OSTR.

6.2.3 Emergency Core Cooling System

Section 13.5 of this SER discusses a potential loss-of-coolant accident. Evaluations performed by the licensee show that air cooling after a loss-of-coolant accident is sufficient to remove decay heat from the fuel and prevent loss of fuel element integrity. The evaluation also shows that dose rates from the uncovered core in the unrestricted area are within regulatory limits. Therefore, the staff concludes that an active emergency core cooling system is not necessary.

6.3 Conclusions

On the basis of its evaluation of the information presented in the licensee's SAR, the staff concludes there is no need for any engineered safety features to mitigate the consequences of the potential accidents analyzed in Chapter 13 of the SAR.

7. INSTRUMENTATION AND CONTROL SYSTEMS

While the licensed power level of the reactor is 1.1 MW(t), the licensee normally operates the reactor at about 1 MW(t) to allow an operating margin for factors such as instrument drift and calibration accuracy.

The OSTR may be operated in one of three modes.

- steady state mode — steady state operation of the reactor at power levels not exceeding 1.1 MW(t)
- square wave mode — step insertion of reactivity rapidly that raises reactor power to a steady-state level up to 1.1 MW(t),
- pulse mode — a large step insertion of reactivity rapidly raises reactor power to create a short duration reactor power pulse of high power (several thousand megawatts)

The specific operating mode is determined by the position of a mode selector switch located on the reactor console. These modes are defined in the TSs by the position of the mode selector switch.

7.1 <u>Summary Description</u>

The I&C systems employed at OSTR are similar to those used by other TRIGA research reactors operating in the United States. Control of the nuclear fission process is achieved using four control rods— one regulating rod, one shim rod, one safety rod, and one transient rod (also used for pulse and square-wave mode operation). The position of each rod is displayed on position indicators located at the control console. Scram is accomplished by deenergizing the magnets for the shim, safety, and regulating rods and by deenergizing the air-operated solenoid valve for the transient rod. In the event of power failure or receipt of a scram signal, all four rods fall freely into the core by gravity.

The I&C system is composed of both nuclear and nonnuclear process and control instrumentation channels that provide the means to safely control the reactor and to avoid or mitigate accidents. All reactor operations are performed at the reactor console which provides controls for the reactor and reactor systems, indication of key reactor operating parameters, and a means for selecting the reactor operating mode (steady state, square wave, or pulse).

Five radiation-based instruments, a wide-range log channel, a wide-range linear channel, a safety channel, a percent-power channel, and a pulse mode channel (called Nvt circuit in the TSs), provide indication of reactor power from intrinsic source range levels to full power (Figure 7.1 of the SAR). The wide-range log channel and wide-range linear channel provide indication of reactor power over the entire normal range from a source level of 0.001 watt(t) to 1.1 MW(t). The safety channel and percent-power channel provide indication above 10 kW(t) (1 percent power) and the pulse mode channel provides indication of peak power above 20 MW(t) while operating in the pulse mode. In addition to log power, the wide-range log channel can also provide an indication of reactor period. Input signals to all of these channels are provided by a fission chamber and two uncompensated ion chambers located near the reactor core. The fission chamber provides an input to the wide-range log channel and wide-range linear channel. One uncompensated ion chamber is used by the safety channel and a second uncompensated ion chamber is used by the percent-power and pulsing channels. If the steady-state reactor power level reaches the maximum power level specified in TS 3.1.1 (1.1 MW(t)), an automatic scram of the reactor will be initiated by either the safety or percent-power channels (the license normally sets this scram at 1.06 MW(t)). During pulse operations, the peak power level is measured by the pulsing channel and displayed on the console.

The primary safety parameter for a TRIGA reactor is fuel temperature. Fuel temperature is measured directly via the fuel temperature channel. Fuel temperature indication is sensed in the IFE, is provided at the reactor console and is recorded on the console data recorder. In the event that the operator does not intervene to manually scram the reactor in response to available indications, exceeding the LSSS of 510 °C fuel temperature specified in TS 2.2 will be prevented automatically by the initiation of a scram signal generated by the fuel temperature channel.

Irradiation facility indication and control of irradiation facility status is available on the console for the pneumatic system and the beam ports. For the pneumatic system, a switch on the console turns the fan blower on and off. In the event that the operator suspects a malfunction or loss of control of the system or samples, the reactor operator can turn the fan blower off. This will prevent a sample from being inserted into the reactor or prevent the return of a sample to the receiving station. Each of the four beam ports may internally contain a wooden plug. A control room annunciator provides indication if the plugs have not been properly installed.

7.2 Design of Instrumentation and Control Systems

The nuclear instrumentation used at the OSTR provides the operator with information needed to properly manipulate the nuclear controls, and initiates automatic protective (scram and interlock) functions.

7.2.1 Design Criteria

Two essential ingredients of an I&C system are reliability and redundancy. Reliability is provided by daily checks and tests for operability, and calibrations are performed on a predetermined schedule in accordance with facility procedures. Redundancy is achieved through multiple instruments and safety systems performing similar functions in all modes of reactor operation (e.g., high-power and high-fuel temperature scrams). Most of these I&C systems were manufactured by GA or other industrial manufacturers of nuclear equipment and have been used successfully with TRIGA reactors throughout the world. The I&C system provides the following:

- information on the status of the reactor and related systems
- the means for insertion and withdrawal of control rods
- automatic control of reactor power level
- the means for detecting over-power, excessive rate of change of power, high-fuel temperature, loss of operability of the power measuring channels, loss of detector voltage, and automatic shutting down the reactor to terminate operation
- radiation monitoring.

7.2.2 Design-Basis Requirements

The fuel temperature and power level scrams provide protection to ensure that the fuel temperature limits will not be exceeded. At OSTR fuel temperature is measured by one of three thermocouples that are imbedded in a single IFE. Since the IFE is located in the area of the core that experiences high-fuel element power levels, it will be exposed to high fuel temperature. TS 2.1 sets the safety limit for the TRIGA fuel at 1150 °C (2100 °F) under any mode of operation. The LSSS as defined in TS 2.2 is 510 °C (950 °F). Exceeding this limit causes a scram of the reactor.

7.2.3 System Description

The I&C system used at the OSTR is composed of process and control instrumentation channels that provide the means to safely control the reactor and to avoid or mitigate accidents. The system provides the operators with audible and/or visual indications of key operating parameters and the means to manually scram the reactor at their discretion. Actuation of the scram logic will occur automatically if setpoints established in the TSs are exceeded. In addition, the I&C system includes various interlocks that prevent a particular action from occurring unless all the prerequisites for that action are satisfied. Required reactor measuring channels are given in TS 3.2.2, Table 1.

Instruments that monitor the status of the reactor include the following:

- Log Power Level (also called wide-range log channel). This channel uses the output of the fission detector and provides indication of both reactor period and power from source level (1 mW(t)) to full steady-state power (1.1 MW(t)). The wide-range log and period signals are displayed on indicators and reactor power is recorded by the console data recorder. In addition, the wide-range log channel provides a rod withdrawal prohibit if the neutron count rate is not greater than 2 counts per second (cps) and prevents pulsing when power is above 1 kW(t). The channel can also produce a reactor power period signal. This channel is required by the TSs in steady-state and square-wave mode.

- Linear Power Level (also called wide range linear channel). This channel detects, displays, and records reactor power level from source level (1 mW(t)) to full steady-state power (1.1 MW(t)). The channel measures the output of the fission detector that is amplified and conditioned to provide a signal that is proportional to power level. The output signal is displayed at the control console and recorded on the console data recorder. In addition, the output signal is used by the servo system for automatic control of reactor power. This channel is required by the TS in steady-state and square-wave mode.

- Safety Channel (called power level in TSs— two required). This channel provides indication of reactor power from 10 kW(t) to 1.1 MW(t) and will initiate an automatic scram of the reactor if reactor power exceeds preset limits. During steady-state operations an automatic scram will be initiated if reactor power reaches the TS limit of 1.1 MW(t) or upon a loss of detector high voltage. This channel is required by the TSs in steady-state and square-wave mode.

- Percent Power Channel (called power level in TSs— two required). This channel provides a redundant function to the safety channel. The channel provides indication of reactor power from 10 kW(t) to 1.1 MW(t) and will initiate an automatic scram of the reactor if reactor power exceeds preset limits. During steady-state operations, the TSs require that an automatic scram be initiated at a reactor power at or below 1.1 MW(t) (the licensee normally sets this scram at 1.06 MW(t)) or upon a loss of detector high voltage. In addition, a nonoperable condition (defined as test switches activated, loss of operating voltage, or loss of detector high voltage) will initiate a scram. This channel is required by the TSs in steady-state and square-wave mode.

- Pulse Mode Channel (also called Nvt circuit). In pulse mode, this channel provides indication of peak power resulting from a pulse. This information is also recorded on the console data recorder. This channel is required by the TSs in pulse mode.

- Fuel Element Temperature. The primary safety parameter for a TRIGA reactor is fuel temperature. The fuel temperature channel provides indication of fuel temperature and records maximum fuel temperature during pulses. This channel consists of an IFE, temperature indicator, and electronic data recorders (pulse mode). In addition to providing fuel temperature information, the fuel temperature channel forms part of the scram logic circuitry. An automatic scram will be initiated when fuel temperature reaches the maximum allowable fuel temperature specified in the TSs (510 °C). This channel is required by the TSs in all modes.

- Servo Control System. In the steady state mode, reactor power may be controlled manually or automatically. In automatic mode (i.e., servo) the wide-range linear channel provides the power-level input to a servo controller which is then compared to a preset signal. The servo controller generates a signal to drive the regulating control rod as required to maintain a constant preset power-level. This system is not required by the TSs.

The following apply to the TS 3.2.2 required minimum measuring channels:

(1) Any single Linear Power Level, Log Power Level or Power Level measuring channel may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration.

(2) If any required measuring channels becomes inoperable while the reactor is operating for reasons other than that identified in Technical Specification 3.2.2 (1) above, the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shutdown.

To perform a channel check, test, or calibration on the power-level measuring channels, it may be necessary to take the channel offline during operation. The surveillances are performed infrequently for short periods of time. Four power-level channels are subject to TS 3.2.2(1). During the surveillance, three redundant power measuring channels remain in service. Because this TS allows the performance of required surveillances while maintaining redundancy of power-level channels it is acceptable to the staff.

Failures of the required measuring channels, if they do not result in a reactor scram, are immediately self-revealing to the reactor operator. TS 3.2.2(2) allows the operator to assess conditions and either quickly correct the cause of the problem or shut down the reactor. Shutting down the reactor is the current action taken by the operator which takes several minutes. This TS will allow for continuity of operation and will prevent equipment failures without safety significance from becoming violations of the TSs. If a failure occurs that is not self-revealing such that the failure lasts for more than 5 minutes, the operator must take immediate action to shut down the reactor upon discovery.

The I&C systems at the OSTR are well designed and maintained. Redundancy in the important ranges of power measurements by nuclear instrumentation is ensured by overlapping ranges of the wide-range log and linear power channel and the safety and percent-power channels. All important nuclear process variables are monitored and conveniently displayed at the reactor console. During a site visit, the staff observed the operation of the I&C system and found that the system provided the types of information necessary to allow the reactor operator to safely and reliably control the reactor power. Based on the above considerations and years of safe operation with the current systems, the staff finds the I&C designs to be acceptable for operation of the OSTR during the period of the renewed license.

7.3 Reactor Control System

Controls for the reactor and reactor systems, indication of key reactor operating parameters, and a means for selecting the reactor operating mode (steady state, square wave or pulse) are provided at the reactor console. Control of the nuclear fission process is achieved using three standard control rods (shim, safety, and regulating rod) and a transient rod. Shutdown (scram) capability is provided by insertion of all four rods. Control rod withdrawal is accomplished by rack and pinion electro-mechanical drives for the standard control rods and a pneumatic-electro-mechanical drive for the transient rod. Shutdown of the reactor (scram) may be accomplished manually, at the discretion of operators in response to observed conditions, or automatically, in response to the scram logic. The position of individual rods is displayed at the main reactor console.

For the transient rod, a scram signal de energizes a solenoid valve on the pneumatic cylinder which vents compressed air and allows the transient rod to fall, by gravity, into the core.

7.3.1 Shim and Safety Control Rods

The drive mechanism for insertion and withdrawal of the shim and safety rods consists of a rack-and-pinion gear system that is electrically driven by a two-phase reversible alternating current (ac)-powered gear reduction motor. Limit switches are used to indicate the up and down position of the magnet and the down position of the rod. Indication of rod position at the reactor console is derived from a 10-turn potentiometer that is coupled to the pinion shaft and displayed at the reactor control console. An interlock ensures that only one control rod can be manually withdrawn at a time.

7.3.2 Regulating Control Rod

The regulating rod is used to maintain constant power during steady-state operations. The control system for the regulating rod allows operators to control its position manually or automatically via the servo controller. The regulating rod control assembly is similar to the shim and safety rods except that the rod drive is coupled by a chain and sprocket to a direct current (dc) stepper motor. There are no specific conditions required to transfer operational control from steady state (manual) to automatic.

7.3.3 Transient Control Rod

In pulse mode, a large power excursion of short duration (a pulse) is produced by rapidly withdrawing the transient rod from the core. This is accomplished by the application of compressed air to the bottom surface of a piston located within an externally threaded, single acting, pneumatic cylinder. The reactivity inserted by the transient rod is dependent on the position of the cylinder prior to applying air. The pneumatic cylinder may be raised or lowered by a motor-driven ball screw nut that acts on the cylinder's external threads. With air applied, energizing the motor in the up or down direction will cause the cylinder, piston, and control rod to move up or down as a unit.

The application of compressed air forces the piston (and the mechanically connected transient rod) to move rapidly upward to contact a shock absorber. As the piston rises the air above the piston is exhausted through vents in the upper portion of the cylinder. Compressed air needed for operation of the pneumatic cylinder is provided by a compressor. During steady-state reactor operations, the transient rod is held against the rod drive carriage by high-pressure air and its position is controlled by the electromechanical portion of the drive mechanism.

7.3.4 Interlocks

Several interlocks are hard wired into the control system circuitry to prevent improper operation. These interlocks are typical of NRC-licensed TRIGA reactors. The interlocks required by the TSs are given in Table 3 of TS 3.2.3 as summarized below:

- Control rod withdrawal is permitted only when the neutron count rate, as measured by the wide-range log power channel, is greater than 2 cps. This prevents reactor startup unless there is reliable indication of the neutron flux level. This interlock is required by the TSs and is in effect only in steady-state mode.

- A permissive interlock prevents the reactor from being pulsed when power exceeds 1 kW(t). This helps to ensure that the fuel temperature safety limit (1,150 °C), will not be exceeded. This interlock is required by the TSs and is in effect only in pulse mode.

- The shim, safety, and regulating rods may not be withdrawn when the reactor is in the pulse mode. This interlock ensures that all pulse reactivity is due to only the transient rod while in pulse mode. Otherwise, control rod removal in pulse mode could add to the inserted reactivity of the transient rod and create an opportunity for exceeding the reactivity insertion design limit of $2.59. This interlock is required by the TSs and is in effect only in pulse mode.

- The simultaneous withdrawal of two control rods is not allowed, thus limiting the reactivity addition rate. The single rod withdrawal interlock prevents a situation whereby the ramped (nonpulse) reactivity insertion rate could exceed that analyzed in Section 13.2.2.2.2 of the SAR through the simultaneous movement of multiple control rods. This interlock is required by the TSs and is in effect in steady-state and square-wave mode.

- Air may not be applied to the transient rod unless it is fully inserted. This prevents the inadvertent pulsing of the reactor while in the steady-state mode. This interlock is required by the TSs and is in effect in the steady-state mode.

- The transient rod interlock limits the rod reactivity insertion to below \$2.55 as required by TS 3.1.4 (note the design limit is \$2.59) and to within values assumed in the accident analysis (SAR Chapter 13.2.2.2.1). The interlock is designed such that, if the electrical (i.e., limit switch) portion fails, a mechanical interlock (i.e., metal bracket) will still keep the reactivity insertion below the criterion. This interlock is required by the TSs and is in effect in pulse and square-wave mode.

- Additional rod withdrawal interlocks are provided to prevent the operator adding reactivity in various operating situations (e.g., period/log test or detector current selector switches are out of the "Operate" position or fuel temperature selector switch is in a position not reading an IFE). These interlocks are part of the interlock circuit associated with the wide-range log power channel (low neutron count rate interlock). These interlocks are not required by the TSs.

The staff concludes that the design of the reactor control system ensures that it can maintain the reactor in a shutdown condition, change reactor power, maintain operation at a fixed power level, and insert a pulse in accordance with reactivity amounts and rates as derived from the SAR analysis and in accordance with the TSs.

7.4 Reactor Protection System

Reactor safety systems are defined in TS 1.22 as follows:

> Reactor safety systems are those systems, including their associated input channels, which are designed to initiate, automatically or manually, a reactor scram for the primary purpose of protecting the reactor.

As specified in TS 3.2.3, Table 2, the minimum reactor safety channels required to be operational include the following:

- Console Scram Button. The manual console scram allows the operator to shut down the reactor if an unsafe or abnormal condition arises. The manual scram is functional in steady-state and square-wave mode. It has no specific setpoint and can be manually initiated by the reactor operator at any time.

- Power Level. Two power-level channels are provided to ensure that the power level is limited to protect against abnormally high fuel temperatures. These channels provide diversity and redundancy to the fuel element temperature channel. The safety channel and the percent-power channel are each separate and individual power-level channels. The signals for each of these channels come from separate ion chambers and are separate and unique. The setpoint for the safety and percent-power channels are normally set to 1.06 MW(t), which is below the licensed power of 1.1 MW(t) allowing for expected and observed instrument fluctuations. Analysis in Section 13 of the SAR shows that this setpoint is sufficient to prevent reactivity additions from control system failures from exceeding the limiting reactivity addition of \$2.59. Two channels are required during steady-state and square-wave mode.

- Fuel Element Temperature. The fuel element temperature channel provides protection to ensure that the reactor can be shut down before the safety limit on the fuel temperature is exceeded. The fuel element temperature scram causes an automatic scram if the IFE senses fuel temperature in excess of the LSSS set point of 510 °C. This safety channel is required during steady-state and square-wave mode.

