KANSAS STATE UNIVERSITY
TRIGA MARK II NUCLEAR REACTOR FACILITY
LICENSE NO. R-88
DOCKET NO. 50-188

SAFETY ANALYSIS REPORT
DATED 21 DECEMBER 2004

REDACTED VERSION*
IN ACCORDANCE WITH
10 CFR 2.390(d)(1)

*Redacted text and figures blacked out or denoted by brackets

K·STATE
Kansas State University
Department of Mechanical and Nuclear Engineering
302 Rathbone Hall
Manhattan, KS 66506-5205
785-532-5610
Fax: 785-532-7057

12/22/2004

Paul M. Whaley
Kansas State University
Nuclear Reactor Facility Manager
112 Ward Hall
Manhattan, Kansas 6506-2506

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

RE: **Additional Information Relative to License Renewal of the Kansas State University Nuclear Reactor Facility (License R-88, Docket 50-188)**

Following submission of information supporting renewal of the KSU TRIGA II facility operating license (including a power uprate to 500 kW) an on-site conference occurred to identify and resolve issues, and a Request for Additional Information (RAI) was issued for the proposed KSU Safety Analysis Report and the Technical Specifications.

The attached material includes (1) a revision to the proposed Safety Analysis Report (addressing both relevant informal recommendations and the Request for Additional Information) , and (2) a tabulation of revisions and a tabulation of how each RAI item was addressed.

Correspondence relating to this information should be directed to P. M. Whaley (address above).

Thank you,

P. M. Whaley

I verify under penalty of perjury that the foregoing is true and correct

12/22/04

MARCIA E. CHACON
NOTARY PUBLIC
STATE OF KANSAS
My Appt. Exp. 5-12-2007

Marcia E Chacon
12/22/2004

CC: D. Hughes, USNRC Project Manager

A020

Safety Analysis Report

Kansas State University

TRIGA Mark II Nuclear Reactor Facility

License R-88

Docket 50-188

21 December 2004

Department of Mechanical and Nuclear Engineering
Kansas State University
302 Rathbone Hall
Manhattan, KS 66506

K-State Nuclear Reactor Facility
110 Ward Hall
Manhattan, KS 66506

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1. The Facility

1.1 Introduction

The Kansas State University nuclear research reactor is owned by Kansas State University (K-State), and operated by the Department of Mechanical and Nuclear Engineering. The reactor was obtained in 1960-1962 through a grant from the United States Atomic Energy Commission and is currently operated under Nuclear Regulatory Commission License R-88 and the regulations of Chapter 1, Title 10, Code of Federal Regulations. The facility supports education and training, research, and public service activities. The reactor facility is located on K-State campus in Manhattan, Kansas, a city of approximately 50,000 residents and 20,000 students, as described in Chapter 2, Site Characteristics.

This report is based on the *Kansas State University TRIGA Mark II Hazards Summary Report* (1961) for the initial operation of the reactor at 100 kW thermal power, the 1968 Safety Analysis Report and Safety Evaluation Report for license amendment to allow 250 kW steady state thermal power (250 MW pulsing capability), and subsequent analyses supporting steady state operations to a maximum of 1,250 kW (pulsing to a nominal $3.00 reactivity insertion. A $3.00 reactivity insertion is expected to result in a peak thermal power of approximately 1,340 MW). Based on proposed reactivity limits, the KSU reactor will only be able to achieve about ½ the proposed maximum power level for steady state operation; therefore thermal-hydraulic and consequence analyses are conservative by at least a factor of 2.

The 1961 KSUTMII Hazards Summary Report identified a set of potential hazards associated with operation of the reactor. The U. S. Atomic Energy Commission reviewed the report and concluded that there is "reasonable assurance that the reactor can be operated at the designated location without undue risk to the health and safety of the public." In 1968, a Safety Analysis evaluated changes in the original hazards analysis for operation a higher steady state power level and the addition of pulsing capability to support a license amendment for operation at 250 kW with pulsing to 250 MW. The U. S. Atomic Energy Commission reviewed the requested amendment and concluded that there is "reasonable assurance that the reactor can be operated at the designated location without undue risk to the health and safety of the public."

This report addresses safety issues associated with operation of the reactor at steady state power levels up to 1,250 kW, and increased pulsing capabilities. The maximum excess reactivity permitted by Technical Specifications cannot achieve a continuous steady state power level greater than about 500 kW; therefore analysis performed for steady state operations at 1,250 kW is extremely conservative in evaluating consequences and characteristics of normal and accident scenarios. This report reflects the as-built condition of the facility, and includes experience with the operation and performance of the reactor, radiation surveys, and personnel exposure histories related to operations to a maximum of 250 kW steady-state power. Where appropriate, radiological characteristics have been extrapolated to reflect operation to a maximum of 1,250 kW. The consequence of routine generation of radioactive effluent and other waste products from steady state operation to a maximum of 1,250 kW is addressed in Chapter 11. Radiation worker and public doses from radiation associated with routine operations are well within the limits of Title 10, Code of Federal Regulations, even under extremely conservative scenarios. The consequence of accident scenarios from operation at 1,250 kW steady-state power and pulsing is

presented in Chapter 13. The consequences of accidents postulated to occur under extremely conservative conditions are well within limits. Therefore, analysis demonstrates that there is still a "reasonable assurance that the reactor can be operated at the designated location without undue risk to the health and safety of the public."

The description of the reactor core and thermal hydraulic analysis presented in Chapter 4, the Secondary Cooling System in Chapter 5, and the Reactor Control System in Chapter 7 are based on 1,250 kW operations.

1.2 Summary & Conclusions on Principal Safety Considerations

Design basis parameters of the KSUTMII are (1) power level, (2) fuel temperature, and (3) fuel loading required to achieve desired power. Limits on the amount of fuel loaded in the core and on the maximum power level ensure the KSU TRIGA Mark II nuclear reactor is an inherently safe reactor.

1.2.1 Safety Considerations

As of July, 1999, there were over 70 TRIGA reactors in use or under construction at universities, government and industrial laboratories, and medical centers in 24 countries. Historically, analysis and testing of TRIGA fuel has demonstrated that fuel cladding integrity is not challenged as long as stress on the cladding remains within yield strength for the cladding temperature. Elevated TRIGA fuel temperatures evolve hydrogen from the zirconium matrix, with concomitant pressure buildup in the cladding. Therefore, the strength of the clad as a function of temperature establishes the upper limit on fuel temperature. Fuel temperature less than limiting values will ensure clad integrity (as evaluated in NUREG 1282) and therefore contain radioactive materials produced by fission in the reactor core.

As a natural-convection cooled system, heat removal capacity is well defined as long as the primary coolant is sub cooled, restricting potential for film boiling. Limiting the potential for film boiling assures fuel and clad temperatures are not capable of challenging cladding-integrity. The maximum heat generated within a fuel element and the bulk water temperature determines the propensity for film boiling. The design basis analysis in Chapter 4 indicates that steady state operation at power levels greater than 1,250 kW in natural convective flow will not lead to film boiling. Analysis indicates that transition boiling may occur during the maximum pulse, but that this condition will not evolve to film boiling.

Negative fuel temperature feedback inherently limits the operation of the reactor. Increases in fuel temperature associated with operation at power regulate maximum possible steady state power, as described in Chapter 4. This coefficient of feedback is a function of the fuel composition and core geometry; within established core systems, the negative temperature coefficient is rather constant with temperature, as described in chapter 4. Excess fuel (above the amount required to establish a critical condition) is required to overcome the negative temperature feedback as operation at power (or pulsing) causes the fuel to heat up. Consequently, maximum possible power using TRIGA fuel is controlled by limiting the amount of fuel loading. [redacted] Limits on total

fuel loading and excess reactivity ensure that the maximum power level will not lead to conditions under which design basis temperatures are possible. A limit on the maximum pulsing reactivity ensures pulsed operations do not lead to conditions under which design basis temperatures are possible.

1.2.2 Consequences of Normal Operations

As indicated in Chapter 11, radiation sources are discharged from the reactor facility in gaseous (airborne), liquid or solid form. These forms are treated individually in subsections of Chapter 11. Airborne radiation sources consist mainly of Argon-41, Nitrogen-16 and Tritium, with Argon 41 the major contributor to off site dose. Limits on Argon-41 and Tritium are tabulated below, with Cesium 137, the other significant isotope of interest for the KSU reactor.

10CFR20: Appendix B to Part 20--Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage								
Atomic No.	Radionuclide	Class	Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion ALI (µCi)	Inhalation ALI (µCi/ml)	Inhalation DAC (µCi/ml)	Air (µCi/ml)	Water (µCi/ml)	Monthly Average Concentration (µCi/ml)
18	Argon-41	Submersion[1]	-	-	3E-6	1E-8	-	-
1	Hydrogen-3	Water, DAC includes absorption	8E+4	8E+4	2E-5	1E-7	1E-3	1E-2
55	Cesium-137	D, all compounds	1E+2	2E+2	6E-8	2E-10	1E-6	1E-5

Gas (HT or T_2) Submersion [1]: Use above values as HT and T_2 oxidize in air and in the body to HTO

A general limit on off site doses from gaseous effluents is also contained in § 20.1101 Radiation protection programs:

> d) To implement the ALARA requirements of § 20.1101 (b), and notwithstanding the requirements in § 20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to § 50.34a, such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions.

Argon 41 is the major contributor to radiation exposure incident to the operation of the K-State reactor. Argon 41 is attributed to neutron activation of natural argon (in air) in the reactor bay atmosphere, rotary specimen rack adjacent to the core, and dissolved in primary coolant. Argon 41 has 1.8 h half-life. Calculations based on 1,250 kW steady state continuous operations show that doses in the reactor bay remain below inhalation DAC. Using extremely conservative

assumptions of operational conditions in concert with the worst-case wind stability class, the off site dose from Argon 41 is slightly less than 10% of the 10 mrem/year limit. A summation of all relative frequencies for winds under Pasquill stability category A (Table B-3) indicates frequency less than 0.6%, i.e., the contribution to off site doses from Argon 41 produced during a year of full power, steady state operations accounts for less than 0.6% of the total dose. All other atmospheric dispersion calculations show that the off site dose from Argon 41 is well within limits, and doses in the reactor bay are below the levels requiring controls of an airborne radioactivity area. Chapter 11 Appendix A shows peak off-site activity concentration during normal operations would be about 4.5×10^{-3} pCi/mL at 53 m downwind under extremely unstable atmospheric conditions, less than the effluent limit of 0.01 pCi/mL. A full year exposure to equilibrium argon concentration for 1,250 kW operations under normal atmospheric conditions would lead to an effective dose of less than 7 mrem, well within applicable limits.

Nitrogen 16 is the major contributor to radiation fields directly over the reactor pool during operation. Nitrogen 16 is produced by a fast neutron reaction with oxygen (as a natural component of water in the core). Nitrogen 16 has a 7:1 second half-life, and consequently does not remain at concentrations capable of contributing significantly to off-site dose. Chapter 11 shows very conservative calculations lead to an expected exposure rate of slightly less than 100 mrem/hr at one meter above the center of the reactor tank during sustained operation at 1,250 kW thermal power. The 22-foot level has radiation monitors directly above the pool and at the rail surrounding access to the pool. Measured exposure rates directly above the pool surface are about 20-30 mR/h at 250 kW operations, and measurements at the rail approach 2 mR/hr. During normal, steady state 500 kW operations dose rate can be expected to achieve 40-60 mrem/hr, and during steady state operations at 1,250 kW the area directly above the pool surface may become a high-radiation area. Therefore, radiation dose rates directly above the reactor pool during expected operations at levels up to 500 kW are within required levels for a radiation area as defined in 10CFR20, and additional administrative controls for access to the area directly above the reactor pool 500 kW to the maximum license power level of 1,250 kW may be required. Installed monitoring systems provide information necessary to identify appropriate access controls.

Tritium is generated by sequential activation of hydrogen (in water) in the core area. Measured tritium specific activity in primary coolant is less than 5×10^{-3} µCi/g. If the reactor bay atmosphere were saturated with this water at 30°C, the water concentration in the air would be less than 3×10^{-5} g/mL and the activity concentration in the atmosphere 1.5×10^{-7} µCi/ml, well below the DAC limit and well below the atmospheric effluent limit with the dilution factor of 200 for discharge from the top of the reactor bay. Even under the extremely conservative assumption that the complete tritium inventory of the reactor pool is released into the reactor bay atmosphere, the tritium concentration would be within limits for an unrestricted area.

No liquid radioactive material is routinely produced by the normal operation of the KSU TRIGA reactor except for miscellaneous neutron activation product impurities in the primary coolant. Non-routine liquid radioactive contamination may be produced during decontamination, maintenance activities (such as resin changes), or occasional level adjustments in the reactor tank or bulk-shield tank. Most releases occur because of condensation in the air-handling unit during summer months. Liquids in the reactor bay floor drains are collected in the reactor bay sump, along with condensate from the air conditioning system. Quantities are small, and these liquids

are released to the sanitary sewerage system after assay and filtration.

Most of the impurities produced in the primary cooling system are deposited in the mechanical filter and demineralizer resins. Therefore, these materials are dealt with as solid waste. The only radionuclides observed are tritium and trace quantities of ^{137}Cs. Typically there are three releases of liquids annually, each amounting to 2.5 m^3. *Even without dilution*, concentrations of these isotopes are well below 10CFR20 Appendix-B effluent concentration limits and monthly sewerage limits. Even unfiltered, untreated primary coolant would meet the liquid effluent limit without further dilution.

1.2.3 Consequences of Potential Accidents

Safety Analysis, Chapter 13, recognizes three classes of accidents for which analysis is required. The maximum hypothetical accident is a fuel element failure with maximum release of fission product inventory, from which the radioactive materials can migrate into the environment. Complete loss of coolant from the reactor pool is the second accident analyzed. The final accident is an insertion of the maximum available positive reactivity. Analysis demonstrates the consequences of reactor accidents are acceptable.

a. Maximum Hypothetical Accident.

Source quantities of radioactive noble gases and iodine are computed and tabulated in Chapter 13 for a maximum hypothetical accident involving cladding failure in a single TRIGA fuel element and the escape of the radionuclides into the environment. Two limiting cases of operation are considered. For short-lived radionuclides, source terms are computed for element failure subsequent to eight hours full-power operation per day for five days. For long-lived radionuclides, source terms are computed for element failure subsequent to continuous operation for 40 years at the average power experienced by the reactor over its first 33 years of operation. Also examined are residual sources still present in fuel, but generated in reactor operations prior to local receipt of the fuel in 1973. Potential consequences of radiological releases are examined.

Under extremely conservative assumptions, potential release and dispersion of a few radioiodines and radioactive strontium inventories exceeds activity levels that permit normal occupancy of the reactor bay. In a 100% release of the inventory of these species, the Derived Air Concentrations for these radioisotopes in the reactor bay could require worker access to be controlled to prevent exceeding the Annual Limit of Intake. The K-State Reactor Emergency Plan controls responses to accidents involving fuel element failure as well as recovery and reentry operations. Unrestricted, uncontrolled, and unmonitored access to the reactor bay following a fuel handling accident is not permitted.

Under extremely conservative assumptions, only a few radionuclides exceed permissible concentrations for release to unrestricted areas, even within the reactor bay. Even in the extremely unlikely event that 100% of the radionuclide inventory is released from a damaged fuel element to the outside atmosphere, very conservative calculations reveal that radionuclides inhaled by persons downwind from the release would lead to organ

doses or effective doses very far below regulatory limits.

Even in the maximum hypothetical accident, no workers or members of the public are at risk of receiving radiation doses in excess of limits prescribed in federal regulations.

b. Loss of Coolant

Although total loss of reactor pool water is considered to be an extremely improbable event, calculations have been made to determine the maximum fuel temperature rise that could be expected to result from such an event taking place after long-term operation at full power of 1,250 kW. Even under extraordinarily conservative assumptions and approximations, the maximum fuel temperature reached in a loss of coolant accident is less than 300°C, well below any safety limit for TRIGA reactor fuel.

Radiation doses from loss of coolant accident under extremely conservative assumptions are computed and have been tabulated in Chapter 13. [redacted]

c. Insertion of Excess Reactivity

Two reactivity accident scenarios are presented. The first is the insertion of 2.1% ($3.00) reactivity at zero power by sudden removal of a control rod. The second is the sudden removal of the same reactivity with the core operating at a power level equivalent to the remainder of the core excess reactivity. Analysis shows that peak fuel temperatures in the first case does not reach fuel temperature limits, with a maximum temperature less than 750°C at the peak in the hot channel for conditions where initial steady state power level is regulated only by the balance of core excess reactivity, while cladding temperature remains below 500°C. In the second case, maximum fuel temperature is calculated at a maximum of less than 870°C at the peak in the hot channel, again with cladding temperature less than 500°C. [redacted]

1.3 General Description of the Facility

1.3.1 Geographical Location

The reactor is located on the campus of Kansas State University, in the City of Manhattan, in Riley County, Kansas. The licensee (through the University Police Department) controls access to Kansas State University facilities and infrastructure. County, city, and university maps are supplied in Chapter 2. The reactor is located in the north wing of Ward Hall. Latitude and longitude, Ward Hall building plans, universal Transverse Mercator coordinates, population details, etc. are provided in Chapter 2.

The operations boundary of the reactor facility encompasses the reactor room and control room. The site boundary encompasses the entire building and adjacent fenced areas controlled by management of the facility.

1.3.2 Principal Characteristics of the Site

The site is in the Flint Hills uplands of northeast Kansas, characterized by glacial sediments. Soil borings reveal modest topsoil, varying levels of silt and clay loams overlying bedrock, limestone, and shale. Groundwater is encountered in sand or gravel layers 18 to 35 feet below existing grade.

The reactor site is located on high ground in the northwest sector of the university campus. Climate is temperate, with typically 32 inches of rain annually. Storm drainage is excellent and sanitary sewerage from the reactor building is collected by a system serving the entire university.

The site is in a seismic risk zone 2. Liquefaction potential of local soils is minimal.

1.3.3 Principal Design Criteria, Operating Characteristics, & Safety Systems

The KSU TRIGA reactor is a water-moderated, water-cooled thermal reactor operated in an open pool. The reactor is fueled with heterogeneous elements clad with stainless steel, consisting of nominally [redacted] enriched uranium in a zirconium hydride matrix. In 1968, the KSUTMII was licensed to operate at a steady-state thermal power of 250 kW with a pulsing thermal power limit of 250 MW. Application is made concurrently with license renewal to operate up to a maximum steady state power level of 1,250 kW steady-state thermal powers and pulsing to $3.00 (nominal 1,340 MW peak power). Reactor cooling is by natural convection. The 250-kW core consists typically of [redacted] elements (a minimum of [redacted] planned for the 1,250-kW core), each containing as much as [redacted] grams of ^{235}U. The reactor core is in the form of a right circular cylinder about 9 in. (23 cm) radius and 15 in. (38 cm) depth, positioned with axis vertical near the base of a cylindrical water tank 1.98 m (6.5 ft.) diameter and 6.25 m (16 ft.) depth. Criticality is controlled and shutdown margin assured by control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite. The 250 kW core originally used three control rods, the 1,250-kW core will be controlled by four. The reactor tank is surrounded on the side and at

the base by a biological shield of reinforced concrete at least 8.2 ft (2.5 m) thick. The tank and shield are in a 4078 m^3 (144,000 ft^3) dynamic confinement building made of reinforced concrete and structural steel, with composite sheathing and aluminum siding. Sectional views of the reactor are shown in Figures 1.1 and 1.2, with a floor layout in Figure 1.3 showing the 0-foot, 12-foot and 22-foot levels of the facility.

1.3.4 Engineered Safety Features

The design of the KSU TRIGA Reactor, licensed in 1962, and the power upgrade to 250 kW in 1968 imposed no requirements for engineered safety features. As discussed in Chapter 13, and from previous analysis, neither forced cooling flow nor shutdown emergency core cooling is required for operation at steady state thermal power as high as 1900 kW, a large margin over the 1,250 kW steady state operations.

1.3.5 Instrumentation and Control (I&C) and Electrical Systems

Instruments and controls are described in Chapter 7, with the electrical power system described in chapter 8. The reactor instrument and control systems include the reactor control system, process instruments, reactor protection system, and radiation safety monitoring systems. As previously noted, there are no engineered safety features at the KSUTMII and therefore no associated instrumentation.

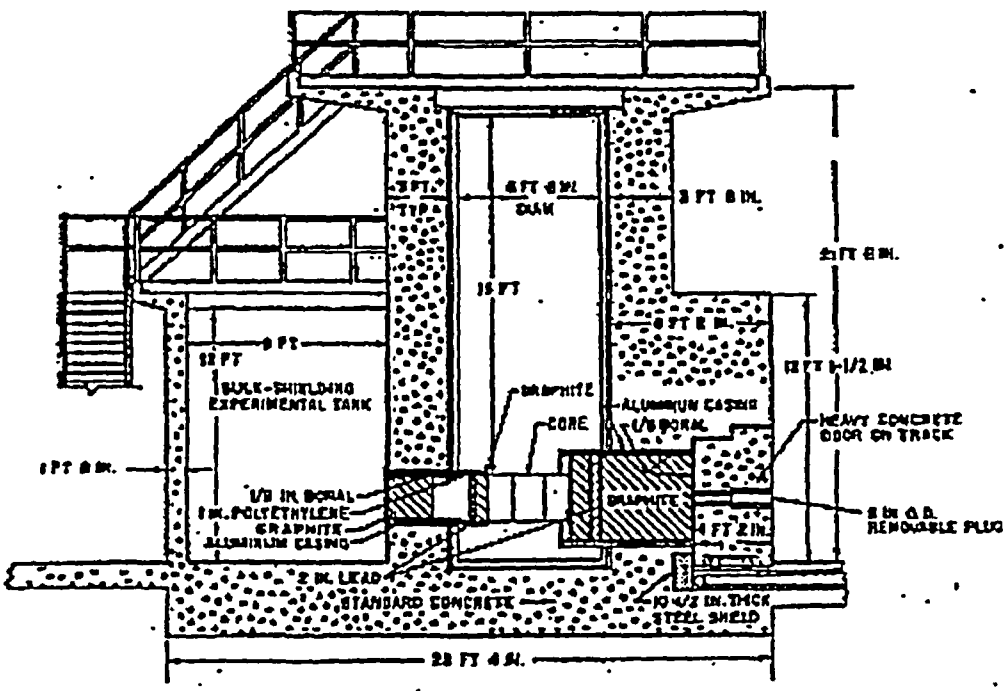


Figure 1.1, Cutaway View of the K-State Reactor.

The bulk of the reactor I&C systems are hard-wired analog systems (primarily manufactured by General Atomics) widely used at various NRC-licensed facilities. A comprehensive upgrade of the reactor's original reactor control system and reactor protection system was accomplished during control room modifications in 1993 and 1994. The original console and vacuum tube instruments were replaced by a solid-state console (previously used in the U.S. Geological Survey's TRIGA Mark I reactor). New General Atomics neutronic measuring channels (N-1000 series) were installed. These channels have optically isolated outputs, allowing other devices to utilize the neutronic data. Installation of a fourth control rod is planned for the near future, with design and hardware supplied by GA. Improvements to air monitoring were accomplished during a modification to the reactor bay HVAC system in 1998.

a. Reactor Control System

The reactor control system includes the mechanical and electrical systems for control rod drives, and instruments that monitor control rod position. Each control rod can be independently manipulated by pushbutton console controls. One control rod can be operated in an automatic mode to regulate reactor power according to a manual setpoint, indicated power on the linear power level monitoring channel and a wide range power level monitoring channel (period) feedback. The wide range power level monitoring channel of the reactor protection system provides interlock signals and actions to the reactor control system. The reactor control system is also interconnected to the reactor protection system through a manual scram bar above the control rod drive switches (allowing the reactor protection system to be actuated manually) and the automatic mode control (as described above).

b. Process Instruments

Primary water temperature is measured in the water box and displayed on the console. A manometer indicates flow rate through the cleanup loop locally. Conductivity probes measure water purity at the entrance and exit of the cleanup loop. A level alarm/switch alerts the operator when the reactor pool water level is low. Primary and secondary flow rates are indicated on local monitors. The makeup water system is instrumented with a flow meter/totalizer to measure water added to the reactor pool or bulk shield tank. Level switches provide indication for low secondary surge tank water level and high reactor bay sump levels. Fuel temperatures for two elements can be monitored on the reactor control console and on an auxiliary panel; the fuel temperature indicator on the auxiliary panel provides input to the reactor protection system.

c. Reactor Protection System

The reactor protection system is designed to ensure reactor and personnel safety by limiting parameters to operation within analyzed operating ranges. Process parameters that can automatically initiate reactor protection system actions include neutron level, rate of rise (period) and fuel temperature. Circuit provisions allow additional, external scrams to be installed when personnel, facility, or experiment protection might require rapid shutdown based on instrument-monitored parameters. A bar above the control rod drive switches allows the scram system to be actuated manually by the reactor operator at the controls.

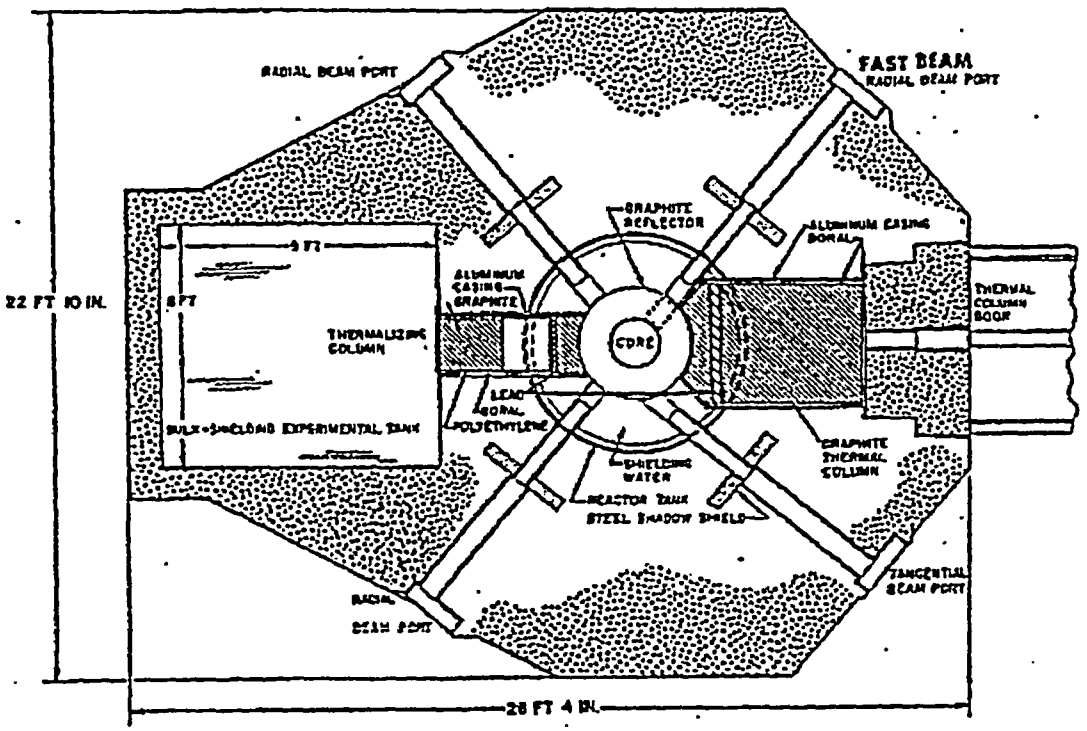


Figure 1.2, Top View of K-State Reactor.

Three neutronic instruments measure reactor power separately: a wide-range logarithmic channel, a multi-range linear channel, and a percent power channel. These provide at least two indications of reactor power from source range to power range. The nuclear instruments of the reactor protection system are integrated into the reactor control system through the automatic power level control system and through rod control interlocks. If a reactor pulse is performed another channel is added to the central thimble to record pulse data. One fuel temperature indicator has a trip relay built into the meter movement for a high fuel element temperature trip. Most of the components of the reactor control system are located within the same enclosure as sections of the reactor control system, although the reactor safety system fuel temperature indicator is mounted on the auxiliary control panel. Since the core is cooled by natural convection, no engineered safety features are necessary for safe reactor shutdown.

d. Radiation Safety Monitoring Systems

Radiation monitors are installed to monitor radiological conditions at the facility. One monitor is stationed on the top of the reactor, with a local, high range indicator and alarm (at 5 R/hr) to initiate evacuation of the reactor bay. One monitor is stationed at the control room door to the reactor bay, with a 2.5-mrem/hr-alarm setpoint. Electrical connections are installed near each beam port, permitting control room and local

indication of radiation levels near an open beam port. The remaining radiation monitors have indicators and alarms both locally and at a central location in the control room.

e. **Electrical Power**

Primary electrical power is provided through the Kansas State University power grid, supplied by an on-campus plant and commercial generators. Main power lines traverse underground tunnels, inhibiting tampering. Loss of electrical power will de-energize the control rod drives, causing the rods to fall by gravity into the core and placing the reactor in a subcritical configuration. Since the core is cooled by natural convection, no emergency power is required for reactor cooling systems. Loss of electrical power does not represent a potential hazard to the reactor. Backup battery systems are provided for emergency lighting and the security system.

1.3.6 Reactor Coolant and other Auxiliary systems

The reactor coolant and auxiliary systems are very simple in design and operation. These systems are required for operation, and not for safety. Many of these systems have been upgraded in recent years to permit extended full power operation of the reactor. Detailed descriptions of the coolant and auxiliary systems equipment and operation are provided in Chapters 4, 5 and 13 of this report.

a. **Reactor Coolant System.**

During full power operation, the nuclear fuel elements in the reactor core are cooled by natural convection of the primary tank water. To remove this bulk heat to the environment, the primary water is circulated through a heat exchanger where the heat is transferred to a secondary cooling loop. A cleanup loop maintains primary water purity with a filter and demineralizer to minimize corrosion and production of long-lived radionuclides that could otherwise occur. The primary coolant provides shielding directly above the reactor core.

b. **Secondary Cooling System**

The secondary cooling system provides the interface for heat rejection from the primary coolant system to the environment. The secondary system is an open system, with the secondary pump discharging through a primary-to-secondary heat exchanger, then through a forced-draft cooling tower. Water returns to an open surge tank (located in the reactor bay) by gravity.

c. **Bulk Shield Tank.**

The bulk shield tank provides radiological controls for an experimental facility, the thermalizing column. Irradiated fuel elements can be stored in the bulk shield tank, or alternately in dry fuel storage pits.

d. **Makeup**

A distillation unit in a room adjacent to the reactor bay provides makeup water through a filter demineralizer unit to the reactor pool or bulk shield tank.

1.3.7 Radioactive Waste Management and Radiation Protection

Operation of the K-State TRIGA reactor produces (low concentration) routine discharges of radioactive gases, periodic batch discharges of (sometimes) slightly contaminated water to sewerage, and small quantities of solid waste. Details of the waste management and radiation protection procedures at the KSU TRIGA reactor are provided in Chapter 11 of this report.

a. **Gaseous Waste**

Maintaining negative pressure controls concentrations of radioactive gases in the reactor bay during operations. An exhaust fan in the roof of the reactor bay, directly over the reactor pool, maintains negative pressure in the reactor bay to ensure that discharges are controlled under conditions of analysis.

b. **Liquid Waste**

Liquid sources are limited generally to tritium-bearing condensate water from the facility air handling system, and occasional releases of tritium-bearing primary coolant from level adjustments in the reactor tank or bulk-shield tank. All reactor bay floor drains and the HVAC condensate drains discharge to a reactor bay sump. Contents of the reactor bay sump are sampled and assayed to assure limits for discharge are met prior to discharge. Sump effluent is filtered prior to discharge to meet NPDES requirements for discharge to campus sewerage. Liquid wastes are released through the sanitary sewerage system after filtration and assay for beta, gamma, and alpha activity.

c. **Solid Waste**

Solid waste is very limited in volume and specific activity. Solid wastes include ion-exchange resin used in reactor-water cleanup, contaminated tools, lab-ware, samples and sample handling material for completed experiments, and anti-contamination clothing associated with reactor experiments and surveillance or maintenance operations. Shipments of solid waste to commercial disposal facilities are made infrequently. Solid wastes shipments are coordinated with the University Radiation Safety Office, Division of Public Safety.

1.3.8 Experimental Facilities and Capabilities

Standard experimental facilities in the KSU TRIGA reactor, as supplied by the vendor, General Atomics, include the central thimble, rotary specimen rack, pneumatic specimen tube, thermal/thermalizing columns, and four beam tubes. Experimental facilities are described in Chapter 10, with auxiliary systems supporting beam tubes in Chapter 9.

a. **Central Thimble**

The reactor is equipped with a central thimble for access to the point of maximum flux in the core. The central thimble consists of an aluminum tube that fits through the center holes of the top and bottom grid plates terminating with a plug below the lower grid plate. The tube is anodized to retard corrosion and wear. The thimble is approximately 20 ft. (6.1 m) in length, made in two sections, with a watertight tube fitting. A removable screen covers the top end of the tube to allow gas relief and to prevent objects from falling into the reactor tank. Although the shield water may be removed to allow extraction of a vertical thermal-neutron and gamma-ray beam (not currently done at the KSU facility at the time this report was completed), four 0.25-in (6.3-mm) holes are located in the tube at the top of the core to prevent expulsion of water from the section of the tube within the reactor core.

b. **Rotary Specimen Rack**

A 40-position rotary specimen rack (RSR) is located in a well in the top of the graphite radial reflector. A rotation mechanism and housing at the 22-ft level of the reactor allows the specimen to be loaded into indexed positions and also rotation of samples for more uniform exposure across the set of co-irradiated samples. The RSR allows large-scale

production of radioisotopes and for activation and irradiation of multiple material samples with neutron and gamma ray flux densities of comparable intensity.

c. **Pneumatic Specimen Tube**

A pneumatic transfer system, permitting applications with short-lived radioisotopes, rapidly conveys a specimen from the reactor core to a remote receiver. The in-core terminus is normally located in the outer ring of fuel-element positions.

d. **Thermal/Thermalizing Columns**

The KSU TRIGA Reactor has two graphite moderated experimental facilities for spectrally tailored experiments requiring well thermalized neutrons. The thermal column, shown in Figures 1.1 and 1.2, has a concrete door acting as shielding. The thermal column is accessible by winching the door, mounted on rails. The space occupied by the rails is covered with steel plate when the thermal column is not in use, shown as the rectangle attached to the reactor pedestal in the 0-foot section of Figure 1.3; towards the top of the figure. The other facility is the thermalizing column, shown in Figures 1.1 and 1.3. Shielding for the thermalizing column is provided by water in the bulk shield tank.

e. **Beam Tube Facilities**

The KSU TRIGA Reactor is provided with four beam tubes. Beam-tube sleeves are welded to the outside surface of the tank and extensions (on axis) are welded to the inside surface. The beam tubes provide beams of neutrons and gamma rays for a variety of experiments. They also provide irradiation facilities for specimens as large as 6 in. (15.2 cm) in diameter. Three of the beam tubes are aligned radial with respect to the center of the core. One of the radial beam tubes is aligned with cylindrical void in the reflector graphite, while the remaining radial beam tubes terminate at the outer edge of the reflector assembly. The fourth beam tube is oriented tangentially with the outer edge of the core.

1.4 Shared Facilities and Equipment

1.5 Comparison with Similar Facilities

The design of the fuel for the KSU TRIGA is similar to that for fuels used in 70 reactors in 24 nations (General Atomics July 1999 data). Of total number of reactors, 45 are currently in operation or under construction with 40 rated for steady-state thermal powers of 250 kW or greater, 22 at 500 kW or greater, and 20 at 1 MW or greater. Nine of the larger power reactors are TRIGA Mark II. The TRIGA Mark II design is a substantial fraction of the 70 reactors using TRIGA fuel world-wide.

In the United States, there have been 26 TRIGA reactors built, with 19 currently in operation (5 TRIGA facilities and 3 non-TRIGA reactors converted to operate with TRIGA fuel at power levels greater than 1,000 kW, as indicated in Table 1.1).

Table 1.1, U.S. TRIGA REACTORS AT 500 kW OR GREATER.

LOCATION	ORGANIZATION	GA MODEL	POWER LEVEL		INITIAL CRITICALITY
			STEADY STATE (kW)	PULSE (MW)	
Pullman, Washington	Washington State University	Conversion	1,000	2,000	1967
Madison, Wisconsin	University of Wisconsin	Conversion	1,000	2,000	1967
College Station, Texas	Texas A&M University	Conversion	1,000	2,000	1968
Bethesda, Maryland	Armed Forces Radiobiology Res. Inst. (AFRRI)	Mark F	1,000	3,300	1962
Denver, Colorado	U.S. Geological Survey	Mark I	1,000	1,200	1969
Corvallis, Oregon	Oregon State University	Mark II	1,000	3,200	1967
University Park, Pennsylvania	Pennsylvania State University	Mark III	1,000	2,000	1965
Austin, Texas	University of Texas	Mark II	1,100	1,600	1992
Sacramento, California	UC Davis	Mark II	2,300	1,200	1990

Table 1.2, U.S. MARK II TRIGA REACTORS.T

LOCATION	ORGANIZATION	POWER		CURRENT STATUS	INTIAL CRITICALITY
		STEADY STATE (kW.t)	PULSE (MW.t)		
Urbana Illinois	University of Illinois	1,500	6,500	Shutdown	1960
Ithaca, New York	Cornell University	500	250	Operating	1962
Corvallis, Oregon	Oregon State University	1,000	3,200	Operating	1967
Sacramento, California	McClellan AFB	2,300	1,200	Operating	1990
Austin, Texas	University of Texas	1,100	1,600	Operating	1992

Major design parameters for the KSU TRIGA are given in Table 1.3. Fuel for the KSU reactor is standard TRIGA fuel having [redacted] of uranium, by weight, enriched up to [redacted] in the ^{235}U isotope. TRIGA fuel is characterized by inherent safety, high fission product retention, and the demonstrated ability to withstand water quenching with no adverse reaction from temperatures to 1150°C. The inherent safety of TRIGA reactors has been demonstrated by extensive experience acquired from similar TRIGA systems throughout the world. This safety arises from the large prompt negative temperature coefficient that is characteristic of uranium-zirconium hydride fuel-moderator elements used in TRIGA systems. As the fuel temperature increases, this coefficient immediately compensates for reactivity insertions. This results in a mechanism whereby reactor power excursions are limited/terminated quickly and safely. Table 1.2 indicates research reactors at 500 kW or above using TRIGA fuel in the U.S..

Table 1.3, KSU TRIGA Reactor Principal Design Parameters

Parameter	Value	Parameter	Value
Max steady-state thermal power	1,250 kW	[redacted]	[redacted]
Maximum pulse reactivity	3.00$	[redacted]	[redacted]
Maximum excess reactivity	2.8 % Δk/k	[redacted]	[redacted]
Number of control rods	4	[redacted]	[redacted]
Regulating rods	1	[redacted]	[redacted]
Shim rods	2	[redacted]	[redacted]
Transient (pulse) rods	1	[redacted]	[redacted]
Minimum shutdown margin	0.71 % Δk/k	Reactor cooling	Natural convection
Integral fuel-moderator material	$U\text{-}ZrH_{1.6\text{-}1.7}$	External moderator	Light water

1.6 Summary of Operations

Reactor utilization since 1980 is summarized in Table 1.4.

Table 1.4, Two-decade operating history for the KSU TRIGA Reactor Facility.

Fiscal year	Total hours of operation	MWh of thermal energy	Typical weekly operation hours	Nuclear eng student users
1981	398	30	8	62
1982	440	47	8	52
1983	501	58	10	51
1984	401	36	8	55
1985	410	36	8	20
1986	469	52	9	22
1987	441	43	9	18
1988	308	31	6	25
1989	257	26	5	27
1990	370	26	7	24
1991	411	35	8	38
1992	449	35	9	38
1993	299	34	*	37
1994	363	38	7	41
1995	394	27	8	29
1996	309	40	6	33
1997	388	28	8	36
1998	458	26	9	26
1999	346	25	7	10

Utilization low in 1993 because of remodeling, console replacement and cooling system replacement.

Since 1980, the following outside users made direct use of the KSU TRIGA reactor services:

- University of Kansas
- University of Nebraska at Lincoln
- Purdue University
- Regional Kidney Disease Program
- Kansas Jr. Academy of Sciences
- National Transportation Safety Board
- Boeing Corporation
- Wolf Creek Nuclear Operating Corp.
- Armed Forces Radiobiology Res. Inst.
- SE Kansas Agricultural Exp. Station
- Kansas Highway Patrol
- Washington University of Saint Louis
- State University of New York
- University of Chicago
- University of Nebraska at Lincoln

Since 1980, the following outside university users made use of the KSU TRIGA reactor's neutron activation analysis services in cooperation with the Geology Department at Kansas State University.

- University of Kansas
- University of Southern California
- University of Georgia
- Louisiana State University
- Cincinnati University
- Nevada-Reno University
- Wichita State University
- Baylor University
- Carleton University

1.7 Compliance with Nuclear Waste Policy Act of 1982

Compliance with Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 for disposal of high-level radioactive waste and spent nuclear fuel is effected through contract between Kansas State University and the U.S. Department of Energy. A copy of the fuel cycle assistance contract is found in Appendix B of this report.

Table 1.5, Major Facility Modifications.

Year	Activity
1960	Authorization of Construction
1962	Initial Criticality, 100 kW
1968	License Amendment for 250 kW and pulsing
1973	Replaced Mark II elements with Mark III
1975	Added new wing to Ward Hall
1993	Replaced secondary cooling system
	Replaced reactor console
	Installed new power level detectors
	Enlarged control room
1999	Changed reactor bay HVAC from positive pressure, distributed unit HVAC system to negative pressure confinement, recalculating HVAC
2001	Increased cooling tower capacity
	Replaced heat exchanger
2004	Replace secondary pump
2005 (scheduled)	Install 4th control rod
	Conversion to 1,250 kW steady state, $3.00 pulsing license

1.8 Facility Modifications and History

Criticality was first achieved on October 16, 1962 at 8:25 p.m. In 1968 pulsing capability was added and the maximum steady-state operating power was increased from 100 kilowatts (kW) to 250 kW.

The original aluminum-clad fuel elements were replaced with stainless-steel clad elements in 1973. With support from the U.S. Department of Energy, coolant system replacement was completed in 1993, as was replacement of the reactor operating console, and enlargement and modernization of the reactor control room. All neutronic instrumentation was replaced in 1994. the secondary cooling system capacity was increased in 2001. Addition of a 4th control rod is scheduled for fall 2002. This description of major facility modifications is presented in tabular form (Table 1.5) to illustrate the timeline.

1.9 Bibliography

Kansas State University TRIGA Mark II Reactor Hazards Summary Report, by R.W. Clack, J.R. Fagan, W.R. Kimel, and S.Z. Mikhail, License R-88, Docket 50-188, 1961.

Analysis of Certain Hazards Associated with Operation of the Kansas State University TRIGA Mark II Reactor at 250 kW Steady State and with Pulsed Operation to $2.00, by R.W. Clack, et al., and the Safety Evaluation by the U.S. Atomic Energy Commission Division of Reactor Licensing, License R-88, Docket 50-188, 1968.

NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," U.S. Nuclear Regulatory Commission, 1987.

2. SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

a. Specification and Location.

The reactor is located on the campus of Kansas State University, in the City of Manhattan, in Riley County, Kansas. It is located in the north wing of Ward Hall, which faces onto 17th St., about 180 meters south of Claflin Rd. Latitude and longitude coordinates of the site are 39°11'30" N, 96°35'2" W. The reactor site, in Universal Transverse Mercator coordinates, is [redacted] Landmarks and nominal locations relative to the reactor are as follows:

Kansas River --	3 km SE
Tuttle Creek Reservoir --	7 km N
Interstate Highway 70 --	13 km S
State Capitol, Topeka --	82 km E
Fort Riley Military Reservation --	12 km W
Manhattan Airport --	9 km SW
U.S. Army Marshall Field Airport --	22 km SW

b. Boundary and Zone Area Maps.

Manhattan is in Riley County, a northeast county in the State of Kansas. The location of Manhattan relative to other counties in the State of Kansas is shown in Figure 2.1 and Chapter 2, Appendix A, Figure 2A.1, along with major highway access to Manhattan.

Manhattan is in the southeast portion of Riley County, with parts of the metropolitan area inside Pottowatomie County. The location of Riley County in Kansas is shown in Figure 2.1; with the location of Manhattan inside Riley County shown in Chapter 2, Appendix A, Figure A2.2.

The location of airports, waterways and public highways surrounding Manhattan is shown in Figure 2.2, and Chapter 2, Appendix A, Figure 2A3.

The location of the reactor near the center of the city of Manhattan is shown in Figure 2.3. Highways and aquatic features directly surrounding the city of Manhattan are shown in Figure 2.4. The location of the reactor on the Kansas State University campus is shown in Figure 2.5

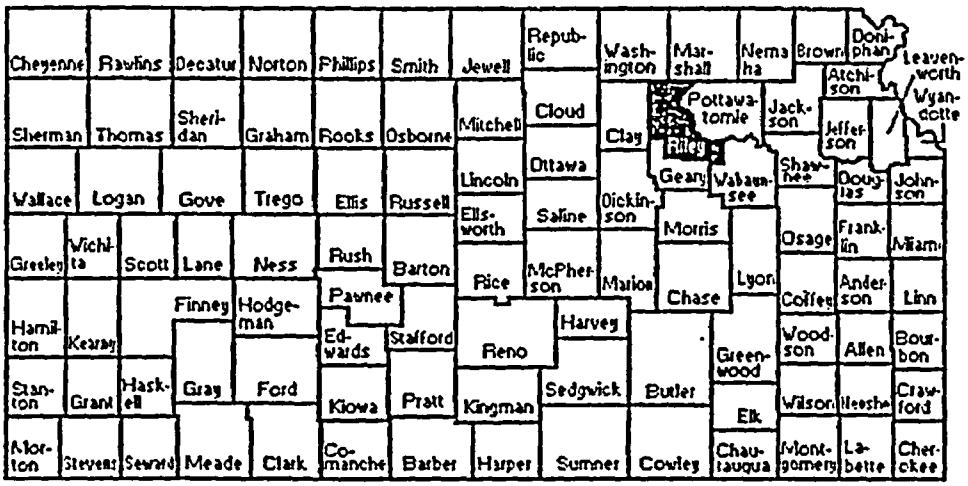


Figure 2.1, Riley County in Relation to Other Counties in Kansas

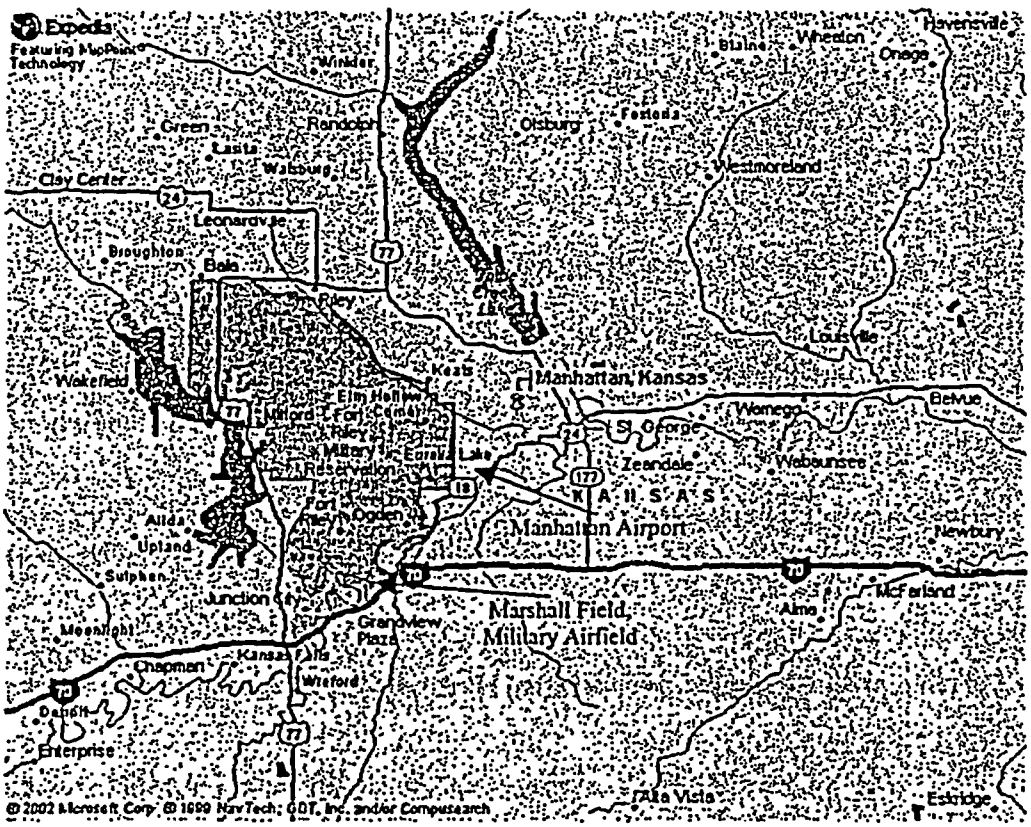


Figure 2.2, Manhattan and Surrounding Items of Interest.

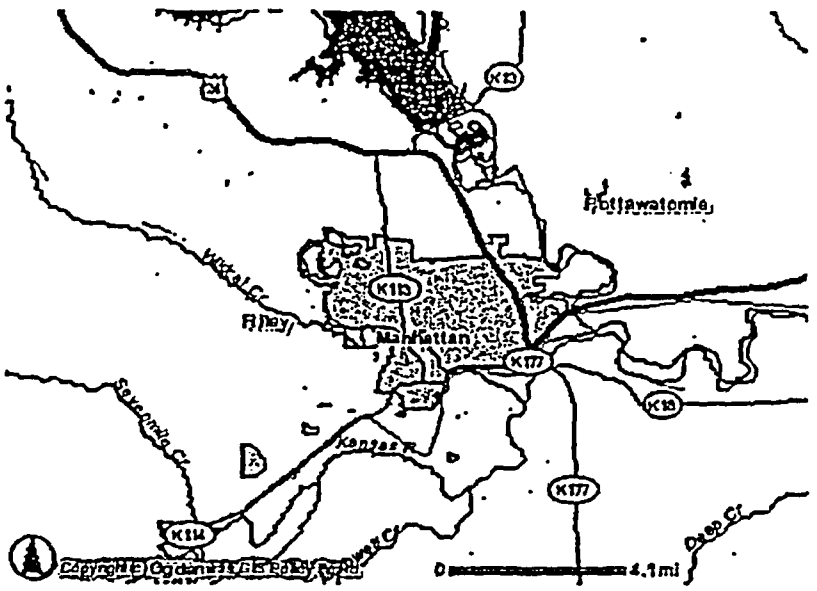


Figure 2.3, Manhattan and Surrounding Highways, Streams, Rivers and Bodies of Water

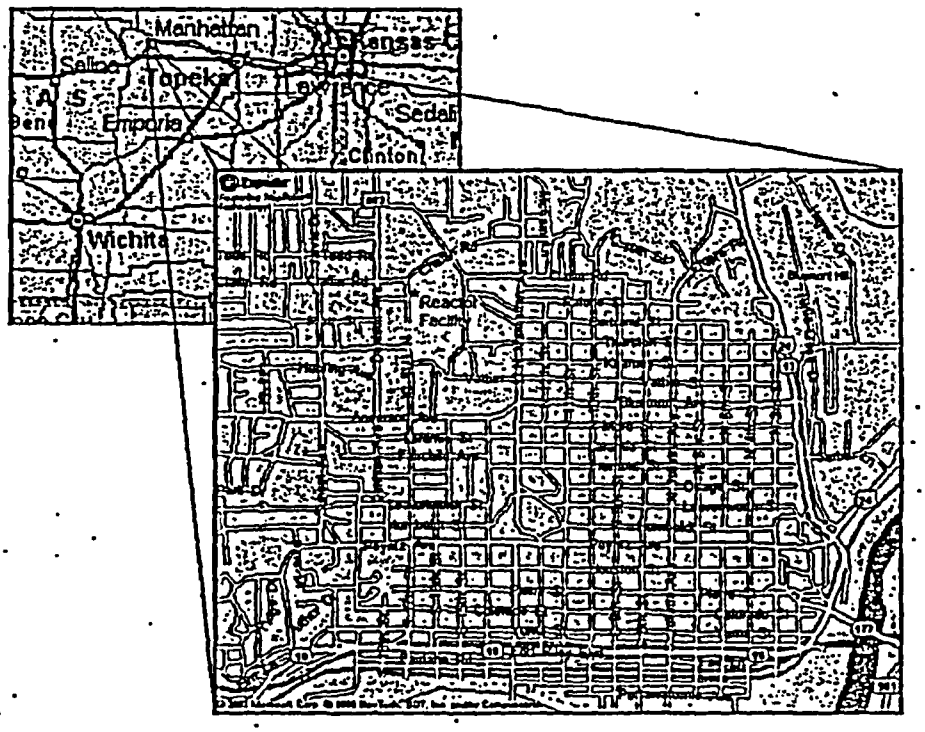


Figure 2.4, Location of the Reactor Within Central Manhattan.

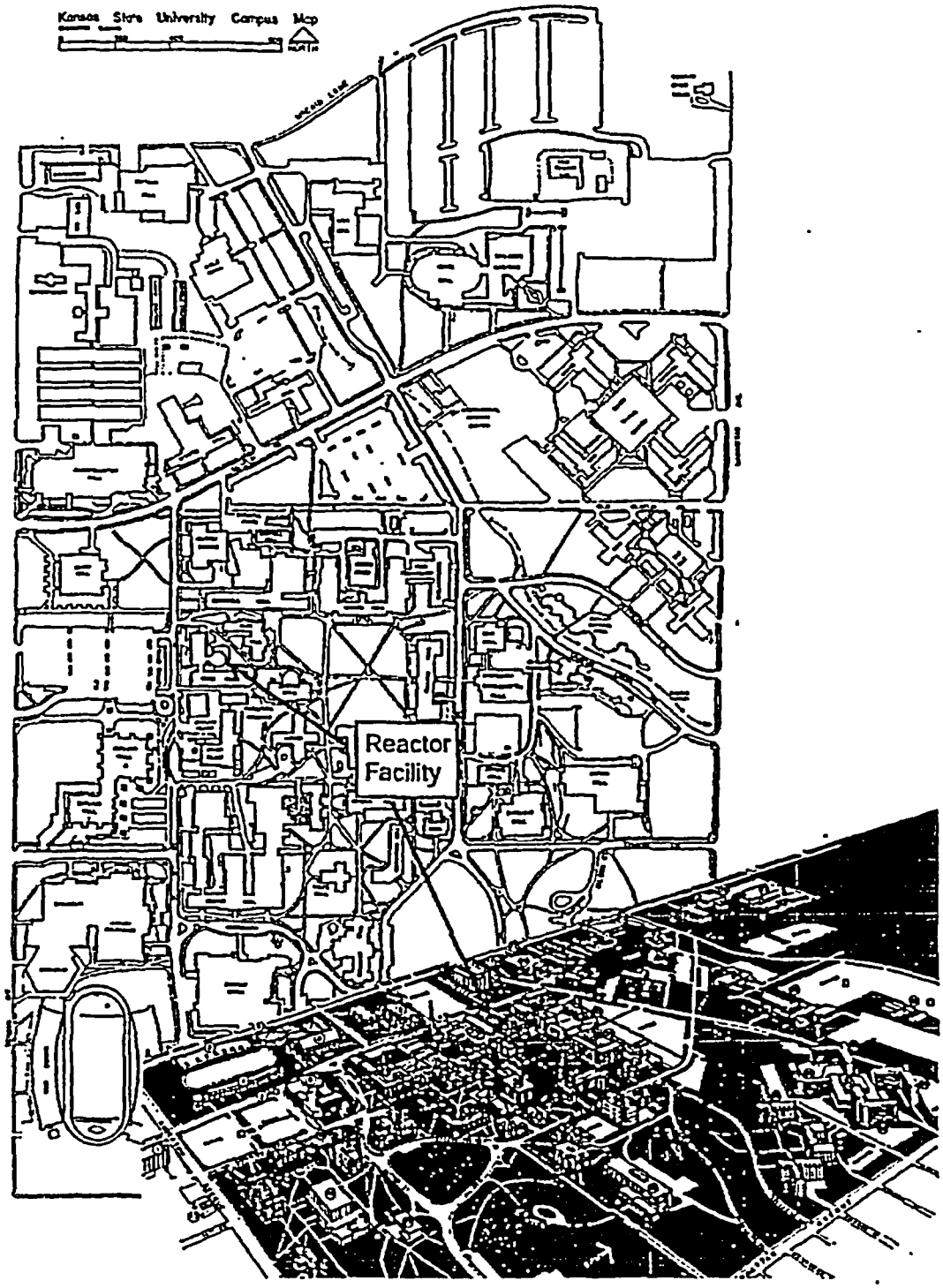


Figure 2.5, Reactor Facility Location within the Central Campus.

The first floor and basement plans for Ward Hall are displayed in Figures 2.6 and 2.7, with the "Reactor Bay" in Figure 2.8. For emergency planning purposes, the operations boundary encompasses the Reactor Facility, Room 110 of Ward Hall; the site boundary encompasses Ward Hall and any adjacent fenced areas, access to which and evacuation from which may be controlled by the management of the Reactor Facility.

2.1.2 Population Distribution

Manhattan, Kansas, home of Kansas State University and the TRIGA reactor, is in Riley County. Junction City, 29 km SW of Manhattan, is located in Geary County. The town of Wamego, 22 km E of Manhattan, is located in Pottawotamie County. Population characteristics of the three counties are described in Table 2.1. A more detailed tabulation of data for Riley County is found in Chapter 2 Appendix B, based on year 1990 census data (year 2000 census data is only partially complete).

Chapter 2, Appendix A, Figures 2A.4 and 2A.5 identifies Census blocks and 1990 population densities in zones extending to a maximum of 12 km and 8 km respectively from the TRIGA Reactor Facility. The census data do not allow for students resident in dormitories on the university campus. As noted above, year 2000 census data is not completely analyzed for distribution but shows a slight decrease in population densities over areas closest to KSU. Population data, prepared for use in accident analysis and listed by radial sector and azimuthal sector, on the sixteen-point compass, are listed in Table 2.2.

Table 2.1, Population Characteristics of Riley, Geary, and Pottawatomie Counties.

	Riley	Geary	Pottawatomie
Families	13,450 (12,262)*	8,191 (7,578)*	4,390 (4,931)*
Households	21,280 (22,137)*	10,676 (10,458)*	5,938 (6,771)*
Urban population	49,743	21,287	3,849
Rural population	17,405	9,166	12,279
Total population	67,139 (62,843)*	30,453 (27,947)*	16,128 (18,209)*

*1990 U.S. Census Data, Database C90STF1C;Year 2000 Data in Parenthesis

TABLE 2.2, Population Distribution Radially from the Reactor Site.

	Radial distance from reactor (km)							
Sector	0-1	1-2	2-4	4-6	6-8	8-10	10-12	Total
N	91	0	182	107	372	42	0	794
NNE	9	485	1741	27	104	2	0	2368
NE	245	201	2032	0	41	180	6	2705
ENE	272	237	1187	271	372	749	503	3591
E	607	969	2238	179	156	388	1709	6246
ESE	439	2134	1863	287	325	88	150	5286
SE	412	2431	2036	14	41	10	68	5012
SSE	586	1900	558	0	0	35	0	3079
S	654	3331	682	77	49	0	0	4793
SSW	254	947	849	96	351	449	11	2957
SW	610	437	1412	101	1924	409	683	5576
WSW	645	448	158	0	57	663	0	1971
W	1601	2568	3542	546	33	611	0	8901
WNW	233	610	3741	388	25	271	360	5628
NW	312	222	1096	11	17	161	78	1897
NNW	347	148	801	22	137	14	37	1506
Total	7317	17068	24118	2126	4004	4072	3605	62310

Source: 1990 U.S. census data, processed using the ArcView Graphical Information System.

2.2 Nearby Industrial, Transportation, & Military Facilities

2.2.1 Locations and Routes

Manhattan, Kansas, hosts light manufacturing and service industries, but no chemical plants or refineries, and no mining or significant quarrying operations. There are no missile sites near the city and no docks, ports, or railroad yards. Natural gas pipelines are for local service, as are electrical distribution lines. The Manhattan airport, which provides general aviation and feeder-airline service is located 9 km from the nuclear reactor. Marshall field, on the Fort Riley military reservation, and 22 km from the nuclear reactor, is a base for rotary-wing army aircraft.

2.2.2 Air Traffic

While the Manhattan airport is located less than 9 km from the nuclear reactor, actual and projected commercial or military aircraft movements are far less than 16,000 annually.

2.2.3 Analysis of Potential Accidents

There are no nearby industrial, transportation, or material facilities that could experience accidents affecting the safety of the nuclear reactor.

2.3 Meteorology

2.3.1 General and Local Climate[1]

Manhattan is located near the geographical center of the United States, and the middle of the temperate zone. For the most part, the city is located along the north bank of the Kansas River, which flows in an easterly direction, and the west bank of the Big Blue River, which flows southerly into the Kansas River. The river valleys, two to four miles wide, are bordered by rolling prairie uplands. Flooding is always a threat along river valleys, but construction of a levee around the city of Manhattan, and the Tuttle Creek Reservoir, on the Big Blue River, has largely alleviated the threat. Kansas State University, site of the TRIGA reactor, is located approximately 25 meters in elevation above the rivers and has never been under threat of flood. Even during the 500-year flood of 1993, which impacted the entire Midwest, the TRIGA reactor was never threatened. Flood waters never penetrated the reactor bay or the basement of Ward Hall. Seventy percent of the annual precipitation normally falls during the months of April through September. The rains of this period are usually of short duration, predominantly of the thunderstorm type. They occur more frequently during the nighttime and early morning hours than at other times of the day. Excessive precipitation rates may occur

1.
[1] Adapted from *Local Climatological Data for Topeka Kansas*, supplied by the National Climatic Data Center, National Oceanic and Atmospheric Administration, Asheville, North Carolina, ISSN 0198-2192 (1995).

with warm-season thunderstorms. Rainfall accumulations of over eight inches in 24 hours have occurred. Tornadoes have occurred in the area on several occasions and caused severe damage and numerous injuries.

Individual summers show wide departures from average conditions. Hottest summers may produce temperatures of 100°F or higher on more than 50 days. On the other hand, 25 percent of the summers pass with two or fewer 100°F days. Similarly, precipitation has shown a wide range for June, July, and August, varying from under three inches to more than 27 inches during the three months. Summers are hot with low relative humidity and persistent southerly winds. Oppressive warm periods with high relative humidity are usually of short duration.

Winter temperatures average about 45°F cooler than summer. Cold spells are seldom prolonged. Only on rare occasions do daytime temperatures fail to rise above freezing. Winter precipitation is often in the form of snow, sleet, or glaze, but storms of such severity to prevent normal movement of traffic or to interfere with scheduled activity are not common.

In the transitional spring and fall seasons, the numerous days of fair weather are interspersed with short intervals of stormy weather. Strong, blustery winds are quite common in late winter and spring. Autumn is characteristically a season of warm days, cool nights, and infrequent precipitation, with cold air inversions gradually increasing in intensity as the season progresses.

Nearly all crops of the temperate zone can be produced in eastern Kansas. Wheat and other small grains, clover, soybeans, fruit, and berries do well, and the area supports extensive dairy and beef cattle operations.

Based on the 1951-1980 period, the average first occurrence of freezing in the fall is October 14, and the average last occurrence in the spring is April 21.

Historical data for severe weather phenomena during the 50-year period ending in 1995 are presented in Table 2.3. Except as indicated, data are from the National Climatic Data Center, National Oceanic and Atmospheric Administration.

2.3.2 Site Meteorology

Multivariate frequency distributions for wind speed, wind direction, and atmospheric stability (Pasquill categories A through G) are listed in Chapter 2, Appendix C. These data were processed from 1991-1996 data and were prepared especially for this report by the National Climatic Data Center, U.S. Oceanic and Atmospheric Administration. They may be used in conjunction with population data presented in Section 2.1.2 to evaluate potential radiation doses associated with hypothetical accidental releases of radionuclides into the atmosphere. The wind-speed data are summarized in a wind rose presented in Figure 2.9. The shading moves from calmest at center to strongest at the isobar lines near the edges of the construct. Using the scale and measuring from the perimeter of the Calm circle, fractional frequencies of occurrence for any direction on the sixteen-point compass may be derived. Directions from which the wind arrives are shown.

Table 2.3, Severe Weather Phenomena for Topeka, Kansas.

Phenomenon	Magnitude	Date
Mean wind speed and direction	9.3 mph at 180°	—
Maximum 2-minute wind speed	44 mph at 340°	June, 1984
Maximum 5-second wind speed	66 mph at 350°	June, 1984
Maximum 24-hour rainfall[a]	5.52 in.	June, 1967
Maximum 24-hour snowfall[a]	15.2 in.	February, 1971
Maximum snow depth	18 in.	March, 1960

[a]Record-breaking incidences in Manhattan are 6.28 in. rainfall in June, 1977 and 18.0 in. snowfall in February, 1900 [Goodin, et al., 1995].

