SAFETY EVALUATION REPORT
Related to the Renewal of the Operating License for the TRIGA Research Reactor at the Kansas State University

Docket No. 50-188

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

March 13, 2008

CONTENTS

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The NRC staff (the staff) conducted this review in response to an application filed by the Kansas State University (the applicant or licensee) for (1) renewal of its facility operating license (R-88) to operate the Kansas State University training reactor and isotopes production, General Atomics (TRIGA) research reactor, (2) an increase in the maximum steady-state power level from 250 kilowatts (thermal) (kW(t)) to 1250 kW(t), and (3) an increase in the pulse reactivity insertion limit to $3.00 from its present limit of $2.00. The reactor facility is on the campus of the Kansas State University in Manhattan, Kansas. In its safety review, the staff considered both onsite observations made by NRC personnel and information submitted by the licensee. On the basis of this review, the staff concludes that the Kansas State University TRIGA reactor can operate in accordance with its application and technical specifications without endangering the health and safety of the public and facility staff.

1. THE FACILITY

1.1 Introduction

The Kansas State University (KSU, applicant, or licensee) nuclear research reactor was constructed during 1960–1962, and was originally licensed in 1962 at a power level of 100 kilowatts (thermal) (kW(t)). In 1968, the license was amended to allow operation at 250 kW(t), with pulsing capability to 250 megawatts (MW). KSU submitted an application to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for a 20-year (yr) renewal of its Class 104c facility operating license (R-88) and an increase in its maximum steady-state power level by means of a letter and supporting documentation dated September 12, 2002. The applicant provided supplemental information on November 11, 2002; November 13, 2002; December 21, 2004; July 6, 2005; September 27, 2005; March 20, 2006; March 30, 2006; June 28, 2006; September 28, 2006; May 17, 2007; June 4, 2007, September 12, 2007; October 11, 2007, and February 6, 2008. Although these supplements provided additional information, they did not expand the scope of the application. This license would authorize the continued operation of the KSU training reactor and isotopes production, General Atomics (TRIGA) research reactor as an NRC-licensed facility with an increase of the steady-state power license limit to 1250 kW(t) from its present limit of 250 kW(t). The license will also allow an increase in the pulse reactivity insertion limit to $3.00 from its present limit of $2.00.

Before issuing the renewed operating license No. R-88, the NRC staff conducted a review based on information in the licensing application, supplemental information, and the licensee's responses to staff requests for additional information (RAIs) and staff questions during site visits. Specifically, the application included financial statements, the safety analysis report (SAR), an environmental report, the Operator Requalification Program, and technical specifications (TSs), also known as Appendix A to the license. The licensee also requested that the staff review and approve a revision of the emergency plan filed with the NRC as part of the application. The licensee has continued to update these documents, both in response to RAIs issued by the staff and as part of its routine document maintenance. Except for the 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 the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. Documents related to this license renewal dated on or after November 24, 1999, 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, if there are problems in accessing the documents located in ADAMS, or if you want access to documents published before November 24, 1999, contact the NRC Public Document Room Reference staff at 1-800-397-4209, 301-415-4737 or by email at pdr@nrc.gov. The licensee is not required by Title 10 of the *Code of Federal Regulations* or by license to have a physical security plan. However, the licensee must comply with the security requirements in Title 10, Section 73.67(f), of the *Code of Federal Regulations* (10 CFR 73.67(f). Qualified staff inspectors verify compliance with these regulations on an established periodicity.

In conducting its safety review, the staff evaluated the facility against the requirements of the following regulations:

- 10 CFR Part 20, "Standards for Protection against Radiation"
- 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material"

- 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”
- 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions”
- 10 CFR Part 55, “Operators’ Licenses”
- 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material”
- 10 CFR Part 73, “Physical Protection of Plans and Materials”

In addition to the above-listed regulations, the staff also evaluated the facility against applicable regulatory guides; relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series; and NRC guidance documents, such as NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Research and Test Reactors.” Because there are no specific accident-related regulations for research reactors, the staff compared calculated dose values for accidents with the standards cited in 10 CFR Part 20. Amendments to 10 CFR Part 20 (Sections 20.1001 through 20.2402 and appendices) became effective on January 1, 1994. Among other items, these amendments changed the dose limits for occupationally exposed persons and members of the public, as well as the concentrations of radioactive material allowed in effluents released from licensed facilities. The licensee must follow the requirements of 10 CFR Part 20, as amended, for all reactor operations.

The purpose of this safety evaluation report (SER) is to summarize the findings of the staff’s safety review of the facility and to delineate the technical details considered in evaluating the radiological safety aspects of continued operation. This SER will serve as the basis for issuing a renewed license for operation of the KSU TRIGA reactor with a thermal power license limit of 1250 kW(t). The reactor can also be operated in a pulse mode with a maximum pulse reactivity addition of $3.00.

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1.2 Summary and Conclusions on Principal Safety Considerations

The staff considered information submitted by the licensee, past operating history recorded in annual reports submitted to the Commission by the licensee, and reports of safety inspections by the NRC staff, as well as first-hand onsite observations. In addition, as part of its licensing review of several TRIGA reactors in the past, the staff obtained laboratory studies and analyses of several accidents postulated for the TRIGA reactor. On the basis of this evaluation and the resolution of principal issues reviewed for the KSU TRIGA reactor, the staff reached the following eight conclusions:

(1) The design, testing, and performance of the KSU TRIGA reactor structure and the systems and components important to safety during normal operation continue to operate as planned, and it is reasonable to expect continued safe operation of the facility.

(2) The licensee’s management organization is adequate to maintain and operate the reactor so that there is no significant radiological risk to facility employees or the public.

(3) The licensee's training, research activities, and security measures are adequate to ensure safe operation of the facility and protection of its special nuclear material (SNM).

(4) The expected consequences of postulated accidents are not likely to exceed the guidelines as specified in 10 CFR Part 20 for doses in unrestricted areas.

(5) Releases of radioactive materials and wastes from the facility are not expected to result in concentrations beyond the limits specified by the Commission's regulations and are as low as reasonably achievable (ALARA).

(6) The licensee's TSs, which contain the operational control limits of the facility, give a high degree of assurance that the facility will be operated in accordance with the assumptions and analyses in the SAR. There has been no significant degradation of equipment, and the TSs will continue to ensure that there will be no significant degradation of equipment.

(7) The financial data submitted with the application show that the licensee has reasonable access to sufficient revenues to cover operating costs and eventually to decommission the reactor facility.

(8) The licensee's emergency plan provides reasonable assurance that the licensee is prepared to assess and respond to emergency events.

On the basis of these conclusions, there is reasonable assurance that KSU can operate its TRIGA reactor in accordance with its application without endangering the public health or safety.

1.2.1 Facility Modifications and History

The reactor first achieved criticality in 1962 at the licensed power level of 100 kW(t). In 1968, a license amendment increased the licensed power level to 250 kW(t) and allowed for pulsing. Since 1968, a number of facility modifications have occurred, including the replacement of the secondary cooling system, heat exchanger, and reactor console; installation of new power-level detectors; increased cooling tower capacity; and enlargement and modernization of the control room. Stainless-steel-clad fuel elements replaced original aluminum-clad fuel elements in 1973. The licensee has scheduled the addition of a fourth control rod to accommodate the power increase to 1250 kW(t).

1.2.2 Reactor Description

The KSU TRIGA reactor was originally designed and constructed to support education, training, research, and public service activities. It is a heterogeneous, water-moderated, water-cooled reactor operated in an open pool. The core is immersed in highly purified water in an open aluminum tank that holds approximately 5,070 gallons (gal) (19,200 liters (L)) of water and is surrounded by reinforced concrete. The core is cooled by natural convection flow. The coolant/moderator is light water, and the reactor core is reflected by light water or graphite. The reactor coolant circulates through an external heat removal and purification system. The reactor facility includes a bulk shield tank next to the reactor core, a pneumatic transfer system, four beam tubes, and a thermal column.

The KSU fuel design is similar to that used by other NRC-licensed TRIGA reactors. The fuel rods are stainless-steel clad, containing the isotope uranium-235. The reactor exhibits a large prompt negative temperature coefficient typical of all TRIGA reactors. Four control rods (three standard rods and one transient/pulsing rod) control reactivity. Limits on total fuel loading, excess reactivity, and maximum pulsing reactivity ensure that operation will not lead to conditions that challenge design-basis temperatures.

1.3 <u>General Description of the Facility</u>

1.3.1 Geographical Location

The reactor is located on the KSU campus, in the city of Manhattan, Kansas, which is located in Riley County. The licensee controls access to the reactor. The operations boundary of the reactor facility encompasses the reactor bay and control room. The site boundary encompasses the reactor facility and adjacent fenced areas. The reactor bay (the confinement building) is a 144,000-cubic-foot (ft^3)(4075-cubic-meter(m^3)) structure made of reinforced concrete and structural steel.

1.3.2 Principal Characteristics of the Site

The site is in the Flint Hills uplands of northeast Kansas, characterized by glacial sediments. Soil bores reveal modest topsoil, varying levels of silt, and clay loams overlying bedrock, limestone, and shale. Ground water exists in sand or gravel layers 18 to 35 feet (ft) below existing grade. The site is located on high ground in the northwest sector of the campus. The climate is temperate, typically experiencing 32 inches (in.) of rain annually. Storm drainage is excellent, and a system serving the entire university collects sanitary sewerage from the reactor building. The site is in a seismic risk zone 2, with minimal liquefaction potential of local soils.

1.3.3 Principal Design Criteria, Operating Characteristics, and Safety Systems

The KSU TRIGA reactor is a light-water-moderated, water-cooled thermal reactor operated in an open pool. The reactor is currently fueled with heterogeneous fuel rods clad with stainless steel and consisting of nearly 20-percent enriched uranium in a zirconium-hydride matrix. Natural circulation of coolant through the core provides reactor cooling.

The core is in the form of a right circular cylinder with a diameter of about 1.5 ft and a height of about 1.25 ft. The core is positioned on the vertical axis near the base of a cylindrical water tank of 13 ft in diameter and 22 ft in height. Control rods in the form of aluminum or stainless-steel-clad boron carbide or borated graphite control criticality and ensure shutdown margin. The licensee will add a fourth control rod to allow for the power increase to 1250 kW(t).

The reactor tank is made of 0.25-in.-thick aluminum, 13 ft in diameter and 22 ft deep, surrounded on the side and bottom by a biological shield of concrete at least 8 ft thick.

1.3.4 Engineered Safety Features

The present licensed reactor power level of 250 kW(t) requires no engineered safety features (ESFs); specifically, neither forced convection cooling nor emergency core cooling are required. As discussed in Chapter 13 of this SER, no ESFs are necessary for power levels up to a steady-state value of 1900 kW(t), a large margin over the proposed 1250 kW(t) steady-state power level.

1.3.5 Instrumentation and Control and Electrical Systems

The four categories of instrumentation and control (I&C) systems include (1) the reactor control system (RCS), (2) process instrumentation, (3) the reactor protection system (RPS), and (4) radiation monitoring systems. Chapter 7 of this SER discusses these systems in detail. Chapter 8 of this SER discusses the electrical system. Most I&C systems are hardwired systems commonly used at TRIGA facilities and manufactured by General Atomics. The licensee upgraded the RCS and RPS in 1993–1994. The licensee will modify the RCS with the addition of a fourth control rod to support the power increase to 1250 kW(t), with design and hardware supplied by General Atomics.

The RCS includes the mechanical and electrical systems for the control rod drives and control rod position indication. Each control rod can be manually inserted or removed from the core, and the control rods are interconnected to the RPS to provide automatic insertion (scram) under the proper conditions.

The process instrumentation system provides for the measurement of process variables that are important for safe operation of the facility. These include temperatures (e.g., coolant and fuel), water level, flow rates, and conductivity. Fuel temperature provides an input to the RPS.

The RPS is designed to ensure reactor and personnel safety by rapidly shutting down the reactor when parameters (e.g., neutron level, rate of power change, and fuel temperature) exceed their limiting values. Operators can also manually initiate reactor shutdown. The system provides for the installation of additional external scrams for specific operations. The RPS interfaces with the RCS.

The radiation monitoring system consists of instrumentation to monitor radiation levels at various points throughout the facility. The system provides indications and alarms but no automatic actions (e.g., reactor scram).

The KSU power grid supplies electrical power to the reactor. Emergency power is not required since the core is cooled by natural convection. A loss of power will result in all of the control rods dropping into the core, thus putting the reactor in a safe configuration. Backup battery systems provide for emergency lighting, fire alarms, security system, and evacuation alarm.

1.3.6 Reactor Coolant and Other Auxiliary Systems

The four water systems that provide for heat removal and shielding are the (1) primary coolant system, (2) secondary coolant system, (3) makeup system, and (4) bulk shielding tank. Chapters 4, 5, and 13 of this SER provide detailed descriptions of these systems.

During full-power operation, natural convection circulates water up through the core and cools the fuel elements. The primary cooling system removes water from the reactor pool and passes it through a heat exchanger which transfers the heat to the secondary cooling system. Primary coolant then returns to the reactor pool. A portion of the primary coolant is diverted through a cleanup loop with a filter and demineralizer for purification and then returned to the pool.

The secondary cooling system removes heat from the primary cooling systems and rejects it to the environment through a forced-draft cooling tower. The water returns from the cooling tower to an open surge tank. Flow in the secondary cooling system is adjusted to accommodate variations in primary coolant temperature and outside air temperature.

The bulk shield tank provides shielding for the thermalizing column. Irradiated fuel elements can also be stored in the shield tank.

A makeup water system compensates for evaporation of primary coolant (primarily from the top of the reactor pool). A distillation unit provides makeup water through a filter demineralizer unit to the pool or bulk shield tank.

1.3.7 Experimental Facilities and Capabilities

Chapter 10 of this SER describes five different experimental facilities—(1) central thimble, (2) rotary specimen rack, (3) pneumatic specimen tube, (4) thermal column, and (5) four beam tubes. These are standard facilities for TRIGA reactors and allow the positioning of samples into or near the reactor core to be irradiated for research purposes.

1.3.8 Radioactive Waste Management and Radiation Protection

The routine operation of the KSU TRIGA reactor results in the discharge of radioactive gases, some periodic discharge of slightly contaminated water to the sewerage system, and small quantities of solid waste. Chapter 11 of this SER discusses waste management and radiation protection procedures.

1.3.9 Safety Considerations of Normal Operations

Normal facility operations produce various radiation sources in gaseous, liquid, and solid forms. The gaseous sources are argon-41, nitrogen-16, and tritium (hydrogen-3). Liquid sources are not routinely produced but result from maintenance activities (such as resin changes) and condensation from the air-handling unit in the summer months. Liquids are typically released to the sanitary sewerage system after assay and filtration, with concentrations well below the 10 CFR Part 20 effluent concentration limits. Solid sources (mostly resulting from activation of primary coolant) are deposited in the mechanical filter and demineralizer resins and are treated as solid waste.

Argon-41, produced as a result of neutron activation of argon-40, has a half-life of 1.8 hours (hr). It is the major contributor to radiation exposure, both on and off site, resulting from normal operations. Calculations based on 1250 kW(t) continuous operation show that doses in the reactor bay are below the inhalation derived air concentration (DAC) limits specified in Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” to 10 CFR Part 20. In addition, under the worst assumptions about the dispersion of offsite radiation caused by atmospheric conditions, the offsite dose from argon-41 is less than 10 percent of the 10 millirem per year (mrem/yr) limit for effluents specified in 10 CFR 20.1101, “Radiation Protection Programs.” Under normal atmospheric conditions, a full-year exposure to equilibrium argon associated with 1250 kW(t) operation would lead to an offsite effective dose of 7 millirem (mrem), which is well within applicable limits.

Nitrogen-16 is primarily responsible for the radiation doses directly over the reactor pool during operation. Produced as a result of neutron activation of oxygen-16 in the reactor coolant, it has a half-life of 7.1 seconds (s). Since its concentration rapidly decreases, it does not contribute significantly to offsite doses. Presently, when the facility is operating at the license limit of 250 kW(t), the measured dose rate directly above the pool surface is 40–60 millirem per hour (mrem/hr). When the facility is operating at 1250 kW(t), the space above the pool may become a high-radiation area, depending on the location, requiring additional administrative controls for access. The licensee’s radiation protection program (see Section 11.1.2) ensure that the requirements of 10 CFR Part 20 are met. (The licensee has calculated nitrogen-16 dose rates above the pool to be 11 milliroentgens per hour (mR/hr) at the ceiling, 100 mR/hr at waist level, and 350 mR/hr at 1 ft above the pool surface.)

Tritium is a product of sequential neutron activation of hydrogen in the reactor coolant. Under the conservative assumption that the complete tritium inventory of the reactor pool is released into the reactor bay atmosphere, the tritium concentration will remain below the limit for an unrestricted area.

1.3.10 Consequences of Potential Accidents

Chapter 13 discusses in detail the accident analysis for the KSU reactor. Specifically, the three accidents defined are (1) a complete loss of coolant from the reactor pool, (2) an insertion of the maximum amount of positive reactivity available, and (3) a fuel element failure in air with maximum release of the fission product inventory (the maximum hypothetical accident (MHA)). Accident analysis focuses on two consequences—the release of radiation and/or an increase in fuel temperature.

As demonstrated in Chapter 13, a complete loss-of-coolant event following long-term operation at 1250 kW(t) will result in a maximum fuel temperature of less than 300 °C, well below any safety limit for TRIGA reactor fuel; therefore, no release of fission products will occur. An insertion of the maximum available amount of positive reactivity will result in fuel and cladding temperatures remaining well below the criteria required to ensure fuel integrity. The MHA, which is a complete loss of fuel cladding while the fuel element is in air, with an attendant 100-percent release of the fission product in the cladding-fuel gap, results in offsite doses that are far below the regulatory limits of 10 CFR Part 20.

1.4 Shared Facilities and Equipment

The KSU TRIGA reactor facility contains the reactor bay, the reactor control room, and all piping and experimental areas. Offices for reactor personnel and others associated with the reactor program are in the reactor building. The university provides the reactor building with electricity, water, heating and ventilation, and a sewerage system. Air from the reactor building is exhausted through a fan to the unrestricted environment (Chapter 3 of this SER discusses water and sewerage, and Chapter 11 addresses controls on discharge to the sewerage system).

1.5 Comparison with Similar Facilities

The KSU TRIGA reactor is similar to other TRIGA research reactors currently licensed to operate by the NRC. The instruments and controls are similar to the newer, nonpower TRIGA reactors licensed by the NRC. Extensive operating experience of TRIGA reactors throughout the world has demonstrated their inherent safety. This safety arises from the prompt negative fuel temperature coefficient that is characteristic of TRIGA fuel. Based on the facility's accident analysis, there are no requirements for forced cooling flow or emergency core cooling.

1.6 Summary of Operations

From 1981–2007, the reactor has operated for 400- 800 hrs per year, or about 8-16 hrs per week. In addition to its use for student education and training, a number of diverse entities from outside the university have used the reactor for research purposes.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(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 licensee enter 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 have entered into contracts with DOE, providing 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 such a contract with DOE, KSU has satisfied the requirements of the Nuclear Waste Policy Act of 1982 as they apply to the KSU TRIGA reactor.

2. SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

The KSU TRIGA reactor is located on the campus of KSU in the city of Manhattan, Kansas, which is located in Riley County. The campus is in the center of the city, surrounded by residential communities with some small businesses. Other prominent landmarks near the reactor site include the Fort Riley Military Reservation (12 kilometers (km) west), Manhattan Regional Airport (9 km southwest), Tuttle Creek Reservoir (7 km north), Kansas River (3 km southeast), the U.S. Army Marshall Field Airport (22 km southwest), and Interstate Highway 70 (13 km south). The area within 12 km (8 miles) of the reactor facility supports a population of about 62,300 (1990 census data), including about 20,000 students.

2.1.2 Population Distribution

The licensee provided the population distribution out to 12 km from the reactor site, based on the 1990 census. (The 2000 census, only partially analyzed, indicates a small decrease in population densities over the areas closest to the facility). The population distribution surrounding the KSU reactor is significantly less dense than the distribution surrounding other TRIGA reactors of comparable power level (e.g., the University of California–Davis.)

2.2 Nearby Industrial, Transportation, and Military Facilities

Light manufacturing and service industries are located in Manhattan, but there are no chemical plants, refineries, mining, or significant quarrying operations. There are no docks, ports, or railroad yards. Consequently, there is little risk of industrial or shipping accidents that could affect the safety of the reactor.

The Manhattan area is approximately 13 km north of Interstate Highway 70, leading from Topeka to the east and Salina to the west. The area is also located along the north bank of the Kansas River and the west bank of the Big Blue River. The reactor facility is located approximately 25 meters (m) in elevation above the rivers and has never been threatened by floods.

The Manhattan airport, approximately 9 km southwest of the facility, provides general aviation and feeder airline service, with about 16,000 flights annually. The Marshall Field Airport on the Fort Riley Military Reservation, 22 km from the reactor, is a base for rotary-wing Army aircraft.

2.3 Meteorology

2.3.1 General and Local Climate

Manhattan, Kansas, is located near the geographical center of the United States and in the middle of the temperate zone. The area is characterized by hot summers (100 °F or higher on more than 50 days), but mild winters (typically about 45 °F cooler than in summer). The average annual rainfall is about 32 in. (80 centimeters (cm)), 70 percent of which occurs from April through September. The construction of a levee around the city of Manhattan and the Tuttle Creek Reservoir has largely alleviated the threat of flooding from the Kansas and Big Blue Rivers. Floodwaters have never penetrated the reactor bay or the basement of the KSU reactor facility.

2.3.2 Site Meteorology

The licensee presented frequency distributions for wind speed, wind direction, and atmospheric stability (Pasquill categories A through G). These data may be used in conjunction with population data to evaluate potential radiation doses associated with hypothetical accidental releases of radiation into the atmosphere. An analysis of this scenario in Chapter 11 of this SER with regard to releases of argon-41 demonstrates that offsite doses over the full range of meteorological conditions will remain below the 10 CFR Part 20 limits.

2.3.3 Sources of Meteorological Data for Emergencies

The applicant conducted dose analyses in the SAR for effluent releases and accidents using conservative conditions from historical meteorological data. Local meteorological measurements for use in assessing actual accidental releases are not available; however, the licensee can obtain regional meteorological data from Internet sources such as the National Weather Service at the Manhattan Municipal Airport or by calling the airport directly. The meteorological data available will enable the licensee to use current conditions to predict the dispersion in the unlikely event of an accident-related release to the environment.

2.4 Geology, Seismology, and Geotechnical Engineering

To evaluate the extent to which a leak of soluble radioactive materials into the ground at the reactor site would contaminate subsurface water, the licensee analyzed the local geology. Test drills in the vicinity of the facility revealed a thin layer of topsoil with varying levels of glacial deposits overlying bedrock, limestone, and shale. Subsurface water exists 18 to 35 ft below existing grade in thin sand and gravel layers near the bedrock surface. The city of Manhattan, as well as the university, draws water from wells. The sand and gravel layers act as filters; however, it is possible for contamination to penetrate these wells. Operating, surveillance, monitoring, inspection, and auditing procedures in place at the reactor facility ensure that (1) regular inspection and monitoring of encapsulated sources occurs, and (2) unencapsulated, soluble radioactive materials are not held in inventory.

The Flint Hills surrounding the city of Manhattan are composed of limestone and shale. Above these are sand and gravel layers that act as filters.

2.4.1 Seismicity

Records reveal that 30 felt earthquakes with epicenters in Kansas have occurred since 1867, as illustrated in a map provided by the licensee in the SAR. Earthquakes of Modified Mercalli intensity VI or greater occur irregularly at intervals of 20 to 40 yrs. The most serious recorded earthquakes in Kansas were intensity VIII and intensity VII events occurring in 1867 and 1906, respectively. In accordance with Federal Emergency Management Agency guidelines, an intensity of VIII will result in slight damage to well-built buildings. No structural damage to well-built buildings will occur with intensity VII or less.

2.4.2 Vibratory Ground Motion

The licensee also provided a map in the SAR of the eight states surrounding Kansas that illustrates the estimated maximum ground accelerations in the region. The contour lines represent the acceleration as a percent of the acceleration caused by gravity, with a probability of exceeding a particular value of no more than 10 percent in 50 yrs (i.e., the contour line labeled “4” means that the probability of exceeding an acceleration greater than 4 percent of the acceleration as a result of gravity is no more than 10 percent in 50 yrs). These low probabilities reflect the history of low-intensity earthquakes.

2.4.3 Surface Faulting

No known faults exist within 8 km of the reactor site.

2.4.4 Liquefaction Potential

The phenomenon of soil liquefaction is associated primarily with medium- to fine-grained saturated soils. Saturated soils cannot support heavy loads, such as foundations for buildings, and earthquake shaking results in soil movement. Sandy, saturated soils are not expected at the facility site and so the potential for local liquefaction is minimal.

2.5 Hydrology

The reactor site is located at an elevation of approximately 1082 ft (330 m) above sea level. This is also approximately 80 ft (25 m) above the highest recorded flooding in the area. The average annual rainfall in the Manhattan area is about 32 in. (80 cm). The reactor facility is located on a region of the campus that is convex upwards (an inverted bowl), minimizing the probability of local flooding. Draining water gathers into storm sewers on the campus and discharges into the Kansas River.

2.6 Staff Evaluation

No hazardous industrial facilities or highway, airport, or rail transportation lines are located near the site. The nearest military facility, a rotary-wing facility, is 22 km away. The staff concludes that no industry, transportation, or military facilities pose significant risk to the continued safe operation of the facility.

This tectonically stable region is characterized by relatively low intensity as well as a relatively low frequency of earthquakes. Therefore, the staff concludes that the history of no significant earthquake damage in the site region supports the conclusion that the risk of seismic-induced damage to the KSU TRIGA is not significant. Furthermore, if the facility building were damaged, the radioactive fuel would be safely contained within the pool’s biological shield. Section 3.4 of this SER also discusses seismic-induced damage to the facility, and Chapter 13 discusses accidents that could be caused by a seismic event, such as loss of coolant.

Given the physical aspects of the area in which the facility is located, there are no unique demographic, geological, hydrological, climatological, or seismic conditions that would preclude continued safe operation of the reactor. From the date the reactor was originally licensed (1962) until the present, there have been no incidents associated with site characteristics that affected the reactor. The staff concludes that no significant changes have occurred that will make the site unsuitable for continued operation.