- High Voltage. The loss of high voltage to the safety channel detector, percent-power channel detector, or fission chamber (which is the source of signal for the wide-range log and linear channels) will result in a reactor scram. The high voltage scram occurs when the high voltage for any of the three detectors drops to 25 percent or less of the nominal operating voltage indicating a detector failure. This safety channel is required during all modes of operation.

- Preset Timer. The preset timer ensures that the reactor power level will reduce to a low level after pulsing. The purpose of the scram is to preclude an unintentional restart or ramped increase to some equilibrium power after a pulse. The channel causes a scram of the transient rod within 15 seconds after a pulse. This safety channel is required during pulse mode.

The following applies to the TS 3.2.3-required minimum reactor safety channels and minimum interlocks:

(1) Any single Linear Power Level, Log Power Level or Power Level safety channel or interlock may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration.

(2) If any required safety channel or interlock becomes inoperable while the reactor is operating for reasons other than that identified in Technical Specification 3.2.3 (1) above, the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shutdown.

To perform a channel check, test, or calibration on the power-level safety channels, it may be necessary to take the channel off line during operation. The surveillances are performed infrequently for short periods of time. Four power-level safety channels are subject to TS 3.2.3(1). During the surveillance, three redundant power-level safety channels remain in service. Because this TS allows the performance of required surveillances while maintaining redundancy of power-level safety channels it is acceptable to the staff.

Failures of the required safety channels or interlocks, if they do not result in a reactor scram, are immediately self-revealing to the reactor operator. TS 3.2.3(2) allows the operator to assess conditions and either quickly correct the cause of the problem or shut down the reactor. Shutting down the reactor is the current action taken by the operator which takes several minutes. This TS will allow for continuity of operation and will prevent equipment failures without safety significance from becoming violations of the TSs because they are corrected or the reactor shut down. If a failure occurs that is not self-revealing such that the failure lasts for more than 5 minutes, the operator must take immediate action to shut down the reactor upon discovery.

An external scram set-point is also available for experiments that are used in situations when the reactor must automatically shut down to reduce the radiation field or control some other aspect of the experiment.

As part of the license renewal application, the licensee requested that the wide-range log power-level channel, which produced a short reactor power period scram at a reactor period of 3 seconds or less, be removed as a required safety channel. There is no safety significance to the period scram signal in a TRIGA reactor. The only function it serves is to prevent the operator from increasing power with a reactor period that is too small to effectively "roll" power over as they are approaching full power. This scenario may result in a high reactor power scram. However, the reactor was designed for pulsing which bounds any short reactor period scenario. Therefore, the elimination of the requirement for a short period scram is acceptable to the staff.

Surveillance requirements for the reactor protection system are given in TS 4.1.a. and TS 4.2.d., e., and f. as follows:

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

4.2 d. A channel check of each of the reactor safety system channels for the intended mode of operation shall be performed prior to each day's operation or prior to each operation extending more than one day.

4.2 e. A channel test of each item in Tables 2 and 3 in section 3.2.3, shall be performed semi-annually.

4.3 f. A channel calibration of the fuel temperature measuring channel shall be performed annually.

Channel check, test and calibration are defined in TS 1.4, 1.5 and 1.3 as follows:

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.

Channel Test: A channel test is the introduction of a signal into the channel for verification 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 which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall include a Channel Test.

These surveillances and their intervals are consistent with the guidance found in ANSI/ANS-15.1 and the intervals used at similar research reactors. Experience has shown that the surveillance frequencies will ensure performance and operability of the systems or components. The staff finds that the specified intervals provide reasonable assurance that I&C component failure and degradation will be detected in a timely manner, and that specified calibration frequencies are adequate to prevent significant drift in instrument setpoints and detection ranges.

The reactor protection system is designed to prevent or mitigate hazards to the reactor and to prevent the escape of radiation. The protection channels and protective responses are sufficient to ensure that no safety limit or LSSSs specified in the TSs will be exceeded, and that the full range of reactor operation poses no undue radiological risk to the health and safety of the public, the facility staff, or the environment.

7.5 Control Console and Display Instruments

The physical layout of the control console and display instruments are within easy sight and reach of the reactor operator. An annunciator panel, located above the console, provides audible and visual warning of off-normal conditions, including the alarm status of various instrumentation.

Irradiation facility status is available on the console for the pneumatic system and the beam ports. For the pneumatic system, a switch on the console turns the fan blower on and off. In the event that the operator suspects a malfunction or loss of control of the system or samples, the reactor operator can turn the fan blower off. Plugs in each of the four beam ports are instrumented such that the reactor operator has indication when the plugs are improperly installed. Irradiation facilities or reactor experiments are instrumented and annunciated on a case-by-case basis.

The staff concludes that the annunciator and alarm indications on the control console give assurance that the status of systems important to adequate and safe operation will be presented to the reactor operator. The staff compared the general arrangement and types of controls and displays provided by the control console to those at similar research reactors and found that the designs are similar. The staff observed the control console during a site visit and found that the control console provides the reactor operator with the types of information and controls necessary to facilitate reliable and safe operation of the reactor.

7.6 Radiation Monitoring Systems

Radiation levels are monitored at various locations throughout the reactor facility. As discussed in Chapter 11 of this SER, there are four radiation monitoring channels required by TS 3.7.1. Airborne radioactivity is monitored by a continuous air monitor (CAM) and a reactor bay exhaust stack monitor. Each unit consists of a monitor, recorder, and associated alarm and warning circuitry. The CAM measures the radioactivity of airborne particulates and gases (gas monitoring is not required by the TSs) at the reactor top. The stack exhaust monitors gaseous and particulate effluents from the reactor bay exhaust stack. A high alarm on the stack monitor will result in a shutdown of the reactor bay ventilation system. Radioactivity in the primary water is monitored by the normally continuously operating primary water activity monitor. This system consists of a local GM detector and a count rate meter located at the control console. A significant increase in primary water activity will actuate an alarm at the reactor console. However, this monitor is not very sensitive and any fission product leakage would likely be detected and alarmed by other monitors (reactor top CAM, first; the stack CAM, second; and the area radiation monitor on the demineralizer tank, third). In addition, the water is sampled monthly for activity with gamma spectroscopy and other methods for beta activity.

Based on its review of the SAR and observations made during an onsite visit, the staff concludes that the radiation monitoring systems described in the SAR provide reasonable assurance that all anticipated sources of radiation will be identified and accurately evaluated.

7.7 Conclusions

On the basis of its evaluation of the information presented in the licensee's SAR, the staff concludes as follows:

- The I&C systems at the OSTR are well designed and maintained. Redundancy in the important ranges of power measurements by nuclear instrumentation is ensured by overlapping ranges of the wide-range log and linear power channel and the safety and percent channels. All important nuclear process variables are monitored and conveniently displayed at the reactor console.

- The design of the reactor control system ensures that it can maintain the reactor in a shutdown condition, change reactor power, maintain operation at a fixed power level, and insert a pulse in accordance with reactivity amounts and rates as derived from the SAR analysis and in accordance with the TSs.

- The reactor protection system is designed to prevent or mitigate hazards to the reactor and to prevent the escape of radiation. The protection channels and protective responses are sufficient to ensure that no safety limit or LSSSs specified in the TSs will be exceeded, and that the full range of reactor operation poses no undue radiological risk to the health and safety of the public, the facility staff, or the environment.

- The radiation monitoring systems described in the SAR provide reasonable assurance that all anticipated sources of radiation will be identified and accurately evaluated.

8. ELECTRIC POWER SYSTEMS

8.1 Normal Electrical Power Systems

The OSTR receives 4-kilovolt (kV), three-phase, 60-hertz, electrical power from two separate services within the campus distribution system via a substation. Power to the reactor facility is supplied from the substation to two transformers. One transformer provides power to the primary and secondary cooling pumps, the cooling tower, and the air compressor. The second transformer is the power supply for all other reactor electrical loads and the majority of electrical loads in the Radiation Center complex. A breaker on this transformer feeds a distribution panel that distributes power to electrical loads throughout the reactor building. Power for reactor control and instrumentation is conditioned by a dedicated 3.1 KVA uninterruptible power supply (UPS) that can also be powered by a battery.

As discussed below, electrical power is not required to safely shut down the reactor and maintain the reactor in a safe-shutdown condition.

8.2 Emergency Electrical Power Systems

As noted in Section 13.9 of this SER, a loss of the normal ac power source will initiate an automatic scram of the reactor by loss of power to the control rod electromagnets and three-way solenoid valve (for the transient rod) causing the control rods to drop, by gravity, into the core. A loss of electric power places the reactor in a safe-shutdown condition. Confirmation of control rod insertion can be accomplished by visual observation of the reactor core. The primary and secondary cooling system pumps will also stop, but the water in the reactor tank is a sufficient heat sink for decay heat from the reactor. Radiation surveillances would be done with handheld instruments if power failed.

While not required by the TSs, the OSTR has an emergency electrical power system. Emergency electrical power can be used to monitor the orderly shutdown of the reactor following loss of normal power. The emergency power system at the OSTR consists of a 6.5-kW propane-fueled generator, the 3.1 KVA UPS, and associated switchgear. Electrical loads supplied by this power source are divided into System A and System B electrical loads. System A loads include the reactor console and control cabinets, the public address system amplifier, and the ventilation system controller. The UPS battery power can carry these System A loads for an amount of time sufficient to allow the emergency generator to start and supply power. An installed bypass switch allows maintenance to be performed on the UPS without losing power to the System A loads. System B loads include partial lighting in the control room and adjacent hallway, closed-circuit television monitors, the stack radiation monitor, and the building fire alarm panel. If normal power is interrupted, the generator will automatically start and supply power to these loads as well as the UPS. Unlike the System A loads, the System B loads will be momentarily interrupted. Battery-powered emergency lighting is also provided to facilitate an orderly evacuation of the facility, if required.

8.3 Conclusions

On the basis of its review, the staff concludes that the normal electrical power system at the OSTR facility provides reasonable assurance of adequate operation. In addition, the staff concludes that loss of normal electrical power will lead to safe shutdown of the facility and that emergency power is not required to maintain the reactor in a safe-shutdown state.

9. AUXILIARY SYSTEMS

9.1 <u>Heating, Ventilation, and Air Conditioning System</u>

The purpose of the heating, ventilation, and air conditioning (HVAC) system (which includes the reactor building which acts as a confinement) is to provide a controlled and monitored pathway for airflow through the reactor building. Pressure, temperature, and humidity of air within the reactor bay of the OSTR are controlled by the HVAC system that is independent of the Radiation Center and laboratory ventilation systems. As discussed in Section 6 of the SER, the HVAC system is not considered an engineered safety feature. The accident analysis in Chapter 13 of the SAR does not depend on the HVAC system to mitigate the consequences of accidents. The HVAC system serves a radiation protection function by minimizing and controlling the release of airborne radioactive effluents during normal operation and accident conditions.

As air enters the reactor bay, it is filtered and conditioned by a heat exchanger that utilizes either hot water for heating or chilled water for cooling that is supplied by OSU campus utilities. The reactor bay is kept at a negative pressure in relation to the outside ambient air by an electronic control system. The HVAC system is designed to automatically contain airborne radioactive material. Exhaust gas and particulate detectors continuously sample the outgoing airflow for radioactive content and shut down the ventilation system if a preset limit for radiation is exceeded in order to isolate the reactor bay from other areas of the Radiation Center complex and the environment. The release point of the system is a stack on the roof of the reactor building. The capability to remotely monitor the reactor bay atmosphere when the ventilation system is isolated also exists.

Additional exhaust air paths from laboratory water system vents, the pneumatic rabbit system, and the argon manifold (an experimental facility) have individual fans that are connected to the reactor bay exhaust. The argon manifold has a high-efficiency particulate air (HEPA) filter on the exhaust for ALARA purposes, but this filter is not required for accident mitigation. All potentially contaminated air is monitored before release to the environment through the building stack. The 23.5-foot stack is located on the reactor bay roof, which places it 65.5 feet above ground level. This height ensures rapid mixing and dispersion at a high elevation. The additional exhaust fans (argon manifold fan and pneumatic rabbit hood fan) are interlocked with the reactor bay supply and exhaust fans to ensure that these fans shut down when the reactor bay ventilation system shuts down. The purpose of ensuring that the hood fan turns off is to maintain a negative pressure differential between the reactor bay and the adjoining rooms. The interlock also prevents restarting the argon exhaust fan until the reactor bay supply and exhaust fans have been in operation for at least 1 minute.

TSs 5.1 c. and d. contain the fundamental design features for the HVAC system as described below:

> 5.1 c. The reactor bay shall be equipped with ventilation systems designed to exhaust air or other gases from the reactor building and release them from a stack at a minimum of 65 feet from ground level.
>
> 5.1 d. Emergency shutdown controls for the ventilation systems shall be located in the control room.

TS 5.1.c. is the basic requirement to have a ventilation system with a controlled air pathway release point. As discussed in Chapter 11 of this SER, the height of the exhaust stack helps to ensure a minimum amount of dispersion and dilution of effluents released from the stack before reaching the ground. TS 5.1.d. ensures that the reactor operator can shut down the system quickly from the control room. This allows airborne radioactive material to be retained in the reactor building, if needed. The reactor operator can isolate the system without having to go out into the reactor bay area. Because TSs 5.1.c. and d. contain the basic design requirements for the ventilation system, they are acceptable to the staff.

The LCO of the system that help to ensure that the ventilation system is in operation to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation are given in TSs 3.5.a. and b. as follows:

3.5 a. The reactor shall not be operated unless the facility ventilation system is operating and the reactor bay pressure is maintained negative with respect to surrounding areas, except for periods of time not to exceed two (2) hours to permit repair, maintenance or testing of the ventilation system.

3.5 b. The ventilation system shall be shutdown upon a high activity alarm from the Exhaust Particulate Radiation Monitor.

TS 3.5.a. contains the basic requirement that the ventilation system be in operation and the reactor bay be at negative pressure during operation. This helps to ensure that the controlled air pathway will be maintained. Shutting down the system if a substantial release of airborne radioactivity occurs helps to keep doses to members of the public ALARA.

TS 3.5.a. allows the system to be shut down for periods of time not to exceed 2 hours for repair, maintenance, or testing. This is to allow continuity of operation. While repair, maintenance, or testing will occur occasionally, the purpose of this TS is not to allow long-term operation without the HVAC being in operation. In the unlikely event of a fission product release with the HVAC shut down, the reactor top continuous air particulate monitor and reactor top area monitor would be used to detect the fission product release because the exhaust gas and particulate monitors would also be out of service. In accordance with TS 3.5.b., the operators would be required to shut the ventilation system down. However, the ventilation system would already be in the shutdown configuration. This configuration was considered by the licensee in the accident analysis discussed in Chapter 13 of the SAR. Measurements of argon-41 concentration at the reactor top during operation indicate that concentrations of argon-41 with the HVAC off would be within regulatory limits. Therefore, the staff concludes having the system shut down for repair, maintenance, or testing is acceptable because having the system shut down would not significantly impact doses to the reactor staff or public during normal and accident conditions. Because TS 3.5.a. and b. contain the basic requirements for operation of the HVAC system in a manner consistent with the ALARA principle, these TSs are acceptable to the staff.

The surveillance requirements for the system are given in TSs 4.5.a. and b as follows:

a. A channel check of the reactor bay confinement ventilation system’s ability to maintain a negative pressure in the reactor bay with respect to surrounding areas shall be performed prior to each day’s operation or prior to each operation extending more than one day.

b. A channel test of the reactor bay confinement ventilation system's ability to be secured shall be performed annually.

These surveillance requirements help to ensure that the system will operate in accordance with the design features and LCO. The surveillance intervals are based on experience which shows that they are acceptable to detect degradation of components and help ensure that the system is operating properly. Therefore, these surveillance requirements are acceptable to the staff.

Chapter 11 of the SAR analyzes the height and flow rate of the stack that exhausts facility air to the unrestricted environment and determines the dose rates for the maximum exposed person in the unrestricted environment. Chapters 11 and 13 of this SER discuss releases from the HVAC system under normal operation and accident conditions.

The NRC staff reviewed the design, operation, and TS requirements of the HVAC system. The staff concludes that the HVAC system is adequate to provide controlled release of airborne radioactive effluents during normal operations and in the event of abnormal or accident conditions. The reactor staff, researchers, and the public will be adequately protected from airborne radioactive hazards related to reactor operations. Based on the staff's review of the operational experience of the facility and TS requirements for operability and testing of the system, the staff concludes that degradation of components will be detected; therefore, there is reasonable assurance that the HVAC system discussed in this section of the SER can continue to operate safely, as limited by the TSs for the proposed license renewal period.

9.2 Handling and Storage of Reactor Fuel

Fuel handling at the OSTR consists of receiving new fuel elements, placing new fuel into storage or use, moving irradiated fuel elements within the core, and placing irradiated fuel elements into and out of storage.

A fuel element inspection tool is used to accurately inspect longitudinal growth and bowing of fuel assemblies in accordance with TS 3.1.6. Secured to the top of the reactor tank, this tool extends into the tank allowing for fuel element inspection while maintaining water shielding over the element. Fuel element bowing is checked by passing a go/no-go cylinder over each fuel element. As the fuel element is passed through this cylinder, it also contacts a spring-loaded bellows assembly. The length of a fuel element is measured by pushing down on the indexing rod until the indexing rod plug moves an incremental amount and touches the indexing plate. This places the fuel element in the required position while the lower surface displaces a plunger which is measured with a dial indicator.

Sufficient storage for the existing fuel inventory is provided. Design features for fuel storage are given in TS 5.4 as follows:

a. All fuel elements shall be stored in a geometrical array where the k-effective is less than 0.9 for all conditions of moderation.

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

The k-effective value for stored fuel of 0.9 represents a change from the pre-license renewal TS limit of 0.8. The k-effective value of 0.9 is commonly used for fuel storage, is recommended in ANSI/ANS-15.1 and NUREG-1537 and is acceptable to the staff. Because the licensee's current fuel storage facilities were designed to meet the 0.8 k-effective limit they also meet the new 0.9 limit. Because criticality and fuel temperature are controlled to acceptable limits, TSs 5.4.a. and b. are acceptable to the staff.

Stored fuel must have sufficient cooling such that the stored fuel elements will not exceed the safety limit. Different storage locations have limitations as to the heat load that can be placed in them under varying conditions of storage such as water versus air cooling. The licensee will perform calculations to show that TS 5.4.b. is met prior to placing fuel in storage.