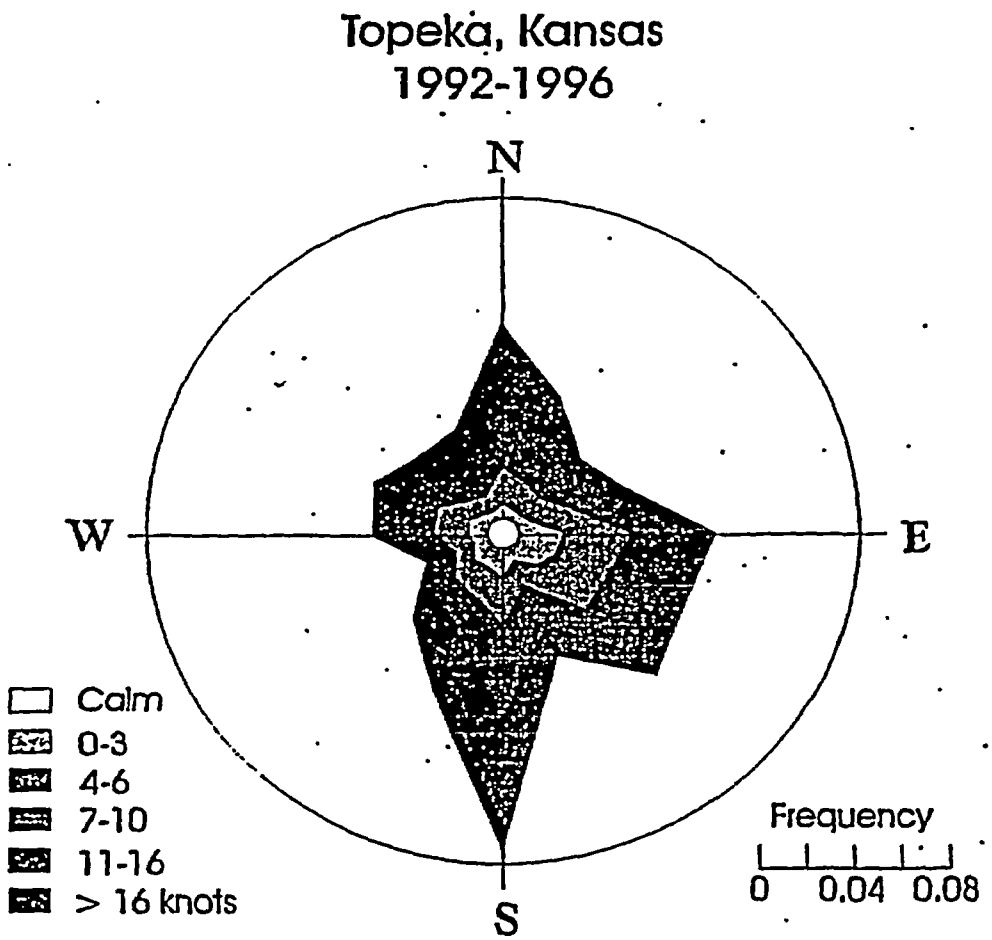


Figure 2.9, Wind-Speed Frequency Data for Topeka, Kansas. National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration

2.4 Geology, Seismology, and Geotechnical Engineering

Test drillings in the vicinity of the reactor reveal a thin layer of topsoil with varying levels of glacial deposits overlying bedrock, limestone and shale. Subsurface water is encountered 18 to 35 feet below existing grade in thin sand and gravel layers near the bedrock surface. The sand and gravel aquifer acts as a temporary trap that could, if charged with radioactively contaminated water, eventually discharge radionuclides into the water-bearing veins underlying the Blue River valley. The city of Manhattan, as well as the university, draws water from wells in this same river valley. It is possible, but highly improbable, that a gross leak of soluble radioactive materials into the ground at the reactor site could eventually deliver contamination to these wells. That such a leak could occur undetected, unchecked, and in sufficient magnitude to produce measurable contamination in a potable water supply appears to have a vanishingly small probability. Operation, surveillance, monitoring, inspection, and auditing procedures in place at the Reactor Facility assure that encapsulated sources are regularly inspected and monitored, and that unencapsulated, soluble radioactive materials are not held in inventory.

2.4.1 Regional Geology

The general physiography of the region is illustrated in Chapter 2, Appendix A, 2A.6. The city of Manhattan, in Riley County, is located in the Flint Hills uplands. To the east is a glaciated region, to the west the Smoky Hills.

2.4.2 Site Geology

As illustrated in Chapter 2, Appendix A, Figures 2A.7 and 2A.8, the university is located in a region of quaternary glacial-fluvial unconsolidated sediment, which is characteristic of much of the city of Manhattan. To the east and south, along the beds of the Big Blue and Kansas Rivers is also unconsolidated terrace deposits and alluvium beds of sediment. The Flint Hills surrounding the city are composed of limestones and shales of the Permian System, shales interleaved with limestone, with regions of unconsolidated sediment along streams.

2.4.3 Seismicity

As seen in Figure 2.10, Manhattan, Kansas is located in seismic risk zone 2, so classified because of several modified Mercali VII-VIII earthquakes that have occurred in the past. Thirty felt earthquakes with epicenters in Kansas have been documented since 1867. These are illustrated in Figure 2.11.

Earthquakes of modified Mercali intensity greater than IV are identified in Figure 2.11. It appears that quakes of intensity VI or greater occur irregularly at intervals of 20 to 40 years.

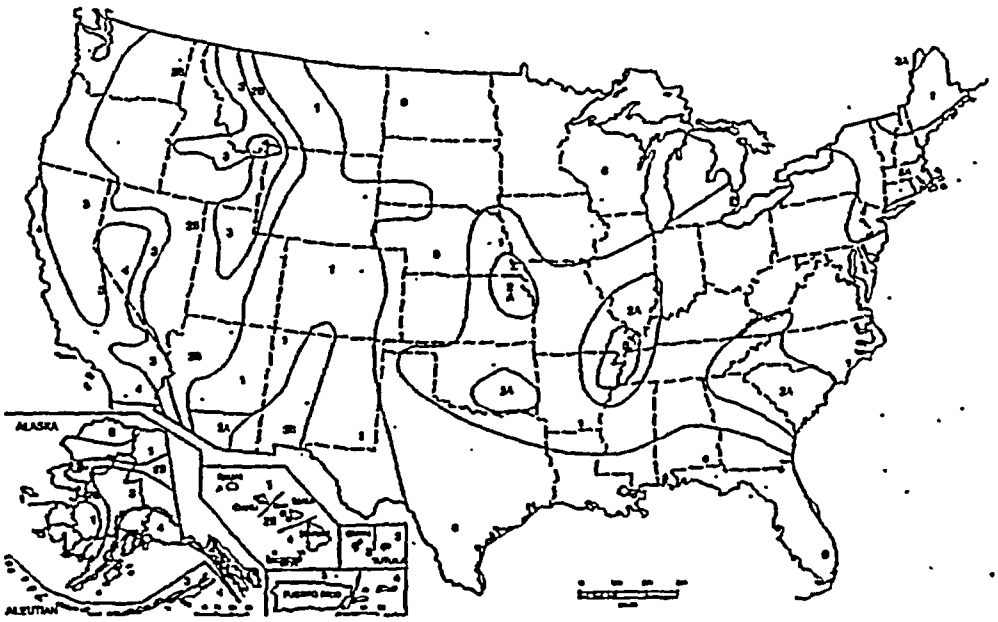
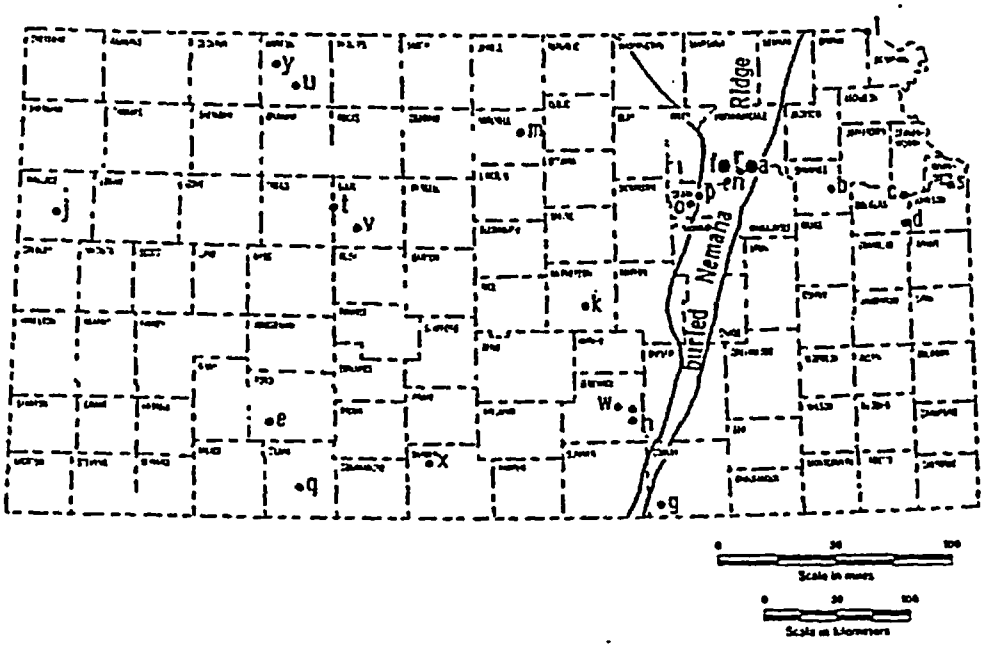


Figure 2.10, U.S. Seismic Risk Map Source: 1997 Uniform Building Code.

EARTHQUAKES IN KANSAS



Explanation

a	1867	VIII	f	1906	VII	k	1927	V	p	1929	V	u	1933	V
b	1875	V	g	1907	IV	l	1927	VI	q	1929	V	v	1942	IV
c	1881	III	h	1919	IV	m	1928	IV	r	1929	V	w	1948	IV
d	1903	II	i	1919	IV	n	1929	V	s	1931	VI	x	1956	VI
e	1904	IV	j	1926	?	o	1929	V	t	1932	VI	y	1961	V

Location and dates of earthquakes in Kansas during the past 110 years. The number following the date is the earthquake intensity on the Modified Mercalli Scale.

Figure 2.11, Recorded Earthquakes in Kansas.
Source: DuBois and Wilson, 1978.

2.4.4 Maximum Earthquake Potential

According to Steeples et al. [1988], the most serious of the recorded earthquakes in Kansas were the intensity VII events that occurred in 1867 and 1906 in the vicinity of Manhattan. Although the structures responsible for these two earthquakes have not been positively identified, the source of seismic activity appears to be movement on the Nemaha Ridge, a buried Precambrian granitic uplift, or the Humboldt Fault along the eastern boundary of the Nemeha Ridge. The Midcontinent Geophysical Anomaly (MGA) is also known to be an important structural feature in the region. The MGA extends from Lake Superior southwestward through central Kansas into Oklahoma. Surface structures associated with the MGA are present in the Manhattan area. These are the Abilene Anticline and the Riley County kimberlite intrusives. Work is under way by the Kansas Geologic Survey, begun under the support of the Nuclear Regulatory Commission, to gain a better understanding of the seismicity of the region and the link between the seismicity and the tectonic setting of the region. DuBois and Wilson [1978] describe the most severe earthquake recorded in Kansas in Figure 2.11; an isoseismal map of the earthquake is illustrated in Figure 2.13

Intensity	**MM VII-VIII**
Time	**2:30 p.m., 24 April 1867**
Epicenter	**Lat 39°10', Long 96°18' near Alma, Kansas, 32 km SE of Manhattan**
Felt area	**300,000 sq. mi.**

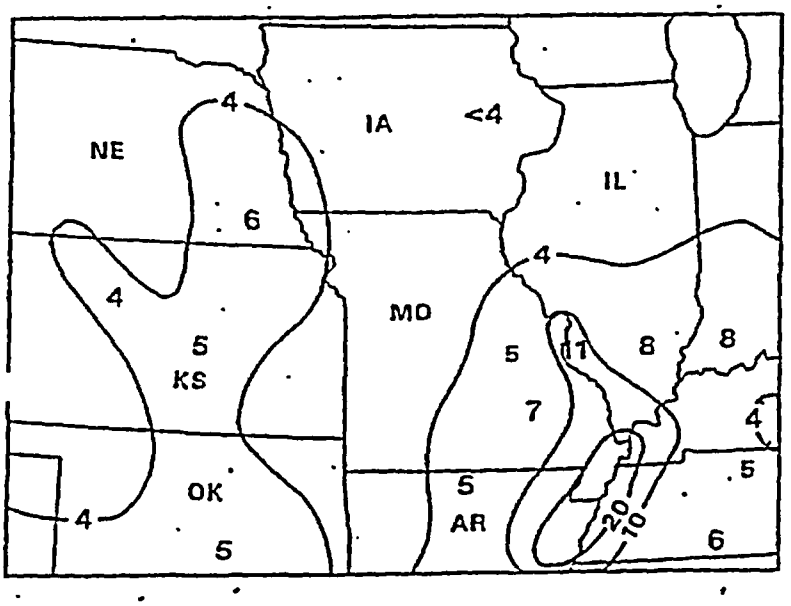


Figure 2.12, Ground-Acceleration Map for the Region.
Source: Underwood, 1990.

ISOSEISMAL MAP OF THE
APRIL 24, 1867 EARTHQUAKE IN KANSAS

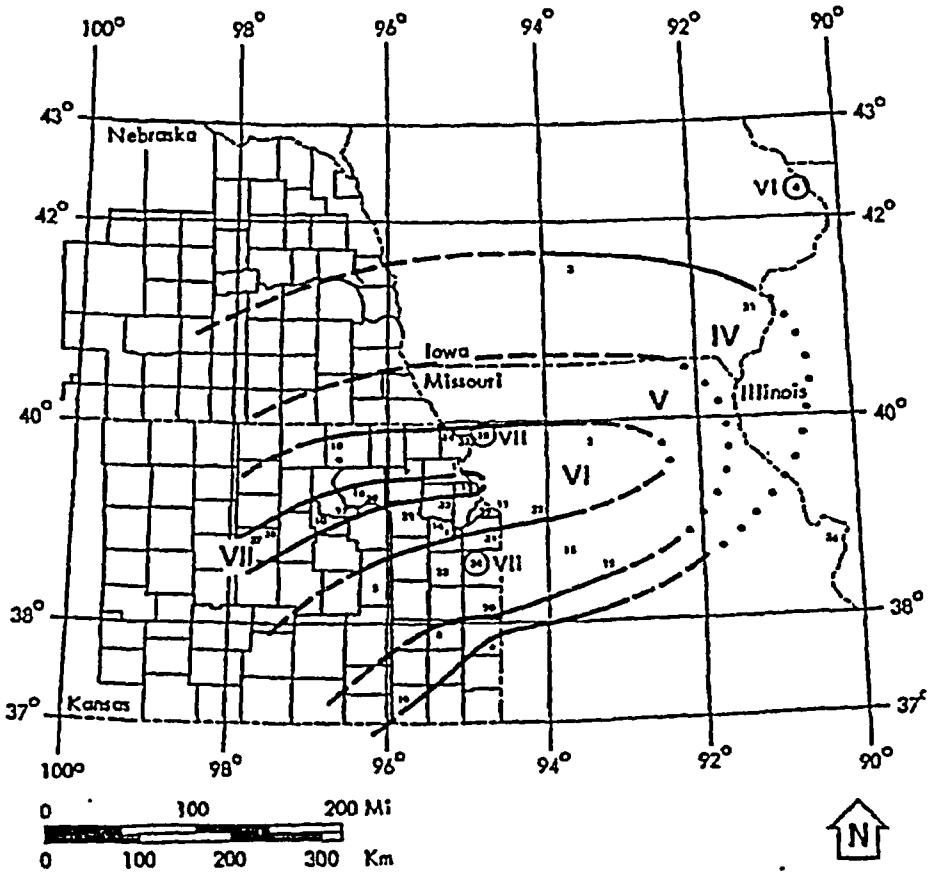


Figure 2.13, Map of the Most Severe Earthquake Recorded in the Region.

2.4.5 Vibratory Ground Motion

Maximum accelerations have been estimated using methods of Algermissen and Perkins [1976]. Figure 2.13, presented by Underwood [1990] from the data of Algermisson and Perkins, is a map of potential maxima in the region. Contour lines represent a probability of no more than 10 percent in 50 years of exceeding in an acceleration expressed as percentage of the acceleration of gravity.

2.4.6 Surface Faulting

There are no known faults within 8 km of the TRIGA reactor site. Historically reported earthquakes in the region are previously enumerated.

2.4.7 Liquefaction Potential

The phenomenon of soil liquefaction is associated primarily with medium to fine grained saturated *cohesionless* soils [Das, 1993]. As indicated in the soil map presented in Chapter 2, Appendix A, Figure 2A.8, such conditions are not met at the site of the TRIGA reactor. Soils are loams, silty loams, and clay loams. Though the reactor is located some 25 meters above the levels of local rivers, and though groundwater depths do vary, sandy soils in the saturated zone are not expected at this site; hence, the liquefaction potential of local soils is quite minimal.

2.5 Hydrology

This topic was addressed in the Hazards Summary Report [Clack et al., 1961] reviewed by the Atomic Energy Commission prior to their granting the original 40-year license for operation of the reactor. Extracts from that report are repeated here.

The reactor site is located at an elevation of 1,082 feet above mean sea level, approximately 65 feet above the highest recorded flooding in the area. Average annual rainfall amounts to about 32 inches in the Manhattan area, with April, May, and June ordinarily being the wetter months.

Terrain features are illustrated in Chapter 2, Appendix A, Figure 2A.8. Soils in the region are primarily silt and clay loams of various classifications. The reactor site is in a region of the university campus that is convex upward, a circumstance that minimizes the probability of local flooding. Surface water not absorbed drains into storm sewers on the campus. The storm sewer system runs through the city of Manhattan and discharges into the Kansas River south of the city. The Kansas River flows eastward through Topeka and Lawrence, joining the Missouri River at Kansas City.

2.6 Bibliography

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Appendices

A. Color Plates

B. Census Data

C. Multivariate Meteorology Frequency Distributions

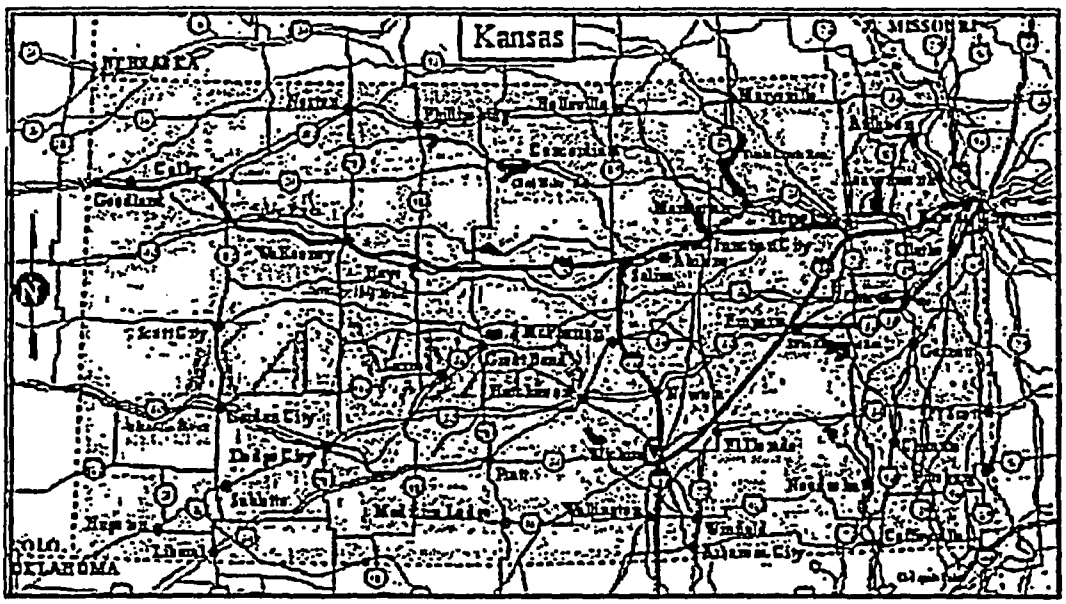

Figure 2A.1, Major Roads in Kansas.

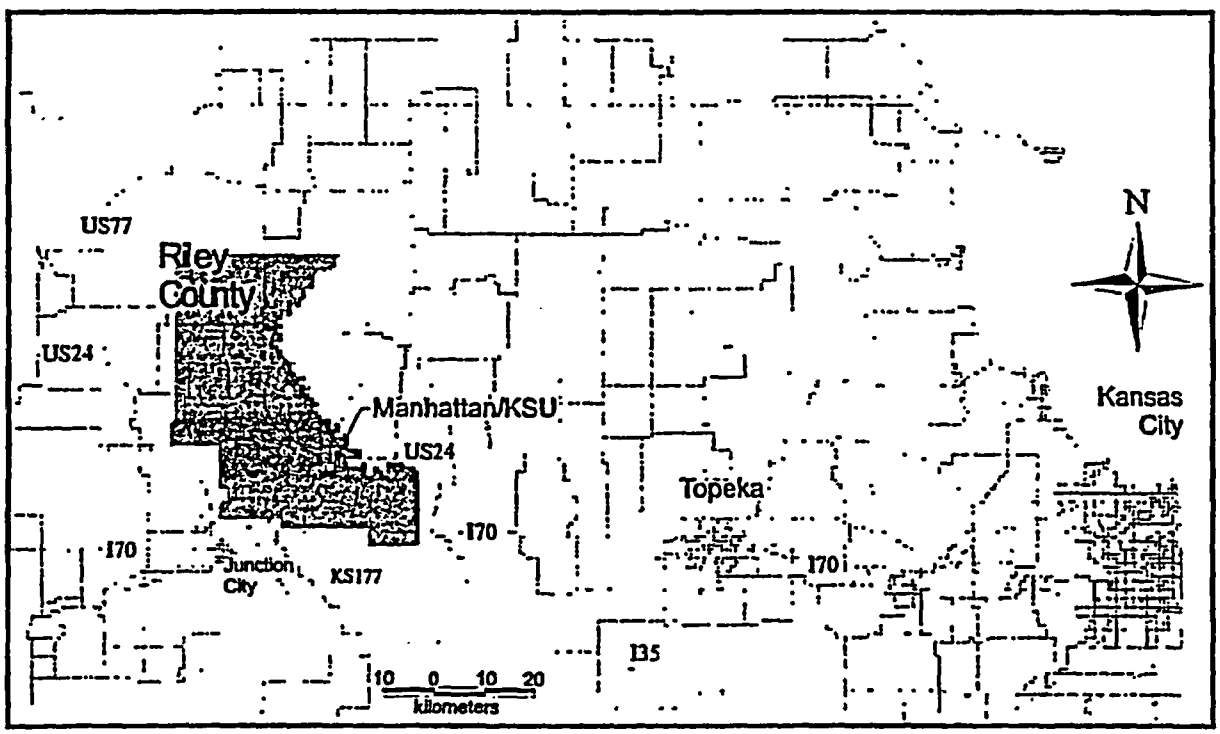

Figure 2A.2, Highway access to Manhattan from Topeka and Kansas City.

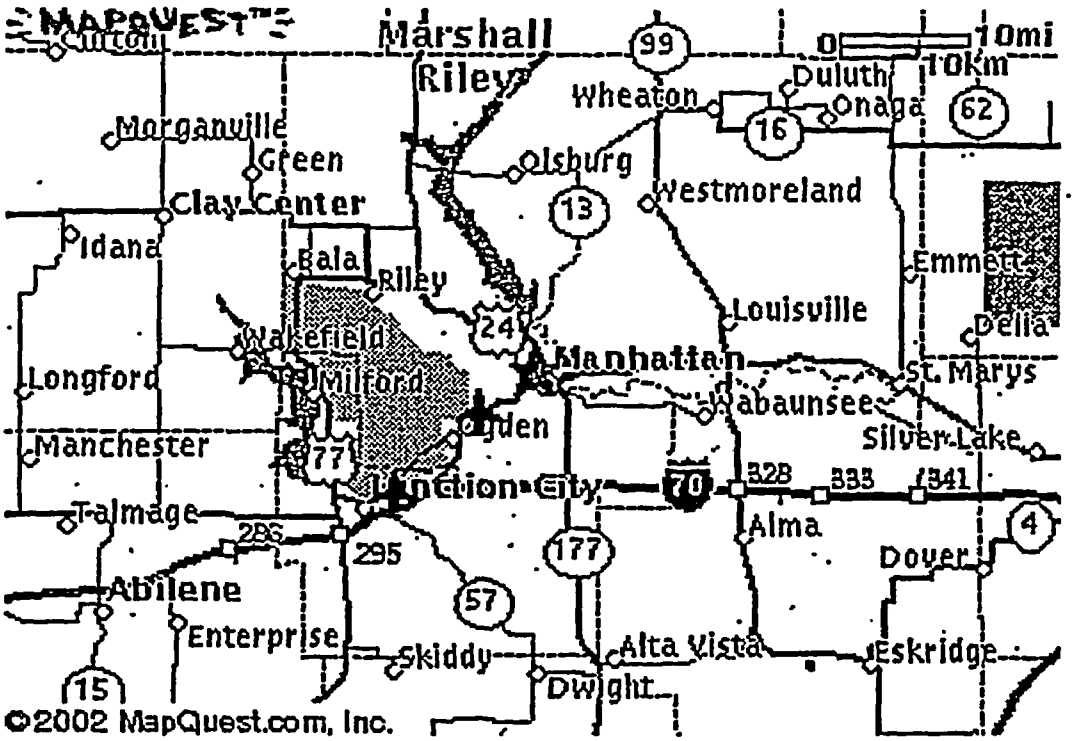

Figure 2A.3, The city of Manhattan, Kansas State University, the TRIGA reactor and nearby transportation features and waterways

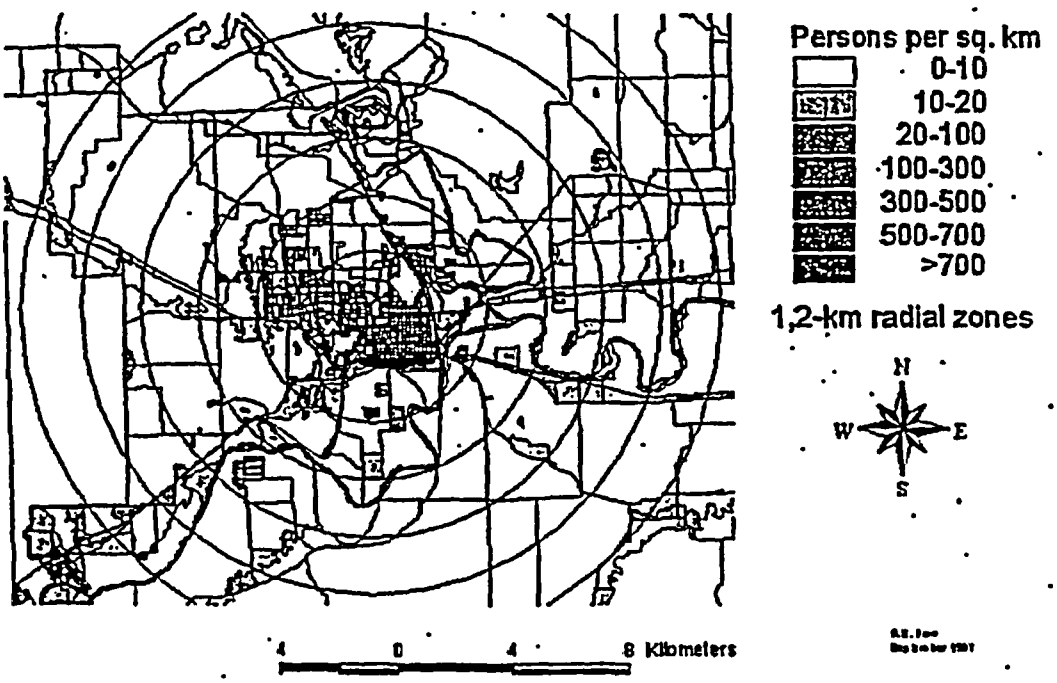

Figure 2A.4, Population Density to About 12 Kilometers from the Facility.

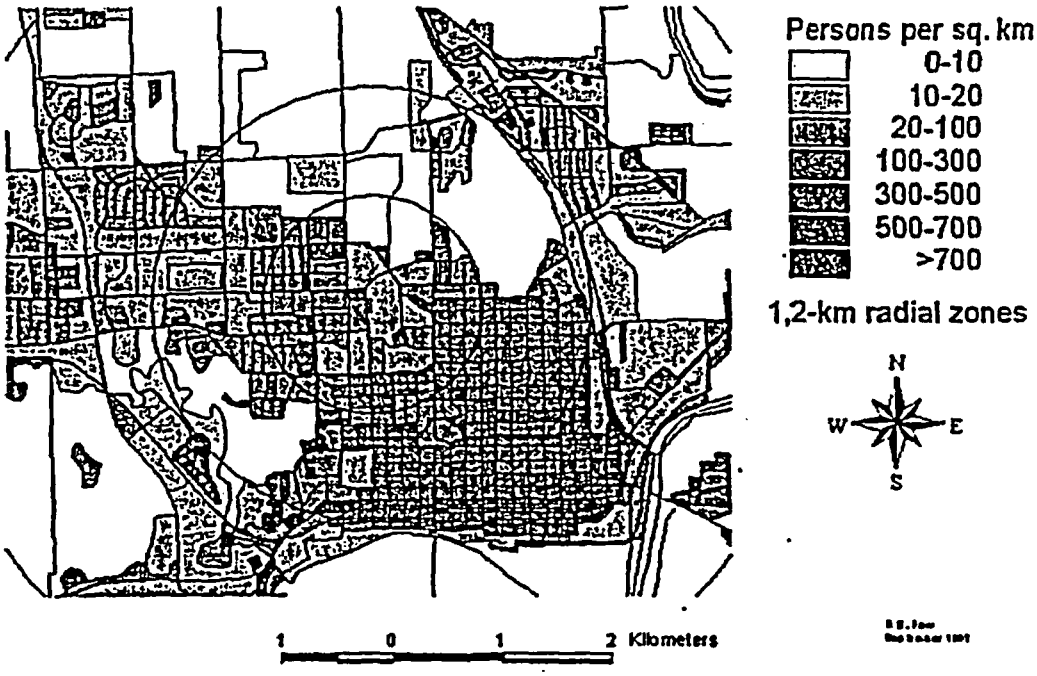

Figure 2A.5, Population Density to About 4 Kilometers from the Facility.

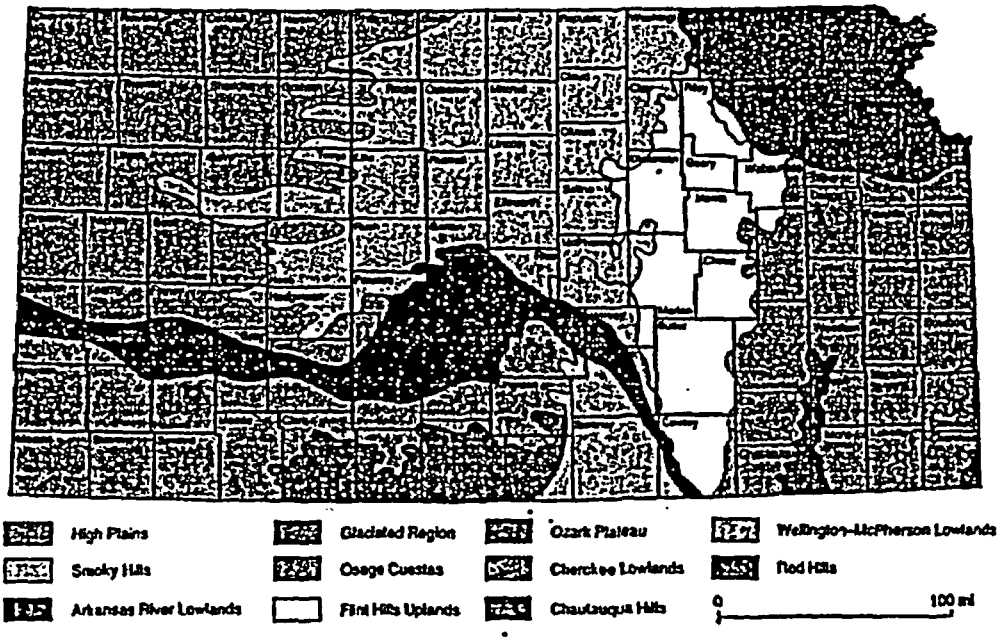

Figure 2A.6, General Physiography of Kansas.
Source: Goodin et al., 1995

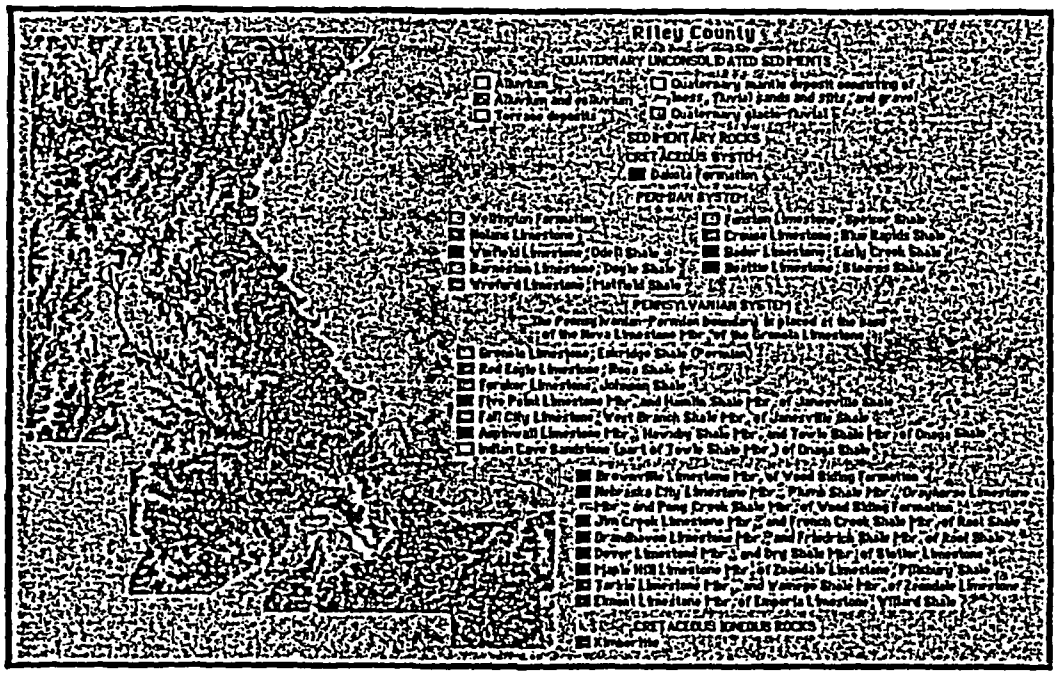

Figure 2A.7, Physiography of Riley County.
Source: Kansas Geologic Survey

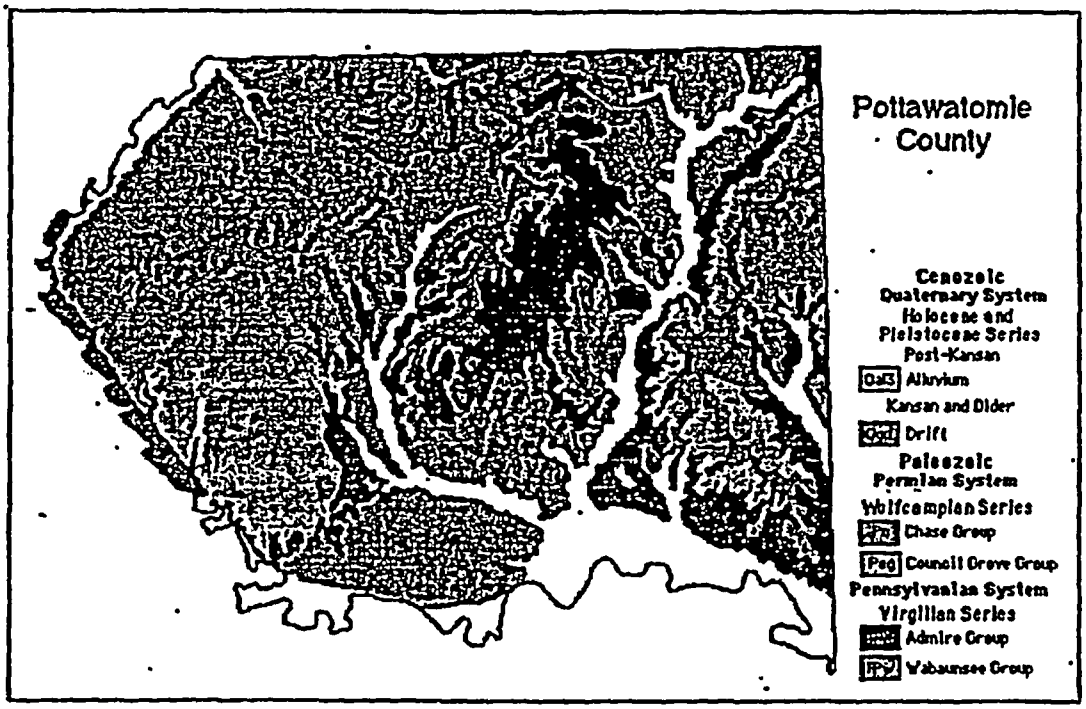

Figure 2A.8: Physiography of Pottawatomie County.
Source: Kansas Geologic Survey

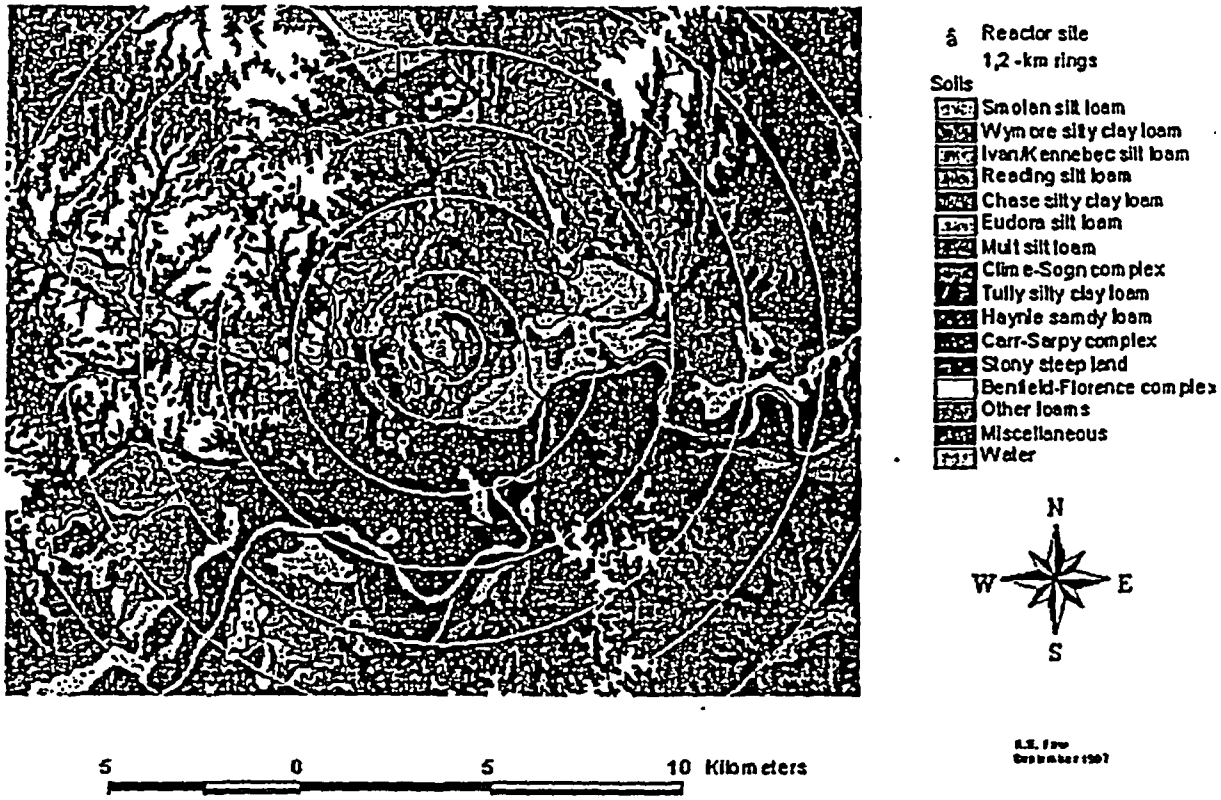

Figure 2A.8, Terrain Surrounding the Facility.

Chapter 2 Appendix B

Population Distribution in Riley County, Kansas

1990 US Census Data, Database: C90STF3B

ZIP Code 66502: ZIP=66502

PERSONS
Universe: Persons
Total.....50229
FAMILIES
Universe: Families
Total.....10598
HOUSEHOLDS
Universe: Households
Total.....17977
URBAN AND RURAL
Universe: Persons
Urban:
  Inside urbanized area.....0
  Outside urbanized area.....37744
Rural:
  Farm.....302
  Nonfarm.....12183
SEX
Universe: Persons
Male.....26004
Female.....24225
RACE
Universe: Persons
White.....45178
Black.....2296
American Indian, Eskimo, or Aleut.....303
Asian or Pacific Islander.....1899
Other race.....553
AGE
Universe: Persons
Under 1 year.....645
1 and 2 years.....1306
3 and 4 years.....1163
5 years.....546
6 years.....689
7 to 9 years.....1537
10 and 11 years.....1095
12 and 13 years.....1032
14 years.....465
15 years.....467

16 years.....429
17 years.....395
18 years.....1486
19 years.....3133
20 years.....3277
21 years.....3176
22 to 24 years.....5531
25 to 29 years.....5114
30 to 34 years.....3749
35 to 39 years.....3178
40 to 44 years.....2771
45 to 49 years.....1867
50 to 54 years.....1187
55 to 59 years.....1146
60 and 61 years.....569
62 to 64 years.....660
65 to 69 years.....1186
70 to 74 years.....850
75 to 79 years.....648
80 to 84 years.....538
85 years and over.....394

SCHOOL ENROLLMENT AND TYPE OF SCHOOL

Universe: Persons 3 years and over

Enrolled in preprimary school:
  Public school.....634
  Private school.....206
Enrolled in elementary or high school:
  Public school.....5787
  Private school.....219
Enrolled in college:
  Public school.....17438
  Private school.....277
Not enrolled in school.....23717

EARNINGS IN 1989

Universe: Households

With earnings.....15831
No earnings.....2146

WAGE OR SALARY INCOME IN 1989

Universe: Households

With wage or salary income.....15494
No wage or salary income.....2483

AGGREGATE WAGE OR SALARY INCOME IN 1989

Universe: Households

Total.....393
136419

PER CAPITA INCOME IN 1989

Universe: Persons

Per capita income in 1989.....10718

Chapter 2 Appendix C
Meteorological Frequency Distributions

Table B-1. Wind rose data excluding calms, for Topeka, Kansas, 1992-1996.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.00740	0.01608	0.02543	0.03054	0.00698	0.00125
NNE	0.00523	0.01359	0.02044	0.01683	0.00299	0.00000
NE	0.00515	0.01060	0.01633	0.00723	0.00099	0.00000
ENE	0.00693	0.01695	0.01609	0.00785	0.00025	0.00000
E	0.01956	0.02830	0.02556	0.00897	0.00000	0.00000
ESE	0.01822	0.02294	0.02469	0.00997	0.00012	0.00000
SE	0.01371	0.02880	0.02942	0.01247	0.00000	0.00000
SSE	0.00649	0.01309	0.02019	0.01296	0.00050	0.00000
S	0.01333	0.02244	0.04189	0.04837	0.00835	0.00050
SSW	0.00998	0.01396	0.01820	0.02368	0.00349	0.00087
SW	0.01018	0.01084	0.01010	0.01334	0.00174	0.00050
WSW	0.00664	0.00848	0.00685	0.00473	0.00050	0.00000
W	0.00798	0.01521	0.01296	0.00984	0.00187	0.00075
WNW	0.00927	0.01421	0.01434	0.01234	0.00212	0.00037
NW	0.00553	0.00910	0.01571	0.01209	0.00124	0.00000
NNW	0.00415	0.00810	0.01433	0.01259	0.00411	0.00175

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-2. Wind rose cumulative data including calms, for Topeka, Kansas, 1992-1996.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.01338	0.02946	0.05489	0.08543	0.09241	0.09366
NNE	0.01121	0.02480	0.04524	0.06207	0.06506	0.06506
NE	0.01113	0.02173	0.03806	0.04529	0.04628	0.04628
ENE	0.01291	0.02986	0.04595	0.05380	0.05405	0.05405
E	0.02554	0.05384	0.07940	0.08837	0.08837	0.08837
ESE	0.02420	0.04714	0.07183	0.08180	0.08192	0.08192
SE	0.01969	0.04849	0.07791	0.09038	0.09038	0.09038
SSE	0.01247	0.02556	0.04575	0.05871	0.05921	0.05921
S	0.01931	0.04175	0.08364	0.13201	0.14036	0.14086
SSW	0.01596	0.02992	0.04812	0.07180	0.07529	0.07616
SW	0.01616	0.02700	0.03710	0.05044	0.05218	0.05268
WSW	0.01262	0.02110	0.02795	0.03268	0.03318	0.03318
W	0.01396	0.02917	0.04213	0.05197	0.05384	0.05459
WNW	0.01525	0.02946	0.04380	0.05614	0.05826	0.05863
NW	0.01151	0.02061	0.03632	0.04841	0.04965	0.04965
NNW	0.01013	0.01823	0.03256	0.04515	0.04926	0.05101

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-3. Relative frequency for winds under Pasquill stability category A, for Topeka, Kansas.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.00045	0.00012	0.00000	0.00000	0.00000	0.00000
NNE	0.00013	0.00025	0.00000	0.00000	0.00000	0.00000
NE	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
ENE	0.00013	0.00025	0.00000	0.00000	0.00000	0.00000
E	0.00013	0.00025	0.00000	0.00000	0.00000	0.00000
ESE	0.00077	0.00037	0.00000	0.00000	0.00000	0.00000
SE	0.00033	0.00062	0.00000	0.00000	0.00000	0.00000
SSE	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
S	0.00019	0.00000	0.00000	0.00000	0.00000	0.00000
SSW	0.00007	0.00012	0.00000	0.00000	0.00000	0.00000
SW	0.00026	0.00012	0.00000	0.00000	0.00000	0.00000
WSW	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
W	0.00007	0.00012	0.00000	0.00000	0.00000	0.00000
WNW	0.00026	0.00012	0.00000	0.00000	0.00000	0.00000
NW	0.00026	0.00012	0.00000	0.00000	0.00000	0.00000
NNW	0.00020	0.00037	0.00000	0.00000	0.00000	0.00000

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-4. Relative frequency for winds under Pasquill stability category B, for Topeka, Kansas.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.00089	0.00100	0.00062	0.00000	0.00000	0.00000
NNE	0.00039	0.00087	0.00062	0.00000	0.00000	0.00000
NE	0.00102	0.00087	0.00100	0.00000	0.00000	0.00000
ENE	0.00045	0.00112	0.00100	0.00000	0.00000	0.00000
E	0.00122	0.00287	0.00150	0.00000	0.00000	0.00000
ESE	0.00108	0.00112	0.00125	0.00000	0.00000	0.00000
SE	0.00143	0.00125	0.00137	0.00000	0.00000	0.00000
SSE	0.00067	0.00075	0.00112	0.00000	0.00000	0.00000
S	0.00065	0.00187	0.00262	0.00000	0.00000	0.00000
SSW	0.00067	0.00137	0.00062	0.00000	0.00000	0.00000
SW	0.00038	0.00025	0.00050	0.00000	0.00000	0.00000
WSW	0.00036	0.00137	0.00037	0.00000	0.00000	0.00000
W	0.00067	0.00075	0.00037	0.00000	0.00000	0.00000
WNW	0.00102	0.00087	0.00112	0.00000	0.00000	0.00000
NW	0.00039	0.00087	0.00075	0.00000	0.00000	0.00000
NNW	0.00032	0.00062	0.00075	0.00000	0.00000	0.00000

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-5. Relative frequency for winds under Pasquill stability category C, for Topeka, Kansas.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.00059	0.00137	0.00337	0.00112	0.00025	0.00000
NNE	0.00029	0.00125	0.00511	0.00062	0.00012	0.00000
NE	0.00022	0.00062	0.00312	0.00037	0.00012	0.00000
ENE	0.00027	0.00224	0.00362	0.00050	0.00000	0.00000
E	0.00091	0.00175	0.00386	0.00087	0.00000	0.00000
ESE	0.00040	0.00100	0.00362	0.00112	0.00000	0.00000
SE	0.00068	0.00212	0.00449	0.00100	0.00000	0.00000
SSE	0.00043	0.00125	0.00461	0.00137	0.00000	0.00000
S	0.00040	0.00212	0.00898	0.00337	0.00112	0.00000
SSW	0.00035	0.00175	0.00374	0.00162	0.00037	0.00000
SW	0.00054	0.00100	0.00249	0.00112	0.00012	0.00000
WSW	0.00037	0.00075	0.00162	0.00037	0.00000	0.00000
W	0.00031	0.00137	0.00337	0.00062	0.00012	0.00000
WNW	0.00054	0.00212	0.00337	0.00037	0.00000	0.00000
NW	0.00015	0.00125	0.00187	0.00050	0.00012	0.00000
NNW	0.00028	0.00112	0.00137	0.00050	0.00012	0.00000

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-6. Relative frequency for winds under Pasquill stability category D, for Topeka, Kansas.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.00121	0.00711	0.01770	0.02942	0.00673	0.00125
NNE	0.00081	0.00424	0.01184	0.01621	0.00287	0.00000
NE	0.00088	0.00362	0.00997	0.00686	0.00087	0.00000
ENE	0.00124	0.00598	0.01097	0.00735	0.00025	0.00000
E	0.00194	0.00910	0.01870	0.00810	0.00000	0.00000
ESE	0.00114	0.00636	0.01683	0.00885	0.00012	0.00000
SE	0.00132	0.00686	0.01970	0.01147	0.00000	0.00000
SSE	0.00096	0.00299	0.01047	0.01159	0.00050	0.00000
S	0.00097	0.00449	0.02082	0.04500	0.00723	0.00050
SSW	0.00036	0.00237	0.00898	0.02206	0.00312	0.00087
SW	0.00024	0.00112	0.00424	0.01222	0.00162	0.00050
WSW	0.00012	0.00125	0.00299	0.00436	0.00050	0.00000
W	0.00042	0.00299	0.00436	0.00922	0.00175	0.00075
WNW	0.00059	0.00337	0.00648	0.01197	0.00212	0.00037
NW	0.00035	0.00224	0.01072	0.01159	0.00112	0.00000
NNW	0.00030	0.00324	0.00997	0.01209	0.00399	0.00175

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-7. Relative frequency for winds under Pasquill stability category E, for Topeka, Kansas.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.00000	0.00212	0.00374	0.00000	0.00000	0.00000
NNE	0.00000	0.00287	0.00287	0.00000	0.00000	0.00000
NE	0.00000	0.00237	0.00224	0.00000	0.00000	0.00000
ENE	0.00000	0.00362	0.00050	0.00000	0.00000	0.00000
E	0.00000	0.00648	0.00150	0.00000	0.00000	0.00000
ESE	0.00000	0.00474	0.00299	0.00000	0.00000	0.00000
SE	0.00000	0.00760	0.00386	0.00000	0.00000	0.00000
SSE	0.00000	0.00274	0.00399	0.00000	0.00000	0.00000
S	0.00000	0.00536	0.00947	0.00000	0.00000	0.00000
SSW	0.00000	0.00262	0.00486	0.00000	0.00000	0.00000
SW	0.00000	0.00274	0.00287	0.00000	0.00000	0.00000
WSW	0.00000	0.00112	0.00187	0.00000	0.00000	0.00000
W	0.00000	0.00337	0.00486	0.00000	0.00000	0.00000
WNW	0.00000	0.00162	0.00337	0.00000	0.00000	0.00000
NW	0.00000	0.00150	0.00237	0.00000	0.00000	0.00000
NNW	0.00000	0.00150	0.00224	0.00000	0.00000	0.00000

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-8. Relative frequency for winds under Pasquill stability category F, for Topeka, Kansas.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.00108	0.00436	0.00000	0.00000	0.00000	0.00000
NNE	0.00162	0.00411	0.00000	0.00000	0.00000	0.00000
NE	0.00144	0.00312	0.00000	0.00000	0.00000	0.00000
ENE	0.00126	0.00374	0.00000	0.00000	0.00000	0.00000
E	0.00302	0.00785	0.00000	0.00000	0.00000	0.00000
ESE	0.00329	0.00935	0.00000	0.00000	0.00000	0.00000
SE	0.00318	0.01035	0.00000	0.00000	0.00000	0.00000
SSE	0.00125	0.00536	0.00000	0.00000	0.00000	0.00000
S	0.00316	0.00860	0.00000	0.00000	0.00000	0.00000
SSW	0.00176	0.00573	0.00000	0.00000	0.00000	0.00000
SW	0.00159	0.00561	0.00000	0.00000	0.00000	0.00000
WSW	0.00101	0.00399	0.00000	0.00000	0.00000	0.00000
W	0.00133	0.00661	0.00000	0.00000	0.00000	0.00000
WNW	0.00168	0.00611	0.00000	0.00000	0.00000	0.00000
NW	0.00159	0.00312	0.00000	0.00000	0.00000	0.00000
NNW	0.00066	0.00125	0.00000	0.00000	0.00000	0.00000

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-9. Relative frequency for winds under Pasquill stability category G, for Topeka, Kansas.

	Wind speed (knots)					
Direction	0-3	4-6	7-10	11-16	17-21	21-
N	0.00318	0.00000	0.00000	0.00000	0.00000	0.00000
NNE	0.00199	0.00000	0.00000	0.00000	0.00000	0.00000
NE	0.00159	0.00000	0.00000	0.00000	0.00000	0.00000
ENE	0.00358	0.00000	0.00000	0.00000	0.00000	0.00000
E	0.01234	0.00000	0.00000	0.00000	0.00000	0.00000
ESE	0.01154	0.00000	0.00000	0.00000	0.00000	0.00000
SE	0.00677	0.00000	0.00000	0.00000	0.00000	0.00000
SSE	0.00318	0.00000	0.00000	0.00000	0.00000	0.00000
S	0.00796	0.00000	0.00000	0.00000	0.00000	0.00000
SSW	0.00677	0.00000	0.00000	0.00000	0.00000	0.00000
SW	0.00717	0.00000	0.00000	0.00000	0.00000	0.00000
WSW	0.00478	0.00000	0.00000	0.00000	0.00000	0.00000
W	0.00518	0.00000	0.00000	0.00000	0.00000	0.00000
WNW	0.00518	0.00000	0.00000	0.00000	0.00000	0.00000
NW	0.00279	0.00000	0.00000	0.00000	0.00000	0.00000
NNW	0.00239	0.00000	0.00000	0.00000	0.00000	0.00000

Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

Table B-10. Relative frequency of occurrence of Pasquill stability categories, for Topeka, Kansas.

	Pasquill stability category						
Direction	A	B	C	D	E	F	G
N	0.00057	0.00251	0.00670	0.06342	0.00586	0.00544	0.00318
NNE	0.00038	0.00188	0.00739	0.03597	0.00574	0.00573	0.00199
NE	0.00000	0.00289	0.00445	0.02220	0.00461	0.00456	0.00159
ENE	0.00038	0.00257	0.00663	0.02579	0.00412	0.00500	0.00358
E	0.00038	0.00559	0.00739	0.03784	0.00798	0.01087	0.01234
ESE	0.00114	0.00345	0.00614	0.03330	0.00773	0.01264	0.01154
SE	0.00095	0.00405	0.00829	0.03935	0.01146	0.01353	0.00677
SSE	0.00000	0.00254	0.00766	0.02651	0.00673	0.00661	0.00318
S	0.00019	0.00514	0.01599	0.07901	0.01483	0.01176	0.00796
SSW	0.00019	0.00266	0.00783	0.03776	0.00748	0.00749	0.00677
SW	0.00038	0.00113	0.00527	0.01994	0.00561	0.00720	0.00717
WSW	0.00000	0.00210	0.00311	0.00922	0.00299	0.00500	0.00478
W	0.00019	0.00179	0.00579	0.01949	0.00823	0.00794	0.00518
WNW	0.00038	0.00301	0.00640	0.02490	0.00499	0.00779	0.00518
NW	0.00038	0.00201	0.00389	0.02602	0.00387	0.00471	0.00279
NNW	0.00057	0.00169	0.00339	0.03134	0.00374	0.00191	0.00239
Total	0.00608	0.04501	0.10632	0.53206	0.10597	0.11818	0.08639
Calms[a]	0.00212	0.00611	0.00324	0.00686	0.00000	0.01795	0.05934

[a]Relative frequence of calms distributed within stability category.
Source: National Climatic Data Center, U.S. National Oceanic and Atmospheric Administration, Station 13996, Topeka, Kansas, 1992-1996.

3. DESIGN OF STRUCTURES, SYSTEMS, & COMPONENTS

This chapter describes the principal architectural and engineering design criteria for the structures, systems, and components that are required to ensure reactor facility safety and protection of the public.

The KSU TRIGA Mark II Nuclear Reactor Facility, which houses the TRIGA reactor, is located in a building constructed for that purpose. The building was constructed in two phases. The first phase was built in 1961 and was identified formally as "Nuclear Science and Engineering Laboratories." The building was named Ward Hall, to honor the late Professor Henry T. Ward, Head of the Chemical Engineering Department from which, in 1958, the Department of Nuclear Engineering was evolved. Construction was completed in 1972 for a major addition to the nuclear science and engineering facilities, known formally as the "Addition to Ward Hall." The original building and the addition are now identified collectively as a single building, namely, Ward Hall.

Building areas include office space, laboratory space, shop facilities, utility service areas, and classrooms, many of which support the activities of the Nuclear Reactor Facility. The TRIGA reactor itself establishes the fundamental requirements for two specific rooms, Room 110, the Reactor Bay, and Room 109, the Control Room. The geographic placement of Ward Hall and the TRIGA reactor are described in Chapter 2. Figures 3.1 and 3.2 illustrate layout of rooms in Ward Hall. The northern wing (Figure 3.1) is the 1961 structure; the southern wing the 1972 addition.

3.1 Design Criteria

3.1.1 General Conditions

The basic design goal of a TRIGA reactor is integrity of the fuel by cladding, acting as a physical containment system for fission products. Fuel design prevents the release of radioactive fission products during routine reactor operation and potential accident conditions. Limits on the amount of fuel loaded in the core (i.e., reactivity) establish a maximum steady state and transient power levels. Maximum possible power levels limit maximum fuel temperatures, which are the basic design constraints for the fuel. Design constraints are described in 3.5.1, Fuel System. Fuel design is detailed in Chapter 4, Reactor Description.

The reactor control system maintains safe shutdown conditions. Since operational limits prevent achieving conditions that could lead to fuel element failure, control system response speed is not significant to protection of fuel integrity. However, to ensure the control system is functioning normally, a test circuit measures rate of drop from full out to full insertion. Design constraints are discussed in 3.5.2, Control Rod Scram System; system design is discussed in Chapter 4, Reactor Description and Chapter 7, Instrumentation and Control Systems.

Facility design also controls personnel exposure to radiation associated with use of the fuel in reactor operation, and the release of effluents such as radioactive gases during normal operation or potential accident conditions. Design of the reactor bay as an air confinement system protects

operating personnel and the general public against any operational hazards such as release of air activation products, namely, ^{41}Ar during normal operation and fission products, namely, halogens and noble gases, during accident conditions. . Design Constraints are discussed in 3.5.3, Confinement and Ventilation System. Release criteria are based on 10 CFR 20. Chapter 11, Radiation Protection Program and Radiological Waste Management, describe control of radiation levels.

Building and structure design for meteorological, hydrological, and seismic effects are discussed in the following sections.

3.1.2 Architectural and Engineering Design Criteria

The reactor vendor was General Atomics Division of General Dynamics Corporation, San Diego, California. Architect-Engineer was Uel. C. Ramey and Associates, Wichita, Kansas. Reactor constructor was Holmes and Narver, Inc., Orange, California. The building code for the State of Kansas at the time of the construction of the original building was the National Building Code, latest edition, as adopted by the Kansas Legislature (according to the State Architect 1960/1961). The Uniform Building Code (UBC) gradually replaced the National Building Code in the late 1960's through the early 1970's. The Design Architect of Record has identified the UBC as the code used to construct the 1972 addition to the original building. The UBC is still the primary code used by the State of Kansas. Table 3.1 lists all codes currently used by the State of Kansas. All state property is exempt from all other local codes per state statute.

The University Architect in office during construction of both the 1961 and 1972 structures has identified practice during the period of construction as developing building programs for each project; the programs outlined in broad terms the spaces required to be designed and any special requirements needed to comply with the applicable regulations (in this case, the Atomic Energy Commission). Copies of the programs are no longer available for review in the State archives. Both the Architect-of-Record and the then University Architect state that the State Architect's office did not have a detailed manual of its expectations of design firms during that time.

The design services provided by both the 1961 and 1972 design teams were those typically delivered by architects and engineers after WWII. The design criteria from the period were eventually codified in the National Building Code, the Uniform Building Code, etc. with augmentation through University programs for construction on campus. However, formal design constraints of the 1961 structure are not available (the architect firm has ceased business, and the Architect-of-Record for the 1961 structure has since died). The Structural Engineer-of-Record for the 1961 building is no longer available, although his firm still performs design services in Kansas. A building program from the University described the design criteria to be used on the addition, according to the Architect-of-Record for the 1972 addition. The Structural Engineer-of-Record for the 1972 addition has retired and closed his practice.

However, some of the structural design constraints were incorporated into building plans, and are therefore available for review. These design constraints include such items as the soil boring log, the allowable maximum soil pressure, drilled pier conditions and various guides for placement and splicing of reinforcing steel design stresses and loads for concrete and structural steel. Selected design constraints applying to the 1961 structure, including the reactor bay and control

room are listed in Table 3.2. Design constraints applying to the 1972 addition are available for inspection. A comparison between drawings for the 1961/1972 constructions and other more recent campus constructions show striking similarities (as noted in section 3.4).

Table 3.1: Codes and their editions used by the State of Kansas.

1.	Uniform Building Code, 1997
2.	Uniform Building Code Standards, 1997
3.	Uniform Mechanical Code, 1997
4.	Uniform Plumbing Code, 1997
5.	National Electric Code, 1999
6.	Kansas State Boiler Code, 1988
7.	American with Disabilities Act Accessibility Guidelines for Buildings and Facilities (ADAAG), without the elevator exemption, published in the Federal Register 7-26-91 (KS.A. 58-1301 et seq.). Uniform Federal Accessibility Standards apply to agencies covered by Section 504 of the Rehabilitation Act of 1973
8.	ASHRAE/IES Standard 90.1-89
9.	Kansas Fire Prevention Code
10.	Underwriter's Laboratories Fire Resistance Directory, 1997
11.	National Fire Protection Association, National Fire Codes and Standards, Latest Edition at Date of Original Contract
12.	American Welding Society, AWS D-10.12-89, AWS D-1.1-96, AWS A-5.8-92, AWS D-10.80
13.	American Institute of Steel Construction-Ninth Edition, as
14.	required
15.	American Concrete Institute, ACI Standards 318-95
16.	Safety Code for Elevators And Escalators, ASME A17.1 Code, 1996
17.	ASME Boiler and Pressure Vessel Code, 1995

Table 3.2: Selected design guides for the 1961 structure.
(G. Hartwell & Co., Structural Engineers)

1.	Allowable maximum soil pressure = 4000 psf. Footings balance for DL.
2.	Piers shall be drilled to material capable of supporting 4000 psf. Bottom of piers shall be belled to indicated sizes and any loose material removed from bottom of excavation before placement of concrete.
3.	Grade beams, wall footings, & tunnels may be poured to lines of next excavation.
4.	Design stresses: concrete - 3000 psi min. at 28 days; reinforcing and structural steel - 20,000 psi min.
5.	Design loads: roof LL - 30 psf; floor LL - 100 psf[1]; loading dock - 7000 lb. concentrated load; crane capacity - 7000 lb.
6.	Concrete columns supporting roof are 12 in. x 12 in. w/4 #5 vertical and #2 ties @ 12 in. centers.

NOTE [1]: This value is extremely conservative; the current University Architect has performed an independent calculation using current methodology showing a floor loading of approximately 350 psf is acceptable

3.1.3 Structural System Design of the 1961 Building

The structural system of the 1961 building is primarily poured-in-place concrete except for the structural steel octagonal shaped dome over the actual reactor. The concrete foundation in the reactor bay has poured-in-place concrete walls 1'-0" thick setting on a continuous footing of 2'-6" W x 1'-0" D. The floor slab is 4" thick, wire-mesh reinforced concrete. The main 1961 building has 30" diameter drilled piers that have a 66" diameter bell resting on bedrock. Connecting each pier is a grade beam upon which rest the remainder of the poured in place columns and beams supporting the main floor and roof structure. The 1961 building was inspected in 1999 for degradation, with no sign of structural movement or damage to the primary structured system of the building and reactor bay. The 1961 building is typical in its detailing and requirements to the poured-in place reinforced concrete structural systems still being designed in the year 2000. There are no evident signs of any special structural design constraints placed on the 1961 building by the Atomic Energy Commission.

The design of the exterior walls are masonry block infilling the area between the concrete columns with a limestone exterior face extending up to the roof. In the reactor bay the masonry, stone and window wall is 7'-0" tall with the remainder of the walls and roof of the octagonal dome being clad in a metal insulated panel. Visual inspection of these components of the building indicate no signs of structural movement or damage. Except for the normal aging of the exterior stone, windows and related components exposed to the elements, the 1961 building in its entirety is in excellent structural condition.