2.7 Conclusions

On the basis of information presented in the licensee’s SAR and summarized above, the staff concludes the following:

- The licensee has provided sufficient information to accurately describe the geology, hydrology, and demography surrounding the KSU TRIGA reactor. There is reasonable assurance that no geologic, hydrologic, or demographic features will render the site unsuitable for continued reactor operation.

- The licensee has discussed nearby manmade facilities and activities (i.e., industrial, transportation, and military) and none pose a significant hazard to reactor operations. There is reasonable assurance that operation of these facilities will not affect reactor operation.
- The applicant provided information on the geologic features and the potential seismic activity at the reactor site in sufficient detail and in a form that can be integrated acceptably into the design bases for structures, systems, and operating characteristics of the reactor. Therefore, the site remains suitable for the continued operation of the reactor.

3. DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The KSU reactor facility was originally constructed in 1961, with an addition completed in 1972. The reactor building was designed and built to meet or exceed contemporary building code requirements. The building is a three-level, rectangular structure housing the reactor. This facility provides space, shielding, and environmental control for radiography and irradiation services. Building areas include office space, laboratory space, shop facilities, utility service areas, and classrooms, many of which support the activities of the reactor facility.

3.1.1 General Conditions

The basic design goal of a TRIGA reactor is to preserve the integrity of the fuel by cladding, which acts as a physical containment system for fission products. Fuel design prevents the release of radioactive fission products during routine operation and credible accident conditions. Limiting conditions of operation (LCO) on the amount of fuel in the core and maximum power level prevent the fuel temperatures from exceeding the safety limit, thereby preserving the integrity of the fuel cladding. Section 3.5 and Chapter 4 of this SER discuss fuel design constraints.

The RCS maintains safe-shutdown conditions by ensuring that control rods can be inserted into the core within a prescribed time. Section 3.5 and Chapters 4 and 7 of this SER discuss RCS design constraints.

Facility design controls personnel exposure to radiation and the release of effluents that accompany normal operation or accident conditions. Section 3.5 of this chapter discusses these facility design constraints.

3.1.2 Architectural and Engineering Design Criteria

The reactor vendor was the General Atomics Division of General Dynamics Corporation. At the time of construction (1961) of the original building, the building code for the State of Kansas was the National Building Code. The 1972 addition to the TRIGA facility was constructed in accordance with the Uniform Building Code (UBC), which replaced the National Building Code. The licensee provided a listing of the various codes applicable to the present structure.

3.1.3 Structural Design of the 1961 Building

The structure of the original 1961 building is primarily poured-in-place concrete, except for the structural steel octagonal-shaped dome over the reactor. The concrete foundation in the bay has poured-in-place concrete walls set on a continuous footing. The building has 30-in. (76.2-cm)-diameter drilled piers resting on limestone bedrock. The licensee’s 1999 inspection for degradation of the 1961 building revealed no sign of structural movement or damage and found that the building is in excellent structural condition.

3.1.4 Structural Design of the 1972 Building

The structure of the 1972 building addition is poured-in-place concrete. The addition is also supported on piers drilled into the limestone bedrock. The 1999 inspection revealed no sign of structural movement or damage. Both the original 1961 building and the 1972 addition meet the structural requirements of buildings that were designed in 2000.

3.1.5 Sanitary Sewerage System

Ward Hall is served by a sanitary sewerage system that services the KSU campus, ultimately exiting at a water treatment facility in the city of Manhattan. The water treatment facility discharges water to the Kansas River.

3.1.6 Storm Sewer

There are no storm sewers on campus. Water flows along streets until it enters Campus Creek, an open creek flowing through the campus, then flows into the city of Manhattan storm sewer system and, ultimately, into the Kansas River.

3.2 Meteorological Damage

The current university architect has determined that "it can be reasonably assumed based on the KSU TRIGA reactor building's performance that the original design of the 1961 and 1972 structures were [was] more than adequate for their intended use." This assessment is based on the building's history in withstanding meteorological damage. In the 1990s, the building withstood snow in excess of 18 in. (45.7 cm); rains at the equivalent of one 1000-yr, two 500-yr, and many 100-yr rainfalls; wind gusts in excess of 110 miles per hour; nearby lightning strikes; and severe hail. These weather events caused no noticeable effect on the building structure or any of its infrastructure systems. Therefore, the NRC staff concludes that the design to mitigate significant wind damage to the facility is acceptable.

3.3 Water Damage

Ward Hall is located on some of the highest ground in the city of Manhattan and the KSU campus. In the floods of 1993, when many areas of Manhattan and much of the Midwest were flooded for weeks, no water entered the building from any point, at or below grade. The location of the building on high ground, along with the sloped grade around the building, allowed surface water to run off quickly. Therefore, the staff concludes that there is reasonable assurance that potential damage to the reactor by flood or ground water is small.

3.4 Seismic Damage

The structures associated with the reactor facility were designed in accordance with codes and standards applicable to the seismic zone designation at the time of construction. (Chapter 13 of this SER considers the failure of the reactor tank and loss of coolant in the event of a very large earthquake; the staff found the consequences of such an event to be acceptable from a public safety standpoint.) The Manhattan area is located in seismic zone 2, as defined in the UBC. The facility was designed and constructed in accordance with this code. Seismic activity in the region has registered as high as Modified Mercalli intensity VIII (in 1867), but only as high as intensity VI in the past 100 yrs. Because of the location of the facility in a low seismic risk zone, the construction codes used, and the design features of the facility that were implemented, as summarized in Sections 3.1.2–3.1.4 of this SER, the staff concludes that the risk of seismic damage to the facility is small.

3.5 Systems and Components

The KSU TRIGA reactor uses a number of diverse systems to reduce and control the potential for exposure to radioactivity as a result of reactor operation. These systems include the fuel and its cladding, the control rod scram system, shielding, and the confinement and ventilation systems.

A preventive maintenance program has been in operation for many years at the facility to conform and comply with the performance requirements of the TSs. The effectiveness of this preventive maintenance program is attested to by the small number and types of malfunctions of equipment over the years of operation. These malfunctions have generally been one of a kind (i.e., no repeats) and/or involved components that were fail safe or self annunciating. (See inspection reports and reports of reportable occurrences from the licensee, Docket No. 50–188.) Therefore, the staff concludes that there appears to be no significant uncompensated deterioration of equipment with time or with operation. Thus, there is reasonable assurance that continued operation for the requested period of renewal will not increase the risk to the public.

3.5.1 Fuel System

The KSU TRIGA reactor is designed to use stainless-steel clad TRIGA fuel elements, with 8.5 percent by weight uranium in a zirconium-hydride matrix (i.e., $ZrH_{1.65}$, which has a hydrogen/zirconium ratio of 1.65), with the uranium enriched to less than 20 percent in uranium-235. The safety limit for the KSU reactor is fuel element temperature, since this parameter affects the performance of the fuel cladding and hence the potential release of fission products. An additional, very important feature of TRIGA fuel is its inherent prompt negative temperature coefficient, which ensures that negative reactivity is quickly added to the reactor when fuel temperature increases, thus providing a natural mechanism for shutting down the reactor.

The major process that can potentially affect the integrity of the fuel cladding of fuel with a hydrogen-zirconium ratio of 1.65 is pressure associated with the release of hydrogen in the zirconium-hydride matrix, which could potentially rupture the cladding.

The release of hydrogen from the fuel matrix produces a pressure within the fuel element that may challenge the cladding. If the stress produced exceeds the ultimate strength of the clad material, a rupture of the clad is possible. The stress in type 304 stainless-steel cladding used in the reactor from hydrogen will equal the ultimate strength at a temperature of about 1150 °C for the $ZrH_{1.65}$ fuel matrix. The safety limit for steady-state fuel temperature is 750 °C, well below the temperatures that challenge the clad strength. The safety limit is protected by the TS-required high-power scram. In addition, there is a non-TS-required fuel temperature scram, set for less than 600 °C if located in the B, C, or D ring of the core. Table 4.1 of this SER shows calculated temperature data for 1250 kW(t) operation. At a bulk coolant temperature equal to the core outlet coolant temperature limit of 50 °C, the calculated centerline fuel temperature is 532 °C, well below the safety limit. The temperature near the fuel centerline (at a 0.69 cm radius), where thermocouples measure “fuel” temperature, is about 5 percent less than the centerline temperature. The licensee states that the actual fuel temperature scram setpoint will be nominally 450 °C if the instrumented element is in the B, C, or D ring of the core. However, as is discussed in chapter 4, the limiting safety system setting of 1250 kW(t) (TS 2.2) will prevent the fuel centerline temperature from reaching the safety limit of 1150 °C (TS 2.1).

3.5.2 Shielding

Design bases for TRIGA shielding derive from the General Atomics shielding design analysis for a 1 MW reactor, which is similar in construction and dimensions to the KSU reactor Design-basis radiation levels are less than 1 megarad per hour at the core boundary, less than 40 rad per hour at the tank boundary, and less than 1 mrem/hr outside the biological shield. Design requirements allow access through the shielding to experimental areas and permit extracting beams of radiation from the shielded volume into the reactor bay. The KSU TRIGA reactor meets the design-basis radiation levels.

3.5.3 Control Rod Scram System

The KSU reactor will be operated at 1250 kW(t) with three standard control rods (it presently has two) and one transient (pulsing) control rod. The neutron-absorbing material is either boron carbide or borated graphite. The rods are nominally 20 in. (50.8 cm) long and clad with aluminum. During operation, the standard rods are held in place with electromagnets that deenergize in response to a scram signal, allowing the rods to drop into the core. Air pressure holds the transient rod in place, with the air vented in response to a scram signal. TS 3.4.3(2) requires that the standard rods insert to 90 percent of full insertion from the full-out position in less than 1 second. The reactor operator manually controls all rods, individually inserting or withdrawing them. Chapters 4 and 7 of this SER discuss the control rod system in detail.

3.5.4 Confinement and Ventilation Systems

The confinement and ventilation systems are designed to control the level of airborne radiation in the reactor bay and to discharge facility air at the top of the confinement structure. The discharge of air during operation maintains a slight negative pressure in the reactor bay and controls argon-41 concentrations within the bay and at the site boundary to within all applicable limits. TS 3.5.3(1) and TS 5.3.3 specify that the confinement and ventilation system shall maintain in-leakage to the reactor bay to control the release of radioactivity to the environment.

A dome-shaped structure of approximately 144,000 ft^3 (4,075 m^3) (TS 5.3.3(2)) free volume surrounds the reactor bay.

3.6 <u>Staff Evaluation and Conclusions</u>

On the basis of the above considerations, the staff concludes that the KSU reactor facility is designed and built to withstand all credible and probable wind, water, and seismic damage contingencies associated with the site.

More than 40 yrs of operation have served to verify the design and performance of the safety systems. The staff confirmed this by reviewing the past operating performance of the facility through the licensee’s annual reports and inspection reports from periodic staff inspections. The staff generically evaluated the reactor fuel design in NUREG-1282, “Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors,” issued in August 1987, and found the design to be acceptable. 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 KSU TRIGA reactor is a water-moderated, water-cooled thermal reactor operated in an open pool with natural convection cooling. The reactor is currently licensed to operate in the steady-state mode at thermal power levels up to and including 250 kW(t). The licensee has requested a renewal of the operating license concurrent with a request to increase the steady-state power limit to 1250 kW(t). The licensee and the staff used the proposed licensed maximum power (1250 kW(t)) to evaluate the thermal-hydraulic aspects of operation. In addition, the licensee proposed a pulse mode of operation with an increase in the maximum reactivity addition from $2.00 to $3.00.

The reactor core is immersed in a reinforced concrete, water-filled, open pool. The core consists of heterogeneous fuel elements with enriched uranium in a zirconium-hydride matrix and a stainless-steel cladding. The pool is spanned by a fixed structure that supports the control rod systems, reactor instrumentation, and some experimental facilities. The core itself is located near the bottom of the pool supported on a structure that rests on the pool floor. The core is in the form of a right circular cylinder with a diameter of 18 in. (45.7 cm) and a height of 15 in. (38.1 cm), positioned with its axis vertical near the base of the cylindrical water tank.

Reactor control is achieved by inserting or withdrawing four neutron-absorbing control rods suspended from drive mechanisms. Heat generated by fission transfers from the fuel to pool water. The primary cooling system circulates the pool water through a heat exchanger in which the heat is transferred to the secondary cooling system and released to the environment by the cooling tower.

4.2 Reactor Core

The original design goal of the General Atomics TRIGA reactor was to provide a completely and inherently safe reactor. The KSU reactor meets this goal by virtue of its use of fuel consisting of enriched uranium in a zirconium-hydride matrix. The main safety-related characteristic of this fuel is the prompt negative temperature coefficient of reactivity, which acts instantaneously to reduce reactor power and thus fuel and clad temperatures. Power-level limits are then based on temperature. The analyses shows that operation at power levels exceeding 1,250 kW(t) (with the proposed core loading, 120ºF inlet water temperature, and with natural convection flow) will not allow film boiling, with its attendant high fuel and clad temperatures. This supports the requested increase in the license limit to 1,250 kW(t).

4.2.1 Reactor Fuel

General Atomics, the fuel vendor, has developed several TRIGA fuel types that vary in uranium loading. The core of the KSU reactor utilizes stainless-steel clad fuel, known as Mark III fuel elements (TS 5.1.3(1)). This type of fuel elements have been successfully used at various TRIGA reactors for many years. The staff generically approved the fuel type in NUREG-1282.

The reactor fuel is a solid, homogeneous mixture of a uranium-zirconium hydride alloy. The hydrogen-to-zirconium atomic ratio within the fuel is 1.65. The hydrogen in the alloy is a neutron moderator. The moderator is dispersed in the fuel matrix, which results in the moderator having the same operating temperature as the fuel. This design feature of the fuel contributes to the ability to safely pulse the reactor, since the prompt negative temperature coefficient will immediately insert negative reactivity as the temperature of the fuel rises.

Each element is clad with a stainless-steel tube with stainless-steel end fixtures. Two sections of graphite are inserted in the tube, one above and one below the fuel, to serve as top and bottom neutron reflectors for the core. The end fixtures are welded to both ends of the tube. TS 3.7.3 specifies standard reactor fuel element physical dimension limits, such as transverse bend and elongation, which are limits typical for this type of fuel. TS 4.7 specifies the frequency of surveillance of fuel integrity. The latter limit provides an indicator of fuel growth and the former prevents contact between adjacent elements. Vertical alignment of each fuel element is provided by top and bottom grid plates. These surveillance limits have been used for many reactor years of operation at the KSU reactor and other TRIGA reactors in the United States and abroad and no cladding ruptures have occurred. The staff concludes that the dimensional limits and surveillance frequency on the fuel are acceptable.

An instrumented element has three chromel-alumel thermocouples embedded in the fuel. This element is placed in the analyzed peak power location in the core to monitor fuel temperature, which is the variable upon which the safety limit is placed. The tip of each thermocouple is located about 0.27 in. (0.69 cm) radially from the axial centerline. One thermocouple is located at the vertical center plane, with the other two located 1 in. above and 1 in. below this plane. The asymmetric location of the thermocouples means that typically the measured temperature is dependent on the orientation of the thermocouple with respect to the core center (as the element is rotated about its axis). Regardless of the radial orientation of the thermocouple, the temperature measured is no less than approximately 5 percent lower than the centerline temperature; therefore, it is a good indication of the peak fuel temperature in that element if the power level is constant. In all other respects, the instrumented element is identical to the standard element.

Graphite dummy elements may be used to fill grid positions in the core. The dummy elements are of the same dimensions as the Mark III fuel elements but are clad with aluminum and have a graphite length of 22 in. (55.9 cm).

The layout for the proposed 1250 kW(t) core will be similar, with additional fuel elements and one additional control rod (control rod positions will be adjusted). The additional fuel elements are required to compensate for the larger power defect from the higher operating temperatures at the higher maximum steady-state power level. The additional control rod is necessary to meet higher reactivity control requirements, such as the minimum shutdown margin in TS 3.3.2, associated with the additional fuel.

TS 6.11(d) requires the following:

> a report within 60 days after criticality of the reactor in writing to the Director, Division of Policy and Rulemaking, US Nuclear Regulatory Commission, Washington, D.C., 20555, resulting from a receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level or the installation of a new core, describing the MEASURED VALUE of the OPERATING conditions or characteristics of the reactor under the new conditions

This TS will ensure the staff that the reactor is operating as analyzed in the SAR and that there is reasonable assurance that the health and safety of the public will not be compromised by the power increase to 1250 kW(t).

Chapter 11 of this SER discusses the consequences of the additional radioactive effluent and other waste products generated from operation at 1250 kW(t). Chapter 13 of this SER discusses the consequences of accident scenarios resulting from operation at 1250 kW(t).

4.2.2 Control Rods

The reactivity of the TRIGA reactor is managed by three control rods, namely a shim rod, a regulating rod, and a transient (or pulse) rod. (The 1250 kW(t) proposed core will contain an additional shim rod.) The rod drives are mounted on a bridge at the top of the reactor tank and are connected to the rods through a connecting rod assembly.

The shim and regulating rods contain a neutron-absorbing boron compound and are clad with aluminum. The rod drive mechanism is an electric motor, allowing the rod's position in the core to be adjusted. The rods are connected to their drives through an electromagnet and armature. Interruption of current to the armature deenergizes the electromagnet, allowing the rods to drop via gravity down into the core.

The transient rod differs from the shim and regulating rods in its diameter and drive mechanism. The transient rod is pneumatically positioned in the core with the use of compressed air. Removal of the compressed air allows the rod to drop into the core. In addition to operating in the steady-state mode, the transient rod can also be used in the pulse mode, since it can be rapidly removed from the core, again using compressed air. Chapter 7 of this SER provides detailed descriptions of the control rod system, control rods, and drives. The rod control system provides for interlocks, which prevent the rods from moving under various circumstances, and reactor scrams, which result in the rapid insertion of control rods. Chapter 7 of this SER also describes interlocks and scrams.

4.2.3 Neutron Moderator and Reflector

The hydrogen in the zirconium-hydride portion of the fuel serves as a neutron moderator. In addition, the light water in the reactor pool provides neutron moderation, while also serving as a heat-removal medium and radiation shield. The location of a neutron moderator within the reactor fuel allows the neutrons to respond rapidly to temperature changes in the fuel, thus providing a nearly instantaneous temperature feedback effect that opposes changes in neutron population and thereby acts to limit rapid power increases.

The neutron reflector provides a means of limiting the number of neutrons that might escape from the core, thus enhancing neutron economy. The neutron reflector is composed of two parts. Above and below each 15-in. (38.1-cm) section of fuel, there is a 3.44-in. (8.74-cm) section of graphite. These sections of graphite reflect neutrons that would escape from the top and bottom of the core back into the core. The escape of neutrons from the sides of the core is mitigated by the radial reflector, a ring-shaped, aluminum-clad block of graphite surrounding the side of the core. The radial reflector has an inside diameter of 18.7 in. (45.7 cm), an outside diameter of 42 in. (106.6 cm), and a height of 22.0 in. (55.9 cm). The reflector assembly rests on an aluminum platform at the bottom of the reactor tank and supports the core grid plates.

4.2.4 Neutron Startup Source

A 2-curie americium-beryllium source is used for reactor startup. The source is encapsulated in stainless steel and housed in an aluminum source holder of the same configuration as a fuel element. Consequently, the source may be positioned in any one of the fuel positions bounded by the upper and lower grid plates.

4.2.5 Core Support Structure

The fuel elements are spaced and supported by two aluminum grid plates. The top grid plate provides lateral support for the fuel elements, while the bottom grid plate supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements. Space is provided for the passage of cooling water around the sides of the bottom grid plate, through the core, and out through the top grid plate. The reflector assembly supports both grid plates.

4.3 Reactor Tank

The reactor core is located in a cylindrical aluminum tank surrounded by a reinforced concrete structure. The reactor tank is a welded aluminum vessel with a diameter of approximately 6.5 ft (1.98 m) and a depth of approximately 20.5 ft (6.25 m). The tank is welded for water tightness. The outside wall of the tank is coated with a bituminous material for corrosion protection. Each experiment penetration through the tank wall has a water collection plenum at the penetration that allows leakage to be identified and measured.

Four beam tubes extend from the tank wall to the outside of the biological shield in the outward direction. Tubes welded to the inside wall extend toward the reactor core. Three of the tubes end at the radial reflector, while the fourth penetrates the reflector, extending to the outside of the core. Two penetrations in the tank allow neutron extraction into a thermal column and a thermalizing column, as described in Chapter 10 of this SER.

A bridge is mounted over the core area and spans the tank. The bridge supports the control rod drives, instrumentation, and other equipment. Access to the pool can be limited by means of an aluminum grating.

4.4 Biological Shield

The reactor tank is surrounded on all sides by a reinforced concrete biological shield, varying in thickness from 8 ft (2.44 m) at core level to 3 ft (0.91 m) at the top of the tank. This structure provides additional radiation shielding for personnel working in the reactor laboratory and protects the reactor core from potentially damaging natural phenomena. The shielding configuration is similar to other facilities operating at power levels up to 1 MW.

4.5 Nuclear Design

The principal method for controlling maximum power, and hence fuel temperature, is through the negative temperature coefficient resulting from the design of TRIGA fuel. Fuel and clad temperatures define the safety limit, which, for the KSU TRIGA reactor, consists of specifying the maximum permissible temperature of the fuel, in accordance with TS 2.1.3, which states the following:

- Stainless-steel clad, high-hydride fuel element temperature shall not exceed 1150 °C (i.e., the accident limit).
- Steady-state fuel temperature shall not exceed 750 °C.

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 4.5.3 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 1250 kW(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.

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. Operating with a maximum long-term, steady-state fuel temperature of 750 °C ensures that the fuel does not have significant time- and temperature-dependent growth.

4.5.1 Design Criteria—Reference Core

The licensee has used accepted heat transfer models and parameters to analyze fuel and clad temperatures for the proposed 1250 kW(t) core. The grid plates have a total of 91 spaces, which can be used for the fuel elements, dummy elements, control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids as required to contain experiments or to limit excess reactivity. The bottom grid plate supports the weight of the fuel elements.

4.5.2 Reactor Core Physics Parameters

The configuration of the proposed core following the upgrade to 1250 kW(t) differs from the present core (250 kW(t)) only in the addition of a fourth control rod, which will replace a graphite dummy element or a void experimental position. Neutron fluxes and radiation dose rates (discussed in Chapter 11 of this SER) will scale linearly with power, but the upgrade will not affect the basic physics parameters (delayed neutron fraction, neutron lifetime, and reactivity coefficients).

4.5.3 Fuel and Clad Temperatures

The thermodynamic quantity of importance is the fuel centerline temperature. The centerline temperature (the highest temperature of the fuel) must not exceed the temperature limits specified by the SLs (see Section 4.5 above). This ensures that clad integrity is maintained. The licensee has used accepted engineering models and thermodynamic parameters to calculate the fuel centerline temperature for the hottest fuel location in the core. Table 4.1 shows the results of temperature calculations for 1250 kW(t) operation. The temperatures are calculated for the fuel element that produces the maximum heat (i.e., the hottest location in the core). The calculations reflect the use of the reference core. Using a core loading with more elements than the reference core would distribute heat production over more elements, hence fuel temperatures and heat flux would be lower and, the results would be conservative.

Table 4.1 Calculated Temperature Data for 1250 kW(t) Operation				
Bulk Water (°C)	Clad/Water Interface (°C)	Gap/Clad Interface (°C)	Fuel/Gap Interface (°C)	Fuel Centerline, (°C)
20.0	21.2	37.7	299.0	503.2
100.0	100.0	116.4	307.8	582.0

For a wide range of bulk water temperature, the hottest fuel centerline temperature is well below the steady-state limit of 750 °C. It should also be noted that TS 3.8.3(1) is an LCO on bulk water temperature (pool exit temperature) of 130 °F (48.9 °C) (by interpolation that would result in a clad/water interface and fuel centerline temperatures of approximately 51 °C and 533 °C, respectively).

The licensee has also calculated the fuel centerline temperature of the hottest fuel element over a range of reactor power. For a water temperature of 27 °C, the centerline fuel temperature remains below 750 °C at power levels up to 1860 kW(t), well above the proposed power limit of 1250 kW(t). For a water temperature of 100 °C, the centerline fuel temperature remains below 750 °C at power levels up to 1540 kW(t).

Keeping centerline temperatures below the safety limit for steady-state operation is a necessary condition for safe operation of the reactor, but is not sufficient on its own. The other important parameter 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 calculated the CHFR over a range of core inlet temperatures for 1250 kW(t) operation. The data clearly show that a very wide margin exists between the operating heat flux and the CHF, even up to unrealistically high core inlet temperatures. Hence, departure from nucleate boiling is not a consideration for the steady-state operation at 1250 kW(t) (the proposed license limit) and the specified coolant temperature of 130 °F at pool exit (TS 3.8.3(1)).

The basic parameter that allows the TRIGA reactor system to operate safely with large step insertions of reactivity is the prompt negative temperature coefficient associated with the TRIGA fuel and core design. This negative temperature coefficient allows operational flexibility in steady-state operation, as the effect of accidental reactivity changes occurring from experimental devices or other incidences is greatly reduced. This coefficient primarily arises from a change in the fuel utilization factor resulting from the heating of the uranium-zirconium hydride fuel-moderator elements (i.e., fewer neutrons are available to cause fission). The coefficient is prompt because the fuel is intermixed with the zirconium-hydride, the predominant moderator; thus, as the fuel temperature increases the moderator temperature rises simultaneously. The heating of the moderator mixed with the heating of the fuel reduces the absorption of neutrons that causes fission relative to the absorption in the water surrounding the fuel, where the temperature did not rise as much. This results in a loss of reactivity.

The present licensed core (250 kW(t)) allows maximum pulse reactivity insertion of $2.00. The limitation of the pulse reactivity to a maximum of $2.00 was based on the allowed use of aluminum clad fuel. The TSs approved with this renewed license only allow stainless-steel clad fuel (TS 5.13(1)). The proposed core (1250 kW(t)) has maximum license pulse reactivity insertion of $3.00. Chapter 13 of this SER provides a detailed analysis of power excursions. The parameter of interest is peak fuel temperature resulting from a pulse. A $3.00 pulse will result in a peak fuel temperature of 746 °C, which is well below the accident safety limit of 1150 °C for stainless-steel clad fuel.