In the past, the licensee operated an AGN-type reactor. This reactor has been permanently shut down. The reactor was decommissioned during the 1978—1980 period. Following decommissioning, the core of the AGN reactor (without the control rods which consisted of fuel) was placed in storage and the authority to possess the material was added to the OSTR license. The AGN reactor license was then terminated by the NRC in 1981. The control rods were removed and shipped to another licensee. Rolled up cadmium sheet was installed in the glory hole and the vacant control rod positions. The licensee stated that the storage configuration meets the requirements of TS 5.4.

Based on the review of the licensee's fuel handling and storage, the staff concludes that there is adequate assurance that fuel elements will be stored and handled in a safe manner.

9.3 Fire Protection Systems and Programs

Fixed thermal fire detection and manual pull stations are located throughout the OSTR complex. They alarm at an annunciator room within the Radiation Center and the OSU Department of Public Safety, which is continuously staffed, and activate an audible alarm throughout the radiation center. Fire response is provided by the Corvallis Fire Department. The Corvallis Fire Department personnel receive annual OSTR-specific training in radiological hazards and OSTR-specific familiarization training, as required under the emergency response plan. Doors to the reactor bay are fire rated. Fire extinguishers are located throughout the radiation center and OSTR. Hydrants are located next to the site boundary to the southeast and west. The floor in the reactor bay is pitched toward a trench so that fire water drains to a holdup tank, where it can be sampled and discharged to the sanitary sewer, if appropriate. This holdup tank was sized to contain the volume of the reactor pool. The staff has toured the facility and notes that housekeeping is good and combustible loading is controlled. Failures of the reactor I&C system caused by fire should cause the reactor to scram.

The staff has reviewed the fire protection systems at the OSTR which are typical for a research reactor. The staff concludes that the fire protection systems are capable of detecting, alarming, and responding to fires. Based on its review, the staff concludes that fire protection at the OSTR is acceptable.

9.4 Communication Systems

The Radiation Center has systems which allow communication within the Radiation Center. Outside telephone capability also exists. These communication systems allow the staff to carry out the communication requirements of the emergency and security plans.

Because the communications systems help ensure that Radiation Center staff can carry out the communication requirements of the emergency and physical security plan, the staff concludes that the communication systems are adequate.

9.5 Possession and Use of Byproduct and Special Nuclear Material

The license for the OSTR authorizes the receipt, possession, and use of special nuclear and byproduct materials. Special nuclear material consists of such material as the uranium-235 in the reactor fuel, fission chambers, and special nuclear material 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 polonium-beryllium and americium-beryllium neutron sources. The licensee also possesses the core of an AGN-type research reactor as discussed above. The possession limits in the license are sufficient for continuity of reactor operation. Other materials are controlled under the State of Oregon Broad Scope License ORE-90005. Designated storage areas for the possession and use of special nuclear and byproduct material as well as source material under the State license are established in the facility. Handling of these materials is controlled by existing operational and health physics procedures. These procedures are written to comply with 10 CFR Part 20 and the OSTR ALARA program described in Chapter 11 of the SAR. These procedures and processes provide reasonable assurance that no uncontrolled release of radioactive material to unrestricted areas will occur.

The restricted area is defined by TS 5.1.a. and all activities performed within this area fall under the jurisdiction of the reactor license. All of the laboratories referred to in Section 9.1.2 of the SAR are located within the reactor building and are 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 licensee states that all current material possession limits will be carried over into the renewed license unchanged. As is current practice, the staff is clarifying the byproduct possession requirement to allow the separation of byproduct material produced in experiments. The limitation on separation of special nuclear material and byproduct material produced in the reactor fuel remains in the license unchanged. The staff has reformatted the license conditions to make them easier to understand and read. The staff has reviewed the possession limits of the license and concludes they are acceptable for continued operation of the reactor.

Based on the staff's review as discussed above and the acceptable results of the NRC inspection program, the staff concludes that the licensee has procedures and equipment in place to safely receive, possess, and use the materials authorized by the reactor license.

9.6 Facility Compressed Air System

Filtered and regulated compressed air is provided to the reactor building for general use and to operate the transient rod air cylinder. The primary system utilizes a 7.5-hp 480 VAC, three-phase skid mounted two-cylinder two-stage reciprocating compressor, water separator/cooler, and a pressure tank. An outlet pressure regulator maintains the supply air pressure at 80 pounds per square inch gauge. Low pressure alarms annunciate on the reactor control room annunciator panel to provide indication of inadequate air supply. Backup service can be provided, if needed, by cross connecting the distribution system to the two compressor systems used for the Radiation Center. Loss of air or deenergizing the three-way solenoid control valve scrams the transient rod.

The staff has reviewed the facility compressed air system and backup. Because compressed air is not needed to prevent malfunctions or reactor accidents, initiate safe reactor shutdown, or prevent uncontrolled release of radioactive material, the staff concludes the system is acceptable.

9.7 Conclusions

On the basis of its evaluation of the information presented in the licensee's SAR, the staff concludes the following:

- The HVAC and confinement systems are designed so that the release of airborne radioactive effluent will be controlled and in compliance with the regulations.
- There is adequate assurance that fuel elements will be stored and handled in a safe manner.
- The fire protection systems are capable of acceptably detecting, alarming and responding to fires.
- Communications systems are adequate to meet emergency plan and physical security plan requirements.
- License conditions for receipt, possession, and use of nuclear material is acceptable.
- The compressed air system is acceptable.

10. EXPERIMENTAL FACILITIES

10.1 Introduction

The OSTR is primarily used as a teaching facility and serves as a source of ionizing and neutron radiation for research and producing different radioisotopes. Experimental facilities include beam ports, thermal column, thermalizing column, pneumatic transfer system, central thimble, vertical irradiation tubes, rotating rack, and argon production facility. In accordance with Subsection 104c of the Atomic Energy Act, as amended (AEA), the facility license allows the licensee to conduct widespread research through the development, review, and approval of new experiments and experimental facilities. The process for carrying out experiments without prior Commission approval is given in 10 CFR 50.59, "Changes, Tests and Experiments." The TSs control the conduct of experiments, and provide means for technical and safety review.

10.2 Experimental Facilities

The following sections describe the experimental facilities at the OSTR and their use. TS 1.14 contains the definition of irradiation facilities. The OSTR experimental facilities allow gamma and neutron irradiation of materials. The design and location of the different facilities provide a spectrum of neutron energies and fluxes. Experimenters can extract radiation beams from the reactor core and perform irradiations in the active region of the core or in the moderator near the core. The experimental facilities are comparable in design, construction, utilization, and purpose to experimental facilities at other similar research reactors. The experimental facilities have been successfully and safely utilized during the period of the current facility license.

Accidents such as loss of coolant and reactivity insertion that experimental facilities could be subject to are discussed in Chapter 13 of this SER. The design, construction, and utilization of the experimental facilities are such that these accidents are extremely unlikely. Chapter 11 of this SER discusses radiation hazards at the OSTR. Access to experimental facilities is controlled by the use of operating and radiation protection procedures. Use of appropriate radiation detection equipment, radiation protection practices (including the ALARA program), and established experiment review procedures provide reasonable assurance that doses from experimental facilities will meet the requirements of 10 CFR Part 20 for personnel and members of the general public.

10.2.1 Beam Port Facilities

Three radial and one tangential tube make up the beam port facilities. The inner sections of the beam ports are constructed of aluminum and the outer stepped portions are made of steel and are cadmium plated to reduce corrosion. Each port is closed on the reactor end of the port, since they are surrounded with reactor pool water. A shutter assembly inside the tube has a lead-lined steel door with a rubber gasket and screw clamps that provide a watertight seal when closed to prevent rapid loss of water from the reactor tank if a port were to develop a serious leak. Axial shielding of each beam port is provided by an inner and outer port plug. The inner plug is cone shaped to fit snugly into the beam port step. It consists of an aluminum case, with a laminate of concrete, boral and lead to prevent both gamma and neutron streaming out of the ports when not in use. There is also an outer wooden plug that provides minimal shielding but is equipped with a pressure switch on the inner surface that can only be deactivated by contact with a specific location on the inner shield plug. This circuit is connected to an annunciator circuit in the control room that provides indication if either beam plug is improperly installed. The outer end of each beam port has a recessed shutter that provides an extension to the beam port when open and provides additional radiation shielding when closed. Each outer shutter is lead filled to provide limited gamma shielding when the shield plugs are removed. They are positioned manually, sometimes with a reach rod. Gamma radiation area monitors are located at strategic locations in the reactor bay area, which includes the experimental area, to alert personnel working there of any change in radiation levels, since an open beam port provides a pathway for radiation leakage. These monitors have both local and control room alarm functions.

10.2.2 Thermal Column

The thermal column consists of two sections. The outer section is a boral-lined, graphite-filled 4-foot by 4-foot aluminum container that consists of a track-mounted door that opens into the experimental area, a graphite assembly, and shielding. Surfaces in contact with the concrete are wrapped with plastic for corrosion protection. The tube section is welded directly to the pool liner and protrudes through the reactor pool, with the other end located directly against the reflector at the core centerline. To reduce the capture gamma flux in the column and the neutron flux in the surrounding concrete, sheets of boral are placed between the aluminum column and the graphite blocks inside the casing. The graphite assembly contains the irradiation positions that consist of horizontal access ports. The consequences from any loss-of-coolant accident created from the failure of the thermal column are bounded by the analysis performed in Section 13.2.4 of the SAR and discussed in Chapter 13 of this SER.

Access to the graphite assembly is gained through the thermal column door. This door is filled with concrete and has a boral sheet affixed to the inner surface of the door. The concrete provides shielding from gamma radiation and the boral provides shielding from thermal neutrons, which would be the major component of any neutron leakage emanating from the column.

10.2.3 Thermalizing Column

The thermalizing column is similar in all respects to the thermal column, except for its orientation in the reactor and its size. Since it is a smaller facility than the thermal column, the radiation and safety implications from this facility are bounded by those of the thermal column.

10.2.4 Pneumatic Transfer System

The pneumatic transfer system rabbit facilitates the rapid transport of small sealed samples to and from the core region and is used for the production of short-lived radioisotopes, primarily to support neutron activation analysis. The rabbit tubing is made of aluminum and penetrates the core to the grid plate with the same orientation as a fuel element. Design of the system is such that it is always under negative pressure relative to the reactor building. Therefore, all air leakage will be inward. This minimizes the argon-41 airborne radiation concentration in the experimental areas. Air supplied from a blower provides the pressure differential that transports the samples. The air from the transfer system passes through a HEPA filter, which minimizes particulate activity, and is then vented from the blower to the stack.

The control assembly is located in the radiochemistry laboratory, which is also the specimen sender/receiver location. By controlling the position of various solenoid valves, air is redirected through the system and either moves the specimen capsule in or out of the reactor. There is also a cutoff switch to the blower motor located in the control room. This precludes unauthorized use by removing the transport capability of the system.

10.2.5 Central Thimble

The central thimble is a long vertical aluminum tube that extends from the reactor bridge through the grid plate to the safety plate below, which supports it. It is located in the center of the core. To provide rigidity, a hexagonal section is mounted to the thimble and affixed to the upper grid plate. The central thimble is used when filled with pool water. However, by placing a special cap on the top of the tube and applying pressurized air from the top of the thimble tube, water in the tube can be forced through small holes near the bottom of the thimble to create an experimental neutron beam.

10.2.6 Vertical Irradiation Tubes

The ICIT and CLICIT are unfueled and similar in shape to the fuel elements. They are used to provide experimental access to different grid locations within the core. All samples placed in either irradiation tube must conform to the material and reactivity requirements specified in the TSs. Experiments performed in both of these tubes are classified as secured experiments.

10.2.7 Rotating Rack

This device rotates samples in a well within the core reflector to produce uniform irradiation of up to 40 samples simultaneously. The assembly consists of a rotary specimen rack, a specimen removal chute, tube and drive-shaft assembly, and drive and indicator assembly. Positioning of the rotor assembly is by a motor/gear arrangement that is mounted from the reactor bridge.

Experiments placed in the rotating rack are considered secured experiments because the experiments will not be moving into or out of the core during reactor operation. Additionally, extensive experience has shown that experiments in the rotating rack have little (i.e., approximately $0.10 at most) or no measurable reactivity worth from the movement of the rack regardless of the material composition. It has been shown in Section 13.2.2.2 of the SAR and discussed in Chapter 13 of this SER that the reactor can accommodate rapid reactivity additions of up to $2.59 and the reactor safety channels adequately protect the reactor from ramped reactivity insertions.

10.2.8 Argon Production Facility

Highly purified argon-40 gas is irradiated in a specialized chamber inside beam port number 4 to produce specific amounts of argon-41. The argon-41 is transferred from the irradiation chamber through 0.25 inch lines to a liquid nitrogen condensing chamber. From the condensing chamber the argon-41 is then transferred to containers for shipping. Valves located inside the beam port are operated by reach rod to limit personnel radiation exposure. Equipment external to the beam port is enclosed within a concrete and lead shielded structure to minimize radiation exposure. Inside, the chamber consists of a 2-liter stainless-steel container with 1/8-inch lines that connect to the transfer system, which is exterior to the biological shield. These lines allow the transfer of argon past both shield plugs, which are also located within the beam port. Typical operating pressure of this system is 40 pounds per square inch atmosphere. Since this system is evacuated prior to each use, any loss of system integrity would be immediately observed.

10.3 Experiment Limits

Since a variety of experiments may be done at the OSTR, restrictions are placed on them to limit the impact they may have on fuel temperature and the amount of radiation that could be released if a failure were to occur. Experimental limits are placed on reactivity, materials, and failures and malfunctions. Experiment malfunction is discussed in Section 13.8 of this SER.

The terms “experiment,” “movable experiment,” “unsecured experiment,” and “secured experiment” are defined in TS 1.10 as follows:

> Experiment: Any operation, hardware, or target (excluding devices such as detectors or foils) which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of their design to carry out experiments is not normally considered an experiment. Specific experiments shall include:
>
> a. Secured Experiment: A secured experiment is any experiment 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, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.
>
> b. Unsecured Experiment: A unsecured experiment is any experiment or component of an experiment that does not meet the definition of a secured experiment.
>
> c. Movable Experiment: A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the core while the reactor is operating.

These are standard definitions used in research reactor TSs and are therefore acceptable to the staff. The definitions of secured, unsecured, and movable experiment are used primarily with experimental reactivity limitations.

TS 3.8.1 limits the effect that all experiments can have on reactivity as described below:

> The reactor shall not be operated unless the following conditions governing experiments exists:
>
> a. The absolute value of the reactivity worth of any single unsecured experiment shall be less than $0.50; and
>
> b. The sum of the absolute values of the reactivity worths of all experiments shall be less than $2.55.

Unsecured experiments are not moved in the reactor during operation but are not restrained like secured experiments. So it is possible that an unsecured experiment may move during operation. Movable experiments (which are also considered unsecured experiments) can be loaded and unloaded from the reactor while the reactor is operating. Because they are not constrained and can be moved during reactor operation, movable experiments are limited to reactivity values less than that which would cause prompt criticality ($1.00). The reactivity limit of $0.50 for single unsecured experiments is designed to prevent prompt criticality from occurring and is substantially below the analyzed pulse design limit of $2.59 (TS 3.1.4 limit of $2.55) for the most limiting case (i.e. EOL FLIP fuel (SAR Section 13.2.2.2.1)).

In TS 3.8..b., the sum of the absolute values of the reactivity worths of all experiments shall be less than $2.55. This is designed to prevent an inadvertent pulse from experiment failure from exceeding the design limit of $2.59 for EOL FLIP fuel.

The reactivity worth limit of $2.55 for the sum of the absolute value of the total worth of all experiments is also designed to ensure that the shutdown margin is met with nonsecured experiments in their most reactive state. The TSs require that the OSTR shutdown margin be at least $0.55 with the most reactive control rod withdrawn. The shut down margin is discussed in Chapter 4 of this SER.

Because the TS 3.8.1 experimental reactivity limits keep the reactivity of experiments within bounds shown to be safe, TS 3.8.1 is acceptable to the staff.

The licensee has requested that limitations on fueled experiments be temporarily removed from the renewed license TSs. The licensee is developing an updated TS on fueled experiments based on contemporary calculational techniques. The licensee will submit the new TS at some point in the future under separate request.

Materials used in experiments are controlled in TS 3.8.2 as described below:

> The reactor shall not be operated unless the following conditions governing experiments exist:
>
> a. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities less than 25 milligrams 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; and
>
> b. Experiments containing corrosive materials shall be doubly encapsulated. The failure of an encapsulation of material that could damage the reactor shall result in removal of the sample and physical inspection of potentially damaged components.

The licensee has requested the removal of a TS requirement in the renewed license allowing the irradiation of up to 0.014-pound-equivalent TNT in the laboratory area for neutron radiography. The licensee does not plan to conduct radiography of these quantities of TNT equivalent under the renewed license. This is acceptable to the staff.

TS 3.8.2-a. states that no explosive material in excess of 25 milligrams is allowed to be irradiated in the reactor. Explosive material up to 25 milligrams 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 irradiation container. The 25 milligram limit is a long standing limit discussed in Regulatory Guide 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” Issued November 1973. Example calculations of the diameter-to-thickness ratios of containers made from stainless steel and aluminum are presented in Section 13.2.6.2 of the SAR. This analysis shows that the limit is safe provided the proper container material with appropriate diameter and wall thickness is used. Therefore, TS 3.8.2.a. is acceptable to the staff.

TS 3.8.2.b. requires experiments containing corrosion material be doubly encapsulated. This requirement reduces the potential for damage to reactor components from corrosive material and is therefore, acceptable to the staff.

TS 3.8.3 places the following limits on experiments such that the radiological consequences of potential failures and malfunctions are limited to the dose limits in 10 CFR Part 20.

> Where the possibility exists that the failure of an experiment under normal operating conditions of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor bay or the unrestricted area, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor bay or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR Part 20, assuming that:
>
> a. 100% of the gases or aerosols escape from the experiment;
>
> b. If the effluent from an irradiation facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape;
>
> c. If the effluent from an irradiation facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these aerosols can escape; and
>
> d. 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, 10% of these vapors can escape.