3.1.4 Structural System Design of the 1972 Building

The structural system of the 1972 building is poured-in-place concrete throughout the building. The building has a full basement and one story above grade. The structural design criterion for indicates the building was designed to support additional floors that have never been constructed. The building, like its 1961 predecessor, is supported on piers that have been drilled into the bedrock. These piers vary in diameters from 2'-6' to 4'-0" depending upon location and loading condition. The exterior foundation walls are generally 1'4" thick and extend approximately 15'-0" below grade to the drilled piers. The foundation walls are heavily reinforced, as are the basement level beams and columns to support the anticipated loads of a three story (above-grade) building. The basement floor is 6" thick and also heavily reinforced. A 28'-0" square area of the basement was designed to function as a "hot cell" area with the sidewalls varying in thickness from 2'-0" to 5'-0" thick. Additionally a poured-in-place roof structure above the "hot cell" was cast-in-place and measures 5'-0" thick. In the center of the "hot cell" room is a 5'-0" diameter by 15'-0" deep well with a 1'-0" diameter by 10'-0" stainless steel auxiliary well extending below the foundation of the first well. The lowest point of the auxiliary well is approximately 25'-0" below the basement floor level. The exterior of the foundation walls and underside of the basement floor were provided with damp-proofing and a foundation underdrain system. Ground water problems have not occurred in the main basement area. However, ground water is sometimes found in the main well and auxiliary well structures to varying depths depending upon subsurface water conditions.

The first floor and roof framing system is poured-in-place concrete beams, columns and reinforced floor slabs. One limited area of structural steel beams, columns and bar joists exists in the small areas where the 1972 structure abuts the 1961 building. Visual inspection of the 1972 building in 1999 indicates no sign of structural movement or damage to the primary structural system of the building. This building like it's predecessor, is typical in its detailing and requirements to the poured-in-place reinforced concrete structural systems being designed in the year 2000. There are numerous signs of special design constraints being placed on the 1972 building by the Nuclear Regulatory Commission. The construction documents contain extensive details on the construction of the "hot cell" area as well as the immediately adjacent basement level laboratories for the "hot cell" lab, neutron generator lab, fuel processing lab, nuclear chemistry lab and radioisotope application lab. The design of the exterior walls matches those in the 1961 building-masonry block infilling the area between the concrete columns with a limestone exterior face extending up to the roof. Visual inspections of these components of the building indicate no signs of structural movement or damage.

3.1.5 Sanitary Sewer System

Ward Hall and the TRIGA Reactor are at the high end of a branch sanitary sewer line which serves the northern portion of the Kansas State University Campus. [redacted] where it joins the primary City of Manhattan sanitary sewer and exits campus. The City of Manhattan sanitary sewer continues on an east/southeasterly direction for approximately 7 miles exiting at the new water

treatment facility constructed by the City of Manhattan in 1995. At this point, after treatment, the water is placed in the Kansas River. A map of the system appears in Figure 3.3.

3.1.6 Storm Sewer

Ward Hall and the TRIGA Reactor are located along some of the higher land of the Kansas State University Campus. Storm sewers do not serve the building. All rain and storm water flows run on grade to the west and north, flowing onto adjacent campus streets. Storm water flows along these streets for approximately 1/4 mile to the north then 1/2 mile to the east/northeast where they enter Campus Creek, an open year around creek flowing through campus. The creek flows in a southeasterly line until it reaches the border of campus where it goes underground into the City of Manhattan storm sewer system. The City of Manhattan storm sewer system extends underground approximately 6 miles to the Kansas River. A map of the system appears in Figure 3.4.

3.2 Meteorological Damage

The available design criteria of both the original building and its addition do not provide any specific insight into how historical data factors on wind velocity, gust factors, recurrence intervals, tornado loading or other factors may have entered into the original design team's consideration. However, in the 30-year life of the 1961/1972 buildings and the direct experience of the current University Architect the entire building complex has an excellent history in withstanding meteorological damage. In the decade of the 90's the building has withstood a snow in excess of 18 inches, rains that came at the equivalent level of one 1000-year, two 500-year and many 100-year rainfalls, wind gusts in excess of 110 mph, nearby lightning strikes and severe hail. Except for the minor interior damage that would come from aging roofing materials, no effect to the building structure or any of its infrastructure system was noted. The current University Architect assesses 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 more than adequate for their intended use."

3.3 Water Damage

The University Architect notes that the mid 90s were exceptionally wet years culminating in the floods of 1993 when many areas of Manhattan were under water. This building is located on some of the highest ground in the city and campus. Only the reactor bay of the original 1961 structure was designed to be 12'-0" below grade as the remainder of the 1961 building was constructed "on grade." However, the 1972 structure was designed to have a full and occupiable basement. During this period, when all of the Midwest was flooded for weeks on end, no difficulties were noted with water entry to the building from any point, at or below grade, except for the minor damaged areas in the roofing previously addressed. The University Architect believes that while the water table in the area of the building was higher during the period of the flood the location of the building on high ground helped keep underground water pressure from building up and seeping into the 1961 reactor bay and the 1972 foundation walls. Additionally, the existing sloped grade around both buildings allows surface water to run off quickly.

3.4 Seismic Damage

The structures associated with the TRIGA reactor were designed in accordance with codes and standards applicable for the seismic zone designation at time of construction. This ensures that the reactor can be returned to operation without structural repairs following an earthquake likely to occur during the lifetime of the plant. Failure of the reactor tank and loss of the coolant in the event of a very large earthquake has been considered in Chapter 13 and the consequences found acceptable for the standpoint of public safety.

Boring logs for the 1961 structure and the 1972 structure are available for inspection, as are logs from the 1999 Ackert Hall Addition site immediately Northwest of this building and slightly downhill. In the view of University Architect (Gerald Carter), the logs are remarkably consistent. All of these soil borings indicate a modest amount of topsoil, varying levels of glacial deposits overlying bedrock, limestone and shale. The 1999 Soils Report notes that, "Cohesionless materials are generally encountered at depths near the bedrock surface." The 1961 boring log notes this level near the bedrock surface as being sand and gravel, the 1969 boring logs make the same comparison. Typically, this is the level where subsurface water is encountered. The 1999 Soils Report notes that ground water was encountered at depths ranging from 18'-0" to 34'-6" below the existing grade. This reflects the ground water level typically found in this area of campus.

The 1999 Soils Report details the recommendations for design of a major multistory addition to the primary biology teaching and research facility, Ackert Hall, on campus. It details the most current requirements for design and construction of structural systems that are typical to those used in the design and construction of the original 1961 building and the 1972 addition. With the sole exception of a 1999 recommendation to use a below grade vapor and drainage system for the foundation floors and walls, a preventative measure now being used around many major structures after the 1993 floods, there are few differences to be noted. The similarity of the 1999 requirements to the 1961/1972 design criteria and actual design is striking to Mr. Carter. The original structural designs of the G. Hartwell and Joseph C. Weakly firms are remarkably similar to the current structural designs of the Ackert Hall Addition. Chief among these similarities is the use of drilled piers extending down to the limestone bedrock. In order to absorb the type of live and dead loads typically found in major structural systems most Kansas State University buildings have utilized drilled piers being cut into the limestone bedrock.

This resolves not only the building loads but also the additional requirements placed on structures for earthquake loads and potential liquefaction problems. However, the potential for liquefaction is so remote in this area that it is not mentioned in the 1999 soil report, nor in any other buildings soil report on the main campus property.

The design recommendations for spread footings, such as those found in the reactor bay are also very similar to those used in the Ackert Hall Addition project as well as other adjacent projects. The only area of difference noted is that current guidelines call for 18" of fill materials is now placed under basement slabs instead of the 2" to 6" typically used in the 1950's and '60's. This difference reflects both the experiences with the recent floods plus the knowledge that some soils in the immediate area are subject to volume change (shrink/swell) with variations in moisture content. It does not appear that the Hartwell firm considered expansive clay soils of its 1961

design. However, in 39+ years there is no evidence of distress or movement caused by expansive soils anywhere around the original building or reactor bay. Given the recent climatological conditions of drought, flood, and then drought again there has been ample time for any soils problem to present itself.

3.5 Systems and Components

The reactor facility design uses a defense in depth concept to reduce and control the potential for exposure to radioactive material generated during reactor operation. Fuel cladding is the principal barrier to the release of radioactive fission products. Shielding (including biological shielding, reactor pool water and bulk shield tank water) controls exposure of personnel to radiation associated with operation of the reactor (during operations, and also to activated material). The control rods assure safe shutdown conditions are maintained when reactor operation is not required. If radioactive material releases associated with reactor operations occur, a dynamic confinement system controls release to the environment.

Cladding integrity is ensured by the fuel system (fuel rod and core design). Fuel cladding surrounding individual fuel elements is the primary barrier to the release of radioactive fission products. The fuel system maintains cladding integrity through interrelated limits on temperature, reactivity, and power to ensure cladding integrity is not capable of being challenged.

Shutdown reactor conditions are initiated and maintained by the control rod scram system. Since inherent shutdown mechanisms of the TRIGA prevent unsafe excursions, the TRIGA system does not rely on speed of control as paramount to the safety of the reactor. The control system ensures reactor shutdown conditions, and controlling power level during operation.

The confinement and ventilation system maintains airflow from the reactor bay to induce a negative air pressure in the bay. A confinement exhaust fan located in the center of the confinement dome normally operates during operation to provide the negative air pressure.

3.5.1 Fuel System

Production of hydrogen from a high temperature zirconium-water reaction is a well-known phenomenon. Zirconium hydride does not exhibit the same chemical reactivity as zirconium, and this reaction is not an issue for TRIGA fuel.

The reactor design bases are predicated on the maximum operational capability for the fuel elements and configuration described in this report. The TRIGA reactor system has three major and interrelated areas that are used to define the reactor design bases:

a. Fuel temperature,

b. Prompt negative temperature coefficient,

c. Reactor power.

Of these three, fuel temperature is the most amenable basis for establishing design constraints fuel temperature is a limit in both steady state and pulse-mode operation. A summary is presented below of the conclusions obtained from the reactor design bases described in this chapter.

Fuel growth and deformation can occur during normal operations at steady state fuel temperatures of 750°C or greater, as described in General Atomics technical report E-117-833. Damage mechanisms include fission recoils and fission gases, strongly influenced by thermal gradients. Operating with maximum long-term, steady state fuel temperature of 750°C does not have significant time- and temperature-dependent fuel growth. Since the KSU reactor will not be operated in the regime vulnerable to this degradation, the damage mechanism is not applicable.

The fuel temperature limit for pulsed mode operation is related to outgassing of hydrogen from the fuel and the subsequent stress produced in the fuel element clad material at elevated temperatures. The strength of the clad as a function of temperature can set the upper limit on the fuel temperature. Fuel temperature limits of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) have been set to preclude the loss of clad integrity (NUREG 1282).

The basic parameter that provides the TRIGA system with a large safety factor in steady-state operation and under transient conditions is the prompt negative temperature coefficient that is rather constant with temperature (-0.01% Δk/k°C). This coefficient is a function of the fuel composition and core geometry. As power and temperature increase, matrix changes cause a shift in the neutron energy spectrum in the fuel to higher energies. The uranium exhibits lower fission cross sections for the higher energy neutrons, thus countering the power increase.

The KSU TRIGA reactor, operating at 500 kW thermal power, is designed to use stainless-steel clad TRIGA fuel elements, with [redacted] percent by weight uranium in a $ZrH_{1.6}$ matrix, the uranium enriched up to [redacted] percent in ^{235}U. Element outside diameter is [redacted] in. ([redacted] cm) and cladding thickness is [redacted] in. ([redacted] cm). Details of the fuel system are presented in Chapter 4, as is fuel thermal characteristics during steady-state and pulse-mode operations. Fuel system behavior in accident conditions is addressed in Chapter 13.

Therefore, the basic safety limit for the TRIGA reactor is the limit on the fuel temperature for both steady state and pulsed-mode operations. Temperature is limited to prevent fuel expansion through phase changes, and to prevent gas pressure buildup.

a. Potential for Zr-Water Reaction

Among the chemical properties of U-ZrH and ZrH, the reaction rate of the hydride with water is also of interest. Since the hydriding reaction is exothermic, water will react more readily with zirconium than with zirconium hydride systems. Zirconium is frequently used in contact with water in reactors, and the zirconium-water reaction is not a safety hazard. Experiments carried out at GA Technologies show that the zirconium hydride systems have a relatively low chemical reactivity with respect to water and air. These tests have involved the quenching with water of both powders and solid specimens of U-ZrH after heating to as high as 850°C, and of solid U-Zr alloy after heating to as high as 1200°C. Tests have also been made to determine the extent to which fission products are removed from the surfaces of the fuel elements at room temperature. Results prove that, because of the high resistance to leaching, a large fraction of the fission products is retained in even completely unclad U-ZrH fuel.

Therefore, temperatures and chemical reactivity of TRIGA fuel matrix ensure that a zirconium water reaction will not occur at magnitudes that could cause hazard to the reactor. Additionally, a large fraction of fission products will be retained in the matrix.

b. Phase/Volume Changes

Two limiting temperatures are of interest, depending on the type of TRIGA fuel used. The TRIGA fuel that is considered low hydride, i.e. with an H/Zr ratio of less than 1.5, has a lower temperature limit than fuel with a higher H/Zr ratio. Figure 3.5 indicates that the higher hydride compositions are single phase, not subject to the large volume changes associated with the phase transformations at approximately 530°C in the lower hydrides. Also, it has been noted [Merton 1962] that the higher hydrides lack any significant thermal diffusion of hydrogen. These two facts preclude concomitant volume changes. The important properties of delta phase U-ZrH are given in Chapter 4, Reactor Description, Table 4.1.

For the rest of the discussion of fuel temperatures, we will concern ourselves with the higher hydride (H/Zr > 1.5) TRIGA fuel clad with 304 stainless steel [0.020 in. (0.508 mm) thick, or a cladding material equivalent in strength at the temperatures discussed. At room temperature the hydride is like a ceramic and shows little ductility. However, at the elevated temperatures of interest for pulsing, the material is found to be more ductile. The effect of very large thermal stress on hydride fuel bodies has been observed in hot cell observations to cause relatively widely spaced cracks which tend to be either radial or normal to the central axis and do not interfere with radial heat flow. Since the segments tend to be orthogonal, their relative positions appear to be quite stable.

Therefore, volume and other physical changes associated with phase change or mechanical deformation of the fuel at high temperatures for the fuel in use at the K-State reactor do not have the potential to mechanically challenge the cladding or affect the ability of the fuel to transfer heat.

c. Internal Fuel Rod Pressure

The limiting effect of fuel temperature is hydrogen gas overpressure. Figure 3.6 relates equilibrium hydrogen pressure over the fuel as a function of temperature for H/Zr ratio of 1.65.

The hydrogen gas over pressure is not in itself detrimental; however, if stress produced by gas pressure within the fuel exceeds ultimate strength of the clad material, a rupture of the fuel clad is possible. While the final conditions of fuel temperature and hydrogen pressure in which such an occurrence could come about are of interest, the mechanisms in obtaining temperatures and pressures of concern are different in the pulsing and steady-state mode of operation, and each mechanism will be discussed independently of the other. In this discussion it will be assumed that the fuel consists of U-ZrH (H/Zr = 1.65) with the uranium being [redacted] wt. %, and further that the cladding can is 304 stainless steel. The clad thickness is [redacted] with an inside clad diameter of [redacted] These fuel parameters have been chosen since they represent the nominal specifications for TRIGA fuel elements. Figure 3.7 shows 304 stainless-steel yield and

ultimate strengths as a function of temperature. In determining hoop stress on the cladding from the internal hydrogen gas pressure the following equation applies [Harvey 1974]:

$$S = \frac{rP}{t},$$

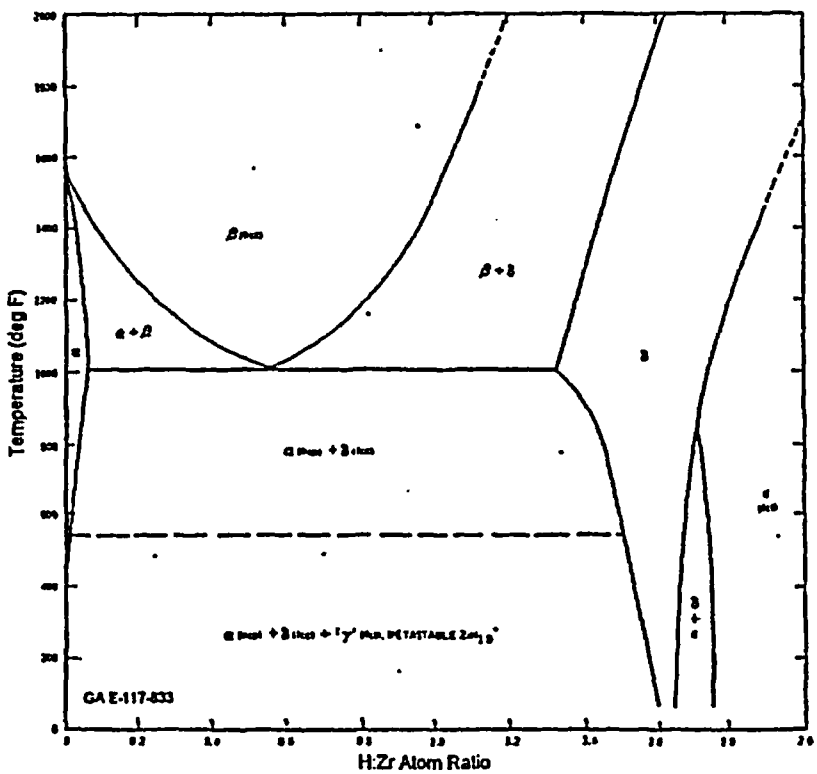

Figure 3.5: Zr-Hydride Phase Diagram

in which S is the stress, P is the internal pressure (in the same units as the stress), r is the (inside) radius of the cladding, and t is the cladding thickness. Thus, for the TRIGA cladding, $S = 36P$.

It is of interest to relate the strength of the clad material at its operating temperature to the *stress* applied to the clad from the internal gas pressure associated with the fuel temperature. Figure 3.8 illustrates the stress applied to the clad as a result of hydrogen dissociation for fuel having a H/Zr ratio of 1.6 to 1.7 as a function of temperature.

There are several mitigating factors that would cause gas pressure to be lower for transient conditions as compared to (predicted) equilibrium values. For example, the gas diffusion rates are finite; surface cooling is believed to be caused by endothermic gas emission, which tends to lower the diffusion constant at the surface; re-absorption takes place concurrently on the cooler hydride surfaces away from the hot spot; there is

evidence for a low permeability oxide film on the fuel surface; and some local heat transfer does take place during the pulse time to cause a less than adiabatic true surface temperature.

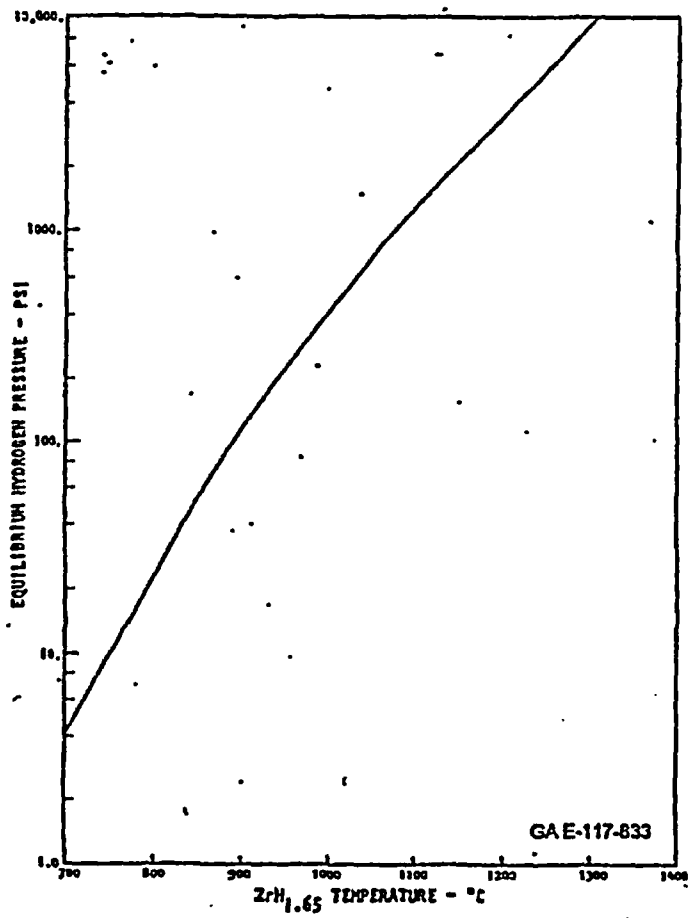

Figure 3.6: Pressure versus Temperature for $ZrH_{1.65}$

The limiting design parameter for TRIGA fuel is therefore related to buildup of pressure through disassociation of hydrogen in the matrix. This free hydrogen is heated by fuel temperature, causing a buildup of pressure on the internal surfaces of the cladding. As illustrated in Figure 3.8, cladding temperatures exceeding about 1000°C (based on a H/Zr ratio of 1.6 to 1.7) has potential to lead to pressures that exceed ultimate tensile strength.

d. Conclusion

Calculations show that a fully bonded fuel element (i.e., cladding temperature at fuel temperature) will not fail at fuel temperatures below about 1000°C. Therefore, design limits for the TRIGA fuel are based on fuel temperature.

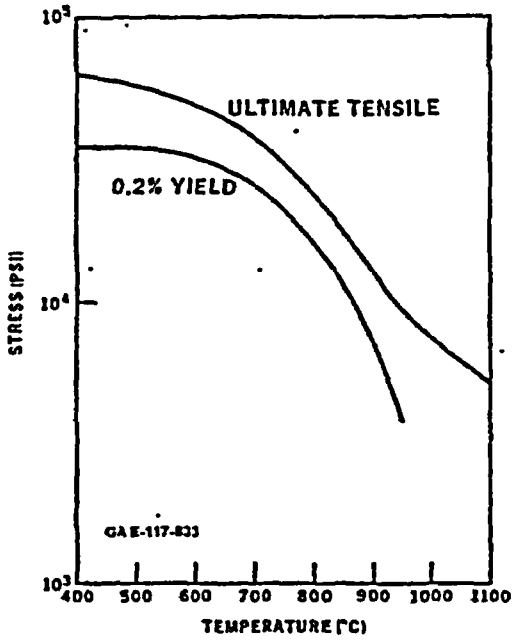

Fig. 3. 7: Temperature dependent stress vs. UTS for type 304 SS cladding

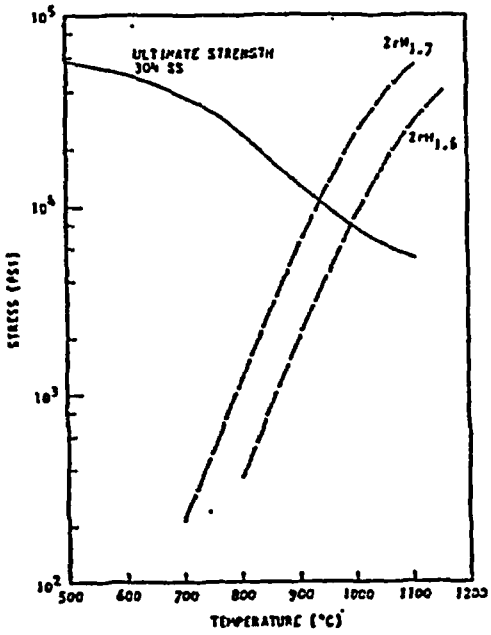

Figure 3.8: Stress Versus Temperature At Hydrogen Compositions

3.5.2 Shielding

Design bases for TRIGA shielding derive from General Atomic shielding design analysis for the 1 MW reactor, which is similar construction and dimensions as the KSU TRIGA reactor. Design basis radiation levels at the core level and at 1 MW, are as follows:

- ≈ 1 Mrad/h at the core boundary
- ≈ 40 rad/h at the tank boundary
- < 1 mrem/h outside the biological shield

Design requirements allow access through the shielding to experimental areas, and permit extracting beams of radiation form the shielded volume into the reactor bay.

3.5.3 Control Rod Scram System

The KSU TRIGA reactor, operating at 500 kW thermal power, is designed to be operated with three standard and one transient (pulsing) control rod. Standard rods are nominally 0.875 in. (2.22 cm) outside diameter, the pulse rod is nominally 1.25 in. (3.18 cm) outside diameter. Both

are 20 in. (50.8 cm) long and are clad with 30 mil (0.0762 cm) aluminum. The control material is either boron carbide or borated graphite. During operation, standard rods are held in place by electromagnets, the pulse rod by air pressure. All are manually withdrawn or inserted by motor-driven gear mechanisms. Upon a scram signal, power to electromagnets is interrupted and air pressure is vented, with all control rods descending by gravity into the core. Standard rods have a maximum drop time of 1 second and the pulse rod has a comparable drop time. Details of the control rod scram system are addressed in Chapters 4 and 7.

3.5.4 Confinement and Ventilation Systems

The confinement and ventilation systems are intended to control the level of airborne radioactive contaminants in the restricted area, and to release reactor bay air in unoccupied area at the top of the confinement structure.

The reactor bay is surrounded by a roughly dome shaped building, as shown in Figure 3.9. As described in Chapter 11, Appendix A, the reactor bay dimensions are approximately as follows: The reactor bay (Ward Hall Room 110) is approximated as a right circular cylinder 36 ft (10.973 m) high and 36.68 ft (11.18 m) radius. The reactor vessel structure is approximated as a right circular cylinder, co-axial with the bay, 22 ft (6.706 m) high and 11 ft (3.3528 m) radius. The free volume is 144,000 ft^3 (4078 m^3). P[redacted] The reactor bay is maintained at a slight negative pressure; the measurement is obtained across the separation of the control room and the reactor bay. [redacted]

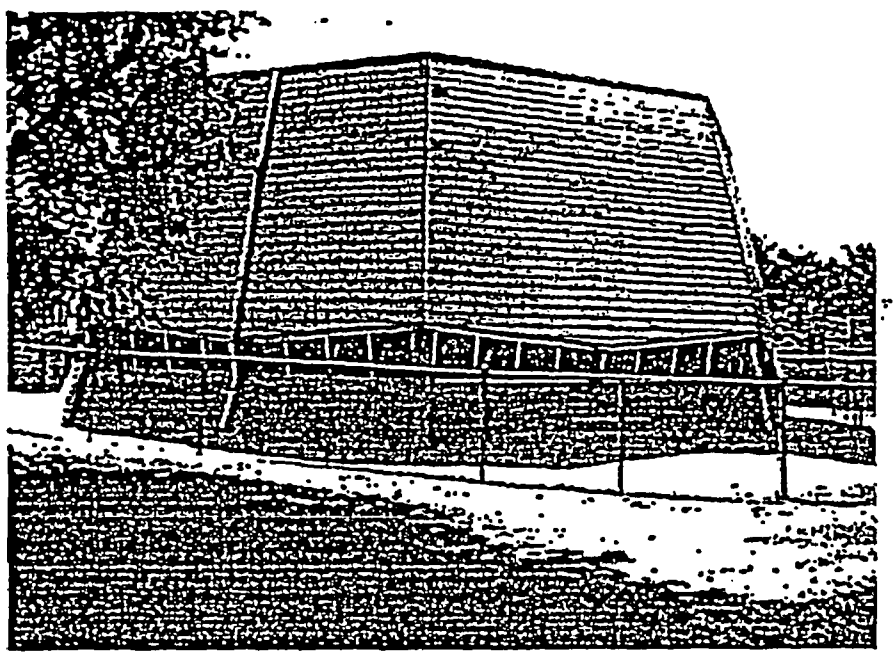

Figure 3.9: Reactor Bay Confinement Structure

In addition to artificial lighting, a series of plate glass windows surround the reactor bay, allowing ambient light into the reactor bay. A security fence restricts access to the windows from the outside.

[redacted]

The ventilation system for the reactor bay exhausts from a roof vent approximately 11 m above grade. As discussed in Chapter 11, Appendix A, this discharge during operation maintains a slight negative pressure on the reactor bay, and controls ^{41}Ar concentrations within the bay and at the site boundary to within all applicable limits.

The control room has a separate fan system that can be used on demand for ventilation. The control room has a window to the environment that may be opened at discretion of the facility staff.

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[1] The main room is 11'-3" by 22'. The foyer is 6'-5" by 6'.

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4. Reactor Description

4.1 Summary Description

The Kansas State University (KSU) Nuclear Reactor Facility, operated by the Department of Mechanical and Nuclear Engineering, is located in Ward Hall on the campus in Manhattan, Kansas. The Department is also the home of the Tate Neutron Activation Analysis Laboratory. The TRIGA reactor was obtained through a 1958 grant from the United States Atomic Energy Commission and is operated under Nuclear Regulatory Commission License R-88 and the regulations of Chapter 1, Title 10, Code of Federal Regulations. Chartered functions of the Nuclear Reactor Facility are to serve as: 1) an educational facility for all students at KSU and nearby universities and colleges, 2) an irradiation facility for researchers at KSU and for others in the central United States, 3) a facility for training nuclear reactor operators, and 4) a demonstration facility to increase public understanding of nuclear energy and nuclear reactor systems.

The KSU TRIGA reactor is a water-moderated, water-cooled thermal reactor operated in an open pool and fueled with heterogeneous elements consisting of nominally [redacted] percent enriched uranium in a zirconium hydride matrix and clad with stainless steel. Principal experimental features of the KSU TRIGA Reactor Facility are:

- Central thimble
- Rotary specimen rack
- Thermalizing column with bulk shielding tank
- Thermal column with removable door
- Beam ports
  - Radial (2)
  - Piercing (fast neutron) (1)
  - Tangential (thermal neutron) (1)

The reactor was licensed in 1962 to operate at a steady-state thermal power of 100 kilowatts (kW). The reactor has been licensed since 1968 to operate at a steady-state thermal power of 250 kW and a pulsing maximum thermal power of 250 MW. Application is made concurrently with license renewal to operate at a maximum of 1,250 kW, with fuel loading to support 500 kW steady state thermal power with pulsing to $3.00 reactivity insertion. All cooling is by natural convection. The 250-kW core consists of 81 fuel elements typically (at least [redacted] planned for the 1,250-kW core), each containing as much as [redacted] grams of ^{235}U. The reactor core is in the form of a right circular cylinder about 23 cm (approximately 9 in.) radius and 38 cm (14.96 in.) depth, positioned with axis vertical near the base of a cylindrical water tank 1.98 m (6.5 ft.) diameter and 6.25 m (20.5 ft.) depth. Criticality is controlled and shutdown margin assured by control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite. Reactivity requirements (i.e., minimum shutdown margin with the most reactive rod fully withdrawn and maximum excess reactivity) can be met for 250 kW with three control rods, but reactivity required to compensate for fuel temperature and fission products for operations at power levels of

500 kW requires four control rods to meet reactivity requirements. A biological shield of reinforced concrete at least 2.5 m (8.2 ft) thick provides radiation shielding at the side and at the base the reactor tank. The tank and shield are in a 4078-m^3 (144,000 $ft.^3$) confinement building made of reinforced concrete and structural steel, with composite sheathing and aluminum siding. Sectional views of the reactor are shown in Figures 4.1 and 4.2.

Criticality was first achieved on October 16, 1962 at 8:25 p.m. In 1968, pulsing capability was added and the maximum steady-state operating power was increased from 100 kW to 250 kW. The original aluminum-clad fuel elements were replaced with stainless-steel clad elements in 1973. Coolant system was replaced (and upgraded in 2000), the reactor operating console was replaced, and the control room was enlarged and modernized in 1993, with support from the U.S. Department of Energy. All neutronic instrumentation was replaced in 1994.

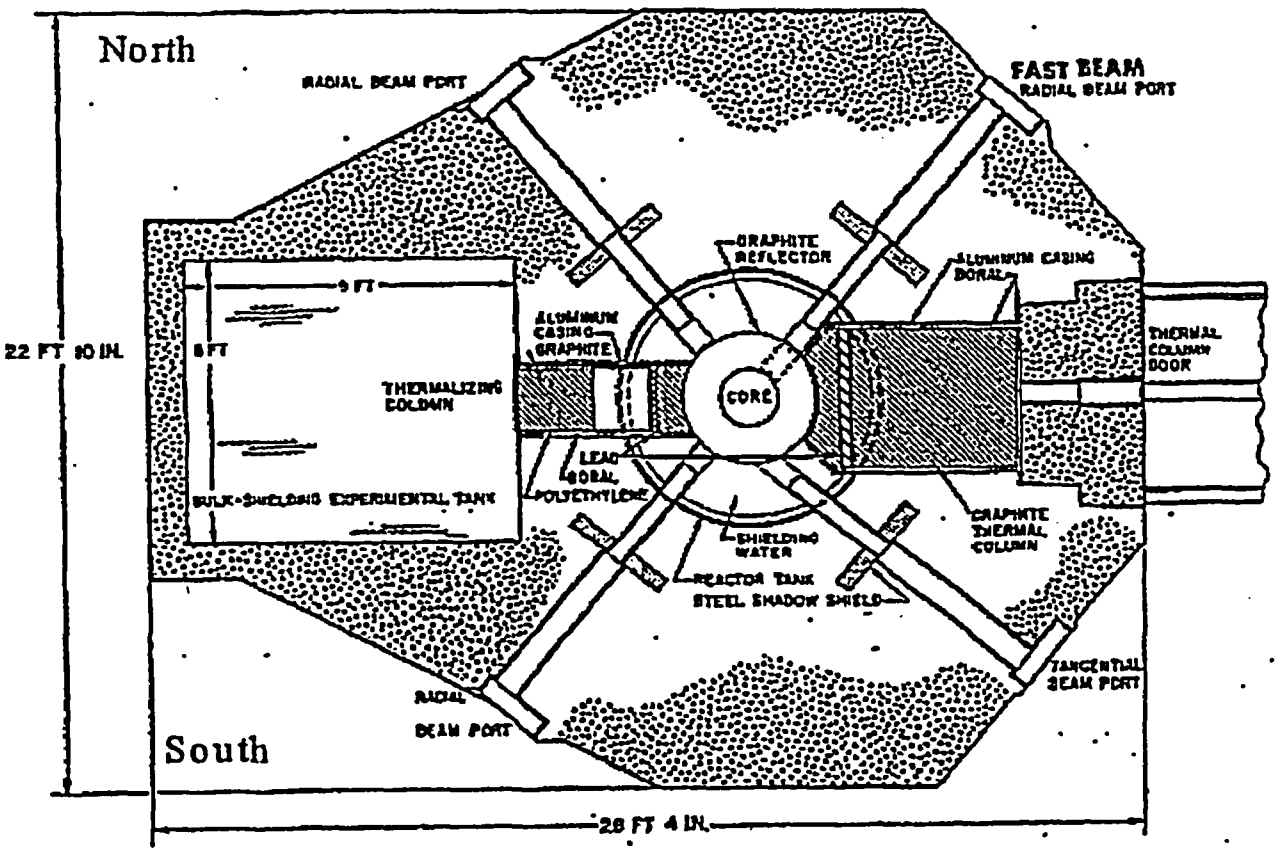


Figure 4. 1, Vertical Section Through the KSU TRIGA Reactor.

4.2 Reactor Core

The General Atomics TRIGA reactor design began in 1956. The original design goal was a completely and inherently safe reactor. Complete safety means that all the available excess reactivity of the reactor can be instantaneously introduced without causing an accident. Inherent safety means that an increase in the temperature of the fuel immediately and automatically results in decreased reactivity through a prompt negative temperature coefficient. These features were accomplished by using enriched uranium fuel in a zirconium hydride matrix.

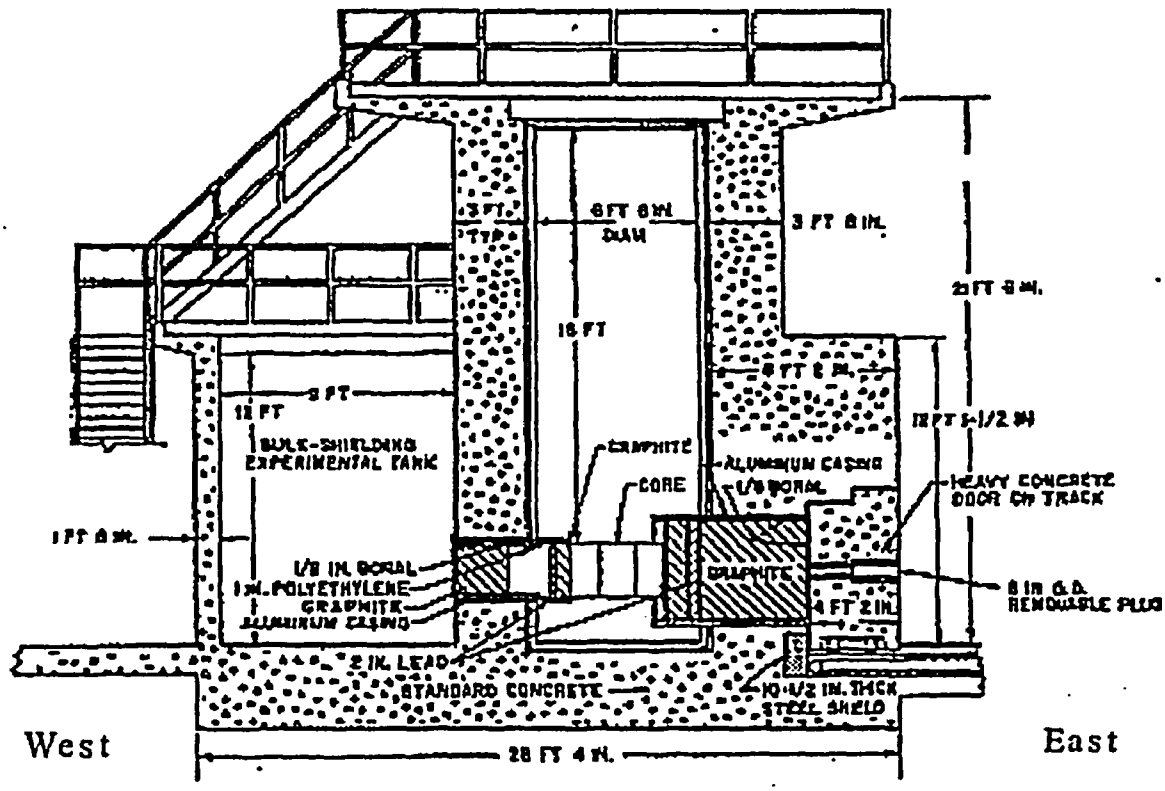


Figure 4. 2, Horizontal Section Through the KSU TRIGA Reactor.

The basic parameter providing the TRIGA system with a large safety factor in steady state and transient operations is a prompt negative temperature coefficient, relatively constant with temperature (-0.01% $\Delta k/k°C$). This coefficient is a function of the fuel composition and core geometry. As power and temperature increase, matrix changes cause a shift in the neutron energy spectrum in the fuel to higher energies. The uranium exhibits lower fission cross sections for the higher energy neutrons, thus countering the power increase. Therefore, fuel and clad temperature automatically limit operation of the reactor.

It is convenient to set a power level limit that is based on temperature. The design bases analysis indicates that operation at up to 1900 kW (with an [redacted] element core and 120°F inlet water temperature) with natural convective flow will not allow film boiling; therefore high fuel and clad temperatures capable of causing loss of clad integrity cannot occur. An [redacted] element core distributes the power over a larger volume of heat generating elements, and therefore using [redacted] elements in analysis results in a less favorable, more conservative thermal hydraulic response.

4.2.1 Reactor Fuel[1]

TRIGA fuel was developed around the concept of inherent safety. A core composition was sought which had a large prompt negative temperature coefficient of reactivity such that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature would automatically cause the power excursion to terminate before any core damage resulted. Zirconium hydride was found to possess a basic mechanism to produce the desired characteristic. Additional advantages were that ZrH has a high heat capacity, results in relatively small core sizes and high flux values due to the high hydrogen content, and could be used effectively in a rugged fuel element size.

TRIGA fuel is designed to assure that fuel and cladding can withstand all credible environmental and radiation conditions during its lifetime at the reactor site. As described in 3.5.1 (Fuel System) and NUREG 1282, fuel temperature limits both steady-state and pulse-mode operation. The fuel temperature limit stems from potential hydrogen outgassing from the fuel and the subsequent stress produced in the fuel element clad material. The maximum temperature limits of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) have been set to limit internal fuel cladding stresses that might challenge clad integrity (NUREG 1282). These limits are the principal design bases for the safety analysis.

a. Dimensions and Physical Properties.

The KSU TRIGA reactor is fueled by stainless steel clad Mark III fuel-elements. Three instrumented aluminum-clad Mark II elements are still available for use in the core. General properties of TRIGA fuel are listed in Table 4.1. The Mark III elements are illustrated in Figure 4.3. To facilitate hydriding in the Mk III elements, a zirconium rod is inserted through a 0.635 cm. (1/4-in.) hole in the center of the active fuel section.

Instrumented elements have three chromel-alumel thermocouples embedded to about 0.762 cm (0.3 in.) from the centerline of the fuel, one at the axial center plane, and one each at 2.54 cm. (1 in.) above and below the center plane. Thermocouple leadout wires pass through a seal in the upper end fixture, and a leadout tube provides a watertight conduit carrying the leadout wires above the water surface in the reactor tank.

[1]Unless otherwise indicated, fuel properties are taken from the General Atomics report of Simnad [1980] and from authorities cited by Simnad.

Graphite dummy elements may be used to fill grid positions in the core. The dummy elements are of the same general dimensions and construction as the fuel-moderator elements. They are clad in aluminum and have a graphite length of 55.88 cm (22 in.).

Table 4.1, Nominal Properties of Mark II and Mark III TRIGA Fuel Elements in use at the KSU Nuclear Reactor Facility.

Property	Mark II	Mark III
Dimensions		
Outside diameter, $D_o = 2r_o$	[redacted]	
Inside diameter, $D_i = 2r_i$		
Overall length		
Length of fuel zone, L		
Length of graphite axial reflectors	4 in. (10.16 cm)	3.44 in (8.738 cm)
End fixtures and cladding	aluminum	304 stainless steel
Cladding thickness	[redacted]	
Burnable poisons	Sm wafers	None
Uranium content		
Weight percent U	[redacted]	[redacted]
^{235}U enrichment percent		
^{235}U content		
Physical properties of fuel excluding cladding		
H/Zr atomic ratio	1.0	1.6
Thermal conductivity (W cm^{-1} K^{-1})	0.18	0.18
Heat capacity [T ≥0°C] (J cm^{-3} K^{-1})		2.04 + 0.00417T
Mechanical properties of delta phase U-ZrH[a]		
Elastic modulus at 20°C		9.1×10^6 psi
Elastic modulus at 650°C		6.0×10^6 psi
Ultimate tensile strength (to 650°C)		24,000 psi
Compressive strength (20°C)		60,000 psi
Compressive yield (20°C)		35,000 psi

[a]Source: Texas SAR [1991].

b. Composition and Phase Properties

The Mark III TRIGA fuel element in use at Kansas State University contains nominally [redacted] by weight of uranium, enriched to [redacted] ^{235}U, as a fine metallic dispersion in a zirconium hydride matrix. The H/Zr ratio is nominally 1.6 (in the face-centered cubic delta phase). The equilibrium hydrogen dissociation pressure is governed by the composition and temperature. For $ZrH_{1.6}$, the equilibrium hydrogen pressure is one atmosphere at about 760°C. The single-phase, high-hydride composition eliminates the problems of density changes associated with phase changes and with thermal diffusion of the hydrogen in earlier designs. Over 25,000 pulses have been performed with the TRIGA fuel elements at General Atomic, with fuel temperatures reaching peaks of about 1150°C.

The zirconium-hydrogen system, whose phase diagram is illustrated in Chapter 3, is essentially a simple eutectoid, with at least four separate hydride phases. The delta and epsilon phases are respectively face-centered cubic and face-centered tetragonal hydride phases. The two phase delta + epsilon region exists between $ZrH_{1.64}$ and $ZrH_{1.74}$ at room temperature, and closes at $ZrH_{1.7}$ at 455°C. From 455°C to about 1050°C, the delta phase is supported by a broadening range of H/Zr ratios.

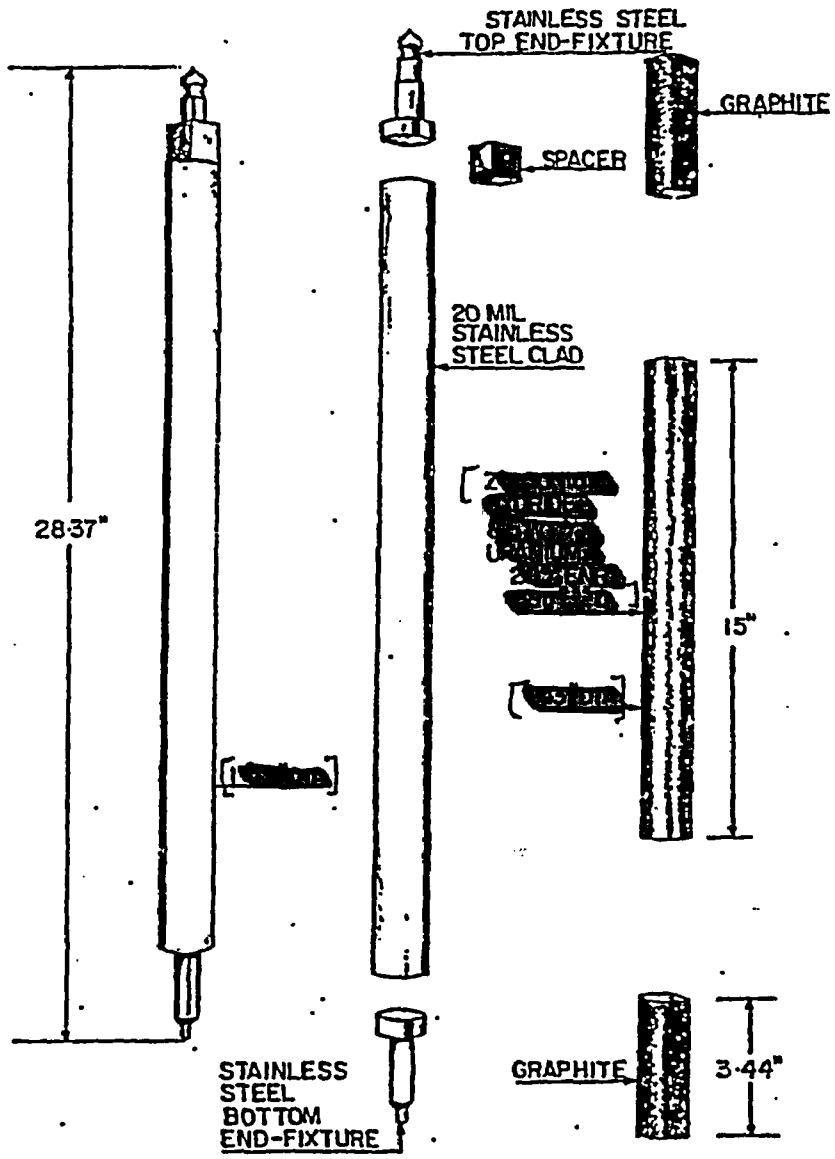


Figure 4.3, TRIGA Fuel Element.

c. Core Layout

A typical layout for a KSU TRIGA II 250-kW core (Core II-18) is illustrated in Figure 4.4. The layout for the 1,250-kW core is expected to be similar, except that the graphite elements will be replaced by fuel elements, one additional control rod will be added, and control rod positions will be adjusted..

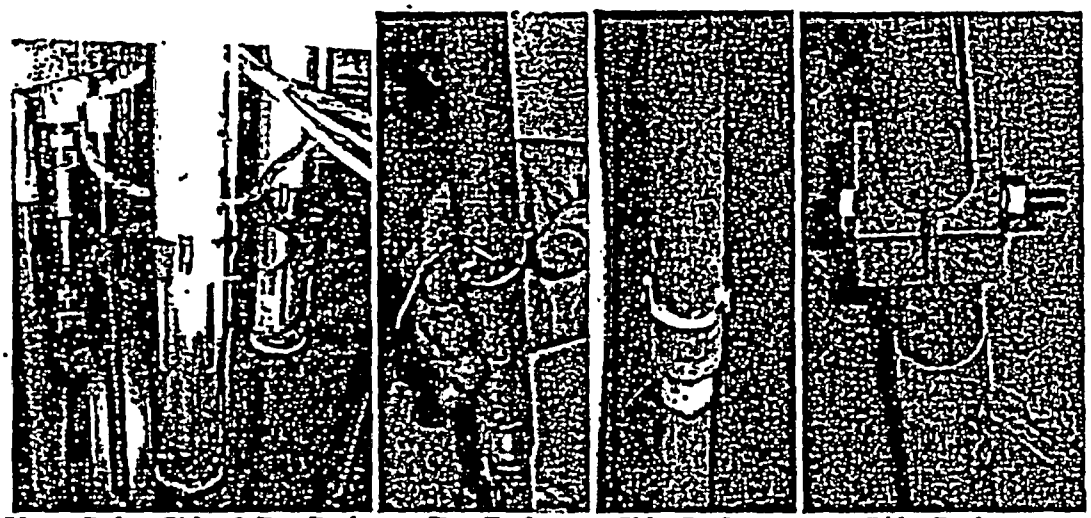
[ ]

The additional fuel elements are required to compensate for higher operating temperatures from the higher maximum steady state power level. The additional control rod is required to meet reactivity control requirements at higher core reactivity associated with the additional fuel. The control rod positions will be different to allow a higher worth pulse rod (the 250 kW pulse rod reactivity worth is $2.00, the 1,250 kW core pulse rod reactivity worth is $3.00), balancing the remaining control rods worth's to meet minimum shutdown margin requirements, and meeting physical constraints imposed by the dimensions of the pool bridge

4.2.2 Control Rods

Control rods are 50.8 cm. (20 in.) long boron carbide or borated graphite, clad with a 0.0762 cm. (30-mil) aluminum sheath. The pulse rod is 3.175 cm. (1.25 in.) diameter. Other rods are 2.225 cm (7/8 in.) diameter.

The control rod drives are connected to control rod clutches through three extension shafts. The clutch and upper extension shaft for standard rods extend through an assembly designed with slots that provides a hydraulic cushion (or buffer) for the rod during a scram, and also limits the bottom position of the control rods so that they do not impact the bottom of the control rod guide tube (in the core). The buffers for two standard rods are shown in the left hand picture below (slotted tubes on the right hand side) along with the top section of the pulse/transient rod extension. The pulse rod drive clutch connects to a solid extension shaft through a pneumatic cylinder; the dimensions of the cylinder limits bottom travel.



Upper Pulse, Shim & Reg Rods Reg Rod Shim Rod Pulse Rod

Figure 4.5, Control Rod Upper Extension Assemblies

The lower extension of the pulse rod is shown on the left hand side of Figure 4.5. The upper extension shaft is a hollow tube, the middle extension is solid. The upper extension shaft is connected to the middle extension shaft with lock wire or a pin and lock wire for standard rods, with a bolted collar for the pulse rod (the mechanical shock during a pulse requires a more durable fastener). Securing the upper control rod extension to the middle extension at one of several holes drilled in the upper part of the middle extension (Figure 4.6) provides adjustment for the control rods necessary to ensure the control rod full-in position is above the bottom of the guide tube.

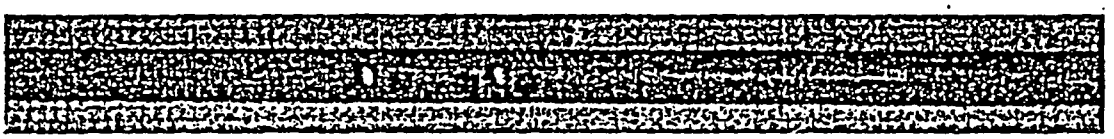
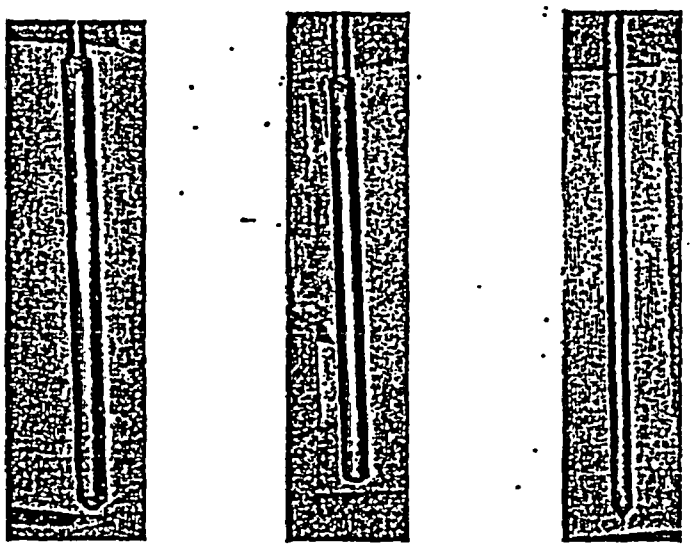


Figure 4.6, Middle Extension Rod Alignment Holes

The middle (solid) extension is similarly connected to the lower extension. The lower extension is hollow, the middle extension fits into the lower extension and a hole drilled in the overlap secures the lower extension to the middle extension. Typically the lower extension has a tighter fit than the upper extension because the lower and middle extension are not separated for inspections and because the interface with upper extension is used to set the bottom position of the control rod. Pictures of the lower connector for the pulse rod and one standard rod are shown at the left in Figure 4.7..

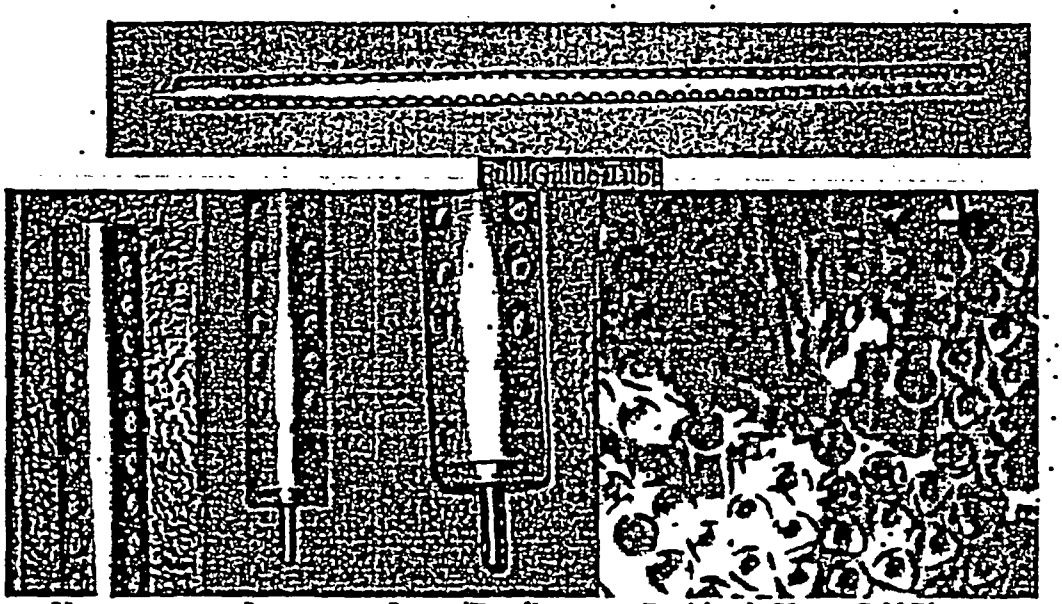
Figure 4.7, Standard & Pulse Rod Lower Coupling

The bottom of the lower extension attaches directly to the control rod. Pictures of the control rods taken during the 2003 control rod inspection are in Figure 4.8. The rods move within control rod guide tubes, shown in Figure 4.9. The guide tubes have perforated walls. Alignment pins in the bottom end fitting of the guide tube fit into holes in the lower guide plate. The alignment pins have a small metal wire in the tip that fits into the lower grid plate; a setscrew inside the bottom of the guide tube pushes the wire against the lower grid plate to secure the guide tube.



Pulse Rod Shim Rod Reg Rod

Figure 4.8, Control Rods During 2003 Inspection



Upper Lower Lower Detail Position in Upper Grid Plate

Figure 4.9, Control Rod Guide Tubes

a. Control Function

While three control rods were adequate to meet Technical Specification requirements for reactivity control with the 100 kW and 250 kW cores, reactivity limits for operation at a maximum power level of 1,250 kW requires four control rods (three standard and one transient/pulsing control rod). The control-rod drives are mounted on a bridge at the top of the reactor tank. The control rod drives are coupled to the control rod through a connecting rod assembly that includes a clutch. The standard rod clutch is an electromagnet; the transient rod clutch is an air-operated shuttle. Scrams cause the clutch to release by de-energizing the magnetic clutch and venting air from the transient rod clutch; gravity causes the rod to fall back into the core. Interlocks ensure operation of the control rods remains within analyzed conditions for reactivity control or limit potential for accident scenarios, while scrams operate at limiting safety system settings. A detailed description of the control-rod system is provided in Chapter 7; a summary of interlocks and scrams is provided below in Table 4.2 and 4.3. Note that (1) the high fuel temperature and period scrams are not required, (2) the fuel temperature scram limiting setpoint depends on core location for the sensor, and (3) the period scram can be prevented by an installed bypass switch.

Table 4.2, Summary of Control Rod Interlocks

INTERLOCK	SETPOINT	FUNCTION/PURPOSE
Source Interlock	2 cps	Inhibit standard rod motion if nuclear instrument startup channel reading is less than instrument sensitivity/ensure nuclear instrument startup channel is operating
Pulse Rod Interlock	Pulse rod inserted	Prevent applying power to pulse rod unless rod inserted/prevent inadvertent pulse
Multiple Rod Withdrawal	Withdraw signal, more than 1 rod	Prevent withdrawal of more than 1 rod/Limit maximum reactivity addition rate
Pulse Mode Interlock	Mode switch in Hi Pulse	Prevent withdrawing standard control rods in pulse mode
Pulse-Power Interlock	10 kW	Prevent pulsing if power level is greater than 10 kW

NOTE: Pulse-Power Interlock normally set at 1 kW

b. Evaluation of Control Rod System

The reactivity worth and speed of travel for the control rods are adequate to allow complete control of the reactor system during operation from a shutdown condition to full power. The TRIGA system does not rely on speed of control as significant for safety of the reactor; scram times for the rods are measured periodically to monitor potential degradation of the control rod system. The inherent shutdown mechanism (temperature feedback) of the TRIGA prevents unsafe excursions and the control system is used only for the planned shutdown of the reactor and to control the power level in steady state operation.

Table 4.3, Summary of Reactor SCRAMs

Measuring Channel	Limiting Trip Setpoint		Actual Setpoint
	Steady State	Pulse	
Linear Channel High Power	110%	N/A	104%
Power Channel High power	110%	N/A	104%
Detector High Voltage	90%	90%	90%
High Fuel Temperature	600°C B Ring element		450°C
	555°C C Ring element		
	480°C D Ring element		
	380°C E Ring element		350°C
Period	N/A	N/A	3 sec

NOTE: Period trip and temperature trip are not required

The reactivity worth of the control system can be varied by the placement of the control rods in the core. The control system may be configured to provide for the excess reactivity needed for approximately 500 kW operations for eight hours per day (including xenon override) and will assure a shutdown margin of at least $0.50.

Nominal speed of the standard control rods is about 12 in. (30.5 cm) per minute (with the stepper motor specifically adjusted to this value), of the transient rod is about 24 in. (61 cm) per minute, with a total travel about 15 in. (38.1 cm). Maximum rate of reactivity change for standard control rods is specified in Technical Specifications.

4.2.3 Neutron Moderator and Reflector

Hydrogen in the Zr-H fuel serves as a neutron moderator. Demineralized light water in the reactor pool also provides neutron moderation (serving also to remove heat from operation of the reactor and as a radiation shield). Water occupies approximately 35% of the core volume. A graphite reflector surrounds the core, except for a cutout containing the rotary specimen rack (described in Chapter 10). Each fuel element contains graphite plugs above and below fuel approximately 3.4 in. in length, acting as top and bottom reflectors.

The radial reflector is a ring-shaped, aluminum-clad, block of graphite surrounding the core radially. The reflector is 0.457-m (18.7 in.) inside diameter, 1.066-m (42 in.) outside diameter, and 0.559-m height. Embedded as a circular well in the reflector is an aluminum housing for a rotary specimen rack, with 40 evenly spaced tubular containers, 3.18-cm (1.25 in.) inside diameter and 27.4-cm (10.8 in.) height. The rotary specimen rack housing is a watertight assembly located in a re-entrant well in the reflector.

The radial reflector assembly rests on an aluminum platform at the bottom of the reactor tank. Four lugs are provided for lifting the assembly. A radial void about 6 inches (15.24 cm) in diameter is located in the reflector such that it aligns with the radial piercing beam port (NE beam port). The reflector supports the core grid plates, with grid plate positions set by alignment fixtures. Graphite inserts within the fuel cladding provide additional reflection. Inserts are placed at both ends of the fuel meat, providing top and bottom reflection.

4.2.4 Neutron Startup Source

A 2-curie americium-beryllium startup source (approximately 2×10^6 n s^{-1}) is used for reactor startup. The source material is encapsulated in stainless steel and is housed in an aluminum-cylinder source holder of approximately the same dimensions as a fuel element. The source holder may be positioned in any one of the fuel positions defined by the upper and lower grid plates. A stainless-steel wire may be threaded through the upper end fixture of the holder for use in relocating the source manually from the 22-ft level (bridge level) of the reactor.

4.2.5 Core Support Structure

The fuel elements are spaced and supported by two 0.75-in. (1.9 cm) thick aluminum grid plates. The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures. Space is provided for the passage of cooling water around the sides of the bottom grid plate and through 36 experiment penetrations. The 1.5-in. (3.8 cm) diameter holes in the upper grid plate serve to space the fuel elements and to allow withdrawal of the elements from the core. Triangular-shaped spacers on the upper end fixtures allow the cooling water to pass through the upper grid plate when the fuel elements are in position. The reflector assembly supports both grid plates.

4.3 Reactor Tank

The KSU TRIGA reactor core support structure rests on the base of a continuous, cylindrical aluminum tank surrounded by a reinforced, standard concrete structure (with a minimum concrete thickness of approximately 249 cm. or 8 ft 2 in), as illustrated in Figures 4.1 and 4.2. The tank is a welded aluminum structure with 0.635 cm. (1/4-in.) thick walls. The tank is approximately 198 cm (6.5-ft) in diameter and approximately 625 cm (20.5-ft) in depth. The exterior of the tank was coated with bituminous material prior to pouring concrete to retard corrosion. Each experiment facility penetration in the tank wall (described below) has a water collection plenum at the penetration. All collection plenums are connected to a leak-off volume through individual lines with isolation valves, with the leak-off volumes monitored by a pressure gauge. The bulk shield tank wall is known to have a small leak into the concrete at the thermalizing column plenum, therefore a separate individual leak-off volume (and pressure gauge) is installed for the bulk shield tank; all other plenums drain to a common volume. In the event of a leak from the pool

through an experiment facility, pressure in the volume will increase. Isolating individual lines allows identification of the specific facility with the leak.

A bridge of steel plates mounted on two rails of structural steel provides support for control rod drives, central thimble, the rotary specimen rack, and instrumentation. The bridge is mounted directly over the core area, and spans the tank. Aluminum grating with clear plastic attached to the bottom is installed that can be lowered over the pool. The grating normally remains up to reduce humidity at electro-mechanical components of the control rod drive system and to prevent the buildup of radioactive gasses at the pool surface during operations. The grating can be lowered during activities that could cause objects or material to fall into the reactor pool.

Four beam tubes extend from the reactor wall to the outside of the concrete biological shield in the outward direction. Tubes welded to the inside of the wall extend toward the reactor core. Three of the tubes (NW, SW, and SE) end at the radial reflector. The NE beam tube penetrates the radial reflector, extending to the outside of the core. Two penetrations in the tank allow neutron extraction into a thermal column and a thermalizing column (described in Chapter 10).

4.4 Biological Shield

The reactor tank is surrounded on all sides by a monolithic reinforced concrete biological shield. The shielding configuration is similar to those at other TRIGA facilities operating at power levels up to 1 MW. Above ground level, the thickness varies from approximately 249 cm. (8 ft 2 in.) at core level to approximately 91 cm. (3 ft.) at the top of the tank.

The massive concrete bulk shield structure provides additional radiation shielding for personnel working in and around the reactor laboratory and provides protection to the reactor core from potentially damaging natural phenomena.

4.5 Nuclear Design

The strong negative temperature coefficient is the principal method for controlling the maximum power (and consequently the maximum fuel temperature) for TRIGA reactors. This coefficient is a function of the fuel composition, core geometry, and temperature. For fuels with [redacted] enrichment, the value is nearly constant at 0.01% Δk/k per °C, only weakly dependent on geometry and temperature.

Fuel and clad temperature define the safety limit. A power level limit is calculated that ensures that the fuel and clad temperature limits will not be exceeded. The design bases analysis indicates that operation at 1,250 kW thermal power with an [redacted]lement across a broad range of core and coolant inlet temperatures with natural convective flow will not allow film boiling that could lead to high fuel and clad temperatures that could cause loss of clad integrity.