The CHFR following a $3.00 pulse from zero power with a 27 °C core inlet temperature remains above unity, thus presenting no threat to fuel-cladding integrity.

The staff notes that time in “core life” considerations are not relevant to these calculations. Unlike a power reactor and some research reactors, the KSU reactor does not operate over a relatively short period of time, shut down, and then refuel. Therefore, there is no definitive “core life.” When core excess reactivity gets too low to allow unrestricted operation within the TS and license limits, additional fuel elements typically are added or fresh elements used to replace burnup elements as necessary. These small core changes do not affect core parameters significantly and thereby do not invalidate these analyses which were done assuming a new core. In addition, with the flexibility to use instrumented elements to measure temperatures in various positions in the core, the fuel temperature (TS 3.3.3(1)), the parameter associated with the safety limit (TS 2.1), can be monitored.

4.6 Thermal-Hydraulic Design and Analysis

The applicant performed the thermal-hydraulic analysis for operation of the KSU reactor using standard heat transfer models and parameters (e.g., thermal conductivities, friction losses). The objective of the thermal-hydraulic analysis is to compute flow rates through the core and temperature rise across the core as a function of reactor power. Table 4.2 below shows the results of the analysis.

Table 4.2 Coolant Flow Rate and Core Temperature Rise for Natural Convection Cooling during Steady-State Operation

Power (kW(t))	Mass Flow Rate (kg/s)	Core ΔT (°C)
50	0.047	3.1
100	0.061	4.7
200	0.077	7.5
300	0.090	9.6
400	0.100	11.5
500	0.108	13.3
750	0.125	17.2
1000	0.139	20.6
1250	0.150	23.8

For normal operation with natural convection cooling, the increases in core temperature (and, therefore, core inlet temperatures increases) are well below those that lead to challenges to the CHFR and fuel centerline temperatures. With a coolant core exit limit of 50 °C (130 °F) and a core ΔT of 24 °C, the core inlet temperature cannot be greater than 26 °C. The licensee's calculations show that the CHFR will be greater than 6 at an inlet temperature of 26 °C and a coolant depth of 13 ft above the core. Therefore, film boiling will not occur and the safety limit will not be exceeded during operation at or below the license power limit of 1250 kW(t) thermal and the coolant core exit limit of 50 °C (130 °F) (even with the core inlet equal to core exit limit of 50 °C (130 °F), the CHFR is greater than 4). This conclusion is reinforced by the experience of other similar TRIGA reactors operating safely at higher power levels using natural circulation cooling.

4.7 Safety Limit

Both steady-state and pulse mode operation are limited by fuel temperature, which is the safety limit described in the TSs. The fuel temperature limits arise from consideration of the stress produced on the fuel cladding by hydrogen outgassing. The stress, which is temperature dependent, must be less than the yield strength of the clad material. The licensee has presented calculations that support the appropriateness of the fuel temperature safety limits.

4.8 Operating Limits

The primary safety consideration is to maintain fuel temperatures below values that might result in fuel damage. In practice, this is achieved by setting limits on other parameters that affect fuel temperature. As previously discussed, these include steady-state power level, fuel temperature during pulse operations, and maximum-step reactivity insertions.

Three other limits affect the safe operation of the reactor—shutdown margin, excess reactivity, and reactivity insertions from experiments. The limit on the minimum shutdown margin ensures that the reactor can be safely shut down from any operational configuration, even if the highest worth scram control rod remains stuck out of the core. The minimum shutdown margin (TS 3.1.3(2)) of $0.50 will ensure that the reactor can be shut down and remain shut down. This minimum shutdown margin must be met with the reactor in any core condition, with the most reactive control rod assumed to be fully withdrawn and the absolute value of all experiments in their most reactive condition. The NRC staff accepts the value of $0.50 as a standard, measurable value for shutdown margin.

The total excess reactivity that may be loaded into the reactor during operation is $4.00 (TS 3.1.3(1)). This amount provides for the various negative reactivity effects associated with operation and use of the reactor and allows for some operational flexibility. Imposing a limit on excess reactivity helps to ensure that the SAR assumptions and analyses are applicable to all operational cores.

TSs 3.6.3(1–2) specify the reactivity limits of experiments. Their purpose is to limit reactivity excursions to less than (or equal to) those analyzed in the accident analysis. TS 3.6.3(1) limits the reactivity worth of experiments to less than $2.00 per experiment. This limit (the sum of the absolute values of reactivity of the individual experiments in the core) is less than the positive reactivity insertion limit of the pulses (see Chapter 13 of this SER) that would be needed to reach the fuel temperature safety limit.

4.8.1 Operating Parameters

The main safety considerations are to maintain fuel temperature below the value and prevent film boiling that would result in fuel damage. The limiting safety system setting limits fuel temperature by controlling maximum power (TS 2.2.3(1)), since there is a direct relationship between reactor power and fuel temperature. TSs 3.8.3(1) and (3) are limits on coolant pool exit temperature and coolant depth over the core that prevents film boiling. Three other parameters of concern include (1) minimum shutdown margin, (2) maximum excess reactivity, and (3) reactivity limits on experiments. The TSs associated with each have been discussed above.

4.8.2 Limiting Safety System Settings

Limiting safety system settings are required to ensure that automatic protective action (reactor shutdown) will occur in sufficient time to prevent safety limits from being exceeded. The TSs (TS 2.2.3(1)) require that the power level does not exceed 1250 kW(t) in the steady-state mode of operation (also the license limit). If operating at 1250 kW(t), the power defect reduces the core excess reactivity, thereby precluding any transient sufficient to exceed the safety limit. This ensures that, in the event of a power excursion, fuel temperatures remain below the accident safety limit of 1150 °C (TS 2.1.3(1)).

4.8.3 Safety Margins

For steady-state operation at 1250 kW(t), the coolant temperature rise through the core is 23.8 °C (Table 4.2 above). Operation at a pool outlet temperature greater than 49.8 °C is limited by TS 3.8.3(1); hence, the maximum core inlet temperature is 26 °C. As discussed in Section 4.6 above, the CHFR is greater than 6 for the operation limits, or, in other words, a factor of 6 margin to departure from nucleate boiling. Therefore, for steady-state operation, a large margin of safety exists for clad temperatures that could cause fuel failure. Also, during steady-state operation at 1250 kW(t) with the bulk coolant temperature at the core outlet temperature limit the maximum fuel centerline temperature is calculated to be 533 °C and the clad temperature is 52 °C (interpolated from Table 4.1). Both of these values are well below the accident fuel temperature safety limit of 1150 °C and clad temperature that would support film boiling. A reactor scram that requires less than 1 second (TS 4.4.2) and occurs during a transient at 1250 kW(t) will shut down the reactor before the safety limit is reached.

Pulse operation with a maximum reactivity insertion of $3.00 will result in a calculated peak fuel temperature of 746 °C. This is 35 percent below the accident safety limit of 1150 °C for stainless-steel clad fuel.

4.9 Staff Evaluation

The fuel design is passively safe because of the large prompt negative temperature coefficient of reactivity. That fuel design feature, along with the core excess limit ($4.00) and the pulse reactivity license limit ($3.00), prevents violation of the safety limit. With the natural circulation cooling, high SLs on the fuel (1150 °C and 750 °C), the limiting safety system setting limits (1250 kW(t)), and other operational TSs, such as the limit on coolant core exit temperature (130 °F) and the scram limit (TS 4.4.2 control rod drop time of 1 second), film boiling is prevented. Margins of safety with regard to fuel temperatures and CHFR are well above minimum requirements. This, along with other considerations of the thermal-hydraulic analyses, the reactor fuel, and the support systems, suggests that there is reasonable assurance that the KSU reactor can operate safely within the license limits and the TSs.

4.10 Conclusions

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

- 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. These TRIGA components were built by General Atomics to the company's design and quality assurance standards. The KSU reactor has been operating for more than 40 yrs as have many other TRIGA reactors of similar design and construction, thus representing hundreds of reactor years of operation. In addition, the KSU reactor includes redundant safety-related systems. The design features of this reactor are similar to those typical of the TRIGA reactors licensed by the NRC at comparable power levels. On the basis of its review of the KSU reactor and its experience with these other facilities, the staff concludes that there is reasonable assurance that the KSU reactor can be operated safely, as limited by the TSs, for the period of the requested renewed license.
- The information provided in the KSU SAR includes thermal-hydraulic analyses for the reactor. These analyses give reasonable assurance that the reactor can be operated at its proposed licensed power level without undue risk to public health and safety.
- The licensee has proposed limits on pulsing the reactor. 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.

- The licensee has discussed and proposed minimum shutdown margin and excess reactivity limits that are acceptable to the staff. The minimum shutdown margin ensures that the reactor can be shut down from any operating condition with the highest worth control rod stuck out of the core. The limit on excess reactivity allows operational flexibility while limiting the reactivity available for reactivity addition accidents.
- The licensee has proposed reactivity limits on experiments (TSs 3.6.3(1–2)). These limits apply to the value of all experiments, moveable experiments, and secured experiments. The licensee has proposed values that are bounded by the pulse reactivity addition analysis. Therefore, failure of experiments will not add unacceptable amounts of reactivity to the reactor.
- The fuel and core design provides reasonable assurance that the KSU TRIGA research reactor can be operated safely at power levels up to 1250 kW(t) and with reactivity additions in the pulse mode of up to $3.00, as limited by the proposed TS requirements.

5. REACTOR COOLANT SYSTEMS

5.1 Summary Description

During full-power operation, natural convection of primary tank water cools the fuel elements. Heat removal from the tank water is accomplished by means of a heat exchanger, wherein primary heat is transferred to a secondary cooling system and then to the environment through a cooling tower. A portion of the primary coolant is directed to a cleanup loop to maintain cleanliness of the primary water, minimizing the circulation of radioactive nuclides and the possibility of corrosion. A makeup water system compensates for losses in primary coolant as a result of evaporation from the open pool.

The cooling systems serve the following five major functions:

(1) remove and dissipate heat generated in the reactor

(2) provide radiation shielding from the core area

(3) control primary water conductivity

(4) control primary water radioactivity

(5) maintain optical clarity of the primary water

The reactor core is cooled by the natural circulation of water in the reactor tank, where the external primary and secondary cooling systems maintain the water temperature at an approximate average of 43.1 °C (the average of maximum core inlet and core exit temperatures).

The applicant will upgrade the primary cooling system to allow continual removal of at least 1250 kW(t) of heat from the reactor tank water. Until the capacity of the cooling system is upgraded, the reactor can be operated safely for periods of time such that the coolant core exit temperature limit is not exceeded. The reactor will contain the necessary equipment and controls to circulate up to 110 gallons per minute (gal/min) (6.9 liters per second (L/s)) of tank water and to limit the temperature of the water to less than 48.9 °C (130 °F) (TS 3.8.3(1)). Instrumentation is provided to monitor system operation, water temperatures, pressure, flow, radioactivity, and conductivity. Tank bulk water outlet and inlet temperatures are continuously recorded. TS 3.8.3(3) requires that the water level be maintained at least 13 ft (4 m) above the reactor core to provide adequate coolant volume, adequate coolant saturation temperature, and the necessary shielding of radiation from the core and any nitrogen-16 produced. The primary and secondary cooling systems are operated and monitored from the reactor control room, with their remote controls and monitoring instrumentation in the reactor room.

5.2 Primary Cooling System

The complete cooling system is designed to transfer heat from the reactor core to the secondary cooling system and to provide radiation shielding directly above the reactor core. The primary portion of the cooling system comprises the reactor tank, a centrifugal pump, one side of a heat exchanger, and a cleanup system. The primary coolant is deionized water. There are two inlets to the primary cooling system, one just beneath the surface of the tank water and another through a skimmer that collects foreign particles on the pool surface. The two inlets provide a suction point for the primary (centrifugal) pump, resulting in a primary flow of approximately 110 gal/min (6.9 L/s).

The output of the primary pump is split into two flowpaths. An approximate flow of 10 gal/min (0.6 L/s) is directed toward the cleanup loop, and the remaining flow is directed to the heat exchanger. The two flowpaths recombine at the outlet of the heat exchanger and return to the reactor tank through a nitrogen-16 diffuser (see Section 5.6 of this SER). Table 5.1 lists the control room instrumentation for the primary coolant system.

Table 5.1 Control Room Instrumentation

Measurement	**Location**	**Device/Output**
Coolant Conductivity	Cleanup Loop Inlet Cleanup Loop Outlet	Platinum Probe/Meter Platinum Probe/Meter
Coolant Temperature	Water Box Heat Exchanger Inlet Heat Exchanger Outlet Reactor Tank (2)	RTD/Meter Transducer/Computer Transducer/Computer 2 Transducers/Computer
Flow Rate	Orifice	2 Transducers/Computer
Radioactivity	Water Box (Cleanup) Pool Surface	Geiger-Mueller Detector/Meter Geiger-Mueller Detector/Meter
Secondary Temperature	Heat Exchanger Inlet Heat Exchanger Outlet	Transducer/Computer Transducer/Computer

In addition to its cooling function, the reactor tank normally provides 16 ft (4.9 m) of shielding directly above the reactor core. (TS 3.8.3(3) requires a minimum of 13 ft for shielding as well as increased saturation temperature.) The pool water level is normally maintained within a few inches from the top of the reactor tank. Loss of primary coolant, either as a result of a pipe break or maintenance operations, is mitigated by valves that allow the isolation of the tank or heat exchanger. In addition, a siphon break is located about 1 ft (30 cm) below the water surface of the tank. The beam ports, when closed, are sealed on the outside by a gasket. Piping connects the beam ports to a manifold containing a pressure gauge, where any rupture would be indicated as a pressure increase.

The system can sense a major loss of coolant in three different ways, illuminating lights on the control panel. Two level sensors located in the reactor building sump indicate a high level in the sump, into which all floor drains in the reactor bay flow. A third level sensor is located at the top of the reactor tank, indicating that the tank level has dropped a few inches below normal level. Increased radiation levels on the remote area monitors will also indicate loss of coolant.

5.3 <u>Secondary Cooling System</u>

The secondary portion of the cooling system consists of a centrifugal pump, surge tank, one side of a heat exchanger, a pneumatic three-way valve, a cooling tower (located outside the reactor building), and a cooling tower fan. The heat exchanger is a plate type, consisting of sandwiched stainless-steel plates alternately carrying primary and secondary cooling water. The secondary cooling system is presently designed for continuous operation at approximately 725 kW(t). Additional plates must be added in order to continuously remove the 1250 kW(t) license power. Until that modification is complete, the licensee may operate up to the license power limit as long as TS 3.8.3(1) (130 °F maximum coolant pool outlet temperature) is met, since no TS requires the operation of the coolant system. (The reactor is operated at least annually to conduct the TS-required calorimetric calibration of the power instrumentation.) The licensee will install additional plates to support the continuous operation at the license power limit of 1250 kW(t) thermal.

5.3.1 Secondary Cooling System Flows

The secondary cooling pump draws water from the surge tank and passes it through the heat exchanger, removing heat from the primary loop. After exiting the heat exchanger, the water, which is controlled by a three-way valve, can take one of two alternate paths. One path bypasses the cooling tower and surge tank, leading directly back to the secondary pump. The second (normal) path flows through the cooling tower and surge tank and back to the pump.

5.3.2 Secondary Cooling Automatic Control System

As previously discussed, secondary system flowpath selection is controlled by a three-way valve, which in turn is controlled by system temperatures. If the outside air temperature is less than negative 23.3 °C (-10 °F), the three-way valve stops cooling tower flow and directs the secondary coolant to the pump. However, if the primary water temperature exceeds 43.3 °C (110 °F), the three-way valve directs flow through the cooling tower regardless of the outside temperature. The operation of the cooling tower fan is determined by the temperature of the secondary water returning from the cooling tower. At 21.1 °C (70 °F), the cooling tower fan starts at low speed. At 32.2 °C (90 °F), the fan switches to high speed. Manual control of fan speed is also available.

5.3.3 Secondary Water Quality

The licensee typically analyzes the secondary water chemistry twice a month. Items tested are pH, chlorine, conductivity, and total alkalinity. To detect possible leaks in the heat exchanger, facility personnel test secondary water monthly for radioactivity, specifically for tritium contamination of the secondary water.

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

TS 3.8.3(2) specifies that primary water conductivity be less than 5 mmho/cm to minimize corrosion of reactor components and production of radioactive materials, as well as to maintain clarity of the pool water. The cleanup system functions to maintain coolant conductivity, radioactivity, and optical clarity within acceptable ranges. The reactor tank, primary coolant system, fuel cladding, tank, and structural components are constructed of aluminum and stainless steel. The use of these materials and the maintenance of highly purified water minimize degradation caused by corrosion and coolant contamination levels.

The cleanup system consists of a water box, fiber cartridge filter, mixed-bed resin demineralizer, flow meter, resistance temperature detection (RTD) probe, Geiger Mueller detector, and two conductivity probes. Primary water enters the cleanup loop and goes into the water box, where temperature, radioactivity, and conductivity are measured. The water then flows through a filter with a 5-micron rating, which removes solid particles that may reduce the effectiveness of the demineralizer resin. Pressure gauges on either side of the filter will indicate any clogging. After leaving the filter, the water passes through a mixed-bed demineralizer and a second conductivity probe and joins the water exiting from the heat exchanger to return to the reactor tank. The primary coolant is sampled monthly for radioactivity, which normally consists primarily of tritium.

5.5 Primary Coolant Makeup Water System

Makeup water can be added to the primary system when necessary to replace any water lost by evaporation or other means. A steam-powered still (with steam supplied by the university power plant) can produce 50 gal (190 L) of distilled water per hr and has a storage capacity of 80 gal (303 L). In addition, the bulk shield tank contains 6,500 gal (24,600 L) of distilled water, which can be used as makeup water.

5.6 Nitrogen-16 Control System

After passing through the cleanup loop and heat exchanger, primary water is directed back to the reactor tank through a diffuser. The diffuser operates anytime the primary coolant pump is running. During normal operation of the reactor, nitrogen-16 is produced. Its half-life is approximately 7 seconds, and it decays with the emission of 6.1 million electron volts of gamma radiation. To retard the flow of water leaving the core toward the pool surface, flow from the diffuser is directed downward toward the top of the core. This increases the time it takes the coolant to reach the top of the pool, thus allowing a significant portion of the nitrogen-16 to decay. A radiation monitor is located directly above the pool (with indication in the control room). With the reactor operating at the 1250 kW(t) license power level, this monitor is expected to measure approximately 350 mR/hr at the pool surface, or approximately 100 mR/hr at 1 m above the bridge. Chapter 11 of this SER discusses radiological effects.

5.7 Auxiliary Systems Using Primary Coolant

Water in the bulk shield tank does not intermix directly with the primary coolant. Hence it will not have any significant effect on the operation of the reactor coolant systems. However, it can be used as a source of distilled makeup water. Water can be pumped from the bulk shield tank into the reactor tank, providing 6,500 gal (24,600 L) of makeup water.

5.8 Staff Evaluation

Within the constraints of the TSs, natural circulation cooling is capable of removing sufficient heat to prevent fuel failure during normal and credible accident conditions (see Chapter 13 of this SER for analyses of credible accidents). The primary and secondary coolant system is adequate to remove 750 kW(t) thermal power continuously. Additional plates must be added to the primary-to-secondary heat exchanger to continuously remove the 1250 kW(t) thermal license power. Until that modification is complete, the license may operate as long as TS 3.8.3(1) (maximum coolant pool outlet temperature of 130 °F) is met. This is no different than when the licensee performs a “pool heat up” calorimetric calibration of the reactor power instruments with the coolant system secured.

The coolant in the tank also functions as shielding during operation, which, along with TSs and administrative controls, meets the 10 CFR Part 20 limits (see Chapter 13 of SER for analyses of the loss-of-coolant accident).

The coolant purification system maintains the purity of the coolant within the conductivity limit of TS 3.8.3(2) (5 mmho/cm). The highly purified water minimizes degradation caused by corrosion of the reactor tank and primary coolant system, fuel cladding, and reactor structural components. In addition, the coolant contamination level from activated corrosion products is minimized. There is reasonable assurance that the reactor coolant systems can continue to function adequately for the duration of the proposed license renewal.

5.9 Conclusions

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

- Within the constraints of the TSs, natural circulation cooling is capable of removing sufficient heat to prevent fuel failure during normal and credible accident conditions and gives reasonable assurance of fuel integrity. Until the primary heat exchanger is modified to allow continuous operation at 1250 kW(t) thermal operation may be restricted by the TS limit on pool outlet temperature (TS 3.8.3(1)).
- TS 3.8.3(2) will ensure that the water purification system controls the conductivity of the primary coolant to limit corrosion of the reactor fuel and other materials in contact with primary coolant to acceptable levels for the duration of the license.
- The design of the reactor pool and the nitrogen-16 diffuser system will provide sufficient shielding and control of nitrogen-16.
- The TSs provide reasonable assurance that the cooling system will operate as designed and be adequate for normal reactor operations as described in the SAR.

6. ENGINEERED SAFETY FEATURES

6.1 Summary Description

The KSU reactor does not require emergency core cooling because of its power-level limit of 1250 kW(t). The accident analysis for a total loss of coolant, after extended operation from the license power limit (see Chapter 13), shows that peak fuel temperatures (see Table 13.3 of this SER) do not exceed 294 °C, which is well below the steady-state and accident SLs.

The KSU reactor facility is designed for confinement. The objective is to restrict the amount of radioactivity released to the environment and to reduce the consequences of such a release. The reactor is housed in a closed room designed to restrict leakage when the reactor is in operation, when the facility is unmanned, or when spent fuel is being handled for eventual placement in an exterior storage cask. The minimum free volume of the reactor room is approximately 144,000 ft^3, and the building will be equipped with a ventilation system capable of exhausting air or other gases from the reactor room at a minimum of 30 ft above ground level. This will allow for a large degree of dispersion and dilution. These features are used in the SAR consequence analyses and are required by TS 5.3.3. In addition, the surveillance aspects of TS 4.5.2 require that the reactor bay negative pressure be verified daily. If the reactor bay differential pressure gauge indicates a negative pressure, then the reactor bay exhaust fan is controlling airflow.

6.2 Staff Evaluation and Conclusions

The NRC staff reviewed the design, maintenance, operation, and TS requirements of the reactor room ventilation system. The staff concludes that the reactor ventilation system equipment and procedures are adequate to provide controlled release of airborne radioactive effluents during normal operations and in the event of abnormal or accident conditions. Furthermore, the reactor staff, researchers, and the public will be adequately protected from airborne radioactive hazards related to reactor operations. On the basis of its 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 and that components will be replaced as needed. Hence, there is reasonable assurance that the systems discussed in this chapter of the SER can continue to operate safely, as limited by the TSs for the proposed license renewal.

7. INSTRUMENTATION AND CONTROL SYSTEM

7.1 Summary Description

The I&C system for the KSU reactor consists of two basic subsystems, the RCS and the RPS. General Atomics manufactured both of these hardwired analog systems, which are widely used at various NRC-licensed facilities.

The RCS consists of the instrumentation channels (neutron monitoring, fuel temperature, water temperature, and conductivity), the control rod drive circuitry, and an automatic flux controller for use in the automatic mode of reactor operation. The neutron monitoring system consists of three independent channels (a wide-range logarithmic channel, a multirange linear channel, and a percent-power channel), providing at least two indications of reactor power from shutdown to full power. (In the pulse mode, an additional channel is installed and calibrated in the central thimble to record pulse data.) The control rod drive circuitry allows the insertion and removal of control rods, measures control rod position, and contains several interlocks that prevent unintentional rapid insertions of reactivity. The control rod drives are of a standard design.

The RPS initiates a reactor scram if any of several parameters in the RCS reach their limiting safety system settings. TS 3.4.3(1) specifies two high reactor power scrams and a manual scram. The TSs do not require other scrams, including high fuel temperature, loss of building power, and short reactor period. The safety limit is protected by the high-power scram. Additional scrams, although not required, provide redundant and diverse added protection against abnormal conditions. A reactor scram results in all control rods falling into the core via gravity, placing the reactor in a subcritical configuration. Since the core is cooled by natural convection flow, no other ESFs are required for safe shutdown.

Display indicators for all instrumentation are located either on the control console or on auxiliary instruments racks adjacent to the control console. They are visible to the reactor operator at all times. At the console, all instruments and controls necessary for reactor operation are within reach of the operator, including an intercom and telephone. Surveillance instruments are located next to the console, with visual and audible alarms to alert the operator to abnormal conditions.

7.2 Design of Instrumentation and Control System

7.2.1 Design Criteria

The KSU reactor is designed to be operated in three modes—(1) manual, (2) automatic, and (3) pulse. The manual and automatic modes are characterized by steady-state reactor conditions. The pulse mode requires the use of the transient (pulse) rod. The manual and automatic reactor control modes are used for reactor operation from source level up to 100 percent of licensed power. The manual mode is used for reactor startup and changes in power level, while the automatic mode is used for steady-state operation.

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 General Atomics or other industrial manufacturers of nuclear equipment and have been used safely with TRIGA reactors throughout the world.

7.2.2 Design-Basis Requirements

The primary function of the RCS is to control the manner by which reactivity is varied in the reactor core. The RCS should prevent the reactor operator from unintentionally inserting large amounts of positive reactivity through various interlock systems, including the ability to withdraw only one control rod at a time, the inability to rapidly eject the transient rod while in the steady-state mode of operation, the inability to withdraw any control rod (other than the transient rod) while in the pulse mode of operation, and the ability to prevent the withdrawal of control rods without a minimum signal from the neutron detectors. (There are no interlocks to prevent the insertion of control rods at any time.) The KSU TRIGA reactor meets all of these requirements.

The primary function of the RPS is to automatically insert all of the control rods into the reactor core when certain parameters reach their setpoints, whether limiting safety systems or redundant systems (reactor scram) (TS 2.2.3(1) and TS 3.4.3(1)). In addition, the reactor must be capable of being scrammed manually (TS 3.4.3(1)). Surveillance testing must verify scram functions (TS 4.4.2), and the control rods must drop into the core within a prescribed time (TS 3.4.3(2)). The KSU reactor satisfies these requirements given the annotated TSs.