The purpose of TS 3.8.3 is to help to ensure that potential releases of radioactive material from experiments is bounded by the dose limits in 10 CFR Part 20 for OSTR staff and members of the public. This includes failures under normal reactor operations, credible reactor accident conditions, and accident conditions in the experiment. The assumptions in the TS are standard research reactor TS assumptions and ensure that source term calculations are conservative. TS 3.8.3 is acceptable to the staff because it limits doses from potential experiment failure or malfunction to 10 CFR Part 20 limits.

The surveillance requirements for experimental limits are given in TS 4.8 as follows:

> a. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.
>
> b. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with Section 3.8 by the Reactor Operations Committee in full accord with Section 6.2.3 of these Technical Specifications, and the procedures which are established for this purpose.

The purpose of TS 4.8 is to ensure that the requirements of TS 3.8 are met. All aspects of TS 3.8 are required to be considered by the licensee, therefore TS 4.8 is acceptable to the staff.

The staff has reviewed the licensee's limitations on experiments. The TS areas suggested by ANSI/ANS-15.1 and NUREG-1537 are covered by the licensee's TSs along with associated surveillance requirements. The technical content of the TSs is consistent with guidance and provides an envelope of performance against which proposed experiments can be evaluated. Therefore, the licensee's limitations on experiments are acceptable to the staff.

10.4 Experiment Review

The review and approval process for experiments is governed by TS 4.8.b., which is discussed above; TS 6.2; and TS 6.5, which is discussed in Section 12 of this SER. The process requires that a safety analysis be developed for each experiment that demonstrates compliance with the TSs. The Reactor Operations Committee (ROC) must review and approve, and the Radiation Center director must approve, in writing, any new experiment involving the reactor, as well as substantive changes to existing experiments. In addition, the ROC reviews all changes made under 10 CFR 50.59 which includes experiments. Minor changes to experiments that do not significantly alter the experiment may be approved by the reactor administrator or reactor supervisor. The NRC considers minor changes to be those that do not raise to the level of a review under 10 CFR 50.59.

The staff has reviewed the licensee's process for experiment review. The TSs adequately describe the processes used to evaluate previously approved experiments and review and approve new experiments, including determining the need for a change to the TSs pursuant to 10 CFR 50.59. In addition, the TSs provide reasonable assurance that administrative oversight will preclude the experiment program from posing a significant risk to the health and safety of the public, facility personnel, experimenters, and the environment.

10.5 Conclusions

On the basis of its evaluation of the information presented in the licensee's SAR, the staff concludes as follows:

- The design of the OSTR experimental facilities, combined with the review and TS requirements applied to experimental research activities, give adequate assurance that experiments are unlikely to fail, are unlikely to release significant radioactivity, and are unlikely to cause damage to either the reactor or its fuel.

- The licensee has a sufficient experimental review process.

- The TSs place acceptable limits on the use of experimental facilities and provide reasonable assurance that experiments are conducted in a safe and controlled manner, as provided by TS 3.8 (limitations on experiments), TS 4.8 (surveillance), and TS 6.2.3 (review of experiments).

11. RADIATION PROTECTION AND RADIOACTIVE WASTE MANAGEMENT

11.1 <u>Radiation Protection</u>

Activities involving radiation at the OSTR are controlled under the radiation protection program. This radiation protection program is designed to meet the requirements of 10 CFR 20.1101, "Radiation Protection Programs," and minimize radiation exposure. The regulations in 10 CFR 20.1101 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 ALARA. The licensee shall periodically (at least annually) review the radiation protection program content and implementation. The NRC inspection program routinely reviews radiation protection and radioactive waste management at the OSTR. The licensee's performance in this area has been acceptable.

11.1.1 Radiation Sources

The descriptions of potential radiation sources, including the inventories of each physical form and their locations, has been reviewed by the staff. This review of the radiation sources included the identification of potential radiation hazards as presented in the SAR and a verification that the hazards were accurately depicted and comprehensively identified.

11.1.1.1 Airborne Radiation Sources

During normal operations the primary airborne sources of radiation are argon-41 and nitrogen-16. Argon-41 results from irradiation of the air in experimental facilities and dissolved air in the reactor pool water. The primary means of argon-41 production is the reactor pool. Other production sources include the open beam ports, the pneumatic irradiation system, and a small contribution from the dry tube (see Chapter 10 of this SER). Nitrogen-16 is produced when oxygen in the pool water is irradiated by the reactor core. The staff has reviewed the licensee's calculations of the production and release of routine airborne radioactive effluents and the resultant doses to the public and OSTR staff.

The dose rate at the top of the reactor tank with the reactor operating at full power is about 100 mRem/hr. Nitrogen-16, argon-41 and radiation from the reactor core contribute to this dose rate. Nitrogen-16 has a very short half-life (about 7 seconds) and the reactor has a core diffuser system (see Chapter 5 of this SER) that creates a water circulation pattern designed to suppress nitrogen-16 transport to the surface of the pool and reduce the reactor pool surface dose rate. The licensee performed calculations in the SAR assuming that the total 100 mrem/hr dose at the pool surface was from nitrogen-16 (a conservative assumption) to determine the occupational airborne immersion dose from nitrogen-16. The result of the calculation was 3.8 mrem/hr at full power. Because of the short half-life of nitrogen-16, exposure to the public is negligible. Only argon-41 has been found in analysis of effluent samples.

The dose from normal operations to a person in the unrestricted area due to an argon-41 release from the stack was also calculated by the licensee. When all irradiation facilities are configured such that the production of argon-41 is maximized, the emission rate of argon-41 in the stack effluent has been measured to be approximately 11 microcuries per second (μCi/s). This configuration of the irradiation facilities is not normal. Facilities such as the beam ports and thermal column are normally valved off to trap air within the facilities. The rotating rack has a nitrogen purge system which displaces air in the system. The normal emission rate which is primarily from the reactor pool is about 10 percent of the maximum. The release point is a stack on the roof of the reactor building that is part of the ventilation system (see Chapter 9 of this SER). The stack is 65 feet above ground with a stack diameter of 1.75 feet and a stack effluent velocity of 3,875 feet per minute. The concentration of argon-41 leaving the stack is calculated at 2.5×10^{-6} microcuries per milliliter (μCi/ml). The dose that this concentration would give to a member of the public was conservatively calculated, assuming that the reactor operates continuously for a year and that the person stands at the point of maximum exposure continuously for the year. These calculations assume that the person is immersed in the argon-41 cloud. Using the worse case atmospheric conditions results in a dose of 5 mrem. This is below the 10 CFR Part 20, Appendix B, Table 2 limit of 50 mrem for inhalation. The licensee also evaluated the dose from the argon-41 cloud passing over a person's head and the person receiving dose from radiation "shine" from the cloud. This method of exposure results in a lower dose than being immersed in the argon-41 cloud on the ground.

TS 3.7.2 limits the concentration of argon-41 discharged into the environment as follows:

> The annual average concentration of 41Ar discharged into the unrestricted area shall not exceed 4×10^{-6} μCi/ml at the point of discharge.

The concentration of 4×10^{-6} μCi/ml in TS 3.7.2 bounds the concentration of the calculated maximum concentration of 2.5×10^{-6} μCi/ml. The concentration in TS 3.7.2 represents a dose of about 8 mrem/yr under the assumptions discussed above. This dose meets the regulatory requirements for radiation dose limits for individual members of the public in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," and the constraint on air emissions of radioactive material in 10 CFR 20.1101(d). Because meeting TS 3.7.2 results in argon-41 emissions that are within the requirements of the regulations, TS 3.7.2 is acceptable to the staff. Because routine effluent releases from the OSTR are within the regulatory limits for radiation dose to members of the public, the staff concludes that the production and control of argon-41 is acceptable.

Because of the design of the ventilation system, occupational exposure to argon-41 in the reactor bay is only from the reactor tank. For argon-41, the derived air concentration (DAC) limit is 3.0×10^{-6} μCi/ml (10 CFR Part 20, Appendix B). The argon-41 concentration in the reactor building from pool release is 2.7×10^{-7} μCi/ml, a factor of 10 below the regulatory limit.

These calculations demonstrate that routine airborne effluent releases from the OSTR are well within 10 CFR Part 20 criteria and are therefore acceptable to the staff.

Releases from accident conditions can occur if the fuel cladding is breached. Conservative calculations performed in Chapter 13 of the SAR demonstrate that, even if a fuel pin clad rupture were to occur in air and the radioactive material quickly released to the environment at ground level, the dose to population directly outside the OSTR would be within 10 CFR Part 20 requirements.

11.1.1.2 Liquid Radiation Sources

Impurities in the primary coolant become activated by neutrons as they pass through the reactor core. The impurity levels are controlled by filters and the demineralzer system. The equilibrium concentrations of predominant radionuclides in the primary coolant are within the

10 CFR Part 20, Appendix B, limits for release to the sewer. As discussed in Chapter 4 of this SER, the equilibrium concentrations of predominant radionuclides are within effluent concentration limits for water given in 10 CFR Part 20, Appendix B.

Nitrogen-16 also contributes to the dose rate from the primary piping, since this piping carries pool water that has been circulated through the reactor core. The pool surface exposure rate is discussed under the gaseous radiation sources above. Measurements taken by the licensee at full-power operation indicate dose rates of 1 to 10 mrem/hr on the surface of the primary piping, all of which is within a designated radiation area.

The licensee samples primary water on a monthly basis for radioactive content to help detect potential fission product leakage from the reactor fuel. There is a primary water monitor that is normally in continuous operation. The monitor does not have a high level of sensitivity for radiation. It is a backup to the reactor top and stack monitors and the demineralizer radiation monitor.

There is also a 2500-gallon holdup tank where potential low-level liquid wastes are accumulated from OSTR operations and stored prior to disposal. The contact dose rate from the holdup tank is typically no greater than background.

Radiation exposures from these liquid radiation sources at the OSTR are small, and access to them is controlled. Therefore, the staff concludes that these sources do not present a significant hazard to either operating personnel or the public.

11.1.1.3 Solid Radiation Sources

The fission products in the reactor fuel constitute the most significant solid radiation source. Water and concrete shielding help to control this source of radiation. Nonfuel sources include activated reactor components, resins from the primary water demineralizer, and irradiated samples. Final radioactivity is estimated before experimental irradiations are performed, so both shielding and storage duration requirements will be known. These sources are controlled by the radiation protection program. Solid radioactive waste handling has not resulted in any significant personnel exposure at OSTR.

11.1.2 Radiation Protection Program

The regulation in 10 CFR 20.1101(a) requires that each licensee shall develop, document, and implement a radiation protection program. OSU has a structured radiation protection program with a health physics staff that is appropriately independent, and equipped with radiation detection capabilities to determine, control, and document occupational radiation exposures at the OSTR. The NRC regularly inspects the radiation protection program and finds that the program as implemented meets the requirements of the regulations. SAR Sections 12.1.2 and 12.1.3 describe the management of the radiation protection program (see Chapter 12 of this SER). SAR Section 11.1.2 states, “The Radiation Center’s Radiation Protection Program is the responsibility of the Radiation Center Director and is under the supervision of the Senior Health Physicist...[and] may report directly to the OSTR Reactor Operations Committee and/or Radiation Safety Committee should such action be deemed necessary.”

The health physics procedure and document control is described in the SAR under Section 11.1.2.3. Radiation protection procedures include testing and calibration of the monitors and detection instrumentation; administrative guidelines for receiving, monitoring, handling, transporting and testing radioactive materials; decontamination; investigation; training; ALARA; and personnel access. SAR Section 11.1.2.5 states that "auditing of the Radiation Protection Program is performed by the Reactor Operations Committee."

All personnel entering the facility are issued the appropriate monitoring devices and any required protective clothing. SAR Section 11.1.2.4 states "All personnel permitted unescorted access to the OSTR reactor building shall receive training in radiation protection as required by 10 CFR 19.12." General levels of training cited in the SAR include storage, transfer, and use of radiation and/or 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/or 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 OSTR vital area shall receive additional training to include access control rules, emergency procedures, dosimetry requirements, key checkout and return, reactor top safety, 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 liquid and gaseous effluents are monitored before release according to TS 3.7.2 to comply with 10 CFR Part 20 limits.

Health physics/radiation protection program procedures are audited on an annual basis by the ROC. This includes all procedures, personnel radiation doses, radioactive material shipments, radiation surveys, and radioactive effluents released to unrestricted areas. The health physics staff, led by the senior health physicist, maintains 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 staff concludes that the OSTR radiation protection program as described in Section 11.1 of the SAR complies with 10 CFR 20.1101(a), is acceptably implemented and provides reasonable assurance that, for all facility activities, the program will protect the staff, the environment, and the public from unacceptable radiation exposures. The organization and oversight of the program is reviewed in Chapter 12 of this SER.

11.1.3 As Low As Reasonably Achievable Program

To comply with the regulations in 10 CFR 20.1101, OSU has 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, "Radiation Protection at Research Reactor Facilities," which is supported by the NRC staff. The Radiation Center director, who has ultimate responsibility for the program, delegates responsibility to the senior health physicist. The program is applied through written procedures and guidelines described in SAR Section 11.1.3. All proposed experiments and operational procedures at the OSTR are reviewed for ways to minimize potential exposure to personnel. The 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 to prevent recurrence. SAR Section

11.1.3 states that “exposure investigations [are] initiated when an individual receives greater than 50 mrem in one month or 500 mrem in a quarter...” The ALARA program is adequately supported by the upper levels of the University management. The NRC regularly inspects the ALARA program and finds that the program as implemented meets the requirements of the regulations. The staff concludes that the OSTR ALARA program as described in Section 11.1.3 of the SAR complies with 10 CFR 20.1101, is acceptably implemented and provides reasonable assurance that for all facility activities, radiation exposure will be maintained ALARA.

11.1.4 Radiation Monitoring and Surveying

The regulations in 10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that–

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

(2) Are reasonable under the circumstances to evaluate–

 (i) The magnitude and extent of radiation levels; and

 (ii) Concentrations or quantities of radioactive material; and

 (iii) The potential radiological hazards.

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

The licensee has a comprehensive set of portable radiation survey instrumentation that covers with sufficient ranges the various types of radiation that may be encountered at the OSTR. The licensee also has other specialized radiation monitoring equipment such as a gamma spectroscopy system and hand and foot monitors.

Systems used at the OSTR for monitoring radiation include area radiation monitors. While only the reactor top area monitor is required by the TSs, area monitors are located at strategic points throughout the facility where radiation levels could exceed normal levels. These area monitors have local audible and visual alarms. The alarm is set at levels agreed upon by the reactor operations supervisor and the senior health physicist based on anticipated normal or potentially abnormal radiation levels.

Airborne activity at the top of the reactor and effluents released up the stack to the environment are monitored by CAM. There is a single-channel airborne particulate monitor at the reactor top and a two-channel airborne particulate and gaseous monitor for effluents leaving the reactor building by the ventilation system stack. Both of these monitors have local and remote alarm capability. The remote alarms are in the control room. For the CAM on the reactor top, a fraction (0.06 percent) of the DAC for cesium-138 is the alarm setpoint (10 CFR Part 20, Appendix B, Table 2). This value permits early detection of a cladding failure and yet is high enough to prevent spurious alarms due to fluctuations in the background count rate.

For the CAM on the stack, the alarm setpoint for the particulate channel is normally the net count rate equivalent of 6.7×10^{-8} microcuries per cubic centimeter ($\mu Ci\ cm^{-3}$) for cesium-138 which is 0.3 percent of the DAC and 83 percent of the annual effluent concentration limit

(10 CFR Part 20, Appendix B, Table 2). This value permits early detection of a cladding failure and yet is high enough to prevent spurious alarms due to fluctuations in the background count rate. The alarm setpoint for the gas channel on the stack monitor is normally set to the net count rate equivalent of 4 $\times 10^{-6}$ μCi cm^{3}, which is the TS annual average concentration limit for argon-41 discussed above.

The following requirements for the radiation monitoring system are found in TS 3.7.1:

> The reactor shall not be operated unless the minimum number of radiation monitoring channels listed in Table 4 are operating.
>
> Table 4— Minimum Radiation Monitoring Channels

Radiation Monitoring Channels	Number
Reactor Top Area Radiation Monitor	1
Continuous Air Particulate Radiation Monitor	1
Exhaust Gas Radiation Monitor	1
Exhaust Particulate Radiation Monitor	1

> Exception: When a single required radiation monitoring channel becomes inoperable, operations may continue only if portable instruments, surveys, or analyses may be substituted for the normally installed monitor within one (1) hour of discovery for periods not to exceed one (1) month.

The staff concludes that TS 3.7.1 requires sufficient monitors to evaluate potential radiation hazards. As the discussion above shows that routine effluent releases are within regulatory limits and the discussion in Chapter 13 of this SER shows that the consequences of accidents are acceptable, having a single required radiation monitoring channel out of service for up to an hour is acceptable to the staff. The substitution of portable instruments, surveys, or analyses for an inoperable channel for a period not to exceed 1 month is acceptable to the staff. One month is a sufficient amount of time to return an inoperable monitor to operation. The substitution is limited to 1 month to prevent the substituted method from becoming permanent. The method used as a substitution would need to show continued regulatory compliance.

The following surveillance requirements for the radiation monitoring systems are given in TS 4.7:

a. A channel check of the radiation monitoring systems in section 3.7.1 shall be performed prior to each day’s operation or prior to each operation extending more than one day.

b. A channel test of the continuous air particulate, exhaust gas, and exhaust particulate radiation monitors shall be performed monthly.

c. A channel calibration of the radiation monitoring systems in section 3.7.1 shall be performed annually.

Calibration of the monitors and survey equipment is performed annually according to ANSI N323A-1997, either in-house or at an appropriate calibration facility. The frequency of channel checks and tests is based on experience and is consistent with frequencies recommended in ANSI/ANS 15.1. The staff concludes that the surveillance requirements in TS 4.7 for the radiation monitoring channels are acceptable.

Radiation and contamination surveys are both performed on a regular basis (daily, weekly, and monthly) by the health physics staff at the OSTR. To confirm that safe radiation working conditions exist, routine radiation level and contamination level surveys of specific areas are performed. Special radiation surveys necessary to support nonroutine facility operations are also performed. In addition, frequent on-the-spot personal observations (and recorded data) are made of work areas to provide advanced warning of needed corrections in the safe use and handling of radiation sources and other radioactive materials. Surveys and observations/corrections ensure and document that the operational and radiation protection programs help to keep doses ALARA.