Increase in maximum thermal power from 250 to 1,250 kW does not affect fundamental aspects of TRIGA fuel and core design, including reactivity feedback coefficients, temperature safety

limits, and fission-product release rates. Thermal hydraulic performance is addressed in Section 4.6.

4.5.1 Design Criteria - Reference Core

The limiting core configuration for this analysis is a compact core defined by the TRIGA Mk II grid plates (Section 4.2.5). The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and graphite dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids in the E or F (outermost two rings) as required to support experiment operations or limit excess reactivity. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures.

4.5.2 Reactor Core Physics Parameters

The limiting core configuration differs from the configuration prior to upgrade only in the addition of a fourth control rod, taking the place of a graphite dummy element or void experimental position. For this reason, core physics is not affected by the upgrade except for linear scaling with power of neutron fluxes and gamma-ray dose rates.

For comparison purposes, a tabulation of total rod worth for each control element from the K-State reactor from a recent rod worth measurement is provided with the values from the Cornell University TRIGA reactor as listed in NUREG 0984 (Safety Evaluation Report Related to the Renewal of the Operating license for the Cornell University TRIGA Research Reactor).

Table 4.4, 250 kW Core Parameters.	
β (effective delayed neutron fraction)	0.007
ℓ (effective neutron lifetime)	43 μS
α_{Tf} (prompt temperature coefficient)	-$0.017 EC^{-1} @ 250kW ~275EC
α_V (void coefficient)	-0.003 1%$^{-1}$ void
α_p (power temperature coefficient – weighted ave)	-$0.006 kW^{-1} to – $0.01 kW^{-1}

Table 4.5, Comparison of Control Rod Worths.			
KSU TRIGA II (250 kW)		Cornell University (500 kW)	
C-3, Shim	$2.88	D-16, Shim	$2.20
D-16, Regulating	$1.58	D-4, Safety	$1.99
D-10, Pulse	$1.96	D-10, Transient	$1.88
-- TBD	TBD	E-1, Regulating	$0.58

The pulse rod is similar to a standard control rod, and the worth of the pulse rod compares well with the comparable standard control rods in similar ring positions. A maximum pulse is analyzed for thermal hydraulic response and maximum fuel temperature.

4.5.3 Fuel and Clad Temperatures

This section analyzes expected fuel and cladding temperatures with realistic modeling of the fuel-cladding gap. Analysis of steady state conditions reveals maximum heat fluxes well below the critical heat flux associated with departure from nucleate boiling. Analysis of pulsed-mode behavior reveals that film boiling is not expected, even during or after pulsing leading to maximum adiabatic fuel temperatures.

Chapter 4, Appendix A of this chapter reproduces a commonly cited analysis of TRIGA fuel and cladding temperatures associated with pulsing operations. The analysis addresses the case of a fuel element at an average temperature of 1000°C immediately following a pulse and estimates the cladding temperature and surface heat flux as a function of time after the pulse. The analysis predicts that, if there is no gap resistance between cladding and fuel, film boiling can occur very shortly after a pulse, with cladding temperature reaching 470°C, but with stresses to the cladding well below the ultimate tensile strength of the stainless steel. However, through comparisons with experimental results, the analysis concludes that an effective gap resistance of 450 Btu hr^{-1} ft^{-2} $°F^{-1}$ (2550 W m^{-2} K^{-1}) is representative of standard TRIGA fuel and, with that gap resistance, film boiling is not expected. This section provides an independent assessment of the expected fuel and cladding thermal conditions associated with both steady-state and pulse-mode operations.

a. Heat Transfer Models

The overall heat transfer coefficient relating heat flux at the surface of the cladding to the difference between the maximum fuel (centerline) temperature and the coolant temperature can be calculated as the sum of the temperature changes through each element from the centerline of the fuel rod to the water coolant, where the subscripts for each of the ΔT's represent changes between bulk water temperature and cladding outer surface, (br_0), changes between cladding outer surface and cladding inner surface (r_0r_i), cladding inner surface and fuel outer surface – gap (g), and the fuel outer surface to centerline (r_icl):

$$T_{cl} = T_b + \Delta T_{br_0} + \Delta T_{r_0r_i} + \Delta T_g + \Delta T_{r_icl} \qquad (1)$$

A standard heat resistance model for this system is:

$$T_o = T_b + q''\left[\frac{1}{h} + \frac{r_o \ln\left(r_o/r_i\right)}{k_c} + \frac{r_o}{r_i h_g} + \frac{r_o}{2k_f}\right] \tag{2}$$

and heat flux is calculated directly as:

$$q'' = U\Delta T = \frac{T_{max} - T_b}{\frac{1}{h} + \frac{r_o \ln(r_o/r_i)}{k_c} + \frac{r_o}{r_i h_g} + \frac{r_o}{2k_f}}, \tag{3}$$

in which r_o and r_i are cladding inner and outer radii, h_g is the gap conductivity, h is the convective heat transfer coefficient, and k_f is the fuel thermal conductivity. The gap conductivity of 2840 W m^{-2} K^{-1} (500 Btu h^{-1} ft^{-2} $°F^{-1}$) is taken from Appendix A. The convective heat transfer coefficient is mode dependent and is determined in context. Parameters are cross-referenced to source in Table 4.6.

Table 4.6: Thermodynamic Values

Parameter	Symbol	Value	Units	Reference
Fuel conductivity	k_f	18	W m^{-1} K^{-1}	Table 13.3
Clad conductivity	k_g	14.9	W m^{-1} K^{-1} (300 K)	Table 13.3
		16.6	W m^{-1} K^{-1} (400 K)	Table 13.3
		19.8	W m^{-1} K^{-1} (600 K)	Table 13.3
Gap resistance	h_g	2840	W m^{-2} K^{-1}	Appendix A
Clad outer radius	r_0	0.018161	M	Table 13.1
Fuel outer radius	r_i	0.018669	M	Table 13.1
Active fuel length	L_f	0.381	M	Table 13.1
No. fuel elements	N	[redacted]	N/A	Chap 13
Axial peaking factor	APF	$\pi/2$	N/A	Table 13.4

General Atomics reports that fuel conductivity over the range of interest has little temperature dependence, so that:

$$\frac{r_o}{2k_f} = 5.1858\text{E-}04\frac{\text{m}^2K}{\text{W}} \tag{4}$$

Gap resistance has been experimentally determined as indicated, so that:

$$\frac{r_o}{r_i h_g} = 3.6196\text{E-}04\frac{\text{m}^2K}{\text{W}} \tag{5}$$

Temperature change across the cladding is temperature dependent, with values quoted at 300 K, 400 K and 600 K. Under expected conditions, the value for 127°C applies so that:

$$\frac{r_o \ln \frac{r_o}{r_i}}{k_c} = 3.103e-5 \frac{m^2 K}{W} \tag{6}$$

Table 4.7, Cladding Heat Transfer Coefficient

Temp (°K)	Temp (°C)	m^2 K w^{-1}
300	27	3:457e-5
400	127	3.103e-5
600	327	2.601e-5

It should be noted that, since these values are less than 10% of the resistance to heat flow attributed to the other components, any errors attributed to calculating this factor are small.

The convection heat transfer coefficient was calculated at various steady state power levels. A graph of the calculated values results in a nearly linear response function.

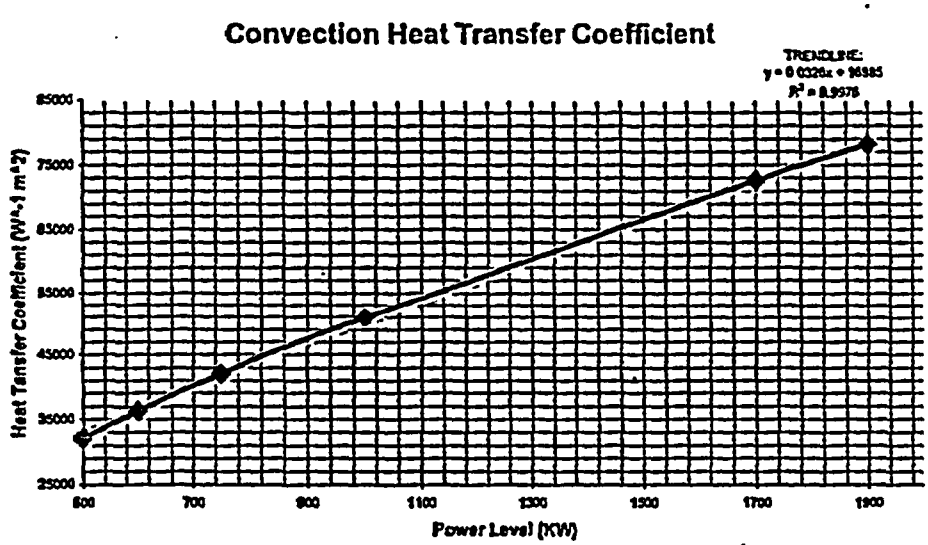


Figure 4.10, Convection Hear Transfer Coefficient versus Power Level

$$\frac{1}{h} = \frac{1}{0.0326P(watts) + 16985} \tag{7}$$

The reference core contains 83 fuel elements; temperature calculations using the reference core are conservative because at least 83 elements are required for steady state 500 kW operations, while analysis assumes 1.25 MW operation. Actual heat production will be less than heat calculated in analysis, so temperatures will be lower. More than 83 elements will distribute heat production across a larger number of fuel elements so that heat flux and temperatures will be lower than calculations based on the reference core. Average heat flux per fuel rod is therefore:

$$q_{av}'' = \frac{power}{area} = \frac{P}{2\pi r_o L_f} \tag{8}$$

With the maximum heat flux of:

$$q_{max}'' = q_{av}''*APF = \frac{P}{83*2\pi r_o L_f}*\frac{\pi}{2} = \frac{P}{332 r_o L_f} = 0.423 m^{-2} P \tag{9}$$

Therefore, core centerline temperature for the fuel rod producing the maximum heat as a function of power can be calculated as:

$$T_o = T_s + 0.423\,P\left[\frac{1}{0.0326\,P + 16985} + 3.103\,e-5 + 3.620\,e-4 + 5.186\,e-4\right] \tag{10}$$

For the purposes of calculation, the two extremes of cladding thermal conductivity were assumed (300 K value and 600 K value) to determine expected centerline temperature as a function of power level. Calculations using both vales are provided graphically, and shows the effect of thermal conductivity changes are minimal. The graph also shows that fuel temperature remains below about 750 °C at power levels up to 1900 kW with pool temperature at 27 °C (300 K), and 1700 kW with pool temperatures at 100 °C.

Finally, temperature calculations for the hottest location in the core were made assuming 1.25 MW steady state power at 20°C and 100°C with the following results:

Table 4.8, Calculated Temperature Data for 1,250 kW Operation

Fuel Centerline °C	Fuel/Gap Interface °C	Gap/Clad Interface °C	Clad/Water Interface °C	Bulk Water °C
503.2	229.0	37.7	21.2	20.0
582.0	307.8	116.4	100.0	100.0

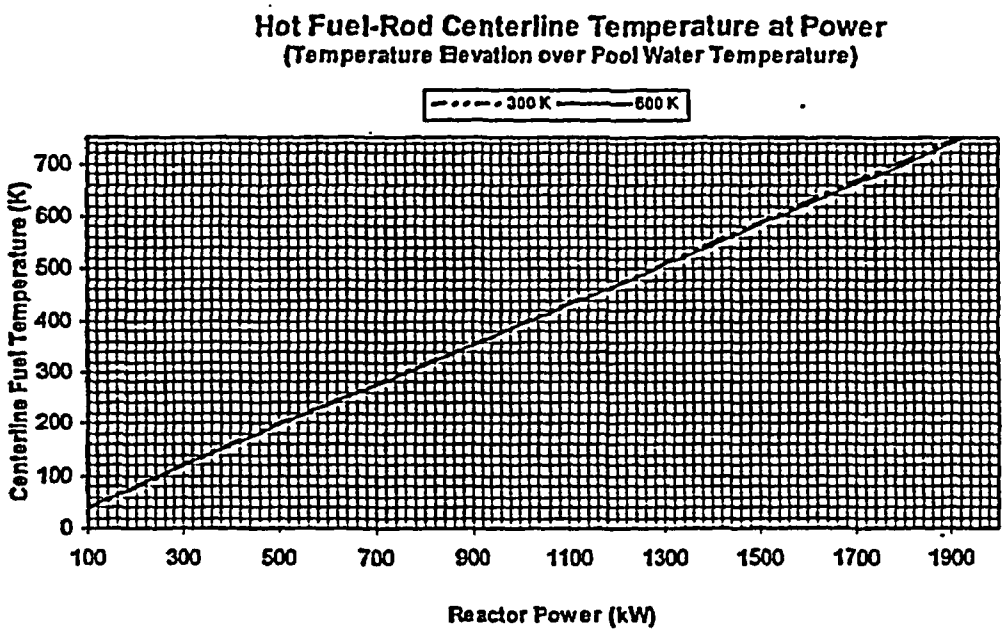


Figure 4.11, Hot Fuel-Rod Centerline Temperature

For subcooled boiling, with water at ambient temperature T_b, the critical heat flux is calculated by (Ivey and Morris 1978)

$$(q''_{chf})_{sub} = (q''_{chf}) * \left[1 + 0.1 * \left(\frac{p_f}{p_g} \right)^{\frac{3}{4}} \frac{c_f (T_{SAT} - T_b)}{\lambda} \right], \tag{11}$$

in which c_f is the heat capacity of the coolant. At the depth of the reactor core in the KSU TRIGA, static pressure is 0.153 MPa and coolant ambient temperature is taken to be 27°C. Thermodynamic and physical properties of water under these conditions are tabulated in Chapter 4 Appendix B.

b. Spatial Power Distribution

The following conservative approximations are made in characterizing the spatial distribution of the power during steady-state operations.

- The hottest fuel element delivers twice the power of the average.

Classically, the radial hot-channel factor for a cylindrical reactor (using R as the physical radius and R_e as the physical radius and the extrapolation distance) is given[2] by:

$$F_r^{\prime} = \frac{1.202 * \left(R/R_e \right)}{J_1\left[2.4048 * \left(R/R_e \right) \right]} \tag{12}$$

with a radial peaking factor of 1.93 for the KSU TRIGA II geometry,. However, TRIGA fuel elements are on the order of a mean free path of thermal neutrons, and there is a significant change in thermal neutron flux across a fuel element. Calculated thermal neutron flux data[3] indicates that the ratio of peak to average neutron flux (peaking factor) for TRIGA cores under a range of conditions (temperature, fuel type, water and graphite reflection) has a small range of 1.36 to 1.40.

Actual power produced in the most limiting actual case is 14% less than power calculated using the assumption; therefore using a peaking factor of 2.0 to determine calculated temperatures and will bound actual temperatures by a large margin, and is extremely conservative.

- The axial distribution of power in the hottest fuel element is sinusoidal, with the peak power a factor of $\pi/2$ times the average, and heat conduction radial only.

The axial factor for power produced within a fuel element is given by:

$$g(z) = 1.514 * \cos\left(\frac{\pi}{2} * \frac{z}{2 * \ell + \ell_{ext}} \right), \tag{13}$$

in which $\ell = L/2$ and ℓ_{ext} is the extrapolation length in graphite, namely, 0.0275 m. The value used to calculate power in the limiting location within the fuel element is therefore 4% higher a power calculated with the actual peaking factor. Actual power produced in the most limiting actual case is 4% less than power calculated using the assumption; therefore calculated temperatures will bound actual temperatures.

- At full power, the thermal power of P_{max} = 1,250 kW is distributed over 83 fuel elements, with a maximum heat flux given by:

[2] Elements of Nuclear Reactor Design, 2nd Edition (1983), J. Weisman, Section 6.3
[3] GA-4361, Calculated Fluxes and Cross Sections for TRIGA Reactors (8/14/1963), G. B. West

$$q''_{\infty} = \frac{\pi * P_{\infty}}{83 * \pi * D_e * L} = 2.414e6 W m^{-1} \tag{14}$$

- The radial and axial distribution of the power within a fuel element is given by

$$q'''(r,z) = q'''_{avg} f(r) g(z), \tag{15}$$

in which r is measured from the vertical axis of the fuel element and z is measured along the axis, from the center of the fuel element. The axial peaking factor follows from the previous assumption of the core axial peaking factor, but (since there is a significant flux depression across a TRIGA fuel element) distribution of power produced across the radius of the fuel the radial peaking factor requires a different approach than the previous radial peaking factor for the core.

- The radial factor is given by:

$$f(r) = \frac{a + cr + er^2}{1 + br + dr^2}, \tag{16}$$

in which the parameters of the rational polynomial approximation are derived from flux-depression calculations for the TRIGA fuel (Ahrens 1999a). Values are: a = 0.82446, b = -0.26315, c = -0.21869, d = -0.01726, and e = +0.04679. The fit is illustrated in Figure 4.11.

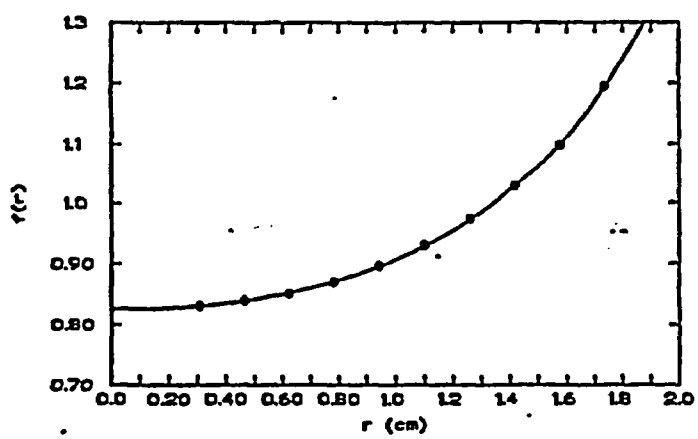


Figure 4.12, Radial Variation of Power Within a TRGIA Fuel Rod.
(Data Points from Monte Carlo Calculations [Ahrens 1999a])

c. **Steady-State Mode of Operation**

Table 4.9 tabulates critical heat flux as a function of coolant temperature from the assumed inlet temperature to the saturation temperature. Also shown in the table is the CHFR, i.e., the ratio of the critical heat flux and the maximum heat flux at full power.

Table 4.9: Critical Heat Flux and CHFR for 1.25 MW Operation at Selected Coolant Temperatures.

T_b (°C)	$T_{sat}-T_b$ (°C)	q''_{cr} (MW m^{-2})	CHFR
27	84.9	6.02	5.8
30	81.9	5.86	5.6
40	71.9	5.33	5.1
60	51.9	4.26	4.1
100	11.9	2.13	2.0
111.9	0.0	1.49	1.4

It is clear from the table that there is a very wide margin between the operating heat flux and the critical heat flux even to unrealistically high pool water temperature, so that film boiling and excessive cladding temperature is not a consideration in steady-state operation. A parametric variation of this calculation for various power levels shows margin to DNBR up to 100°C pool temperature for power levels greater than 1.9 MW.

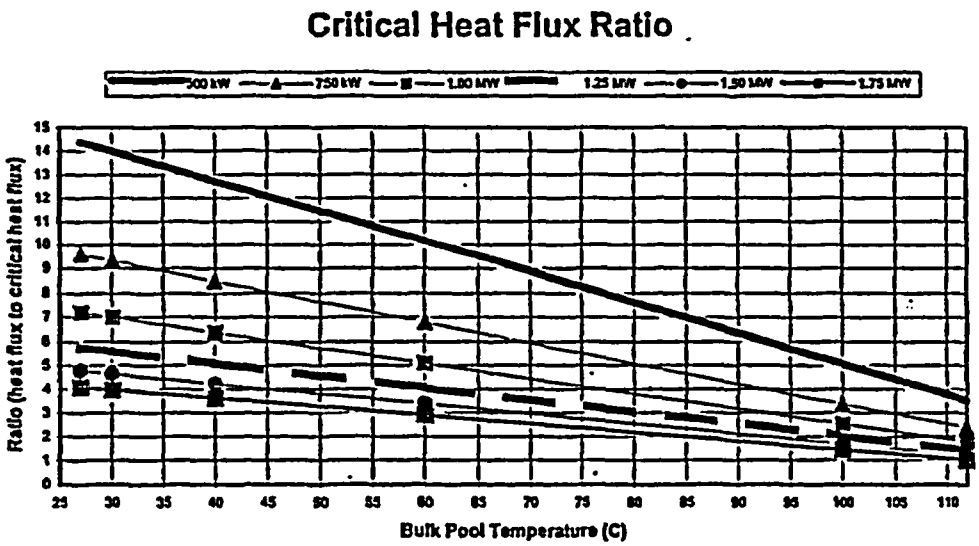


Figure 4.13, Ratio of Departure from Nucleate Boiling to Critical Heat Flux at Various Power Levels

d. Pulsed Mode of Operation

Transient calculations have been performed using a custom computer code TASCOT for transient and steady state two-dimensional conduction calculations (Ahrens 1999). For these calculations, the initial axial and radial temperature distribution of fuel temperature was based on Eqs. (6) and (7), with the peak fuel temperature set to 746 °C, i.e., a temperature rise of 719 °C above 27 °C ambient temperature. The temperature rise is computed in Chapter 13, Section 13.2.3 for a 2.1% ($3.00) pulse from zero power and a 0.7% ($1.00) pulse from power operation. In the TASCOT calculations, thermal conductivity was set to 0.18 W cm^{-1} K^{-1} (Table 4.1) and the overall heat transfer coefficient U was set to 0.21 W cm^{-1} K^{-1}. The convective heat transfer coefficient was based on the boiling heat transfer coefficient computed using the formulation (Chen 1963, Collier and Thome 1994)

$$q'' = h_b(T_w - T_{sat}) = h(T_w - T_b). \tag{17}$$

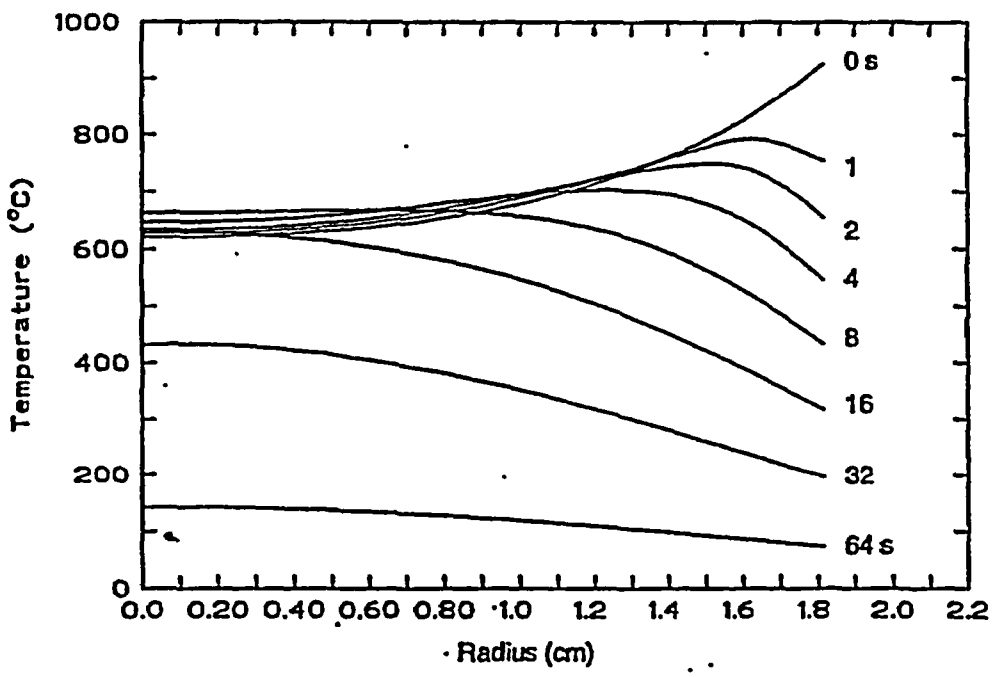


Figure 4. 14, Midplane Radial Variation of Temperature Within the Fuel Subsequent to a $3.00 Pulse.

The boiling heat transfer coefficient is given by the correlation (Forster & Zuber 1955)

$$h_b = 0.00122 * \left[\frac{k_f^{0.79} * c_{pf}^{0.45} * \lambda^{0.51}}{\sigma^{0.5} * \mu_f^{0.29} * \rho_g^{0.24} * \left(\nu_g - \nu_v\right)^{0.75} * T_{sat}^{0.75}} \right] * \left(T_w - T_{sat}\right)^{0.99}, \qquad (18)$$

in which T_w is the cladding outside temperature, T_{sat} the saturation temperature (111.9 °C), and T_b the coolant ambient temperature (27°C). Fluid-property symbols and values are given in Appendix B. Subscripts f and g refer respectively to liquid and vapor phases. The overall heat transfer coefficient U varies negligibly for ambient temperatures from 20 o 60 °C, and has the value 0.21 W cm^{-1} K^{-1} at T_b = 27 °C.

Figure 4.14 illustrates the radial variation of temperature within the fuel, at the midplane of the core, as a function of time after the pulse. Table 4.8 lists temperatures and heat fluxes as function of time after a 2.1% ($3.00) reactivity insertion in a reactor initially at zero power. The CHFR is based on the critical heat flux of 1.49 MW m^{-1} from Eqs. (3) and (4) and from Table 4.2 for saturated boiling. Figure 4A.3 of Appendix A, using the Ellion data, indicates a Leidenfrost temperature in excess of 500°C. Thus transition boiling, but not fully developed film boiling might be expected for a short time after the end of a pulse.

4.6 Thermal Hydraulic Design and Analysis

A balance between the buoyancy driven pressure gain and the frictional and acceleration pressure losses accrued by the coolant in its passage through the core determines the coolant mass flow rate through the core, and the corresponding coolant temperature rise. The buoyancy pressure gain is given by

$$\Delta p_g = \rho_o \beta_o \Delta T g L, \qquad (19)$$

in which ρ_o and β_o are the density and volumetric expansion coefficient at core inlet conditions (27°C, 0.15285 Mpa), g is the acceleration of gravity, 9.8 cm^2 s^{-1}, ΔT is the temperature rise through the core, and L is the height of the core (between gridplates), namely, 0.556 m. The frictional pressure loss is given by

$$\Delta p_f = \frac{\dot{m}^2 f L}{2 A^2 D_h \rho_o}, \qquad (20)$$

in which $\dot{m}$ is the coolant mass flow rate (kg s^{-1}) in a unit cell approximated as the equivalent annulus surrounding a single fuel element, A is the flow area, namely, 0.00062 m^2, and D_h is the hydraulic diameter, namely, 0.02127 m. The friction factor f for laminar flow through the annular area is given by 100 Re^{-1} (Shah & London 1978), in which the Reynolds number is given by $D_h \dot{m} / A \mu_o$ in which μ_o is the dynamic viscosity at core inlet conditions.

Table 4.10, Heat Flux and Fuel Temperatures Following a $3.00 Pulse from Zero Power, with 27(°C) Coolant Ambient Temperature.

Time (s)	Q'' (W/m²)	CHFR	Fuel outside Temp. (°C)	Clad surface Temp. (°C)
0	-	-	953	-
1	3.57×10^5	4.2	781	224
2	7.34×10^5	2.0	683	432
4	8.52×10^5	1.7	574	498
8	7.54×10^5	2.0	461	443
16	5.71×10^5	2.6	344	342
32	3.46×10^5	4.3	224	218
64	1.04×10^5	14.4	100	84

Entrance of coolant into the core is from the side, above the lower grid plate (see Section 4.2.5), and the entrance pressure loss would be expected to be negligible. The exit contraction loss is given by

$$\Delta p_e = \frac{\dot{m}^2 K}{2\rho_o A^2}. \tag{21}$$

The coefficient K is calculated from geometry of an equilateral-triangle spacer in a circular opening, for which

$$K \cong \left[\frac{A_t}{A_c}\right]^2 = \left[\frac{3 * R^2 \sin 60^\circ \cos 60^\circ}{\pi * R^2}\right] = 0.171, \tag{22}$$

where R is the radius of the opening in the upper grid plate. Equations (11) through (13), solved simultaneously yield the mass flow rates per fuel element, and coolant temperature rises through the core listed in Table 4.9.

Table 4.11, Coolant Flow Rate and Temperature Rise for Natural-Convection Cooling the TRIGA Reactor During Steady-State Operations.

P (kW)	$\dot{m}$ (kg/s)	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

4.7 Safety Limit

As described in 3.5.1 (Fuel System) and NUREG 1282, fuel temperature limits both steady-state and pulse-mode operation. The fuel temperature limit stems from potential hydrogen outgassing from the fuel and the subsequent stress produced in the fuel element clad material by heated hydrogen gas. Yield strength of cladding material decreases at a temperature of 500°C; consequently, limits on fuel temperature change for cladding temperatures greater than 500°C. A maximum temperature of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) will limit internal fuel cladding stresses that might lead to clad integrity (NUREG 1282) challenges.

4.8 Operating Limits

4.8.1 Operating Parameters

The main safety consideration is to maintain the fuel temperature below the value that would result in fuel damage. Setting limits on other operating parameters, that is, limiting safety system settings, controls the fuel temperature. The operating parameters established for the KSU TRIGA reactor are:

- Steady-state power level
- Fuel temperature measured by thermocouple during pulsing operations
- Maximum step reactivity insertion of transient rod

4.8.2 Limiting Safety System Settings

Heat transfer characteristics (from the fuel to the pool) controls fuel temperature during normal operations. As long as thermal hydraulic conditions do not cause critical heat flux to be exceeded, fuel temperature remains well below any limiting value. Figure 4.13 illustrates that critical heat flux is not reached over a wide range of pool temperatures and power levels. As indicated in Table 4.9, the ratio of actual to critical heat flux is at least 2.0 for temperatures less than 100°C bulk pool water temperature for 1.25 MW operation. Operation at less than 1.25 MW ensures fuel temperature limits are not exceeded by a wide margin.

Limits on the maximum excess reactivity assure that operations during pulsing do not produce a power level (and generate the amount of energy) that would cause fuel-cladding temperature to exceed these limits; no other safety limit is required for pulsed operation.

4.8.3 Safety Margins

For 1,250 kW steady-state operations, the critical heat flux ratio indicated in Table 4.9 ranges from 15.3 for pool water at room temperature (20°C) to 10.2 at 60 °C (pool temperatures are controlled to less than 48°C for operational concerns). Even at pool water temperatures

approaching boiling, the margin remains above 3. Therefore, margins to conditions that could cause excessive temperatures during steady state operations while cladding temperatures is below 500°C are extremely large.

Normal pulsed operations initiated from power levels below 10 kW with a $3.00 reactivity insertion result in maximum hot spot temperatures of 746°C, a 34% margin to the fuel temperature limit. As indicated in Chapter 13, pulsed reactivity insertions of $3.00 from initial conditions of power operation can result in a maximum hot spot temperature of 869°C. Although administratively controlled and limited by an interlock, this pulse would still result in a 15% margin to the fuel temperature safety limit for cladding temperatures below 500°C.

Analysis shows that cladding temperatures will remain below 500°C when fuel is in water except following large pulses. However, mechanisms that can cause cladding temperature to achieve 500°C (invoking a 950°C fuel temperature limit) automatically limit fuel temperature as heat is transferred from the fuel to the cladding.

Immediately following a maximum pulsed reactivity additions, heat transfer driven by fuel temperature can cause cladding temperature to rise above 500°C, but the heat transfer simultaneously cools the fuel to much less than 950°C.

If fuel rods are placed in an air environment immediately following long-term, high power operation, cladding temperature can essentially equilibrate with fuel temperature. In worst-case air-cooling scenarios, cladding temperature can exceed 500°C, but fuel temperature is significantly lower than the temperature limit for cladding temperatures greater than 500°C.

4.9 Bibliography

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Appendix 4-A
Post-Pulse Fuel and Cladding Temperature

This discussion is reproduced from Safety Analysis Reports for the University of Texas Reactor Facility (UTA 1991) and the McClellan Nuclear Radiation Center (MNRC 1998).

The following discussion relates the element clad temperature and the maximum fuel temperature during a short time after a pulse. The radial temperature distribution in the fuel element immediately following a pulse is very similar to the power distribution shown in Figure 4A.1. This initial steep thermal gradient at the fuel surface results in some heat transfer during the time of the pulse so that the true peak temperature does not quite reach the adiabatic peak temperature. A large temperature gradient is also impressed upon the clad which can result in a high heat flux from the clad into the water. If the heat flux is sufficiently high, film boiling may occur and form an insulating jacket of steam around the fuel elements permitting the clad temperature to tend to approach the fuel temperature. Evidence has been obtained experimentally which shows that film boiling has occurred occasionally for some fuel elements in the Advanced TRIGA Prototype Reactor located at GA Technologies [Coffer 1964]. The consequence of this film boiling was discoloration of the clad surface.

Thermal transient calculations were made using the RAT computer code. RAT is a 2-D transient heat transport code developed to account for fluid flow and temperature dependent material properties. Calculations show that if film boiling occurs after a pulse it may take place either at the time of maximum heat flux from the clad, before the bulk temperature of the coolant has changed appreciably, or it may take place at a much later time when the bulk temperature of the coolant has approached the saturation temperature, resulting in a markedly reduced threshold for film boiling. Data obtained by Johnson et al. [1961] for transient heating of ribbons in 100°F water, showed burnout fluxes of 0.9 to 2.0 Mbtu ft^{-2} hr^{-1} for e-folding periods from 5 to 90 milliseconds. On the other hand, sufficient bulk heating of the coolant channel between fuel elements can take place in several tenths of a second to lower the departure from nucleate boiling (DNB) point to approximately 0.4 Mbtu ft^{-2} hr^{-1}. It is shown, on the basis of the following analysis, that the second mode is the most likely; i.e., when film boiling occurs it takes place under essentially steady-state conditions at local water temperatures near saturation.

A value for the temperature that may be reached by the clad if film boiling occurs was obtained in the following manner. A transient thermal calculation was performed using the radial and axial power distributions in Figures 4A.1and 4A.2, respectively, under the assumption that the thermal resistance at the fuel-clad interface was nonexistent. A boiling heat transfer model, as shown in Figure 4A.3, was used in order to obtain an upper limit for the clad temperature rise. The model used the data of McAdams [1954] for subcooled boiling and the work of Sparrow and Cess [1962] for the film boiling regime. A conservative estimate was obtained for the minimum heat flux in film boiling by using the correlations of Speigler et al. [1963], Zuber [1959], and Rohsenow and Choi [1961] to find the minimum temperature point at which film boiling could occur. This calculation gave an upper limit of 760°C clad temperature for a peak initial fuel temperature of 1000°C, as shown in Figure. 4A.4. Fuel temperature distributions for this case are shown in Figure 4A.5 and the heat flux into the water from the clad is shown in Figure 4A.6. In this limiting case, DNB occurred only 13 milliseconds after the pulse, conservatively calculated

assuming a steady-state DNB correlation. Subsequently, experimental transition and film boiling data were found to have been reported by Ellion [9] for water conditions similar to those for the TRIGA system. The Ellion data show the minimum heat flux, used in the limiting calculation described above, was conservative by a factor of 5. An appropriate correction was made which resulted in a more realistic estimate of 470°C as the maximum clad temperature expected if film boiling occurs. This result is in agreement with experimental evidence obtained for clad temperatures of 400°C to 500°C for TRIGA Mark F fuel elements which have been operated under film boiling conditions [Coffer et al. 1965].

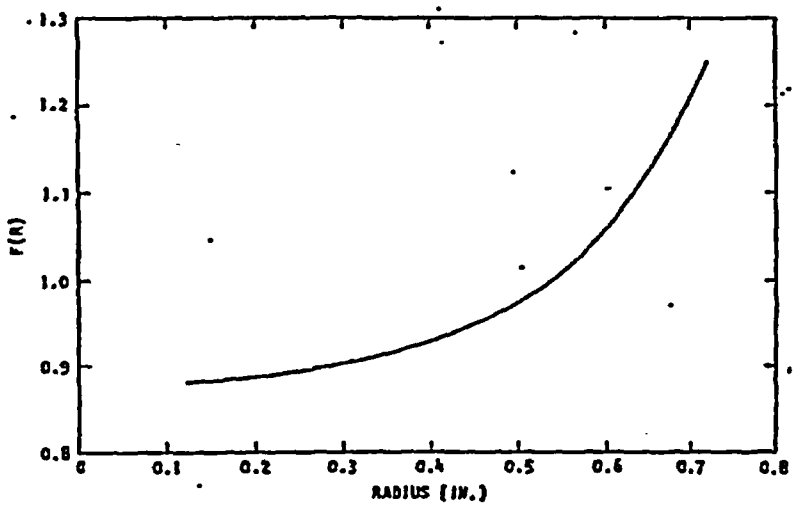

Figure 4A.1. Representative Radial Variation of Power Within the TRIGA Fuel Rod

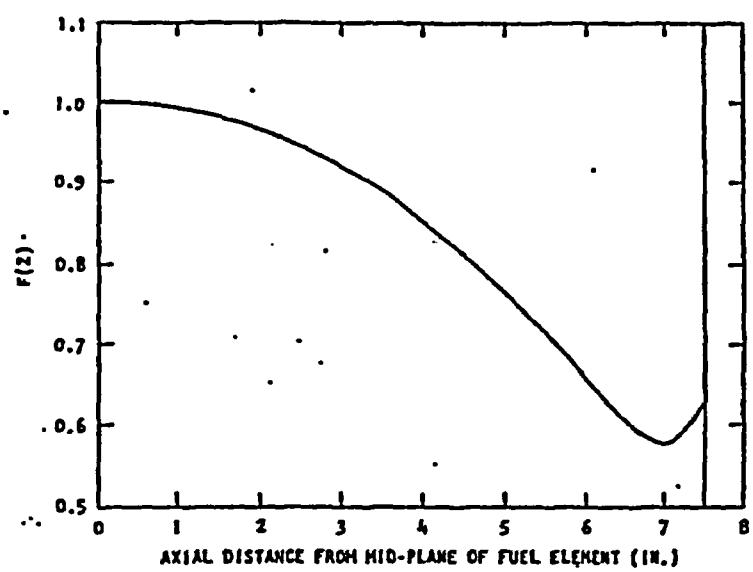

Figure 4A.2, Representative Axial Variation of Power Within the TRIGA Fuel Rod.

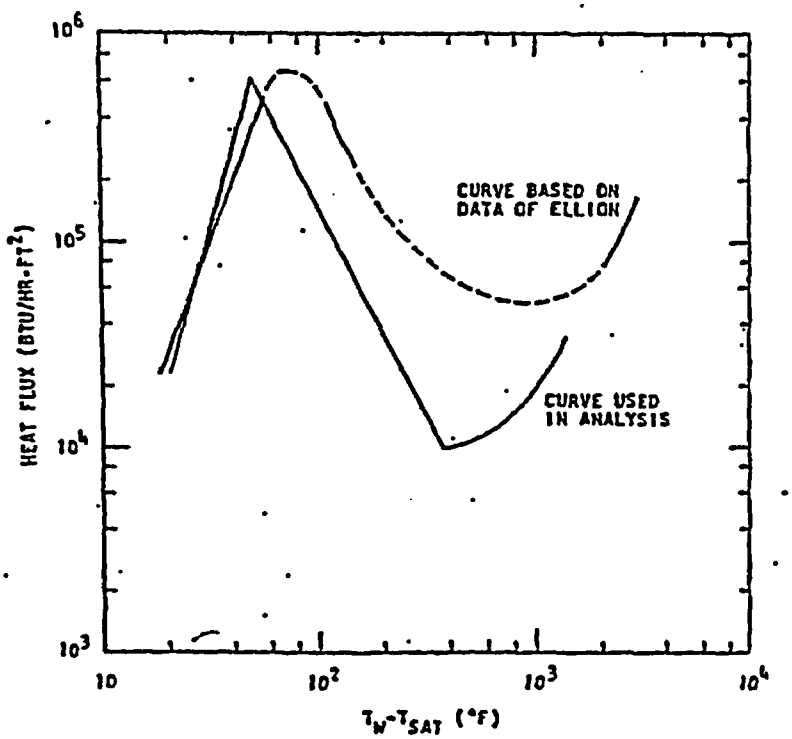

Figure 4A.3, Subcooled Boiling Heat Transfer for Water.

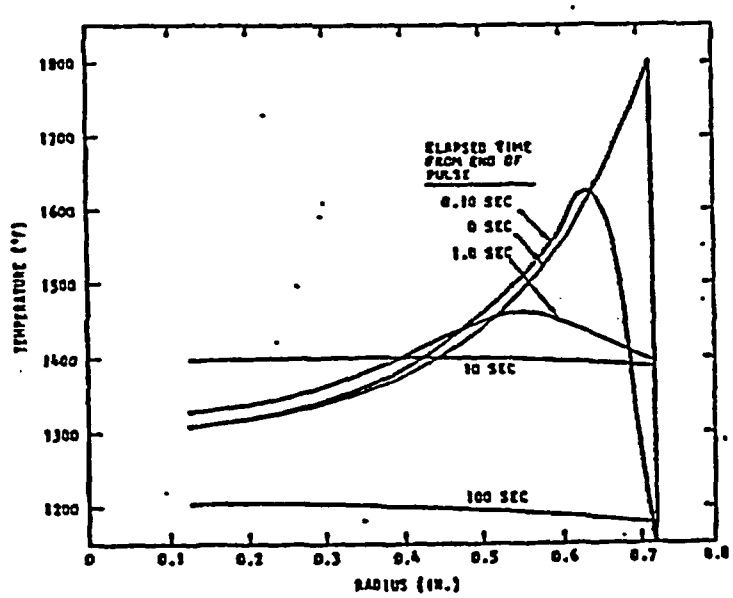

Figure 4A.4, Fuel Body Temperature at the Midplane of a Well-Bonded Fuel Element After Pulse.

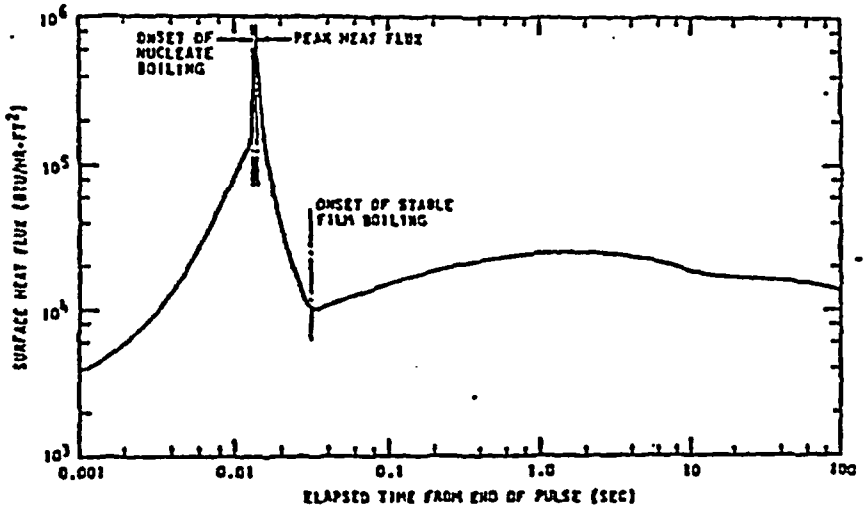

Figure 4A.5, Surface Heat Flux at the Midplane of a Well Bonded Fuel Element After a Pulse.

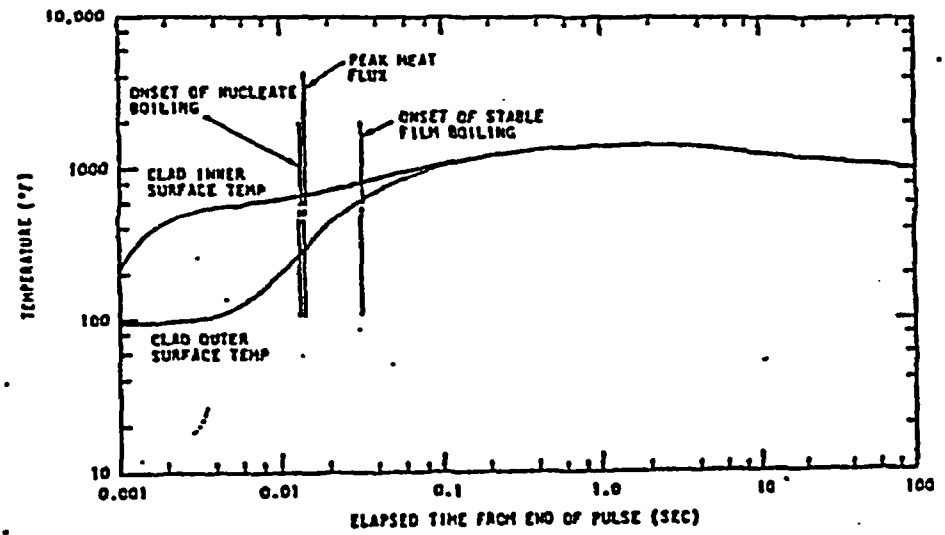

Figure 4A.6, Clad Temperature at Midpoint of Well-Bonded Fuel Element.

The preceding analysis assessing the maximum clad temperatures associated with film boiling

assumed no thermal resistance at fuel-clad interface. Measurements of fuel temperatures as a function of steady-state power level provide evidence that after operating at high fuel temperatures, a permanent gap is produced between the fuel body and the clad by fuel expansion. This gap exists at all temperatures below the maximum operating temperature. (See, for example, Figure 16 in the Coffer report [1965].) The gap thickness varies with fuel temperature and clad temperature so that cooling of the fuel or overheating of the clad tends to widen the gap and decrease the heat transfer rate. Additional thermal resistance due to oxide and other films on the fuel and clad surfaces is expected. Experimental and theoretical studies of thermal contact resistance have been reported [Fenech and Rohsenow 1959, Graff 1960, Fenech and Henry 1962] which provide insight into the mechanisms involved. They do not, however, permit quantitative prediction of this application because the basic data required for input are presently not fully known. Instead, several transient thermal computations were made using the RAT code. Each of these was made with an assumed value for the effective gap conductance, in order to determine the effective gap coefficient for which departure from nucleate boiling is incipient. These results were then compared with the incipient film boiling conditions of the 1000°C peak fuel temperature case.

For convenience, the calculations were made using the same initial temperature distribution as was used for the preceding calculation. The calculations assumed a coolant flow velocity of 1 ft per second, which is within the range of flow velocities computed for natural convection under various steady-state conditions for these reactors. The calculations did not use a *complete boiling curve heat transfer model, but instead, included a convection cooled region* (no boiling) and a subcooled nucleate boiling region without employing an upper DNB limit. The results were analyzed by inspection using the extended steady-state correlation of Bernath [1960] which has been reported by Spano [1964] to give agreement with SPERT II burnout results within the experimental uncertainties in flow rate.

The transient thermal calculations were performed using effective gap conductances of 500, 375, and 250 Btu ft^{-2} hr^{-1} $°F^{-1}$. The resulting wall temperature distributions were inspected to determine the axial wall position and time after the pulse which gave the closest approach between the local computed surface heat flux and the DNB heat flux according to Bernath. The axial distribution of the computed and critical heat fluxes for each of the three cases at the time of closest approach is given in Figures 4A.7 through 4A.9. If the minimum approach to DNB is corrected to TRIGA Mark F conditions and cross-plotted, an estimate of the effective gap conductance of 450 Btu ft^{-2} hr^{-1} $°F^{-1}$ is obtained for incipient burnout so that the case using 500 is thought to be representative of standard TRIGA fuel.

The surface heat flux at the midplane of the element is shown in Figure 4A.10 with gap conductance as a parameter. It may be observed that the maximum heat flux is approximately proportional to the heat transfer coefficient of the gap, and the time lag after the pulse for which the peak occurs is also increased by about the same factor. The closest approach to DNB in these calculations did not necessarily occur at these times and places, however, as indicated on the curves of Figures 4A.7 through 4A.9. The initial DNB point occurred near the core outlet for a local heat flux of about 340 kBtu ft^{-2} hr^{-1} $°F^{-1}$ according to the more conservative Bernath correlation at a local water temperature approaching saturation.

This analysis indicates that after operation of the reactor at steady-state power levels of 1 MW(t), or after pulsing to equivalent fuel temperatures, the heat flux through the clad is reduced and therefore reduces the likelihood of reaching a regime where there is a departure from nucleate boiling. From the foregoing analysis, a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1000°C is conservatively estimated to be 470°C.

As can be seen from Figure 4.7, the ultimate strength of the clad at a temperature of 470°C is 59,000 psi. If the stress produced by the hydrogen over pressure in the can is less than 59,000 psi, the fuel element will not undergo loss of containment. Referring to Figure 4.8, and considering U-ZrH fuel with a peak temperature of 1000°C, one finds the stress on the clad to be 12,600 psi. Further studies show that the hydrogen pressure that would result from a transient for which the peak fuel temperature is 1150°C would not produce a stress in the clad in excess of its ultimate strength. TRIGA fuel with a hydrogen to zirconium ratio of at least 1.65 has been pulsed to temperatures of about 1150°C without damage to the clad [Dee et al. 1966].

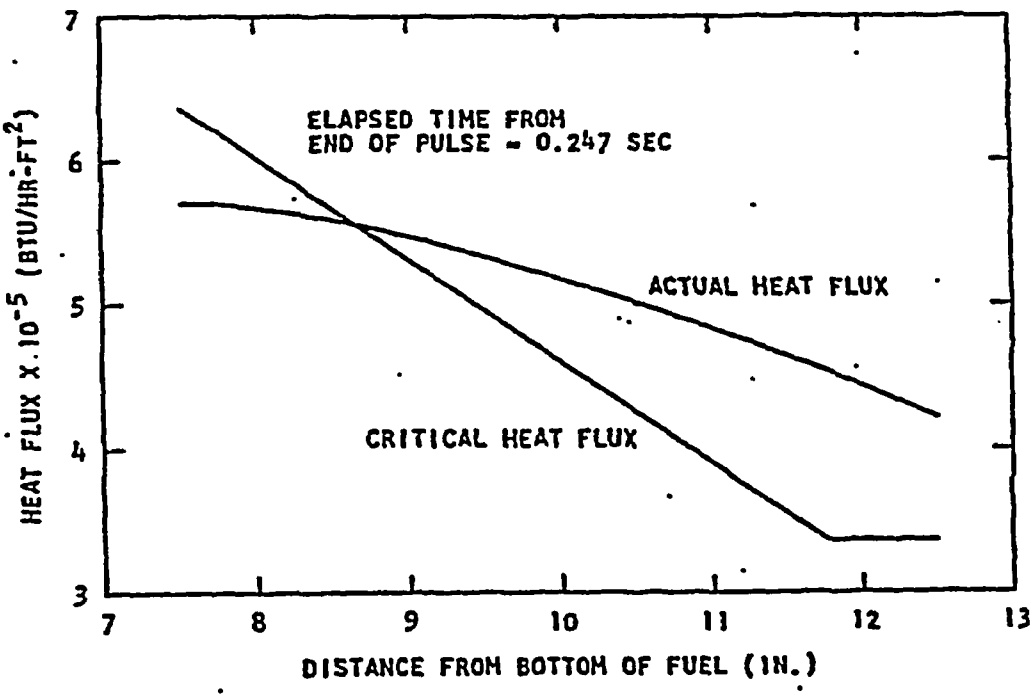

Figure 4A.7, Surface Heat Flux Distribution for Standard Non-Gapped (h_{gap}= 500 Btu/h ft^2 °F) Fuel Element After a Pulse.

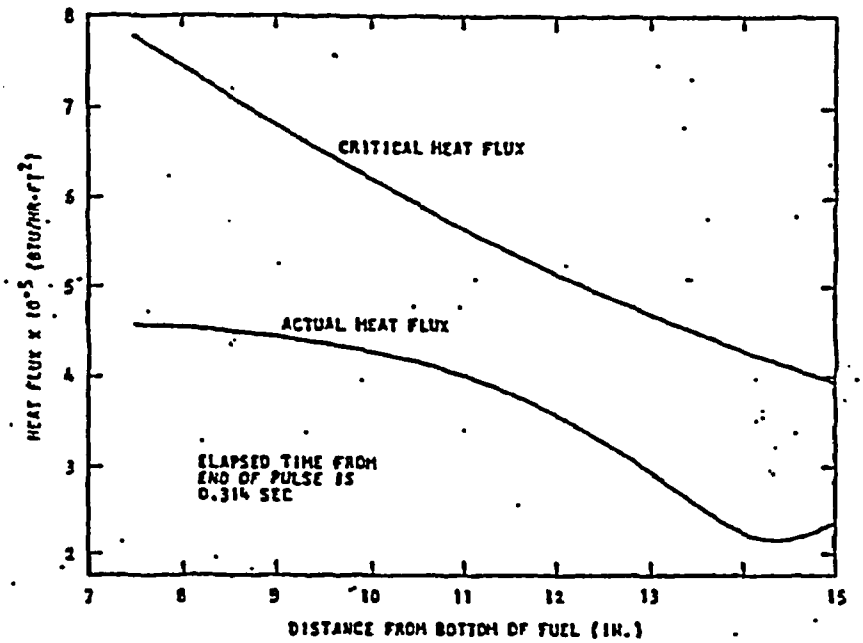

Figure 4A.8, Surface Heat-Flux Distribution for Standard Non-Gapped Fuel Element (h_{gap}= 375 Btu/h ft^2 °F) After a Pulse.

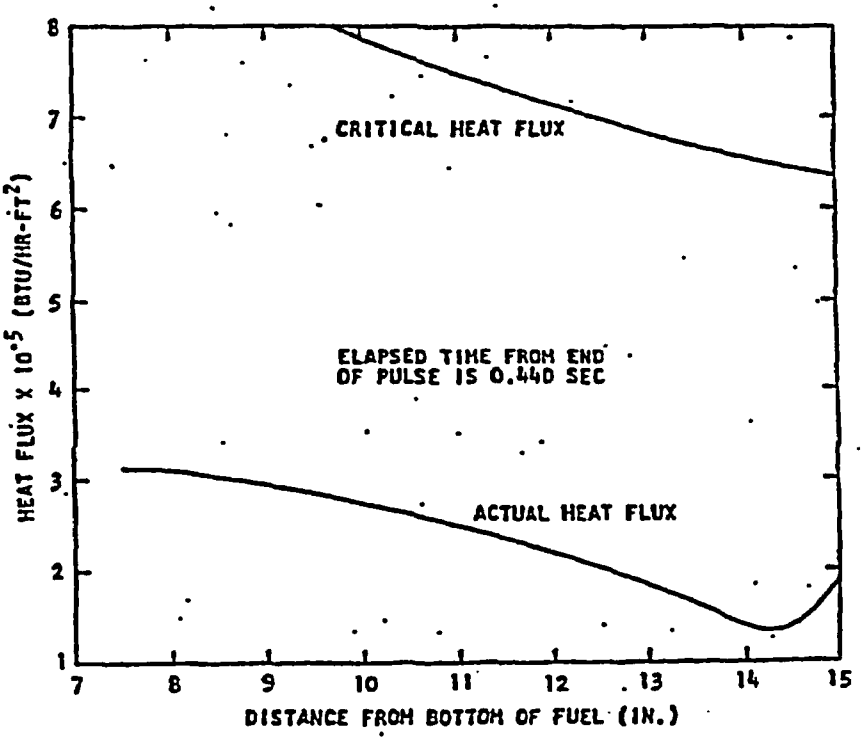

Figure 4A.9, Surface Heat-Flux Distribution for Standard Non-Gapped Fuel Element (h_{gap}= 250 Btu/h ft^2 °F) After a Pulse.

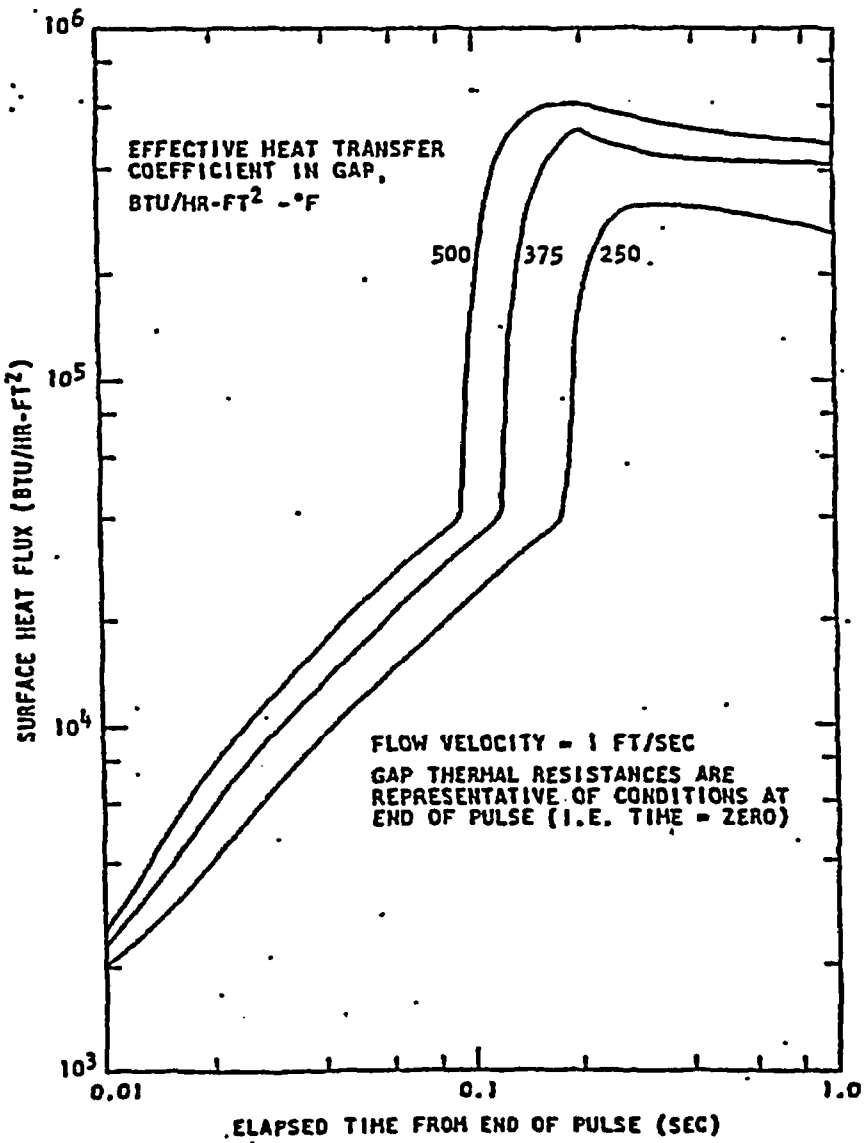

Figure 4A.10, Surface Heat Flux at Midpoint vs. Time for Standard Non-Gapped Fuel Element After a Pulse.

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Appendix B
Water Properties at Nominal Operating Conditions

The following data are from the NBS/NRC Steam Tables, by L. Haar, J.S. Gallagher, and G.S. Kell, Hemisphere 1984.

Saturated Water at 0.153 MPa		
TEMPERATURE, T	111.9	°C
PRESSURE, P	0.153	MPa
HEAT OF VAPORIZATION, λ	2.22E+03	kJ kg^{-1}
SURFACE TENSION, σ	5.66E-02	J m^{-2}
ρ, density (kg m^{-3})	9.50E+02	8.78E-01
C_v, heat capacity (kJ kg^{-1} K^{-1})	3.71E+00	1.55E+00
C_p, heat capacity (kJ kg^{-1} K^{-1})	4.24E+00	2.09E+00
s, entropy (kJ kg^{-1} K^{-1})	1.44E+00	7.22E+00
i, enthalpy (kJ kg^{-1})	4.70E+02	2.69E+03
u, internal energy (kJ kg^{-1})	4.69E+02	2.52E+03
sonic speed (m s^{-1})	1.53E+03	4.79E+02
k, thermal conductivity (W m^{-1} K^{-1})	6.82E-01	2.65E-02
μ, dynamic viscosity (kg m^{-1}s^{-1})	2.50E-04	1.27E-05
ν, kinematic viscosity (m^{2}s^{-1})	2.64E-07	1.45E-05
α, thermal diffusivity (m^{2}s^{-1})	1.70E-07	1.44E-05
Pr, Prandtl Number	1.55E+00	1.00E+00
β, volumetric expansion coefficient	8.14E-04	2.86E-03

Subcooled Water at 27°C, 0.153 MPa

TEMPERATURE	(T)	27 °C
PRESSURE	(P)	0.153 MPa
DENSITY	(ρ)	9.97E+02 kg m^{-3}

C_v, heat capacity (kJ kg^{-1} K^{-1})	4.18E+00
C_p, heat capacity (kJ kg^{-1} K^{-1})	4.18E+00
s, entropy (kJ kg^{-1} K^{-1})	3.95E-01
i, enthalpy (kJ kg^{-1})	1.19E+02
u, internal energy (kJ kg^{-1})	1.18E+02
sonic speed (m s^{-1})	1.50E+03
k, thermal conductivity (W m^{-1}K^{-1})	6.11E-01
μ, dynamic viscosity (kg m^{-1}s^{-1})	8.52E-04
ν, kinematic viscosity (m^{2}s^{-1})	8.55E-07
α, thermal diffusivity (m^{2}s^{-1})	1.46E-07
Pr, Prandtl Number	5.88E+00
β, volumetric expansion coefficient	2.78E-04

5. REACTOR COOLANT SYSTEMS

The reactor coolant systems are very simple in design and operation. These systems are required from an operational standpoint, and not safety. Potential hazards associated with the primary cooling system are minimal. Many of these systems have been upgraded in recent years to permit extended full power operation of the reactor. A general overview of the reactor coolant system is follows.

During full power operation, the fuel elements in the reactor core are cooled by natural convection of the primary tank water. To remove bulk heat to the environment, primary water is circulated through a heat exchanger where the heat is transferred to a secondary cooling loop. Water passing through the secondary side of the heat exchanger is then passed through a forced-draft cooling tower to transfer heat to outside air.

A cleanup loop maintains cleanliness of the primary water to minimize production of long-lived radionuclides and minimize corrosion. Radioactive contamination of the primary water does not present a hazard to workers (in fact, activities are low enough to be directly released to sanitary sewerage). Primary coolant does provide shielding directly above the reactor core, and loss of this shielding would elevate radiation levels in the reactor bay, especially directly above the reactor tank.

Loss of primary water would also deprive the reactor fuel of its principal means of cooling. However design analysis of TRIGA fuel shows that it may be cooled by natural convection in air, without risk of fuel failure. In the event of a water loss, makeup water for the primary system is available from a still or the bulk shield tank. If repairs of the primary tank require draining, fuel elements can be stored in either dry storage pits or the bulk shield tank.

The primary and secondary cooling system flows are isolated by a plate-type heat exchanger. A rupture in the heat exchanger could cause mixing of the coolant streams. Therefore, although the primary coolant is of little radiological significance, the secondary coolant is monitored periodically for radioactivity to detect even a small breach between the two systems.

5.1 Summary Description

The reactor cooling system serves 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 (to minimize corrosion of reactor components, particularly the fuel elements)
4. Control primary water radioactive contamination (by removing nearly all particulate and soluble impurities)
5. Maintain optical clarity of the primary water

The primary system contains de-ionized water and is open to atmosphere. The reactor core is cooled by natural convection alone. To permit extended operation at full licensed power, bulk heat is transferred by forced convection across a heat exchanger to the secondary cooling system. The secondary cooling system then transfers the heat to the environment via a cooling tower, using service water treated for corrosion and biological growth.

This cooling system combination provides enough heat removal for continuous full-power operation. In addition to the cooling system, the reactor is provided with a bulk-shielding tank. This 6500-gallon (25 kL) tank contains distilled water, and can be used to supplement make-up water for the primary tank (using a makeup water system independent of the above drawing) or provide temporary fuel storage. Makeup water for both systems is provided by a steam-powered still.

In normal (automatic) control of the cooling systems, a single backlit pushbutton switch on the control console energizes the primary and secondary pump, and the cooling tower fan control system. Individual controls for system components are located on the 0-foot level. In addition to control systems, several cooling system parameters can be monitored in the control room. Measurement functions available to the reactor operator are presented in Table 5.1.

Table 5.1: Control Room Instrumentation

Measurement	Location	Device / Output
Primary Water Conductivity	Cleanup Loop Inlet	Platinum Probe / Meter
	Cleanup Loop Outlet	Platinum Probe / Meter
Primary Water Temperature	Water Box	RTD / Meter
	Heat Exchanger Inlet	Transducer / Computer
	Heat Exchanger Outlet	Transducer / Computer
	Reactor Tank (× 2)	2 Transducers / Computer
Primary Flow Rate	Orifice	2 Transducers / Computer
Primary Radioactivity	Water Box	G-M Tube / Meter
	Pool Surface	G-M Tube/ Meter
Secondary Water Temperature	Heat Exchanger Inlet	Transducer / Computer
	Heat Exchanger Outlet	Transducer / Computer

5.2 Primary Coolant System

Principal functional requirements of the primary coolant system are to (1) transfer heat from the reactor core to the secondary cooling system, and (2) provide radiation shielding directly above the reactor core. Although natural convection cools the reactor core, primary bulk water temperature should be kept below 130°F (48.9 °C). This limit could easily be reached during extended operation at full licensed power, hence the need for a heat transfer loop. The second requirement involves the 16-feet (4.9 m) of shielding that the water provides directly above the reactor core.

The primary coolant system layout is shown in Figure 5.1. The system consists principally of a reactor tank, a centrifugal pump, a pressure orifice, one side of the heat exchanger, and a cleanup loop (described in Section 5.4). Since the primary coolant is de-ionized water, system components and piping are constructed of either aluminum or stainless steel. Most of the piping is aluminum of nominal 2.5-in (6.4 cm) diameter.