In addition to the RCS and RPS, a radiation monitoring system (see Chapter 11 of this SER) is an essential part of the overall I&C system, both for personnel protection measures and emergency assessment actions. Area radiation monitors should provide the reactor operator with information on the actual radiation environment inside the reactor building, including alarms to warn personnel of dangerous conditions. Other instruments should signal the presence of dispersible radioactive materials, which indicates possible fuel cladding damage. The KSU reactor satisfies these requirements.

7.2.3 System Description

Most of the I&C system is located on a control console at which the reactor operator sits. Table 7.1 below lists the instrumentation available on the console.

Table 7.1 Console Instrumentation	
Function	**Description**
Console Power	Push Button Switch
Magnet Power/Scram Reset	Key Switch
Control Rod Drive Position	Push Button Switches
Apply Air to Pulse Rod	Push Button Switch
Rod Position Indicators	LED Displays
Mode Selector	Rotary Switch
Automatic Power Demand Control	10-Turn Potentiometer
Manual Scram Bar	Bar Covering Scram Switches
Fuel #2 and Water Temperature	Display and Selector Switch
Scram Status, Source Interlock, Low Air Pressure, Hi and Hi-Hi Sump Level, Surge Tank Level and Makeup, Upper and Lower Doors, Cooling System Power	Indicators and Control Switches
Wide-Range Logarithmic Power Channel	General Atomics NLW-1000 Channel
Multirange Linear Power Channel	General Atomics NMP-1000 Channel
Percent Power and Pulsing Channel	General Atomics NPP-1000 Channel
Source Interlock Override	Key Switch
Period Scram Override	Key Switch

The remainder of the I&C system is located in an instrument rack adjacent to the console and is immediately accessible to the reactor operator. Table 7.2 below lists the instrumentation on this rack.

Table 7.2 Instrumentation Rack
Rod Drop Timer
Pneumatic System Controls
Strip Chart Recorder
Spare High Voltage Supply
Battery Charger
NIM Rack for Experiments
Fuel Temperature #1 & Water Monitor
Conductivity Monitor
Area Radiation Monitors
Alarm Panel
Cooling Fan

Because the instrumentation rack contains general-use equipment, its configuration may change to accommodate the installation of new equipment or modifications, without affecting function.

7.2.4 System Performance Analysis

The current I&C system meets the design bases for the facility. Instrument upgrades have replaced and improved the I&C system from the original General Atomics system. The licensee states that those upgrades have increased the reliability of the I&C system and that few unanticipated reactor shutdowns occur. Daily checkouts are performed, and test procedures ensure that all equipment is maintained in operational status. A line conditioner provides regulated power to the instruments, protecting the equipment from electrical disruptions.

7.3 Reactor Control System

The RCS consists of the instrumentation channels, control rod drive circuitry and interlocks, and an automatic flux controller, all mounted on the console or instrument rack as previously discussed. The RCS measures several key reactor parameters, including power, fuel temperature, water temperature, and water conductivity.

7.3.1 Neutronic Instruments (Reactor Power)

There are three channels of neutron instrumentation, each capable of measuring reactor power independently. The first channel is the wide-range log channel (NLW-1000), which uses a fission chamber to measure power over the range from shutdown to full power. The channel also provides a period scram at +3 seconds. To prevent the operator from removing control rods with insufficient indication of neutron level, the channel provides a protective interlock at 2 counts per second, below which the shim, safety, and regulating rods may not be withdrawn. Another protective interlock prevents pulsing the reactor if reactor power is greater than 10 kW(t) (a later section of this SER discuses this interlock). The channel is tested for operability at the start of each operating day. No high-power scram is associated with this channel.

The second channel of neutron instrumentation is the multirange linear power channel (NMP-1000), which uses a compensated ion chamber for thermal neutron detection. This channel has automatic or manual ranging in order to display the appropriate decade of power. The channel provides a high-power scram (TS 2.2.3(1) and (TS 3.4.3(1)). A scram is also associated with a loss of high voltage to the detector (TS 3.4.3 A.2). TS 4.3.2 specifies the surveillance requirements. Both scrams are bypassed when the reactor mode selector switch is set to "Hi Pulse."

The third channel, or the power-range channel (NPP-1000), uses an uncompensated ion chamber that indicates percent power in the upper two decades of the power range. The channel provides a high-power scram (TS 2.2.3(1) and (TS 3.4.3(1)). A scram is also associated with a loss of high voltage to the detector (TS 3.4.3 A.2). TS 4.3.2 specifies the surveillance requirements. In addition to its power-measuring function when the reactor is in the manual (steady-state) mode, this channel can also operate in the pulse mode, measuring pulse power and energy. During pulsing operations (with the reactor mode selector switched to "Hi Pulse"), the power-range channel switches to a subchannel of the percent-power channel (capable of monitoring maximum pulse power).

An added pulsing channel, a BF_3 chamber, can be inserted into the central thimble of the core to measure pulse power and energy. This instrument is not an LCO and no required control functions are associated with this channel.

The reactor operator can obtain automatic reactor control by switching from manual operation to automatic operation using the mode selector switch on the control console. All instrumentation, safety, and interlock circuitry described for manual operation apply to operation in automatic mode. The regulating rod automatically controls the reactor power in accordance with the power demand set by the operator with thumbwheel switches. When reactor power, as measured by the linear multirange channel, is above or below the power demand, the regulating rod is servo-controlled to return reactor power to the demand level. No alarms are associated with the servo-control system.

7.3.2 Temperature

The licensee makes a variety of temperature measurements throughout the facility. Fuel element temperature is an important parameter and is measured with the use of instrumented fuel elements. Three of the six fuel elements in the B-ring are configured with chromel-alumel thermocouples. Each of the three instrumented fuel elements have three thermocouples embedded 0.30 in. (0.76 cm) below the fuel surface, one located at the midpoint of the element and the other two at positions ± 1 in. (2.5 cm) from the midpoint. An averaged value from the three thermocouples is used as a measure of fuel temperature.

The temperature in the primary cleanup loop is measured with a nickel-alloy thermistor and is displayed on a console meter, which is shared with fuel temperature via a rotary switch (see fuel temperature #2 in Table 7.1). Another indication of fuel temperature (see fuel temperature #2 in Table 7.2) is located in the instrument rack and can initiate a reactor scram if the temperature exceeds a preset value (normally 400 °C).

Reactor operators can read several other temperatures from a computer in the control room. Fuel temperatures from other fuel elements, inlet and exit temperatures of both the primary and secondary sides of the heat exchanger, and bulk pool temperature (for high-temperature alarm) are available.

7.3.3 Water Conductivity

Maintaining low water conductivity is necessary to prevent possible corrosion, deionizer degradation, or slow leakage of fission products from degraded cladding and is a requirement of TS 3.8.3(2). Primary water conductivity is measured at the inlet and outlet of the purification loop (see Table 7.2).

7.3.4 Control Rod Drives

Four control rods are necessary for reactor operations at 1250 kW(t). The rods are identified as the regulating rod, shim rod, safety rod, and transient (or pulse) rod. The present configuration uses a regulating rod, shim rod, and transient rod; the licensee will add a safety rod to support the 1250 kW(t) operation. The regulating, shim, and safety rods share identical control circuitry and are referred to as "standard" control rod drives. The standard control rod drives are motor operated, while the transient rod drive is pneumatically operated. All rods can be individually scrammed.

7.3.4.1 Standard Control Rod Drives

An electric motor-actuated linear drive positions each standard control rod. The rods are magnetically coupled to the drive. The control rod is connected to the drive at the lower end of a connecting rod. The upper end of the connecting rod terminates in an armature. Above the armature is a draw tube with an electromagnet at its lower end. When the electromagnet is energized, it is magnetically coupled to the armature. (A key-locked switch controls the application of power to the electromagnet. When the reactor is not operating, the key is either in the possession of a licensed operator or kept in a locked box.) The drive motor raises and lowers the draw tube, which in turn raises and lowers the control rod. The operator can thus position the control rod in any portion of its range. In the event of a reactor scram, the magnet is deenergized, releasing the armature. The connecting rod and control rod then drop into the core via gravity. Therefore, a loss of electrical power will result in the reactor being placed in a safe configuration (i.e., all rods inserted into the core).

Manual rod control is accomplished by push buttons on the control console. "Up" and "down" switches allow the operator to manually move the drive (and hence the rod) over the full range of travel from full in to full out, about 15 in., at a speed of about 12 in. (30 cm) per minute. Control rod position indicators are light-emitting diode (LED) display indicators, which receive a variable voltage input from 10-turn potentiometers actuated by the drive motors. Three microswitches activate lights on the control console to indicate (1) the draw tube is full up, (2) the draw tube is full down, and (3) the armature and magnet are coupled (in contact). A fourth microswitch is actuated when the control rod is full down. The system allows the operator to manually control each rod and to know its complete status.

The rod drive control circuit is unique in that it only uses microswitches for its control functions, eliminating the need for relays with their attendant reliability problems. The operator initiates rod movement by the motor-control push switches. Clockwise rotation of the drive motor (drive up) or counterclockwise rotation (drive down) is produced by shifting the phase between the windings of the motor with a capacitor.

If the control rod separates from the drive and drops into the core, actuation of the rod down limit switch causes the drive to move into the core. The contact light will extinguish, since the rod is down but the drive is not (the drive down limit switch has not been actuated). This alerts the operator that the rod has dropped.

7.3.4.2 Transient (Pulse) Rod Drive

The transient rod is a pneumatically operated system. Positioning the rod is accomplished by the application of air pressure through a three-way energized solenoid valve.

In the manual and automatic modes of operation, in which the transient rod may be used as a control rod, air pressure is applied to a piston, which is connected to the transient rod by a piston rod. The air pressure supports the rod and holds it at the position desired by the operator. This position is controlled by moving the shock absorber by means of a motor and worm gear, and the piston is held against the shock absorber by air pressure. Measurements of rod position and indication are identical to the standard control rods. In the event of a reactor scram, the solenoid valve is deenergized by the scram circuitry and shifts position, venting air from the piston to the atmosphere. The rod then drops into the core.

The transient rod is also used to insert a pulse of reactivity into the core. This is accomplished by supplying a burst of air that drives the piston, and hence the rod, upward. The amount of reactivity to be inserted, which is determined by the operator, is limited by the distance the rod is allowed to travel. This is accomplished by positioning the shock absorber at a height that is related to the desired amount of reactivity to be inserted. After the pulse is inserted, a variable timer deenergizes the solenoid valve, and the transient rod falls back into the core. Typically, the operator then manually scrams all control rods. This method of inserting a pulse is standard in NRC-licensed TRIGA research reactors.

7.3.4.3 Interlocks

Several interlocks are hardwired into the control system circuitry to prevent improper operation. These interlocks are typical of NRC-licensed TRIGA reactors and can be described as follows:

- An interlock prevents control rod withdrawal (shim, regulating, and safety rods only), unless the count rate neutron channel is indicating, above the minimum sensitivity of the NLW-1000 power-level monitoring channel (TS 3.3.3(2)). The licensee stated that, operationally, this requirement is set at 2 counts per second, which is greater than the minimum sensitivity of the channel. This interlock ensures that a neutron detector is functioning during startup and provides the mechanism for implementing the pulse interlock as described below. TS 4.3.2 requires daily surveillance of this interlock.

  The interlock may be bypassed (1) during fuel-loading operations when core inventory is not high enough to multiply the source above 2 cp, or (2) if channel operability can be verified by a source check.

- Air may not be applied to the pulse rod if the pulse rod shock absorber is above its full-down position and the reactor is in the steady-state mode. This interlock prevents the inadvertent pulsing of a reactor in the steady-state mode. The TSs refer to this as the pulse rod interlock (see TS 3.4.3(1)). TS 4.4.2 specifies the required surveillance.

Pulse operations are likely to exceed the maximum range of the power-level instruments used in the steady-state mode; therefore, this interlock ensures that power-level monitoring is configured for pulse operations.

- An interlock prevents simultaneous withdrawal of two or more control rods when the reactor is in the steady-state mode.

  The TSs for the original 100-kW(t) facility operating license included a maximum reactivity addition rate; this interlock ensured that the maximum reactivity addition rate was not challenged by inserting reactivity from multiple control rods. The facility operating license was subsequently revised (as indicated in Chapter 1) to include pulsing operations, where up to $2.50 can be added in a fraction of a second, effectively removing the safety basis for the reactivity addition rate. The renewed license will have a maximum license pulse reactivity limit of $3.00. Chapter 13 of this SER provides a detailed analysis of power excursions. The parameter of interest is peak fuel temperature resulting from a pulse. A $3.00 pulse will result in a peak fuel temperature of 746 °C, which is well below the accident safety limit of 1150 °C for stainless-steel clad fuel. The simultaneous rod withdrawal interlock is not part of the TSs; however, the licensee is retaining it as a functional part of the control system as a good operating practice.

- Only the pulse rod can be withdrawn when the reactor is in the pulse mode to limit control rod reactivity addition during a pulse (i.e., to the pulse rod only). This interlock does not prevent the scramming of any control rod. The interlock function is provided manually by engaging the source interlock with a pushbutton switch on the NLW-1000 instrument to configure the channel after the reactor mode selector switch is placed in the pulse mode. TS 3.4.3(1) specifies this interlock, which is known as the control rod (standard) position interlock. TS 4.2.2 specifies the required surveillance.

  The amount of reactivity added during a pulse is controlled by the position of the pulse rod. The analysis assumed reactivity addition to the maximum nominal value of the pulse rod. This interlock prevents movement of additional control rods during pulsing operations, thus minimizing the possibility of pulsing the reactor at an unknown reactivity level and reactor condition.

- An interlock prevents reactor pulses from being fired if the reactor power is above 10 kW(t) (normally set at 1 kW(t)).

  Previous safety bases did not address operations with elevated fuel temperature, while current accident analysis considers pulsing while operating at power. This interlock is a redundant feature to prevent exceeding the safety limit during a pulse and is not specified by a TS. The primary features that prevent exceeding the safety limit during pulsing are the $3.00 limit on the pulse reactivity (TS 3.2.3(1)) and the $4.00 limit on core excess reactivity (TS 3.1.3(1)). The reactivity accident analyses in Section 13.2.3 of this SER show that the excess reactivity limit, along with a pulse magnitude limited by the remaining available reactivity, after the power defect, prevents any combination of power operation and pulse from exceeding the safety limit of 1150 °C. The licensee is retaining this interlock as a functional part of the control system as a good operating practice.

7.4 Reactor Protection System

The RPS initiates an automatic reactor scram if any of several parameters reach their limiting safety systems settings or if the operator manually initiates a scram. The reactor is shut down by the dropping of control rods into the core. (Other reactor scrams may be installed in the RPS to accommodate special conditions, such as shutting down the reactor if personnel attempt to enter an area near an open beam port.) In the manual and automatic modes, automatic scrams are initiated by any of the following conditions:

- high power as measured by the multirange linear channel (required by TS 2.2.3(1) and TS 3.4.3(1))
- high power as measured by the power-range channel (required by TS 2.2.3(1) and TS 3.4.3(1))
- loss of high voltage for the nuclear instrumentation (required by TS 3.3.4 A.2))
- high fuel temperature (the scram is not required by TS but the measuring channel is required by TS 3.3.3(1))
- short reactor period (not required by a TS)
- operator action (a manual scram is required by TS 3.4.3(1))

The fuel temperature measurement requires an element that contains three chromel-alumel thermocouples embedded in the fuel. These instrumented elements are very expensive, and the thermocouples often fail and cannot be replaced. In addition, a number of factors can affect the ability of the thermocouple to measure the peak fuel temperature in the core. For example, the thermocouples are not placed within the element in the location of peak temperature, the temperature is dependent on the location of the element in the core (e.g., close or far from the core center, next to a control rod or not, next to a water hole or experiment or not), burnup of the element, and the coupling of the thermocouple to the fuel matrix and the width of the clad fuel gap. In addition, thermocouples do not follow transients well. All of these factors contribute to the fuel temperature being a poor parameter upon which to base the limiting safety system setting. However, the reactor power level must be known to meet the license limit, and it is easily measured and calibrated. The licensee has shown that reactor power is a good surrogate for fuel temperature, since temperature is a monotonic function of power. Therefore, even though the safety limit is based on temperature of the fuel, the limiting safety system setting is based on reactor power.

A short period scram is not required in a TRIGA pulsing reactor to prevent release of fission products from credible reactivity insertion accidents. See Section 13.2.3 of this SER for more discussion on reactivity accidents.

In the pulse mode, the period scram and the linear channel high-power scram are disabled. Loss of high-voltage scrams for the wide-range log channel and the multirange linear channel are also disabled. Pulse power is measured by the power-range channel. Also, as previously discussed, to terminate the pulse, a variable timer inserts a scram of the transient rod only while in the pulse mode and after the pulse is initiated.

Reactor operators check the operability of all scrams daily before reactor startup (TS 4.3.2).

7.5 Engineered Safety Features (ESF) Actuation Systems

There are no ESF actuation systems. Control rod insertion is accomplished by gravity, and core cooling is provided by natural convection cooling. Therefore, ESF systems are not required; however, Chapter 6 of this document discusses facility design features that support confinement, and the TSs based on them, which help control and mitigate radioactive material releases.

7.6 Control Console and Display Instruments

The physical layout of the control console and display instruments places them within sight and easy reach of the reactor operator. All push buttons required for reactor control are located on the console, and a computer display on top of the console shows reactor status information on a single screen.

The operator can immediately reach other control devices in the control room. Directly behind the operator are the circuit breakers that interrupt power to electrical devices in the control room and reactor bay. A Halon fire extinguisher is located next to the breakers for use in fighting electrical fires. A wall-mounted box in the control room has illuminated switches to indicate the presence of personnel in the reactor bay.

7.7 Radiation Monitoring Systems

Radiation monitoring systems are used throughout the reactor facility, and the indicators for the various detectors are visible to the reactor operator in the control room. Geiger-Mueller detectors are located at the reactor pool surface and the cleanup loop (see Table 5.1 of this SER). A 5-roentgen per hour (R/hr) monitor on the 22-ft level (reactor bridge) serves as an evacuation alarm. An air monitoring system, also on the 22-ft level, measures iodine, particulate, and noble gas activity. A continuous air monitor is located on the 12-ft level. This monitor senses any changes in airborne contamination in the reactor bay relative to background levels. Numerous other remote Geiger-Mueller detectors are located throughout the facility—at the top of the reactor tank, above the bulk shield tank, near the ion exchanger in the primary coolant system, and directly over the primary water tank. These detectors have readouts locally and in the control room, with visual indicators for normal, alert, and alarm conditions. The control room alarm has an audible signal as well. Finally, an independent monitor with visual and audible alarms is located above the entry door to the reactor bay. In addition to these fixed instruments, many portable survey instruments are located throughout the bay. Film and ring badges are issued to the staff, and neutron/gamma-sensitive pocket ion chambers are available in the control room for visitors.

7.8 Staff Evaluation

The RCS, RPS, and process instrumentation are designed to ensure safe operation and shutdown of the facility. The reactor operator can easily access all controls, instrumentation, and indications necessary for safe operation of the reactor. There is reasonable assurance that the I&C systems can continue to operate safely for the duration of the license renewal. Redundancy and diversity in the RPS reduces the risk of common-mode failures of the system. The key for the switch that controls the magnet power is in the possession of a licensed operator, in the console with a licensed operator in control, or in a locked box. This helps the licensee maintain compliance with the requirements of 10 CFR 50.54(k).

7.9 Conclusions

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

- The licensee has shown that all nuclear and process parameters important to the safe and effective operation of the reactor will be displayed at the control console. The display devices for these parameters are easily understood and readily observable by an operator positioned at the reactor controls. The control console design and operator interface are sufficient to promote safe reactor operation.
- The RCS 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 components and devices of the RCS are designed to sense all parameters necessary for facility operations with acceptable accuracy and reliability.
- The reactor safety system is designed to prevent or mitigate hazards to the reactor or the escape of radiation, so that the full range of normal operation poses no undue radiological risk to the health and safety of the public, the facility staff, or the environment.
- The annunciator and alarm panels on the control console provide assurance that systems important to adequate and safe reactor operation will function properly.
- The multiplicity of radiation detection systems ensures that personnel are adequately warned of the presence of abnormal levels of radiation.
- The locking system on the control console reasonably ensures that the reactor facility will not be operated by unauthorized personnel.

8. ELECTRICAL POWER SYSTEMS

The reactor's primary electrical power, provided through the KSU power grid, is supplied by on-campus plant and commercial generators. Power lines are located underground. Loss of electrical power will result in all of the control rods dropping into the core, thus placing the reactor in a safe configuration. Since the core is cooled by natural convection, no emergency power is required for cooling. Backup battery systems are provided for emergency lighting, the security system, and the 22-ft-level evacuation alarm.

8.1 Normal Electrical Power Systems

Utility voltage (4160 volts) is delivered by underground cables to three transformers inside a locked, fenced area immediately outside the reactor bay, within the facility boundary. One transformer supplies building power and the control room breaker box. A second transformer supplies the breakers in the reactor bay for the cooling system and recirculation ventilator. The third transformer (normally disconnected) can supply power for experiments that may require large amounts of electrical power.

The control room breaker box supplies all electrical outlets throughout the facility, a line conditioner, and power to the reactor bay crane. The line conditioner provides isolated, regulated power for reactor instruments and control systems so that they are not affected by minor fluctuations. Interruption of normal power will deenergize the line conditioner, which must be manually reset following the return of normal power.

The bulk of electrical wiring is in shielded conduits as required by commercial electrical codes. Instrument wires from the I&C system are routed through a subfloor conduit in the control room into a wire tray leading to the upper level of the reactor bay. A secondary tray is routed around the perimeter of the 12-ft level of the reactor. Grounding straps are used to ensure common ground between the control room and instrument locations. All electrical devices are installed according to the electrical codes.

8.2 Emergency Electrical Power Systems

The emergency electrical power systems provide lighting and surveillance for emergency conditions and to maintain physical security. Battery backup power is used for emergency lighting, the university fire alarm system, the evacuation alarm, and the security system. The backup systems have regular maintenance schedules and periodic surveillance to ensure operability.

Emergency lighting, with self-contained batteries, is located at the upper level of the reactor bay, at all exits from the bay, exit sign illumination, and fire alarm sensors and pull stations. The KSU fire safety personnel test these systems periodically. Backup batteries, with annual maintenance and testing, also supply the security system.

8.3 Staff Evaluation

The electrical power system is capable of supporting the reactor facility in its normal operation. The emergency power system is capable of providing necessary support systems (e.g., lighting, monitoring for emergency conditions, evacuation system, and physical security systems) in the event of loss of normal power. Loss of power will cause a safe shutdown of the reactor.

8.4 Conclusions

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

- The design bases and functional characteristics of the normal and emergency electrical power systems were reviewed, and the proposed systems are capable of providing the necessary range of services.
- The design and operating characteristics of the source of emergency electrical power are basic and reliable, ensuring availability if needed.
- The design of the normal and emergency electrical power systems will not interfere with safe facility shutdown or lead to reactor damage if the systems malfunction during normal reactor operation.

9. AUXILIARY SYSTEMS

The systems described in this chapter are not directly required for reactor operation but are used in support of the reactor for normal and emergency operations.

9.1 Heating, Ventilation, and Air Conditioning Systems

Heating and air conditioning of the reactor bay and control room are provided by either steam (heating) or chilled water (air conditioning) from the university's physical plant, with a thermostat in the facility to regulate temperatures. The KSU Department of Facilities provides routine maintenance and service of these systems.

Section 3.5.4 of this SER describes the ventilation system for the reactor bay. In addition to the bay system, the control room has a separate fan system that is available on demand for ventilation. The system exhausts from a duct out of the roof immediately above the control room.

9.2 Handling and Storage of Reactor Fuel

The fuel elements that are on site are located in three separate areas—the reactor core, the reactor tank, or fuel storage pits. Fuel elements in the reactor core do not present any safety concerns since the core is designed for exactly that purpose. Fuel elements in the reactor tank are stored in aluminum racks on the wall of the tank. The spacing of the elements in the racks and the maximum number of fuel elements in each rack (six) prevent accidental criticality. Control of spacing within a fuel storage rack is not required to limit the effective neutron multiplication factor of the array. Since each storage rack is limited to six elements, criticality is not possible even if the racks were loaded with fresh fuel elements containing the maximum amount of uranium.

Fuel may also be stored in locked fuel storage pits. The pits are used to store fuel elements that were damaged in handling or shipping, and they are inspected annually.

TS 5.2.3 specifies that all fuel elements or fueled devices shall be in a safe, stable geometry such that the k_{eff} of the arrangement is less that 0.8 and adequate cooling by air or water is available. The licensee's radiation protection program assure that sufficient shielding is provided to minimize personnel exposure ensure that the requirements of 10 CFR Part 20 are met.

Fuel elements are individually handled with a special tool that connects to the end fitting on the element. The access to the tool is controlled. The TSs require that any fuel movement (e.g., from the core to an in-pool storage rack) be performed in the presence of a licensed senior reactor operator, a reactor operator at the control console to monitor conditions in the reactor and reactor bay, and one other person trained in fuel-handling operations. Transfers of activated fuel elements are accomplished with the use of a fuel element cask and the reactor bay polar crane. Facility procedures govern fuel handling.

TS 3.7.3 specifies that fuel elements be inspected for bowing and elongation. A measurement tool for this purpose is located on the inside of the reactor tank. Visual inspection of the fuel element is accomplished with a periscope.

9.3 Fire Protection Systems and Programs

Fire protection systems are maintained and serviced by the university's Campus Fire Safety, Department of Environmental Health and Safety, Division of Public Safety. The building fire alarm system is part of a campuswide network. Fire alarm signals are sent to the campus police station via a line-monitored system, where the locations of the alarm as well as building maps are displayed on a computer terminal.

A wide array of fire detection and firefighting systems are within the facility. Pull stations are located in the control room and the reactor bay. Smoke detectors are located throughout the bay and control room that alert campus police. Fire extinguishers are readily available to reactor personnel. A Halon fire extinguisher is located in the control room for electrical fires. Carbon dioxide extinguishers are located in the hallway outside the control room and on the 22-ft level, the 12-ft level, and the 0-ft level of the bay. Dry chemical extinguishers are located on the 12-ft level, the 0-ft-level stairs, and next to the cooling system pumps. Reactor personnel visually inspect all extinguishers monthly. Campus fire safety personnel perform pressure testing and general maintenance annually.