Based on its review, the staff concludes that the equipment used by the licensee is appropriate for detecting the types and intensities of radiation likely to be encountered within the facility at appropriate frequencies to ensure compliance with 10 CFR 20.1501(a) and (b).

11.1.5 Radiation Exposure Control and Dosimetry

Reactor shielding based on GA design analysis for 1-MW TRIGA reactors was built and is maintained and upgraded as necessary. Measured radiation levels at the shield surface are normally less than 1—2 mrem/hr.

The ventilation system as described in Chapter 9 of this SER and above maintains the reactor bay at negative pressure with respect to outside areas and maintains argon-41 and nitrogen-16 levels below the limits prescribed in 10 CFR 20.1201, “Occupational Dose Limits for Adults.”

Personnel exposure is monitored by thermoluminescent dosimeters, CR-39 Track Etch dosimeters and pocket ion chambers, which are assigned to individuals who have the potential to be exposed to radiation. These personnel monitoring devices measure all types of radiation found at the OSTR.

Personnel protective equipment is used as needed. Facilities and equipment to decontaminate persons are available, if needed. Procedures exist governing the use of this equipment.

Internal dosimetry is evaluated by two bioassay methods. OSTR uses a liquid scintillation counter to analyze urine for the presence of tritium. A thyroid counting program is used to analyze iodine uptake in the thyroid in vivo. The Pacific Northwest National Laboratory performs whole body in vivo counting using gamma spectroscopy. Abnormal dose reading investigation (ADRI) is performed if a dose in any reporting period exceeds 1 percent of the applicable regulatory limit in 10 CFR Part 20, or when a visitor’s ADRI measured dose exceeds 10 mrem. Dosimetry records are kept for the life of the facility.

A review of the average and highest annual dose equivalent incurred by OSTR staff demonstrate compliance with the facility’s ALARA program as well as the efficacy of the control and radiation protection. The regulations in 10 CFR 20.1201 limit annual occupational total effective dose equivalent (TEDE) dose to less than 5 rem and shallow dose to skin of whole body or any extremity to 50 rem. The staff examined the average and highest individual annual exposures for the time period 1999—2007. The average annual dose over that period was 56 mrem TEDE and 144 mrem extremities. The greatest individual annual dose was 153 mrem TEDE and 615 mrem for extremities. All OSTR staff received less than the Federal limits. Based on a review of OSTR staff exposures which are significantly below regulatory limits, the staff concludes that the control and radiation protection program is adequate in ensuring safety.

The staff has reviewed the licensee's control of exposure and dosimetry program. Historic doses to the OSTR staff show that exposures are controlled through the radiation protection and ALARA programs. The licensee dosimetry program monitors external and internal radiation doses of all individuals required to be monitored. The staff concludes that the licensee's control of exposures and dosimetry program are acceptable.

11.1.6 Contamination Control

Contamination surveys are performed on a daily, weekly, and monthly basis, depending on the frequency that radioactive material is used or handled. Handling of any radioactive material within the OSTR is controlled by written procedures. The facility surveys have routinely shown no detectable contamination in nonradiological areas of the facility. The NRC inspection program has confirmed that the licensee has effective contamination control. The staff concludes that adequate controls exist to prevent the spread of radiological contamination within the facility.

11.1.7 Environmental Monitoring

Environmental monitoring consists of taking various environmental samples at 25 different locations, generally within 1000 feet of the OSTR. The environmental monitoring program is audited by the ROC on a quarterly basis to ensure that the program contains an adequate number of samples, locations, and frequency of collection. It also verifies that analysis of the samples has sufficient sensitivity to ensure that the overall program would provide an early indication of any environmental impact caused by OSTR operation. Direct gamma radiation measurements are performed monthly. Soil, vegetation, and water samples are completed quarterly. Doses recorded by the offsite dosimetry and direct dose rate measurements have consistently totaled approximately 80 mrem/yr. This dose is attributable to natural background radiation, which, for the area of Oregon where the OSTR is located is about 110 mrem/yr. TS 6.8.3.b. specifies that records of offsite environmental monitoring surveys shall be retained for the lifetime of the reactor facility. The staff concludes that the environmental monitoring program is sufficient to assess the radiological impact of the OSTR on the environment.

11.2 Radioactive Waste Management

The purpose of the radioactive waste management program is to minimize radioactive waste and ensure that it is properly handled, stored, and disposed of. Section 11.2 of the SAR describes the radioactive waste management program, addressing solid, liquid, and gaseous radioactive wastes. All radioactive waste handling operations are controlled by procedure and overseen by the OSTR heath physics staff.

TS 6.7.1 specifies that, annually, the facility reports the following to the NRC:

(e) 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 prior to 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 percent of the concentration allowed or recommended, a statement to this effect is sufficient.

11.2.1 Solid Waste

Low-level solid waste from laboratory experiments or disposable protective clothing items is accumulated and stored in authorized containers. Activated equipment and activated irradiation samples are stored in the reactor bay area for reuse or to decay to low-level activity limits. When filled, the low-level waste containers are sealed and transferred to OSU radiation safety until they are shipped off campus by a licensed carrier to a licensed facility for disposal. Approximately 20 cubic feet of solid radioactive waste is generated per year from OSTR operations. Adequate 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.

11.2.2 Liquid Waste

Normal operation of the OSTR does not produce significant liquid radioactive waste. The policy of the OSTR is not to routinely release 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 system equipment. These liquid wastes are collected and stored in a 2500-gallon holdup tank until determination of final disposition. All non-sewer drains within the reactor building drain to this holdup tank. The tank has a level indicator with high and full alarms which are monitored by reactor operators during the day and remotely at night. The hold-up tank is sampled and tested for radioactive materials on a monthly basis.

Liquid radioactive waste from the holdup tank is released to the sanitary system in accordance with the approved discharge permit from the Oregon Natural Resource Conservation Commission. All releases are governed by written procedures to ensure they are within the limits stated in 10 CFR Part 20, Appendix B, Table 3. Fresh water can be added to the holdup tank if dilution is necessary prior to discharge.

11.2.3 Gaseous Waste

Argon-41 is the only gaseous waste produced during OSTR operations. There are no special offgas collection systems or storage tanks for it. The amounts that are produced at the OSTR typically mix with the normal facility atmosphere and are discharged with the ventilation system exhaust. This radioactive gas is considered an effluent as opposed to a waste gas and is discussed above.

11.2.4 Radioactive Waste Management Conclusion

The staff has reviewed the facility radioactive waste management program and concludes that there is reasonable assurance that radioactive waste released from the facility will neither exceed applicable regulations nor pose unacceptable radiation risk to the environment and the public.

11.3 <u>Conclusions</u>

On the basis of its evaluation of the information presented in the licensee's SAR, observations of the licensee's operations and results of the NRC inspection program, the staff concludes as follows:

- The OSTR radiation protection program complies with the requirements in 10 CFR 20.1101(a), is acceptably implemented, and provides reasonable assurance that the staff, the environment, and the public are protected from unacceptable radiation exposures. The radiation protection program is acceptably staffed and equipped. The radiation protection staff has adequate lines of authority and communication to carry out the program.

- The OSTR ALARA program complies with the requirements of 10 CFR 20.1101(b). Review of controls for radioactive material in the OSTR provides reasonable assurance that radiation doses to the environment, the public, and facility personnel will be ALARA.

- The results of radiation surveys carried out at the OSTR, doses to the persons issued dosimetry, and results of the environment monitoring program help verify that the radiation protection and ALARA programs are effective.

- 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 argon-41 and nitrogen-16 and control the potential for facility staff exposures. Conservative calculations of the quantities of these gases released into restricted and unrestricted areas give reasonable assurance that doses to the OSTR 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 environment and the public.

12. CONDUCT OF OPERATIONS

Conduct of operations for the OSTR includes administrative controls over facility operation and the facility emergency and security plans. The administrative controls discussed in this section of the SER are the facility organization, training, operational review and audits, procedures, required actions, reports and records. This section of the SER addresses Chapter 12 of the SAR and Section 6 of the TSs.

The primary guidance for the development of administrative control TSs for research reactor operation is ANSI/ANS-15.1. The licensee's TSs are based on the 1990 and 2007 versions of the standard. The NRC staff supports the use of ANSI/ANS 15.1 for Section 6 TSs as discussed in Chapter 14 of NUREG-1537. The staff used the 1990 and 2007 versions of ANSI/ANS-15.1 in its review of the licensee's administrative controls. In some cases, the wording of TSs proposed by the licensee was not identical to that given in ANSI/ANS 15.1 and NUREG-1537. However, the staff reviewed these cases and determined that the licensee's proposed TSs met the intent of the guidance and were acceptable.

12.1 <u>Organization</u>

Responsibility for the safe operation of the reactor facility is vested within the chain of command shown in SAR Section 12.1 and Figure 12.1; TSs 6.1, 6.1.1, and 6.1.2; and related TS Figure 1. The licensee's organization is shown in Figure 12.1 below.

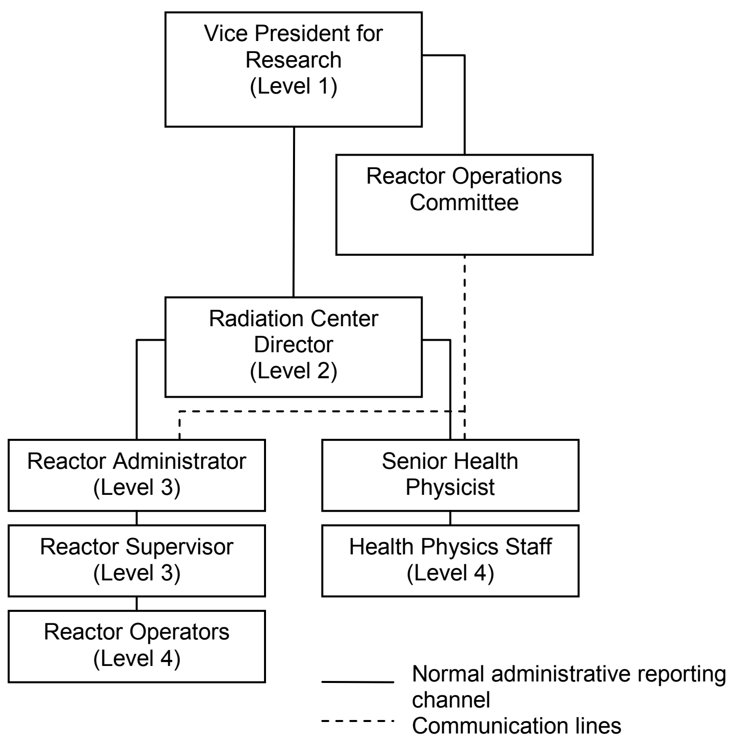

Figure 12.1 Radiation Center organization

The Radiation Center director is Level 2 management and is responsible for the safe operation and maintenance of the entire radiation center including the OSTR. The radiation center director reports to the Vice President for Research of the University (Level 1) and then to the President of the University. The ROC (discussed below) provides for the safety review and audit functions for the reactor and reports to the Vice President for Research of the University and then to the President of the University.

TS 6.1 describes the organization as follows:

> Individuals at the various management levels, in addition to being responsible for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, technical specifications, and federal regulations.

TS 6.1.2 discusses the responsibilities of each of the organizational levels given in the organizational chart as follows:

a. Vice-President for Research (Level 1): The Vice-President for Research is responsible for University Centers, Institutes, and Program organizations representing Oregon State University.

b. Radiation Center Director (Level 2): The Radiation Center Director reports to the Vice-President for Research and is accountable for ensuring that all regulatory requirements, including implementation, are in accordance with all requirements of the USNRC and the *Code of Federal Regulations*.

c. Reactor Administrator (Level 3): The Reactor Administrator reports to the Radiation Center Director and is responsible for guidance, oversight, and technical support of reactor operations.

d. Senior Health Physicist (Level 3): The Senior Health Physicist reports to the Radiation Center Director and is responsible for directing the activities of health physics personnel including implementation of the radiation safety program.

e. Reactor Supervisor (Level 3): The Reactor Supervisor reports to the Reactor Administrator and is responsible for directing the activities of the reactor operators and senior reactor operators and for the day-to-day operation and maintenance of the reactor.

f. Reactor Operator and Senior Reactor Operator (Level 4): The Reactor Operator and Senior Reactor Operator report to the Reactor Supervisor and are primarily involved in the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor-related equipment.

Individuals at all levels are defined as being responsible for safeguarding the health and safety of the public and adhering to all requirements of the license, TSs, and NRC regulations.

The organization as described in TSs 6.1 and 6.1.1, and Figure 1 of the TSs, and the Radiation Center staff responsibility, as described in TS 6.1.2 and SAR Section 12.1, is consistent with the guidance in ANSI/ANS 15.1 and NUREG 1537, and is therefore acceptable to the staff.

The licensee discussed the minimum staffing necessary to safely operate the OSTR in Section 12.1.3 of the SAR and in TS 6.1.3 as follows:

a. The minimum staffing when the reactor is operating shall be:

   1. A reactor operator or the Reactor Supervisor in the control room;

   2. A second person present in the Radiation Center Complex able to carry out prescribed instructions; and

   3. If neither of these two individuals is the Reactor Supervisor, the Reactor Supervisor shall be readily available on call. Readily available on call means an individual who:

      i. Has been specifically designated and the designation is known to the operator on duty;

      ii. Can be rapidly contacted by phone by the operator on duty; and

      iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).

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

   1. Radiation Center Director
   2. Reactor Administrator
   3. Senior Health Physicist
   4. Any Licensed Reactor or Senior Reactor Operator

c. Events requiring the direction of the Reactor Supervisor:

   1. Initial startup and approach to power of the day;

   2. All fuel or control-rod relocations within the reactor core region;

   3. Relocation of any in-core experiment or irradiation facility with a reactivity worth greater than one dollar; and

   4. Recovery from unplanned or unscheduled shutdown or significant power reduction.

The regulation in 10 CFR 50.54(k) states “An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility.” OSTR TS 6.1.3 requires an operator in the control room. All portions of the control room are easily accessible to the controls; an adjacent room is not considered to be part of the control room.

The regulation in 10 CFR 50.54(m)(1) states “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." TS 6.1.3.c. requires a senior reactor operator be present for those events listed in 10 CFR 50.54(m)(1).

The staff evaluated the requirements of TS 6.1.3 and found that they are consistent with the guidance in ANSI/ANS-15.1 and NUREG-1537, and that they satisfy the requirements of 10 CFR 50.54(k) and 10 CFR 50.54(m)(1). TS 6.1.3 is acceptable to the NRC staff.

12.2 Training

The licensee discussed selection and training of personnel for key positions in Sections 12.1.4 and 12.10 of the SAR and in TS 6.1.4. TS 6.1.4 reads as follows:

> The selection, training and requalification of operations personnel should be in accordance with ANSI/ANS 15.4-1988; R1999, "Standard for the Selection and Training of Personnel for Research Reactors."

As part of the license renewal the staff reviewed the "Requalification Program for Licensed Operators of the Oregon State TRIGA Reactor." The reactor supervisor serves as the training coordinator and is responsible for the implementation, coordination, and operation of the requalification program including the training of new operators. The plan discusses the schedule of training, lectures and written examinations; on-the-job training; oral and operating examinations; document review requirements; overall evaluation of operators; absence from licensed activities; exemptions to the program; recordkeeping; and administration of the program. The staff concludes that it meets the applicable regulations of 10 CFR Part 55, "Operators' License," and follows the guidelines of ANSI/ANS-15.4, "Selection and Training of Personnel for Research Reactors," issued in 1988.

The licensee uses ANSI/ANS 15.4 as guidance for the selection and training of personnel. The NRC staff supports the use of ANSI/ANS 15.4 by licensees for selection and training of personnel. Based on the licensee's use of ANSI/ANS 15.4 and the licensee's requalification program meeting the requirements of 10 CFR Part 55, the staff concludes that TS 6.1.4 is acceptable.

12.3 Review and Audit Activities

The ROC exists to review and audit matters relating to the safe operation of the facility and the health and safety of the public and the environment. Overall responsibility, composition and qualification of the ROC, as well as charter and rules, are described in SAR Sections 12.2.1 and 12.2.2 and TSs 6.2, 6.2.1, 6.2.2, and 6.2.3.

These TSs read as follows:

> 6.2 Review And Audit
>
> The Reactor Operations Committee (ROC) shall have primary responsibility for review and audit of the safety aspects of reactor facility operations. Minutes, findings or reports of the ROC shall be presented to Level 1 and Level 2 management within ninety (90) days of completion.

6.2.1 Composition and Qualifications

An ROC of at least five (5) members knowledgeable in fields which relate to reactor engineering and nuclear safety shall review and evaluate the safety aspects associated with the operation and use of the facility. The jurisdiction of the ROC shall include all nuclear operations in the facility and associated general safety standards.

6.2.2 Charter and Rules

The operations of the ROC shall be in accordance with a written charter including provisions for:

a. Meeting frequency (at least annually);

b. Voting rules;

c. Quorums (5 members, no more than two voting members may be of the operating staff at any time);

d. Method of submission and content of presentation to the committee;

e. Use of subcommittees; and

f. Review, approval, and dissemination of minutes.

The facility organizational structure (Figure 12.1) shows that the ROC reports to the Vice President for Research which is considered a Level 1 position. The ROC conducts its review and audit functions in accordance with a written charter, which includes provisions for meeting frequency, voting rules, quorums, method of submission, content of presentations to the ROC, use of subcommittees, and minutes. The charter was reviewed and found to be in agreement with the SAR, NUREG-1537, and the recommendations of ANSI/ANS-15.1.

NUREG-1537 and ANSI/ANS 15.1 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 ROC charter establishes a quorum of five members, which guarantees that operations will not be a majority. The rules of the ROC, as outlined in TS 6.2, the ROC charter and the SAR, are consistent with the guidelines of ANSI/ANS-15.1. These aspects of the ROC are acceptable to the staff.

The review functions of the ROC are listed in TS 6.2.3 as follows:

Review Function

The responsibilities of the ROC, or designated Subcommittee thereof, include, but are not limited to, the following:

a. Review all changes made under 10 CFR 50.59;

b. Review of all new procedures and substantive changes to existing procedures;

c. Review of proposed changes to the technical specifications, license or charter;

d. Review of violations of technical specifications, license, charter or violations of internal procedures or instructions having safety significance;

e. Review of operating abnormalities having safety significance;

f. Review of all events from reports required in sections 6.6.1 and 6.7.2 of these Technical Specifications;

g. Review of audit reports.