Cooling system inlet piping has two suctions, one just under the water surface and a second suction through a skimmer, which collects foreign particles on the pool surface. An installed valve may be used to isolate the skimmer. The tank itself provides 22-ft (6.7 m) of head (9.5 psig). This water is pulled into a self-priming, direct-coupled, centrifugal pump located on the 0-foot level of the reactor bay. The pump boosts water pressure to about 60 psig (410 kPa), with a flow rate of approximately 110 gallons per minute (6.94 L s^{-1}).

From the primary pump, the flow is split, with about 10 gallons per minute (0.63 L s^{-1}) diverted through a cleanup loop and the remainder passing through an orifice to the heat exchanger, also located on the 0-foot level. The orifice provides the necessary pressure drop to flow water through the cleanup loop. Exit pressure on the orifice (and hence the entrance pressure to the heat exchanger) is approximately 30 psig (210 kPa).

A plate-type compact heat exchanger is used to remove heat from the primary coolant (see Figure 5.2). The heat exchanger consists of sandwiched stainless steel plates alternately carrying primary and secondary cooling water. The heat exchanger has a transfer capacity of 682 kW

(2,327,080. BUT h^{-1}) under normal conditions. The unit was originally pressure tested at 130 psig (0.89 MPa). Four temperature transducers are located in both inlet and outlet streams, as well as two in the reactor tank, for performance monitoring of the heat exchanger. Additional plates may be added to the heat exchanger to provide expand cooling capacity. The compact heat exchanger and temperature probes were installed in 1993.

In addition to cooling functions, the primary tank provides 16 ft (4.9 m) of shielding directly above the reactor core. Water level is normally kept within a few inches of the top of the reactor tank. To prevent loss of primary coolant during maintenance operations, four valves allow isolation of the heat exchanger or reactor tank. In the event of a rupture in the primary piping, a siphon break is located about one foot below the water surface of the tank. The siphon break is a small hole in the pipe that allows air into the pipe if water level drops below the break point, breaking the siphon. The beam ports, when closed, are sealed on the outside by a gasket. A system of pipes connects the beam ports to the manifold with a pressure gauge, where any rupture would be indicated as increased pressure.

If a major loss of coolant were to occur, there are three level sensors that would illuminate lights on the control panel. Two sensors are located in the reactor bay sump, activating when the sump level is high. Since all floor drains in the reactor bay connect to the sump, any leaks would accumulate there. A third sensor is located at the top of the tank, activating if the tank level drops a few inches below normal operating levels. If these indicators fail, loss of coolant would be apparent by increased radiation readings on the remote area monitors. The increased radiation levels directly above the reactor core could present a radiological hazard on the 22-foot level.

The primary reactor tank serves as a shutdown cooling pool and spent fuel pool, therefore the impact of loss of primary water on cooling fuel elements was evaluated. General Atomics calculations for aluminum-clad fuel show that after shutdown from infinite operation at 250 kW that the maximum temperature at the fuel cladding interface would be less than 150°C. For these conditions, the pressure exerted by trapped air and fission products is about 660 psi (4.6 MPa), whereas the yield stress for the aluminum cladding is >5000 psi (34 MPa) at 150°C. The Division of Reactor Licensing validated these findings in the 1968 Safety Evaluation. Current fuel inventory has stainless steel cladding, with two spare aluminum clad instrumented elements. Stainless steel has greater yield strength, hence greater resistance to cladding failure. Technical Specification G.2 allows for spent fuel elements to be stored in air or water.

5.3 Secondary Cooling System

The secondary cooling system is designed for continuous operation at approximately 2,437,500 BTU h^{-1} (723 kW). The secondary cooling system circulates water from the heat exchanger through a cooling tower. The system uses service water that is treated to minimize corrosion and bacteriological growth. The system consists of a surge tank, centrifugal pump, heat exchanger, and cooling tower. The system is regulated by an automatic control system consisting of several temperature sensors, a pneumatic three-way valve, and electrical controls. With the exception of the cooling tower, all equipment is located on the 0-foot level of the reactor bay.

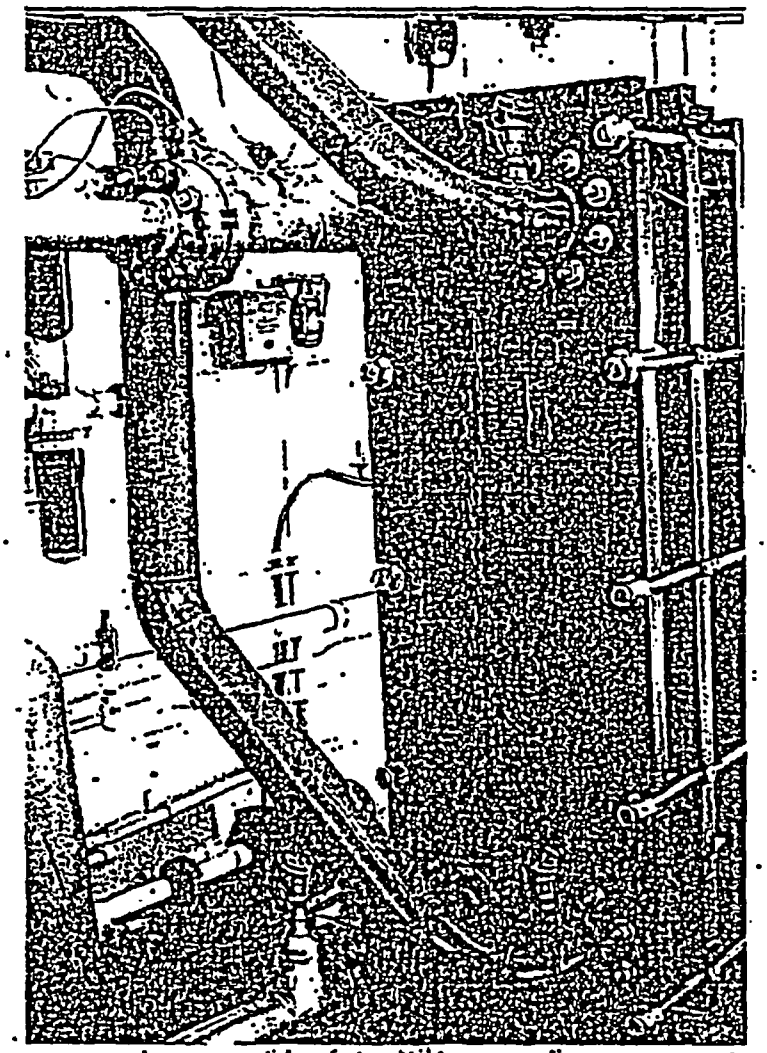

Figure 5.2: Heat Exchanger

5.3.1 Secondary Cooling System Flows

Water is drawn into the secondary cooling system from a surge tank. The tank has a volume greater than that of the cooling tower and stores this water when the tower is not in use. The tank has a sight gauge that is visible from the control room, and an indicator on the control console, which illuminates on low water level. Service water makeup can be initiated from the control room by resetting a five-minute timer (to prevent overfilling) that energizes an electric valve connected to potable water. The makeup line terminates above the tank, providing an air-gap for protection of potable water.

The water from the surge tank is then drawn into the system by a direct-coupled, self-priming, centrifugal pump (see Figure 5.3). When the controls are in a normal configuration, the pump is

energized (with the primary pump and cooling tower fans) from a backlit push button switch in the control room. Normal flow rate through the system is 250 gallons per minute (15.8 $L \cdot s^{-1}$).

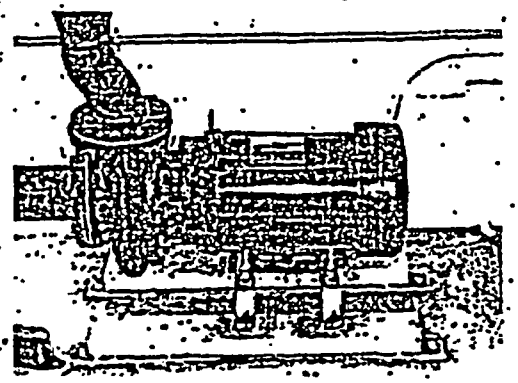

Figure 5.3: Secondary Cooling Pump

After leaving the pump the water passes through the heat exchanger, receiving heat from the primary loop. The temperature transducers on the heat exchanger provide temperature data in the control room. Several manual temperature gauges are also located at various locations in the reactor bay.

Normally, water leaving the heat exchanger is then sent to the cooling tower (Figure 5.4) and returned to the surge tank. However during cold weather conditions (when outside air is less than –10 °F, -23 °C), the water bypasses the cooling tower at a three-way valve and is returned directly to the pump. Any water in the cooling tower then drains into the surge tank. The cooling tower is an induced-draft, crossflow design constructed of galvanized steel with a 146-ton cooling capacity. It is located outside the reactor building, outside a fenced area at about the 12-foot level of the reactor. The tower contains 142 gallons (538 L) of water during operation. It has a two-speed fan controlled by the secondary automatic control system. The tower was replaced along with the surge tank in 1991. The tower was upgraded in 2001 to permit 500 kW operations.

5.3.2 Secondary Cooling Automatic Control System

The secondary automatic control system has three basic control functions:

1. To maintain the primary water at a set temperature.
2. To control the cooling tower fan speed to maintain secondary water at a set temperature.
3. To prevent cooling water freeze-up during cold weather operation (less than –10 °F {-23 °C}).

The secondary automatic control system performs these functions by pneumatically controlling the three-way valve and electrically controlling the cooling tower fan. The secondary automatic control system is normally initiated by energizing the primary cooling pump. The three-way

valve allows secondary water to flow to the cooling tower under normal conditions. If the outside air temperature is less than –10 °F (-23.3 °C), then the three-way valve stops cooling tower flow. This outside air temperature limit is bypassed when the primary water temperature exceeds 110 °F (43.3 °C) thus reestablishing cooling tower flow. The temperature of the secondary water returning from the cooling tower controls cooling tower fan speed. At 70 °F (21.1 °C), the cooling tower fan starts at low speed and switches to high speed when the temperature reaches 90°F (32.2 °C). Manual control of fan speed is available at the 0-foot level breaker.

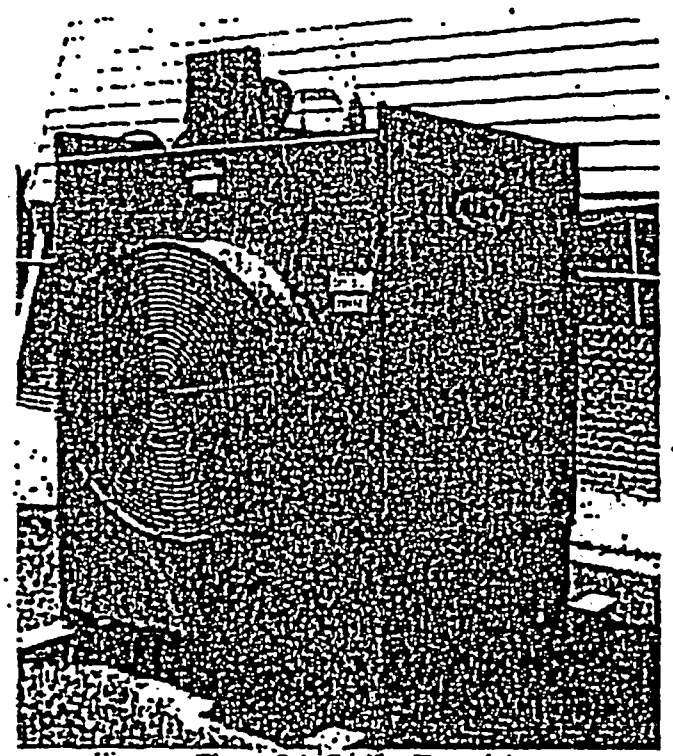

Figure 5.4: Cooling Tower

5.3.3 Secondary Water Quality

Secondary water chemistry analysis is typically performed twice a month. Items tested are pH, chlorine, conductivity, and total alkalinity. Suggested manufacturers limits on pH are 7.6 to 8.6, and our typical values are 7.5 to 8.0. Chlorine levels and conductivity should be less than 4 times the maximum city concentration. Chlorine levels and conductivity of the city water are 2-4 ppm at point of use and 0.4 to 0.7 mS cm^{-1}. Upon reaching a conductivity value 4 times that of service water, the cooling water is replaced. Total alkalinity should be less than 270 ppm, although this limit has never been reached during normal operation.

To detect possible leaks in the heat exchanger, the secondary water is tested monthly for radioactivity. Since the primary tank has 22-feet (6.7 m) of static head, as opposed to 3 to 9-feet (0.9 to 2.7 m) in the secondary (depends on surge tank level), a breach in the heat exchanger would result in flow from the primary to secondary cooling system when the cooling system is secured. Leaks while the cooling system is operating are unlikely; the heat exchanger was

pressure tested to 150 psig, and the maximum pressure differential across the heat exchanger (primary pump running, secondary pump off) is 30 psig. A small breach in the heat exchanger would be evidenced by tritium contamination of the secondary water. A larger breach would be indicated by loss of primary coolant from the reactor tank. Primary water typically remains within 10CFR20 limits for release to sanitary sewers, and is not a hazard even if a leak were to occur.

5.4 Primary Coolant Cleanup System

Technical specification requires average monthly water conductivity be kept below 2 $\mu S\ cm^{-1}$ to minimize corrosion of reactor components and production of radioactive materials, as well as maintaining optical quality of the coolant. To maintain this low conductivity the primary coolant system was constructed with an integral cleanup loop, located on the 0-foot level of the reactor bay. As shown in Figures 5.1 and 5.5, this cleanup loop consists of a water box, a fiber cartridge filter, a mixed-bed resin demineralizer, a flow meter, two conductivity probes, a RTD temperature probe, and a Geiger tube. Conductivity measurement equipment was replaced in 1990. Connecting piping is 1-in (2.5 cm) aluminum.

Water is drawn out of the primary system after leaving the primary pump and returns to the system with the water exiting the heat exchanger. A flow orifice (Figure 5.6) is installed in the main primary loop before the heat exchanger to provide the necessary pressure drop to force water through the cleanup loop. Upon entering the cleanup loop the water passes through the water box, where temperature, conductivity, and radioactivity are measured. Normal inlet conductivity is 0.8 to 1.3 $\mu S\ cm^{-1}$.

The water is then passed through a filter assembly that contains two replaceable cartridges of at least 5-micron rating. Pressure gauges are supplied on either side of the filter to indicate clogging. In addition to improving the optical clarity of the water in the reactor tank, the removal of solid particles extends the life of the demineralizer resin.

The water then passes through a mixed-bed demineralizer to remove soluble impurities. The typical resin in use is Amberlite ARN-150. After the demineralizer, the water passes a second conductivity probe. Typical exit conductivity is 0.05 to 0.2 $\mu S\ cm^{-1}$. Flow rate through this loop is regulated by a plug-type flow meter with a 0 to 28 gallon per minute range. A manual valve accomplishes regulation. Normal flow rate through this loop is 10 gallons per minute. Normal pressure in the loop is 55 to 60 psig.

Primary coolant is sampled every month for radioactivity. The only nuclide of significance that is normally detected in the coolant is tritium and is usually in the range of 200 to 1500 $pCi \cdot mL^{-1}$. In the event of a fuel cladding failure, the cleanup loop could be used to remove radioactive contaminants.

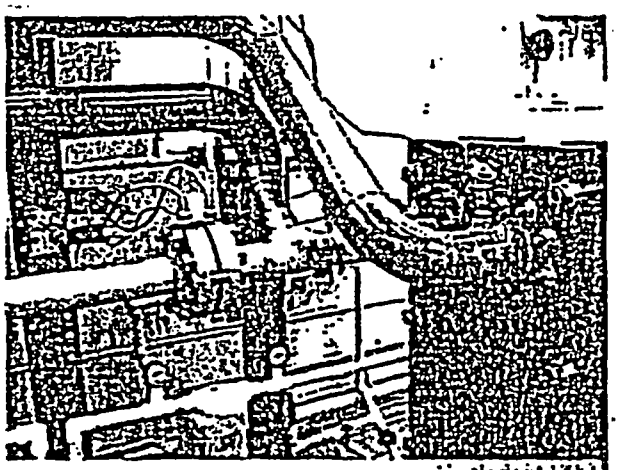

Figure 5.6: Flow Orifice

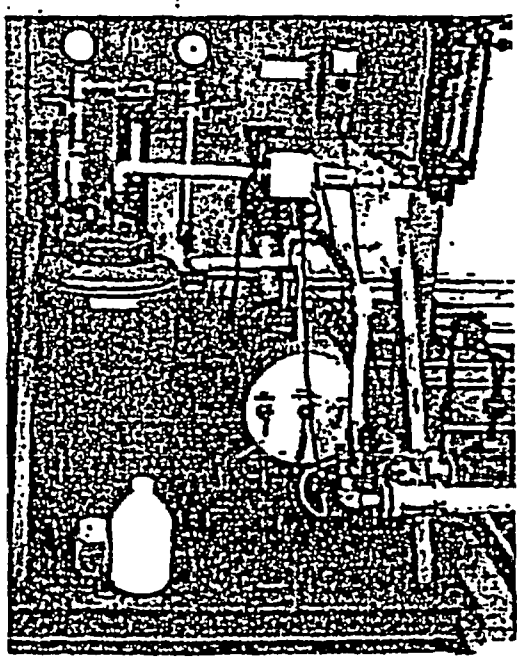

Figure 5.5: Primary Coolant Cleanup Loop

5.5 Makeup Water System

Makeup water for the primary system and the bulk shield tank is provided by a still located in the basement adjacent to the reactor bay (Figure 5.7). The still is a steam-powered unit, with steam supplied by the University's power plant. The steam is distributed in underground passages making service relatively secure from ordinary hazards of weather. The still is rated to produce 50 gallons distilled water per hour (3.2 L min^{-1}) and has a storage capacity of 80 gallons (300 L). Input water to the still is conditioned by a water softener to minimize scaling in the boiler section. In addition to the still, the bulk shield tank contains 6500 gallons (25 kL) of distilled water, which could be used as makeup water.

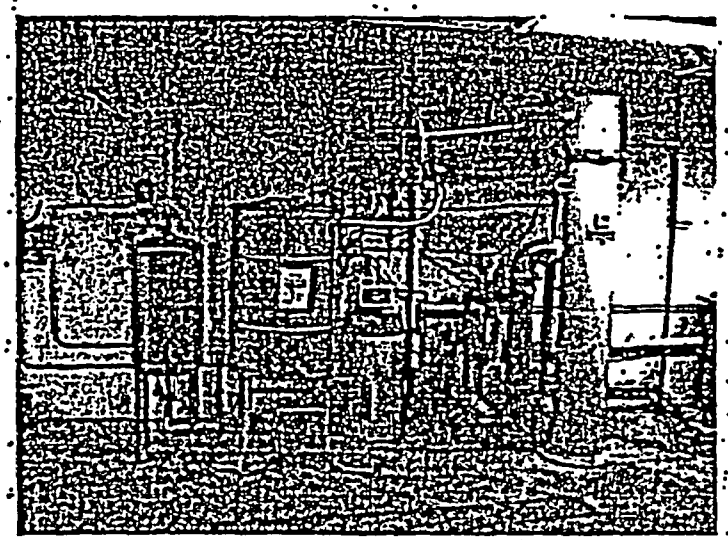

Figure 5.7: Makeup Water System

5.6 Nitrogen-16 Control System

The cooling systems return line to the reactor pool enters the pool through a diffuser. The diffuser is constructed to induce a helical flow pattern in the reactor tank. This extends transport time of the convection flow of water from the core to allow much of the nitrogen-16 generated during operation to decay before reaching the pool surface. A radiation monitor directly above the pool surface provides the control room operator with information to prompt exposure controls (generally energizing the primary cooling pump to initiate the helical flow for decay, or limiting access to the area directly over the pool). Pool surface monitor radiation measurements at 250 kW directly above the pool surface are typically 20 to 30 $mR \cdot h^{-1}$ from all sources with the primary cooling system operating, and is expected to be 90 $mR \cdot h^{-1}$ at 500 kW operation, expected to be approximately 350 $mR \cdot h^{-1}$ at 1,250 kW operation (slightly less than 100 $mR \cdot h^{-1}$ at 1 meter above the bridge).

A radiation monitor at the rail around the pool provides the control room operator with information to prompt exposure controls for personnel on the 22-foot level but not directly over the reactor pool. At 250 kW, radiation levels at the rail are less than 2 mR/hr.

5.7 Auxiliary Systems Using Primary Coolant

Although the bulk shield tank does not circulate with the primary coolant, it can be used as a source of distilled water. Shielding of the thermalizing column by this tank is only needed for operation. The bulk shield tank water circulation system has the provision to allow for water to be pumped directly from the bulk shield tank into the primary tank, providing 6500 gallons (25 kL) of makeup water. The bulk shield tank can also be used for temporary fuel storage during maintenance operations involving the primary tank. Makeup water for the bulk shield tank is provided by the same still as used for the primary system.

5.8 Bibliography

GA 7860, Safety Analysis Report for TRIGA Reactors using Aluminum-Clad Fuel, 6 March 1967 General Atomics/General Dynamics.

Safety Evaluation by the Division of Reactor Licensing for Increase in Power Level, Skovholt, Donald J. . 26 June 1968.

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6. Engineered Safety Features

As discussed in Chapter 13, from previous analysis, and from experience at other TRIGA reactors, emergency core cooling is not required for operations at steady state thermal powers below 1900 kW. No engineered safety features are required for the KSU reactor because the steady state power limit is 500kW.

6.1 Bibliography

Kansas State University TRIGA Mark II Reactor Hazards Summary Report, by R.W. Clack, J.R. Fagan, W.R. Kimel, and S.Z. Mikhail, License R-88, Docket 50-188, 1961.

Analysis of Certain Hazards Associated with Operation of the Kansas State University TRIGA Mark II Reactor at 250 kW Steady State and with Pulsed Operation to $2.00, by R.W. Clack, et al., and the Safety Evaluation by the U.S. Atomic Energy Commission Division of Reactor Licensing, License R-88, Docket 50-188, 1968.

NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," U.S. Nuclear Regulatory Commission, 1987.

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7. PPINSTRUMENTATION AND CONTROL SYSTEMS

Much of the reactor's original instrumentation and control (I&C) systems were replaced during the control room modifications in 1993 and 1994. The original console and vacuum-tube instruments were replaced by a surplus solid-state console obtained from U.S. Geological Survey's TRIGA Mark I reactor. This console was then outfitted with new N-1000 series neutronic channels from General Atomics. These channels have optically isolated outputs, allowing other devices to utilize the neutronic data.

7.1 Summary Description

The bulk of the reactor I&C systems are hard-wired analog systems primarily manufactured by General Atomics and widely used at various NRC-licensed facilities. The general layout of these systems is shown in Figure 7.1.

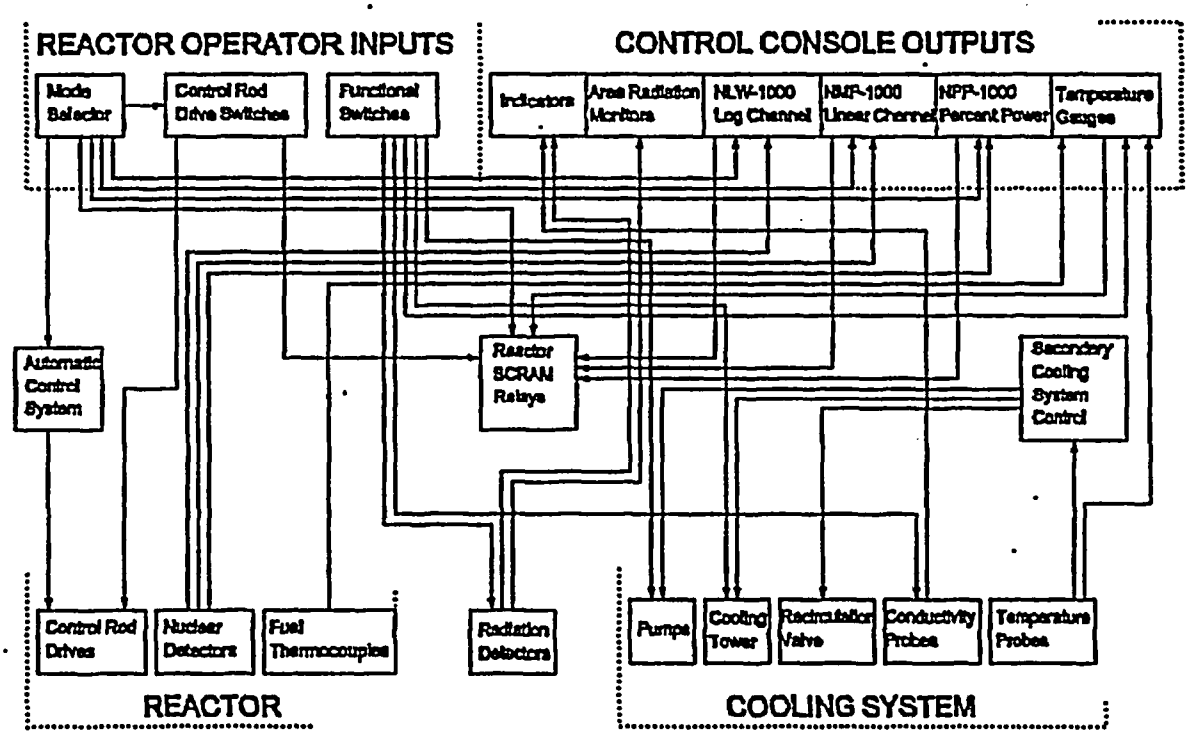


Figure 7.1, Inter-connectivity Diagram.

The reactor control system (RCS) consists of the instrumentation channels, the control rod drive circuitry and interlocks, and an automatic flux controller. The RCS measures several key reactor parameters including power, fuel temperature, water temperature, and water conductivity. Three

neutronic instruments measure reactor power separately: a wide-range logarithmic channel, a multi-range linear channel, and a percent power channel. These provide at least two indications of reactor power from source range to power range. Additionally, if a reactor pulse is performed another channel is added to the central thimble to record pulse data. Fuel temperatures can be monitored on both the console and on an auxiliary panel. Primary water temperature is displayed on the console and measured by an RTD in the water box. Titanium electrodes at the entrance and exit of the cleanup loop measure water conductivity.

The control rod drives and their associated circuitry are simple in design. A rotary switch configures the primary mode of operation, namely automatic, steady-state, or pulse mode. Numerical indicators give drive position, with illuminated switches to manipulate the rods and to indicate rod and drive status. Several interlocks are incorporated to prevent unintentional rapid insertions of reactivity, except in pulse mode. An automatic control system links the RCS with the neutronic channels providing regulation of the power level.

The reactor protection system (RPS) is a component of the RCS instruments. The RPS will initiate a reactor scram if any of several measured parameters in the RCS are outside of their limited safety system settings. The reactor scram effectively places the reactor in a subcritical configuration by releasing the control rods from their respective drives. Since the rods are no longer physically attached to the drive, they fall into the reactor core by gravity. High reactor power, high fuel temperature, loss of detector high voltage, loss of building power, and short reactor period will automatically cause all of the control rods to be dropped into the reactor core. A bar above the control rod drive switches allows this system to be actuated manually. Since the core is cooled by natural convection, no other engineered safety features are necessary for safe reactor shutdown.

The control console and display instruments are primarily housed in a control console, with auxiliary instruments located in a rack next to the console. At the console, the reactor operator has direct control over mode of operation, control rod drive positions, cooling system operation, opening of reactor bay doors, and manual scram of the reactor. Display instruments located in the control console provide measurements of reactor power, control rod positions, primary water temperature, and fuel temperature. Indicators in the console display scram information; low air pressure, low primary water level, high reactor sump water level, sump high water level, sump overflow water level, secondary surge tank level low, source interlock status; reactor bay upper door open, reactor bay lower door open, thermal column door open, person on stairway, and rod drive status. Secondary surge tank makeup is controlled with a backlit pushbutton that indicates surge tank low level and surge tank makeup valve operation. An intercom system on the console provides communication to numerous locations around the reactor bay and staff offices. In the auxiliary rack, the operator can control pneumatic transfer system operation, actuate timers, and add water to the secondary cooling system. Display instruments located in this rack include, primary water conductivity, water activity, remote area radiation monitors, fuel temperature, and a strip-chart output of reactor power. Several audible alarms indicate high radiation levels in the primary coolant and at various locations throughout the reactor bay. A breaker-box in the control room provides control over electrical devices in the reactor facility, including ventilation systems.

Radiation protection instruments are distributed throughout the reactor bay. All instruments have visual indication of radiation level, visible alarm conditions, and audible alarms. Radiation area

monitors (RAM) strategically cover potential radiation areas throughout the reactor bay. A combined pool surface and primary water monitor indicated water activity. A 5 $R \cdot h^{-1}$ evacuation alarm is located on the 22-foot level. A continuous air monitor (CAM) is energized during reactor operation.

The human-machine interface principles incorporated into control room design allow the reactor to be operated by a single individual. All monitoring instruments are visible to the reactor operator at the console. The 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 signal the operator to abnormal conditions.

7.2 Design of Instrumentation and Control System

7.2.1 Design Criteria

Reliability of essential equipment is ensured through redundancy. Multiple instruments and safety systems perform similar functions for all modes of reactor operation. The construction and installation of instruments was performed according to applicable regulations at the time of introduction. However, all crucial instruments are checked daily for calibration and operability. Testing and calibration procedures exist for repair and general service. The majority of these I&C systems were manufactured by General Atomics, or other industrial manufacturers of nuclear equipment. Crucial systems to be considered include neutronic instruments, control rod drives, radiation monitors, and control systems.

Redundancy is designed into each of these systems. During steady state operation, a minimum of two neutronic channels provide reactor power level indication, two of which provide high power level RPS actuation (scram). These neutronic instruments are tested prior to reactor operation for demonstration of scram capability. Two are also tested for operability by internal calibration tests. There are two fuel temperature indications. The control rod drives drop their rods into the reactor core upon loss of power or RPS actuation, providing sufficient shutdown margin with even the most reactive rod stuck out. Multiple remote area radiation monitors cover important areas, including two directly above the reactor core and two monitoring primary coolant activity.

The maximum steady state power level for KSU TRIGA Mark II reactor is proposed to be 500 kW. Similar reactors operate up to 2 MW with 2 GW pulses. Therefore, the limited safety systems settings associated with reactor are extremely conservative when compared to the safety limits of the reactor. Thus the reactor has a considerable safety margin.

7.2.2 Design-Basis Requirements

The primary function of the RCS is to govern the manner in which reactivity is varied in the reactor core. The RCS system should prevent the reactor operator from unintentionally inserting large amounts of reactivity, through various interlock systems. The operator should only be able to remove one rod at a time from the reactor core, preventing large insertion rates. The pulse rod must not be able to be rapidly ejected from the core while in steady-state operation. Furthermore, the pulse rod should be the only rod that can be withdrawn in pulse mode, preventing

supercritical pulses. There should also be an interlock to prevent startup without a power level signal above the minimum instrument sensitivity, preventing unmonitored or unanticipated criticality. Rod position indicators should show the rod position to 0.2% or total travel for accurate reactivity calculations.

Another primary function of the RCS is to provide the reactor operator with reactor status information. Reactor power, a crucial parameter, requires at least two instruments to provide confirmation of reactor power from shutdown to operating levels. Three instruments are used to cover this range: a wide-range logarithmic channel, a multi-range linear channel, and a percent power channel. Accuracy of measurement at full rated power increases accordingly with the refinement of scale. The log channel provides gross reactor power indication and is accurate to 20% of scale, the linear channel is accurate to 5% of scale, and the percent power channel is accurate to 3% of scale. The percent power channel will also display pulse parameters for large pulses. These instruments are calibrated annually and checked for operability at the start of each operating day. An additional channel is installed and calibrated in the central thimble to record pulse data. Fuel temperature must be monitored during pulsing operation.

The primary function of the RPS is to automatically insert the control rods into the reactor core when certain parameters deviate from limited safety system settings. Several scrams involve the neutronic channels in the RCS. If 110% rated power level is exceeded in steady state mode, one of two trip-points will scram the reactor. Failure of the high voltage power supplies for operating neutronic channels will also cause a scram. Manual scram will be available in all modes of reactor operation. Rod drop times for the standard rods will be measured regularly to ensure proper RPS function. No other ESF features are required in this design.

The primary function of the radiation monitoring instruments is for personnel protection measures and emergency assessment actions. The area monitors provide the reactor operator with information regarding the actual radiation environment inside the reactor bay. With this knowledge, reactor users can be informed of possible hazards. A 5 $R \cdot h^{-1}$ monitor on the 22-foot level signals personnel to evacuate the reactor bay. A number of survey instruments (ion chambers, rem balls, G-M counters) are also available to personnel. Other instruments such as the constant air monitor, pool surface monitor, and water box monitor indicate the presence of dispersible radioactive materials, an indication of possible fuel cladding failures.

The control room is designed so a single operator can manipulate all significant controls without leaving the room. The reactor operator should be able to de-energize all equipment and experiments in the reactor bay. The control room should provide sufficient ventilation to provide cooling of the reactor instruments.

7.2.3 System Description

The overall system layout is depicted in Figure 7.2. The majority of the RCS is housed in a General Atomics (GA) console originally manufactured for the USGS reactor, which is shown with modifications in Figure 7.2. A detailed description of this figure is provided in Table 7.1. Figure 7.3 shows a representative layout of the auxiliary instrumentation rack. Since the instrument racks are general use equipment, configuration may be changed to allow better utilization of space, installation of new equipment, support specific equipment modifications, etc. without affecting function. The functions of each piece of equipment in this configuration are discussed in following sections with additional figures showing location and layout.

7.2.4 System Performance Analysis

The system performance of the current I&C systems surpasses the original equipment. Reliability has been high, with few unanticipated reactor shutdowns. Since daily checkouts are performed, any discrepancies would be observed and corrected in a prompt manner. The opto-isolated outputs of the neutronic channels allow the data to be utilized by other devices without concern over those devices affecting the channels. A line conditioner provides regulated power to the instruments, protecting the equipment from electrical disruptions.

7.2.5 Conclusion

The current I&C systems outperform the original equipment supplied with the reactor, while meeting all of the necessary design bases for the facility. The human design factors used in control room development allow the reactor to be operated by a single individual. Checkout and testing procedures ensure that all equipment is maintained in operational status.

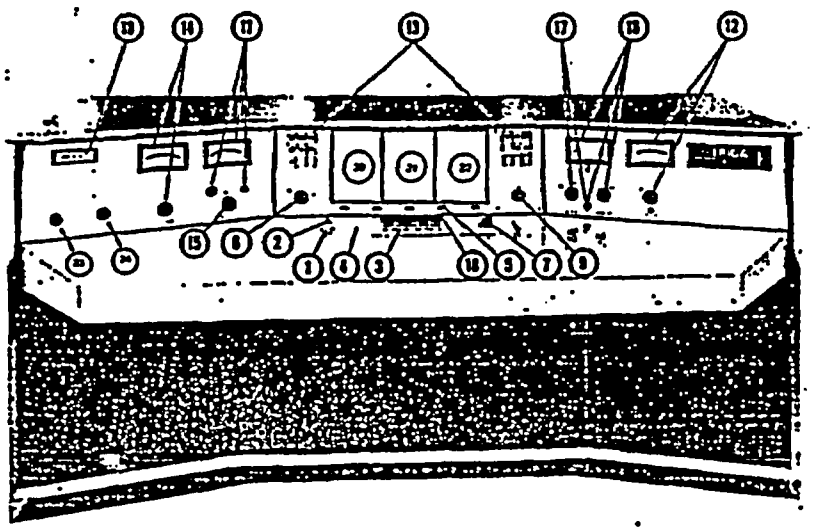


Figure 7.2, USGS TRIGA Console with Modifications.

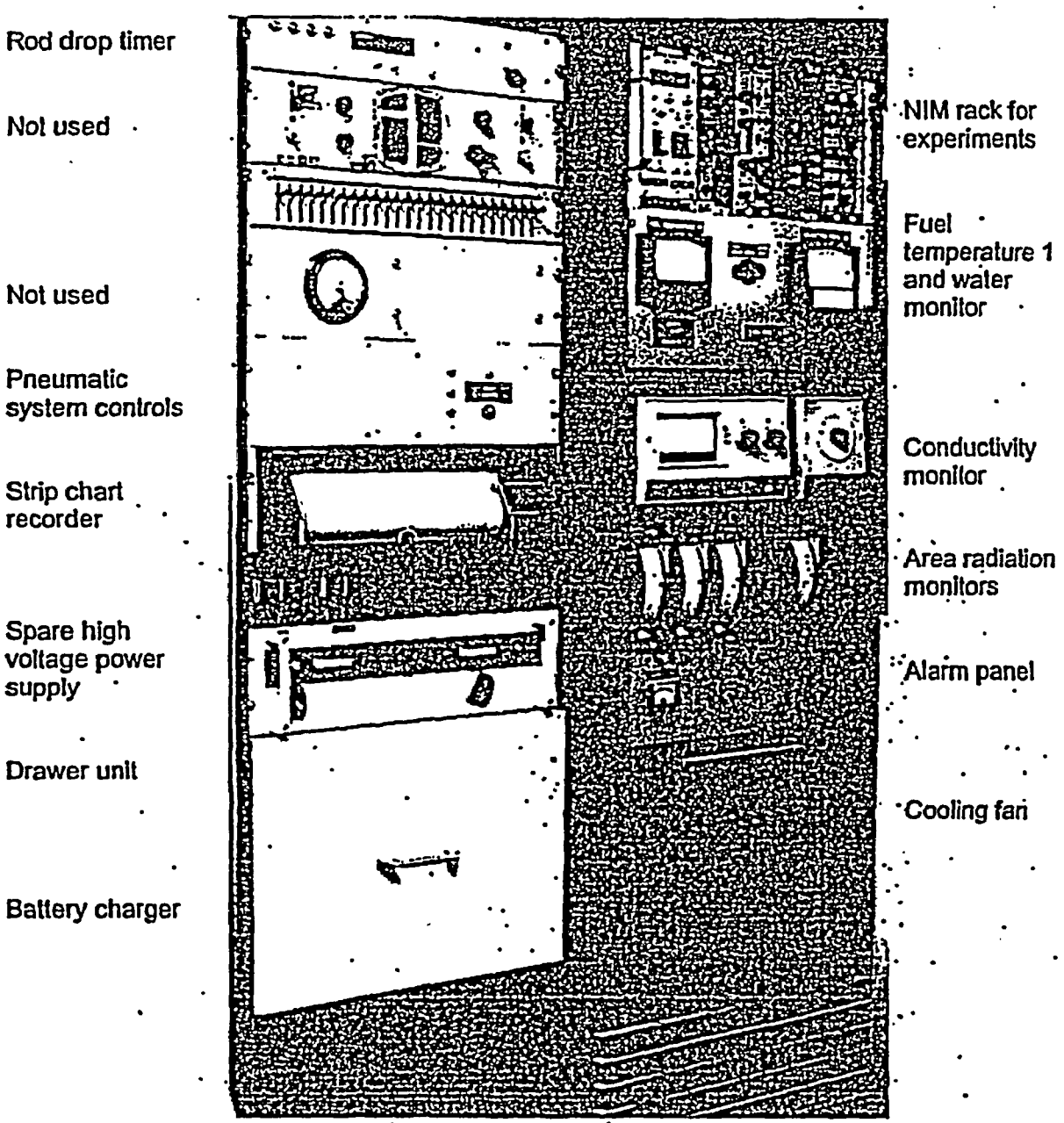


Figure 7.3, Instrumentation Rack.

Table 7.1, Description of Figure 7.2.

Number	Function	Description
1	Console Power	Push Button Switch
2	Magnet Power / Scram Reset	Key Switch
3	Control Rod Drive Position	Push Button Switches
4	Apply Air to Pulse Rod	Push Button Switch
5	Rod Position Indicators	LED Displays
6	Mode Selector	Rotary Switch
7	Removed	Backup Range Switch
9	Automatic Power Demand Control	10 Turn Potentiometer
10	Manual Scram Bar	Bar Covering Scram Switches
11	(not used)	Backup Period Channel
12	Fuel #2 & Water Temperature	Display and Selector Switch
13	Scram Status, Source Interlock, Low Air Pressure, Hi & Hi-Hi sump level, surge tank level & makeup, Upper & Lower Doors, and Cooling System Power	Indicators and Control Switches
14	(not used)	Backup Count Rate Channel
15	(not used)	Backup Log Channel
16-17	(not used)	Backup Percent Power Channel
18	(not used)	N/A
20	Wide Range Log Power Channel	GA NLW-1000 Channel
21	Multi-Range Linear Power Channel	GA NMP-1000 Channel
22	Percent Power and Pulsing Channel	GA NPP-1000 Channel
23	Source Interlock Override	Key Switch
24	Period Scram Override	Key Switch

7.3 Reactor Control System

The bulk of the reactor control system (RCS) is housed in the USGS console shown in Figure 7.2. The remainder is contained in the auxiliary rack-mount panel next to the console, shown in Figure 7.3. The RCS consists of the instrumentation channels, the control rod drive circuitry and interlocks, and an automatic flux controller. These are shown in Figure 7.4. The RCS measures several key reactor parameters including power, fuel temperature, water temperature, and water conductivity.

7.3.1 Neutronic Instruments (Reactor Power)

Three neutronic instruments measure reactor power separately: a wide-range logarithmic channel, a multi-range linear channel, and a percent power channel, as shown in Figure 7.5. Wiring diagrams and calibration procedures are found in the instrument maintenance manuals listed in the bibliography.

The wide-range log channel uses a fission counter for detecting thermal neutrons in the range of 1.4 to 1.4 x 10^5 nv, and provides approximately 0.7 counts·nv^{-1}. The detector has an aluminum case, an aluminum electrode, a U_3O_8 (>90% enriched in ^{235}U) coating as the neutron sensitive material, and an argon-nitrogen mixture for a fill gas. A preamplifier is used to minimize noise and signal loss from the detector to the console, and it is located on the 12-foot level.

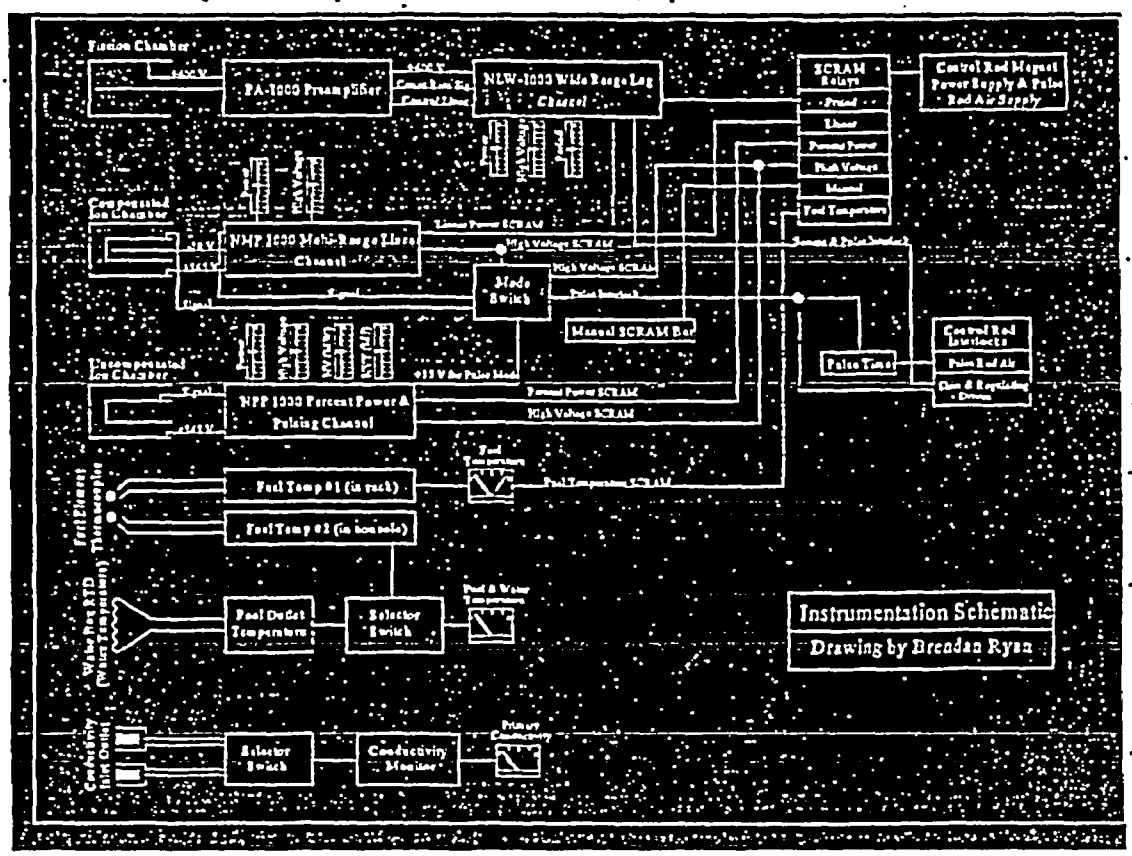


Figure 7.4, Instrumentation Diagram.

The remainder of the channel circuitry is located in the NLW-1000 unit in the central console. The NLW-1000 unit supplies the high voltage for the detector and power for the preamplifier. The instrument switches from pulse mode operation to current mode as reactor power increases out of the source range, allowing the instrument to measure reactor power in the upper ranges. Three displays indicate reactor power, high voltage, and reactor period. The power signal is permanently recorded via an opto-isolated output to a strip-chart recorder located in the instrumentation rack. The period meter has a scram at 3 sec and there is a high voltage scram, both of which are bypassed in pulse mode. This channel also provides a protective interlock which prevents rod withdrawal when indicated neutron flux is < 2 cps, which is also activated in pulse mode to prevent removal of the shim, safety and regulating rods. Another interlock

prevents pulsing when reactor power is above 10 kW (normally set at 1 kW). The unit has two calibration checks in pulse mode, two in current mode, and checks for the period and high voltage scrams.

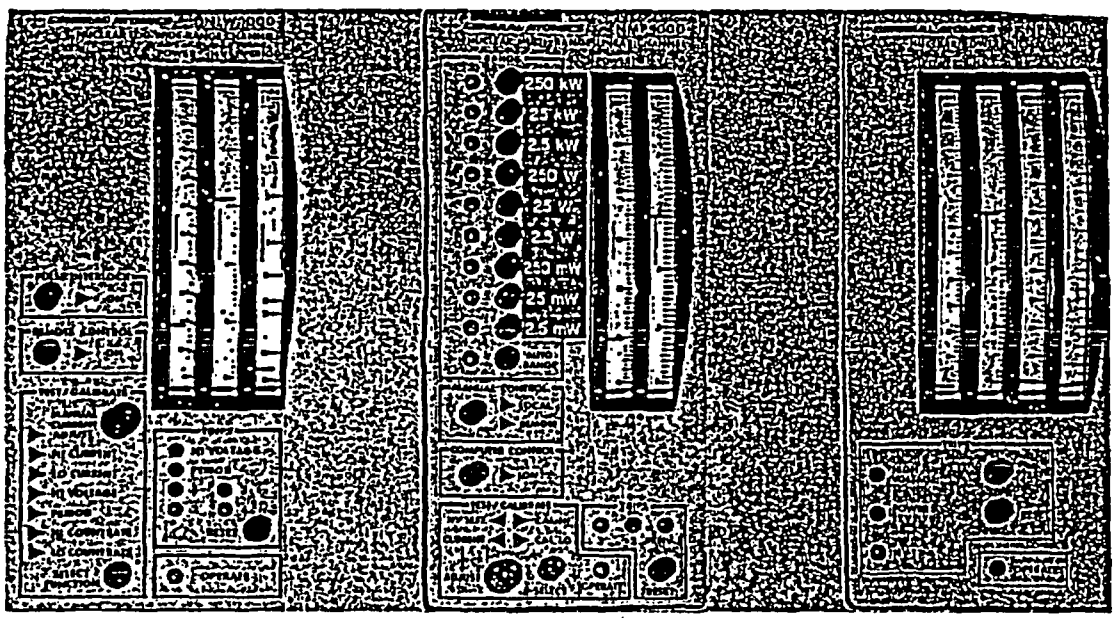


Figure 7.5, N-1000 Series Instruments.

The second channel provides multi-range linear power indication. This channel uses a compensated ion chamber for detection of thermal neutrons. The linear channel detector signal goes directly to the NMP-1000 unit in the center console, which in turn supplies high and compensation voltages. This unit features automatic or manual ranging to select the appropriate decade of power displayed. The instrument provides two indicators, power and high voltage. The power signal is permanently recorded via an opto-isolated output to a strip-chart recorder located in the instrumentation rack. In addition there is also a high power level scram (normally set for 104% nominal rated power) and a high voltage scram. The signal from the detector and the high voltage scram are bypassed in the pulsing mode. The unit has two calibration checks, an auto-ranging test feature, and checks for high power level and high voltage scrams.

Power range indication of neutron flux is provided by an uncompensated ion chamber signal, which indicates percentage of power in the upper two decades of the power range. The uncompensated ion chamber is virtually identical in construction to the compensated ion chamber, but no gamma compensation is provided in the circuitry. The detector sends its signal to the NPP-1000 instrument in the center console, which provides a visual indication of reactor power, high voltage, nv, and nvt measurements. The NPP-1000 supplies the high voltage for the detector. There is a high power scram (normally set for 104% of full power) and a high voltage scram. In pulse mode, the channel is designed to read off the maximum power and integral output of a reactor pulse. However, the pulse output readings are measured in reference to the 250 MW maximum. Hence an additional channel is added to the central thimble to permit recovery of data from pulses of various magnitudes. The unit has checks for high power and high

voltage scrams.

An added pulsing channel consists of a small BF_3 chamber, which can be inserted into the central thimble of the reactor core. A separate high voltage supply powers the instrument and a multi-range picoammeter reads the detector current. A reference voltage output of the picoammeter is sent to a computer in the control room, which collects the pulse data. This channel is calibrated prior to pulsing operations and range selected in advance based upon the anticipated peak power.

7.3.2 Temperature

Temperature indications for the primary water and specific B-Ring fuel elements are provided on the front section of the control panel and in the instrumentation rack. The instrumented fuel elements have three chromel-alumel thermocouples in the fuel element that are used for temperature indication on the console or in the instrumentation rack. The thermocouples are located 0.76 cm (0.3-in) below the fuel surface, spaced at the midpoint of the element and at ±, 2.5 (1 in.) cm from the midpoint; an averaged value from all three thermocouples is typically used for instrument readings. The temperature in the primary cleanup loop is a nickel alloy thermistor, and is displayed on a console meter, which is shared with fuel temperature via a rotary switch. Another indication of fuel temperature is located in the instrumentation rack with the capability of initiating reactor scram if the measured fuel temperature exceeds a preset value (normally 400 °C).

Several other temperature measurements can be obtained from the computer in the control room. The computer can read two additional fuel thermocouples from other fuel elements in various positions in the reactor core. Additionally AD590 temperature transducers are located on the inlet and exit of both the primary and secondary sides of the heat exchanger to evaluate performance. Two other transducers are located in the reactor tank for bulk pool temperature measurement and high temperature alarm.

7.3.3 Water Conductivity

Primary water conductivity is measured at the inlet and outlet of the purification loop by titanium electrode cells that send signals to a bridge circuit in the instrumentation rack. The bridge circuit is automatically temperature compensated and nulled to provide good conductivity measurements over all reactor conditions. The inlet and outlet conductivities provide a good indication of the overall purity of the primary water and the effectiveness of the ion exchanger.

7.3.4 Control Rod Drives

Four control rods are required for reactor operations at 1,250 kW to meet reactivity control requirements: a shim rod, a regulating rod, a transient rod, and a safety rod. The shim, regulating and safety rods share identical control circuitry (Figure 7.7) and provide coarse and fine power control. Two of the rod drives are original, analog systems. One of the rod drives uses a stepper motor. Drive position is determined by voltage drop across a potentiometer that is adjusted as the control rod drive is moved. The position indicator for the analog motors is attached to a shaft

coupled to the drive motor shaft with a setscrew, while the stepper motor is connected to the position indicator with a chain drive. The pulse rod is designed so that it can be rapidly ejected from the core to a preset height to initiate a reactor pulse. However, it still functions as a normal control rod in steady state mode. All rods can be individually scrammed without shutting down the reactor.

a. Standard Control Rod Drives

The rod drive mechanism (see Figure 7.6) is an electric motor actuated linear drive, equipped with a magnetic coupler. Its purpose is to adjust the reactor control rod position. In the analog drive motor, a 110-V, 60-cps, two phase motor drives a pinion gear and a 10-turn potentiometer. The potentiometer provides rod position indication. The pinion engages a rack attached to the magnet drawtube. An electromagnet mounted on the lower end of the drawtube engages an iron armature that screws into the end of a long connecting rod which terminates (at its lower end) in the control rod.

The magnet, armature, and upper portion of the connecting rod are housed in a tubular barrel that extends well below the reactor water line. Located part way down the connecting rod is a piston equipped with a stainless steel piston ring. Rotation of the motor shaft rotates the pinion, thus raising or lowering the magnet draw tube. If the magnet is energized, the armature and connecting rod will follow the draw tube so that the control rod is withdrawn from or inserted into the reactor core. In the event of a reactor scram, the magnet will be de-energized and release the armature. The connecting rod, piston, and control rod will then drop, thus reinserting the control rod into the reactor. Since the upper portion of the barrel is well ventilated, the piston will move freely through this range. However, when the connecting rod is within 2-in (5 cm). of the bottom of its travel, the piston is restrained by the dashpot action of the restricted ports in the lower end of the barrel. This restraint cushions bottoming impact. Control rod drop times are measured semi-annually and must be less than one second.

The analog rod drive motor is dynamically braked and held by an electrically locked motor. In the static condition, both windings are energized with the same phase (see Figure 7.7), electrically locking the motor. Clockwise (up) or counter-clockwise (down) rotation is enabled by shifting the phase between the windings with a 1-μF capacitor; motor control switches allow the appropriate phase shift. The stepper motor operates using phase switched direct current power. The motor shaft advances 200 steps per revolution (1.8 degrees per step). Since current is maintained on the motor winding when the motor is not being stepped, high holding torque is maintained. A translator module drives the stepping motor.

Three microswitches limit and control the travel of the magnet drawtube. Actuation of the magnet up limit microswitch (S901) applies line voltage to one winding therefore allowing only the phase shift, which gives counter-clockwise rotation.

Actuation of the magnet down limit microswitch (S902) applies line voltage to the other winding therefore allowing only the phase shift that gives clockwise rotation. Actuation of the rod down microswitch (S903A) causes the phase shift for counter-clockwise

rotation. Therefore, if the control rod drops, the magnet drawtube drives down until the magnet down limit microswitch locks the rotor. Since the rod down microswitch drives the magnet draw tube down, then the rod down microswitch must be open before the magnet down microswitch during coupled withdrawal of the control rod.

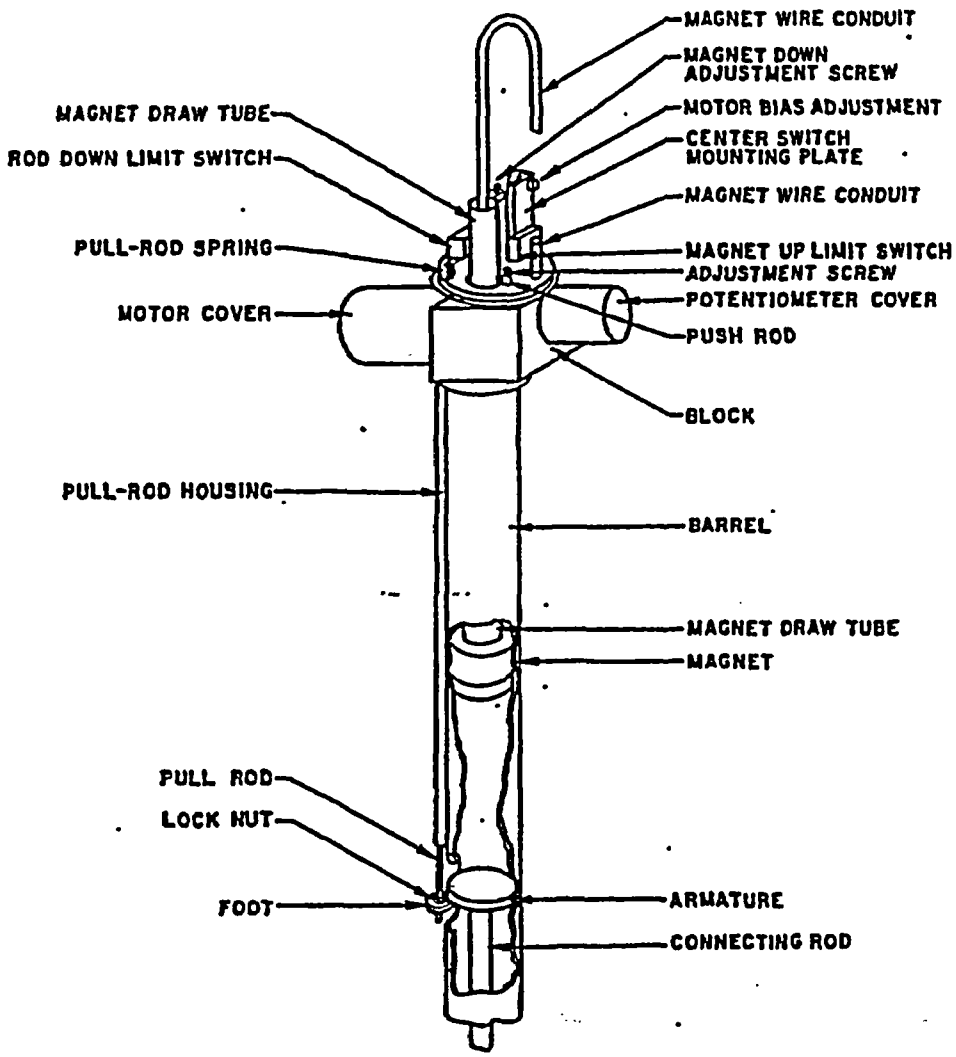
Three lights indicate that, 1) the magnet drawtube is full up, 2) the magnet drawtube is full down, and 3) the armature and magnet are coupled. When the magnet drawtube is full up, microswitch (S901) is actuated opening the short across the magnet up light (DS321). When the magnet draw tube is full down, microswitch (S902) is actuated opening the short across the magnet down light (DS324). When the control rod drops, the non-actuated magnet down microswitch (S902) and the actuated rod down microswitch (S903B) short the contact light (DS317) indicating separation of the magnet and the armature.

Other features of the circuit are an adjustable bias resistor (R902), a 220-ohm surge resistor, 50-ohm current limiting resistors. The adjustable bias resistor compensates for the torque applied by the weight of the control rod and the magnet drawtube. The 220-ohm surge resistor limits the capacitor current surge during switching. The 50-ohm current limiting resistors limit the currents in the 12-volt indicating circuits when the indicating lamps are shorted.

The unconventional circuit employed in the rod-drive system minimizes the number of switch contacts required. Therefore, relays with their attendant reliability problems are not required. It should be noted that the rod drive units are identical both mechanically and electrically (with the exception that one unit uses a stepper motor) and they are, therefore, interchangeable.

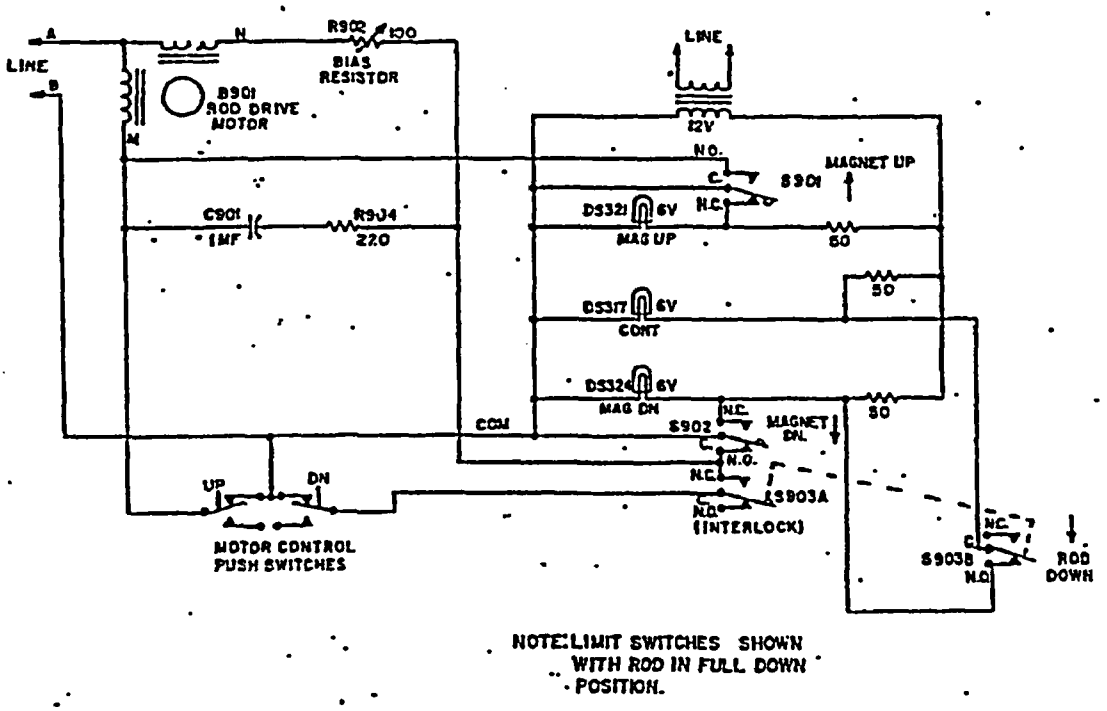
The rod position indicators are three digit, LED display indicators that receive a variable DC voltage input from 10-turn potentiometers that are driven by the respective rod drive motors. The digital display is simply a voltmeter, since the voltage across the potentiometer is directly related to the control rod position. The position indicators have their own variable power supplies and are therefore completely independent. The indicator systems are located in the control console except for the 10-turn potentiometers on the drive and the associated wiring.

Normal rod motion speed is about 12-in. per minute. Using rod speed, rod position indication at UP and DOWN limit switch positions, and respective rod worth curves, the operator can determine the reactivity insertion rate for a given interval of rod motion.



Standard rod drive mechanism

Figure 7.6, Regulating, Safety and Shim Rod Drive.



Rod drive, motor control, and indicator lamp circuit

Figure 7.7, Control Rod Drive Circuit.

b. Transient Rod Drive

The transient rod-drive (Figure 7.8) is an electrically controlled, pneumatically operated, mechanically limited system. The transient rod and aluminum extension rod are mechanically connected to a pneumatically driven piston inside a worm gear and ball-screw assembly. The system is housed on a steel support structure mounted above the reactor tank. A three-way solenoid valve mounted below the support controls air to the piston. The throw of the piston, and hence the amount of reactivity inserted into the core during pulsing operations, is regulated by adjusting the worm gear and ball-screw assembly. The adjustment is made from the central console by actuating a reversible motor drive, which is coupled to a worm gear and a 10-turn potentiometer for position indication. The operation of the position indicator is identical to that of the shim, safety and regulating control rods. The drive circuitry is identical to the shim, safety and regulating rods (Figure 7.7), except for that the motor is not continuous energized. Relays in the drive unit allow it act similarly to the electrically locked motors. Since air is used to support the rod, there is no compensation for rod weight. The remaining differences involve the pneumatic relay controls.

The solenoid valve is actuated by means of the console mounted transient rod fire and air release (scram) switches. When the rod fire switch is depressed, the solenoid valve opens, admitting air to the cylinder, coupling the piston and rod to the shock absorber. Depressing the air release switch de-energizes the solenoid valve, which removes air from the cylinder and vents air to the atmosphere. In the event of a reactor scram, the solenoid will be de-energized via the scram circuitry, which will allow the transient rod to drop into the core after the air is removed. Micro-switches are used to indicate the extreme positions, up or down, of the shock absorber. In steady state mode, an interlock prevents actuation of the rod fire switch if the drive is not in its fully down position.