9.4 Communication Systems

The reactor facility has an intercom system. The control unit is mounted above the control console, with a foot switch underneath the console for hands-free communication. Speakers are located at the reactor manager's office, the reactor operators' office, the control room door, the lower level door to the reactor bay, the neutron activation analysis laboratory, the 22-ft level above the tank, the 22-ft level near the pool surface, and next to each of the four beam ports.

Telephones at the facility share a common line and are located in the reactor manager's office, the reactor operators' office, the control room, and the 22-ft level and 0-ft level of the reactor bay.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

Reportable quantities of radioactive materials are possessed under the university's State radioactive materials license, the reactor facility license, and a separate NRC SNM license. The reactor fuel is the property of DOE. KSU owns several radioactive sources. Radioactive materials, including SNM, are inspected for contamination and inventoried on a quarterly basis. Several areas are designated for storage of these materials.

Byproduct material produced in the reactor for research purposes is transferred to the State radioactive materials license and is recorded in a transfer log. The KSU Department of Environmental Health and Safety, Division of Public Safety, maintains the State license, which is administered by the University Radiation Safety Committee (URSC). Only individuals listed under the license may receive materials. Normally, a member of the reactor facility staff is also approved by the committee to receive byproduct and SNM under the State license. The State sets possession limits and the URSC determines use limits. Transfers off-campus to other licensees must first go through the Department of Environmental Health and Safety, Division of Public Safety. The facility has several sources for reactor startup, research, and instrumentation calibration purposes that are possessed under this license. Low-level wastes generated under the reactor license are coordinated with the Department of Environmental Health and Safety, Division of Public Safety.

Reactor fuel comprises the bulk of SNM at the facility. This fuel is owned by DOE and possessed by KSU under the reactor facility license (R-88). Also under License R-88 are fission chambers containing SNM, owned by KSU. The license sets possession limits at 4.2 kg uranium-235 in enrichments less than 20 percent, and 90 gm uranium-235 in any enrichment. KSU possesses small quantities of SNM for experiments the under the State of Kansas SNM License 38-C011-01.

Storage locations for radioactive materials include the reactor bay, source cave, safe, and designated laboratories. Section 9.2 of this SER describes fuel storage locations. A shielded source cave, located in the northeast corner of the reactor bay, is used for storing large sources and low-level wastes. A shielded, locked safe in the reactor bay can be used to store small sources. Laboratories storing radioactive materials include the panoramic irradiation room, the beta-shielding laboratory, the radiation detection laboratory, and the neutron activation analysis laboratory.

9.6 <u>Cover Gas Control in Closed Primary Coolant Systems</u>

The KSU reactor has an open primary coolant system and, therefore, no cover gas. Nitrogen-16 is controlled as described in Chapters 5 and 11 of this SER. The use of helium in the pneumatic system (see Section 10.2.4 of this SER) reduces the possible inventory of radioactive argon.

9.7 <u>Other Auxiliary Systems</u>

9.7.1 Reactor Sump System

All floor drains and the ventilator condensate line feed into the reactor sump, a square cavity covered with steel plates and a capacity of about 1,056 gal (4,000 L). A sump discharge system allows the sump liquid to be recycled through filters and discharged through a separate filter to ensure that insoluble radioactive materials are not discharged to sanitary sewers. Before discharge, liquid samples are analyzed for specific activity. In the past, the only isotope normally discharged has been tritium from primary water in quantities well below the 10 CFR Part 20 limits. A facility procedure controls the operation of the sump water discharge system.

9.7.2 Reactor Bay Polar Crane

A polar crane in the reactor bay is used to manipulate loads up to 8000 pounds (lbs). Even though the power to the crane is supplied by a breaker in the control room, a disconnect is located on the crane to give personnel operating the crane local and positive control over the power. A lockable breaker on the west wall of the bay permits securing power to the crane.

9.7.3 Associated Laboratories

The neutron activation analysis laboratory, used for gamma-ray spectroscopy, is located in the basement of Ward Hall, outside the operations boundary of the reactor. The radiation detection laboratory, also outside the operations boundary, is a student laboratory. Another room contains a panoramic irradiator for performing gamma-ray instrument calibrations.

9.8 <u>Staff Evaluation and Conclusions</u>

The fuel-handling and storage system designs, as well as sealed source and waste storage, have been in use by this licensee since the facility was originally licensed 40 yrs ago. No significant degradation of the equipment by corrosion or aging has occurred. Both the TSs and the radiation protection program require handling procedures (see Chapter 12 of this SER). There is reasonable assurance that the licensee can continue to handle and store fuel for the life of the renewed license.

Fire protection is maintained by the KSU Campus Fire Safety, Department of Environmental Health and Safety, Division of Public Safety. There is reasonable assurance that the fire protection systems and practices will continue for the life of the renewed license.

The communication systems have been adequate under the present license. There is no reason to doubt their adequacy for continued operation under the renewed license.

10. EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

10.1.1 Experimental Programs

The KSU reactor provides educational and training services to support the Department of Mechanical and Nuclear Engineering degree programs. Other university departments are also supported. The reactor produces radioisotopes for research, including tracer/gauging and radioanalysis applications. The reactor also supports research using extracted beams.

10.1.2 Experimental Facilities

The KSU reactor supports a variety of experimental irradiation facilities which are either in the reactor core or pool or are external to the core. Facilities in the core or pool include a central thimble, a rotary specimen rack, a pneumatic transfer (“rabbit”) tube, some fuel element locations into which dry tubes may be inserted, a thermal column, and a thermalizing column.

In addition to these facilities, four beam tubes penetrate the pool walls to allow for the extraction of neutrons and gamma rays. Three of the tubes are oriented radially with respect to the center of the core. The fourth tube is tangential to the outer edge of the core.

10.1.3 Experiment Monitoring and Control

Applicable experiment procedures identify specific monitoring requirements for individual experiments. In general, there are requirements for leak detection, radiation monitoring, and the installation of external scrams in the RPS.

The beam tubes, thermal column, and thermalizing column interface with the walls of the reactor pool. Consequently, coolant leakage into these facilities is possible. Piping from these facilities is directed to a leak-off volume that is instrumented with pressure monitors that will detect a partially or fully filled volume. The monitors are located on the outside of the north wall of the pool. Watertight seals are installed at the interface between the beam tubes and the reactor bay.

At each beam port, electrical lines are installed that permit the connection of area radiation monitors. In addition, the RPS contains a channel designated as an “external” scram. Sensors for parameters related to experiments can be connected into this channel and cause a reactor scram if those parameters exceed limits.

10.1.4 Experiment Review and Approvals

The Reactor Safeguards Committee (RSC) reviews and approves experiments before they are conducted. The reactor supervisor or nuclear reactor facility manager (hereafter referred to as the facility manager) may schedule an experiment for performance after the experiment and procedures are approved.

10.2 Experimental Facilities

The reactor is a multipurpose research facility, with irradiation locations inside the core boundary, in the reflector, outside the reflector, and outside the biological shielding. Experiments may be installed in these locations.

10.2.1 In-Core Facilities

Irradiation facilities within the core boundary include upper grid plate fuel element spaces, a series of smaller penetrations in the upper grid plate, and the central thimble.

10.2.1.1 Available Fuel Element Spaces

Experiments may be inserted in spaces designed for fuel elements. The experiment is typically inserted through a dry tube with an “S” bend to minimize streaming radiation at the pool surface. A dry tube may be lined with cadmium to support specifically tailored neutron irradiations.

10.2.1.2 Small Upper Grid Plate Penetrations

The upper grid plate has a series of small (about 0.25 in.) holes to permit flux-mapping experiments.

10.2.1.3 Central Thimble

The reactor is equipped with a central thimble for access to the point of maximum neutron flux in the core. The thimble is an anodized aluminum tube that fits through the center holes in the top and bottom grid plates, terminating with a plug at a point about 19 cm below the lower grid plate.

10.2.2 In-Tank, Ex-Core Facilities

Irradiations may be performed above and below the core and adjacent to the radial reflector. Irradiations are also authorized in the bulk shield tank and at the outer face of the thermal column.

10.2.3 In-Reflector Facilities

The in-reflector facilities include a thermal column with dry irradiation space, a thermalizing column that provides irradiation space in the bulk shield tank, and a rotary specimen rack inside the radial reflector.

10.2.3.1 Thermal Column

The thermal column is a large, boral-lined, graphite-filled aluminum container, about 4 ft (1.2 m) square in cross section and about 5 ft (1.5 m) in length. The outer portion is embedded in the concrete biological shield, and the inner portion is welded to the aluminum reactor tank. The container is open toward the reactor bay, with the opening covered by a steel door. Blocks of nuclear-grade graphite, which occupy the entire volume, thermalize neutrons for use in experiments outside the column.

10.2.3.2 Thermalizing Column

The bulk shielding experimental tank is about 12 ft (3.6 m) deep, 8 ft (2.4 m) wide, and 9 ft (2.7 m) long and is filled with water for shielding. The thermalizing column is similar to the thermal column but smaller, extending from the bulk shielding tank through the concrete biological shield to the aluminum reactor tank. When the column is not in use, an aluminum cover plate keeps water out of the column. As discussed above, blocks of graphite in the column provide for the thermalization of neutrons.

10.2.3.3 Rotary Specimen Rack

A 40-position rotary specimen rack is located in a well in the top of the graphite radial reflector. The rotary specimen rack allows for the large-scale neutron and gamma irradiation of radioisotopes and for the activation and irradiation of multiple material samples. The samples are manually loaded from the top of the reactor (at the 22-ft level) through a watertight tube into the rotary specimen rack. As each sample is loaded, the rack is rotated to allow subsequent samples to be loaded.

10.2.4 Automatic Transfer Facilities

The pneumatic transfer system accommodates the transfer of individual small specimens (i.e., rabbits) into and out of the reactor core. Specimens are placed in a small polyethylene capsule which in turn is placed into a receiver. The rabbit travels through aluminum and polyethylene tubing to the terminus in the core, normally in the outer ring of fuel element positions. Once the irradiation is complete, the rabbit returns along the same path to the receiver. Directional gas flow (helium) moves the rabbit between the receiver and terminus (helium gas is used to minimize the production of argon-41).

10.2.5 Beam Ports

The reactor is provided with four beam ports (tubes). Beam tube sleeves are welded to the outside surface of the reactor tank to allow extraction of neutrons and gamma rays for a variety of experiments and irradiation of specimens as large as 6 in. (15 cm) in diameter. Three of the beam tubes are oriented radially with respect to the center of the core. The fourth tube is tangential to the outer edge of the core. All radial tubes terminate at the outer edge of the reflector assembly. The beam tubes are in two sections—the inside section with a 6-in. (15-cm) diameter and the outside section with an 8-in. (20.3-cm) diameter. The outer section contains a vent line in the argon vent system, thus permitting the purging of accumulated gases.

Each section of the tube is fitted with a plug for shielding purposes. The outer end of the tube (at the reactor bay) is fitted with a shutter and a gasket-sealed door, which is locked to prevent the inadvertent opening of the tube and the loss of water in the event of a beam tube leak.

10.3 Experiment Review

A wide array of experiments have been documented and approved for execution. The experiment and review process is conducted in accordance with approved facility administrative procedures. Before an experiment can be conducted, a request for the experiment is submitted to the reactor supervisor, who verifies that the experiment meets the limitations on experiments in TS 3.6.3, TS 4.6.2, and TS 5.4.3; the regulations; and, in particular, the requirements of 10 CFR 50.59, “Changes, Tests and Experiments.” The reactor supervisor then schedules the experiment after verifying that it has been approved by the RSC. If the proposed experiment was not previously approved, it is considered a new experiment. To perform a new experiment following the specifications in TS 6.4, an analysis must be done to verify that the specifications in the above TSs will be met, and the analysis package of the proposed experiment is then presented to the RSC for approval.

10.3.1 Planning and Scheduling of New Experiments

As mentioned, new experiments require the approval of the RSC before they are performed. To support the RSC review, a written description of each proposed new experiment must be prepared, with sufficient detail to allow evaluation of experiment safety. The information, at a minimum, must include the purpose of the experiment, background, procedure, experimental methods, description of equipment, and summary of potential effects on the reactor. In addition, a determination must be made using the criteria in 10 CFR 50.59 that the new experiment can be done without NRC staff approval. TS 6.2(c) specifies the diversity and expertise of the RSC membership. The university radiation safety officer is an ex officio member of the RSC (TS 6.2(c).5), and the committee quorum requires attendance by all ex officio members (TS 6.2(e)). In accordance with TS 6.2(f), any permissive action of the RSC requires both the affirmative vote of the university radiation safety officer and the majority vote of the members present. (Chapter 12 of this SER further discusses these aspects of the RSC.) After RSC review and approval, the experiment may be scheduled.

10.3.2 Review Criteria

The RSC shall consider new experiments in terms of their effect on reactor operation and the possibility and consequences of failure, including consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and reactivity effects. Before approval, the RSC must conclude that the experiment will not constitute a significant hazard to the integrity of the core or to the safety of personnel. Evaluation of the proposed experiment must include, at a minimum, a finding that the likelihood of the following occurrences is minimal or acceptable in both normal and failure modes:

- breach of fission product barriers, which could occur through reactivity effects, thermal effects, mechanical forces, and/or chemical attack
- interference with RCS functions, which could occur through local flux perturbations or mechanical forces that can affect shielding or confinement
- introduction or exacerbation of radiological hazards, which could occur through irradiation of dispersible material, mechanical instability, inadequate shielding, and/or inadequate controls for safe handling
- interferences with other experiments or operations activities, which could occur through reactivity effects from more than one source, degradation of performance of shared systems, physical interruption of operational activities or egress, toxic or noxious industrial hazards, or unanticipated effects of pulsing

The RSC must also determine that the proposed experiment complies with the TSs. If an event or new information challenges the original evaluation, the RSC shall review the experiment approval and determine whether the original approval is still valid before the experiment continues.

10.4 Staff Evaluation

The design and construction of the KSU experimental facilities are similar to other TRIGA facilities that have been operated safely for many reactor years. The KSU operating and maintenance experience, as reviewed by the staff in annual and special reports, and reports by

NRC inspections demonstrate that the KSU management and personnel can operate the experimental program safely. The limitations on experiments that are specified in the TSs are generally consistent with the guidance in ANSI/ANS-15.1, “Development of Technical Specifications for Research Reactors,” and Regulatory Guide 2.2, “Development of Technical Specifications for Experiments in Research Reactors,” issued November 1973, and comparable to those at reactor facilities similar to the KSU reactor. The administrative procedures for review and approval, as required by the TSs, provide reasonable assurance that experiments are unlikely to fail, release significant radioactivity, damage the reactor, or prevent safe shutdown of the reactor. However, if there is a failure of the experiment, the review and approval process, along with the TSs and the requirements of 10 CFR 50.59, provide reasonable assurance that the consequences are less than those of the MHA as analyzed in the KSU SAR and evaluated in Chapter 13 of this SER. There is reasonable assurance that the reactor can continue to operate safely with its experimental program, as specified by the TSs, for the duration of the renewed license.

10.5 Conclusions

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

- The licensee has demonstrated its reliance on an independent safety committee to conduct reviews of all experiments. The diversity and expertise of the committee’s membership are appropriate to its function (see Chapter 12 of this SER for more discussion).
- The procedures and methods used ensure a detailed review of all potential safety and radiological risks that an experiment may pose to the facility or the public.
- Administrative controls are adequate to protect operations personnel, experimenters, and the general public from radiation and other potential hazards caused by experiments.
- The design and planned operation and use of experimental facilities will not result in operation of the reactor outside TS limits (TS 3.6). The design of experimental facilities ensures that facility staff and public radiation doses are within the limits of the regulations and the facility’s ALARA program.
- 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 4.6.2 (surveillance), TS 5.4 (design of experiments), and TS 6.4 (proposals for experiments).

11. RADIATION PROTECTION AND WASTE MANAGEMENT

11.1 <u>Radiation Protection</u>

The radiation protection program for the KSU research reactor facility is designed to meet the requirements of 10 CFR 20.1101(b). The program also incorporates requirements of the State of Kansas. The goal of the program is to control radiation exposures and radioactivity releases to levels that are ALARA without significantly restricting operation of the facility for purposes of education and research. The program is executed in coordination with the Office of Radiation Safety, Department of Public Safety, for KSU. The RSC has review and approval, as well as audit responsibility, for the program (see Chapter 12 of this SER for more details).

Certain aspects of the program address radioactive materials regulated by the State of Kansas (an Agreement State) under License C0011-01. The radiation protection program was developed following the guidance of ANSI/ANS-15.11, "Radiation Protection at Research Reactor Facilities," issued in October 1992.

11.1.1 Radiation Sources

Radiation sources present in the reactor facility may be in gaseous (airborne), liquid, or solid form. Airborne sources consist mainly of argon-41, nitrogen-16, and tritium (hydrogen-3). Liquid sources are mostly condensate water from the facility air-handling system, with occasional releases of primary coolant from level adjustments in the reactor tank or bulk shield tank to support maintenance and operations. Solid sources consist of reactor fuel, a startup neutron source, and fixed radioisotope sources used for instrument calibration. Solid sources also consist of wastes, such as ion-exchange resins, contaminated tools, labware, and anticontamination clothing associated with reactor experiments and surveillance or maintenance operations.

11.1.1.1 Airborne Radiation Sources

During normal operation of the reactor facility, the following three airborne radiation sources are produced:

(1) Argon-41 is produced as a result of neutron activation of naturally occurring argon-40 in the reactor bay air (neutron flux in the reactor bay is insignificant except in beam tube experiments), air in the rotary specimen rack, and air dissolved in primary coolant. Argon-41 has a half-life of 1.8 hr. The argon-41 produced ultimately migrates to the reactor bay and may result in occupational exposure during normal operations. In the appendix to Chapter 11 of the SAR, the licensee performed detailed calculations for argon-41 doses at full power (i.e.,1250 kW(t)). If all of the argon-41 activity from the rotary specimen rack and primary coolant were instantly dispersed into the reactor bay atmosphere, the DAC would be about 7.2×10^{-7} microcuries per milliliter (µCi/mL), which is below the Appendix B to 10 CFR Part 20, occupational DAC limit of 3×10^{-6} µCi/mL.

The licensee has also calculated the offsite impact of argon-41 resulting from normal operations. The peak offsite effluent activity concentration would be about 0.004 picocurie per milliliter (pCi/mL) at 443 ft (135 m) downwind under slightly unfavorable atmospheric conditions. This concentration is less than half of the 10 CFR Part 20 effluent limit of 0.01 pCi/mL. A full year of operation at the maximum power level of 1250 kW(t) would result in a dose of less than 4 mrem, well below the maximum allowed 10 mrem from effluents specified in 10 CFR 20.1101(d).

(2) Nitrogen-16 is produced by oxygen-16 neutron capture in the primary coolant. It is produced in the core volume and rises to the pool surface, resulting in a cylindrical volume source within the reactor tank. It thus leads to significant exposure rates above the tank. The half-life of nitrogen-16, however, is only 7 seconds, and so airborne concentrations of nitrogen-16 in the reactor bay and off site are negligible compared to the direct dose above the reactor tank. Operation at the licensed power level of 1250 kW(t) is expected to lead to a dose rate at 1 m above the center of the tank of approximately 100 mR/hr from nitrogen-16 decay. (These calculations do not take any credit for the nitrogen-16 diffuser system, which is always operating during normal operations and serves to reduce the accumulation of nitrogen-16 near the top of the pool.)

(3) Tritium (hydrogen-3) is produced as a result of neutron absorption in hydrogen in the primary coolant, with a half-life of about 12 yrs. Tritium assays are performed monthly, and extrapolating from history (based on operation at the present full-power level of 250 kW(t)), the tritium concentration at 1250 kW(t) would be less than 3.42×10^{-8} µCi/mL, which is below the DAC limit of 2×10^{-5} µCi/mL and the atmospheric effluent limit of 1×10^{-7} µCi/mL.

11.1.1.2 Liquid Radioactive Sources

During normal operation of the reactor, the only production of liquid radioactive materials occurs through neutron activation of impurities in the primary coolant. Many of these impurities are captured in mechanical filtration or ion exchange in the demineralizer resins and are treated as solid waste. The radioactive liquids from resin changes are collected in sump tanks, along with condensate from the air-handling system. These liquids are released to the sanitary sewer after assay and filtration. The only radionuclides observed are tritium and trace quantities of cesium-137. Typically, there are three releases annually, with concentrations of 2×10^{-4} µCi/mL of tritium and 2×10^{-7} µCi/mL of cesium. Even without dilution, these values are well below the 10 CFR Part 20 effluent concentration limits of 1×10^{-3} µCi/mL and 1×10^{-6} µCi/mL, respectively. Activation products in the primary coolant, such as magnesium-27, aluminum-28, and manganese-56, are of such low concentrations and have such short half-lives that they are not detected in surveillance programs.

11.1.1.3 Solid Radioactive Sources

The primary solid radioactive source associated with reactor operations is the enriched uranium in the fuel and fission chambers. Because the actual inventory of fuel and other sources changes continuously in normal operation, their exact amount is not known but in all cases is less than the license limit. Section 11.2 of this SER addresses the disposition of solid wastes other than fuel.

11.1.2 Radiation Protection Program

The radiation protection program is conducted in accordance with the requirements of 10 CFR Part 20. Its goal is to limit radiation exposures and radioactivity releases to an ALARA level.

11.1.2.1 Management and Administration

The facility manager is responsible for the preparation, audit, and review of the radiation protection program. The RSC reviews the activities of the manager and the semiannual audits prepared by the manager during the committee’s annual inspections (TS 6.2(b).7).

The reactor supervisor is responsible for training, surveillance, and recordkeeping. ALARA activities are incumbent upon all radiation workers associated with the facility.

Substantive changes in the radiation protection program require the approval of the RSC.

Editorial changes, or changes to appendices, may be made on the manager’s authority. Changes to the radiation protection program apply automatically to operating or emergency procedures. As with any procedure, the reactor supervisor or facility manager may override elements of the program on a temporary emergency basis as long as the emergency changes are brought promptly to the attention of the RSC.

11.1.2.2 Training

The reactor supervisor is responsible for the implementation of radiation protection training. Personnel who need access to the facility, but who are not reactor staff, are either escorted by trained personnel or are given unescorted access training. Radiation training for licensed operators is integrated with the operator training and requalification program.

The goal of unescorted access training is to provide the knowledge and skills necessary to control personnel exposure to radiation associated with the operation of the reactor. Specific knowledge and skills required to meet the goal have been developed as learning objectives, and training material is based on these objectives. Such training explicitly addresses the specific requirements of 10 CFR Part 19, “Notices, Instructions and Reports to Workers: Inspections and Investigations,” 10 CFR Part 20, the radiation protection plan, and the emergency plan. A facility walk-through is part of the training.

All persons granted unescorted access to the facility must receive the training and must score at least 70 percent on a written examination on radiation safety and emergency preparedness. The facility must keep examinations on file for audit purposes for at least 3 yrs.

The reactor staff performs health physics functions following approved procedures. Therefore, procedure training for the licensed reactor staff includes additional radiological training. Examinations are prepared and administered in accordance with the requalification plan.

11.1.2.3 ALARA Program

The licensee is committed to keeping both occupational and public radiation exposure ALARA. Specifically, the goal of the program is to ensure that exposures are no greater than 10 percent and 50 percent of the occupational limits and the public limits specified in 10 CFR Part 20. Any action (e.g., operations, experiments) that might result in as much as one-half of the annual ALARA dose limit to any one individual in one calendar quarter requires a formal ALARA review and report. Any staff member, experimenter, or member of the RSC may call for an ALARA review of a proposed action. Only with the approval of the reactor supervisor and the endorsement of the reactor manager can the action proceed.

The ALARA review and report considers a number of topics, including features for external radiation control, contamination control, effluent control, operations, and operations planning. The report discusses how these topics affect personnel exposure and recommends specific actions. The reactor manager audits the implementation of the ALARA program semiannually as part of a general audit of the radiation protection program.

11.1.2.4 Radiation Monitoring and Surveillance

The structure of the radiation monitoring program ensures that all three categories of radiation sources—airborne, liquid, and solid—are detected and assessed in a timely manner. This is accomplished through the combination of surveillances and radiation monitoring equipment.

Radiation monitoring surveillance requirements of the radiation protection program include monthly surveillances (swipe surveys of reactor bay and control room), quarterly surveillances (source inventory and leak tests), and semiannual surveillances (environmental). Table 11.1 provides a summary of typical radiation monitoring equipment. (Because equipment is updated and replaced periodically, the equipment in the table should be considered as a representative rather than an exact listing.) The instrumentation is calibrated according to written procedures. Facility staff maintains calibration records, and the facility manager audits them annually; calibration stickers are affixed to the instruments.

Table 11.1 Radiation Monitoring and Surveillance Equipment

Item	Location	Function
Continuous air monitors (2)		
Effluent monitor	22-ft level	Measure radioiodine, noble gases, and particulates
Reactor room air monitor	12 ft level	Measure radioiodine in room air
Area radiation G-M monitors (3)	0-ft level 12-ft level 22-ft level	Measure gamma-ray exposure rates in accessible areas of the reactor room
Pool surface G-M monitor	22-ft level	Measure exposure rate at pool surface (N-16 and Ar-41)
Evacuation alarm G-M monitor	22-ft level	Measure high level gamma-ray exposure rate (5 R h^{-1})
Entrance G-M monitor	Control room	Measure exposure rate at reactor room entrance
Portable ion chamber meters (3)	0, 12, 22-ft levels	Measure gamma-ray exposure rate, sense beta particles
Portable G-M survey meters (3)	0, 12, 22-ft levels	Measure gamma-ray exposure rate, sense beta particles
Portable neutron survey meters (2)	0-ft & 22-ft levels	Measure ambient dose rate
Fixed alpha/beta counter	Room 11 Ward Hall	Wipe-test assay
Liquid scintillation spectrometer	University Radiation Safety Office	Counts liquid and wipe-test samples
Gamma-spectroscopy systems (3)	Room 11 Ward Hall	Gamma-ray assay
Direct reading pocket dosimeters	Control room	Personnel gamma/neutron dose
High volume air sampler	Control room	Emergency sample collection

11.1.2.5 Radiation Exposure Control and Dosimetry

Radiation exposure control depends on many factors, including design features such as shielding, ventilation, containment, entry control for high radiation areas, protective equipment, personnel exposure, estimates of annual radiation exposures for specific locations within the facility, and dosimetry. Facility design features and the radiation protection program combine to support the ALARA principle.