The licensee considers reviews of changes in equipment and systems and reviews of experiments to fall under item a above. The particular items specified for review by the ROC were compared with the guidelines of the NUREG-1537 and ANS 15.1 and are acceptable.

TS 6.2.4 requires the performance of an annual audit of reactor operations by the ROC or a subcommittee of the ROC. TS 6.2.4 reads as follows:

> 6.2.4 Audit Function
>
> The ROC or a Subcommittee thereof shall audit reactor operations at least annually. The annual audit shall include at least the following:
>
> a. facility operations for conformance to the technical specifications and applicable license or charter conditions;
>
> b. the retraining and requalification program for the operating staff;
>
> c. the results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety; and
>
> d. the Emergency Response Plan and implementing procedures.

Audits of various aspects of the heath physics program are specified in Chapter 11 of the SAR and the ROC charter and are a requirement of the regulations in 10 CFR 20.1102(c). The licensee specifies that no individual responsible for an area may conduct the audit of that area. Deficiencies that are uncovered by an audit that affect reactor safety are immediately reported to Level 1 management as a reportable event. Written reports of audit findings are submitted to Level 1 and Level 2 management within 90 days after completion of the audit.

The staff reviewed TS 6.2 and SAR Section 12.2 and found the licensee's review and audit functions acceptable, the committee members appear to be well qualified and have a wide spectrum of expertise, the charter and rules are acceptable, and the items that the committee will review and audit are comprehensive and acceptable. These findings were based on a comparison with NUREG-1537 and ANSI/ANS-15.1.

12.4 Radiation Protection

Radiation safety and radiation protection services are the responsibility of the senior health physicist (Level 3 management), who reports to the director of the Radiation Center. The senior health physicist has a line of communication to the reactor administrator. The senior health physicist and staff provide the radiation protection function for the OSTR. They have the authority and responsibility to halt any perceived unsafe practices. TS 6.3 contains the administrative control related to radiation safety as follows:

> The Senior Health Physicist shall be responsible for implementation of the radiation safety program. The requirements of the radiation safety program are established in 10 CFR Part 20. The program should use the guidelines of the ANSI/ANS 15.11—1993; R2004, "Radiation Protection at Research Reactor Facilities".

Radiation safety oversight is provided by the OSU Radiation Safety Committee, which reports through the Vice President for Finance and Administration to the President of the University. The senior health physicist may bring concerns directly to the ROC or the Radiation Safety Committee. Additional discussion of radiation safety can be found in Chapter 11 of the SER. The administrative control TS for radiation safety is consistent with ANSI/ANS-15.1 and NUREG-1537 and is therefore acceptable to the staff.

12.5 Procedures

The licensee has specified in Section 12.3 of the SAR and in TS 6.4 the type of written procedures that must be prepared and approved prior to use. TS 6.4 reads as follows:

> 6.4 Procedures
>
> Written operating procedures shall be adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items:
>
> a. Startup, operation and shutdown of the reactor;
>
> b. Fuel loading, unloading, and movement within the reactor;
>
> c. Maintenance of major components of systems that could have an effect on reactor safety;
>
> d. Surveillance checks, calibrations, and inspections required by the technical specifications or those that have an effect on reactor safety;
>
> e. Radiation protection;
>
> f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity;
>
> g. Shipping of radioactive materials;
>
> h. Implementation of the Emergency Response Plan.

Substantive changes to the above procedures shall be made only after review by the ROC. Except for radiation protection procedures, unsubstantive changes shall be approved prior to implementation by the Reactor Administrator and documented by the Reactor Administrator within 120 days of implementation. Unsubstantive changes to radiation protection procedures shall be approved prior to implementation by the SHP and documented by the Senior Health Physicist within 120 days of implementation.

Temporary deviations from the procedures may be made by the responsible senior reactor operator in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported by the next working day to the Reactor Administrator.

As shown above, TS 6.4 specifies the areas to be covered by procedures. TSs 6.2.3 and 6.4 specify that new procedures and substantive changes to procedures must be approved by the ROC. The process for unsubstantive and temporary changes to procedures is outlined in TS 6.4.

The TSs for procedures are consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537. The process and method for procedures established in TS 6.4 ensures adequate management control and proper review of procedures. The staff concludes that the procedural requirements in TS 6.4 provide reasonable assurance of the safe operation of the reactor and proper administration of the facility.

12.6 Experiments

Administrative controls for the review and approval of experiments are given in TS 6.5 as follows:

Experiments Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures. Procedures related to experiment review and approval shall include:

a. All new experiments or class of experiments shall be reviewed by the ROC and approved in writing by the Level 2 or designated alternates prior to initiation;

b. Substantive changes to previously approved experiments shall be made only after review by the ROC and approved in writing by the Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

TS 6.5 contains the requirements for the review and approval of experiments. The purpose of the TS is to ensure that all experiments initially have a high level of review and that changes to existing approved experiments undergo a level of review appropriate to the significance of the change. TS 6.5 is consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537 and is therefore acceptable to the staff.

12.7 Required Actions

The licensee has defined the required actions for events in TS 6.6. TS 6.6.1 contains those actions to be taken in case of a safety limit violation as described below:

In the event a safety limit (fuel temperature) is exceeded:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the USNRC;

b. An immediate notification of the occurrence shall be made to the Reactor Administrator, Radiation Center Director and Chairperson, ROC; and

c. A report, and any applicable followup report shall be prepared and reviewed by the ROC. The report shall describe the following:

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

   2. Effects of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public; and

   3. Corrective action to be taken to prevent recurrence.

TS 6.7.2.a.1. contains the requirement to report the violation of the safety limit to the NRC not later than the next working day. The actions proposed by the licensee are consistent with the guidance of ANSI/ANS 15.1 and NUREG-1537 and meet the requirements given in 10 CFR 50.36(d)(1) for actions to be taken if a safety limit is exceeded. Therefore, TS 6.6.1 is acceptable to the staff.

TS 6.6.2 contains those actions to be taken in case of reportable occurrences other than exceeding a safety limit. Reportable occurrences are discussed in TS 6.7.2 below.

The following actions are required by TS 6.6.2:

For all events which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours under Section 6.7.2, except a safety limit violation, the following actions shall be taken:

a. the reactor shall be secured and the Reactor Administrator notified;

b. operations shall not resume unless authorized by the Reactor Administrator;

c. the Reactor Operations Committee shall review the occurrence at their next scheduled meeting; and

d. a report shall be submitted to the NRC in accordance with Section 6.7.2 of these Technical Specifications.

The NRC will be notified of important events by the licensee in a timely manner. These actions are consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537 and are therefore acceptable to the staff.

Because these actions are consistent with those recommended in NUREG-1537 and ANSI/ANS-15.1, the staff concludes that appropriate action will be taken in case a safety limit is exceeded or in case of other reportable actions.

12.8 Reports

TS 6.7 contains the requirements for reports made by the licensee to the NRC. TS 6.7.1 details the requirements for the annual operating report and TS 6.7.2 details special reports. Special reports include occurrences that are reportable within 24 hours with a followup 14-day written report and changes that need to be reported to the NRC in writing within 30 days.

The contents of the annual operating report that must be submitted to the NRC are as follows:

> An annual report shall be created and submitted by the Radiation Center Director to the USNRC by November 1 of each year consisting of:
>
> a. a brief summary of operating experience including the energy produced by the reactor and the hours the reactor was critical;
>
> b. the number of unplanned shutdowns, including reasons therefore;
>
> c. a tabulation of major preventative and corrective maintenance operations having safety significance;
>
> d. 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 10 CFR 50.59;
>
> e. 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 prior to 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 percent of the concentration allowed or recommended, a statement to this effect is sufficient;
>
> f. a summarized result of environmental surveys performed outside the facility; and
>
> g. a summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed.

The specified content of the annual operating report is consistent with NUREG 1537 and ANSI/ANS-15.1 and is acceptable to the staff.

The following required special reports, are listed in TS 6.7.2:

> In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made by the Radiation Center Director to the NRC as follows:
>
> a. a report not later than the following working day by telephone and confirmed in writing by facsimile to the NRC Operations Center, to be followed by a written report that describes the circumstances of the event within 14 days to the NRC Document Control Desk of any of the following:
>
>   1. violation of the safety limit;
>   2. release of radioactivity from the site above allowed limits;
>   3. operation with actual safety system settings from required systems less conservative than the limiting safety system setting;
>   4. operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in Section 3;
>   5. a reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required;
>   6. an unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded;
>   7. abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable; or
>   8. an observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.
>
> b. a report within 30 days in writing to the NRC, Document Control Desk, Washington, D.C. of:
>
>   1. Permanent changes in the facility organization involving Level 1—2 personnel; and
>   2. significant changes in the transient or accident analyses as described in the Safety Analysis Report.

These reporting requirements are consistent with the guidance of ANSI/ANS-15.1 and NUREG-1537 and are therefore acceptable to the NRC staff.

12.9 <u>Records</u>

The following records required to be retained by the licensee are listed in TS 6.8:

6.8 Records

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

a. normal reactor operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year);
b. principal maintenance activities;
c. reportable occurrences;
d. surveillance activities required by the Technical Specifications;
e. reactor facility radiation and contamination surveys;
f. experiments performed with the reactor;
g. fuel inventories, receipts, and shipments;
h. approved changes to the operating procedures; and
i. Reactor Operations Committee meetings and audit reports.

6.8.2 Records to be Retained for at Least One Certification Cycle

a. Records of retraining and requalification of certified reactor operators and senior reactor operators shall be retained at all times the individual is employed until the certification is renewed.

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

a. gaseous and liquid radioactive effluents released to the environs;
b. offsite environmental monitoring surveys;
c. radiation exposures for all personnel monitored;
d. drawings of the reactor facility; and
e. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

The staff compared TS 6.8 with NUREG-1537 and ANSI/ANS-15.1. The requirements in TS 6.8 were consistent with the guidance and therefore are acceptable to the staff.

12.10 Emergency Planning

Regulations in 10 CFR 50.54(q) and (r) require that a licensee authorized to possess and/or operate a research reactor shall follow and maintain in effect an emergency plan that meets the requirements of Appendix E to 10 CFR Part 50. The staff reviewed the Oregon State University Radiation Center and Oregon State TRIGA Reactor Emergency Response Plan (EP) dated March 10, 2006 (ADAMS Accession No. ML060480275), against NUREG 0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors," issued October 1983; Regulatory Guide 2.6, Revision 1, "Emergency Planning for Research and Test Reactors," issued March 1983; ANSI/ANS-15.16, "Emergency Planning for Research Reactors," issued 1982; and NRC Information Notice 97-34, "Deficiencies in Licensee Submittals Regarding Terminology for Radiological Emergency Action Levels in Accordance with the New Part 20," issued June 1997. The staff concluded that the emergency plan is in accordance with the guidance and regulations. The licensee has demonstrated the ability to make changes to the emergency plan in accordance with 10 CFR 50.54(q). Accordingly, the staff concludes that the emergency plan provides reasonable assurance that the licensee can respond appropriately to a variety of emergency situations and that the emergency plan will be adequately maintained during the period of the renewed license.

12.11 Security Planning

Because the facility license authorizes possession of special nuclear material of strategic significance, the licensee must maintain security measures that satisfy the requirements of 10 CFR 73.67, "Licensee Fixed Site and In-Transit Requirements for the Physical Protection of Special Nuclear Material of Moderate and Low Strategic Significance," and 10 CFR 73.60, "Additional Requirements for Physical Protection at Nonpower Reactors" as appropriate. The licensee meets these requirements and the NRC inspects the licensee's measures for physical security and protection of special nuclear material on a routine basis. A recent inspection verified that the licensee's security measures satisfy applicable regulations and are acceptable.

12.13 Conclusions

On the basis of the above discussions, the staff concludes that the licensee has sufficiently experienced oversight, management structure, and procedures to provide reasonable assurance that the reactor will continue to be managed in a way that will cause no significant risk to the health and safety of the public. The staff has reviewed SAR Chapter 12 and Section 6 of the TSs which discuss the licensee's proposed organization, training including operator requalification, review and audit activities, administration of radiation protection activities, procedures, experiment review, required actions, and records and reports against the guidance given in the 1990 and 2007 versions of ANSI/ANS-15.1, which is supported by the NRC staff for the conduct of operations and NUREG-1537. The licensee's proposed conduct of operations in the areas reviewed is consistent with the guidance of the ANS standard and NUREG-1537. The staff also reviewed Section 6 of the TSs against 10 CFR 50.36 "Technical Specifications," including 10 CFR 50.36(d)(5) and (7) and concludes that the TSs meet the requirements of the regulations.

The staff has reviewed the Oregon State University TRIGA Reactor (OSTR) Physical Security Plan, and concludes that the licensee's security plan meets the requirements of 10 CFR 73.67 and 10 CFR 73.60 for protecting the special nuclear material associated with the facility.

The staff has reviewed the Oregon State University Radiation Center and Oregon State TRIGA Reactor Emergency Response Plan and concludes that the licensee's emergency plan meets the requirements of the regulations in 10 CFR Part 50, Appendix E and is therefore acceptable.

13. ACCIDENT ANALYSIS

The accident analyses presented in the OSTR SAR helped establish safety limits and limiting conditions that are imposed on the OSTR through the TSs. The licensee analyzed potential reactor transients and other hypothetical accidents. The licensee's analysis has included the potential effects of natural hazards as well as potential accidents involving the operation of the reactor. The staff evaluated the licensee's analytical assumptions, methods, and results. In addition, the NRC staff has obtained independent analyses of accidents with TRIGA-fueled reactors (NUREG/CR-2387), and has compared those results with accidents analyzed by the licensee. Since the TSs limit steady-state and pulsed operations below the bounds established for safe operation of TRIGA reactors, none of the potential accidents considered in the OSTR SAR would lead to significant public exposure.

The basic limit used to ensure safe operations of TRIGA reactors is the fuel temperature, which is measured in an IFE. The NRC has accepted (NUREG-1537, Appendix 14.1) that no fuel damage or cladding failure is expected if the fuel temperature never exceeds 1150 °C when the cladding temperature is less than 500 °C. This is the safety limit given in TS 2.1. If the cladding temperature is greater than 500 °C no damage is expected if the fuel temperature does not exceed 950 °C. This is the criterion used for a loss-of-coolant accident. The TSs were designed to maintain the fuel temperature far enough below these temperatures during either steady-state or pulsed operation to ensure significant safety margin.

TRIGA reactors are licensed by the NRC with steady-state power levels up to 2.3 MW(t) with natural circulation of the coolant water. At power levels of about 1.5 MW(t), the saturation fission product inventory would be small enough that an instantaneous loss of coolant would not result in fuel damage. The OSTR is nominally operated at 1 MW(t) with a limit of 1.1 MW(t) set in TS 3.1.1.

The typical energy generation of the reactor is 15 to 20 megawatt hour (MWh) per week with an expected maximum operation of 70 MWh per week, so that any routine operation of the OSTR will not result in a saturated fission product inventory. However, because there is no limitation on operating time in the license or TSs, the safety analyses assumed an inventory of saturated fission products. In the pulsed mode, GA performed numerous tests with step insertions up to $5.00 before any fuel damage became apparent. TS 3.1.4 imposes a step reactivity insertion limit for pulsing of $2.55, keeping the OSTR within the safety envelope established by GA and approved by the NRC.

None of the credible accidents postulated would lead to the failure of the cladding of any fuel elements or the uncontrolled release of fission products. However, the licensee postulated an enveloping event involving the rupture of the cladding of an irradiated fuel element in air which is the standard enveloping event for TRIGA-design research reactors. This event would lead to the maximum potential radiation hazard from fuel failure to facility personnel and members of the public. The licensee makes no assumptions as to the cause of the failure. The licensee evaluated only the potential consequences of this event, not the likelihood or mechanisms of the event's occurrence. This worst-case scenario for this accident has been designated as the MHA and for a TRIGA reactor, for purposes of classification, is referred to as the "fuel-handling accident." The licensee and the staff have evaluated other possible accident sequences that originate in the intact reactor core; none pose a significant risk of cladding failure or release of fission products. If this cladding were ruptured, noble gases and halogen fission products could escape from the gap between the fuel and the cladding.

In NUREG-1537 a total of nine accidents are suggested, though the OSTR SAR states that not all are credible for the OSTR. Two additional accidents are discussed in the SAR. The following accidents are considered:

- fuel element drop accident or cladding failure out of the pool (MHA)
- insertion of reactivity
- loss-of-coolant accident (LOCA)
- loss-of-coolant flow
- mishandling or malfunction of fuel underwater
- experiment malfunction
- loss of normal electrical power
- external events
- mishandling or malfunction of equipment
- beam port flooding
- metal-water reactions

13.1 Fuel Element Drop Accident or Cladding Failure Out of the Pool (MHA)

The failure of fuel element cladding outside the pool has been designated the MHA for the OSTR. Such a failure could be from improper handling of fuel, a manufacturing defect, or corrosion.

The worst-case scenario assumes that the cladding of the most irradiated fuel element fails in the reactor room air. This occurs after a long run at full licensed power so the inventories of all radionuclides of significance (except krypton-85) in the scenario are at their maximum values. The licensee did not try to develop a detailed mechanistic scenario, but assumed that the cladding of one fuel element fails and that all of the fission products accumulated in the gap are released abruptly. These nuclides would diffuse in the ambient air of the reactor room and be released to the environment.

Several series of experiments at GA have obtained data on the species and fractions of fission products released from U-ZrH_x under various conditions (Simnad et al., 1976; Foushee and Peters, 1971; Baldwin, Foushee, and Greenwood, 1980). The noble gases were the principal species found to be released. GA developed a correlation for fission product release fraction based on fuel temperature. When the fuel specimen was irradiated at temperatures below about 350 °C, the fraction of the total inventory that was released could be summarized as a constant independent of operating temperature. The release fraction increased as temperature increased above 350 °C. The licensee assumed an average fuel temperature of 500 °C.