In the pulse mode, a variable timer (usually six seconds) de-energizes the solenoid valve after the pulse is initiated. The shock absorber will remain in its preset position until the mode selector switch is taken to steady state. In the steady state mode of operation, the adjustable (normally six second) timer is disengaged and the cylinder remains pressurized. If the air supply for the pulse rod drive should drop below approximately 45 psig, an amber low air pressure warning light will be actuated on the control console. Loss of air pressure will cause the rod to fall into the core.

c. **Interlocks**

Several interlocks are built into the control system of the reactor to prevent improper operation. These interlocks are hard-wired into the control rod drive circuitry. They are stated below:

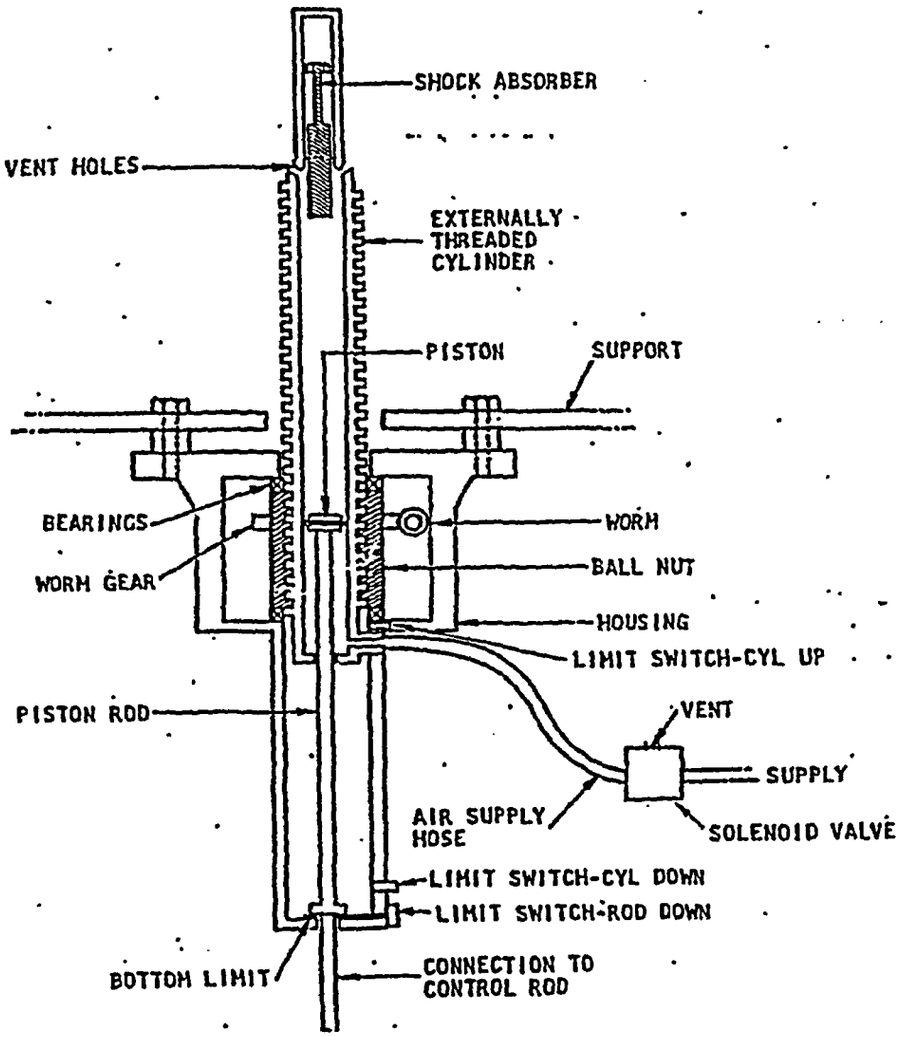
1. No control rod withdrawal (shim, regulating and safety rods only) is possible unless the count rate neutron channel is indicating > 2 cps. This interlock prevents the possibility of a startup without a functional power level startup channel.

   The low count rate interlock may be bypassed during fuel loading operations when core inventory is not high enough to multiply the source above 2 cps.

2. 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.

3. There is no simultaneous withdrawal of two or more control rods when the reactor is in the steady state mode. This interlock prevents violation of the maximum reactivity insertion rate of the reactor.

4. The pulse rod is the only control rod that can be withdrawn when the reactor is in the PULSE mode (this does not prevent the scramming of any control rod). This interlock minimizes the possibility of pulsing a supercritical reactor. This interlock is provided by the source interlock, which is engaged when the log channel is placed in pulse mode.



Transient rod drive mechanism

Figure 7.8, Transient Rod Drive.

Additionally, there is an interlock that prevents reactor pulses from being fired if the reactor power is above 10 kW (normally set at 1 kW). There is also a key switch for bypassing the source interlock during fuel loading operations to check for criticality.

7.4 Reactor Protection System

The reactor protection system (RPS) will initiate a reactor scram if any of several measured parameters are outside their limited safety system settings (LSSS). The reactor scram effectively shuts down the reactor by de-energizing portions of the shim, regulating, transient, and safety rod drives, causing the control rods to drop into the reactor core by gravity. The shim, safety and regulating rod drives utilize electromagnets to hold up their respective control rods. The pulse rod drive utilizes an electric solenoid to apply compressed air to the pulse rod. High reactor power, high fuel temperature, loss of detector high voltage, or short reactor period will automatically cause the control rods to be dropped into the reactor core. The reactor operator may manually scram the reactor as well by means of a scram bar on the console. If beam ports are in use in their open configuration, an additional reactor scram may be added to reduce radiation levels if personnel attempt to enter the beam area.

During steady state operation, the high reactor power scram, the high voltage power supply failure scrams for all neutronic channels, and the manual scram are required for operation. Although the period scram is normally in operation, it can be bypassed provided that the reactor operator calculate reactor period for each rod movement and that the calculated reactor period is greater than one second. The fuel temperature scram is active, but it is not necessary since the fuel temperature setpoint (usually set for 400°C) will normally not be reached during steady state mode.

In pulse mode, the mode selector switch is set to the HI PULSE position, interrupting detector signal to the linear channel. When the pulse interlock is activated (to initiate the source interlock) to prevent withdrawal of the shim, safety and regulating rods, the detector signal to the logarithmic wide range detector is interrupted. Consequently, the period scram and the linear high power scrams are disabled. High voltage scrams for the linear and log channels are also disconnected as large pulses produce detector currents that may temporarily overload the power supplies. The percent power channel increases its range to the pulse range (the percent power channel high voltage power supply scram remains active). The uncompensated ion chamber for the percent power channel is positioned further away from the reactor core, allowing for measurements of both steady state and pulsing power levels without excessive current from the high voltage supply.

7.5 Engineered Safety Features Actuation Systems

There are no engineered safety features actuation systems. Control rod insertion is provided by gravity and core cooling is provided by natural convection in water or air. Therefore, ESF systems are not required in this design.

7.6 Control Console and Display Instruments

The control console and display instruments are shown in Figures 7.2-7.5, 7.9, 7.10 and 7.12. Their layout (Figure 7.11) provides a single operator with all relevant reactor information. All push buttons required for general operation are located on the control console, within easy reach by a seated operator. All instruments on both the console and instrumentation rack are visible

from this seated position. In addition to the various analog displays, a computer display on top of console can also be used to show relevant reactor status information on a single screen.

There are several additional pieces of equipment in the control room. [redacted] to interrupt power to electrical devices in the control room and reactor bay (see Figure 7.13a). A halon fire extinguisher is located next to the breakers for use in fighting electrical fires. Current core and facility configuration is shown in a display cabinet (Figure 7.13.b). [redacted] A local radiation area monitor (including indicator and alarm) is located above the door in the control room to the reactor bay.

7.7 Radiation Monitoring Systems

Radiation monitoring systems are employed throughout the reactor facility: G-M detectors at the reactor pool surface and cleanup loop, 7 remote area monitor channels (3 general area or process monitors, 4 channels for beam ports – 11 beam port channels is currently instrumented, with the remainder scheduled for instrumentation near term), a 5 $R \cdot h^{-1}$ evacuation alarm, several air activity monitors, and numerous portable radiation monitors, including those for contamination monitoring. Additionally, an independent monitor with visual and audible alarms is located above the door to the reactor bay.

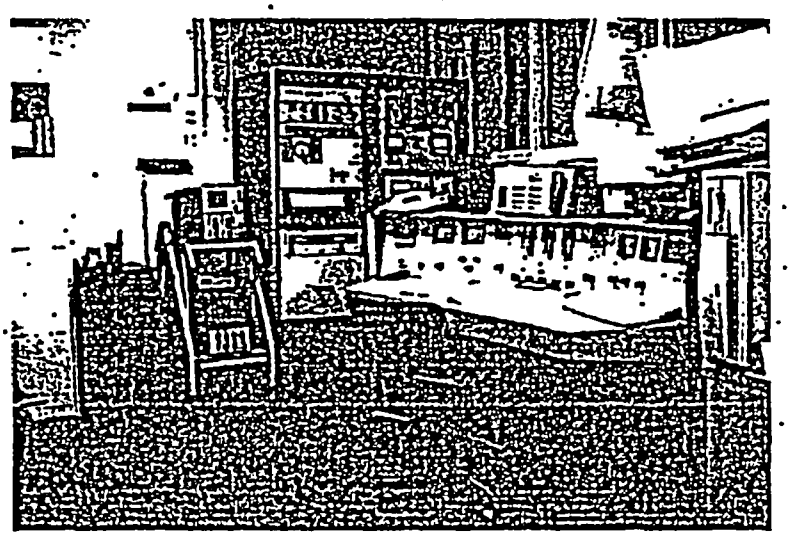


Figure 7.9, Control Room Overall View.

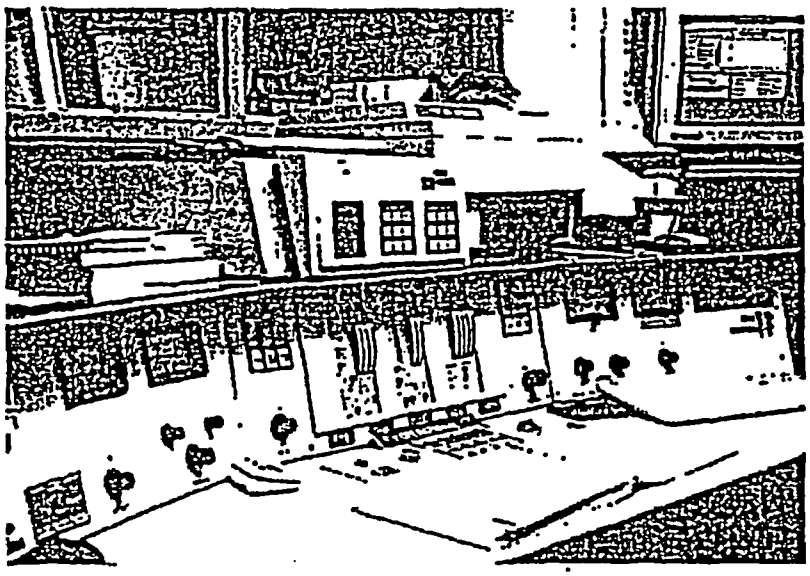


Figure 7.10, Control Console Operator's View.

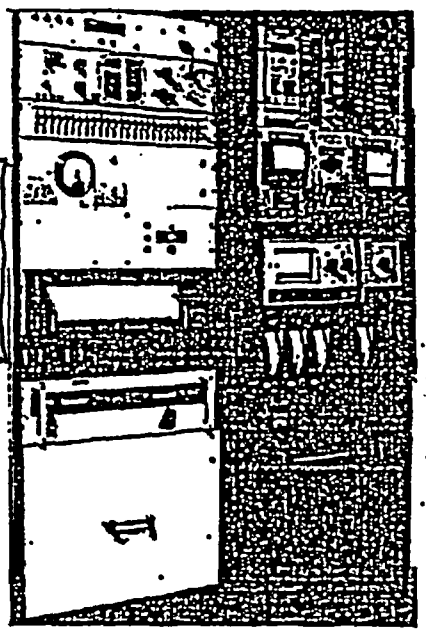


Figure 7.12, Instrumentation Rack Operator's View.

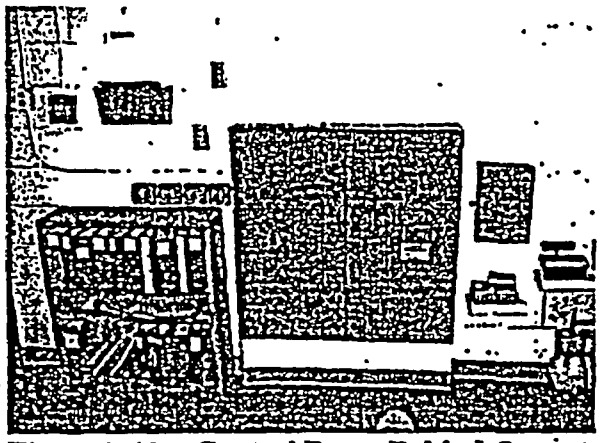


Figure 7.13a, Control Room Behind Operator.

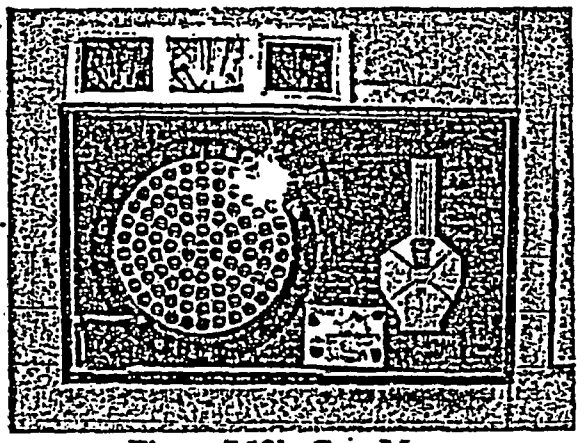


Figure 7.13b, Core Map.

The radiation monitor in the instrumentation rack provides an indication of radiation levels directly above the primary pool water surface. A meter on the right-hand section of the rack indicates a radiation dose rate of 100 $mR \cdot h^{-1}$ full scale, utilizing a GM tube detector measures coolant activity in the cleanup loop water box. The water box monitor has an audible alarm, with a reset button.

The remote area monitors utilize G-M detectors located throughout the reactor bay (typical unit illustrated in Figure 7.14). Permanent locations are: at the top of the reactor tank, above the bulk shield tank, and near the ion exchanger in the primary coolant system, and directly over the primary water tank. Each beam port has signal and power lines to support installing a beam port monitor.. The detectors feature an analog readout in both the control room and locally with visual indicators for normal, alert, and alarm conditions. The control room alarm has an audible signal as well.

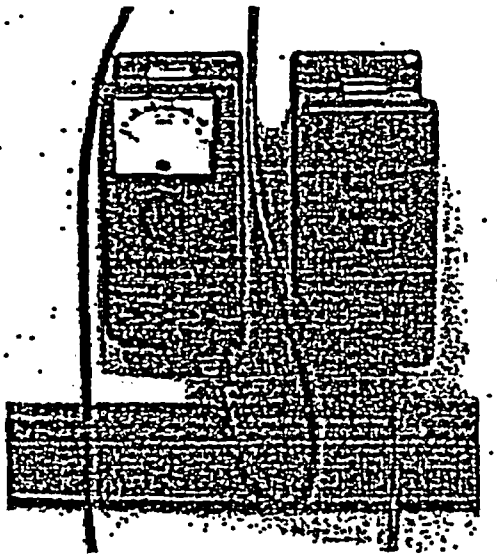


Figure 7.14, Typical Area Monitor Instillation.

A 5 $R \cdot h^{-1}$ monitor on the 22-foot level serves an evacuation alarm. The alarm signals a 100+ dB audible alarm in the reactor bay.

There are numerous ion chamber and G-M portable survey instruments through the reactor bay and control room for gamma and beta surveys. These instruments are calibrated semi-annually. A rem-ball and a Bonner Sphere set are located in the reactor bay for neutron measurements. The rem-ball is calibrated annually. Low-level G-M counters in the reactor bay used for contamination monitoring. These are calibrated annually. Neutron and gamma sensitive pocket ion chambers are available in the control room for tours and personnel monitoring, and are calibrated semi-annually. Film and ring badges are issued to regular staff, which are read monthly and quarterly respectively.

An air monitoring system (Figure 7.15) samples air over the reactor pool. These monitors should sense any changes in radioactive discharge from the reactor pool to the environment. Channels are provided for monitoring particulate, noble gas and iodine activity.

An independent air monitoring system, a continuous air monitor (Figure 7.16), is stationed on the 12-foot level. This monitor should sense any changes in airborne contamination in the reactor bay. The continuous air monitor has active background discrimination. One channel monitors radioactive contamination on a filter exposed to airflow from the reactor bay. A second channel monitors background radiation levels for a background subtraction

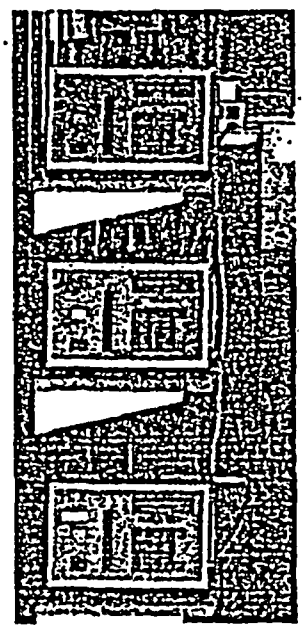


Figure 7.15: Air Monitoring System

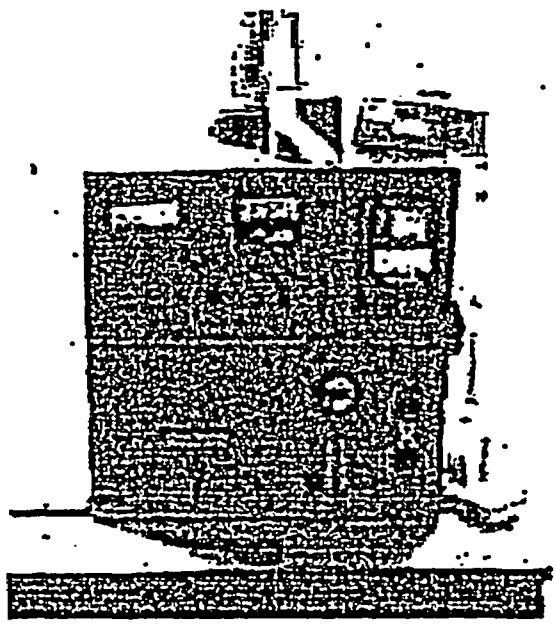


Figure 7.16: Continuous Air Monitor

7.8 Bibliography

250-kW TRIGA Mark II Pulsing Reactor Mechanical Maintenance and Operation Manual for KSU, General Atomics, GA-3399; 1 August 1962.

NMP-1000 Linear Power Channel: Operation and Maintenance Manual General Atomics, , E-117-1017, Rev. 1; 1991.

NLW-1000 Log Power Channel: Operation and Maintenance Manual, General Atomics, E117-1019; 1992.

NPP-1000 Percent Power Channel: Operation and Maintenance Manual, General Atomics, , E117-1010, Rev. 2; 1992.

TRIGA Mark I Reactor Instrumentation Manual for USGS, General Atomics, GA-9039; 1 November 1968. (For console acquired from USGS)

TRIGA Mark II Pulsing Reactor Instrumentation Maintenance Manual for KSU, General Atomics, GA-3114, no date.

8. ELECTRICAL POWER SYSTEMS

Primary electrical power is provided through the Kansas State University power grid, supplied by an on-campus plant and commercial generators. Main power lines traverse underground tunnels, thus inhibiting tampering. Loss of electrical power automatically places the reactor in a subcritical, secured configuration. Loss of electrical power will de-energize the control rod drives, causing the rods to fall by gravity into the core, and therefore does not represent a potential hazard to the reactor. Since the core is cooled by natural convection, no emergency power is required for reactor cooling systems. Backup battery systems are provided for emergency lighting, the security system, and the 22-foot level evacuation alarm.

8.1 Normal Electrical Power Systems

The design basis for the normal electrical power systems is to provide sufficient current for normal operations. The original instrumentation and control systems were mostly vacuum-tube designs; hence there is more than adequate supply for the newer solid-state devices. Safe-shutdown is an automatic consequence of loss of electrical power and hence there are no major electrical backup systems. The reactor has no exclusive electrical supply and distribution, but derives from the building transformers. Supplied power is standard 60 Hz AC, available in 110 V single phase, 220 V single phase, and 480 V three phase configurations.

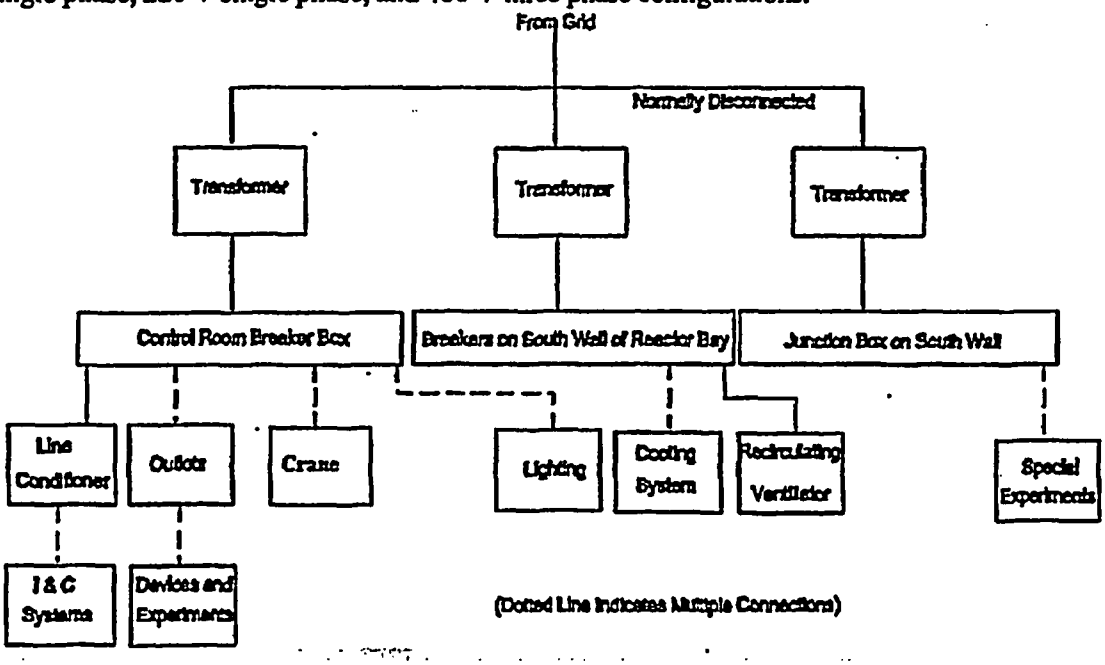


Figure 8.1, Electrical Distribution Layout.

Utility voltage (4160 kV) is delivered via underground cables to three oil-filled switches inside a locked, fenced area immediately outside the reactor bay, but within the facility boundary (Fig

8.1). These switches connect power to step-down transformers for the original Ward hall (including control room power), reactor bay service loads, experiment services, the 1972 addition to Ward Hall, and Cardwell Hall (math and physics building, north of Ward Hall).

One transformer supplies building power and the control room breaker box. A second primary transformer supplies the breakers in the reactor bay for the cooling system and recirculation ventilator. A third dedicated service located in the reactor bay can be used to supply reactor experiments requiring large amounts of electrical power, which is normally disconnected at the transformer.

The control room breaker box supplies all electrical outlets throughout the facility, lighting, a line conditioner, and the reactor bay crane. From this source, a line conditioner in the reactor bay provides isolated, regulated power for reactor instruments and control systems, such that they are not affected by minor electrical fluctuations.. An interruption of electrical power will cause the line conditioner to de-energize and it must be manually reset to resume operations. All standard reactor experiments utilizing the outlets in the reactor bay can be de-energized from the control room breakers.

Although the cooling system control breakers are supplied from a separate line, a relay network provides control from a switch located on the control console. Another breaker supplies a recirculation ventilator inside the reactor bay, which cannot be de-energized from the control room. A large electrical distribution center is located near the cooling system breakers in the reactor bay that can be used to supply reactor experiments, although this system is normally disconnected at the transformer.

The bulk of electrical wiring is in shielded conduits as per commercial electrical codes. Instrument wires from the reactor instrumentation and control system run through a sub floor conduit in the control room into a wire tray leading to the upper level of the reactor bay. A secondary tray runs the perimeter of the 12-foot level of the reactor. Most instruments use self-shielded cables. The signal cable for the log power channel, being sensitive to electrical noise, runs inside a flexible metal conduit in the wire tray from the upper level of the reactor to the control console. Grounding straps are used to ensure common ground between the control room and instrument locations.

All electrical devices were installed according to the electrical codes in existence at the time of their introduction. During the control room modifications in 1993, the control room wiring was inspected and several systems were re-wired to meet current electrical codes.

8.2 Emergency Electrical Power Systems

The design basis for emergency electrical power systems is to provide lighting and surveillance for emergency conditions and to maintain physical security. Consequently, battery backup power is used for emergency lighting, the University fire alarm system, the evacuation alarm, and the security system. All backup systems have regular maintenance schedules and are periodically tested for operation.

Internal batteries supply emergency lighting at the upper level of the reactor and at all exits to the reactor bay. Exit signs are similarly illuminated. The fire alarm system has battery backup for

sensors and pull-stations. These systems are maintained and periodically tested by KSU Fire Safety personnel.

Backup batteries supply the security system, but description of the system is limited to our Physical Security Plan as per 10CFR2.790.d. Maintenance and testing of the batteries is performed annually.

8.3 Bibliography

Ward Hall Blueprints. Available from the Department of Facilities, Kansas State University.

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9. AUXILIARY SYSTEMS

The systems covered 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 is provided by either steam or chilled water from the University Physical Plant. Because of this design, heating and cooling cannot be simultaneously provided. Consequently, the system gets changed over from steam to chilled water and vice versa twice a year.

The reactor bay was specifically designed for handling radioactive materials. The reactor bay was originally equipped with four unit ventilators on the 12 foot level which drew intake from outside air and discharged into the reactor bay. These units have been disabled, removed and blocked. In-leakage and a single exhaust fan at the top and center of the confinement dome provides outside air to the reactor bay. Prior to disabling the unit ventilators, a large unit ventilator was installed to support an experiment in the reactor bay with adjustable ducting between the 12-foot level and the top of the reactor bay. The unit has a coil connected to the HVAC lines and is now used for heating and cooling in the reactor bay. A thermostat was installed to permit temperature adjustments. The electrical disconnect for the unit is located near the cooling system breakers, behind the heat exchanger. The condensate line drains into the reactor sump. Routine maintenance and service of these systems is the responsibility of the KSU Department of Facilities. All ventilator filters in the reactor bay are surveyed for radionuclide deposition upon removal.

The control room features a single thermostat that can vary the temperature of the air coming from the main intake unit located in room 108, adjacent to the control room. The main intake unit supplies the bulk temperature change and secondary coils inside the control room duct allow zone control. The control room also features an exhaust fan, which ducts out of the roof immediately above the control room. A switch on the control room wall manually controls the exhaust fan. A single window to the outside may be opened from the control room; the window is covered with a metal barrier for security.

9.2 Handling and Storage of Reactor Fuel

The majority of the reactor fuel is stored in the reactor tank, either in the reactor core or racks surrounding the reactor tank. Racks may be added to the bulk shield tank for fuel storage during fuel transfer operations. The racks are fabricated of aluminum and allow only for single row spacing of up to six elements, whose spacing in the rack is sufficiently far apart to prevent accidental criticalities.

Fuel may also be stored in 10 fuel storage pits which extend 3 m below the floor of the 0 foot level of the reactor bay. The fuel storage pits are constructed of 25.4 cm diameter Schedule 40 steel pipe, welded closed at the base with lockable covers. Due to water seepage into one of the

pits, the pits are inspected annually. These pits are currently used to store elements that were damaged in handling or shipping. There are also two 25.4 cm beamport plug holders built into the walls of the reactor bay, one of which is used to store new fuel elements.

Fuel is individually handled by means of a standard TRIGA element tool that grapples the end fitting on the element. The tool is 7.6 m long and requires two persons when moving fuel. One person is required to be a senior reactor operator, the other trained in fuel handling operations. Additionally, a reactor operator is also required to be at the reactor console to monitor conditions in the reactor and reactor bay.

To facilitate transfers of activated fuel elements, a single fuel element cask is used. The reactor bay crane is used to move the cask. The fuel element is drawn in through the base of the cask and a shutter is slid under the element. The cask is steel over lead. The cask is 35.6 cm in diameter and 1 m in height, has a 5.1 cm internal cavity and weighs 1134 kg (2500 lbs).

For fuel inspection, there is a periscope mounted in the reactor tank for visual inspection of cladding. The periscope is counterweighted, and can be lowered all the way to the reactor core level. It offers a 24° view in water, with a 48° apparent field of view. For fuel element gauging, a tool is permanently mounted to the side of the tank. A GO-NO-GO tube is used to check for swelling or bowing within 1.57 mm of manufactured tolerances. The technical specification limitation is 1.59 mm. A dial indicator is used to measure element lengths to the nearest 0.03 mm against reference standards. Fuel elements elongation is limited by technical specification to 2.54 mm.

With a very small burn-up of fissile material per year, the KSU reactor has infrequent shipments of fuel elements off-site. Commercial NRC-certified transport casks are used for off-site transport of used fuel, principally the Battelle BMI-1 cask (to be replaced by the GE-2000 or other suitable cask). New fuel elements are delivered by General Atomics in their own licensed container.

9.3 Fire Protection Systems and Programs

Fire protection systems are maintained and serviced by the Campus Fire Safety, Department of Environmental Health and Safety, Division of Public Safety. The building fire alarm system was installed in 1995 and is part of a campus-wide 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. The reactor has two pull-stations one in the control room and one in the reactor bay. There are two smoke detectors located about at the 20-foot level inside the reactor bay on the north and south walls. The reactor also has an additional system, consisting of seven smoke detectors in the reactor bay and one in the control room. This system is associated with the reactor Physical Security system and will also notify the campus police.

There are eight fire extinguishers readily available to reactor personnel. A halon extinguisher is located in the control room for electrical fires. For general purpose fires, carbon dioxide fire extinguishers are located in the hallway outside the control room, on the 22 foot level of the reactor bay, on the 12 foot level, and on the 0 foot level. For equipment fires, dry chemical extinguishers are located on the 12 foot level, on the 0 foot stairs, and next to the cooling system

pumps. Reactor personnel visually inspect all extinguishers monthly. Campus Fire Safety performs pressure testing and general maintenance on an annual basis.

9.4 Communication Systems

The reactor facility has an intercom system controlled by a 3M D-120 commercial unit, installed in 1996. The control unit is mounted immediately above the control console, with a foot switch underneath the console for hands-free communication. Speaker locations include: Reactor Manager's Office, Reactor Operators' Office, control room door, lower level door to reactor bay, Neutron Activation Analysis laboratory, 22 foot level above tank, 22 foot level near pool surface, and one next to each of the four beam ports.

Telephones at the facility share a common line. They are located in the Reactor Manager's office, the Reactor Operators' office, the control room, at the 22 foot level of the reactor bay, and at the 0 foot level. The NAA lab has a telephone with a separate line. The offices and control room also have high-speed Ethernet connections that can link with the reactor computer or outside machines.

9.5 Possession & Use of Byproduct, Source, & 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 special nuclear materials license. The reactor fuel is the property of the Department of Energy. Several radioactive sources are owned by Kansas State University. Radioactive materials, including special nuclear material (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 License and recorded in a radioactive transfer log. The State license is maintained by the KSU Department of Environmental Health and Safety, Division of Public Safety and administered by the University Radiation Safety Committee. Only individuals listed under the license are permitted to receive materials. Normally, a member of reactor staff is also approved by the Committee to receive byproduct and special nuclear material under the state license. Possession limits are set by the State, and the University Radiation Safety Committee 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 State of Kansas license are transferred to the Department of Public Safety for disposal under the State license. Disposal of low-level wastes generated under the reactor license is coordinated with the Department of Environmental Health and Safety, Division of Public Safety.

SNM inventory is reported to the Nuclear Assurance Corporation under Reporting Identification Symbol (RIS) ZKL. The reactor fuel comprises the bulk of SNM at the facility. This fuel is owned by the Department of Energy and possessed under the Reactor Facility license R-88. Also

under license R-88 are fission chambers containing SNM, owned by KSU. Possession limits are set by the license R-88 at 3.98 kg ^{235}U in enrichments less than 20% and 0.02 kg ^{235}U in enrichments up to 100%. Small quantities of SNM for experiments are owned by KSU 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. Fuel storage locations are listed in Section 9.2. 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 Cover Gas Control in Closed Primary Coolant Systems

The KSU reactor has an open primary coolant system and hence has no cover gas control. Nitrogen 16 is controlled as described in Chapters 5 and 11 by forcing convection cooling flow form the reactor core into a helical pattern (to enhance time delays for more decay). Using helium in the pneumatic system (instead of air) controls the possible inventory of radioactive argon.

9.7 Other Auxiliary Systems

9.7.1 Reactor Sump System

All floor drains and the recirculating ventilator condensate line feed into the reactor sump. The reactor sump is a square cavity covered by steel plates has a capacity of 3.8 kL. It is also connected to the sub-floor tracks used for moving the thermal column door, giving it an overall capacity of 4.5 kL. A sump discharge system was installed in 1997 that permits the sump 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 drawn for analysis of specific activity. Historically, the only isotope normally discharged was tritium from primary water, in quantities well below 10CFR20 limits.

9.7.2 Reactor Bay Polar Crane

A polar crane in the reactor bay is used to manipulate loads of up to 3630 kg (8000 lb). Various lifting straps and attachments are available for handling varied loads. A breaker in the reactor control room supplies power to the crane. A lockable breaker on the west wall of the reactor bay permits securing power to the crane. A disconnect on the crane permits personnel servicing the crane a local and positive control over power to the crane. A basket is attached to the outside edge of the crane for changing light bulbs in the ceiling of the reactor bay. Due to safety concerns, lighting was installed around the periphery of the reactor bay, reducing the need for the overhead lights and the basket.

9.7.3 Beam Facilities

A complete description of the beam tube facilities is provided in Chapter 10. Auxiliary systems that support the beam tubes are described below.

a. Thermal Column Door

The thermal column door rolls on railroad tracks, which are deeply recessed, into the floor. It is opened and closed by means of steel cables and a winch. A key lock secures the winch. The steel cables were replaced in 1997. Indicator lights on the control console indicating the position of the door were also added when the cables were replaced.

b. Beamport Plug Handling

A special cask is available for use in removal and storage of beamport plugs. It allows the plugs to be drawn directly from the port into the cask. A lead shutter can be closed over the open end, further reducing exposures.

c. Beam Facility Vents and Drains

All air-filled spaces in beam facilities are connected by pipes to a manifold mounted on the outside of the reactor. Additional pipes near the beamport doors have valves that can be opened to allow insertion of a tool to open the lead shield doors or to vent the beamport. Beamport doors are padlocked shut. Over the last several years, the void space near the thermalizing column has been collecting water. The source of the water is likely from the bulk shield tank, since the maximum observed pressures correspond to the level of the bulk shield tank and not the primary. With the bulk shield tank water level about 6 in. below the top of the tank, the water leakage is essentially secured. Regardless of the source, there was concern that this water might pass through the manifold and fill the other beam lines, or obscure evidence of leakage from other experimental facilities. Hence in 1996, the thermalizing column beam port line was connected to a separate manifold to permit collection of fluids.

d. Pneumatic Transfer System (Rabbit)

The pneumatic transfer system or rabbit is used to rapidly transport samples between an in-core location and the Neutron Activation Analysis Laboratory. From commercial cylinders, compressed helium fills small tanks at either end of the system. Pressure is limited by release valves at 275 kPa (40 psi). The system is operated from the instrumentation rack (Fig 7.3) in the control room, where the reactor operator sets the direction of motion by positioning vent valves and applies the helium by another valve. Indicator lights show position of valves. A light indicates low helium pressure.

9.7.4 Associated Laboratories

The C.C. Tate Memorial Neutron Activation Analysis Laboratory is located in the basement of Ward Hall. It features several germanium detectors and associated electronics for gamma ray spectroscopy. There is also an alpha counter with beta and gamma discrimination. The Radiation Detection Laboratory is a student lab, but contains many additional pieces of detection equipment. Another room contains a panoramic irradiator for performing gamma-ray instrument calibrations. NBS-traceable alpha, beta, and neutron sources are also available.

9.8 Bibliography

None

10. Experimental Facilities and Utilization

10.1 Summary Description

10.1.1 Experimental Programs

The K-State reactor provides educational and training services to support the Department of Mechanical and Nuclear Engineering Bachelor of Mechanical Engineering with a Nuclear Option as well as Masters and Doctoral programs in Nuclear Engineering. Other K-State departments with nuclear, radiation or instrumental analysis components in other curricula are also supported.

The K-State reactor produces radioisotopes for research, including both tracer/gauging and radioanalysis applications. The K-State reactor performs extracted beam research.

10.1.2 Experimental Facilities

Sectional views of the reactor are shown in Chapter 1. For orientation of experimental facilities with respect to the reactor tank and concrete shield, the reader is referred to Figs. 1.1 and 1.2 of Chapter 1. Components of experimental facilities located in the reactor pool are illustrated in Figure 10.1. Principal experimental features of the KSU TRIGA Reactor Facility include:

- Central thimble
- Rotary specimen rack
- Thermalizing column (with bulk shielding tank)
- Thermal column (with removable door)
- Beam ports
  - Radial (2)
  - Piercing (fast neutron) (1)
  - Tangential (thermal neutron) (1)

10.1.3 Experiment Monitoring and Control

Monitoring requirements for individual experiments are identified in the applicable experiment procedure. Experiment monitoring which is a routine part of facility equipment includes a system to detect leakage from reactor pool wall into the beam tubes or thermal/thermalizing column, and to minimize leakage into the reactor bay, capabilities for installing an area radiation monitor channel in the vicinity of an open beam port, and the capability for installing an external scram in the reactor protection system.

a. Leak Detection System

The interface of experimental facilities (beam ports, thermal column and thermalizing column) and the reactor pool liner contains an open plenum with piping connected to a leak off volume.. The leak detection piping is connected to a single volume, except that a separate leak off volume and pressure gauge has been installed for the thermalizing column. If the pool leaks into the experiment facilities, the water will overflow the

plenum and fill the leak off volume. Pressure monitors in the leak off volume indicate when the volume is partially or fully filled. The leak off volumes and the pressure monitors are located on the north wall of the biological shielding near the northwest (radial) beam port.

Watertight seals are installed at the interface between the beam ports and the reactor bay. With the original seal attached to the inside of the beam port door, the seals are designed to be used when beam ports are not in use. If the pool wall fails, this seal will prevent complete loss of water from the pool.

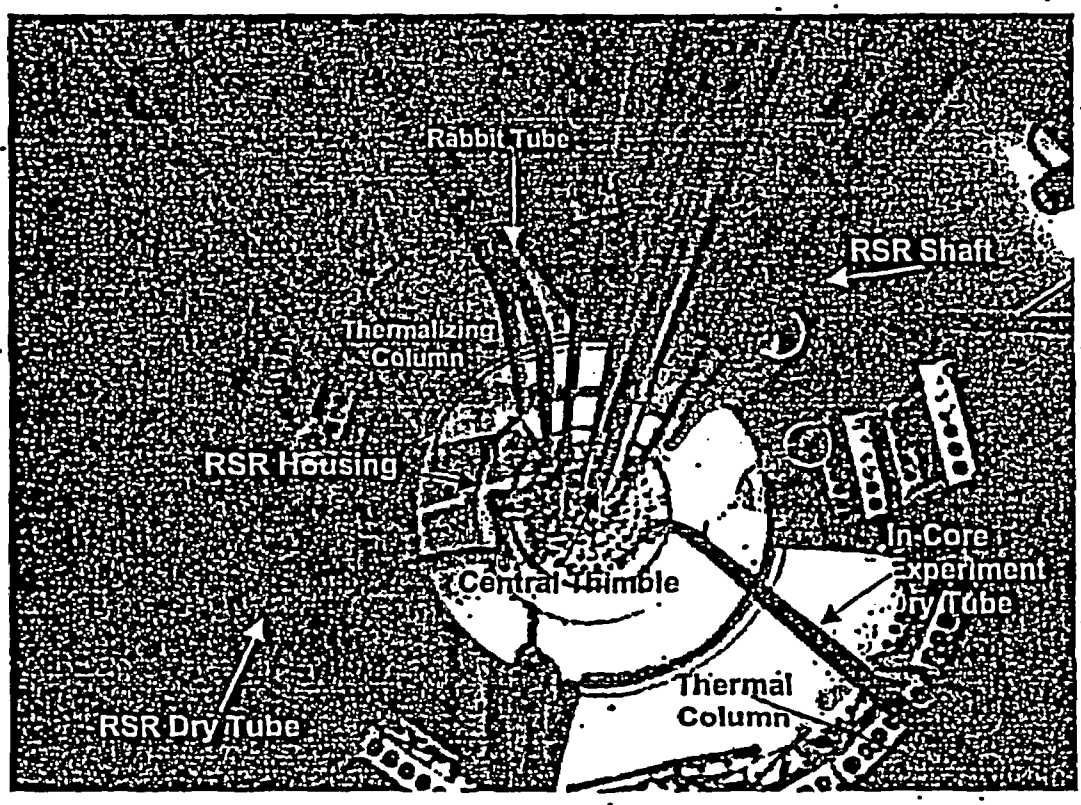

Figure 10.1: Experiment Facility Components in the Reactor Pool

b. Area Radiation Monitor

Electrical lines are installed to support area radiation monitors at each beam port.

c. **External Scram**

One scram relay has been designated as the "external scram channel." The external scram channel is not normally in use. In the event that potential consequences associated with an experiment requires protection for the facility, personnel or the experiment, sensors may be connected to cause a reactor scram based on significant parameters.

10.1.4 Experiment Review and Approvals

Experiments are reviewed and approved by the Reactor Safeguards Committee prior to performance as a reactor experiment procedure. The Reactor Supervisor or Nuclear Reactor Facility Manager may schedule an approved experiment for performance.

The Reactor Safeguards Committee approves reactor operations prior to performance via Reactor Operating Procedures. Operations supporting education, training and requalification without insertion of material in the experimental facilities is considered an operation.

10.2 Experimental Facilities

The K-State reactor is a flexible, multi-use facility with irradiation facilities inside the core boundary, in the reflector, outside the reflector, and outside the biological shielding. One of the in-core facilities is a pneumatic sample delivery system capable of providing samples directly to the neutron activation analysis laboratory.

10.2.1 In Core Facilities

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

a. **Available Fuel Element Spaces**

Experiments may be inserted in spaces designed for fuel elements. Core 18-II has one empty space, two spaces occupied by graphite "dummy" rods, and one space occupied by a dry tube. Typical dry tube configuration uses a modified "S" bend to minimize streaming radiation at the pool surface. One dry tube is lined with cadmium to support spectrally tailored neutron irradiations.

b. **Small Upper Grid Plate Penetrations**

The upper grid plate was fabricated with a series of small (slightly larger than ¼") holes to permit flux-mapping experiments. These penetrations may also be used to irradiate targets with suitable geometry.

c. **Central Thimble**

The reactor is equipped with a central thimble for access to the point of maximum flux in the core. Procedures limit the maximum power permitted with fueled experiments in the central thimble. A removable screen at the top end of the thimble allows gas relief and prevents objects from falling through the reactor tank covers.

The central thimble is an aluminum tube that fits through the center holes of the top and bottom grid plates terminating with a plug at a point approximately 7.5 in. (19 cm) below the lower grid plate. The tube is anodized to retard corrosion and wear. Although the shield water may be removed to allow extraction of a vertical thermal-neutron and gamma-ray beam (not done at the KSU facility), four 0.25-in (6.3-mm) holes are located in the tube at the top of the core to prevent expulsion of water from the section of the tube within the reactor core. Dimensions of the tube are 1.5-in. OD (3.81 cm), with 0.083-in (2.0 mm) wall thickness. The thimble is approximately 20 ft. (6.1 m) in length, made in two sections, with a watertight tube fitting.

10.2.2 In Tank, Ex Core Facilities

Experiment procedures authorize irradiations above and below the core (note that the fuel contain san internal reflector at the top and bottom of each element) and adjacent to the radial reflector. Irradiation is also authorized in the bulk shield tank, and at the outer face of the thermal column.

10.2.3 In Reflector Facilities

The principal, traditional in-reflector facilities include a "thermal column" with dry irradiation space, and a thermalizing column provides irradiation space in the bulk shield tank. A third facility is nested inside the radial reflector, the rotary specimen rack.

a. ***Thermal Column***

The thermal column is a large, boral-lined, graphite-filled aluminum container, with outside dimensions 4 ft (1.2 m) square in cross section and approximately 5.35 ft (1.6 m) in length. The thermal-column liner is a seal welded container fabricated from 0.5-in. (12.7-mm) aluminum plate. The outer portion is embedded in the concrete shield and the inner portion is welded to the aluminum reactor tank. The exterior surfaces are coated with plastic for corrosion protection. The portion of the thermal column welded to the aluminum tank extends to the graphite reflector and matches the contour of the reflector over a 100-degree angle. In a vertical plane, the column extends approximately 13 in. (33 cm) above and below the reflector, with the centerlines of the column and reflector coincident.

The aluminum container is open toward the reactor room. Blocks of nuclear-grade graphite occupy the entire void except for a 2-in (0.79 cm) thick lead curtain located within the column. The individual blocks are approximately 4-in (10.2-cm) square in cross section, the longest being 50 in. (1.27 m) in length.

Twelve graphite blocks serve as removable foil stringers. These blocks are machined 1/16 in. (1.6 mm) undersize for easy removal and insertion. The central block is aligned with a stringer-access plug in the thermal column door. To gain access to other than the central stringer, it is necessary to roll out the thermal-column door on its tracks.

Surrounding the graphite on the inside of the aluminum casing on all sides are 1/8-in. (3.2 mm) sheets of boral. This is done to reduce capture-gamma-ray production in the surrounding concrete shield.

b. Thermalizing Column

The bulk-shielding experimental tank is 12 ft (3.7 m) deep, 8 ft (2.46 m) wide, and 9 ft (2.8 m) long. The tank is waterproofed with an epoxy coating and is filled with water for shielding. The thermalizing column is similar to the thermal column, but smaller. Its outer section extends from the bulk-shielding experimental tank through the concrete shielding and to the aluminum reactor tank. The inner section of the column is welded to the tank, and extends to the reflector assembly and matches its contour. The division of the column into two sections allows for thermal expansion during reactor operation. The column is 2 ft (61 cm) square in cross section by approximately 4.33 ft (1.32 m) in length. It is fabricated from 0.5-in. (12.7-mm) thick seal-welded aluminum. The horizontal centerline is coincident with the centerline of the active fuel lattice. The exterior surfaces in contact with concrete are coated with plastic for corrosion protection.

An aluminum cover plate 0.625 in. (15.8 mm) thick, held in place by twenty 5/8-in. anodized aluminum bolts, keeps water out of the thermalizing column. A 0.25-in. (6.44-mm) neoprene gasket provides the seal. In the region of the concrete shield, the aluminum container is lined with 0.125-in. (3.2-mm) boral sheets that extend 3 ft. 2.125 in. (96.8 mm) inward from the bulk-shielding tank.

At the inner end, nearest the reactor core, the column is filled with graphite blocks to an axial thickness of 8 in. (20.3 cm). All blocks are nuclear-grade graphite, 4-in (10.2-cm) square in cross section. This wall of graphite is backed by a 2-in. (50.8-mm) slab of lead. In the outer portion of the container, where the aluminum is lined with boral, 1 in. (2.54 cm) thick polyethylene sheets line the boral. The graphite, in the form of 4-in (10.2-cm) square blocks, is stacked 24 in. (61 cm) thick from the outer edge of the column.

c. Rotary Specimen Rack

A rotary, 40-position rotary specimen rack (RSR) is located in a well in the top of the graphite radial reflector. The RSR allows large-scale production of radioisotopes and for activation and irradiation of multiple material samples with neutron and gamma ray flux densities of comparable intensity. Specimen positions are 1.25 in. (3.18 cm) diameter by 10.8 in. (27.4) cm depth. Samples are manually loaded from the top of the reactor through a water-tight tube into the RSR. The rack may be rotated (repositioned) manually from the top of the reactor. Figure 10.2 illustrates the design features of the RSR, and Figs. 10.3 and 10.4 are respectively photographs of the RSR during construction and the rotation mechanism and housing at the 22-ft level of the reactor.

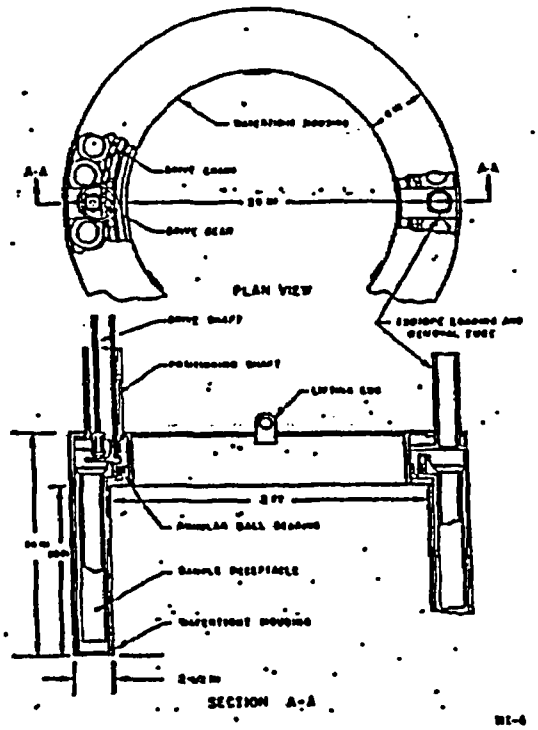

Figure 10.2: Rotary Specimen Rack

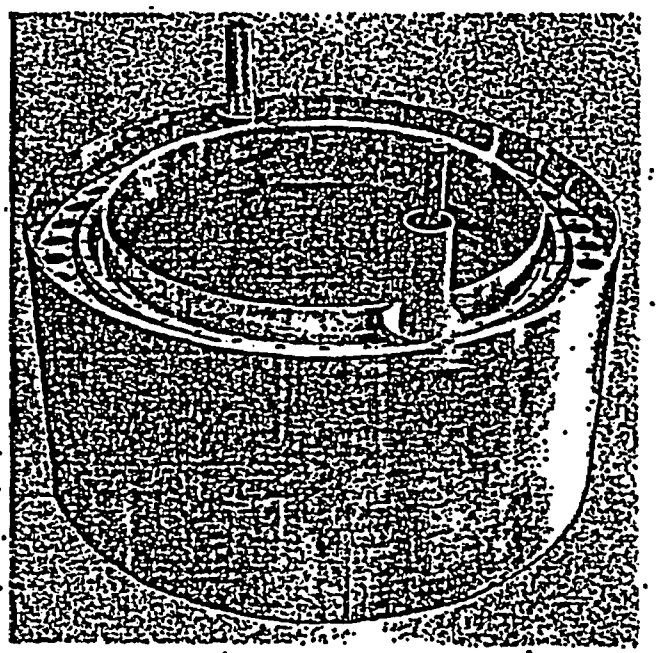

Figure 10.3: RSR during construction.

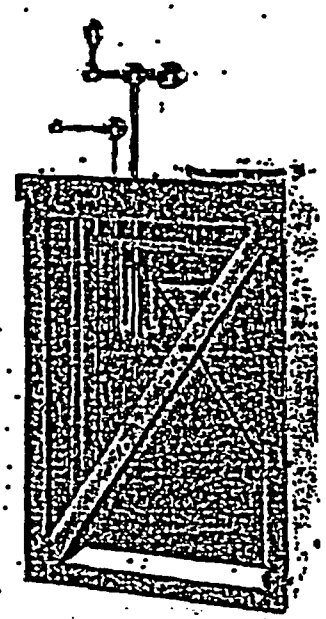

Figure 10.4: RSR rotation mechanism.

10.2.4 Automatic Transfer Facilities

A pneumatic transfer system, permitting applications with short-lived radioisotopes, rapidly conveys a specimen from the reactor core to a remote receiver. The in-core terminus is normally located in the outer ring of fuel-element positions. The sample capsule (rabbit) is made of polyethylene, approximately 14.2 mm ID and 100 mm length. It is conveyed to a receiver/sender station via aluminum and polyethylene tubing nominally 1.25-in. OD (3.18 cm) and at least 1.08 in. (27.4 mm) ID, with radii of curvature no less than 2 ft (61 cm). Figure 10.4 is a schematic diagram of the transfer system, as originally supplied. The gas supply has been replaced by compressed helium gas. The in-tank and in-core portion of the pneumatic transfer system is illustrated in Fig. 10.5.

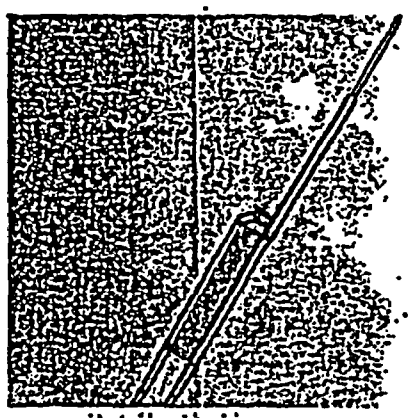

Figure 10.5: In-tank and in-core portions of the pneumatic transfer system.

10.2.5 Beam Ports

The KSU TRIGA Reactor is provided with four beam tubes. Beam-tube sleeves are welded to the outside surface of the tank to allow extraction of neutrons and gamma rays for a variety of experiments, and irradiation facilities for specimens as large as 6 in. (15.2 cm) in diameter. Three of the beam tubes are oriented radial with respect to the center of the core. The fourth is tangential to the outer edge of the core. All radial tubes terminate at the outer edge of the reflector assembly, but one is aligned with cylindrical void in the reflector graphite. In order that this void clears the rotary specimen rack in the reflector, all beam-tube axes are 2.75 in. (7.0 cm) below the centerline of the core.

The four beam tubes are in two sections within the concrete shield. The inner section is 6.065 in. (15.2 cm) inside diameter. The sleeve has a 6.315-in (16 cm) diameter. A step is incorporated into the design, with the outer section 8 in. (20.3 cm) in diameter. The outer section is made of cadmium plated steel. A 0.5-in. (12.7-mm) line in the argon vent system leads from the outer section, thus permitting purge of accumulated gases. The inner sections of the beam tubes are aluminum, coated on the outside with plastic to prevent corrosion when in contact with concrete. The gap between the inner tube and the sleeve prevents stresses resulting from thermal expansion in the aluminum tank.

A 4-in. (10.2-cm) thick steel shadow shield, 40-in. (102 cm) square, is placed around each beam tube in the concrete shield to provide additional shielding for the area adjacent to the beam port. The shield is placed around the inner tube, at its juncture with the outer. When beam tubes are not in use, special shielding is provided along the tube axes. The original shielding configuration is fabricated in four sections: an inner concrete-filled plug, an outer wooden plug, a lead-filled shutter, and a lead-lined door.

The inner section of the beam-tube shielding is a concrete-filled aluminum plug (Fig. 10.6) approximately 48 in. (1.22 m) long. It consists of a 42.5-in. (1.08-m) long 6-in. (15.2-cm) diameter inner portion, which is rigidly joined to a 5-in (12.7 cm) long, 7.875-in (20-cm) diameter steel outer portion. The plug weighs approximately 180 pounds (82 kg). The shielding in the inner plug consists of 1/8 in. (3.2 mm) of boral on the inner end, followed by 4 in. (10.2 cm) of lead, 36 in. (0.92 m) of borated normal-density concrete, and the 5-in. steel outer portion. This outer portion is equipped with a threaded hole for attaching the beam-tube-plug handling tool.

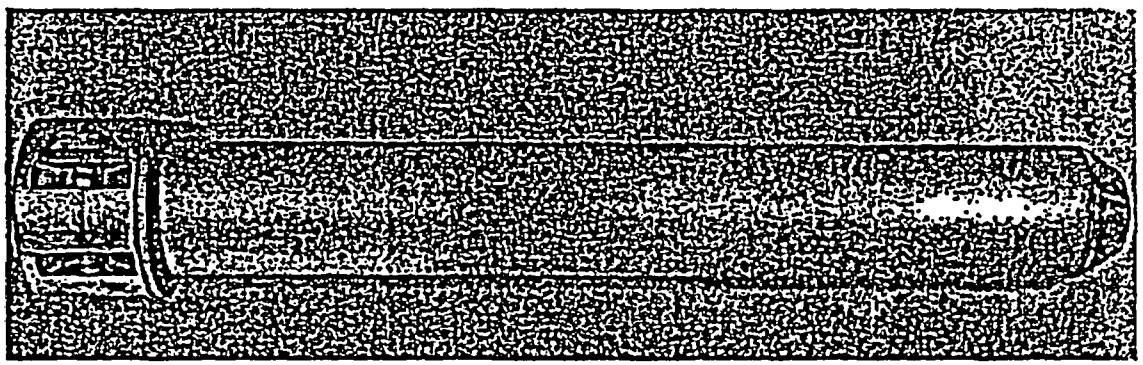

Figure 10.6: Inner beam tube plug.

The outer beam-tube shielding (Fig. 10.7) is a wooden plug, 48 in. (1.20 m) long and 8 in. (20.3 cm) diameter. The plug weighs about 45 pounds (20 kg) and has a handle for manual removal. Experience has shown the outer wooden beam port plug is not required to control radiation levels at acceptable levels, and the use of wooden plugs in the beam ports is optional.

The outer end of the beam tube is equipped with a lead-filled safety shutter and door to provide limited gamma ray shielding when the plugs are removed. The shutter (Fig. 10.8) is contained in a rectangular steel box recessed into the outer surface of the concrete shield. The shutter is 9 in. (23 cm) square and 4.5 in. (11.4 cm) thick. It is welded from 0.25-in (6.3-mm) steel plate and filled with lead. On one side of the shutter is welded a threaded socket for connection of a push rod.

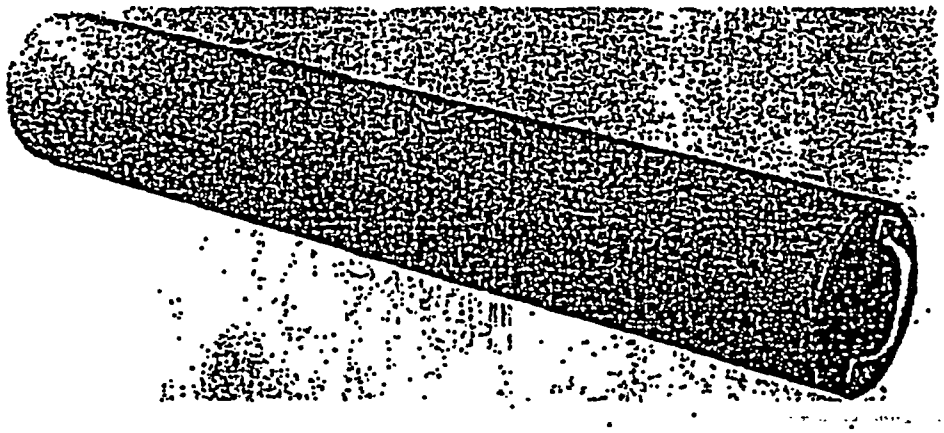

Figure 10.7: Outer beam tube plug.

The shutter box is equipped with a door made of 0.375-in (9.5 mm) steel lined with 1.25 in. (3.2 cm) of lead for additional shielding. The door is hinged at one side and equipped with a rubber gasket and six clamps that permit the door to be sealed against the possibility of rapid loss of shielding water in the event of a major beam-port leak. A valve mounted to the side of the steel box (not shown in the figure) provides access for the push rod operator (open position) and maintains the water seal on the safety shutter (closed position).

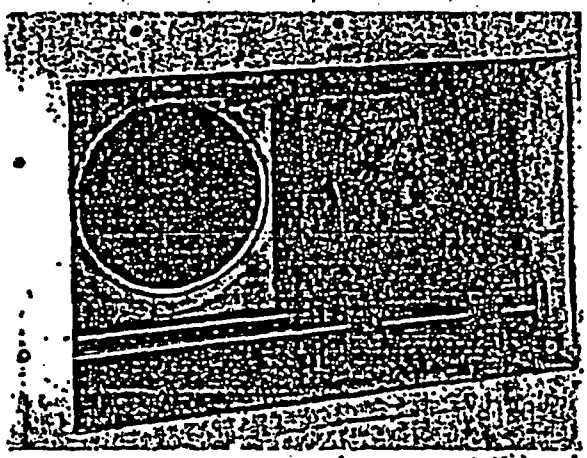

Figure 10.8: Beam tube safety shutter.

10.3 Experiment Review

A wide array of experiments have been documented and approved for execution in the operational history of the facility. The experiment review and approval process is conducted in accordance with approved facility administrative procedures. If an experiment falls within the scope of an approved experiment, a request for operation is submitted to the Reactor Supervisor. (or designated alternate). The Reactor Supervisor (or designated alternate) verifies that operation is within the scope of an approved experiment, and approves the request by signature so that the experiment may be scheduled. If it is determined that the proposed experiment does not fall with

n the scope of a previously approve experiment or if the experiment potentially involves an unreviewed safety question, the experiment is considered a "new" experiment.

10.3.1 Planning and Scheduling of New Experiments

New experiments require approval of the Reactor Safeguards Committee prior to implementation. As noted in Chapter 12, the Radiation Safety Officer must concur for permissive decisions. To support Committee review, a written description of each proposed new experiment must be prepared, with sufficient detail to enable evaluation of experiment safety. The Committee shall make an evaluation as to whether new contemplated experiments, procedures, facility modifications (and/or changes thereto) meet review criteria, and approve experimental operations (with or without changes or additional constraints) or prohibit the experiment from being performed. The following information is a minimum for the proposed experiment:

- Purpose of the experiment
- Background (if appropriate)
- Procedure - to include a description of the experimental methods to be used and a description of the equipment to be used. A sketch of the physical layout and a tabular list of equipment necessary for the experiment is recommended if appropriate.
- A summary of various effects that the experiment could cause, or that could interact with the experiment, or including:
  - Reactivity Effects
  - Thermal-Hydraulic Effects
  - Mechanical Stress Effects
- References.

The Reactor Safeguards Committee may require additional information to determine that an experiment is acceptable; the experiment shall not be scheduled until the Committee has reviewed the proposed experiment, including any supplemental information requested by the Committee.

10.3.2 Review Criteria

The Reactor Safeguards Committee shall consider new experiments in terms of effect on reactor operation and the possibility and consequences of failure, including, where significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and reactivity effects. Before approval, the Committee shall conclude that in their judgment the experiment by virtue of its nature and/or design will not constitute a significant hazard to the integrity of the core or to the safety of personnel. Evaluation of the proposed experiment shall include (as a minimum, not limited to) that the likelihood of occurrences listed below are 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 reactor control system 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 - e.g., electrical, potable water, etc., physical interruption of operational activities or egress, toxic or noxious industrial hazards, unanticipated effects of pulsing. Note this evaluation should also consider potential for fire or personnel exposure to toxic/noxious material)

- Determination that the proposed activity is in compliance with Technical Specifications

If an event or new information challenges the original evaluation, the Committee shall review the experiment approval and determine if the original approval is still valid prior to a continuation of the experimental program. When container failure is discovered that has released material with potential to damage the reactor fuel or structure (by corrosion or other means), physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Safeguards Committee and determined to be satisfactory before operation of the reactor is resumed.

10.4 Bibliography

"250-kW TRIGA Mark II Pulsing Reactor Mechanical Maintenance and Operating Manual," Report GA-3399, General Atomic Division, General Dynamics, 1962.

"License R-88, Docket 50-188, Amended Facility License for the KSU TRIGA Mark II Nuclear Reactor," Nuclear Regulatory Commission, Revised 21 Dec 1998 including Amendment 12.

"Kansas State University TRIGA Mark II Reactor Facility Operations Manual."

"Kansas State University TRIGA Mark II Reactor Facility Administrative Plan."

USNRC Regulatory Guide 2.4, "Review of Experiments for Research Reactors" (July 1976)

11. Radiation Protection and Waste Management

This chapter deals with the overall radiation protection program for the KSU TRIGA Mark II Nuclear Reactor Facility and the associated practices for management of radioactive wastes. The chapter identifies radiation sources that may be present during normal operation of the reactor and on the various procedures followed to monitor and control these sources. The chapter also identifies expected personnel radiation exposures due to normal operations. Supporting calculations are found in the Safety Analysis Report Appendix A.

11.1 Radiation Protection

The Radiation Protection Program for the K-State nuclear research reactor facility was prepared to meet the requirements of Title 10, Part 20.1101, Code of Federal Regulations (10CFR20). The Program also incorporates requirements of the State of Kansas. The Program seeks to control radiation exposures and radioactivity releases to a level that is As Low As Reasonably Achievable (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 Kansas State University. The Program is reviewed and approved by the Reactor Safeguards Committee for the Reactor Facility.

Certain aspects of the Program deal with radioactive materials regulated by the State of Kansas (an Agreement state) under license C0011-01. Therefore, the University Radiation Safety Committee (responsible for administration of the State license) reviewed the Program. The Radiation Protection Program was developed following the guidance of the American National Standard *Radiation Protection at Research Reactor Facilities* [1] and Regulatory Guides issued by the NRC [2-7].

11.1.1 Radiation Sources

Radiation sources present in the reactor facility may be in gaseous (airborne), liquid, solid, or form. These forms are treated individually in successive subsections.

Airborne sources consist mainly of ^{41}Ar (1.8 h half-life), attributable to neutron activation of natural ^{40}Ar in air in the reactor bay, in the rotary specimen rack adjacent to the core, and dissolved in the primary coolant. The nuclide ^{16}N (7.1 s half-life) is produced in the primary coolant as a result of the $^{16}O(n,p)^{16}N$ reaction. Because of its short half-life, ^{16}N contributes negligibly to off-site radiation exposure, but is the major source of radiation dose to the area above the reactor pool.

Liquid sources are principally limited to condensate water from the facility air handling system, which occasionally contains small concentrations of tritium. There are occasional releases of tritium-bearing 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 such as those used for instrumentation calibration. Solid waste is another solid source, very limited in volume and specific activity. Solid wastes include, ion-exchange resin used in reactor-water

cleanup, and contaminated tools, labware and anti-contamination clothing associated with reactor experiments and surveillance or maintenance operations.

a. Airborne Radiation Sources

During normal operation of the reactor facility, there are three major potential airborne sources, ^{3}H, ^{16}N, and ^{41}Ar. Airborne tritium occurs with evaporation of primary coolant, and contributes negligibly to personnel or public radiation exposure. Assumptions and calculations used to assess production and radiological impact of ^{16}N, and ^{41}Ar sources during normal operations are discussed in Chapter 11, Appendix A and summarized here.