The RSC requires monitoring of workers and members of the public for radiation exposure. The licensee achieves these objectives through the following procedures:

- Authorization for access—Personnel who enter the control room or the reactor bay must hold authorization for unescorted access or be under the direct supervision of an escort who holds authorization for unescorted access.
- Access control during operation—When the reactor is operating, the licensed operator at the controls is responsible for controlling access to the control room and reactor bay. The 22-ft-level access has line-of-sight to the control room and has radiation monitoring equipment positioned directly over the pool surface. The operator at the controls is responsible for appropriately controlling access to the 22-ft level based on radiological conditions.
- Neutron dosimetry—If the potential exists for exposure of personnel to neutrons within the bay, personnel entering the bay must have neutron-sensitive individual monitoring.
- Exposure records for access during operation—Personnel who enter the reactor bay during reactor operation must have a record of accumulated dose measured by a gamma-sensitive individual monitoring device; at the discretion of the operator at the controls, a single individual monitoring device may be used for individual monitoring of no more than two people who agree to stand together in the bay.
- Exposure records for access during nonoperating conditions—Personnel who enter the reactor bay while the reactor is secured must have a record of accumulated dose either by measurement through individual monitoring or based on assessment of data from individual monitoring devices or surveys.
- Recordkeeping—Records are required to confirm that personnel exposures are less than 10 percent of applicable limits.
- Records of prior occupational exposures—The Office of Radiation Safety permanently maintains these records (NRC Form 4).
- Records of occupational personnel monitoring—The Office of Radiation Safety permanently maintains these records (NRC Form 5). Forms in use include the monthly report for the university as a whole, monthly summary report for the facility, and quarterly report on extremity exposures for the university as a whole.
- Records of doses to individual members of the public—The facility maintains permanent self-reading dosimeter records in a logbook. Results of measurements or calculations used to assess accidental releases of radioactive effluents to the environment are maintained on file permanently in the facility.

11.1.3 Contamination Control

The licensee controls potential contamination in the facility by using trained personnel following written procedures and by operating a monitoring program designed to detect contamination in a timely manner. The most likely contamination sites are sample ports at the rotary specimen rack and central thimble and at a sample-handling table for receiving irradiated samples. These sites are covered by removable absorbent paper pads with plastic backing and are periodically monitored. If contaminated, pads are removed and treated as solid radioactive waste. While working at potentially contaminated sites, workers wear protective gloves, protective clothing, and footwear. Workers must perform surveys to ensure that no contamination is present on their hands, clothing, shoes, or other areas before leaving the sites. If contamination is detected, a check of the exposed areas of the body and clothing is required, with monitoring control points established for this purpose. Materials, tools, and equipment are monitored for contamination before removal from contaminated areas or from restricted areas likely to be contaminated. Facility staff and visiting researchers are trained on the risks of contamination and on techniques for avoiding, limiting, and controlling contamination. The licensee provided sample locations for routine monitoring of surface and waterborne contamination. On a monthly basis, 100 square centimeter (cm^2) swipe surveys and 1 milliliter water samples are analyzed for contamination.

11.1.4 Environmental Monitoring

Environmental monitoring is required to ensure compliance with 10 CFR Part 20 and the TSs. Installed monitoring systems include area radiation monitors, airborne contamination monitors, and a radiation monitor at the pool surface. The RSC may require additional monitoring and has established the following requirements for contamination and radiation survey surveillances:

- Area radiation monitors—Monitors at the 22-ft level (and the 0-ft level if beam ports are open) are required for reactor operation. The monitor is calibrated in accordance with the radiation protection program and required by TS 4.3.2.
- Airborne contamination monitors—The facility uses two air monitoring systems; one is the continuous air monitor on the 12-ft level and the other (a gaseous effluent monitor) is in the exhaust plume path from the reactor pool to the bay exhaust system. TS 4.3.2 and TS 4.5.2, respectively, specify the surveillance for these monitors.
- Pool surface monitor—A radiation monitor is stationed directly over the pool surface. The monitor is calibrated in accordance with the radiation protection program and required by TS 4.3.2.
- Additional monitoring—The RSC imposes additional requirements through the radiation protection program.
- Contamination surveys—The Division of Public Safety, University Radiation Safety Office, maintains an independent contamination monitoring program under the Kansas State radioactive material license. As required by 10 CFR Part 20, contamination surveys are conducted to ensure compliance with regulations to evaluate the magnitude and extent of contamination levels, concentrations or quantities of radioactive materials, and potential radiological hazards.

- Radiation surveys—Monthly surveys are conducted for radiation levels with the reactor not in use. Additionally, semiannual environmental monitoring is conducted, involving measurement of both gamma and neutron dose rates within the facility operations boundary and at the site boundary with the reactor at full power.
- Monitoring for conditions requiring evacuation—An evacuation alarm (high radiation level) is required at the 22-ft level of the reactor bay. Response testing of the alarm is performed in accordance with facility procedures.

11.2 Radioactive Waste Management

The KSU reactor program generates very small quantities of radioactive waste because of the type of program carried out at the facility and a conscious effort to minimize waste volumes. The objective of the radioactive waste management program is to ensure that radioactive waste is minimized and that it is properly handled, stored, and disposed. The operator license training and requalification program is incorporated in the training associated with waste management functions.

11.2.1 Radioactive Waste Management Program

Liquid wastes are released through the sanitary sewerage system after filtration and assay for beta, gamma, and alpha activity. Solid wastes generated under the Kansas State license are transferred to the University Radiation Safety Office, Division of Public Safety, where they are combined with other solid radioactive wastes from the university, allowed to decay, and disposed of under the aegis of the State of Kansas. Solid wastes generated under the reactor license are generally allowed to decay, with subsequent disposal coordinated by the University Radiation Safety Office.

11.2.2 Radioactive Waste Controls

Radioactive solid waste is generally considered to be any item or substance no longer of use to the facility, which contains or is suspected to contain radioactivity above background levels. When possible, solid radioactive waste is initially segregated at the point of origin from items that are not considered waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the future need for the items and materials involved. Since Kansas is an Agreement State, radioactive materials generated for research and experiments under the Federal byproduct material license of the facility are transferred to the State of Kansas license. Solid wastes resulting from experiments and activities conducted under the State of Kansas license are then physically transferred to the University Radiation Safety Office, Division of Public Safety. Solid reactor waste is stored for decay until disposal.

Liquid wastes in the facility are held temporarily in storage tanks within the facility until pumped into the sanitary sewerage system. Liquid wastes are primarily condensate from the facility air conditioning system and are very slightly radioactive because of the presence of tritium from the primary coolant and bulk shield tank evaporation. To ensure compliance with 10 CFR Part 20, liquid wastes are assayed for alpha, beta, and gamma activity before release and are filtered to prevent particulates from being released with the liquids.

Although argon-41 is released from the facility, this release is not considered to be waste in the same sense as liquid and solid wastes. It is an effluent that is a routine occurrence in the operation of the facility. Sampling of argon-41 is done continuously by a continuous air monitor, and total argon-41 releases are limited by TS 3.5.3(2) to 30 curies per year to ensure that the requirements of 10 CFR 20.1101(d) are met.

11.3 Staff Evaluation

The radiation protection program receives appropriate support from administrators and managers. The KSU organizational structure and distribution of responsibilities provide clear oversight of the facility radiation protection program with organizational independence from operations. The program is reviewed and audited by the RSC, which meets the annual program review requirement in 10 CFR 20.1101(c). Training for radiation protection is an integral part of the operator licensing and requalification program. university radiation safety officer attendance is necessary for a quorum in the RSC, and any permissive action of the RSC requires both the affirmative vote of the university radiation safety officer and the majority vote of the members present.. This provides the university radiation safety officer decisive authority on reactor and radiation safety issues. (See Chapter 12 of this SER for more discussion on the RSC.) The NRC staff routinely inspects the reactor operations, including the radiation protection program. A staff review of those inspection reports has not indicated significant deficiencies in the KSU radiation protection program. The staff has reviewed the calculations provided by the licensee for gaseous effluent production, release, and the resultant doses to the personnel and the public. Administrative procedures are adequate to ensure that those exposures are within regulatory limits. There are adequate monitoring and surveillance controls in place to ensure that the principles of ALARA are achieved, and the licensee's surveys verify that conclusion.

11.4 Conclusions

On the basis of the review of information presented in the licensee's SAR annual reports and the NRC inspection reports, the staff concludes the following:

- The radiation protection program is acceptably implemented and the reactor staff adequately trained.
- The radiation monitoring and sampling equipment give reasonable assurance that radiation will be detected, monitored, and sampled in a manner consistent with regulatory requirements and the ALARA program.
- Effluent and environmental monitoring programs conducted by personnel are adequate to identify significant releases of radioactivity promptly and to predict and identify maximum exposures to individuals in the unrestricted area.
- There is reasonable assurance that personnel and procedures will continue to protect the health and safety of the public, the facility staff, and the environment from significant radiation exposures related to normal reactor operations for the duration of the license.
- Waste management activities at the reactor facility have been and are expected to continue to be conducted in a manner consistent with both 10 CFR Part 20 and ALARA principles.

- TSs, systems, and procedures limit the production of argon-41 and nitrogen-16 and control potential exposures to facility staff and the public. Computations by the licensee—and reviewed by the staff—show that the quantities of effluent gas (i.e., argon-41) released beyond the boundary of the reactor facility will result in potential doses to the public far below applicable 10 CFR Part 20 limits.

12. CONDUCT OF OPERATIONS

The conduct of operations involves the administrative aspects of facility operations and the facility's emergency, security, quality assurance, and reactor operator requalification plans. Administrative aspects of facility operations are the facility organization, training, operational review and audits, procedures, reporting, and recordkeeping.

12.1 Organization

KSU, a land-grant institution governed by a Board of Regents appointed by the Governor, holds the operating license (R-88, Docket Number 50-100) for the TRIGA reactor. TS 6.1 specifies the organizational structure and individual responsibilities of the management and staff. The chief executive officer of the university is the president. Figure 12.1 outlines the nuclear reactor facility's organization and management structure and identifies the president as the licensee for the facility.

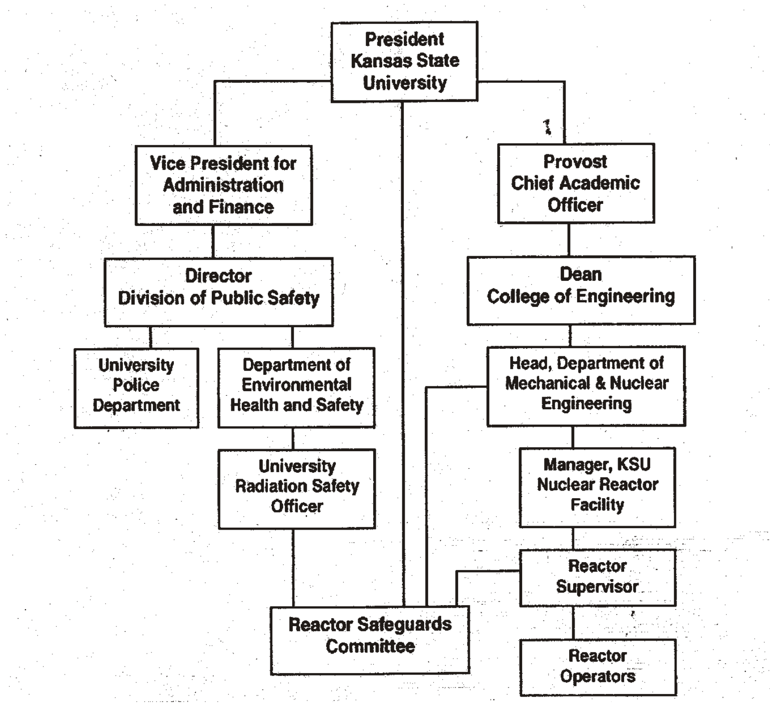

Figure 12.1 Organization and management structure for the KSU nuclear reactor facility

A university provost administers academic instruction and research for the university. Individual colleges manage these functions, with the College of Engineering responsible for the Department of Mechanical and Nuclear Engineering. The Department of Mechanical and Nuclear Engineering is directly responsible for managing the reactor.

The Department of Mechanical and Nuclear Engineering appoints a facility manager to directly manage the reactor. The facility manager delegates the responsibility to supervise and coordinate daily operations to a reactor supervisor (the facility manager may also perform these functions). The facility manager and reactor supervisor hold senior reactor operator licenses issued by the NRC. Additional licensed reactor operators (or senior reactor operators) perform operations and maintenance functions under the supervision of the reactor supervisor.

The vice president for administration and finance is responsible for safety at the university. KSU provides management and independent environment, safety, and health oversight functions for the university, implemented through the Division of Public Safety. Safety functions are administered by two sections of the Division of Public Safety—the University Police Department and the Department of Environmental Health and Safety.

The RSC (composed of members specified by TS 6.2(c) and appointed by the president of KSU, with the exception of certain ex officio members, including the head of the Department of Mechanical and Nuclear Engineering, the university radiation safety officer, and the facility manager) performs the review and audit of nuclear operations for the president. The committee meets at frequencies specified in TS 6.2(g). The committee reports to the president but also advises the facility manager and the head of the Department of Mechanical and Nuclear Engineering.

Responsibility for facility operations therefore extends from the government of the State of Kansas through the Board of Regents, the KSU president, the KSU provost, and the dean of the College of Engineering to the operating unit head of the Department of Mechanical and Nuclear Engineering and the reactor staff, including the facility manager, the reactor supervisor, and reactor operators.

12.1.1 Structure

As indicated in Figure 12.1 above, the KSU president is the licensee for the nuclear reactor facility. The reactor is under the direct control of the facility manager, who reports through the academic administrative structure to the president.

The vice president for administration and finance provides environment, safety, and health oversight and expertise, independent of facility line management. In addition to the reactor license, KSU administers a broad radioactive material license. The university radiation safety officer is the university's broad licensee and manages the radioactive material (i.e., byproduct and nonreactor SNM) inventory and the university's radiation safety program for ionizing radiation.

The URSC (reporting to the vice president for administration and finance) maintains oversight and control of radiation protection functions for the university. The committee prepares and distributes radiological controls for possession and use of radioactive materials in the radiation protection program. The URSC has authorized the facility manager to possess and transfer radioactive material under the State broad license.

University requirements and regulations under 20 CFR 20.1101 are combined in a comprehensive reactor radiation protection program. In accordance with this program, the reactor staff fulfills most routine radiation protection functions at the reactor, with review and oversight by the university radiation safety officer. The university radiation safety officer manages the radiation worker exposure monitoring system (and distribution of related records), as well as radioactive material inventory control.

12.1.2 Responsibility

Operating staff responsibilities are categorized below:

12.1.2.1 Reactor Operations Line Management

- President—As chief executive officer of the university, the president is responsible for safe operation of the reactor, protection of the public health and safety, and protection of the environment. The line of authority and responsibility for reactor operations extends through the provost and dean of the College of Engineering to the head of the Department of Mechanical and Nuclear Engineering. Environment, safety, and health compliance management and independent oversight functions are distributed through the vice president of administration and finance to the manager of the Division of Public Safety, Department of Environmental Health and Safety.

- Head of the Department of Mechanical and Nuclear Engineering—The department head is the appointment authority for the facility manager, reactor supervisor, and all reactor operators and senior reactor operators. The department head is responsible for providing resources required for safe operations of the reactor facility. The department head is the chair of the URSC, which reports to the president of the university and is responsible for approval of all plans and procedures for reactor operations, operational audits, and recordkeeping.

- Nuclear Reactor Facility Manager—The facility manager, who may also serve as reactor supervisor, is directly responsible to the head of the Department of Mechanical and Nuclear Engineering for all aspects of facility operation. The facility manager may hold academic and research responsibilities beyond those associated with the reactor facility. The facility manager can delegate responsibility for operation and use of the reactor to the reactor supervisor.

- Reactor Supervisor—In the operation of the reactor, the reactor supervisor has such duties as may be delegated by the facility manager. The reactor supervisor's nominal duties include reactor scheduling and responsibility for all records on reactor operation as are required by appropriate Federal licenses and regulations, laws and regulations of the State of Kansas, and regulations of KSU, including the KSU TRIGA Mark II reactor operations manual.

  The reactor supervisor is responsible for ensuring that the reactor is operated only while a properly qualified and licensed reactor operator is present. The reactor supervisor maintains the reactor operations manual, which includes prescribed operating procedures for all routine modes of reactor operation, loading and unloading procedures, startup procedures, maintenance schedule, testing procedures, operational references, and other appropriate information as determined by the reactor supervisor and the facility manager. The reactor supervisor is responsible for ensuring that the reactor is operated in strict accordance with the reactor operations manual and the facility license.

- Reactor Operator—Operators (reactor operators and senior reactor operators) report directly to the reactor supervisor and/or the facility manager. Operators are responsible for knowing the status and condition of the facility and for ensuring that both facility personnel and the general public are protected from exposure to radiation consistent with approved policies and procedures. Operators are responsible for operation of the reactor in accordance with the TSs, operating procedures, and experiment procedures.

  Operators are responsible for ensuring that only authorized personnel (trainees for senior operator and operator positions, as well as students enrolled in academic courses making use of the reactor, as permitted by 10 CFR Part 55) manipulate controls under the direction of a licensed operator or senior operator. Operation includes startup, shutdown, routine I&C checkout, recordkeeping, routine maintenance, and such other duties as may be described in the operations manual and/or as directed by the reactor supervisor.

  During fuel movement, a reactor operator must be at the reactor control console, and a senior operator must be inside the reactor bay directing fuel operations.

12.1.2.2 Environment, Safety, and Health Staff

- Vice President for Administration and Finance—The vice president for administration and finance is responsible for safety at the university, as discussed in Section 12.1.

- University Police Department—The University Police Department is responsible for law enforcement functions and institutional physical security. The University Police Department is the primary interface for external agencies during response to events at the reactor facility.

- Manager, Department of Environmental Health and Safety—The Department of Environmental Health and Safety is responsible for compliance issues related to environment, safety, and health. To meet these requirements, the department has positions for a university radiation safety officer, occupational safety manager, hazardous material manager, and safety and security officer.

- University Radiation Safety Officer—The university radiation safety officer reports to the manager of the Department of Environmental Health and Safety. The university radiation safety officer, or an authorized representative, shall be available (upon due notice) for advice and consultation on radiation surveys and radiation safety in connection with isotope production and radiation streaming problems that may result from reactor operation or experimentation. The university radiation safety officer serves ex officio as a member of the URSC. The university radiation safety officer also serves ex officio as a member of the RSC, with any action (i.e., concerning potential radiation exposure or radioactive effluents) of the committee requiring university radiation safety officer approval.

12.1.2.3 Principal Advisory and Oversight Committees

- Reactor Safeguards Committee—The RSC is composed of members appointed by the university president, upon the recommendation of the chair of the committee. The TSs explicitly define the composition and membership qualifications of the committee. The RSC is responsible for approval of all plans and procedures for reactor operations, audit of reactor operations, and recordkeeping.

- University Radiation Safety Committee—The URSC is an advisory committee for the vice president for administration and finance and the university radiation safety officer. Under authority of the vice president for administration and finance, the URSC authorizes conditions for use of radioactive material and authorizes users (by name) to acquire and possess radioactive materials. The reactor radiation protection program incorporates requirements of the URSC.

12.2 Staffing

Whenever the reactor is not secured, it is under the direction of an NRC-licensed senior operator who is designated as reactor supervisor. The reactor supervisor shall be on call, within a 20-min travel time to the facility, and cognizant of reactor operations.

Whenever the reactor is not secured, an NRC-licensed reactor operator (or senior reactor operator) who meets the requirements of the operator requalification program is at the reactor control console and directly responsible for control manipulations. A call list of reactor facility personnel, management, and radiation safety personnel is available in the reactor control room for use by the reactor operator at the controls.

During fuel movement, a reactor operator is at the reactor operating console, and a senior reactor operator is inside the reactor bay directing fuel operations.

Only the reactor operator at the controls or personnel authorized by, and under the direct supervision of, the reactor operator at the controls shall manipulate the controls. Whenever the reactor is not secured, operation of equipment that has the potential to affect reactivity or power level shall be manipulated only with the knowledge and consent of the reactor operator at the controls. The reactor operator at the controls may authorize students enrolled in an academic course making use of the reactor or persons training to qualify for an operator license in accordance with the approved reactor operator requalification plan to manipulate reactivity controls.

12.3 Selection and Training of Personnel

The reactor facility maintains a training and selection program to prepare trainees for examination by the NRC in pursuit of operator or senior operator licenses. Section 12.12.2.1 of this SER describes medical qualification for the operator license program.

Personnel qualified for unescorted access or under direct supervision of personnel qualified for unescorted access are permitted access to the reactor bay. Unescorted access is granted to personnel who meet the security requirements and are trained and examined in the knowledge and skills necessary to control personnel exposure to radiation associated with the operation of the reactor. Training includes the KSU radiological protection program and the reactor emergency plan. The radiological protection program requires specific instruction in the risks of occupational exposure, the risks of prenatal exposure, the provisions of 10 CFR Part 19 and 10 CFR Part 20, and a tour of the reactor facility.

12.4 Radiation Safety

The licensee developed the radiation protection program for the KSU TRIGA to meet the requirements of 10 CFR Part 20. Development of the radiation protection program used the guidance of ANSI/ANS-15.11 and various NRC regulatory guides. The program also addresses radioactive materials regulated by the State of Kansas (an Agreement State) under License 38-C011-01. The goal of the program is to limit radiation exposures and radioactivity releases to a level that is ALARA while allowing operation of the facility to perform its mission of education and research. The program is executed in coordination with the KSU Department of Environmental Health and Safety, Division of Public Safety, University Radiation Safety Office. The RSC for the reactor facility has reviewed and approved the program. The URSC has independently reviewed the program.

The reactor staff fulfills most of the functions described in the program. The university radiation safety officer maintains oversight of the facility by direct observation and review of specific activities and by acting as a member of the RSC and the URSC. The university radiation safety officer approval is required for RSC approval of any item under review. The university radiation safety officer has the authority to stop, in the interest of safety, any experiment involving radiation on the KSU campus. All campus radioactive materials users must eliminate any known unsafe practice or report the issue to the university radiation safety officer.

Chapter 11 of this SER describes the radiation safety program in detail.

12.5 Review and Audit Activities

Review and audit activities are oversight functions essential to the safe operation of the facility and the protection of the public health and safety. The TSs, emergency plan, radiation protection program, and the reactor administrative plan require a set of internal surveillances, reviews, and audits conducted by the reactor staff and the facility manager, culminating in a semiannual management audit of operations.

The RSC holds oversight responsibility and authority (TS 6.2(a)). Oversight of the facility manager’s performance and review of management audits are the responsibility of the RSC and are evaluated formally at periodic intervals (required at least once each year by TS 6.2(b).7). In addition to periodic, scheduled reviews, the RSC is available to conduct reviews upon request (TS 6.2(g)). The RSC reviews and approves administrative controls, such as program plans and procedures, before their implementation (TS 6.2(b)).

The university radiation safety officer conducts periodic laboratory safety audits and regularly participates in radiation surveillances to review the conduct of radiation surveys by the facility staff.

12.5.1 Reactor Safeguards Committee Composition and Qualifications

With the exception of ex officio members, RSC members are appointed by the university president, upon the recommendation of the committee chair. Composition and membership qualifications of the committee (as specified in the TS 6.2(c)) provide expertise to evaluate reactor management, facilities, experimental programs, operating and experiment procedures, and radiological hazards.

The head of the Department of Mechanical and Nuclear Engineering, or a designated deputy, is an ex officio member, serves as the committee chair (TS 6.2(c).6) and has the authority and responsibility to allocate resources that ensure safe reactor operations. The university radiation safety officer is an ex officio committee member with veto power over permissive committee decisions (TS 6.2(c) and TS 6.2(f)). The facility manager is also an ex officio member of the committee (the reactor supervisor may attend if the facility manager is not available, but is nonvoting). These ex officio committee members must be in attendance to meet the quorum requirements (TS 6.2(e)).

At least one committee member shall be proficient in reactor and nuclear science or engineering (TS 6.2(c)1). At least one committee member shall be proficient in chemistry, geology, or chemical engineering (TS 6.2(c)2), and at least one committee member shall be proficient in the biological effects of radiation (TS 6.2(c)3). One individual may have and represent expertise in more than one area, but the committee, described below, shall consist of at least seven members, five of which shall be faculty members (TS 6.2(c)).

12.5.1.1 Charter and Rules

The committee is required to meet at least semiannually. Sections 12.2.3 and 12.2.4 of this SER described the specific review and audit activities prescribed for these meetings. The chair of the committee, or a designee, may call additional meetings. At the discretion of the chair or the chair's designee, the committee may be polled in lieu of a meeting, and such a poll will constitute committee action subject to the same requirements as those for an actual meeting.

Any permissive action of the committee requires an affirmative vote of the university radiation safety officer as well as a majority of the members present. A quorum consists of not less than a majority of the full committee, including all ex officio members.

Minutes of meetings are distributed to the dean of the College of Engineering, the provost, and the university president. Recorded affirmative votes on proposed new or revised experiments or procedures shall indicate that the committee has determined that proposed actions do not, involve changes in the facility design, or TSs that require prior NRC approval, and can be taken without endangering the health and safety of workers or the public or constituting a significant hazard to the integrity of the reactor core.