Because the noble gases do not condense or combine chemically, it is assumed that any released from the cladding will diffuse in the air until their radioactive decay. On the other hand, the iodines are chemically active, and are not volatile below about 180 °C. Therefore, some of the radioiodines will be trapped by materials with which they come in contact, such as water, and reactor or building structures. Evidence indicates that most of these iodines will either not become or not remain airborne under many accident scenarios applicable to research reactors. However, to be certain that the fuel-cladding failure scenarios discussed below led to upper limit dose estimates for all events, the licensee assumed that 25 percent of the iodines in the gap also become airborne and could be released to the environment. This assumption will lead to computed thyroid doses that could be higher than actual doses.

The staff has reviewed the licensee's methods for computing the dose within and beyond the confines of the reactor facility in case of a fission product release. Three scenarios were considered by the staff and the licensee as discussed in the following section. Doses to the most exposed worker, the location of the nearest permanent residence, and the most exposed member of the public were analyzed.

13.1.1 Scenario for Failure of Fuel Element in Air

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

- The highest power density FLIP fuel element failed outside of the pool after 1 continuous year of operation at 1 MW(t). Hence, the radioactive halogen and noble gas fission products (with the exception of krypton-85) are at saturation.
- No credit is given for decay time.
- The SAR assumed an average fuel temperature of 500 °C, which is conservatively higher than the expected value of 350 °C given the power of the reactor and the power peaking of the highest power density fuel element. This results in a release fraction of 1.22×10^{-4} from the fuel to the gap.
- All of the gap activity is released to the reactor room, and is instantly mixed uniformly with the air.
- All of the noble gases from the gap are assumed to be available for release to the environment. Twenty-five percent of the halogens are available for release to the environment.
- All releases from the reactor building to the environment are assumed to occur at ground level, which takes no credit for release height or building wake effects, which is conservative.
- The most stable atmospheric class (Pasquill F) was assumed.
- The closest person in the unrestricted environment was assumed to be 33 feet from the reactor building. The nearest office building is 330 feet from the reactor building. The nearest permanent residence is 875 feet away from the reactor building. Persons remain for the entire duration of the event.

Three scenarios were considered:

- Scenario A assumes the north wall of the reactor building has vanished. In this scenario, it takes 8.52 seconds for the radioactive gases to be released to the outside atmosphere.
- In Scenario B, the gases are released to the outside atmosphere at a rate equal to the ventilation system release rate of 4.39×10^{6} cubic centimeters per second ($cm^{3}s^{-1}$). In this scenario, it takes 14.7 minutes to release the gases.

- In Scenario C, the reactor room is assumed to leak to the outside at a rate of 1.69×10^4 cm^3s^{-1}, requiring 63.7 hours to release the fission products to the outside. The room leakage rate was conservatively assumed to arise due to a pressure differential between the reactor room and the outside through the combination of a rise in internal room temperature and a drop in outside atmospheric pressure.

The licensee's fission product inventories are similar to inventories quoted for other TRIGA reactors. As expected, the highest dose to a member of the public occurred in Scenario A, and the highest dose to the OSTR staff occurred in Scenario C.

For Scenario A, the calculations for the dose (TEDE) to a member of the public standing 33 feet from the reactor building is 19 mrem. The dose at the nearest office building is 3 mrem and at the nearest residence is less than 1 mrem. These doses are within the regulatory limit given in 10 CFR 20.1301 for doses for individual members of the public of 100 mrem. The doses for Scenario B and C are lower.

For Scenario C, which is based on a 5-minute building evacuation time for the OSTR staff, the occupational TEDE was calculated to be 23 mrem. This is within the regulatory limit given in 10 CFR 20.1201(a)(1)(i) of 5000 mrem. The thyroid dose (sum of the deep dose equivalent and the committed dose equivalent) was calculated to be 505 mrem. This is within the regulatory limit given in 10 CFR 20.1201(a)(1)(ii) of 50 rems (50,000 mrem). The licensee stated that 5 minutes is a reasonable amount of time to perform an evacuation including the time it takes to check out systems to eliminate a potential false alarm.

The licensee also considered the exposure from building shine due to the source term inside the reactor room. This is not a significant contributor to dose outside the reactor room due to the shielding effect of the reactor building wall.

In accordance with the discussions and analysis above, the staff concludes that, if a fuel element from the reactor were to release noble gas and halogen fission products accumulated in the fuel-cladding gap, radiation doses to both occupational personnel and to the public in unrestricted areas would be below the limits of 10 CFR Part 20. The staff also concludes that the licensee has the capability to evaluate airborne releases of radioactive materials.

13.2 Insertion of Reactivity

The Nordheim-Fuchs model was used to determine the maximum step reactivity insertion that would result in a peak fuel temperature of 1100 °C (fuel damage may occur at fuel temperatures above 1150 °C which is the safety limit). The Nordheim-Fuchs model is a longstanding model used for this purpose. The calculations were performed for EOL and BOL FLIP fuels along with the low-enrichment standard fuels. The most limiting value of step reactivity insertion was found to be $2.59 for EOL FLIP fuel.

Given the limiting value of $2.59, the licensee proposed the following TS limit on pulse mode operation of $2.55:

TS 3.1.4 The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical block and electrical interlock on the transient rod, such that the reactivity insertion shall not exceed $2.55.

TS 3.1.4 is acceptable to the staff because it limits pulsing reactivity insertions to values shown to protect the fuel safety limit.

For the uncontrolled withdrawal of a single control rod scenario, the transient control rod is assumed to be withdrawn from the core. The transient control rod is used because it has the highest total worth and, therefore, the highest worth reactivity addition per second. At an initial steady-state power of 100 watts(t), the maximum reactivity insertion resulting from the withdrawal of the transient rod is $1.06 before reaching the reactor scram set point at 1.06 MW(t) that initiates a reactor scram. The time for the maximum reactivity insertion also considered a 0.5-second instrumentation response time based on the slowest responding instrument channel. After the scram signal is initiated the power rise is stopped and power starts decreasing as the other control rods are inserted into the reactor core in no more than 2 seconds (TS limit). The maximum reactivity insertion of $1.06 is well below the limiting reactivity insertion of $2.55 where the fuel temperature stays below the safety limit. Therefore, this event poses no significant threat to the reactor.

Initiating the transient control rod withdrawal scenario from steady-state power of 1 MW(t), the maximum reactivity insertion is $0.15 before initiating a reactor scram and again this is well below the $2.55 limit. Therefore, this event poses no significant threat to the reactor.

For the case of all of the control rods being simultaneously withdrawn, starting at 100 watts(t), the reactivity insertion is $1.23, again well below the limits set in the TSs. Therefore, this event poses no significant threat to the reactor.

13.3 <u>Beam Port Flooding</u>

There are four beam ports in the OSTR which are normally filled with either inert gas or air. If all four of these beam ports were to instantaneously flood, the estimated positive insertion of reactivity would be approximately $1.00, well below the $2.55 reactivity insertion limit (TS 3.1.4). Therefore this event poses no significant threat.

13.4 <u>Metal-Water Reactions</u>

Quench tests have shown that no problems occur if the fuel temperature is below 2200 °F. This temperature is above the safety limit and is not expected to be reached under any circumstances. Therefore this event poses no significant threat.

13.5 <u>Loss-of-Coolant Accident</u>

There are two scenarios that could result in a significant loss of coolant from the reactor tank: pumping of the water from the reactor tank or a tank or tank penetration failure. Pumping all of the coolant water from the reactor tank accidentally is not possible, due to the location of the piping and siphon breaks in the piping.

Tank failures due to corrosion or other failures that lead to slow loss of water will be detected by the reactor staff. A drop in pool level would set off a low level alarm. During normal work hours the reactor staff would respond to the alarm. During off hours, the alarm is monitored by the University who would contact the reactor staff. Slow leaks could be compensated for by adding coolant to the reactor pool.

The only means for a rapid loss of coolant is some catastrophic failure of the reactor tank or penetrations, such as an earthquake beyond that for which the reactor was designed to withstand or dropping a heavy load into the reactor pool. If the cladding temperature is greater than 500 °C, as could be expected during a loss-of-coolant accident with air cooling, no fuel damage is expected if the fuel temperature does not exceed 950 °C. As the cladding temperature increases, the material properties of the cladding degrade and the amount of internal pressure (yield stress) needed to cause cladding failure decreases.

If the tank were instantly drained of coolant, analysis performed by TRIGA licensees and GA show that air circulation would be adequate to prevent fuel damage by removing the decay heat of the fuel. For FLIP fuel in a 1 MW(t) reactor operated for an infinite amount of time, studies show that if power in a fuel element is equal to or less than 24 kW(t), the fuel temperature will not exceed the limit. If reactor operations are limited to 70 MWh per week, which is typical of operation of the OSTR, the power in a fuel element needed for clad failure increases to 28 kW(t). These conclusions are supported by other studies performed for TRIGA fuels which show that, in general, as long as the operating power is less than 1.5 MW(t), the fuel cladding should not breach during a loss-of-coolant accident. As shown in Chapter 4 of this SER, the hot rod in the reactor has a power level less than 19 kW(t), well below the 24 kW(t) limit. A more likely scenario than complete draining of the reactor pool is that the core is only partially uncovered. GA concluded that the temperature rise for a partial loss of coolant is less severe than for a complete loss of coolant. This was verified experimentally by GA and is discussed in "Simulated Loss-of-Coolant Accident for TRIGA Reactors," by J.R. Shoptaugh, Jr. Also, TRIGA reactors have negative void coefficients, so even if the reactor tank were to drain while the reactor was at full power without scram, the reactor would shut down upon loss of water from the core.

The main consequence from a catastrophic loss of coolant is the gamma ray dose from the exposed core. For the licensee's analysis, the fission product inventory was assumed to be that for 1 full year operation at 1 MW(t). Coolant was instantly lost. Both of these assumptions are very conservative. No credit was given for attenuation of radiation through the fuel or through the grid plate. The dose was calculated for three locations— at the grating at the top of the reactor tank from direct radiation from the core, at a point within the reactor room from scattered radiation, and at the nearest fence boundary from scattered radiation. The highest dose rate is directly above the core and is 10,000 rem/hr at 10 seconds after shutdown. For the scattered radiation within the reactor building the highest dose rate would be 327 mrem/hr at 10 seconds after shutdown. After 1 day, the dose rate drops to 39 mrem/hr which would allow reactor staff to reenter the reactor room to perform recovery operations. At the fence the dose rate would be 0.015 mrem/hr, well within regulatory limits.

A related concern on a loss of all water from the reactor tank is potential offsite contamination with radioactive water. The reactor pool contains 4600 gallons. Most of the water will drain into the pipe trench which empties into the waste holdup tank. The total capacity of the trench and tank is 5950 gallons. The licensee initially stated the holdup tank was routinely pumped down to a level of 1800 gallons, effectively reducing the combined capacity of the tank and trench to 4150 gallons. This is not enough to contain the full amount of water in the reactor pool in an accident. In response to an RAI, the licensee agreed to change the OSTR procedures to pump down to a level of 1350 gallons, thus ensuring that there is always a 4600-gallon capacity in the trench and waste holdup tank. As noted in Chapters 4 and 9 of this SER, the equilibrium concentrations of predominant radionuclides in the primary coolant are within the 10 CFR Part 20, Appendix B limits for release to the sewer. The licensee's actions to contain coolant lost from the reactor tank during a loss-of-coolant accident are acceptable to the staff.

The lifting of heavy loads in the reactor bay is covered under Oregon State TRIGA Reactor Operating Procedure 23, "Crane Operation Procedures." This procedure details the authorization for use of the crane, rigging and lifting procedures, and limitations of use. That procedure prohibits the movement of heavy loads over the biological shielding except as part of a task that would specifically require it and the movement is approved by the reactor supervisor. The licensee's control of heavy loads over the reactor is acceptable to the staff.

The staff has reviewed the loss-of-coolant accident. The loss-of-coolant accident does not result in damage to the reactor fuel. Doses within the reactor room and at the site boundary are below the limits of 10 CFR Part 20 and are therefore acceptable to the staff.

13.6 Loss of Coolant Flow

This event is the loss of ability to cool the primary coolant in the reactor pool. This occurrence is not likely to result in any consequences. Numerous alarms (bulk water temperature, water level, water flow, and radiation monitors) are available to signal the need for operator action to shut down the reactor. Even if there were a loss in the ability of the primary and secondary cooling systems to remove heat from the primary coolant, and the reactor remained at full power, it would take more than 9 hours for the water level to evaporate down to the top of the core. As the water level dropped past the top of the core, the negative void coefficient of reactivity would shut down the reactor. Makeup water could be easily provided from external sources by the operators. Because the reactor operators have multiple indicators of the loss-of-coolant flow, the staff concludes that the reactor will be shut down in a timely manner if loss-of-coolant flow occurred.

13.7 Mishandling or Malfunction of Fuel Underwater

Cladding might fail if a fuel element were to be dropped underwater during transfer or during operation in the reactor due to a manufacturing defect, corrosion, or overheating of the fuel element. This accident is bounded by the MHA. The three cases of releases that were analyzed for the MHA were also considered for this situation. However, for this case most of the halogens will be scrubbed by the primary coolant in the reactor pool, so the doses will be lower than for the MHA. Occupational doses were calculated based on a 5-minute evacuation time. The licensee stated that the 5-minute evacuation time is a reasonable amount of time to perform an evacuation including checking out systems to ensure that it is not a false alarm.

The maximum dose (TEDE) to a person in the unrestricted area at the site boundary (33 feet) is 6 mrem. The maximum occupational TEDE was calculated to be 6 mrem. The thyroid dose (sum of the deep dose equivalent and the committed dose equivalent) was calculated to be 31 mrem. These values are well within the requirements of the regulations given above.

As discussed above, the licensee established a reactor operating procedure for lifting heavy loads in the reactor bay that details the authorization process for the various steps involved in moving a heavy load. Moving a heavy load over the biological shield requires special authorization by the reactor supervisor.

13.8 Experiment Malfunction

A malfunction of an experiment can result in over pressurizing an experiment or generating more radioisotopes than expected. The over pressurizing of an experiment can result in a sudden release of radioisotopes or damage to the core. It could also result in a step insertion of reactivity. Therefore, there are procedures, requirements and limitations set in TSs 3.8, 4.8, and 6.5 and in the operating procedures regarding the control and review of all reactor experiments. Limitations on experiments are discussed in detail in Chapter 10 and 12 of this SER. The review process of a proposed experiment includes a safety analysis that assesses the complete range of safety issues such as the generation of radionuclides; the reactivity worth of the experiment, material properties such as chemical, physical, explosive, and corrosive characteristics of each experiment; and potential failures and malfunctions.

TS 3.8.1 limits the absolute value of the reactivity worth of any single unsecured experiment to less than $0.50, a value well within the reactivity addition limit of TS 3.1.4. The sum of the absolute values of the reactivity worth of all experiments shall be less than $2.55, which is within the reactivity additional limit of TS 3.1.4. Potential reactivity malfunctions are bounded by the discussion of the insertion of reactivity accident discussed in Section 13.2 above.

TS 3.8.2 limits the irradiation of explosives in the reactor to less than 25 milligrams. In addition, a condition for irradiation is to show that 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 in which the explosive is irradiated. In Section 13.2.6.2 of the SAR, the licensee presents some example calculations to show container design characteristics.

TS 3.8.3 addresses experiment failures and malfunctions. Where the possibility exists that the failure of an experiment could, under normal operating conditions of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment, release radioactive gases or aerosols to the reactor bay or the unrestricted area, the quantity and type of material in the experiment is limited such that the airborne radioactivity in the reactor bay or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR Part 20.

Based on the licensee's analyses of experiment malfunction and the limitations placed on the conduct of experiments by the TSs, the staff finds the risk of experiment malfunction to be acceptably low.

13.9 <u>Loss of Normal Electrical Power</u>

Although the OSTR has an emergency electric power generator, the emergency electrical power system is not necessary to safely shut down the reactor and is not required to ensure public health and safety. In the event of a loss of electrical power without emergency power, all control rods would insert into the core automatically by gravity due to the loss of power to the electromagnets and for the transient rod, the three-way solenoid valve which holds the control rods in position. Upon loss of electrical power, the primary and secondary coolant pumps would stop. Reactor decay heat would be dissipated through natural circulation in the primary coolant. 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. The OSTR has a backup power system that powers the reactor console and control instrumentation allowing the operator to monitor the shut down of the reactor even if electrical power is lost. However this is not needed as the operator can easily verify shutdown of the reactor manually by visually inspecting the core from the reactor top. The staff concludes that loss of normal electric power poses little risk to the health and safety of the public of the OSTR staff.

13.10 External Events

External events are discussed in Chapter 3 of this SER. Hurricanes, floods, and tornadoes are very rare for the area around the OSTR site, so these external events were not specifically considered in the SAR. The seismic activity in the area is also low, and the building was designed and built to exceed seismic building code requirements. The potential consequence of external events would be bounded by the MHA.

13.11 Mishandling or Malfunction of Equipment

The TRIGA control systems have several interlocks and automatic shutdown circuits built into them. Since TRIGA reactors are designed for large step insertions of reactivity with no fuel damage, the SAR states that no credible accident that will result in damage to the reactor can occur as a result of an situation where equipment was mishandled or malfunctioned.

13.12 Conclusions

The staff has reviewed the licensee's analyses of potential accidents at the reactor facility. The staff concludes that the licensee has postulated and analyzed sufficient accident initiating events and scenarios. On the basis of its evaluation of the information presented in the licensee's SAR, the staff concludes as follows:

- Using conservative assumptions, the MHA will not result in occupational radiation exposure of the facility staff or radiation exposure of the public in excess of the applicable NRC limits in 10 CFR Part 20.

- For accidents involving insertions of excess reactivity, the licensee has demonstrated that a pulse reactivity limit of $2.55 will result in peak fuel temperatures below the safety limit (TS 2.1). Insertions of excess reactivity resulting both from the uncontrolled withdrawal of the transient rod and the simultaneous withdrawal of all control rods are less than $2.59 and hence do not pose a threat to fuel integrity.

- The loss-of-coolant accident does not result in damage to the reactor fuel. Doses within the reactor room and at the site boundary are below the limits of 10 CFR Part 20.

- On the basis of its review, the staff concludes that there is reasonable assurance that no credible accident would cause significant radiological risk to the facility staff, the environment, or the public.