Fuel element failure, although not expected, could occur while the reactor is operating normally (e.g., associated with manufacturing defects or cladding corrosion), and would result in a small penetration of the cladding through which fission products would be slowly released into the primary coolant. Some of these fission products, primarily the noble gases, would migrate from the coolant to the air of the reactor room. Although this type of failure could occur during normal operation, its occurrence is not normal and no normal operation would take place until the failed element were located and removed from the core. Thus, failure of a single element is evaluated in Chapter 13 as an abnormal, accident situation.

Tritium in the Reactor Bay

A 5-year average of tritium assay (performed monthly) indicates specific activity in the primary coolant of 228 pCi/ml for 250 kW operations. If the reactor bay atmosphere were saturated with this water at 30°C, the water concentration in the air would be less than 3×10^{-5} g mL^{-1} and the activity concentration in the atmosphere would be less than 6.84E-09 µCi/ml. Based on history, tritium concentration at 500 kW would be less than 1.37×10^{-8} µCi/ml, and tritium concentration at 1,250 kW would be less than 3.42×10^{-8} µCi/ml. In all cases, tritium concentrations are well below the 10CFR20 Appendix-B DAC of 2×10^{-5} µCi mL^{-1} and the atmospheric effluent limit of 10^{-7} µCi mL^{-1}. Even the primary coolant would meet the liquid effluent limit of 10^{-3} µCi mL^{-1} without further dilution.

^{41}Ar in the Reactor Bay

Production of ^{41}Ar arises from neutron absorption in natural argon present in air spaces, notably the rotary specimen rack (RSR), and dissolved in primary coolant. Occupational exposure to ^{41}Ar during normal operation of the KSU TRIGA reactor can therefore occur in the reactor bay. According to Appendix B of 10CFR20, the submersion DAC for occupational exposure is 3×10^{-6} µCi mL^{-1} and the effluent limit is 1×10^{-8} µCi mL^{-1}.

<u>Exhaust of the rotary specimen rack:</u>[1] As shown in Chapter 11 Appendix A, the equilibrium activity of ^{41}Ar in the RSR is 0.56 Ci for sustained operation at 500 kW

[1]Exhaust of other air volumes, e.g., beam tube or pneumatic transfer system, would release lesser quantities of ^{41}Ar into the reactor bay.

thermal power, 1.4 Ci at 1,250 kW. If this activity were instantly dispersed into the reactor bay atmosphere, under normal ventilation conditions, and a worker were continuously exposed thereafter, the cumulative exposure would be at 4.4×10^{-10} µCi h mL^{-1}; well below the occupational exposure limit for 2000 hours at the DAC, 3×10^{-5} µCi h mL^{-1}.

Release from primary coolant: As shown in Chapter 11 Appendix A, even with extremely conservative assumptions, during sustained operation at full power with ventilation, the steady-state activity concentration of ^{41}Ar in the reactor-bay atmosphere would be 7.2×10^{-7} µCi mL^{-1}, less than the occupational DAC.

Offsite Impact of ^{41}Ar. As shown in Chapter 11 Appendix A, the peak off-site activity concentration during normal operations would be about 0.003903 pCi mL^{-1} at 135 m downwind under slightly unstable atmospheric conditions, occurring 0.6% of total time. This concentration is less than the effluent limit of 0.01 pCi mL^{-1}. A full year of operation at the maximum power level maximum concentration would result in an effective dose at the receptor with the maximum concentration of only about 0.16 mrem, well within applicable limits. The highest dose to a location occurs at 2140 meters with a dose of 3.8 mrem, well below the maximum allowed 10 mrem from effluents.

^{16}N in the Reactor Bay

An additional source of airborne radioactivity during normal reactor operations is ^{16}N, which is generated by absorption of fast neutrons in ^{16}O present in the primary coolant within the core. Compared to molecular oxygen, dissolved oxygen or oxygen in the RSR airspace is of negligible significance.

After ^{16}N is produced in the core region, it rises to the tank surface and spreads to form a cylindrical volume source within the reactor tank, thereby leading to significant exposure rates above the tank. As discussed in Chapter 11, Appendix A, ^{16}N created in the coolant is likely to remain in anionic form and in solution, with negligible release in the gaseous state. Furthermore, because of the only 7.13-second half-life of ^{16}N, airborne concentrations, on-site and off-site, are of negligible significance compare to the direct dose from ^{16}N dissolved in the primary coolant.

As shown in Chapter 11 Appendix A, conservative calculations lead to an expected exposure rate of approximately 25 mR h^{-1} at one meter above the center of the reactor tank during sustained operation at 500 kW thermal power, increasing to nearly 100 mR h^{-1} at 1,250 kW.

b. Liquid Radioactive Sources

During normal operation of the KSU TRIGA reactor, the only production of liquid radioactive materials occurs through neutron activation of impurities in the primary coolant. Most of this material is captured in mechanical filtration or ion exchange in demineralizer resin; therefore, these materials are dealt with as solid waste. Non-routine liquid radioactive material could result from decontamination or maintenance activities such as resin changes. These liquids are collected in sump tanks along with condensate

from the air conditioning system. Quantities are small and these liquids are released to the sanitary sewerage system after assay and filtration. Most of these are generated from condensate drains of the air-conditioning system. The only radionuclides observed are tritium and (sporadically) trace quantities of ^{137}Cs. Typically there are three releases of liquids annually, each amounting to 2.5 m^3. Typical concentrations are 2×10^{-4} µCi mL^{-1} of ^{3}H and 2×10^{-7} µCi mL^{-1} of ^{137}Cs. Even without dilution, all these are well below 10CFR20 Appendix-B effluent concentration limits of 1×10^{-3} µCi mL^{-1} of ^{3}H and 1×10^{-6} µCi mL^{-1} of ^{137}Cs, and monthly sewerage limits of 1×10^{-2} µCi mL^{-1} of ^{3}H and 1×10^{-5} µCi mL^{-1} of ^{137}Cs.

The only significant liquid radioactive source at the KSU TRIGA reactor is the primary coolant. The only measured radionuclides are ^{41}Ar, ^{16}N, and ^{3}H, and their consequences are examined in Section Chapter 11 Appendix A. Activation products such as ^{27}Mg, ^{28}Al, and ^{56}Mn, while undoubtedly present, are of such low concentrations and have such short half-lives that they are not detected in surveillance programs. Similarly, ^{16}N present in primary coolant as it passes through the cooling system undoubtedly contributes to ambient dose rates inside the reactor bay but surveillance at full power reveals no contributions of 1 mR h^{-1} or greater.

c. **Solid Radioactive Sources**

The solid radioactive sources associated with KSU TRIGA reactor operations are summarized in Table 11-1. Because the actual inventory of fuel and other sources continuously changes in normal operation, the information in the table is to be considered representative rather than an exact inventory.

[redacted]

Solid and liquid wastes are not included in Table 11-1a and 11-1.b. These sources are addressed in Section 11.2.

11.1.2 Radiation Protection Program

The Radiation Protection Program was prepared by personnel of the Kansas State University TRIGA Mk II Nuclear Reactor Facility in response to the requirements of Title 10, Part 20.1101, Code of Federal Regulations (10CFR20). The goal of the Program is the limitation of radiation exposures and radioactivity releases to a level that is as low as reasonably achievable without seriously 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, Kansas State University. It has been reviewed and approved by the Reactor Safeguards Committee for the Reactor Facility. Certain aspects of the Program deal with radioactive materials regulated by the State of Kansas (an Agreement state) under license C0011-01 and the Program has been reviewed by the University Radiation Safety Committee, which is responsible for administration of that license. The Radiation Protection Program is designed to meet requirements of 10CFR20. It has been developed following the guidance of the American National Standard *Radiation Protection at Research Reactor Facilities* [1] and Regulatory Guides issued by the NRC [2-7].

[redacted]

a. Management and Administration

Preparation, audit, and review of the Radiation Protection Program is the responsibility of the Nuclear Reactor Nuclear Reactor Facility Manager. The Reactor Safeguards Committee (chaired by the Head of the Department of Nuclear Engineering) reviews the activities of the Manager and semi-annual audits prepared by the Manager. The Reactor Safeguards Committee examines records required by the Radiation Protection Program as well as audit reports by the Manager during their semi-annual inspections.

Training, surveillance and record keeping are the responsibility of the Reactor Supervisor who reports to the Nuclear Reactor Facility Manager. ALARA activities, for which record keeping is the particular responsibility of the Reactor Supervisor, are incumbent upon all radiation workers associated with the Reactor Facility.

Substantive changes in the Radiation Protection Program require approval of the Reactor Safeguards Committee. Editorial changes, or changes to appendices, may be made on the authority of the Nuclear Reactor Facility Manager. Changes made to the Radiation Protection Program apply automatically to operating or emergency procedures; corresponding Program changes may be made without further consideration by the Reactor Safeguards Committee. As with procedures, the Reactor Supervisor or Nuclear Reactor Facility Manager may override elements of the Program on a temporary

emergency basis so long as the emergency changes are brought promptly to the attention of the Safeguards Committee.

b. Training

Implementation of training for radiation protection is the responsibility of the Reactor Supervisor. There are two categories of radiological training at the K-State reactor. Personnel who need access to the facility, but are not reactor staff, are either escorted by trained personnel or provided unescorted access training. Radiation training for licensed operators and staff is integrated with the training and requalification program.

The goal of unescorted access training is to provide knowledge and skills necessary to control personnel exposure to radiation associated with the operation of the KSU nuclear reactor. Specific knowledge and skills required to meet the goal have been developed as learning objectives, and training material is based on these learning objectives. Specific training requirements of 10 CFR 19, 10 CFR 20, the Radiation Protection Plan, and the Emergency Plan are explicitly addressed. A facility walkthrough is incorporated.

All persons granted unescorted access to the Reactor Facility must receive the training and must complete without assistance a written examination over radiation safety and emergency preparedness. An examination score of at least 70 percent is required. Examinations must be retained on file for audit purposes for at least three years.

The reactor staff accomplishes health physics functions at the K-State reactor following approved procedures. Therefore, procedure training for the licensed reactor staff training includes additional radiological training. Examinations for reactor staff training are prepared and implemented in accordance with the K-State reactor requalification plan.

11.1.3 ALARA Program

a. Policy and Objectives

Management of the Reactor Facility is committed to keeping both occupational and public radiation exposure as low as is reasonably achievable (ALARA). The specific goal of the ALARA program is to assure that actual exposures are no greater than 10 percent of the occupational limits and 50 percent of the public limits prescribed by 10CFR20, namely, ALARA goals of:

Workers:	≤ 500 mrem annual TEDE
	≤ 5 rem annual dose equivalent to any organ except the lens of the eye
	≤ 1.5 rem annual dose equivalent to the lens of the eye
	≤ 5 rem annual dose equivalent to the skin
	≤ 50 mrem dose equivalent to the fetus during pregnancy
Public:	≤ 50 mrem annual TEDE

b. Implementation of the ALARA Program

Planning and scheduling of operations and experiments, education and training, and facility design are the responsibilities of the Reactor Supervisor and/or the Nuclear Reactor Facility Manager. Any action which, in either of their opinions, might lead to as much as half the annual ALARA dose limit (Section 6.1) to any one individual in one calendar quarter requires a formal ALARA review and report. Any staff member or experimenter, or any member of the Reactor Safeguards Committee may call for an ALARA review of a proposed action. Under any of these circumstances, it is the responsibility of the Reactor Supervisor to conduct an ALARA review and report. Only with the approval of the Reactor Supervisor and the endorsement of the Nuclear Reactor Facility Manager, may the action proceed.

c. Elements of the ALARA Review and Report

The following topics shall be considered, if applicable. The report shall include discussion of how these topics affect personnel exposure and specific actions recommended, categorized by topic:

Features for External Radiation Control

Shielding and construction materials
Radioactive material storage and disposal
Monitoring systems
Facility layout
Control of access to high and very high radiation areas

Contamination Control

Ventilation and filtration
Containment of contamination
Confinement of contamination spread
Construction materials to facilitate decontamination
Facility layout

Effluent Control

Gaseous effluents
Liquid effluents
Effluent monitoring

Operations and Operations Planning

Assessment of potential individual and collective exposures
Application of *shielding, time, and distance* for dose reduction
Use of ventilation and decontamination to reduce collective dose
Provision of special radiac or communications instrumentation
Provision of special personnel training and practice
Provision of special supervision and surveillance

Provision of special clothing or other protective gear

d. **Review and Audit**

Implementation of the ALARA Program is audited semi-annually by the Nuclear Reactor Facility Manager as part of the general audit of the Radiation Protection Program.

11.1.4 Radiation Monitoring and Surveillance

The radiation monitoring program for the KSU reactor is structured to ensure that all three categories of radiation sources—air, liquid, and solid—are detected and assessed in a timely manner. Surveillances are

a. **Surveillances**

Radiation monitoring surveillance requirements are imposed by the Reactor Safeguards Committee through the Radiation Protection Program (independent of the Emergency Plan) for:

Monthly	Wipe test reactor bay and control room
Quarterly	Source inventory and leak test
Semi-annually	Environmental surveillance (radiation levels at full power)

b. **Radiation Monitoring Equipment**

Radiation monitoring equipment used in the KSU reactor program is summarized in Table 11.2. Because equipment is updated and replaced as technology and performance requires, the equipment in the table should be considered as representative rather than exact.

c. **Instrument Calibration**

Radiation monitoring instrumentation is calibrated according to written procedures. Whenever possible, NIST traceable sources are used for the calibration. The Nuclear Reactor Facility or the University Radiation Safety Office (Office of Public Safety) are responsible for calibration of the Table 11.2 instruments on site. Calibration records are maintained by the facility staff and audited annually by the Nuclear Reactor Facility Manager. Calibration stickers containing pertinent information are affixed to instruments.

Table 11.2, Radiation Monitoring and Surveillance Equipment at the KSU TRIGA.

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

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control depends on many different factors including facility design features, operating procedures, training, proper equipment, etc. Training and procedures have been discussed in Section 11.1.2. This section deals with design features such as shielding, ventilation, containment, entry control for high radiation areas, protective equipment, personnel exposure, and estimates of annual radiation exposures for specific locations within the facility. Dosimetry records and trends are also included.

a. Shielding

The biological shielding around the KSU TRIGA reactor is the principal design feature for control of radiation exposure during operation. The shielding is based on TRIGA shield designs used successfully at many other similar reactors.

The reactor is designed so that radiation from the core area can be extracted into the 0 foot level for research, education, and training purposes. When the beam port shielding is removed, additional measures are required to control radiation exposure by either restricting access to areas of elevated radiation fields or providing a new shielding configuration for access. When a new shielding configuration is established (or modified), radiation surveys are required to determine access and monitoring requirements. When a specific configuration is used power levels higher than previous testing, additional radiation surveys are required to ensure adequate controls at the new power level. Monitoring is not required for a well-defined shielding configuration that previously met radiological limits, but is not prohibited.

b. Personnel Exposure

Regulation 10CFR20.1502 requires monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the limits prescribed in 10CFR20.1201.

The regulation also requires monitoring of any individuals entering a high or very high radiation area within which an individual could receive a dose equivalent of 0.1 rem in one hour. According to Regulatory Guide 8.7 [2], if a prospective evaluation of likely doses indicates that an individual is not likely to exceed 10 percent of any applicable limit, then there are no requirements for recordkeeping or reporting. Likewise, Regulatory Guide 8.34 [3] indicates that, if individual monitoring results serve as confirmatory measures, but monitoring is not required by 10CFR20.1502, then such results are not subject to the individual dose recordkeeping requirements of 10CFR20.2106(a) even though they may be used to satisfy 10CFR20.1501 requirements.

Table 11.3 lists results of a 12-year survey of occupational exposures at the KSU TRIGA Reactor Facility. There have been no instances of any exposures in excess of 10 percent of the above limits. Thus, retrospectively, only confirmatory monitoring is required and 10CFR20.2106(a) recordkeeping requirements do not apply, so long as there are no significant changes in the facility, operating procedures, or occupational expectations. For operation at 500 kW, records of exposure are not likely to result in increased record keeping requirements. If, in the view of supervisory personnel (Reactor Supervisor, Nuclear Reactor Facility Manager, or Radiation Safety Officer), any action under consideration might lead to exposures in excess of 10 percent of any applicable limit, then the ALARA program is triggered. A consequence of ALARA program planning, which is described in Section 11.1.3, might be the imposition of federally required recordkeeping procedures.

Table 11.3. Representative Occupational Exposures at 250 kW.

Year	Numbers of persons in annual-dose categories			
	Immeasurable	< 0.1 rem	0.1-0.5 rem	> 0.5 rem
1992	28	0	0	0
1991	23	0	0	0
1990	20	0	0	0
1989	19	1	0	0
1988	23	3	1	0
1987	23	0	0	0
1986	26	1	0	0
1985	31	8	0	0
1984	33	1	0	0
1983	29	2	0	0
1982	26	7	0	0
1981	11	23	0	0

Monitoring of workers and members of the public for radiation exposure required by the Reactor Safeguards Committee and is described in Facility Procedure 9. Principal objectives of Procedure 9 include:

Authorization for Access

Personnel who enter the control room or the reactor bay will either hold authorization for unescorted access, or be under direct supervision of an escort (i.e., escorted individuals can be observed by and hear instructions of the escort) who holds authorization for unescorted access.

Access Control During Operation

When the reactor is operating, the licensed reactor operator (or senior reactor operator) at the controls shall be responsible for controlling access to the control room and the reactor bay.

The 22-foot level access has a line of sight to the control room, and has radiation monitoring positioned directly over the pool surface and mounted on the rail surrounding the pool. The operator at the controls is responsible for appropriately controlling access to the 22-foot level based on radiological conditions.

Neutron Dosimetry

If there is potential for exposure of personnel to neutrons within the reactor bay, personnel who enter the reactor bay shall have neutron sensitive individual monitoring; this individual monitor shall be assigned to single individuals

Exposure Records for Access During Operation

Personnel who enter the reactor bay during reactor operation shall have a record of accumulated dose measured by a gamma sensitive individual monitoring device; at the

discretion of the reactor 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 reactor bay.

Exposure Records for Access During Non-Operating Conditions

Personnel who enter the reactor bay while the reactor is secured shall have a record of accumulated dose either by measurement through individual monitoring or based on assessment of data from individual monitoring devices or surveys.

c. Record Keeping

Although the Reactor Facility is likely exempt from federally required record keeping requirements of 10CFR20.2106(a), certain records are required in confirmation that personnel exposures are less than 10 percent of applicable limits.

Records of Prior Occupational Exposures

These records (NRC Form 4) are initially obtained, and then maintained permanently by the Office of Radiation Safety.

Records of Occupational Personnel Monitoring

The Office of Radiation Safety permanently maintains these records (NRC Form 5). Forms in use include monthly report for the University as a whole, monthly summary report for the Nuclear Reactor Facility, and quarterly report on extremity exposures for the University as a whole.

Records of Doses to Individual Members of the Public

Self-reading dosimeter records are kept in a logbook maintained by the Nuclear Reactor Facility. Such records are kept permanently. Results of measurements or calculations used to assess accidental releases of radioactive effluents to the environment are to be retained on file permanently in the Reactor Facility.

11.1.6 Contamination Control

Potential contamination is controlled at the KSU TRIGA reactor by using trained personnel following written procedures controls radioactive contamination, and by operating a monitoring program designed to detect contamination in a timely manner.

There are no areas within the reactor laboratory with continuing removable contamination. More likely sites of contamination are sample ports at the rotary specimen rack (RSR) and central thimble (CT) and at a sample-handling table for receiving irradiated samples. These sites are covered by removable absorbent paper pads with plastic backing, and are routinely monitored on a periodic basis. In some cases, an

enclosed work area is provided. If contaminated, pads are removed and treated as solid radioactive waste. While working at this or other potentially contaminated sites, workers wear protective gloves, and, if necessary, protective clothing and footwear. Workers are required to perform surveys to assure that no contamination is present on hands, clothing, shoes, etc., before leaving workstations where contamination is likely to occur. If contamination is detected, then 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.

Reactor Facility staff and visiting researchers are trained on the risks of contamination and on techniques for avoiding, limiting, and controlling contamination.

Table 11.4 lists sample locations for routine monitoring of surface and waterborne contamination control measures. On a monthly basis, 100 cm^2 swipe tests and 1 mL water samples are analyzed for contamination.

Table 11.4, Representative Contamination Sampling Locations.

SWIPE TESTS	WATER SAMPLES
0-foot level	Bulk shield tank
Floor	Primary coolant
Door handle	Secondary coolant
^{252}Cf Irradiator	
Source safe	
Source cave	
Sink	
Cleanup system	
12-foot level	
Floor	
Door handle	
Control room	
22-ft level	
Floor	
Sample table	
RSR loading port	
CT loading port	

Acceptable surface contamination levels for unconditional release are given in Table. 11.5, as provided in the approved *Radiation Protection Program, KSU TRIGA Mark II Nuclear Reactor Facility*. Limits on average contamination levels for unconditional release are calculated based on survey areas smaller than 1 m^2. Limits on maximum contamination levels for unconditional release are calculated based on survey areas smaller than 100 cm^2.

11.1.7 Environmental Monitoring

Environs monitoring is required to assure compliance with 10CFR20, Subpart F and with Technical Specifications for the Facility operating license. Installed monitoring systems include area radiation monitors, airborne contamination monitors, and a radiation monitor at the pool surface. The Reactor Safeguards Committee may require additional monitoring and has (via the Radiation Protection Program) established requirements for contamination and radiation survey surveillances.

a. Area Radiation Monitors

Area radiation monitors at the 22-ft level (and the 0 foot level; if beam ports are open) are required for reactor operation. Area radiation monitor calibration I accomplished as required by Technical Specifications in accordance with facility procedures.

b. Airborne Contamination Monitors

The facility uses two air monitoring systems, one on the 12-foot level and one in the exhaust-plume path from the reactor pool to the reactor bay exhaust system. Airborne contamination monitor calibration is accomplished as required by Technical Specifications in accordance with facility procedures.

c. Pool Surface Monitor

A radiation monitor is stationed directly over the pool surface. The pool surface monitor is calibrated in accordance with requirements of the Radiation protection program using facility procedures.

d. Additional Monitoring

The Reactor Safeguards Committee imposes additional requirements through the Radiation Protection Program.

Contamination Surveys

Contamination monitoring requirements and surveillances addressed in 11.1.6 prevent track-out of radioactive contamination from the reactor facilities to the environment. The K-State 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 20.1501, contamination surveys are conducted to ensure compliance with regulations reasonable under the circumstances to evaluate the magnitude and extent of radiation levels; concentrations or quantities of radioactive material; and potential radiological hazards.

Guidance has been promulgated in IE Circular No. 81-07 (Control of Radioactively Contaminated Materials) for releasing materials from restricted to unrestricted areas:

Based on the studies of residual radioactivity limits for decommissioning (NUREG-06132 and NUREG-07073), it can be concluded that surfaces uniformly contaminated at levels of 5000 dpm/100cm2 (beta-gamma activity from nuclear power reactors) would result in potential doses that total less than 5 mrem/yr. Therefore, it can be concluded that for the potentially undetected contamination of discrete items and materials at levels below 5000 dpm/100cm2, the potential dose to any individual will be significantly less than 5 mrem/yr even if the accumulation of numerous items contaminated at this level is considered.

The contamination monitoring using portable survey instruments or laboratory measurements should be performed with instrumentation and techniques (survey scanning speed, counting times, background radiation levels) necessary to detect 5000 dpm/100 cm2 total and 1000 dpm/100 cm2 removable beta/gamma contamination. Instruments should be calibrated with radiation sources having consistent energy spectrum and instrument response with the radionuclides being measured. If alpha contamination is suspected appropriate surveys and/or laboratory measurements capable of detecting 100 dpm/100 cm2 fixed and 20 dpm/100 cm2 removable alpha activity should be performed.

Radiation Surveys

Semi-annual environmental monitoring is conducted, involving measurement of both gamma-ray and neutron dose rates within the facility operations boundary and at the site boundary with the reactor at full-power operation.

Monthly surveys are conducted at the KSU TRIGA reactor for radiation levels with the reactor not in use. These are supplemented by semiannual monitoring of both neutron and gamma radiation levels during full power operations. Neutron dose rates are entirely negligible.

Gamma-ray exposure-rate data, based on semi-annual measurements over the ten-year period 1988-1998 (250 kW operation) is indicated in Table 11.6. Uncertainties are 1 standard deviation.. Source terms are related to reactor power levels; therefore maximum radiation levels during operation at 500 kW should not exceed twice the maximum historical values In Table 11.6.

Monitoring for Conditions Requiring Evacuation

An evacuation alarm (high radiation level) is required at the 22-ft level of the reactor. Response testing of the alarm is performed in accordance with facility procedures.

Table 11.6, Radiation Survey Results for 250 kW Operation.		
Level	Location	Maximum exposure rate (mR h^{-1})
22-ft	Pool surface	38 ± 9
	1-m above pool	9 ± 5
	Above bioshield	1 ± 1
12-ft	Above BST	13 ± 10
	Outside bioshield	2 ± 2
	Control room entry	0.07 ± 0.04
	Site boundary	0.03 ± 0.02
0-ft	Outside bioshield	0.02 ± 0.02
	Source cave	3 ± 3
	Cleanup/lab space	1 ± 1

11.2 Radioactive Waste Management

The KSU TRIGA reactor generates very small quantities of radioactive waste, as indicated in Section 11.1.1. Training for waste management functions are incorporated in operator license training and requalification program.

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 of containing radioactivity above background levels. Volumes of waste at the KSU TRIGA Reactor are small, and the nature of the waste items is limited and of known characterization, there is rarely question of what is or is not radioactive waste. Equipment and components are categorized as waste by the staff. Consumable supplies such as absorbent materials or protective clothing are declared radioactive waste if radioactivity above background is found to be present.

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. Kansas being an "agreement state," radioactive materials generated for

research and experiments under the federal byproduct material license of the Reactor Facility are transferred to State of Kansas license for conduct of the activities. Solid wastes resulting from experiments and activities conducted under the State of Kansas license are then physically transferred to the University Radiation Safety Office (RSO), Division of Public Safety, with representatives of the RSO and the Reactor staff certifying the transfer. Solid reactor waste is stored for decay until disposal, coordinated through the RSO.

Liquid wastes in the Reactor Facility are held in temporarily in storage tanks within the facility until pumped into the sanitary sewerage system of the University. Liquid wastes are primarily condensate from the building air-conditioning system and are very slightly radioactive because of the presence of tritium due to evaporation from the primary coolant and the bulk shield tank. To assure compliance with 10CFR20.2003, liquids wastes are assayed for alpha, beta, and gamma activity prior to release and are filtered to assure that no particulates are released along with liquids.

Although ^{41}Ar is released from the KSU TRIGA Reactor Facility, this release is not considered to be waste in the same sense as liquid and solid wastes. Rather, it is an effluent, which is routine part of the operation of the facility. A complete description of ^{41}Ar production and dispersal is provided in Chapter 11 Appendix A.

11.2.3 Release of Radioactive Waste

The history of liquid releases since 1995 is tabulated in Table 11.7. Note the seasonal release of air-conditioning condensate.

Table 11.7, Liquid Releases.

Date	Quantity released (m^3)	Measured pCi m^{-1} L above background		
		Alpha	Beta[a]	Gamma[b]
11 Aug 99	2.5	0	38	0
6 Jul 99	2.5	0	26	0.034
26 Aug 98	2.5	0	55	0.10
27 Jul 98	25[c]	0	67	0
21 Jul 98	2.5	0	206	0.23
26 Jun 98	2.5	0	89	0.27
16 Oct 97	2.7	0	120	0
8 Aug 97	2.3	0	250	0
27 May 97	0.7	0	0	0
5 Dec 96	3.3	0	150	0
12 Aug 96	3.0	0.004[d]	193	0
26 Jun 96	3.2	0	116	0
5 Sep 95	3.0	0	135	0
15 Aug 95	4.2	0	138	0
17 Jul 95	4.2	0	54	0
7 Jun 95	3.2	0	54	0

[a] Tritium as HTO (LSC analysis)
[b] ^{137}Cs-^{137m}Ba
[c] Draining bulk shield tank
[d] Unidentified anomaly

11.3 Bibliography

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Chapter 11 Appendix A

Radiological Impact of ^{41}Ar and ^{16}N During Normal Operations

A.1 Introduction

Normal operation of the KSU reactor results in two potential source terms for radioactive gaseous effluent at significant levels, ^{41}Ar and ^{16}N. There are variations in experimental configuration and possible scenarios where the production of ^{41}Ar may be different than the routine operations; these scenarios do not produce not long term, routine radioactive effluent but are assessed to determine if the amount of radioactive effluent is so high as to impact the annual exposure that might result from routine operations.

A.1.1 Purpose

The purpose of this appendix is to show the methods and calculations used to predict the production, concentrations, and dose rates from ^{41}Ar and ^{16}N associated with normal operation of the KSU TRIGA Mk. II nuclear reactor.

The nuclide ^{41}Ar is produced by thermal neutron absorption by natural ^{40}Ar in the atmosphere and in air dissolved in the reactor cooling water. The activation product appears in the reactor room (bay) and is subsequently released to the atmosphere through the reactor bay ventilation exhaust stream.

The nuclide ^{16}N is produced by fast neutron interactions with oxygen. The only source of ^{16}N in the reactor that needs consideration results from interactions of neutrons with oxygen in the cooling water as it passes through the reactor core. Any interaction with oxygen in the atmosphere is relatively insignificant and is neglected in this analysis.

A portion of the ^{16}N produced in the core is eventually released from the top of the reactor tank into the reactor bay. The half-life of ^{16}N is only 7.14 seconds, so its radiological consequences outside the reactor bay are insignificant.

Although not expected, the cladding of a fuel element could fail during normal operations as a result of corrosion or manufacturing defect. Should a failure occur, a fraction of the fission products, essentially the noble gases and halogens, would be released to the reactor tank and, in part, ultimately become airborne and released to the atmosphere via building ventilation. This operational occurrence, taking place in air, is addressed in Chapter 13 as the design basis accident for the TRIGA reactor.

Neutron interactions with structural and control materials, including cladding, as well as materials irradiated for experimental purposes, result in the formation of activation products. These products are in the nature of fixed sources and are mainly a source of occupational radiation

exposure. Administrative controls preclude the significant formation of airborne activation products, other than the aforementioned ^{41}Ar.

A.1.2 Radiological Standards

The concentration to dose rate (effective dose equivalent) conversion factor for submersion in an infinite atmosphere of ^{41}Ar is as follows: 2.17×10^{-10} Sv h^{-1} per Bq m^{-3}, or 0.803 mrem/h per pCi/ml (EPA 1993). For 2000 hours annual occupational exposure and 50 mSv maximum permissible annual exposure, this translates to a derived air concentration of 1.15×10^{5} Bq m^{-3} or 3.1×10^{-6} µCi cm^{-3} (3×10^{-6} µCi cm^{-3} as specified in 10CFR20). For 8766 hours annual public exposure and 1 mSv maximum permissible annual exposure, this translates to a derived air concentration of 526 Bq m^{-3} or 1.4×10^{-8} µCi cm^{-3} (1×10^{-8} µCi cm^{-3} as specified in 10CFR20).

A.1.3 KSU TRIGA Design Bases

General System Parameters

The calculations for ^{41}Ar and ^{16}N releases during normal operations are based on the following system parameters.

Table A.1, General System Parameters for Normal Operations at 500 kWt Full Power.

Parameter	Symbol	Value
Reactor steady power	P	1,250,000 W
Core coolant mass flow rate[a]	w	0.150 kg s^{-1}
Core coolant density	ρ	1.0 g cm^{-3}
Core avg. thermal neutron flux at full power (E ring)[b]	ϕ_{th}	2.05×10^{13} n $cm^{-2}s^{-1}$
Core avg. fast neutron flux at full power (E ring)[b]	ϕ_f	3.00×10^{13} n $cm^{-2}s^{-1}$
Thermal neutron flux in RSR at full power	ϕ_{RSR}	9.00×10^{12} n $cm^{-2}s^{-1}$
Total neutron flux per watt at fast (piercing) beam port		4250 n $cm^{-2}s^{-1}$ (0.5 MeV avg)
Total neutron flux per watt at tangential beam port		1400 n $cm^{-2}s^{-1}$ (0.1 MeV avg)
Fuel element heated length	L	0.381 m
Flow cross sectional area per fuel element[a]	A	6.2 cm^2
Mass flow rate per fuel element[a]	$\dot{m}$	108 g s^{-1}
Reactor tank diameter		1.98 m
Reactor tank depth		6.25 m
Reactor tank water depth above core		4.88 m (16 ft)
Coolant volume in reactor tank	V_c	1.92×10^{7} cm^3
Air volume in reactor bay (144,000 ft^3)	V_{bay}	4.078×10^{9} cm^3
Air volume in rotary specimen rack	V_{RSR}	3.75×10^{4} cm^3
Ventilation rate for reactor bay (air changes hourly)[d]	λ_v	0.368 h^{-1}

[a]See §4.6 of this report.
[b]See §5.8 of Operations Manual.
[c]See §13.2.2.2 of this report
[d]See letter B.C. Ryan (KSU) to Theodore Michaels (NRC) 15 Jan 99.

Reactor Core Parameters

Modeling of the reactor core for radiation transport calculations is based on the following approximations. For purposes of radiation shielding calculations the TRIGA reactor core may be approximated as a right circular cylinder 0.4572 m (18 in.) diameter (OD of F ring). The fuel region is 0.381 m (15 in.) high. On each end axially is a graphite zone 0.0874 m (3.44 in.) high and an aluminum grid plate 0.0191 m (0.75 in.) thick. In [redacted] locations, there are [redacted] fuel elements, 3 standard control rods and 1 transient control rod, 1 void location, 1 central thimble (void), 1 source (assume void), and 1 pneumatic transfer site (assume void). The fuel region may be treated as a homogeneous zone, as may be the axial graphite zones and the grid plates.

Fuel elements are 1.43-in. (3.6 cm) ID and 1.47-in (3.7 cm) OD, clad with type 304 stainless steel[1]. Fuel density is 5996 kg m^{-3}. Fuel composition is [redacted] uranium, [redacted] $ZrH_{1.65}$. The uranium is [redacted] ^{235}U and [redacted] ^{238}U. Steel density is 7900 kg m^{-3}. Standard control rods are 0.875-in. OD, the transient rod 1.25-in. OD. Both types of rods are clad with 30-mil thick aluminum (2700 kg m^{-3} density). The control material may be approximated as pure graphite, with density 1700 kg m^{-3}.

In radiation transport calculations, the core is modeled conservatively as a central homogenous fuel zone (air density neglected) bounded on either end by a homogeneous axial reflector zone, and by a 0.75-in. thick aluminum grid plate, treated as a homogeneous solid. Densities of the homogenous zones are as follow:

Fuel	3602 kg m^{-3}
Reflector	1147 kg m^{-3}
Grid Plate	2700 kg m^{-3}

Composition of the three zones, by weight fraction, are given in the following table.

Table A.2, Compositions of Homogenized Core Zones.

Element	Mass Fraction	Element	Mass Fraction
Fuel Zone		*Axial Reflector Zone*	
C	0.0617	C	0.7920
Al	0.0010	Al	0.0033
H	0.0139	Mn	0.0041
Zr	0.7841	Cr	0.0368
Mn	0.0013	Ni	0.0164
Cr	0.0117	Fe	0.1474
Ni	0.0052		
Fe	0.0469	*Grid Plate*	
U	0.0741	Al	1.0000

[1] Composition, by weight, 2% Mn, 18% Cr, 8% Ni, balance Fe.

Reactor Bay Parameters

For purposes of radiation dose calculations within the reactor bay, the dimensions are approximated as follows:

The reactor bay is approximated as a right circular cylinder 36 ft (10.973 m) high and 36.68 ft (11.18 m) radius. The reactor vessel structure is approximated as a right circular cylinder, co-axial with the bay, 22 ft (6.706 m) high and 11 ft (3.3528 m) radius. The free volume is 144,000 ft^3 (4078 m^3). The site boundary, at its nearest approach to the reactor bay, is about 2 m beyond the bay boundary, that is, at a radius of 13.13 m from the center of the reactor.

A.2 Radiological Assessment of ^{41}Ar Sources

A.2.1 Production of ^{41}Ar from Beams

Operation with a fully open beam port is not a routine operational condition. Beam port operations normally have shielding, collimation and beam stops that prevent a full beam from penetrating the column defined by the beam port into air volume between the reactor and the reactor bay wall. Operating experience with neutron radiography performed at 10 kW involves a neutron flux of 2×10^7 cm^{-2} s^{-1} or less. We assume here that this is the flux density along the beam port, which has a cross sectional area of 324 cm^2 (8-in diameter). In other words, in a radiography operation $S = 6.48 \times 10^9$ neutrons per second enter the atmosphere essentially in a parallel beam. The microscopic cross section for thermal neutron absorption in ^{40}Ar is 0.66 barns, so the macroscopic cross section for thermal neutron absorption in ^{40}Ar in air (0.0129 weight fraction) is $\mu = 1.54 \times 10^{-7}$ cm^{-1}. The maximum distance of travel of a neutron is from the reactor tank wall to the exterior wall of the reactor bay, namely, about $L_b = 1020$ cm. The decay constant for ^{41}Ar is $\lambda_r = 0.380$ h^{-1} and the ventilation rate is $\lambda_v = 0.368$ h^{-1}. The volume of the reactor bay is $V_{bay} = 4.08 \times 10^9$ cm^3. Thus, the activity concentration of airborne ^{41}Ar after sustained operation with an open beam port at 10 kW is given by:

$$C(\mathrm{Bq/m^3}) = \frac{S\lambda_r(1-e^{-\mu L_b})}{V_{bay}(\lambda_r+\lambda_v)} = 1.27 \times 10^{-4}\,\mathrm{Bq\,cm^{-3}} \qquad (1)$$

or 3.42×10^{-9} µCi mL^{-1} in conventional units. Operations at maximum power are not performed for radiography, and radiography is not performed long enough to achieve equilibrium ^{41}Ar. Therefore, scaling the calculation for sustained operations at 1,250 kW provides an extremely conservative bound on ^{41}Ar production. Scaling the 10 kW ^{41}Ar production value to 1,250 kW results in 4.28×10^{-7} µCi mL^{-1} which is slightly above submersion DAC for occupational exposure; however, conditions for the source term are related to a very unusual set of conditions (open beam port with no shielding) that are not continuous in two respects. Shielding for radiography external to the bema port limits the beam to less than ½ of the analyzed volume. Radiography configuration is implemented only for radiography operations, a small fraction of all operations. Typically radiography occurs less than 1 day per month. Radiography operations are inherently discontinuous as the purpose of individual operations are met when the image is

obtained. Typically a day of radiography operations involves less than 4 hours of operation at full power. These conservatisms assure DAC and effluent limits are met with no further consideration.

A.2.2 Production of ^{41}Ar in Rotary Specimen Rack

The air volume in the rotary specimen rack does not freely exchange with the air in the reactor bay; there is no motive force for circulation and the rotary specimen rack opening is routinely covered during operation. If the rotary specimen rack were to flood, water would force the air volume in the RSR into the reactor bay. The air volume of the rotary specimen rack (RSR) is approximately[2] $V_{RSR} = 3.75 \times 10^4$ cm^3 (HSR p. 28) and the thermal neutron flux density in the RSR is $\phi_{th} = 9.0 \times 10^{12}$ cm^{-2} s^{-1} at 1,250 kW thermal power. After sustained operation at full power, the equilibrium ^{41}Ar activity (Bq) in the RSR volume is given by

$$A_0 = \mu * \phi * V_{RSR} = 5.2x10^{10} \tag{2}$$

or, 1.4 Ci in conventional units. If this activity were flushed in to the reactor-bay atmosphere as a result of a water leak into the RSR, the initial activity concentration would be $A_o/V_{bay} = 3.5 \times 10^{-4}$ µCi mL^{-1}. This would instantaneously be well above even the occupational DAC for ^{41}Ar. However, with radioactive decay and ventilation, the concentration would decline in time according to

$$A(t) = A_o e^{-(\lambda_r + \lambda_v)t}. \tag{3}$$

If a worker were exposed to the full course of the decay, cumulative concentration (µCi h mL^{-1}) in the reactor bay would be

$$\frac{1}{V_{bay}} * \int_0^\infty dt * A(t) = \frac{A_0}{V_{bay} * (\lambda_r + \lambda_v)} = 1.6x10^{-4}\,\mu Ci\,h\,ml \tag{4}$$

The value 1.6×10^{-6} µCi s mL^{-1}, or 4.4×10^{-10} µCi h mL^{-1}, is well below the 3×10^{-6} µCi h mL^{-1} annual limit of 2000 DAC hours specified in 10CFR20/EPA-520/1-88-020.

A.2.3 Production of ^{41}Ar from Coolant Water

The reactor tank water surface is open to the reactor bay. Radioactive ^{41}Ar is circulated in the pool by convection heating, and freely exchanges with the reactor bay atmosphere during normal operation. The ^{41}Ar activity in the reactor tank water results from irradiation of the air dissolved in the water. The following calculations evaluate the rate at which ^{41}Ar escapes from the water into the reactor bay. The following variables plus those in Table A.1 are used in the calculations of ^{41}Ar concentrations in the core region, in the reactor tank outside the core, and in the reactor bay air.

[2] Approximated as a section of a cylindrical annulus, with 28-in. OD, 24-in. ID, and 14-in. height.

V_{core} = volume of core region ([redacted] elements) = [redacted] × A × L = [redacted])

C_{40} = ^{40}Ar atomic density (cm^{-3}) in coolant

σ = microscopic cross section (cm^2)

ρ = density (1.0 g cm^{-3})

ν = volumetric flow rate through core (cm^3 s^{-1})

τ = residence time for coolant in core at full power (s)

T = out-of-core cycle time for coolant.

Although [redacted] elements are assumed for thermal hydraulic analysis, the actual core water to non-water ratio is 33%; therefore, the active region of the core contains 5160 cm^3 of water. The saturated concentrations of argon in water at the coolant inlet temperature of 27°C is approximately 6.1 × 10^{-5} g per cm^3 of water (Dorsey 1940). If it is assumed that air is saturated with water vapor above the water tank (27 mm Hg vapor pressure at 27°C) and that the mole fraction of argon in dry air is 0.0094, the partial pressure of argon in air above the tank is 0.0094(760 - 27) = 6.9 mm Hg. By Henry's law, the concentration in water at the inlet temperature is 6.1 × 10^{-5} × 6.9/760 = 5.5 × 10^{-7} g cm^{-3} (C_{40} = 8.3 × 10^{15} atoms cm^{-3}).

The number of atoms per second of ^{41}Ar produced in the core is $C_{40} \times V_{core} \times \sigma\phi_{th}$ = 2.25 × 10^9. Activity is calculated as the product of the isotope concentration and the mean lifetime. If it were assumed that all atoms escape to the containment volume, the steady-state activity concentrations in the reactor bay atmosphere would be:

$$Bq_{cm} := \frac{C_{40}\, V_{core}\, \sigma_{40}\, \phi_{th}\, \lambda_\tau}{V_{bay}(\lambda_\tau + \lambda_\nu)} \tag{5}$$

Table A.2, Variables in Ar^{41} Calculations		
Variable	Units	Basis
$C_{40} := .83\ 10^{16}$	cm^{-3}	Calculated above
$V_{core} := 5160$	cm^3	33% of core volume
$\sigma_{40} := .66\ 10^{-24}$	cm^{-2}	Cross section
$\phi_{th} := .205 \cdot 10^{14}$	cm^{-2}s^{-1}	Table A.1
$\lambda_\tau := .000106$	s^{-1}	Radiological decay constant
$V_{bay} := .4077625909\ 10^{10}$	cm^3	144,000 cu ft
$\lambda_\nu := .000102$	s^{-1}	Bay effluent flow constant

The equilibrium ^{41}Ar concentration during full power steady state operation at 1,250 kW in the reactor bay would be 0.072 Bq cm^{-3} (1.9 × 10^{-6} µCi mL^{-1}) without ventilation and 0.027 Bq cm^{-3} (7.2 × 10^{-7} µCi mL^{-1}) with ventilation.

Environmental Protection Agency, Federal Guidance Report No. 11 (EPA FG 11 – Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion) lists the DAC for ^{41}Ar as (Table 1.b) $3X10^{-6}$ µCi/ml, or 3 pCi/ml. Therefore, equilibrium ^{41}Ar concentration during full power steady state operation at 1,250 kW is less than DAC and there are no restrictions on activities in the reactor bay imposed by the normal Ar^{41} production mode.

Even with the extremely conservative approximation of 100% release of ^{41}Ar to the atmosphere, the estimated steady state concentration, under ventilation, is less than the DAC.

A.2.4 Maximum Impact of ^{41}Ar Outside the Operations Boundary

Although there are three modes of ^{41}Ar production, only the release of radioactive argon dissolved in water occurs routinely. The ^{41}Ar produced in the reactor bay during normal operations is released to the atmosphere via an exhaust fan at approximately height $h = 11$ meters above grade. The flow rate is 4.17×10^5 cm^3 s^{-1} (884 cfm). At the steady state concentration computed in the previous section, the release rate would be $Q = 1.29$ µCi s^{-1}. The maximum downwind concentration (pCi cm^{-3}), at grade, may be computed using the Sutton formula (Slade 1968):

$$C_{max} = \frac{2Q}{e\pi\bar{u}h^2}\frac{C_z}{C_y}, \tag{6}$$

in which $\bar{u}$ is the mean wind speed (m s^{-1}), $e = 2.718$, and C_y and C_z are diffusion parameters in the crosswind and vertical directions respectively. The maximum concentration downwind occurs at distance d (m) given by

$$d = (h / C_z)^{\frac{2}{2-n}}, \tag{7}$$

in which the parameter n is associated with the wind stability condition. In this calculation, we adopt the values of n and C_z use in the McClellan AFB SAR. Mean wind speeds, by stability class are inferred from the data in Chapter 2. Calculations are shown in Table 2.3.

Table A.3, Atmospheric Dispersion Calculations.

Pasquill stability class	u (m s^{-1})	n	C_y (m$^{n/2}$)	C_z (m$^{n/2}$)	d (m)	C_{max} (pCi cm^{-3})
Extremely unstable (A)	1.6	0.2	0.31	0.31	53	0.003903
Slightly unstable (C)	4.0	0.25	0.15	0.15	135	0.001560
Slightly stable (E)	3.5	0.33	$4C_z$	0.075	393	0.000445
Extremely stable (G)	0.77	0.5	$8C_z$	0.035	2140	0.001013

The dose conversion factor provided by EPA FG 11 for ^{41}Ar (Table 2.3) is 2.17 X 10^{-10} Sv/hr per Bq/m^3, or (using the provided conversion factor of 3.7 X 10^{15}) 8.03 X 10^5 mrem/hr per μCi/cm^3, 8.03 X 10^{-1} mrem/hr per pCi/cm^3. Using the highest maximum concentration of Table A.3 (0.003903 pCi cm^{-3}) at steady state full power operation for a full year (8760 hours) with observed frequency of class A stability (see Appendix 2.C) would result in a dose less than 1 mrem/year. Frequency of occurrence and the concentration at the maximum dose will occur from class C conditions, with a maximum annual dose of 1.7 mrem. The maximum concentration at the highest frequency (class G) is 0.001013 pCi cm^{-3}, with a dose of 3.8 mrem.

The assumed 24-7 operating history is not feasible for the KSU reactor, which has an average operating time for two decades of about 8 hours/week. Additionally, a full power, continuous operation would require a significant quantity of new fuel.

Note that over the full range of conditions examined in Table 2.3, the peak downwind concentration is substantially below the DAC of 3 pCi cm^{-3} established in 10CFR20 Appendix B, and less than the permissible effluent concentration of 1 X 10^{-8} μCi/cm^3 for all meteorological conditions except the set of conditions with the lowest frequency of occurrence; for that stability classification, the instantaneous effluent concentration is slightly higher than the DAC.

A.2.5 Radiological Assessment of ^{16}N Sources

Nitrogen-16 is generated by the reaction of fast neutrons with oxygen and the only significant source results from reactions with oxygen in the liquid coolant of the reactor. The nuclide has a half-life of 7.13 s (decay constant $\lambda_{16} = 0.0972\ s^{-1} = 350\ h^{-1}$) and emits, predominantly, 6.13-MeV gamma rays. According to the McClellan AFB SAR, the effective cross section for the $^{16}O(n,p)^{16}N$ reaction, averaged over the fast-neutron energy spectrum in the TRIGA or over the fission-neutron spectrum is $\sigma_{np} = 2.1 \times 10^{-29} cm^2$.

The atomic density C_N (cm^{-3}) of the nuclide as it leaves the reactor core is given in terms of the oxygen density in water, $C_O = 3.34 \times 10^{22}$, as

$$C_N = \frac{\phi_f * C_0 * \sigma_{n,p}}{\lambda_{16}} * \left(1 - e^{-\lambda * t}\right) \tag{8}$$

where time in the core is represented by t. Fast-neutron flux varies linearly with reactor power. Time in core is a function of convection flow rate, a function of reactor power (see Chapter 4). As power increases, the rate of production increase from increased neutron flux is mitigated by a reduced time in the core from the increase in core cooling flow rate.

As the warmed coolant leaves the core, it passes through 1.5-in diameter ($A_{xp} = 11.4\ cm^2$) channels in the upper grid plate, but the triflute upper end fixture of the fuel element restricts the flow. This leaves a flow area for each element of:

$$A_0 \cong A_{xp} * \left[1 - \left(\frac{3}{\pi}\right) * \sin 30^\circ * \cos 30^\circ\right] = 6.69 cm^2 \tag{9}$$

Operation at power requires primary cooling; primary cooling enters the pool through a flow diverter approximately 2 feet (61 cm) above the core exit, 14 feet (427 cm) below the pool surface. Core exit is at 16 feet (488 cm) below the pool surface. The flow diverter induces mixing and avoids the direct rise from the core to the pool surface (which could otherwise occur through a chimney effect from core heating). A rough estimate of hydraulic diameter of the core exit (based on total flow area) is about 13 cm; calculations show the contributions to total dose rates at the pool surface are negligible at 160-200 cm below the surface of the pool, 22-25 times the hydraulic diameter of the exit into the pool. Exit flows are a small fraction of mixing flow, and under these conditions it is considered adequate to use a nuclide concentration reduced by the ratio of the total core exit surface area (approximately 555 cm^2 for 83 elements) and the pool (with a total surface area of approximately 30900 cm^2); mixing reduces the concentration of ^{16}N from the core exit by 0.018. Therefore, concentration of the radionuclide used in calculation is reduced from core exit by dilution.

Because of the short ½ life, the concentration of ^{16}N is also reduced by decay during transit. Since it is difficult to characterize flow velocity field from core exit to total mixing, flow rate from the core to the surface is conservatively assumed as core exit flow rate for dose rate calculations.

Dose rate calculations were modeled as a set of disk sources, each disk containing the appropriate volume source term multiplied times the difference between the disk locations. The appropriate volume source strength for each disk source calculation was modified by exponential decay of ^{16}N, with the time element calculated from core exit surface area, flow rate, and distance form the core exit. Does rate calculations were based on the two major emissions, 6.13 MeV (69%) and 7.11 MeV (5%). Total dose rate at each disk (where x is the distance form the disk to the pool surface) was therefore calculated as:

$$\dot{D} = \sum_E \left[\frac{k(E) * E * S_v \Delta d}{2} * \sum_i Ai * \left(E_1(\mu_i x) - \left(E_1(\mu_i x * \sec\theta)\right)\right) \right] \qquad (10)$$

Where:

- $k(E): \dfrac{R * h^{-1}}{MeV * cm^{-1} * s^{-1}}$
- $S_v = S_0(flow) * \exp\left(-\lambda * \dfrac{x * A_{channel}}{\dot{m}_{H_2O} * \rho_{H_2O}^{-1}} \right)$ (see Chapter 4 for coolant flow rate)
- A_i *Taylor buildup factor*
- μ_i *linear attenuation coefficient*, mod*ified by Taylor buildup factor* a_i
- $\theta = \arctan\left(\dfrac{x}{R_{pool}} \right)$

Parametric variation on the distance between the disk sources showed little improvement in convergence for separations smaller than 2 cm, and essentially no improvement below 1 cm; therefore ½ cm was used for final calculations. Locations of interest for dose calculations include 30 cm (1 ft) above the pool surface (i.e., pool surface monitor), waist high (approximately 130 cm/51 in. above the pool, 100 cm/39 in. above the bridge), and at the ceiling over the pool.(549 cm/18 ft above the pool).

Table A.4, Dose Rate (mR h^{-1}) Above Pool

KW	30 cm	130 cm	549 cm
50	0.5	0.2	0.0
100	3.5	1.1	1.3
200	15.8	4.7	0.5
300	35.7	10.5	1.2
400	60.0	17.5	1.9
500	87.2	25.3	2.8
750	166.3	47.7	5.2
1000	255.3	72.8	7.9
1250	347.9	98.8	10.6

Only a small proportion of the ^{16}N atoms present near the tank surface are actually transferred to the air of the reactor bay. Upon its formation, the ^{16}N recoil atom has various degrees of ionization. According to Mittl and Theys (1961) practically all ^{16}N combines with oxygen and hydrogen atoms in high purity water, and most combines in an anion form, which has a tendency to remain in the water. In this consideration, and in consideration of the very short half life of the nuclide, the occupational consequences of any airborne ^{16}N are deemed negligible in comparison to consequences from the shine from the reactor tank. Similarly, off-site radiological consequences from airborne ^{16}N are deemed negligible in comparison to those of ^{41}Ar.

A.3 Bibliography

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12. CONDUCT OF OPERATIONS

This chapter describes the conduct of operations at the KSU TRIGA Mark II Nuclear Reactor Facility. The conduct of operations involves the administrative aspects of facility operations, the facility emergency plan, the physical security plan, and the requalification plan. This chapter of the Safety Analysis Report forms the basis of Section 6 of the Technical Specifications (Chapter 14).

12.1 Organization

The operating license R-88, Docket 50-188, for the reactor is held by Kansas State University. Kansas State University is a land-grant institution governed by a Board of Regents (appointed by the Governor of the State). The Chief Executive Officer of the university is the President. The organizational structure (as shown in Fig. 12.1) identifies the President (a representative of the State of Kansas via the Board of Regents) as the licensee 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 management of the reactor.

The Department of Nuclear Engineering appoints a Nuclear Reactor Facility Manager (unclassified appointment, equivalent to a faculty appointment) for direct management of the reactor. The Nuclear Reactor Facility Manager delegates a Reactor Supervisor with the responsibility for direct supervision and coordination of daily operations (the Nuclear Reactor Facility Manager may hold these functions). The Nuclear Reactor Facility Manager and the Reactor Supervisor hold Senior Reactor Operator licenses issued by the USNRC. Additional licensed Reactor Operators (or Senior Reactor Operators) perform operations and maintenance functions under supervision of the Reactor Supervisor.

The Vice President for Administration and Finance is responsible for safety at the university. Kansas State University provides management and independent environment, safety and health oversight functions for the University, implemented though 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 University Police Department is responsible for law enforcement functions and institutional physical security. The University Police Department provides support for response to events. University Police Department is the primary interface for external agencies during response to events at the reactor facility.

The Department of Environmental Health is responsible for compliance issues related to environment, safety and health. To meet these requirements, the Division maintains a Radiation Safety Officer, Occupational Safety Manager, Hazardous Material Manager, and (fire) Safety and Security Officer. A partial listing of germane Department of Public Safety management functions include (but are not limited to):

- Environmental management (air, water, & waste)
- Ionizing & non-ionizing radiation
- Sanitation and water quality
- Fire prevention and emergency equipment
- Accident prevention and investigation
- Occupational safety and health
- Industrial hygiene and toxicology
- Indoor air quality
- Asbestos & PCB's (TSCA)
- Laboratory safety & chemical hygiene

A Reactor Safeguards Committee (composed of members of the Mechanical and Nuclear Engineering Department and other K-State faculty appointed by the President, the MNE chair, the University Radiation Safety Officer, and the Reactor Supervisor) performs the review and audit of nuclear operations for the President. The committee meets at frequencies specified in Technical Specifications. The Committee reports to the President, but also advises the Nuclear Reactor 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 President of Kansas State University, the Provost of Kansas State University, the Dean of the college of Engineering, to the operating unit (Department of Mechanical and Nuclear Engineering) and the reactor staff, including the Nuclear Reactor Facility Manager, the Reactor Supervisor and Reactor Operators.

12.1.1 Structure

As indicated on Figure 12.1, *Organization structure for the KSU TRIGA Mark II Nuclear Reactor Facility*, the K-State President is the licensee for the KSU Nuclear Reactor Facility. The reactor is under the direct control of the Nuclear Reactor Facility Manager, who reports through the academic administrative structure to the President.

Environment, safety and health oversight and expertise is provided through the Vice President for Administration and Finance, independent of facility line management. In addition to the reactor license, Kansas State University administers a broad radioactive material license. The Radiation Safety Officer is the University broad licensee, and manages radioactive material (i.e., byproduct and non-reactor special nuclear material) inventory and the University radiation safety program for ionizing radiation.

A University Radiation Safety Committee (reporting to the Vice President for Administration and Finance) maintains oversight and control of radiation protection functions for the University. Radiological controls for possession and use of radioactive materials at K-State in the University Radiation Protection Program are prepared and distributed by the University Radiation Safety Committee. The University Radiation Safety Committee has authorized the Nuclear Reactor Facility Manager to possess and transfer radioactive material under the State broad license.

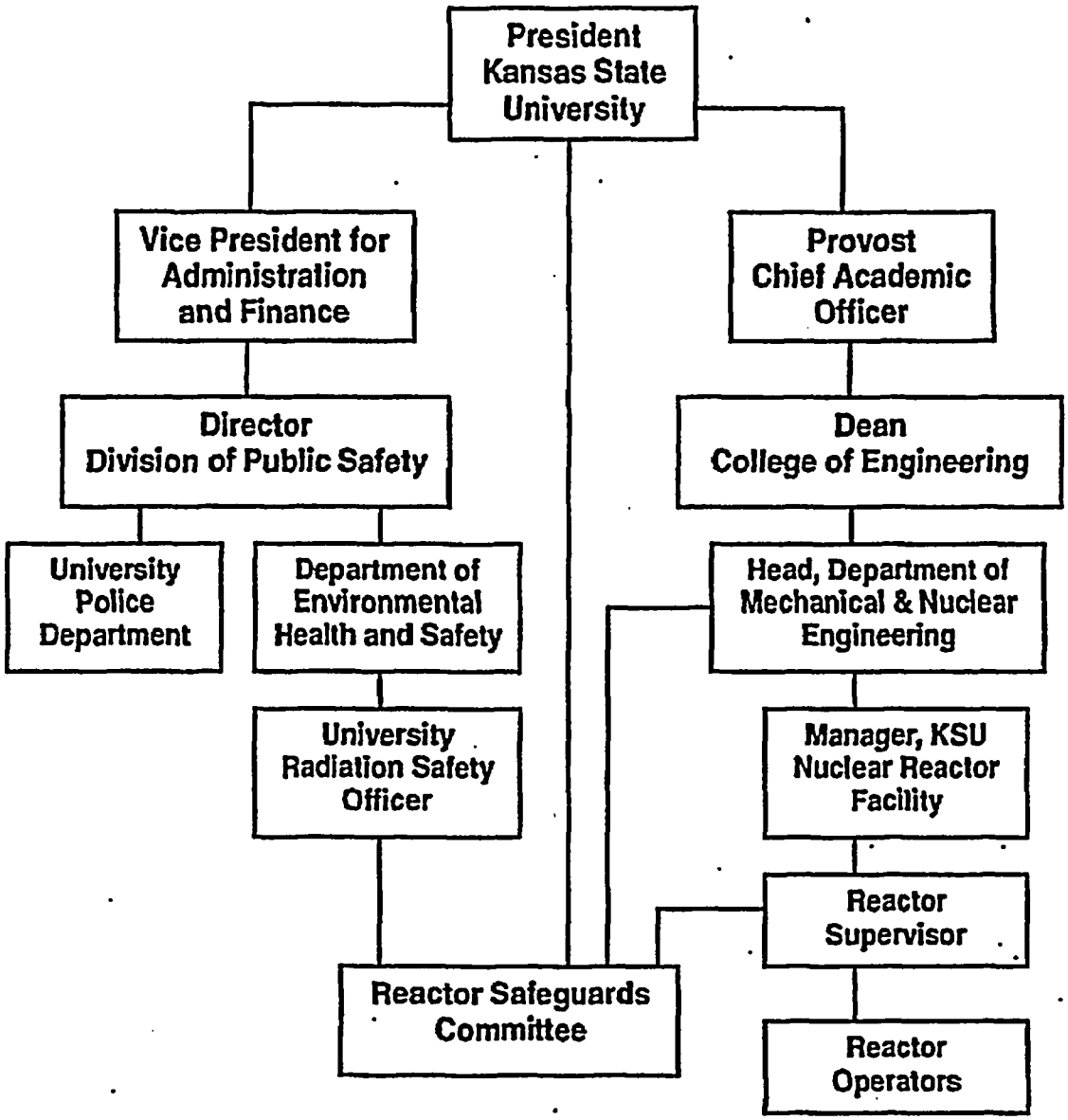

Figure 12.1: Organization and Management Structure for the K-State Reactor

University requirements and 10CFR requirements are combined in a comprehensive Reactor Radiation Protection Program. In accordance with the Reactor Radiation Protection Program, the reactor staff fulfills most routine radiation protection functions at the K-State reactor, with review and oversight by the Radiation Safety Officer. The 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

A description of responsibilities is provided in three categories. Responsibility for safe operation of the reactor is described in Section I. Independent environment, safety and health compliance and oversight is described in Section II. Advisory and oversight committees are described in Section III.

a. Reactor Operations Line Management

President:

As chief executive officer for the University, the President is responsible for safe operation of the reactor, protection of the health and safety of the public, and protection of the environment. The line of authority and responsibility for reactor operations extends through the Provost and Dean 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 Safety and Health.

Head of the Department of Mechanical and Nuclear Engineering

The Department Head is the appointment authority for the Nuclear Reactor 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 facilities. The Department Head is the Chair of the Reactor Safeguards Committee, which reports to the President, and which is responsible for approval of all plans and procedures for reactor operations and for audit of reactor operations and record keeping.

Nuclear Reactor Facility Manager

The Nuclear Reactor 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 Manager may hold academic and research responsibilities beyond those associated with the Reactor Facility. The Nuclear Reactor Facility Manager is authorized to delegate responsibility for operation and use of the reactor to the Reactor Supervisor.

Reactor Supervisor

The Reactor Supervisor has such duties, in regard to the operation of the reactor, as may be delegated by the Nuclear Reactor Facility Manager but whose nominal duties include reactor scheduling, responsibility for all records with regard to reactor operation as are required by appropriate federal licenses and regulations, laws and regulations of the State of Kansas and regulations of Kansas State University including the Kansas State University TRIGA MARK II Reactor Operations Manual.

The Reactor Supervisor is responsible for assuring that the reactor is operated only while a properly qualified and licensed Reactor Operator is present. The Reactor Supervisor is responsible for maintenance of a Reactor Operations Manual, the manual to include prescribed operating procedures for all routine modes of operation of the reactor, procedures for loading and unloading, start-up procedures, maintenance schedule, testing procedures, operational references, and other appropriate information as determined by the Reactor Supervisor and the Nuclear Reactor Facility Manager. The Reactor Supervisor is responsible for determining that the reactor is operated in strict accordance with the Operations Manual and the Facility License.

Reactor Operator

Operators (Reactor Operators and Senior Reactor Operators) report directly to the Reactor Supervisor and/or the Nuclear Reactor Facility Manager. Operators are responsible for knowing the status and condition of the facility, and ensuring that both personnel within the facility 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 Technical Specifications, operating procedures sand experiment procedures.

Operators are responsible for ensuring 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 10CFR55) manipulate controls under the direction of a licensed reactor operator or senior operator. Operation includes start-up, shutdown, routine instrumentation and control checkout, record keeping, 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 operating console, and a senior operator inside the reactor bay directing fuel operations.

b. 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. Kansas State University provides management and independent environment, safety and health oversight functions for the University. This responsibility is implemented though the Department of Environmental Safety and Health, 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.

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 Radiation Safety Officer, Occupational Safety Manager, Hazardous Material Manager, and Safety and Security Officer. A partial listing of germane Department of Public Safety management functions include (but are not limited to):

- Environmental management (air, water, & waste)
- Ionizing & non-ionizing radiation
- Sanitation and water quality
- Fire prevention and emergency equipment
- Accident prevention and investigation
- Occupational safety and health
- Industrial hygiene and toxicology
- Indoor air quality
- Asbestos & PCB's (TSCA)
- Laboratory safety & chemical hygiene

Radiation Safety Officer

The Radiation Safety Officer reports to the Manager of the Department of Environmental Health and Safety. The Radiation Safety Officer, or an authorized representative, shall be available (upon due notice) for advice and consultation regarding radiation surveys and radiation safety in connection with isotope production and radiation streaming problems as might arise in connection with reactor operation or experimentation. The Radiation Safety Officer is *ex officio* a member of the Kansas State University Radiation Safety Committee. The Radiation Safety Officer serves *ex officio* as a member of the Reactor Safeguards Committee, with any action (i.e., concerning potential radiation exposure or radioactive effluents) of the Committee requiring approval of the Radiation Safety Officer.

c. Principal Advisory and Oversight Committees

Reactor Safeguards Committee

The Reactor Safeguards Committee is composed of members appointed by the President of the university, upon the recommendation of the Chairman of the Committee, and ex-officio for specific positions. Composition and membership qualifications of the Committee are explicitly stated in Technical Specifications. The Reactor Safeguards Committee is responsible for approval of all plans and procedures for reactor operations and for audit of reactor operations and record keeping.