12.5.1.2 Review Function

The responsibilities of the RSC shall include but are not limited to the following:

- review and approval of rules, procedures, and proposed TSs (TS 6.2(b)1)
- review and approval of all proposed changes in the facility that could have a significant effect on safety and of all proposed changes in rules, procedures, and TSs, in accordance with procedures in Section 6.3 (TS 6.2(b)2
- review and approval of experiments using the reactor in accordance with procedures and criteria in Section 6.4 (TS 6.2(b)3)

- determination of whether changes in the facility as described in the SAR (as updated), changes in the procedures as described in the final SAR (as updated), and the conduct of tests or experiments not described in the SAR (as updated) may be accomplished in accordance with 10 CFR 50.59 without obtaining prior NRC approval via license amendment pursuant to 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit" (TS 6.2(b)4)
- review of abnormal performance of plant equipment and operating anomalies (TS 6.2(b)5)
- review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50 (TS 6.2(b)6)
- inspection of the facility, review of safety measures, and audit of operations at a frequency of not less than once a year, including operation and operations records of the facility (TS 6.2(b)7)
- requalification of the facility manager and/or the reactor supervisor (TS 6.2(b)8)
- review of container failures in which released materials have the potential for damaging reactor fuel or structural components, including the results of physical inspection, evaluation of consequences, and need for corrective actions (TS 6.2(b)9)

12.5.1.3 Audit Function

Not less than once a year, the RSC inspects the facility, reviews safety measures, and audits operations, including operations records (TS 6.2(b)7). The inspections include but are not limited to the following:

- inspection of reactor operating records
- inspection of maintenance activity records
- inspection of health physics records
- review of the effectiveness of training and requalification activities
- review of radiological surveillance records
- inspection of the reactor facility

12.6 Procedures

Before initiating any of the activities listed below, the RSC, as well as the facility manager, approve written procedures. Under conditions specified by the RSC, the facility manager may make changes in procedures or experiments subject to validation by the RSC (TS 6.3(b)). A periodic review of procedures will be performed and documented in a timely manner to ensure that they are current. Procedures shall be adequate to ensure the safety of the reactor, persons in the facility, and the public, but will not preclude the use of independent judgment and action, should the situation require. Specific activities listed in TS 6.3(a) require procedures; however, administrative requirements to determine the need for procedures are in TS 6.3(c). The following actions will typically require reviewed written procedures:

12.6.1 Reactor Operations

- startup, operation, and shutdown of the reactor
- fuel loading, unloading, and movement within the reactor
- control rod removal or replacement
- routine maintenance, testing, and calibration of control rod drives and other systems that could affect reactor safety
- changes to administrative controls for operations, maintenance, conduct of experiments, and conduct of facility tours
- implementation of procedures for the emergency plan or physical security plan

12.6.2 Health Physics

- testing and calibration of area radiation monitors, facility air monitors, and fixed and portable radiological surveillance instruments
- conduct of radiological surveillance measurements
- release of contaminated materials to the University Radiation Safety Office, Division of Public Safety
- changes to accountability for SNMs

12.7 Required Actions

The two categories of required actions are violations of facility SLs and reportable events.

12.7.1 Violations of Facility Safety Limits

If a safety limit is not met, the actions specified in TS 6.8 and summarized below shall occur:

- The reactor shall be shut down and reactor operations secured.
- An immediate report shall be made to the chair of the RSC
- The licensee shall provide notifications and reports to the NRC as required in TS 6.11 (summarized in Section 12.8 below).
- Operations shall not resume until authorized by the NRC.

12.7.2 Occurrences Reportable to the U.S. Nuclear Regulatory Commission

In addition to the reporting requirements, if a reportable event, as defined in TS 6.9(a) occurs, the following actions, required by TS 6.9(b), shall take place:

- The reactor supervisor shall be notified.
- If a reactor shutdown is required, only the facility manager can authorize the resumption of normal operations.
- A report shall be prepared as specified in TS 6.9(b)2 and submitted to the RSC for review.
- A report shall be submitted as specified in TS 6.11.

12.8 Reports to the U.S. Nuclear Regulatory Commission

All reports shall address (to the extent known or possible) the impact of the event on the safety and health of the public, workers, and the facility (e.g., whether or not the event resulted in property damage, personal injury, or exposure). Reports (including initial reports, to the extent possible) shall describe, analyze, and evaluate safety implications and outline the corrective measures taken or planned to prevent recurrence of the event. (See TS 6.8 c) and TS 6.9 b)2.)

12.8.1 Immediate Notification

The licensee shall make a report within 24 hrs of discovery of any accidental release of radioactivity above permissible limits in unrestricted areas, violation of an safety limit, or reportable occurrence by (1) telephone, and (2) facsimile, telegraph, or email to the Region III administrator, in accordance with 10 CFR 50.36(7)(ii), and to the NRC Operations Center in accordance with TS 6.11(a).

12.8.2 10-Day Notification

The licensee, as required by TS 6.11(b), shall submit a written report within 10 days to the NRC for any accidental release of radioactivity above permissible limits in unrestricted areas, any violation of an safety limit, or any reportable occurrence.

The licensee shall prepare a report of an safety limit violation or a reportable occurrence describing the analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence (TS 6.8(a) and TS 6.9(b)).

In the event of any accidental release of radioactivity above permissible limits in unrestricted areas, the licensee shall prepare a report that describes, analyzes, and evaluates the safety implications and outlines the corrective measures taken or planned to prevent recurrence (TS 6.11(b).1).

12.8.3 30-Day Notification

The licensee shall make a written report, as specified in TS 6.11(d), within 30 days to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, for any of the following:

- change in the chair of the Department of Mechanical and Nuclear Engineering or a change in reactor manager (TS 6.11(c).3)
- significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during reactor operation (TS 6.11(c).1)
- significant change in the transient or accident analysis as described in the SAR (TS 6.11(c).2)

12.8.4 Other Reports

The licensee must submit a written report to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, within 60 days after criticality of the reactor resulting from receipt of a new facility license or an amendment to the license authorizing an increase in power level or the installation of a new core. The report must describe the measured values of the operating conditions or characteristics of the reactor under the new conditions. TS 6.11(d) specifies this reporting requirement. This TS requires a report after the KSU increases power to the new license limit evaluated by this SER and authorized by the renewed license.

Within 60 days after completion of the first calendar year of operation, and at intervals not to exceed 12 months thereafter, the licensee must submit a routine report to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555. This report shall provide the following information:

- narrative summary of reactor operating experience
- energy generated by the reactor (in megawatt-hours)
- unscheduled shutdowns, including corrective action taken to prevent recurrence
- major preventive and corrective maintenance with safety significance
- summary of each change to the reactor facility or procedures, tests, and experiments, in accordance with the requirements of 10 CFR 50.59
- summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of KSU, determined at or before the point of release or discharge
- description of any environmental surveys performed outside the facility
- summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposure
- a brief summary of the results of radiation and contamination surveys performed within the facility

TS 6.11(e) specifies this reporting requirement.

12.9 Record Retention

There are three categories of record retention. General operating records (as noted) must be kept for 5 yrs. Records relating to requalification are kept for the duration of the individual's employment or for a complete training cycle. Records related to radiation (releases or exposure) and as-built facility drawings are kept for the life of the facility.

12.9.1 5-Year Retention Schedule

In addition to the requirements of 10 CFR Part 20 and 10 CFR Part 50, the licensee must prepare records and logs for the following items and retain them for at least 5 yrs:

- normal facility operation, including power levels
- principal maintenance activities
- reportable occurrences

- equipment and component surveillance activities
- experiments performed with the reactor
- all emergency reactor scrams, including reasons for emergency shutdowns

TS 6.10(a) specifies these reporting requirements.

12.9.2 Certification Cycle

The licensee must maintain records of retraining and requalification of certified operations personnel at all times the individual is employed or until the certification is renewed. This meets the requirements of 10 CFR 55.59(c)(5)(i).

12.9.3 Life-of-the-Facility Records

The licensee must maintain the following records for the life of the facility:

- gaseous and liquid radioactive effluents released to the environs
- offsite environmental monitoring surveys required by the TSs
- fuel inventories and transfers
- facility radiation and contamination surveys
- radiation exposures for all personnel monitored
- corrected and as-built facility drawings

TS 6.10(b) specifies these reporting requirements.

12.10 Emergency Planning

The licensee has established and followed an emergency plan in accordance with NRC regulations. In addition, the licensee has established emergency procedures, which the RSC has reviewed and approved, to cover all foreseeable emergency conditions potentially hazardous to facility staff or to the public, including, but not limited to, conditions involving an uncontrolled reactor excursion or an uncontrolled release of radioactivity. The licensee submitted and supplemented a revised emergency plan as part of the license renewal. The staff reviewed this plan for compliance with applicable portions of Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” to 10 CFR Part 50 and Regulatory Guide 0849 issued in 1983. By separate letter (ADAMS Accession No. ML062700235), the staff approved the KSU emergency plan in August 2006.

12.11 Security Planning

In accordance with NRC regulations, because the reactor license authorizes possession of SNM of low strategic significance, the licensee must maintain security in accordance with the provisions of 10 CFR 73.67(f). NRC inspectors have inspected the KSU security procedures and their implementation and have determined that KSU meets the requirements of this regulation. The NRC inspection program will continue to verify that the requirements are met.

12.12 <u>Operator Training and Requalification</u>

The KSU reactor facility maintains a training and selection program to prepare trainees for examination by the NRC in pursuit of operator licenses. Examinations are based on those of the NRC and include both written and operating tests in accordance with the requirements of 10 CFR Part 55.

12.12.1 Operator Training

In preparation for their NRC examinations, trainees must satisfactorily complete study in the following areas:

- theory and operating principles
- operating characteristics
- I&C
- protection systems
- operating and emergency procedures
- radiation control and safety
- TSs
- Title 10 of the *Code of Federal Regulations*

12.12.2 Requalification Program

The licensee submitted a requalification plan as part of the license renewal, which meets the requirements of 10 CFR 55.59. This plan was reviewed and, with this SER, is reapproved by the NRC staff. The requalification program follows a 2-yr cycle as of January 1, 1974. The program provides for operator medical certification, on-the-job-training elements and proficiency, lectures, examinations, and records. The program identifies periodic and special requirements associated with medical certification, maintaining operational proficiency, operator examinations, training lectures, and records.

12.12.2.1 Medical Certification

The NRC licenses operators based on physician evaluation and facility management certification that the licensee's medical condition and general health will not adversely affect the performance of assigned operator job duties or cause operation errors endangering the public health and safety.

The facility manager has primary responsibility to ensure that medically qualified personnel are on duty. Medical qualification of the facility manager, if licensed, is the responsibility of the chair of the RSC.

The requalification program identifies requirements for licensed operators to maintain certification that they are medically qualified to operate the reactor, including annual reexamination and notification of significant health changes, should they occur.

12.12.2.2 Proficiency

During each 2-yr cycle, each licensed operator maintains proficiency in reactivity manipulations by performing tasks that demonstrate skill with reactivity control systems, as specified in the requalification program. The licensee documents and distributes information on changes in facility design, operating procedures, facility license, and abnormal and emergency procedures to ensure that all licensed operators are cognizant of facility conditions and requirements.

12.12.2.3 Examinations

The requalification program specifies two sets of annual examinations—written exams and operating exams. The program specifies that examinations should be based on a representative sample of questions covering areas in sufficient depth to evaluate operator understanding and capabilities. The program specifies that examinations should evaluate the knowledge, skills, and abilities required to perform as a reactor operator/senior reactor operator, as appropriate.

Requirements for attending formal training lectures are based on the results of annual written examinations administered to all licensed personnel, according to criteria specified in the requalification program. The facility manager or the reactor supervisor prepares and grades the examinations.

Requirements for additional training are based on the results of annual operational examinations covering normal, abnormal, and emergency operating procedures, according to criteria specified in the requalification program. The facility manager or the reactor supervisor prepares and administers the operational examinations, as specified in the requalification program.

12.12.2.4 Lectures

The requalification program provides guidance for preparing training material based on how the material relates to job performance.

12.12.2.5 Records

Records demonstrating successful participation in the requalification program are maintained as specified in Section 12.6 of this SER. These include operator training record folders, records of RSC reviews, operating logs, annual training records, and operator biennial requalification records. Operator biennial requalification records are retained for one completed cycle.

12.13 <u>Staff Evaluation</u>

The licensee has described administrative aspects of facility operations and has presented an organizational structure that contains all organizational relationships important to safety, including a review and audit function and a radiation safety function. The organizational structure clearly shows the lines of responsibility for safe operation of the facility and for the protection of the health and safety of the public, the facility staff, and the environment. The ultimate responsibility lies with the president of KSU. In addition, the staff reviewed past annual reports and inspection reports that have demonstrated the licensee's ability to operate the facility safely.

12.14 Conclusions

On the basis of the above discussions, the staff concludes that the licensee has sufficient experience, 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 the licensee's proposed organization, training, operational review and audits, procedures, required actions, and records and reports against the guidance given in ANSI/ANS-15.1-1990, "American National Standard for the Development of Technical Specifications for Research Reactors," This standard is adopted by the NRC staff for the conduct of operations by including it as acceptable review criteria in the "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria," NUREG-1537 Part 2. The licensee's proposed conduct of operations in these areas is consistent with the guidance of the standard and is, therefore, acceptable to the staff.

Training for staff members will be conducted at an acceptable level. The licensee has submitted a reactor operator requalification plan containing information that meets the requirements of 10 CFR Part 55. The plan gives reasonable assurance that the reactor facility will be operated by competent operators and is acceptable.

As discussed in section 12.11 of this SER. NRC inspectors have inspected the KSU security procedures and their implementation and will continue to verify that the requirements are met with periodic inspections. The staff concludes that there is reasonable assurance that the licensee will maintain physical security in accordance with the requirements of 10 CFR 73.67(f). The staff finds physical security planning acceptable.

The staff concludes that the licensee's emergency plan meets the requirements of the regulations in Appendix E to 10 CFR Part 50, and is, therefore, acceptable.

13. ACCIDENT ANALYSIS

This chapter provides information and analysis to demonstrate that the health and safety of the public and facility staff are protected in the event of an accident at the reactor facility. The licensee analyzed anticipated potential reactor transients and other potential and hypothetical accidents and the potential consequences of such events on the reactor fuel and on the radiological health and safety of the public. The staff evaluated the licensee's assumptions, analytic methods, and results.

13.1 <u>Accident-Initiating Events and Scenarios</u>

Chapter 4 of this SER discusses the reactor physics and thermal-hydraulic conditions associated with the normal long-term operation of the KSU TRIGA reactor at a power level of 1250 kW(t). The analyses of accidents in this section further evaluate the consequences of off-normal behavior. These analyses assume steady-state operation at 1250 kW(t) and pulsing operation with a $3.00 reactivity insertion with an estimated peak power of 1340 MW.

The maximum allowable fuel temperature imposes limits for manual, automatic, and pulse modes of operation. These limits stem from the out-gassing of hydrogen from uranium-zirconium hydride fuel and the subsequent stress produced in the cladding material of the fuel elements. The strength of the stainless-steel cladding as a function of its temperature and the hydrogen pressure as a function of fuel temperature establishes an upper limit on the fuel temperature without fuel failure.

NUREG-1537, "Guidelines for Preparing and Reviewing Application for the Licensing of Non-Power Reactors, Format and Content," issued in 1996, and NUREG-2387, "Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors," issued in 1982, suggest credible and MHAs for TRIGA reactors. The licensee has selected the following three scenarios, which are typical for TRIGA reactor accident analysis:

(1) loss of coolant

(2) insertion of excess reactivity

(3) fuel cladding failure, which is the MHA

The licensee analyzed the loss-of-coolant accident to demonstrate that maximum fuel element temperature does not exceed acceptable limits. It also analyzed the dose from scattered radiation from the uncovered core.

Insertion of excess reactivity is the inadvertent rapid insertion (pulse insertion) of positive reactivity which, if large enough, could produce a transient resulting in fuel overheating and a possible breach of cladding integrity. The inherent prompt negative temperature response characteristic of TRIGA fuel is a safety factor for this type of postulated accident.

The MHA is a hypothetical accident that bounds the consequences of all credible accidents. The MHA for the KSU reactor has been defined as a cladding rupture in air of a single fuel element operated at peak power density for an extended time with no decay, followed by instantaneous release of airborne and gaseous fission products from the fuel-clad gap into the reactor room and subsequently into the unrestricted area. This accident has been analyzed to determine the limiting or bounding potential radiation doses to the reactor staff and to the general public in unrestricted areas.

13.2 Accident Analysis and Determination of Consequences

13.2.1 Notation and Fuel Properties

As a basis for the engineering calculations performed in support of the accident scenarios, Tables 13.1 shows the neutronic properties of the TRIGA Mark II fuel elements. Table 13.2 summarizes the assumptions and design-basis values used in the analyses.

Table 13.1 Neutronic Properties of TRIGA Mark III $ZrH_{1.6}$ Fuel Elements

Property	Symbol	Value
Effective delayed neutron fraction	β	0.007
Effective neutron lifetime	l	43 μsec
Temperature coefficient of reactivity	α	-0.000115 K^{-1}

Table 13.2 Core Conditions Basis for Calculations

Steady-state maximum power, P_o	1250 kW(t)
Fuel mass per element	
Heat capacity per element at T (°C)	805.0 + 1.646T(JK^{-1})
Minimum number of fuel elements, N	
Core radial peaking factor	2
Axial peaking factor	π/2
Excess reactivity	$4.00 (2.8% Δk/k)
Maximum pulsing reactivity insertion	$3.00 (2.1% Δk/k)
Excess reactivity at maximum power	$1.16 (0.81% Δk/k)
Fuel average temperature at maximum power	285 °C

The calculations supporting the accident scenarios described below use the parameters in Tables 13.1 and 13.2 and other parameters provided by the licensee..

13.2.2 Loss of Reactor Coolant

Although the total loss of reactor coolant is considered extremely improbable, the licensee analyzed the event to determine the resulting maximum fuel temperature. The purpose of the analysis was to demonstrate that the maximum fuel temperature, under the most conservative assumptions, is well below the accident safety limit for TRIGA fuel. That safety limit is (1) 1150 °C when clad temperature is below 500 °C, or (2) 950 °C when clad temperature can equal the fuel temperature (the latter is the likely case if the element is in air rather than in water). The loss-of-coolant accident is analyzed under the following assumptions:

- The reactor is assumed to have been operating for an infinite time at full power (i.e., 1250 kW(t)), at the time coolant is lost. This is very conservative considering the typical operating schedule for KSU.
- Coolant loss is instantaneous. This is very conservative when credible water loss scenarios are evaluated. In addition, many of the more credible scenarios do not lead to complete loss of water. For example, if a beam tube were to rupture, the core would not be completely uncovered, and the peak fuel temperatures reached would be significantly lower than those resulting from the instantaneous and complete loss-of-coolant scenario.
- Reactor scram occurs simultaneously with coolant loss.
- Decay heat is from fission product gamma and x-rays, beta particles, and electrons. Effects of delayed neutrons are neglected.
- Thermal power is distributed among the proposed number of fuel elements, with a radial peak-to-average ratio of 2.0. In individual elements, thermal power is distributed axially according to a sinusoidal function.
- Cladding and gap resistance are assumed to be negligible, that is, cladding temperature is assumed to be equal to the temperature at the outside surface of the fuel. (Note: Because of the thermal properties and thickness of the cladding and gap, the temperature drop across the cladding is only about 0.5 °C. Hence, this assumption is not necessarily conservative, but merely simplifying.)
- Cooling of the fuel occurs via natural convection to air at an inlet temperature equal to 300 °K (equals core inlet temperature 27 °C). Radiative cooling and conduction to the grid plates are neglected.
- Heat transfer in the fuel is one-dimensional (axial conduction is neglected).
- Heat transfer in the fuel is treated as steady state (at any one instant, heat transfer is described by steady-state conduction and convection equations).

These assumptions are very conservative because any one of them can cause a maximum fuel temperature rise.

The following is the licensee's analysis as presented in its SAR. The staff has verified these calculations.

Decay Heat Power

To calculate a fuel temperature increase following the loss of coolant, it is first necessary to calculate the time-dependent power of the heat source (i.e., the decay heat of the fuel). This is described by a decay heat function, R(t), which has been evaluated by the CINDER computer code (England et al., 1976). The function R(t) is the ratio of the decay power at time t after shutdown, $P_d(t)$, to the power before shutdown, P_o:

$R(t) = P_d(t)/P_o$, or $P_d(t) = P_oR(t)$ (13-1)

The worst possible case occurs in the fuel element that generates twice the power as the average element (Table 13.2). The average fuel element generates a power equal to the total power divided by the number of fuel elements, N, and so the time-dependent power of the worst-case fuel element is as follows:

$P_d(t) = 2P_oR(t)/N$ (13-2)

At time t = 0, R(0) has its maximum value of 0.0526, and the power of the worst-case fuel element, $P_d(0)$, has a maximum value of 1.584 kW(t).

Calculation of Maximum Air Temperature

The decay heat in the fuel transfers to air that is flowing up through the core, and so it is necessary to calculate the increase in air temperature resulting from the heat transfer from the fuel element. Two opposing processes govern the flow of air through the core—the buoyancy that causes the air to rise and the friction loss that opposes its motion. Each of these processes is independently calculated from first principles.

The temperature of the air as it exits the top of the core (T_o) is equal to the temperature of the air at the core inlet (T_i) plus the temperature increase:

$T_o(t) = T_i(t) + P_d/wc$ (13-3)

Where: w = mass flow rate of the air (kg/s) and

c = the specific heat capacity of the air (1030 J/kg°K).

As the heated air rises up through the core, the buoyancy pressure drop of the rising air is given by the following:

$\Delta p_b(t) = 0.190\ R(t)/w$ (13-4)

The constant (0.190) takes into account the travel length of airflow, the density of air, the air inlet temperature (300°K), and the effective cross-sectional area of flow.

Opposing the rising air is the frictional pressure drop, given by the following:

$\Delta p_f = 1780\ w$ (13-5)

Where: the constant takes into account flow characteristics such as Reynold's number, viscosity, and area of flow. Equating equations 13.4 and 13.5 leads to the following:

$w = 0.0103\ [R(t)]^{1/2}$ (13-6)

Using equations 13-2, 13-3, and 13-6, the temperature rise of the air as it passes up through the core is given by the following:

$T_o(t) - T_i(t) = 1140\,[R(t)]^{1/2}$ (13-7)

The maximum air temperature increase occurs at time t = 0, when R(t) equals 0.0526. Therefore, the maximum increase is 261 °K (above the 300 °K inlet temperature).

Fuel and Cladding Temperatures

After calculating the decay heat function and the temperature rise of the air rising up through the core, fuel and cladding temperatures can be computed. The maximum heat flux (watts per square meter (W/m^2)) from the hottest fuel element is given by the following:

$Q_{max} = 4.235 \times 10^5\, R(t)$

With a design power of 1250 kW(t), the constant takes into account a radial power-peaking factor of 2, an axial power-peaking factor of π/2, and fuel surface area. The local value of the heat flux is given by the following:

$Q(z) = Q_{max} \sin(\pi z/L_f)$ (13-8)

Where: z is the distance along the fuel element measured from the inlet, with the assumed sinusoidal axial variation of heat flux.

This results in an air temperature along the fuel element given by the following:

$T_{air}(z) = T_i + 0.3075\, R(t)[1 - \cos(\pi z/L_f)]$ (13-9)

The cladding surface temperature that results is given by the following:

$T_{clad}(z) = T_{air}(z) + Q(z)/h$ (13-10)

Where: $h = 5240\ W/m^2\ °K$ is the heat transfer coefficient.

Finally, the fuel element centerline temperature is given by the following:

$T_{fuel}(z) = T_{clad}(z) + Q(z) r_i / 2k_f$ (13-11)

Where: k_f is the fuel thermal conductivity.

Table 13.3 tabulates the results of equations (13-8) through (13-11) at the start of the accident (t = 0). For all subsequent times, because the decay heat is decreasing, all of the calculated heat fluxes and temperatures will be lower.

Table 13.3 Initial (t=0) Fuel and Cladding Temperatures

z/Lf	**Q** (W/m2)	T_{air} (°C)	T_{clad} (°C)	T_{fuel} (°C)
0.00	0	27	27	27
0.10	17,209	33	36	45
0.20	32,733	52	58	75
0.30	45,053	81	89	112
0.40	52,963	117	127	154
0.50	55,688	158	168	197
0.60	52,963	198	208	235
0.70	45,053	235	243	266
0.80	32,733	263	269	286
0.90	17,209	282	285	294
1.00	0	288	288	288

For the maximum fuel centerline temperatures, which occur at time t = 0, the safety limit temperatures have a very large margin. The highest fuel centerline temperature is 294 °C. The margin to the safety limit is so high as to negate the nonconservative assumption regarding the lack of a temperature drop across the gap and cladding. As mentioned in the loss-of-coolant accident assumptions, a more likely scenario is that the core is only partially uncovered. This conclusion that the temperature rise for a partial loss of coolant is less severe was verified experimentally by General Atomics in “Simulated Loss-of-Coolant Accident for TRIGA Reactors,” by J.R. Shoptaugh, Jr. The staff concludes that fuel and cladding integrity will not be threatened by a total loss-of-coolant accident.

The other aspect of the loss-of-coolant accident to be analyzed is the radiation levels and subsequent doses that result from an uncovered core. Once again, the calculations use extremely conservative assumptions. Specifically, the reactor is assumed to have been operating at full power (1250 kW(t)) continuously for 1 yr, followed by an instantaneous loss of coolant. This assumption maximizes the gamma-ray source strength. The source strength is determined by the ORIGEN-2 code. Having determined the source strength, radiation transport calculations are performed using the Monte Carlo N-particle transport (MCNP) code.

Modeling of the reactor core and reactor bay for radiation transport purposes is performed using the geometries shown in Figures 13.1 and 13.2 below.

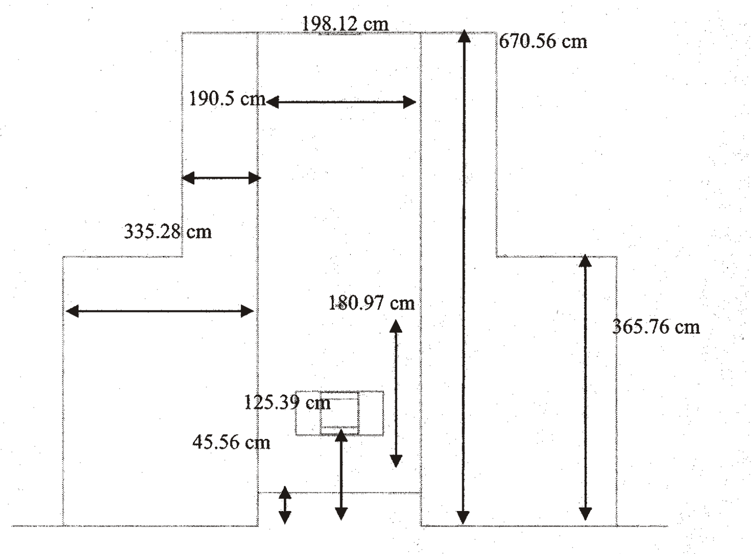

Figure 13.1 Core and biological shielding, MCNP model

The model approximates the reactor core as a right circular cylinder with a diameter of 45.6 cm and a height of 38.1 cm. On the top and bottom of the core are a graphite zone and aluminum grid plates. The core contains fuel elements, three standard control rods and one transient control rod, one void location, one central thimble (void), one source (void), and one pneumatic transfer site (void). The biological shielding is approximated as two concrete cylinders. The lower cylinder is 335.38 cm thick at the bottom. At the 12-ft level, the thickness is 190.5 cm.