14. TECHNICAL SPECIFICATIONS

The staff has evaluated the TSs as part of its review of the application for renewal of Facility License No. R-106. The TSs define certain features, characteristics, and conditions governing the operation of the OSTR. The TSs are explicitly included in the renewed license as Appendix A. The staff reviewed the format and content of the TSs for consistency with the guidance found in ANSI/ANS-15.1 and NUREG-1537. Other chapters of this SER discuss the evaluation of individual TSs. The staff specifically evaluated the content of the TSs to determine if the TSs meet the requirements in 10 CFR 50.36. The staff concluded that the OSTR TSs do meet the requirements of the regulations. The staff based this conclusion on the following findings:

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

- The OSTR 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 license will include the TSs. To satisfy the requirements of 10 CFR 50.36(b), the licensee provided TSs derived from analyses in the OSTR SAR.

- To satisfy the requirements of 10 CFR 50.36(d)(1), the licensee provided TSs specifying a safety limit on the fuel temperature and LSSSs for the reactor protection system to preclude reaching the safety limit.

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

- The TSs contain surveillance requirements that satisfy the requirements of 10 CFR 50.36(d)(3).

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

- The TSs contain administrative controls that satisfy the requirements of 10 CFR 50.36(d)(5). The licensee’s administrative controls contain requirements for initial notification, written reports, and records that meet the requirements specified in 10 CFR 50.36(d)(1),(2),(7), and (8).

The staff finds the TSs to be acceptable and concludes that normal operation of the OSTR within the limits of the 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 occupational exposures. The staff also finds that the TSs provide reasonable assurance that the facility will be operated as analyzed in the OSTR SAR, and adherence to the TSs will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 13 of this SER.

15. FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Operate the Reactor

As stated in 10 CFR 50.33(f), "Except for an electric utility applicant for a license to operate a utilization facility of the type described in 10 CFR 50.21(b) or 10 CFR 50.22, [an application shall state] information sufficient to demonstrate to the Commission the financial qualifications of the applicant to carry out, in accordance with the regulations of this chapter, the activities for which the permit or license is sought."

OSU does not qualify as an "electric utility," as defined in 10 CFR 50.2, "Definitions." Further, pursuant to 10 CFR 50.33(f)(2), the application to renew or extend the term of any operating license for a nonpower reactor shall include financial information that is required in an application for an initial license. Therefore, the staff has determined that OSU must meet the financial qualifications requirements pursuant to 10 CFR 50.33(f), and is therefore subject to a full financial qualifications review by the NRC. OSU must provide information to demonstrate that it possesses or has reasonable assurance of obtaining the necessary funds to cover estimated operating costs for the period of the license. It must submit estimates for the total annual operating costs for each of the first 5 years of facility operations from the time of expected license renewal, and indicate the source(s) of funds to cover those costs.

OSU submitted an estimated expense and income table for the reactor facility for the 5-year period ranging from FY2008 to FY2012. The table shows that projected expenses range from approximately $621,000 in FY2008 to $755,000 in FY2012, and that they would be covered by income mostly from State of Oregon funds appropriated to the University. Other sources of income include grants and awards. The staff reviewed the licensee's estimated operating costs and projected sources of funds, and found them to be reasonable.

Based on its review, the staff finds that OSU has demonstrated reasonable assurance of obtaining the necessary funds to cover the estimated facility operations costs for the period of the license. Accordingly, the NRC staff has determined that OSU has met the financial qualifications requirements pursuant to 10 CFR 50.33(f), and is financially qualified to hold the renewed license for the OSTR.

15.2 Financial Ability to Decommission the Facility

The NRC has determined that the requirements to provide reasonable assurance of decommissioning funding are necessary to ensure the adequate protection of public health and safety. The regulation at 10 CFR 50.33(k) requires that an application for an operating license for a utilization facility contain information to demonstrate how reasonable assurance will be provided and that funds will be available to decommission the facility. The regulation at 10 CFR 50.75(d) requires that each non-power reactor applicant for or holder of an operating license shall submit a decommissioning report which contains a cost estimate for decommissioning the facility, an indication of the funding method(s) to be used to provide funding 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 specified in 10 CFR 50.75(e)(1). The licensee has elected to use a statement of intent to provide financial assurance, as allowed by 10 CFR 50.75(e)(1)(iv) for a Federal, State, or local government licensee. The statement of intent must contain a cost estimate for decommissioning and indicate that funds for decommissioning will be obtained when necessary.

The licensee's statement of intent, dated July 31, 2007, contains a decommissioning cost estimate of $12.2 million for the DECON option, and states, "When a decision is made to terminate the facility license and decommission the facility, the University will request legislative appropriation of funds, or otherwise provide funds sufficiently in advance of decommissioning to prevent delay of required activities." The cost estimate includes the costs for labor, energy, and waste burial in the same proportions to total cost as those in the cost adjustment factor formula for power reactors at 10 CFR 50.75(c)(2).

The staff evaluated the reasonableness of the licensee's decommissioning cost estimate using two bases. First, the OSTR was used as a reference reactor in the study "Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors" (published as NUREG/CR-1756). The licensee used the reference costs in this study (which included costs for waste burial, labor, energy, other miscellaneous costs, and a 25-percent contingency factor) as the initial basis for the OSTR decommissioning cost estimate, then escalated the estimate for inflation on a yearly basis until 1993. Second, in 1993, the licensee escalated its cost estimate and established a new baseline for its estimates based on a report prepared by the Non-destructive Testing Information Analysis Center entitled "Cost Estimate for the Decommissioning of the McClellan Nuclear Radiation Center." Based on its review of the information noted herein, the NRC staff concludes that the decommissioning cost estimate for the OSTR is reasonable. The licensee stated that it will update its decommissioning cost estimate using the 10 CFR 50.75(c)(2) methodology described above. Changes in labor costs will be obtained from the U.S. Bureau of Labor Statistics (BLS) index "Employment cost index for total compensation, private industry workers, by region and area, not seasonally adjusted (Base 1982);" changes in energy costs will be obtained from the BLS index "Producer price index and percent change for selected commodity grouping (Base 1982) (Commodity 05-43 Industrial Electrical Power, and Commodity 05-73 Diesel Fuel)," and changes in waste disposal costs will be obtained from NUREG-1307 for pressurized water reactors for the low-level radioactive waste disposal facility in the State of Washington.

To support the statement of intent and the licensee's qualifications to use a statement of intent, the licensee stated that OSU is a State [of Oregon] agency and provided references which corroborate this statement. The Oregon State Board of Higher Education is itself an agency of the State of Oregon, and since it has jurisdiction over OSU, OSU is therefore a State agency. The licensee also provided information supporting its representation that the decommissioning funding obligations of OSU are backed by the full faith and credit of the State of Oregon. Further, the licensee provided a memo from the OSU Vice President for Finance and Administration which identifies the Vice President for Research (currently Dr. John Cassidy, the signator of the statement of intent) as having authority to enter contracts on behalf of the University; therefore, Dr. Cassidy has the authority to execute the statement of intent and bind the University financially.

The staff reviewed the licensee's information on decommissioning funding assurance as described above and finds that the licensee is a State Government licensee under 10 CFR 50.75(e)(1)(iv), the statement of intent is acceptable, the decommissioning cost estimate for the DECON option is reasonable, and the OSU means of adjusting the cost estimate and associated funding level periodically over the life of the facility are reasonable.

15.3 Foreign Ownership, Control, or Domination

Section 104d of the AEA prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The NRC regulation at 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application, OSU is an agency of the State of Oregon and is not a corporation, and there is no foreign control of the University. The NRC staff does not know or have reason to believe that OSU will be owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government.

15.4 Nuclear Indemnity

The staff notes that the licensee currently has an indemnity agreement with the Commission, and said agreement does not have a termination date. Therefore, OSU will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.71, "Scope," OSU, as an educational institution licensee, is not required to provide nuclear liability insurance. The Commission will indemnify OSU 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, "Form of Indemnity Agreement with Nonprofit Educational Institutions," for up to $500 million and above $250,000. Also, OSU is not required to purchase property insurance under 10 CFR 50.54(w).

15.5 Conclusions

The staff reviewed the financial status of the licensee and concludes that there is reasonable assurance that the necessary funds will be available to support the continued safe operation of the OSTR and, when necessary, to shut down the facility and carry out decommissioning activities.

16. PRIOR REACTOR UTILIZATION

As detailed in previous sections of this SER, the NRC staff concludes that continued operation of the OSTR will not pose a significant radiological risk. The bases for these conclusions include the assumption that the facility systems and components are in good working condition. Systems and components that perform safety-related functions must be maintained or replaced to ensure that they continue to protect adequately against accidents. Such systems and components found at the OSTR include the fuel cladding and the reactor safety system.

Because the staff has concluded that the reactor was initially designed and constructed to be inherently safe, the staff must also consider whether operation will cause significant degradation in safety features. Furthermore, because loss of integrity of fuel cladding is the MHA, the staff has considered mechanisms which could increase the likelihood of failure. Possible mechanisms are (1) radiation degradation of fuel element cladding strength, (2) high internal pressure caused by high temperature leading to exceeding the elastic limits of the cladding, (3) corrosion or erosion of the cladding leading to thinning or other weakening, (4) mechanical damage as a result of handling or experimental use, and (5) degradation of safety components or systems.

16.1 Fuel Element Aging

Section 4.2.1 of this SER describes the reactor fuel. TRIGA fuel has a large negative prompt temperature coefficient of reactivity, high fission product retention, chemical stability when quenched from high temperatures in water, and dimensional stability over a range of temperatures. The most conservative maximum fuel temperature estimate of 588 °C for an IFE temperature of 510 °C is sufficiently below the 750 °C temperature where swelling of the fuel element might be expected. Therefore, there is reasonable assurance that fuel swelling has been precluded during past operation. Further, the licensee helps to ensure the integrity of the fuel through TS 3.1.6 inspections of the fuel cladding to detect gross failure or visual deterioration. Attributes inspected include the fuel element transverse bend and length, and a visual inspection for bulges or other cladding defects. As stated in Chapter 4 of the SER, the design and development program for the fuel provides reasonable assurance that the fuel will function safely in the OSTR core for the renewal period without adversely affecting public health safety.

In TS 3.1.4, the licensee has limits on pulsing the reactor. The maximum reactivity addition for pulsing helps to ensure that the reactor can be safely pulsed without fuel damage. The large, prompt negative temperature coefficient of reactivity of the uranium-zirconium hydride fuel moderator provides a basis for safe operation of the reactor in the nonpulsing mode and is the essential characteristic supporting the operational capability of the reactor in pulse mode. In the pulsed mode of operation, TS 3.1.4 limits step insertion of reactivity to $2.55. GA performed numerous tests with step insertions up to $5.00 before any fuel damage became apparent; therefore, the self-imposed limit of $2.55 is well within the safety envelope established by GA and approved by the NRC.

The FLIP TRIGA fuel in the core has been in use since 1976 and has been subjected to moderate burnup levels of uranium-235. FLIP TRIGA fuel at the GA Mark F reactor has been subjected to higher burnup, with no observable degradation of cladding as a result of radiation.

Fuel aging should be considered normal with use of the reactor and is expected to occur gradually. There is some evidence that the $U\text{-}ZrH_x$ fuel tends to fragment with use, probably as a result of the stresses caused by high temperature gradients and a high rate of heating during pulsing (GA-4314, 1980). Some of the possible consequences of fragmentation are a decease in thermal conductivity across cracks, leading to higher central fuel temperatures during steady-state operation (temperature distributions during pulsing would not be affected significantly by changes in conductivity because a pulse is completed before significant heat redistribution by conduction occurs), and more fission products would be released into the cracks in the fuel.

With regard to the first item above, hot cell examinations of thermally stressed hydride fuel bodies have shown relatively widely spaced radial cracks that would cause minimal interference with radial heat flow (GA-4314, 1980). However, after pulsing, TRIGA reactors have exhibited small increases in both steady-state fuel temperatures and power reactivity coefficients. At power levels of 500 kW(t), temperatures have increased by approximately 20 °C and power reactivity coefficients by approximately 20 percent (GA-5400, 1965). GA has attributed these changes to an increased gap between the fuel material and cladding caused by rapid fuel expansion during pulse heating, which reduces the heat transfer. Experience has shown that the observed changes occur mostly during the first several pulses and have essentially saturated after 100 pulses.

Two mechanisms for fission product release from TRIGA fuel have been proposed (GA-4314, 1980). The first mechanism is fission fragment recoil into gaps within the fuel cladding. This effect predominates up to about 400 °C and is independent of fuel temperature. GA has postulated that in a closed system such as exists in a TRIGA fuel element, fragmentation of the fuel material within the cladding will not cause an increase in the fission product release fraction (GA-8597, 1968). The reason for this is that the total free volume available for fission products remains constant within the confines of the cladding. Under these conditions, the formation of a new gap or widening of an existing gap must cause a corresponding narrowing of the existing gap at some other location. Such a narrowing allows more fission fragments to traverse the gap and become embedded in the fuel or cladding material on the other side. In a closed system in which the density of the fuel is constant, the average gap size and, therefore, the fission product release rate remains constant, independent of the degree to which fuel material is broken up.

Above approximately 400 °C, the controlling mechanism for fission product release is diffusion through the ZrH, and the amount accumulated in the gap is dependent on fuel temperature and fuel surface-to-volume ratios. The licensee has accounted for the impact of fission product release with increasing fuel temperature in the analysis of the MHA.

Water flow through the core is obtained by natural thermal convection. Natural convection cooling does not generate the coolant velocities or pressures necessary to erode the cladding.

The design of the in-pool structures and components minimizes the chance for mechanical impact. Reactor components are contained between top and bottom aluminum grid plates. The core support assembly, which is supported and located by the reflector can, accurately positions and aligns fuel elements for all anticipated operating conditions. The core support assembly also provides acceptable guides and supports for the control rods, the fuel moderator and graphite dummy elements, guide tubes, and the pneumatic transfer tube. The design ensures that no deleterious mechanical impact will damage the reactor during fuel handling.

The fuel is handled as infrequently as possible, consistent with periodic surveillance. Any indications of possible damage or degradation are investigated. The only experiments which are placed near the core are isolated from the fuel cladding by a water gap and at least one metal barrier, such as the pneumatic tube or the central thimble.

TS 3.3 places requirements on the conductivity and pH of the primary coolant. TS 4.3 specifies surveillance intervals for the chemical properties of the coolant. These TSs help to ensure that no significant corrosion of the cladding has occurred or will occur.

The staff concludes that the likely processes of aging of the U-ZrH_x fuel moderator under OSTR operating conditions would not cause significant changes in the operating temperature of the fuel or affect the accumulation of gaseous fission products within the cladding. The staff also concludes that there is reasonable assurance that fuel aging will not significantly increase the likelihood of fuel-cladding failure, or the quantity of gaseous fission products available for release in the event of a loss of cladding integrity for TRIGA fuel operated under the conditions of the OSTR.

16.2 Aging of Safety Components

The reactor contains redundant safety-related measuring channels and control rods. Under most normal operating conditions, failure of all but one control rod would not prevent a reactor shutdown to a safe condition. In addition, safety system failures normally lead to a reactor scram. Important parameters (such as reactor power) have redundant safety systems. If a failure occurs that would prevent a scram upon receipt of a valid scram signal in one channel, the redundant channel would scram the reactor.

The electrical design of the reactor safety system (e.g., safety channel circuitry, control rod magnets) helps to preclude accidents as a result of failure of system components. Failure or removal for maintenance of safety-related I&C components causes a safe reactor shutdown. TS 4.2 specifies surveillance requirements for the reactor safety system. These requirements ensure that gradual degradation of system components will be detected. Additionally, the OSTR staff performs regular preventive and corrective maintenance and replaces system components as necessary. Nevertheless, some equipment malfunctions have occurred. The staff's review indicates that most of these malfunctions were one of a kind and typical of even industrial-quality electrical and mechanical instrumentation and components. The staff concludes that there is no indication of significant degradation of the instrumentation and components, and there is strong evidence that the OSTR staff will remedy any future degradation with prompt corrective action.

The staff did not consider prior utilization of other systems and components because degradation would occur gradually, be readily detectable, or not affect the likelihood of accidents. Some examples include degradation of the reactor pool liner, secondary coolant pump, or chart recorders. Section 4.3 of this SER discusses potential pool liner degradation.

16.3 Conclusions

In addition to the considerations discussed above, the NRC staff reviewed licensee event reports and inspection reports. On the basis of this review and the preceding considerations, the NRC staff concludes that there has been no significant degradation of facility systems or components. The NRC staff further concludes that the surveillance requirements in the TSs provide reasonable assurance that the facility will continue to be adequately monitored for degradation of systems and components.

17. CONCLUSIONS

On the basis of its evaluation of the application as discussed in the previous chapters of this SER, the staff concludes the following:

- The application for license renewal dated October 5, 2004, as supplemented, complies with the standards and requirements of the AEA, 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 well as the provisions of the AEA and the rules and regulations of the Commission.

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

- The 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.

18. References

For details with respect to the application for renewal, see the licensee's letter dated October 5, 2004 (ADAMS Accession Nos. ML043270077 and ML071430452), as supplemented by letters dated August 8, 2005 (ADAMS Accession No. ML052290051); May 24, 2006 (ADAMS Accession No. ML061510355); November 10, 2006 (ADAMS Accession No. ML063210182); November 21, 2006 (ADAMS Accession No. ML063320500); July 10, 2007 (ADAMS Accession No. ML072150361 and ML072150362); July 27, 2007 (ADAMS Accession No. ML072150363); July 31, 2007 (ADAMS Accession No. ML 072190043), August 6, 2007 (ADAMS Accession No. ML072340580), April 14, 2008 (ADAMS Accession No. ML081150194), August 6, 2008 (ADAMS Accession No. ML082261409), and August 11, 2008 (ADAMS Accession No. ML082270383). The dates and associated ADAMS accession numbers of NRC RAIs are May 15, 2006 (ADAMS Accession No. ML061310209), October 3, 2006 (ADAMS Accession No. ML062060026), May 21, 2007 (ADAMS Accession No. ML071300010), February 4, 2007 (ADAMS Accession No. ML072890036), and March 19, 2008 (ADAMS Accession No. ML080710139). For details with respect to the issuance of the renewed facility license, see renewed Facility License No. R-106 (ADAMS Accession No. ML082520047), the related safety evaluation report (ADAMS Accession No. ML071430452), and the related environmental assessment dated September 8, 2008 (ADAMS Accession No. ML061650197).

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