As shown in Figure 13.2, the model approximates the reactor bay as a hemispherical dome that covers a right circular cylinder.

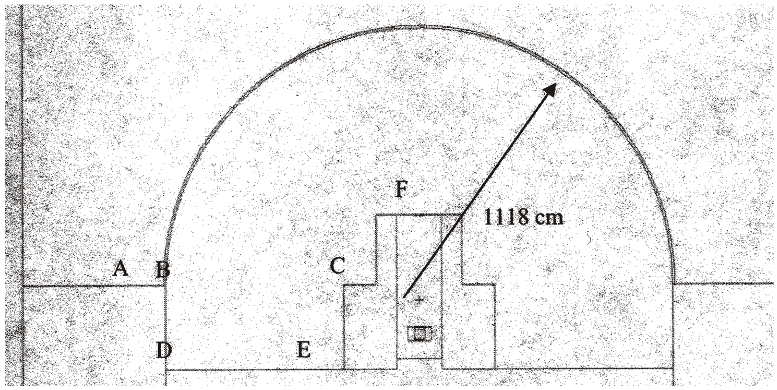

Figure 13.2 Reactor bay, MCNP model

The radius of the hemisphere is 1118 cm as measured from the reactor core. The right circular cylinder also has a radius of 1118 cm and is 365.8 cm high. The dome is a concrete slab 10 cm thick. The reactor bay free-air volume is 144,000 ft^3 (4,078 m^3).

Radiation doses are calculated at various receptor locations within the reactor bay (on site), as shown in Figure 13.2, and at locations outside the reactor bay (off site), as measured from a fence defining the controlled area. The receptor locations are labeled A through F as described below. (The fence is about 2 m from the bay boundary.)

Key	Receptor Location	Radius (m)	Elevation (m)
A	2 m beyond dome (radius of controlled area)	13.18	4.66
B	1 m above grade/12-ft level, 30 cm inside the dome	11.18	1.00
C	1 m above 12-ft working platform, 30 cm from biological shielding	2.21	4.66
D	1 m above the reactor bay floor, 30 cm from wall	11.18	1.00
E	1 m above the reactor bay floor, 30 cm from the biological shielding	3.65	1.00
F	1 m above the 22-ft level, reactor center	0	7.71

Table 13.4 summarizes the results for onsite and offsite dose rates (R/hr), showing dose rates inside the reactor bay and at several locations in the unrestricted area outside the fence (about 2 m from the building). As indicated by the results, the doses to workers who evacuate the reactor bay within a few minutes after an instantaneous loss of coolant are well within the NRC limits for annual routine occupational doses as stated in 10 CFR 20.1201, “Occupational Dose Limits for Adults.”

Table 13.4 Gamma-Ray Ambient (Deep) Dose Rates (R/hr) at Selected Locations for Times Following Loss-of-Coolant Accident after Operation for 1 Year at 1250 kW(t) Power

	Time after accident				
Onsite elev.—radius from center of reactor bay/core	0 hrs	1 hr	24 hrs	30 days	180 days
22 ft (center)	1.48×10^{4}	5.03×10^{3}	1.43×10^{3}	3.41×10^{2}	6.19×10^{1}
12 ft (2.2 m)	1.28×10^{1}	4.44	1.55x100	3.61x101	7.49×10^{-2}
0 ft (3.6 m)	4.79	1.71	5.78×10^{-1}	1.38×10^{-1}	2.94×10^{-2}
12 ft (10.2 m)	1.28×10^{1}	4.75	1.58	3.92×10^{-1}	8.16×10^{-2}
0 ft (10.2 m)	1.09×10^{1}	3.90	1.28	3.18×10^{-1}	6.62×10^{-2}
Offsite—radius from center of reactor bay/core					
13.28 m	2.59×10^{-1}	2.97×10^{-2}	3.23×10^{-2}	8.70×10^{-3}	1.82×10^{-3}
20 m	1.15×10^{-1}	1.29×10^{-2}	1.35×10^{-2}	3.21×10^{-3}	6.13×10^{-4}
30 m	6.56×10^{-2}	7.10×10^{-3}	6.75×10^{-3}	1.87×10^{-3}	3.55×10^{-4}
40 m	4.59×10^{-2}	4.97×10^{-3}	5.13×10^{-3}	1.35×10^{-3}	2.36×10^{-4}
50 m	3.08×10^{-2}	3.56×10^{-3}	3.45×10^{-3}	8.37×10^{-4}	1.72×10^{-4}
70 m	1.98×10^{-2}	2.31×10^{-3}	2.01×10^{-3}	5.39×10^{-4}	1.06×10^{-4}
100 m	1.05×10^{-2}	1.25×10^{-3}	1.30×10^{-3}	3.05×10^{-4}	5.63×10^{-5}

With the exception of dose rates at time t = 0 at 13.28 m and 20 m, projected doses to the general public in the unrestricted area around the facility would take longer than 1 hr to exceed the 100 mrem annual limit for the general public stated in 10 CFR 20.1301, “Dose Limits for Individual Members of the Public.” (For example, with the exception noted above, the highest dose rate of 6.56×10^{-2} R/hr occurs at 30 m at time t = 0. This results in a dose rate of 65.6 mR/hr and falls off to a rate of 7 mR/hr in 1 hr. A member of the public at that location for a few minutes would receive a dose of a few mR.) The area surrounding the reactor facility is under the control of KSU, and exposures outside the reactor bay environment can be limited by controlling access in accordance with the emergency plan and procedures. In accordance with the emergency plan, this event would be classified as an Alert, with reentry to the reactor or other onsite areas requiring approval of the RSC. The staff concludes that KSU has complete authority to control access to campus locations in accordance with the emergency plan and procedures.

13.2.3 Insertion of Excess Reactivity

The licensee analyzed the rapid insertion of excess reactivity to ensure that there will be no challenges to the SLs and thereby the integrity of the TRIGA fuel. As discussed in Section 13.2.2 of this SER, the safety limit is 1150 °C when clad temperature is below 500 °C, which is the case when there is water in the pool. The rapid insertion of excess reactivity will result in a power excursion leading to rapid heating of the fuel. The negative reactivity feedback inherent in TRIGA fuel limits the excursion. The analyzed scenario involves the insertion of $3.00 of positive reactivity at zero power by rapid removal of a control rod. The analysis is predicated upon the following very conservative assumptions:

- The reactor is operating with the proposed number of fuel rods (conservative because it maximizes the power produced per fuel rod).
- Reactor and coolant ambient (zero power) temperature is 27 °C (= To) (Assured by an interlock that prevents pulsing if power is greater than 10 kW(t)).
- Maximum reactivity insertion for pulsing is $3.00 (= 2.1% Δk/k) (TS 3.2.3(1)).

The analysis first requires a calculation of the heat capacity of the fuel. The specific heat capacity of the fuel is given by the following:

$$c_{pf} = 340.1 + 0.6952\ T\ (°C)$$

If there are N fuel elements, each with a mass of M, the total heat capacity is given by the following:

$$K = (M)(N \text{ fuel elements})(c_{pf})$$

$$K = C_o + C_1 T(°C)$$

Where: $C_o = 6.682\text{x}10^4$ and

$C_1 = 136.6$.

The Fuchs-Nordheim model for power excursions is the basis for the evaluation of this scenario. (This model, developed by Fuchs and Nordheim in the 1940s to describe and predict prompt critical excursions and adapted by General Atomics to describe and predict TRIGA reactor prompt critical pulsing, has been used in previous licensing considerations for TRIGA reactors.) In accordance with that model, a reactivity insertion of ρ will cause a power increase from an initial power of P_o to a power of P_{max}, given by the following:

$$P_{max} - P_o = C_o(\rho - \beta)^2 / 2\alpha l + C_1(\rho - \beta)^3 / 6\alpha^2 l$$

Where: β = delayed neutron fraction,

α = absolute value of the temperature coefficient of reactivity, and

l = effective neutron lifetime.

(Table 13.1 of this SER contains values for these parameters.) A reactivity insertion of $3.00 will result in a power increase of 1430 MW. According to the Fuchs-Nordheim model, the temperature increase resulting from the reactivity insertion is given by the following:

$$T_{max} - T_o = 1.90(\rho - \beta)/\alpha$$

The temperature increase as calculated by this equation is the core average temperature increase and is equal to 229 °C. The maximum temperature increase will occur in the fuel element with the highest peaking factors, as listed in Table 13.2 of this SER. The core radial peaking factor has a value of 2, while the axial peaking factor has a value of π/2. Their product is π, and so the maximum temperature increase is π x 229 °C or 719 °C. Therefore, the maximum temperature is 746 °C (719 °C + 27 °C), which is well below the safety limit of 1150 °C.

The temperature increase only depends on the amount of reactivity insertion, not on the power at which the insertion occurred. Hence, this analysis also applies to a reactivity insertion caused by an experiment failure at full power, where reactivity of any independent or combination of more than one interdependent experiment is limited to $2.00 (TS 3.6). The insertion of the maximum reactivity of $2.00 results in a peak temperature increase of 443 °C. Since the peak fuel temperature at full power is 503 °C (Table 4.2), the resulting fuel temperature following the experiment failure would be 946 °C, well below the safety limit. TS 5.4.3(1) limits unsecured experiments to a maximum reactivity of $1.00. In this latter case, movement of a $1.00 experiment during operation would cause a power excursion but not a prompt critical pulse. The power-level scrams in TS 3.3.3(1) give reasonable assurance that a safety limit will not be exceeded.

The limit on excess reactivity for the KSU TRIGA is $4.00 (TS 3.1.3(1)). Consider the scenario in which $3.00 of that reactivity is reserved for a maximum pulse. The remaining $1.00 of reactivity will allow operation to 107 kW(t). Operation at that power level results in a peak fuel temperature of 150 °C. The calculated peak temperature of 716 °C for a $3.00 pulse added to the peak temperature for operation at 107 kW(t) is 866 °C. This postulated scenario would also be prevented by the 10 kW(t) interlock (part of the control system but not a TS). This shows that the excess reactivity limit, along with a pulse limited by the magnitude of the remaining available reactivity, prevents any combination of power operation and pulse exceeding the safety limit of 1150 °C.

The insertion of the maximum possible reactivity will result in a peak fuel temperature of 746 °C, well below the accident safety limit. Since the maximum rapid insertion of reactivity (pulse insertion) is limited to $3.00 (TS 3.2.3(1)), the reactor fuel will not approach the limit at which fuel cladding failure could lead to fission products escaping into the reactor coolant. Therefore, there is reasonable assurance that no radiation exposures will occur as a result of this event.

13.2.4 Single Element Failure in Air

The typical MHA for TRIGA reactors involves failure of the cladding of a single fuel element after extended reactor operations, followed by instantaneous release of the fission products directly into the air of the reactor bay. The radionuclide inventory therefore immediately affects facility personnel and subsequently affects the general public following the release of the radionuclides to the environment.

Inventories of radioactive fission products are calculated using the ORIGEN code. ORIGEN is a code developed by Oak Ridge Laboratory and is in the Office of Nuclear Reactor Regulation Licensing Code Catalog as approved for licensee use. The fraction of noble gases and iodine contained in that inventory which is actually released from the clad-fuel gap is assumed to be 1×10^{-4}. This conservative value is prescribed in NUREG-2387 and is used in analyses for other TRIGA reactor facilities. The fraction of radionuclides released, other than noble gases and iodine (particulates), is assumed to be 1×10^{-6}, again an estimate from NUREG-2387. The maximum potential exposure to facility personnel assumes that the radionuclide release is held within the reactor bay. (Calculations that take into account ventilation of the reactor bay are also performed.) Potential exposure to the public assumes unrestricted release of the radionuclides in a plume from the reactor bay exhaust fan.

The fission product inventory in a single fuel element that is available for release is calculated based on a two-period model of reactor operation. The first period (historical operations) assumes that the reactor has operated at its average thermal power continuously for 40 yrs. That average thermal power is 3.5 kW(t). The second period assumes that the reactor operates at its (new license limit) full power of 1250 kW(t) continuously for another 40 yrs. The fission product inventory in a single fuel element at the end of the second period is the sum of the inventory remaining from the first period plus the inventory generated from the second period.

The single element failure occurs at the end of the second 40-yr period. The ORIGEN code calculates the activities of noble gases, iodine isotopes, and particulates that would be released from the fuel element. The activities resulting from the calculations are then compared to the annual limit on intake (ALI), the DAC, and the effluent limit (EL) that are listed in Appendix B to 10 CFR Part 20. (The ORIGEN calculation results in the activities of 688 isotopes, some listed in Appendix B to 10 CFR Part 20 and some not listed.)

The licensee has presented a comprehensive analysis of gaseous (noble gases and iodines) and particulate radionuclides resulting from the ORIGEN calculations. The analysis provides ALI, DAC, and EL for the following four groups of radionuclides:

(1) gaseous radionuclides identified in the ORIGEN calculations that are listed in Appendix B to 10 CFR Part 20

(2) gaseous radionuclides identified in the ORIGEN calculations that are not listed in Appendix B to 10 CFR Part 20

(3) particulate radionuclides identified in the ORIGEN calculations that are listed in Appendix B to 10 CFR Part 20

(4) particulate radionuclides identified in the ORIGEN calculations that are not listed in Appendix B to 10 CFR Part 20

It is convenient to express the ALI, DAC, and EL calculations in terms of fractions relative to the 10 CFR Part 20, Appendix B limits. Thus, a total fraction less than unity means that the limit has not been exceeded. Table 13.5 summarizes the results and presents ratios of the various limits. For example, the effluent limit ratio is 7.2×10^{-2}, meaning that the effluent concentration is 7.2 percent of the limits found in Table 2, Column 1, of Appendix B. The ELs never exceed 10 CFR Part 20 limits. (The staff has independently performed the calculations leading to the above conclusion using a slightly different methodology and has obtained results that are in close agreement with the licensee's calculations.)

Table 13.5 Releases Compared to Limiting Values

Category	ALI fraction	DAC fraction, without vent.	DAC fraction, with vent.	EL fraction
Gaseous, listed	2.1×10^{3}	2.9×10^{1}	4.0×10^{-1}	4.7×10^{-2}
Gaseous, unlisted	2.7	1.5×10^{-4}	1.5×10^{-4}	1.7×10^{-5}
Particulate, listed	4.9×10^{2}	1.6×10^{2}	7.0×10^{-2}	2.4×10^{-2}
Particulate, unlisted	3.9×10^{1}	3.3×10^{-2}	4.5×10^{-3}	5.2×10^{-4}
Total Fraction	2.6×10^{3}	1.9×10^{2}	4.8×10^{-1}	7.2×10^{-2}

Two DAC fractions are presented, with and without ventilation of the reactor bay. With ventilation (since loss of electrical power would be considered a second failure that makes a hypothetical accident even more incredible), DAC values never exceed limits (maximum DAC fraction = 4.8×10^{-1}). This corresponds to a maximum dose of approximately 2.5 rem. Without ventilation, however, the maximum DAC fraction (1.9×10^{2}) exceeds limits. This can be mitigated by restricting access to the reactor bay. Emergency Plan Procedure EPP-13 controls recovery and reentry into the reactor bay.

The ALI fraction is greater than unity (2.6×10^{3}). However, there is no scenario in which a single individual can collect and breathe all of the radioisotopes that are released. Restricting access to the reactor bay and operation of the ventilation system will both result in limits not being exceeded. In accordance with the emergency plan, the emergency director strictly controls access to the reactor bay.

Worst-case doses to members of the general public would be approximately 3.6 mrem total effective dose equivalent at a receptor location 30 m from the ventilation exhaust. This value is below 10 CFR Part 20 limits for doses to members of the general public.

13.3 Staff Evaluation

The applicant has postulated and analyzed sufficient accident-initiating events and scenarios to demonstrate that the reactor design, management, operating limits, and procedures are planned in a manner such that SLs will not be exceeded and radiation exposure to the facility staff and the public will not exceed the NRC dose limits in 10 CFR Part 20.

13.4 Conclusions

The staff concludes that the licensee has postulated and analyzed sufficient accident-initiating events and scenarios to demonstrate that the reactor design, management, operating limits, and procedures are planned in a manner such that radiation exposure to the facility staff and the public will not exceed the NRC limits in 10 CFR Part 20. On the basis of its review of the licensee's accident analysis, the staff concludes the following:

- Under the least favorable conditions, the MHA of the failure of a fuel element cladding in air, with reactor bay ventilation and restricted bay access, will not result in occupational radiation exposure of the facility staff or radiation exposure of the general public in excess of applicable NRC limits in 10 CFR Part 20.
- For accidents involving insertions of excess reactivity and loss of coolant, the licensee has demonstrated that there is no projected significant damage to the reactor or fuel, except the damage or malfunction assumed as part of the different accident scenarios analyzed. Furthermore, the loss-of-coolant accident will not result in doses to the general public exceeding 10 CFR Part 20 limits. The TS excess reactivity limit, along with a pulse limited by the magnitude of the remaining available reactivity, prevents any combination of power operation and pulse exceeding the safety limit of 1150 °C. The TS reactivity limit on nonsecured experiments ($1.00) provides reasonable assurance that the safety limit would not be exceeded if the experiment were to fail to insert the total reactivity worth. In fact, there is reasonable assurance that a failure of an experiment of maximum worth ($2.00) at full licensed power will not cause the safety limit to be exceeded.

14. TECHNICAL SPECIFICATIONS

14.1 Summary

The staff has evaluated the licensee's TSs as part of its review of the application for renewal of Facility License No. R-75. The TSs define certain features, characteristics, and conditions governing the operation of the KSU TRIGA reactor. As specified by license condition 3.B, Appendix A to the renewed license explicitly includes the TSs. The staff has reviewed the format and content of the TSs using the guidance found in ANSI/ANS-15.1 and NUREG-1537. Other parts of this SER discuss the staff's evaluations of individual TSs to determine whether they meet the requirements specified in 10 CFR 50.36, "Technical Specifications."

14.2 Staff Evaluation

The staff evaluated the KSU TRIGA reactor TSs against the requirements of the regulations. The staff finds the following:

- To satisfy the requirements of 10 CFR 50.36(a), the licensee provided proposed TSs with the application for license renewal. The proposed TSs include appropriate bases for the TSs.
- The KSU TRIGA reactor 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 TSs were appropriately derived from the safety analyses.
- To satisfy the requirements of 10 CFR 50.36(c)(1), the licensee provided TSs specifying an safety limit on the fuel temperature and limiting safety system settings for the RPS 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(c)(2)(ii).
- The TSs contain surveillance requirements that satisfy the requirements of 10 CFR 50.36(c)(3).
- The TSs contain design features that satisfy the requirements of 10 CFR 50.36(c)(4).
- The TSs contain administrative controls that satisfy the requirements of 10 CFR 50.36(c)(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(c)(1), (2), (7), and (8).

14.3 Conclusions

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

15. FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Operate the Reactor

The KSU TRIGA reactor has been in continuous operation since 1962. From 1962 to 1997, the Department of Nuclear Engineering operated the reactor. In 1997, the Departments of Mechanical Engineering and Nuclear Engineering merged to become a single department. The budget for the reactor is part of the department budget. The department has substantial resources; it supports a student body of 424 undergraduate students and 61 graduate students, a full-time faculty of 24, and $2.5 million of external funding for research support.

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.”

The Kansas State University (KSU, the applicant) does not qualify as an “electric utility,” as defined in 10 CFR 50.2. Further, pursuant to 10 CFR 50.33(f)(2), the application to renew or extend the term of any operating license for a non-power reactor shall include financial information that is required in an application for an initial license. Therefore, the staff has determined that KSU 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. KSU 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.

KSU submitted an estimated expense and income table for the reactor facility for the 5-year period from 2007 to 2012. The table showed that projected expenses ranged from approximately $183,000 to $224,000 per year, and would be covered by income mostly from State of Kansas funding to KSU. Other sources of funds for the reactor facility are expected from DOE-funded research and sales of commercial training and calibration services. The staff reviewed the applicant’s estimated operating costs and projected sources of funds, and found them to be reasonable.

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

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 applicant 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 applicant's statement of intent, dated October 12, 2007, contains a preliminary decommissioning cost estimate of $1,031,084 for the DECON option, and states that 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 disposal of radioactive materials, labor, energy, and other costs (tools and supplies), and a 25% contingency. In reviewing the cost estimate, the staff took into consideration several factors, including, but not limited to, power level, size of facility, and operational history. The staff also considered as a point of reference the reference research reactor in NUREG/CR-1756. The applicant stated that it will update its decommissioning cost estimate using the following methodology: the costs for labor, energy and other costs will be escalated using annual changes in the U.S. Bureau of Labor Statistics Consumer Price Index (All Urban Consumers - CPI-U, U.S. City Average, All Items); and costs of radioactive waste disposal will be escalated using biennial changes in the disposal costs of bio shield concrete as provided in NUREG-1307.

To support the statement of intent and the applicant's qualifications to use a statement of intent, the applicant also provided an opinion letter from the KSU attorney regarding information requirements of 10 CFR 50.33(d), stating that "Kansas State University is not a corporation, but rather an agency of the State of Kansas." Further, the applicant provided information showing that the decommissioning funding obligations of KSU are backed by the full faith and credit of the state of Kansas. Further, the President of KSU, currently Jon Wefald (the signator of the statement of intent), is by statute the CEO of KSU, and under a Board of Regents policy, the CEO is specifically authorized to execute contracts on behalf of the educational institution (KSU).

The staff reviewed the applicant's information on decommissioning funding assurance as described above and finds that the applicant 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 KSU's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable.

15.3 Foreign Ownership, Control, or Domination

Section 104d of the Atomic Energy Act, as amended (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, KSU is a non-profit educational institution, an agency of the State of Kansas 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 KSU 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 applicant currently has an indemnity agreement with the Commission, and said agreement does not have a termination date. Therefore, KSU will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.71, KSU, as a nonprofit educational institution licensee, is not required to provide nuclear liability insurance. The Commission will indemnify KSU for any claims arising out of a nuclear incident under the Price-Anderson Act (Section 170 of the Atomic Energy Act, as amended) and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95, for up to $500 million and above $250,000. Also, KSU 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 thee KSU TRIGA and, when necessary, to shut down the facility and carry out decommissioning activities.

16. OTHER LICENSE CONSIDERATIONS

The KSU TRIGA reactor was initially licensed in 1961 to operate at a power level of 100 kW(t). In 1968, a license amendment allowed operation at 250 kW(t). This license renewal allows continued operation of the reactor for an additional 20 yrs at a power level of 1250 kW(t). The staff concludes that the reactor was initially designed and constructed to requirements typical of TRIGA reactors operated safely at power levels up to 2 MW(t). It is reasonable to assume that the KSU TRIGA will continue to operate safely at the increased power level. During the license application review, the staff considered whether prior operation would cause significant degradation in the capability of components and systems to continue to perform their safety functions. Because fuel cladding is the component most responsible for preventing release of fission products to the environment, the staff considered mechanisms that could lead to detrimental changes in cladding integrity. Those mechanisms include radiation degradation of cladding integrity, high fuel temperature and temperature cycling effects on the mechanical properties of the cladding, corrosion, damage from handling or experimental use, and degradation of safety components or systems. The staff found that there is reasonable assurance that the fuel cladding can continue to provide a barrier to release of fission products. In addition, the analyses evaluated in Chapter 13 of this SER show that if an MHA release occurs the limits of 10 CFR Part 20 would not be exceeded in the unrestricted areas.

The KSU TRIGA reactor is typical of a large number of TRIGA reactors operating within the United States and overseas. For the KSU TRIGA reactor, the mechanisms that could result in changes to cladding integrity, such as power density and maximum fuel temperatures, coolant flow rates and temperatures, conductivity, and pH of primary coolant, are comparable to those of other licensed operating TRIGA reactors. In addition, the KSU reactor personnel perform regular surveillances and preventive maintenance. This gives the staff confidence that the KSU reactor can continue to operate safely.

As part of the license renewal process, the NRC staff reviewed the four most recent inspection reports—August 2000 (ADAMS Accession No. ML003755548), September 2002 (ADAMS Accession No. ML022770654), November 2004 (ADAMS Accession No. ML043640526), and June/July 2006 (ADAMS Accession No. ML062540358). Inspection findings with regard to experiments, fuel handling, surveillance, maintenance, and effluent releases are particularly important. In every case, the staff determined that the licensee’s program complied with NRC requirements directed toward the protection of public health and safety. Furthermore, issues of aging or degradation of performance were not of concern. The staff concludes that there has been no significant degradation of equipment and that facility management will continue to maintain and operate the reactor so that there is no significant increase in the radiological risk to facility staff or the public.

This license renewal includes an increase in the licensed power level from 250 kW(t) to 1250 kW(t). The power increase will require the addition of a control rod and modification of the primary coolant heat exchanger to meet all TSs and allow extended operation at the power limit. In accordance with the requirements of TS 6.11(d), the licensee is required to submit a (startup) report describing the measured value of the operating conditions or characteristics of the reactor under the operating limits.

17. CONCLUSIONS

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

- The application filed by KSU for renewal of an operating license for a TRIGA research reactor complies with the requirements of the Atomic Energy Act of 1954, as amended (AEA), as well as the Commission's regulations set forth in Chapter I, "Nuclear Regulatory Commission," of Title 10 of the *Code of Federal Regulations*.
- The facility will operate in conformance with the application (as amended), as well as the provisions of the AEA and the rules and regulations of the Commission.
- The licensee has provided reasonable assurance that (1) the activities authorized by the operating license can be conducted without endangering the public health and safety, and (2) such activities will be conducted in compliance with the Commission's regulations as set forth in Chapter I of Title 10 of the *Code of Federal Regulations*.
- The licensee is technically and financially qualified to engage in the activities authorized by the license in accordance with the Commission's regulations as set forth in Chapter I of Title 10 of the *Code of Federal Regulations*.
- The issuance of this license will not be inimical to the common defense and security or to the public health and safety.

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