University Radiation Safety Committee

The University Radiation Safety Committee is an advisory committee for the Vice President for Administration and Finance and the Radiation safety Officer. Under authority of the Vice President of Administration and Finance, the Radiation Safety Committee authorizes conditions for use of radioactive material at K-State, and authorizes users (by name) to acquire and posses radioactive materials. The Reactor Radiation Protection Program incorporates requirements of the Radiation Safety Committee.

12.1.3 Staffing

Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator who is designated as Reactor Supervisor. The Supervisor shall be on call, within twenty minutes travel time to the facility, and cognizant of reactor operations.

Whenever the reactor is not secured, a (USNRC licensed) Reactor Operator (or Senior Reactor Operator) who meets requirements of the Operator Requalification Program shall be at the reactor control console, and directly responsible for control manipulations. A call list of Reactor Facility Personnel, management, and radiation safety personnel shall be available in the reactor control room for use by the Reactor Operator at the controls.

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

Only the Reactor Operator at the controls or personnel authorized by, and under 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 persons to manipulate reactivity controls who are training either as (1) a student enrolled in academic course making use of the reactor, (2) to qualify for an operator license, or (3) in accordance the approved Reactor Operator requalification program.

12.1.4 Selection and Training of Personnel

The KSU Reactor Facility maintains a training and selection program to prepare trainees for examination by the Nuclear Regulatory Commission in pursuit of operator or senior operator permits. Medical qualification for operator license program is described in Section 12.10.

Access is permitted to the reactor bay for personnel qualified for unescorted access or under the direct supervision of personnel qualified for unescorted access. Unescorted access qualification is granted after training and examination in knowledge and skills necessary to control personnel exposure to radiation associated with the operation of the KSU nuclear reactor. Training includes familiarization with salient sections of 10 CFR 19 Notices, Instructions and Reports to Workers, and Investigations, 10 CFR 20 Standards for Protection Against Radiation, the KSU Radiological Protection Program, and the KSU Reactor Emergency Plan. The KSU Radiological

Protection Program requires specific instruction in the risks of occupational exposure, the risks of prenatal exposure, provisos of 10CFR19 and 10CFR20, and a tour of the reactor facility.

12.1.5 Radiation Safety

The Radiation Protection Program for the Kansas State University TRIGA Mk II Nuclear Reactor Facility was prepared in response to the requirements of Title 10, Part 20.1101 (Code of Federal Regulations, 10CFR20). The Radiation Protection Program was developed following the guidance of the American National Standard *Radiation Protection at Research Reactor Facilities* and Regulatory Guides issued by the NRC. The Program also deals with radioactive materials regulated by the State of Kansas (an Agreement state) under license 38-C011-01. The goal of the Program is the limitation of radiation exposures and radioactivity releases to a level that is as low as reasonably achievable without seriously restricting operation of the Facility for purposes of education and research. The Program is executed in coordination with the Kansas State University Department of Environmental Safety and Health, Division of Public Safety, Radiation Safety Office. The program was been reviewed and approved by the Reactor Safeguards Committee for the Reactor Facility; the University Radiation Safety Committee has independently reviewed the program.

The reactor staff fulfills most of the functions described in the program. The Radiation Safety Officer maintains oversight of the facility by direct observation and review of specific activities, and by acting as a member of the Reactor Safeguards Committee and the University Radiation Safety Committee. Radiation Safety Officer approval is required for Reactor Safeguards Committee approval of any item under review. The Radiation Safety Officer is empowered to stop, in the interest of safety, any experiment involving radiation on the K-State campus. All campus radioactive materials users are required to eliminate any known unsafe practice or report the issue to the Radiation Safety Officer.

The details of the radiation safety program are described in Chapter 11.

12.2 Review and Audit Activities

Review and audit activities are oversight actions essential to the safe operation of the facility and the protection of the health and safety of the public. The Technical Specifications, 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 Nuclear Reactor Facility Manager, culminating in a semi-annual management audit of operations.

The Reactor Safeguards Committee holds oversight responsibility and authority. Oversight of the Manager's performance and review of managerial audits is the responsibility of the Safeguards Committee, evaluated formally at periodic intervals. In addition to periodic, scheduled reviews, the Reactor Safeguards Committee is available to conduct reviews on request. Review and approval of administrative controls, such as programs plans and procedures, is performed by the Reactor Safeguards Committee prior to implementation.

The Radiation Safety Officer conducts periodic laboratory safety audits, and participates in radiation surveillances on a periodic basis to review facility staff conduct of radiation surveys.

12.2.1 Reactor Safeguards Committee Composition and Qualifications

With the exception of *ex-officio* members, Reactor Safeguards Committee members are appointed by the President of the University, upon the recommendation of the Chairman of the Committee. Composition and membership qualifications of the Committee (as specified in the proposed Technical Specifications) provide expertise to evaluate reactor management, plant facilities, experimental programs, operating and experiment procedures, and radiological hazards.

The Head of the Department of Mechanical and Nuclear Engineering is the Committee Chair, and has the authority and responsibility to allocate resources that ensure safe reactor operations. The University Radiation Safety Officer is also an *ex officio* member of the Committee, with veto power over permissive Committee decisions. The Nuclear Reactor Facility Manager is also an *ex officio* member, non-voting, of the Committee. These *ex officio* Committee members (or designated alternates) are required to attend all meetings where permissive Committee decisions are made.

At least one Committee member shall be a Mechanical and Nuclear Engineering faculty member with expertise in reactor physics, nuclear engineering or nuclear science. At least one Committee member shall have expertise in chemistry, geology, or chemical engineering. At least one Committee member shall have expertise in the biological effects of radiation. One individual may have and represent expertise in more than one area, but the Committee shall consist of at least seven members.

a. Charter and Rules

The Committee is required to meet semi-annually, as a minimum. Specific review and audit activities are prescribed for these meetings, described in Sections 12.2.3 and 12.2.4. The Chair of the Committee or his designee may call additional meetings. At the discretion of the Chair or his designee, the Committee may be polled in lieu of a meeting; such a poll shall constitute Committee action subject to the same requirements as for an actual meeting.

Any permissive action of the Committee requires affirmative vote of the University Radiation Safety Officer as well as a majority vote of the members present. A quorum consists of not less than a majority of the full Committee, including *ex officio* voting members.

Minutes of meetings of the Reactor Safeguards Committee are distributed to the Dean of Engineering, the Provost, and the President of the University. Recorded affirmative votes on proposed new or revised experiments or procedures shall indicate that the Committee determines that proposed actions do not involve unreviewed safety questions, changes in the facility as designed, or changes in Technical Specifications, and could 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.

b. Review Function

The responsibilities of the Reactor Safeguards Committee shall include but are not limited to the following:

Review of proposed new or revised experiments.

Review of proposed new or revised procedures.

Review of proposed modifications of the reactor, the reactor bay, or the reactor control room.

Determination of whether items 1 through 3 involve unreviewed safety questions, changes in the facility as designed, or changes in Technical Specifications. The Nuclear Reactor Facility Manager may make this determination in the form of verifying an evaluation/determination.

Review of proposed revisions to Technical Specifications

Review of proposed changes to the Safety Analysis Report

Review and approval of audits of the Radiation Protection Program, the Physical Security Plan, and the Emergency Plan performed by the Nuclear Reactor Facility Manager.

Review of all operating anomalies and equipment failures.

Review of all reportable events.

Review of results of NRC inspections.

Review of critiques of emergency exercises.

Requalification of the Nuclear Reactor Facility Manager or Reactor Supervisor.[1]

c. Audit Function

The Reactor Safeguards Committee shall audit reactor operations and health physics during semi-annual inspections. The inspections shall 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.

12.3 Procedures

Written procedures shall be prepared and approved prior to initiating any of the activities listed in this section. The Nuclear Reactor Facility Manager and the Reactor Safeguards Committee shall

[1] It is the responsibility of the Safeguards Committee to reach a decision on the requalification of the licensed person administering examinations to other operators and senior operators.

approve the procedures. Under conditions specified by the Reactor Safeguards Committee, the Nuclear Reactor Facility Manager may make changes in procedures or experiments subject to validation by the Reactor Safeguards Committee. A periodic review of procedures will be performed and documented in a timely manner to assure they are current. Procedures shall be adequate to assure the safe operation of the reactor, but will not preclude the use of independent judgment and action, should the situation require. The following are actions that will typically require reviewed written procedures.

12.3.1 Reactor Operations

1. Startup, operation, and shutdown of the reactor
2. Fuel loading, unloading, and movement within the reactor.
3. Control rod removal or replacement.
4. Routine maintenance, testing, and calibration of control rod drives and other systems that could have an effect on reactor safety.
5. Administrative controls for operations, maintenance, conduct of experiments, and conduct of tours of the Reactor Facility.
6. Implementing procedures for the Emergency Plan or Physical Security Plan.

12.3.2 Health Physics

1. Testing and calibration of area radiation monitors, facility air monitors, and fixed and portable radiological surveillance instruments.
2. Conduct of radiological surveillance measurements.
3. Release of contaminated materials to the University Radiation Safety Office.
4. Accountability for special nuclear materials.

12.4 Required Actions

Two categories of required actions are addressed, violations of facility safety limits and reportable events.

12.4.1 Violation of Facility Safety Limit

In the event that a Safety Limit is not met,

a. The reactor shall be shutdown, and reactor operations secured.
b. The Reactor Supervisor and Nuclear Reactor Facility Manager shall be notified.

c. The safety limit violation shall be reported to the Nuclear Regulatory Commission within 24 hours by telephone, confirmed via written statement by email, fax or telegraph
d. A safety limit violation report shall be prepared within 14 days of the event to describe:
   (1) Applicable circumstances leading to the violation including (where known) cause and contributing factors
   (2) Effect of the violation on reactor facility components, systems, and structures
   (3) Effect of the violation on the health and safety of the personnel and the public
   (4) Corrective action taken to prevent recurrence
e. The Reactor Safety Review Committee shall review the report and any followup reports
f. The report and any followup reports shall be submitted to the Nuclear Regulatory Commission.
g. Operations shall not resume until the USNRC Manager of the Division of Reactor Licensing approves resumption.

12.4.2 Occurrences Reportable to the U.S. Nuclear Regulatory Committee

In the event of a reportable occurrence, as defined in the Technical Specifications, and in addition to the reporting requirements,

a. The Reactor Supervisor and Nuclear Reactor Facility Manager shall be notified
b. If a reactor shutdown is required, resumption of normal operations shall be authorized by the Nuclear Reactor Facility Manager
c. The event shall be reviewed by the Reactor Safeguards during a normally scheduled meeting

12.5 Reports to the Nuclear Regulatory Commission

All written reports shall be sent within prescribed intervals to the United States Nuclear Regulatory Commission, Washington, D.C., 20555, Attn: Document Control Desk.

All reports shall address (to the extent known or possible) the impact of the event on 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.

12.5.1 Immediate Notification

A report shall be made within 24 hours of discovery of a violation of safety limit or reportable occurrence by (1) telephone and (2) fax, telegraph or electronic mail to the NRC Operation Center.

12.5.2 14-Day Notification

A report shall be made within 14 days in writing to the NRC Operation Center for any violation of safety limit or reportable occurrence

12.5.3 Thirty-Day Notification

A report shall be made within 30 days in writing to the Manager, Non-Power Reactors and Decommissioning Project Manager, US. Nuclear Regulatory Commission, Washington, D.C, for:

a) Any permanent changes in Nuclear Reactor Facility Manager or Head of the Department of Mechanical and Nuclear Engineering
b) Any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during operation of the reactor;
b) Any significant change in the transient or accident analysis as described in the Safety Analysis Report.

12.5.4 Other Reports

A report is required within 60 days after criticality of the reactor in writing to the NRC Operation Center 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 values of the operating conditions or characteristics of the reactor under the new conditions.

A routine report is required to the US. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, within 60 days after completion of the first calendar year of operating and at intervals not to exceed 12 months, thereafter, providing the following information:

a) A narrative summary of reactor operating experience
b) The energy generated by the reactor (in megawatt-hours);
c) Unscheduled shutdowns, including corrective action taken to prevent recurrence
d) Major preventative and corrective maintenance with safety significance
e) Major changes in the reactor facility, including a summary of safety evaluations leading to the conclusions that no Unreviewed safety questions were involved

f) Major changes in procedures, including a summary of safety evaluations leading to the conclusions that no Unreviewed safety questions were involved

g) New tests or experiments (or both) that are significantly different from those previously performed and are not described in the Safety Analysis Report, including a summary of safety evaluations leading to the conclusions that no unreviewed safety questions were involved

h) A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of Kansas State University determined at or before the point of release or discharge.

The summary shall include, to the extent practicable, an estimate of individual radionuclides present in the effluent. If the estimated average release is less than 25% of the allowed or recommended value, a statement to this effect is sufficient.

i) A summarized result of environmental surveys performed outside the facility

j) A summary of radiation exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed or recommended.

12.6 Record Retention

There are three categories of record retention. General operating records (as noted) are required to be kept for five years. Records related to requalification are kept for the duration of the individuals' employment or for a complete training cycle. Records related to radiation (releases or exposure) are kept for the life of the facility.

12.6.1 Five-Year Retention Schedule

In addition to the requirements of applicable Code of Federal regulations (Title 10, Parts 20 and 50), records and logs shall be prepared and retained for a period of at least 5 years for the following items as a minimum.

a) Normal plant operation, including power levels;
b) Principal maintenance activities;
c) Reportable occurrences;
d) Equipment and component surveillance activities;
e) Experiments performed with the reactor;
f) All emergency reactor scrams, including reasons for emergency shutdowns.

12.6.2 Certification Cycle

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

12.6.3 Life-of-the-Facility Records

The following records shall be maintained for the life of the facility:

a) Gaseous and liquid radioactive effluents released to the environs
b) Offsite environmental monitoring surveys required by Technical Specifications
c) Fuel inventories and transfers
d) Facility radiation and contamination surveys
e) Radiation exposures for all personnel monitored
f) Corrected and as-built facility drawings

12.7 Emergency Planning

An emergency plan shall be established and followed in accordance with NRC regulations. The plan shall be reviewed and approved by the Reactor Safeguards Committee prior to its submission to the NRC. In addition, emergency procedures that have been reviewed and approved by the Reactor Safeguards Committee shall be established to cover all foreseeable emergency conditions potentially hazardous to persons within the Laboratory or to the public, including, but not limited to, those involving an uncontrolled reactor excursion or an uncontrolled release of radioactivity.

12.8 Security Planning

Administrative controls for protection of the reactor plant shall be established and followed in accordance with NRC regulations.

12.9 Operator Training and Requalification

The KSU Reactor Facility maintains a training and selection program to prepare trainees for examination by the Nuclear Regulatory Commission in pursuit of (senior) operator permits. Examinations are based on those of the Nuclear Regulatory Commission and include both written and practical tests. In preparation, trainees must satisfactorily complete study for the following areas:

1. Theory and operating principles
2. Operating characteristics
3. Instrumentation and control
4. Protection systems
5. Operating and emergency procedures
6. Radiation control and safety
7. Technical specifications
8. Title 10, Code of Federal Regulations

12.9.1 Requalification Program

The proposed Requalification Program follows a two-year cycle as of 1 Jan 1974. The program provides for operator medical certification, on the job training elements and proficiency, lectures, examinations, and records. The proposed Program identifies periodic and special requirements associated with medical certification, maintaining operational proficiency, operator examinations, training lectures, and records.

a. Medical Certification

The USNRC 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 operational errors endangering public health and safety.

The Nuclear Reactor Facility Manager has primary responsibility to assure medically qualified personnel are on-duty. Medical qualification of the Nuclear Reactor Facility Manager, if licensed, is the responsibility of the Chair of the Reactor Safeguards Committee.

The proposed Requalification Program identifies requirements to maintain medical certification that the licensed operator is medically qualified to operate the reactor, including annual reexamination and notifications of significant changes (should they occur).

b. Proficiency

During each two-year cycle, each licensed operator will maintain proficiency in reactivity manipulations by performing manipulations that demonstrate skill with reactivity control systems, as specified in the Requalification Program. Changes in facility design, operating procedures, facility license, and abnormal or emergency procedures will be documented and distributed to ensure all licensed operators are cognizant of facility conditions and requirements.

c. Examinations

The proposed 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 depth required to evaluate trainee understanding and capabilities. The program specifies that examinations should be based on evaluating knowledge, skills, and ability required to perform as a reactor operator/senior reactor operator, as appropriate.

Requirements for attending formal training lectures will be determined based on the results of annual written examinations administered to all licensed personnel, according to criteria specified in the Requalification Plan. The examinations will be prepared and graded by the Nuclear Reactor Facility Manager or the Reactor Supervisor.

Requirements for additional training will be determined based on the results of annual operational examinations covering normal, abnormal and emergency operating procedures (according to criteria specified in the Requalification Plan). The Nuclear Reactor Facility Manager or the Reactor Supervisor will prepare and administer the operational examinations, as specified in the Requalification Program.

c. **Lectures**

The proposed Requalification Program provides guidance for preparing training material based on objectives based on operational needs, i.e., objectives based on how the material relates to job performance.

d. **Records**

Records demonstrating successful participation the Reunification Program will be maintained as specified in the Program, which includes operator training record folders, records of Reactor Safeguards Committee reviews, operating logs, annual training records, and retention of operator biennial requalification records for one completed cycle.

12.10 Medical Certification of Licensed Operators and Senior Operators

The primary responsibility for assuring that medically qualified personnel are on-duty rests with the Nuclear Reactor Facility Manager. Medical qualification of the Nuclear Reactor Facility Manager, if licensed, is the responsibility of the Chair of the Reactor Safeguards Committee.

Licensed personnel should be examined biennially for continued medical qualification. Medical Examination Report Forms are supplied by the Facility and must be signed by the examining physician, with the physician's license number noted on the form.

The Nuclear Reactor Facility Manager reviews the medical report forms and makes a determination of whether the licensee (a) should be denied a license on medical grounds, (b) should have no license restrictions on medical grounds, or (c) should have certain specified restrictions. This information is entered on a Facility Medical Examiner Review Form and signed by the Nuclear Reactor Facility Manager. The Chair of the Reactor Safeguards Committee must approve the recommendation of the Manager, with signature on the same form.

The approved recommendations of the Nuclear Reactor Facility Manager are submitted to the Nuclear Regulatory Commission using NRC Form 396.

12.11 Bibliography

ANSI/ANS-15.1, "Development of Technical Specifications for Research Reactors," American National Standards Institute/American Nuclear Society, La Grange Park, Illinois, 1990.

ANSI/ANS-15.11 (Final Draft), "American National Standard Radiation Protection at Research Facilities," American Nuclear Society, La Grange Park, Illinois, October, 1992.

13. ACCIDENT ANALYSIS

This chapter provides information and analysis to demonstrate that the health and safety of the public and workers are protected in the event of equipment malfunctions or other abnormalities in reactor behavior. The analysis demonstrates that facility design features, limiting safety system settings, and limiting conditions for operation ensure that no credible accident could lead to unacceptable radiological consequences to people or the environment.

13.1 Accident Initiating Events and Scenarios

This chapter deals with analysis of abnormal operating conditions and consequent effects on safety to the reactor, the public, and operations personnel. Three conditions to be analyzed are:

- Loss of coolant
- Insertion of excess reactivity
- Fuel encapsulation failure — the maximum hypothetical accident (MHA)

These are the three conditions considered in the initial licensing of the Reactor Facility in 1962 for 100-kW steady-state operation and in the 1968 upgrade of the license permitting 250-kW steady state operation and 250-MW pulsing operation. The analysis presented here treats the same conditions, but for steady-state operation at 1,250 kW and pulsing operation to a $3.00 reactivity insertion, estimated peak power of 1,340 MW.

The maximum hypothetical accident for a TRIGA reactor is the failure of the encapsulation of one fuel element, in air, resulting in the release of gaseous fission products to the atmosphere. Failure in air could result from a fuel-handling accident or, possibly, failure in the event of a loss of reactor coolant. Failure under water, leading ultimately to atmospheric release of fission products, could possibly result from insertion of excess reactivity or operation with damaged fuel. This chapter addresses the several scenarios potentially leading to fuel failure, and then the potential consequences, should failure occur in air.

13.2 Accident Analysis and Determination of Consequences

13.2.1 Notation and Fuel Properties

Tables 13.1-13.3 identify physical characteristics of the TRIGA Mark II fuel. Table 13.4 identifies the assumptions and design basis values used in the accident analyses.

Table 13.1, Dimensions of TRIGA MkII $ZrH_{1.6}$ Fuel Elements.

Property of Individual Element	Symbol	Value
Length of fuel zone	L_f	0.381 m
Fuel radius	r_i	0.018161 m
Clad outside radius	r_o	0.018669 m
Fuel volume	V_f	0.000417 m^3
Clad volume	V_c	0.0000224 m^3
Fuel mass	M_f	2.5014 kg
Clad mass	M_c	0.1845 kg
Wt. Fraction U in fuel	x_u	[redacted]
Wt. Fraction $ZrH_{1.6}$ in fuel	x_m	[redacted]

Source: Training Manual, KSU TRIGA Nuclear Reactor Facility, 1998.

Table 13.2. Neutronic Properties of TRIGA MkII $ZrH_{1.6}$ Fuel Elements.

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

Source: West et al. (1967).

Table 13.3, Thermal and Mechanical Properties of TRIGA MkII $ZrH_{1.6}$ Fuel Elements and Type 304 Stainless Steel Cladding.

Property	Symbol	Value	Temp.
Fuel			
Density	ρ_f	5996 kg m^{-1}	
Thermal conductivity	k_f	18 W $m^{-1}K^{-1}$	All
Heat capacity, $c_{pf} = 340.1 + 0.6952T(°C)$	c_{pf}	340.1 J $kg^{-1}K^{-1}$	0 °C
Cladding			
Density	ρ_c	7900 kg m^{-1}	300 K
Thermal conductivity	k_c	14.9 W $m^{-1}K^{-1}$	300 K
		16.6	400 K
		19.8	600 K
Heat capacity	c_{pc}	477 J $kg^{-1}K^{-1}$	300 K
		515	400 K
Yield strength		250 Mpa	400 °C
Tensile strength		455 Mpa	400 °C

Source: fuel properties from Simnad (1980); cladding properties from Incropera and DeWitt (1990) and from Metals Handbook (1961).

Table 13.4, KSU TRIGA Core-Conditions Basis for Calculations.

Steady state maximum power, P_o	1.259 kW
Fuel mass per element	2.367 kg
Heat capacity per element at T (°C)	$805.0 + 1.646T$ (J K^{-1})
Minimum number of fuel elements, N	[redacted]
Core radial peaking factor	2
Axial peaking factor	$\pi/2$
Excess reactivity	\$4.00 (2.8% Δk/k)
Maximum pulsing reactivity insertion	\$3.00 (2.1% Δk/k)
Excess reactivity at 500 kW maximum power[a]	\$1.16 (0.81% Δk/k)
Fuel average temperature at 500 kW maximum power[a]	285 °C

[a]Source: Data from GA Torrey Pines TRIGA reactor

13.2.2 Loss of Reactor Coolant

Although total loss of reactor pool water is considered to be an extremely improbable event, calculations have been made to determine the maximum fuel temperature rise that could be expected to result from such an event taking place after long-term operation at full power of 500 kW. Limiting design basis parameters and values are addressed by Simnad (1980) as follows:

> *Fuel-moderator temperature is the basic limit of TRIGA reactor operation. This limit stems from the out-gassing of hydrogen from the ZrH_x and the subsequent stress produced in the fuel element clad material. The strength of the clad as a function of temperature can set the upper limit on the fuel temperature. A fuel temperature safety limit of 1150°C for pulsing, stainless steel U-$ZrH_{1.65}$ ... fuel is used as a design value to preclude the loss of clad integrity when the clad temperature is below 500°C. When clad temperatures can equal the fuel temperature, the fuel temperature limit is 950°C. There is also a steady-state operational fuel temperature design limit of 750°C based on consideration of irradiation- and fission-product-induced fuel growth and deformation....*

As this section demonstrates, even under extraordinarily conservative assumptions and approximations, the maximum fuel temperature reached in a loss of coolant accident is 290°C, well below any safety limit for TRIGA reactor fuel. Conservatism notwithstanding, the margin between computed temperature and design limits is sufficiently great to accommodate a design margin of at least a factor of two.

a. Initial Conditions, Assumptions, and Approximations

The following conditions establish an extremely conservative scenario for analysis of the loss of coolant accident.

- The reactor is assumed to have been operating for infinite time at power $P_o = 500$ kW at the time coolant is lost.
- Coolant loss is assumed to be instantaneous.
- Reactor scram is assumed to occur 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 N = [illegible] 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, i.e., cladding temperature is assumed to be equal to the temperature at the outside surface of the fuel matrix.

- Cooling of the fuel occurs via natural convection to air at inlet temperature T_i = 300°K. Radiative cooling and conduction to the grid plates are neglected.

- Heat transfer in the fuel is one dimensional, i.e., axial conduction is neglected, and fuel is assumed to be uniform in thermophysical properties.

- Heat transfer in the fuel is treated on the basis of pseudo-steady-state behavior, i.e., at any one instant, heat transfer is described by steady-state conduction and convection equations.[1]

b. *Core Geometry*

The following data on core geometry are derived from the KSU TRIGA Mechanical Maintenance and Operating Manual (1962). The core contains 90 fuel positions in five circular rings (B - F), plus the central thimble (A ring). The upper grid plate is 0.495 m in diameter and 0.019 m thick. Holes to position the fuel are 0.03823 m diameter and the central thimble is very slightly larger in diameter, 0.0384 m.

Cooling water passes through the differential area between the triangular spacer block on the top of each fuel element and the round holes in the upper grid plate. The nominal diametral clearance between the tips of the spacer blocks and the grid plate is approximately 0.001 m.

The lower grid plate is 0.405 m diameter, with 36 holes, 0.0159 m diameter, for water flow. However, the bulk of the water flow is through the annular space provided between the top of the lower grid plate and the bottom of the reflector. The radial reflector is D_r = 0.457 m inside diameter and 0.559 m height.

The effective hydraulic diameter for flow through the core, with an experiment in place in the central thimble, is given by

$$D_h = \frac{4 * A_{flow}}{P_{wet}} = \frac{4*(\frac{\pi * D_r^2}{4} - 91*\pi*r^2)}{\pi * D_r + 91*2*\pi*r_0} = 0.02127m \quad \text{(Equation 13.2.2.1)}$$

[1] See Todreas & Kazemi (1990) or El-Wakil (1971) for steady-state conduction equations.

If, in thermal-hydraulic calculations, one approximates conditions as flow through an annular section around any one fuel element, the outer radius of the annulus, say r_a, is given by

$$D_h = \frac{4\pi(r_a^2 - r_o^2)}{2\pi r_o}, \text{ or}$$

$$r_a = \sqrt{D_h r_o / 2 + r_o^2} = 0.02339 \text{ m}, \qquad \text{(Equation 13.2.2.2)}$$

The flow area A_c per fuel rod is $\pi(r_a^2 - r_o^2) = \pi r_o D_h / 2 = 0.0006238 \text{ m}^2$.

The total length of a fuel rod is 0.7206 m (28.37 in.), of which the length of the fuel matrix, the heated length, is $L_f = 0.3810$ (15 in). The lengths of upper and lower axial reflectors, L_u and L_l, are each 0.0874 m (3.44 in). Beneath the lower reflector is a bottom end fixture of length L_l about 0.0824 m (3.245 in.). Above the upper reflector is a triangular spacer of length L_t about 0.0191 m (0.75 in.) and an upper end fitting of length L_o about 0.0634 m (2.495 in.). The zone between grid plates is $L_e = L_l + L_t + L_f + L_u = 0.6382$ m.

c. **Decay Power**

The time dependence of the thermal power in the core as a function of time after shutdown is based on calculations by the CINDER code [England et al. 1976] as reported by George, LaBauve, and England [1980, 1982]. Sample results are presented in Figure 13.1 and Table 13.5 as the function $R(t)$ defined as the ratio of the thermal power $P_d(t)$ from gamma ray and beta particle decay at time t after shutdown to the steady power P_o prior to shutdown, based on 200 MeV energy release per fission.

For the purpose of this analysis, the time dependence function for 1 to 10^6 s may be approximated as

$$R(t) = \frac{0.04856 + 0.1189x - 0.0103x^2 + 0.000228x^3}{1 + 2.5481x - 0.19632x^2 + 0.05417x^3}, \qquad \text{(Equation 13.2.2.3)}$$

in which x is the natural log of the time after shutdown, in seconds. Time dependence of the thermal source in the worst-case element is given by

$$P_d(t) = \frac{2P_o}{N} R(t), \qquad \text{(Equation 13.2.2.4)}$$

in which the worst case element generates twice the power as the core average, P_o is the total-core thermal power prior to shutdown (1,250 kW), and N [83] is the minimum number of fuel elements required for operation.

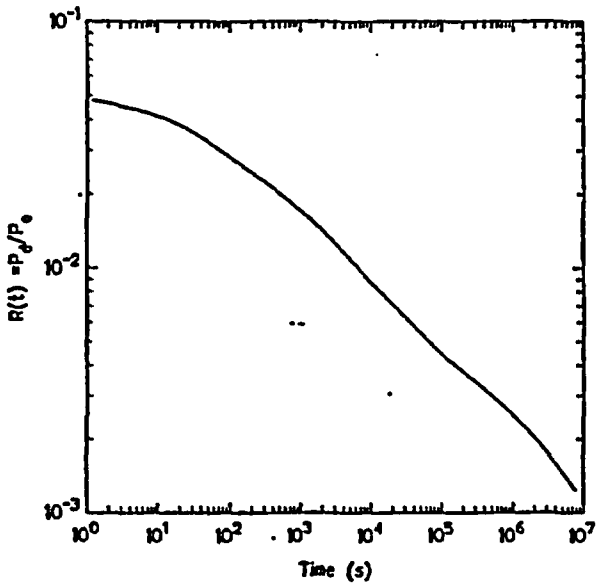

Figure 13.1. Decay Heat Function $R(t)$.

Table 13.5, Decay Heat Function for Thermal Fission of ^{235}U.

Time t (s)	$R(t) = P_d(t)/P_o$
0	0.0526
1 (10^0)	0.0486
10 (10^1)	0.0418
100 (10^2)	0.0282
1000 (10^3)	0.0172
10,000 (10^4)	0.0087
100,000 (10^{45})	0.0044
1,000,000 (10^6)	0.0025

d. Maximum Air Temperature

The fundamental relationships between buoyancy driven differential pressure and pressure losses from friction provide an independent verification for results of the previous calculations.

Buoyancy Driven Pressure Difference

The total mass flow rate w (kg/s) associated with the worst-case fuel element is determined by a balance between the buoyancy driven pressure difference vertically across the core and the frictional pressure loss within the core, which is discussed in the next section. The temperature rise across the core is

$$\Delta T_m(t) = T_o(t) - T_i = \frac{P_d(t)}{wc_{pm}^{air}} = \frac{2P_oR(t)}{Nwc_{pm}^{air}}, \qquad \text{(Equation 13.2.2.5)}$$

in which the heat capacity of the air evaluated at the inlet air temperature. At time zero, for example,

$$\Delta T_m(0) = T_o(0) - T_i \cong 0.6153 / w. \qquad \text{(Equation 13.2.2.6)}$$

Air inlet temperature T_i is assumed to remain constant at 27°C. Suppose ρ_i and ρ_o are respectively the densities of air at the inlet and outlet temperatures.[2] Suppose further that the effective chimney height is H. The chimney height is the distance between the center of the zone in which the air is heated and the center of the zone in which the air is cooled. Evaluation of the latter is difficult to determine because of uncertainties in mixing of the air after it leaves the upper grid plate. Here we follow the lead of the UT SAR (1991) and choose 10 hydraulic diameters as the effective distance. Thus, H is given by $L_f/2 + L_t + 10D_h = 0.422$ m. and the buoyancy pressure difference is given by

$$\Delta p_b = (\rho_i - \rho_0) * g * H \qquad \text{(Equation 13.2.2.7)}$$

in which g is the acceleration of gravity, 9.8 m s^{-2}. Since

$$\rho_i - \rho_0 \cong \frac{353 * (T_o - T_i)}{T_i^2}, \qquad \text{(Equation 13.2.2.8)}$$

and $T_i = 300\ K$, it follows from Eqs. (13.2.2-5) and (13.2.2-8) that

$$\Delta p_b(t) = 0.190 R(t) / w, \qquad \text{(Equation 13.2.2.9)}$$

Frictional Pressure Difference

In this calculation, only frictional losses within the core, computed on the basis of the equivalent annulus model, are accounted for. Based on air inlet density and an air mass flow rate per fuel rod of w, the frictional pressure difference is given by

$$\Delta p_f = \frac{L_c f w^2}{2\rho_i D_h A_c^2}. \qquad \text{(Equation 13.2.2.10)}$$

The laminar-flow (Moody) friction factor f for the equivalent annulus model, with $r_o / r_p = 1.193$ is given by Sparrow and Loeffler (1959) as

$$f = 100 / \mathrm{Re}, \qquad \text{(Equation 13.2.2.11)}$$

Re, the Reynolds number, is given by $D_h w / \mu_i A_c$, and μ_i is the dynamic viscosity of the air at the inlet temperature.[3] Equation (13.2.2-10) may be rewritten as

[2] Density at 1 atm, for air as an ideal gas, is given by ρ (kg/m^3) = 353.0/T(°K). Heat capacity, from 300 to 700 °K is 1030 J/kgK ± 3% (Incropera and DeWitt, 1990).

[3] Dynamic viscosity, over the range 250 - 1000 °K is given by $10^7\mu$ (Ns/m^2) = -106.2941 + 16.81986[T(°K)]$^{1/2}$ or $\mu = 1.85 \times 10^{-5}$ Ns/m^2 at 300 °K. (Incropera and DeWitt, 1990).

$$\Delta p_f = \frac{0.1416 L_c \mu_i T_i w}{D_h^2 A_c} = 1780 w, \qquad \text{(Equation 13.2.2.12)}$$

Equating the frictional pressure drop with the buoyancy pressure driving force, using Equations (13.2.2-9) and (13.2.2-12),

$$w = 0.0103\sqrt{R(t)}, \qquad \text{(Equation 13.2.2.13)}$$

or

$$T_o(t) - T_i = \Delta T(t) = 1140\sqrt{R(t)}. \qquad \text{(Equation 13.2.2.14)}$$

Results

For R(t) at time zero of 0.0526, maximum air temperature rise above 300 K is 261 K.

e. Fuel and Cladding Temperature Distribution

With design power P_o = 1,250 kW, a factor of two radial peak to average power, and a fuel surface area in the heated zone equal to $A_f = 2\pi r_o L_f$ = 0.04469 m^2, the worst-case average heat flux at post-accident time t in the heated zone is:

$$q''_{avg} = 2 * 1.25e6 * R(t) / A_f N = 3.37e5 \text{ W/m}^2. \qquad \text{(13.2.2-15)}$$

With the conservative approximation that the axial variation of heat flux is sinusoidal, the local value of the heat flux (W/m^2) is given by:

$$q''(z) = q''_{max} \sin(\pi / L_f), \qquad \text{(13.2.2-16)}$$

in which z is the distance along the fuel channel, measured from the inlet and $q''_{max} = (\pi/2) q''_{avg} = 4.235 \times 10^5 R(t)$ W/m^2. Similarly, the local value of the air temperature in the coolant channel is given by

$$T_{air}(z) = T_i + \frac{0.6153 R(t)}{2}\left[1 - \cos(\pi / L_f)\right]. \qquad \text{(13.2.2-17)}$$

According to Dwyer and Berry (1970), the Nusselt number for laminar flow in a cooling channel is approximately Nu = 4.24. The corresponding heat transfer coefficient is

$$h = \frac{k_{air} \text{Nu}}{D_h}. \qquad \text{(13.2.2-18)}$$

By using as an approximation the air thermal conductivity[4] of 26.3 W/mK at 300°K, one computes h = 5240 W/m^{2}K, and the cladding surface temperature

[4] For the range 200 to 1000°K, data of Incropera and DeWitt (1990) is very well fit by the formula $k_{air} = -22.055 + 2.8057\sqrt{T(°K)}$ in units of W/mK.

$$T_{clad}(z) = T_{air}(z) + q''(z)/h. \qquad (13.2.2\text{-}19)$$

By using the fuel thermal conductivity k_f = 18 W/mK, and neglecting the temperature drop across the cladding, one computes the fuel centerline temperature as

$$T_{fuel}(z) = T_{clad}(z) + \frac{q''(z) r_o}{2k_f}. \qquad (13.2.2\text{-}20)$$

Fuel and cladding temperatures are reported in Table 13.6 and illustrated in Figure 13.2 for the case of zero time post accident. This is based on three conservative assumptions: equilibrium fission product buildup at full power, instantaneous loss of coolant when the reactor scram occurs, and equilibrium temperature based on initial decay heat production.

Table 13.6, Post-Accident Fuel and Cladding Temperatures.

z/L_f	q'' (W/m^2)	T_{air} (°K)	T_{clad} (°K)	T_{fuel} (°K)
0.00	0	300	300	300
0.10	17209	306	309	318
0.20	32733	325	331	348
0.30	45053	354	362	385
0.40	52963	390	400	427
0.50	55688	431	441	470
0.60	52963	471	481	508
0.70	45053	508	516	539
0.80	32733	536	542	559
0.90	17209	555	558	567
1.00	0	561	561	561

The K-State reactor does not operate 24 hours per day, 7 days per week, so that the total inventory of fission products is significantly lower than the assumed value. During actual loss of coolant, overall heat transfer coefficient will be based on water rather than air. There is at least 16 feet of water over the core that has to drain before the core is uncovered. Experiments reported in GA-6596 "Simulated Loss-of-Coolant Accident for TRIGA Reactors" (General Atomics, August 18, 1965) demonstrated that with a constant and continuous heat production, temperature rises to about 50% of the equilibrium temperature in approximately 30 minutes (1800 seconds). At 10^3 seconds, R(t) is 0.0172, 33% of the heat production following shutdown. Equilibration takes a significant amount of time, while heat production is decaying.

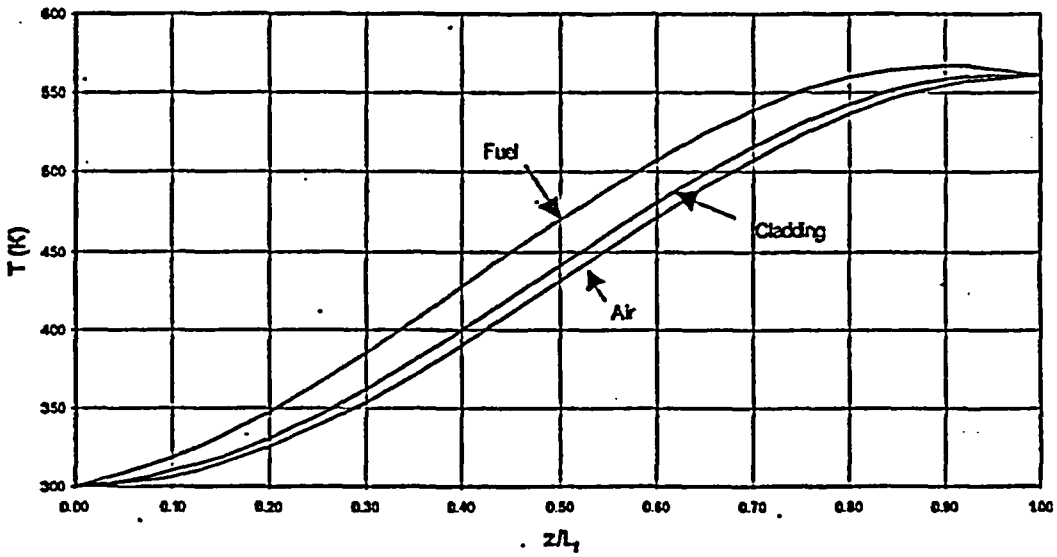

Figure 4.13, Axial Variation of fuel, cladding, and air temperature immediately following a loss of coolant accident, with equilibrium fission product heating.

f. Radiation Levels from the Uncovered Core

Although there is only a very remote possibility that the primary coolant and reactor shielding water will be totally lost, direct and scattered dose rates from an uncovered core following 1,250 kW operations have been calculated.

This section describes calculations of on-site and off-site radiological consequences of the loss-of-coolant accident. Extremely conservative assumptions are made in the calculations, namely, operation at 1,250 kW for one year followed by instant and simultaneous shutdown and loss of coolant. The reactor core is homogenized and the ORIGEN-2 [CCC-371] is used to define gamma-ray source strengths, by energy group. Radiation transport calculations are performed using the MCNP code (Briesmeister 1997), with doses evaluated at the 22-ft elevation of the reactor (one meter above the operating deck), in the reactor bay at the 0-foot elevation (one meter above the floor, 30-cm from the outer wall, and at one-meter above ground level at the site boundary and at points beyond the site boundary extending to 100 m radial distance from the core.

Modeling of the reactor core for radiation transport calculations is based on the following approximations. The TRIGA reactor core is approximated as a right circular cylinder 0.4572 m diameter (OD of F ring). The fuel region is 0.381 m (15 in.) high. On each end axially is a graphite zone 0.0874 m (3.44 in.) high and an aluminum grid plate 0.0191 m (0.75 in.) thick. In [redacted] fuel locations, there are [redacted] fuel elements, 3 standard control rods and 1 transient control rod, 1 void location, 1 central thimble (void), 1 source (assume void), and 1 pneumatic transfer site (assume void). The fuel region may is treated as a homogeneous zone, as are the axial graphite zones and the grid plates.

Fuel elements are 1.43-in. ID and 1.47-in OD, clad with type 304 stainless steel[5]. Fuel density is 5996 kg/m^3. Fuel composition is [redacted]% uranium, [redacted]% $ZrH_{1.65}$. The uranium is [redacted]% ^{235}U and [redacted]% ^{238}U. Steel density is 7900 kg/m^3. Standard control rods are 0.875-in. OD, the transient rod 1.25-in. OD. Both types of rods are clad with 30-mil thick aluminum (2700 kg/m^3 density). The control material may be approximated as pure graphite, with density 1700 kg/m^3.

In radiation transport calculations, the core is modeled conservatively as a central homogenous fuel zone (air density neglected) bounded on either end by a homogeneous axial reflector zone; and by a 0.75-in. thick aluminum grid plate, treated as a homogeneous solid. Densities of the homogenous zones are as follow:

Fuel	3602 kg/m^3
Reflector	1147 kg/m^3
Grid Plate	2700 kg/m^3

Composition of the three zones, by weight fraction, are given in Table 13.7.

Table 13.7, Compositions of Homogenized Core Zones.

Element	Mass Fraction	Element	Mass Fraction
Fuel Zone		*Axial Reflector Zone*	
C	0.0617	C	0.7920
Al	0.0010	Al	0.0033
H	0.0139	Mn	0.0041
Zr	0.7841	Cr	0.0368
Mn	0.0013	Ni	0.0164
Cr	0.0117	Fe	0.1474
Ni	0.0052		
Fe	0.0469	*Grid Plate*	
U	0.0741	Al	1.0000

The reactor bay is approximated as a right circular cylinder 36 ft (10.973 m) high and 36.68 ft (11.18 m) radius. The reactor vessel structure is approximated as a right circular cylinder, co-axial with the bay, 22 ft (6.706 m) high and 11 ft (3.3528 m) radius. The free volume is 144,000 ft^3 (4078 m^3). The site boundary, at its nearest approach to the reactor bay, is about 2 m beyond the bay boundary, that is, at a radius of 13.13 m from the center of the reactor.

Gamma-ray source strengths vs. times after shutdown are listed in Table 13.8 for an operating time of one year and a thermal power of 1,250 kW. The source strengths are gamma rays per second, by group, for the entire reactor core. In the MCNP calculations, the source is assumed to be uniformly distributed within the core.

The roof of the reactor bay is modeled as a concrete slab 10 cm thick and with density 2.35 g/cm^3. In fact, the roof is a composite structure, so reflection from the concrete slab conservatively models gamma-ray transport and dose rates at the site boundary.

[5] Composition, by weight, 2% Mn, 18% Cr, 8% Ni, balance Fe.

Table 13.9 reveals substantial dose rates above the void reactor tank with long-term 500-kW operation. One day after loss of coolant, the dose rate is 17 Sv/h (1700 rem/h). However, dose rates inside the reactor bay at the 0-ft. and 12-ft. elevations are only about 4 rem/h immediately after coolant loss, declining to about 1.5 rem/h after 1 hour and 100 mrem/h after 30 days. These dose rates show that personnel could occupy the reactor bay shortly after the loss of coolant to undertake mitigating actions without exceeding NRC occupational dose limits.

Dose rates at other receptor locations are shown in Table 13.9 as indicators of areas that should be checked following a loss of reactor pool water event. These dose rates are unrealistically conservative in view of the assumption of one-year operation at full power (not possible) and scattering from a thick concrete roof. Dose rates at the site boundary (facility fence) decline from about 3 rem/h immediately after the loss of coolant to 350 mrem/h after 1 day. Kansas State University has complete control over access to campus locations within the zones defined by receptor locations in the analysis.

Table 13.8,. Full Core Gamma-Ray Sources Strengths (Number Per Second) Following Operation for One Year at 1,250 kW Thermal Power.

	Time after shutdown				
E (MeV)	0	1 h	24 h	30 days	180 days
1.00E-02	8.72E+15	2.92E+15	1.05E+15	2.43E+14	6.20E+13
2.50E-02	2.09E+15	7.50E+14	3.04E+14	5.97E+13	1.32E+13
3.75E-02	1.74E+15	7.37E+14	3.97E+14	6.72E+13	1.45E+13
5.75E-02	1.81E+15	5.50E+14	1.77E+14	4.15E+13	1.20E+13
8.50E-02	1.50E+15	5.20E+14	2.33E+14	3.21E+13	8.39E+12
1.25E-01	1.50E+15	7.65E+14	4.81E+14	8.98E+13	1.31E+13
2.25E-01	3.11E+15	1.03E+15	4.44E+14	2.42E+13	6.83E+12
3.75E-01	1.94E+15	5.72E+14	2.20E+14	3.81E+13	3.41E+12
5.75E-01	3.25E+15	1.71E+15	7.54E+14	1.01E+14	1.08E+13
8.50E-01	4.16E+15	2.26E+15	8.94E+14	4.51E+14	1.15E+14
1.25E+00	2.22E+15	8.15E+14	7.83E+13	1.62E+12	6.01E+11
1.75E+00	9.30E+14	5.14E+14	2.23E+14	4.78E+13	7.72E+10
2.25E+00	5.00E+14	2.19E+14	6.69E+12	1.12E+12	3.72E+11
2.75E+00	1.99E+14	7.32E+13	8.13E+12	1.82E+12	1.55E+09
3.50E+00	1.16E+14	1.72E+13	7.12E+10	1.50E+10	9.37E+07
5.00E+00	6.23E+13	2.42E+11	6.96E+08	4.67E+00	4.63E+00
7.00E+00	5.05E+11	5.27E-01	5.27E-01	5.26E-01	5.22E-01
9.50E+00	9.49E+07	5.98E-02	5.98E-02	5.98E-02	5.93E-02
Total	3.39E+16	1.35E+16	5.27E+15	1.20E+15	2.61E+14
MeV/s	1.43E+16	6.28E+15	2.02E+15	5.77E+14	1.13E+14

Dose rates vs. post accident times are listed in Table 13.9 for a number of receiver locations. Locations are specified at radial distances from the center of the reactor bay. The22-foot level provides direct access to the reactor pool, the 12-foot level access to valves controlling water flow to the reactor pool. The 0-foot level provides access to valves and pumps, which allow water to be added to the reactor pool. The 13-meter distance marks a zone defined by the fence surrounding the reactor bay.

g. Conclusions

Although a loss of pool water is considered to be an extremely improbable event, calculations show the maximum fuel temperature that could be expected to result from such an event (after long-term operation at full power of 1,250 kW) is 294°C, well below any safety limit for TRIGA reactor fuel.

Maximum possible dose rates resulting from a complete loss of pool water permit mitigating actions. The area surrounding the reactor is under control of the Kansas State University, and exposures outside the reactor bay environment can be limited by controlling access appropriately. Kansas State University has complete authority to control access to campus locations.

Table 13.9, Gamma-Ray Ambient (Deep) Dose Rates (R/h) at Selected Locations for Times Following Loss of Coolant After Operation for One Year at 1,250 kW Thermal Power.

	Time post accident				
	0	1 h	24 h	30 d	180 d
On-site (elev.)					
22 ft. (center)	3.75E+06	1.38E+06	4.25E+05	1.03E+05	1.98E+04
12 ft. (boundary)	1.08E+03	4.00E+02	1.43E+02	3.25E+01	7.00E+00
0 ft. (boundary	9.75E+02	3.75E+02	1.23E+02	3.00E+01	6.25E+00
Off-site (radius from center of reactor bay)					
13 m	6.75E+02	2.75E+02	8.75E+01	2.15E+01	4.50E+00
15 m	5.00E+02	2.00E+02	6.75E+01	1.60E+01	3.25E+00
20 m	2.50E+02	1.00E+02	3.50E+01	8.75E+00	1.73E+00
30 m	8.25E+01	3.25E+01	1.10E+01	2.75E+00	5.50E-01
40 m	3.75E+01	1.40E+01	5.25E+00	1.18E+00	2.33E-01
50 m	1.93E+01	7.25E+00	2.43E+00	6.25E-01	1.23E-01
70 m	6.50E+00	2.75E+00	9.00E-01	2.18E-01	4.00E-02
100 m	2.35E+00	9.50E-01	2.75E-01	6.75E-02	1.33E-02

13.2.3 Insertion of Excess Reactivity

Rapid compensation of a reactivity insertion is the distinguishing design feature of the TRIGA reactor. Characteristics of a slow (ramp) reactivity insertion are less severe than a rapid transient since temperature feedback will occur rapidly enough to limit the maximum power achieved during the transient. Analyses of plausible accident scenarios reveal no challenges to safety limits for the TRIGA. The fuel-integrity safety limit, according to Simnad (1980), may be stated as follows:

> *Fuel-moderator temperature is the basic limit of TRIGA reactor operation. This limit stems from the out-gassing of hydrogen from the ZrH_x and the subsequent stress produced in the fuel element clad material. The strength of the clad as a function of temperature can set the upper limit on the fuel temperature. A fuel temperature safety limit of 1150°C for pulsing, stainless steel U-$ZrH_{1.65}$ ... fuel is used as a design value to preclude the loss of clad integrity when the clad temperature is below 500°C. When clad temperatures can equal the fuel temperature, the fuel temperature limit is 950°C. ....*

Two reactivity accident scenarios are presented. The first is the insertion of 2.1% reactivity at zero power by sudden removal of a control rod. The second is the sudden removal of the same reactivity with the core operating at the maximum power level permitted by the balance of the core excess reactivity (i.e., core excess less $3.00). Movements of control rods for the first case are controlled, in part, administratively, while movements for the second are prevented by control circuit design.

As the analysis shows, in neither scenario does the peak fuel temperature approach the temperature limit. The nearest approach is 869°C, incurred by a pulse insertion of 0.7% while the reactor is operating at a steady power of 94 kW, an action prevented both by administrative requirements and by interlocks.

a. Initial Conditions, Assumptions, and Approximations

The following conditions establish an extremely conservative scenario for analysis of insertion of excess reactivity.

- The reactor operates with a minimum of N = [redacted] fuel rods.
- Reactor and coolant ambient (zero power) temperature is 27°C.
- Maximum reactivity insertion for pulsing or for the worth of experiments is set at $3.00, $\delta k_{max} = 2.1\%$ or $\rho_{max} = 0.021$.
- Reactor power equivalent to the core excess reactivity of $1.00, i.e., $\delta k = \delta k_{ex} - \delta k_{max} = 0.7\%$ ($\rho = 0.007$) is $P_o = .107$ kW and the maximum fuel temperature at that power is $T_o = 150°C$. Basis: Data for the Torrey Pines TRIGA, as included in the KSU TRIGA Operations Manual.
- A control rod interlock preventing pulsing operations from power levels greater than a maximum of 10 kW is not credited
- Conservative hot channel factors as calculated in 4.5.3 are used

b. Computational Model for Power Excursions

The following relationships are for the Fuchs-Nordheim model, modified by Scalletar, for power excursions, as described for the TRIGA reactor by West et al. (1967).

If the heat capacity of the fuel is given by , $c_{pf} = 340.1 + 0.6952T(°C)$ (J/kg°K), and there are N fuel elements, each of mass $m_f = 2.367$ kg, then the overall core heat capacity is given by

$$K(\mathrm{J/K^\circ}) = m_f N c_{pf} = C_o + C_1 T(^\circ C) \qquad (13.2.3\text{-}1)$$

in which $C_o = 6.682 \times 10^4$ and $C_1 = 136.6$. If P_o (W) is the reactor power at the initiation of the pulse power excursion, and ρ_o is the magnitude of an initiating step change in reactivity, then the maximum power increase (W) is given by

$$P_{max} - P_o = \frac{(\rho-\beta)^2 C_o}{2\alpha\ell} + \frac{(\rho-\beta)^3 C_1}{6\alpha^2\ell}. \qquad (13.2.3\text{-}2)$$

A maximum pulse of \$3.00 would result in a power rise of approximately 1430 MW(t). If T_o is the average core temperature at the start of the excursion, the maximum temperature rise (°K) is given by

$$T_{max} - T_0 = \left[\frac{2*(\rho-\beta)}{\alpha}\right]*\left[-\frac{3}{8}*(\sigma-1) \pm \frac{3}{8}*\sqrt{(\sigma-1)^2 + \frac{16}{3}*\sigma}\right] \qquad (13.2.3\text{-}3)$$

in which

$$\sigma = \frac{\alpha C_o}{(\rho-\beta)C_1}, \qquad (13.2.3\text{-}4)$$

and the "+" sign applies when $\sigma > 1$. Although there are nonlinear terms in the model, calculation of temperature change as a function of temperature shows a nearly linear response. Therefore the major factor in determining core-average peak temperature is the amount of reactivity available to pulse. To remain within 1000°C at all locations, core average temperature cannot exceed 318°C (based on Table 13.2.1.4 peaking factors).

The maximum worth of the pulse rod is \$3.00, therefore peak temperature following a pulse was calculated based on a maximum reactivity available from the pulse rod at reference core temperature (27°C). The core-average temperature change is 229°C, with a hot spot change of 716°C (based on Table 13.2.1.4 peaking factors). Therefore, core-average peak temperature for a \$3.00 pulse from 27°C is 256°C, with a hot spot temperature of 746°C.

With \$3.00 reserved for a maximum pulse, the reactor has remaining reactivity \$1.00 greater than critical to support operation at power. Reactivity of \$1.00 allows operation 94 kW; 94 kW operation results in an average fuel temperature of 48°C, and a hot-spot

fuel temperature of 150°C. A $3.00 pulse increases this hot spot temperature from 150°C to 869°C.

Change in Peak Temperature Versus Pulsed Reactivity

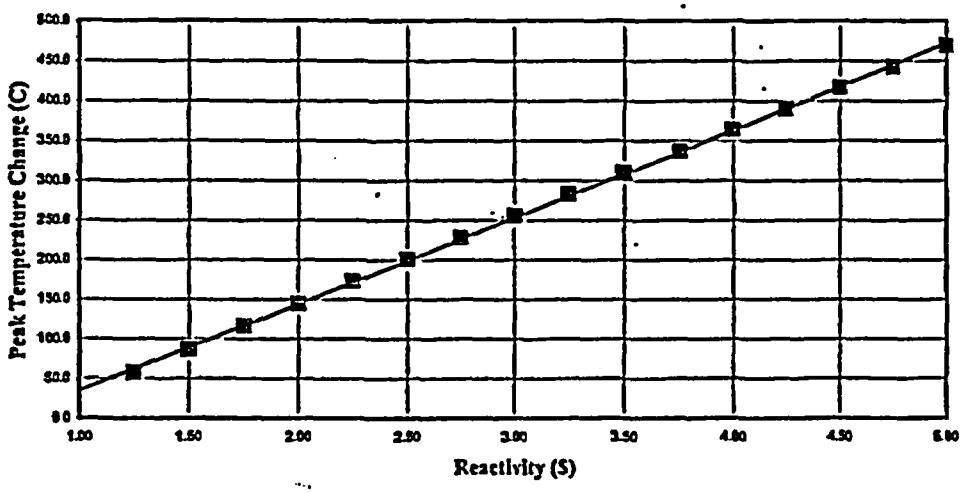

Figure 13.3, Change in Peak Temperature Versus Pulsed Reactivity Insertion

The postulated scenarios do not result in fuel damage, but physical aspects of system prevent these scenarios from occurring. It is not possible to achieve full power operation with the pulse rod fully inserted; since the pulse rod is partially withdrawn with air applied to the pulse solenoid, it physically cannot be pulsed. Although not required to ensure the safety of the reactor, an interlock prevents pulsing from power levels greater than a maximum of 10 kW.

c. Conclusions

Insertion of the maximum possible reactivity without initial temperature feedback (i.e., fuel temperature is too low to limit core available reactivity) results in a peak hot spot fuel temperature of 746°C, well below the safety limit.

Insertion of the maximum possible reactivity with initial temperature feedback (i.e., fuel temperature limits available) results in a peak hot spot fuel temperature of 869°C, well below the safety limit.

13.2.4 Single Element Failure in Air

Source quantities of radioactive noble gases and iodine are computed and tabulated for a maximum hypothetical accident involving cladding failure in a single TRIGA fuel element and the escape of the radionuclides into the environment. Two limiting cases of operation are considered. For short lived radionuclides, source terms are computed for element failure subsequent to eight hours full-power operation per day for five days. For long-lived radionuclides, source terms are computed for element failure subsequent to continuous operation

for 40 years at the average power experienced by the reactor over its first 33 years of operation. Also examined are residual sources still present in fuel, but generated in reactor operations prior to local receipt of the fuel in 1973. Potential consequences of radiological releases are examined. Even in the maximum hypothetical accident, no workers or members of the public are at risk of receiving radiation doses in excess of limits prescribed in federal regulations.

a. Assumptions and Approximations

Following are assumptions and approximations applied to calculations.

1. For long-lived radionuclides, calculations of radionuclide inventory in fuel are based on continuous operation prior to fuel failure for 40 years at the average thermal power experienced by the reactor during its first 33 years of operation, namely, 3.50 kW.

2. For short-lived radionuclides, calculations of radionuclide inventory in fuel are based on operation at the full thermal power of 1,250 kW for eight hours per day, for five successive days prior to fuel failure, an average of 31.25 kW-hr/day.

3. Radionuclide inventory in one "worst-case" fuel element is based on 81 elements in the core for the historical period and [redacted] elements for full power operation, which is the case for core II-16, [redacted] grams of ^{235}U per element [NUREG-2382], and a value of 2.0 as a very conservative value of the ratio of the maximum power in the core to the average power. Thus, for the historical period, the worst case element has operated at a thermal power of (3500/81)×2 = 86.42 W, and for the one-week full-power operation, (500/[redacted]×2 = [redacted] kW.

4. The fraction of noble gases and iodine contained within the fuel that is actually released is 1.0×10^{-4}. This is a very conservative value prescribed in NUREG 2387 [Hawley and Kathren, 1982] and may be compared to the value of 1.5×10^{-5} measured at General Atomics [Simnad et al., 1976] and used in SARs for other reactor facilities [NUREG-1390, 1990].

5. The fractional release of particulates (radionuclides other than noble gases and iodine) is 1.0×10^{-6}, a very conservative estimate used by Hawley and Kathren [1982].

b. Radionuclide Inventory Buildup and Decay

Consider a mass of ^{235}U yielding thermal power P (kW) due to thermal-neutron induced fission. The fission rate is related to the thermal power by the factor $k = 3.12 \times 10^{13}$ fissions per second per kW.[6] Consider also a fission product radionuclide, which is produced with yield Y, and which decays with rate constant λ. It is easily shown that the equilibrium activity A_∞ (Bq) of the fission product, which exists when the rate of creation by fission is equal to the rate of loss by decay, is given by $A_\infty = kPY$. Here it should be noted that the power must be small enough or the uranium mass large enough that the

[6] Note that the product of k and yield Y may be stated as $3.12 \times 10^{13} \times Y$ Bq/kW or $843 \times Y$ Ci/kW.

depletion of the ^{235}U is negligible.[7] Starting at time $t = 0$, the buildup of activity is given by

$$A(t) = A_{\infty} * \left(1 - e^{-\lambda * t}\right) \qquad (13.2.4\text{-}1)$$

For times much greater than the half-life of the radionuclide, $A \approx A_{\infty}$, and for times much less than the half-life, $A(t) = A_{\infty} * \lambda * t$. If the fission process ceases at time t_1, the specific activity at later time t is given by

$$A(t) = A_{\infty} * \left(1 - e^{-\lambda * t}\right) * e^{-\lambda * (t - t_1)} \qquad (13.4.2\text{-}2)$$

Consider the fission product ^{131}I, which has a half-life of 8.04 days ($\lambda = 0.00359\ h^{-1}$) and a chain (cumulative) fission product yield of about 0.031. At a thermal power of 1 kW, the equilibrium activity is about $A_{\infty} = 9.67 \times 10^{11}$ Bq (26.1 Ci). After only four hours of operation, though, the activity is only about 0.37 Ci. For equilibrium operation at 3.5 kW, distributed over 81 fuel elements, the average activity per element would be $26.1 \times 3.5 \div 81 = 1.13$ Ci per fuel element. The worst case element would contain twice this activity. With a release fraction of 1.0×10^{-4}, the activity available for release would be about $1.13 \times 2 \times 1.0 \times 10^{-4} = 2.26 \times 10^{-4}$ Ci. This type of calculation is performed by the ORIGEN code [CCC-371] for hundreds of fission products and for arbitrary times and power levels of operation as well as arbitrary times of decay after conclusion of reactor operation. The code accounts for branched decay chains. It also may account for depletion of ^{235}U and ingrowth of ^{239}Pu, although those features were not invoked in the calculations reported here because of minimal depletion in TRIGA fuel elements.

c. Data From Origen Calculations

ORIGEN-2.1 calculation output files are included as Appendices A and B; App. A contains data for the buildup of long-lived radionuclides over the 40-year entire operating history of the KSU TRIGA reactor, modeled as 86.42 W continuous thermal power. App. B contains data for the buildup of relatively short-lived radionuclides during a worst-case scenario modeled as 8-hours/day operations at 12.05 kW thermal power for five consecutive days. Tabulated results for Appendices A and B are μCi activities, by nuclide, immediately after reactor shutdown, and at 1, 2, 3, 7, and 14 days after shutdown. In Appendix A, which deals with relatively long-lived radionuclides, data are provided only for those nuclides present at activities greater than 100 mCi in a single fuel element at 1 day after reactor shutdown. In Appendix B, which deals with relatively short-lived radionuclides, data are provided only for those nuclides present at activities greater than 100 mCi in a single fuel element immediately after reactor shutdown. In both Appendices A and B, the activities per element are multiplied by the release fractions previously cited, thus yielding maximum activities available for release in a maximum hypothetical accident.

[7] Negligible burnup is modeled in ORIGEN calculations by setting the fuel mass very large (1 tonne) and the thermal power very low (1 kW or less).

d. Reference Case Source Terms

Appendices A and B data for worst case TRIGA fuel element are compared; greater values for any one isotope are selected as reference case source terms for maximum hypothetical accident. Data are presented in Table 13.10 for halogens/noble gases, and Table 13.11 for particulate radionuclides.

e. Derived Quantities

The raw data of Tables 13.10 and 13.11 are activities potentially released from a single worst-case fuel element that has experienced a cladding failure. This activity may itself be compared to the annual limit of intake (ALI) to gauge the potential risk to an individual worker. By dividing the activity by the 144,000 ft^3 free volume of the reactor bay in the Nuclear Reactor Facility, one obtains an air concentration (specific activity) that may be compared to the derived air concentration (DAC) for occupational exposure as given 10CFR20 or in EPA federal guidance [Eckerman et al., 1988].

Table 13.10, Reference Case Halogen & Noble Gas Activities Potentially Released in Maximum Hypothetical Accident at KSU TRIGA Mk. II Nnuclear Reactor. [1]

		Available activity (μCi) at time in days after reactor shutdown						
Element	Nuclide	0	1	2	3	7	14	28
Br	83	12110	13	0	0	0	0	0
Br	84	25388	0	0	0	0	0	0
I	131	8490	8245	7695	7130	5125	2810	840
I	132	23280	20943	16930	13685	5838	1318	68
I	133	69950	33530	15073	6773	278	0	0
I	134	192228	0	0	0	0	0	0
I	135	98750	7980	645	53	0	0	0
Kr	85	45	45	45	45	45	45	45
Kr	87	64498	0	0	0	0	0	0
Kr	88	79048	225	0	0	0	0	0
Kr	83M	9953	48	0	0	0	0	0
Kr	85M	23368	580	15	0	0	0	0
Xe	133	20363	24105	24010	22390	14110	5673	895
Xe	135	58955	30518	6595	1195	0	0	0
Xe	138	158548	0	0	0	0	0	0
Xe	133M	35	33	28	20	8	0	0
Xe	135M	205	15	0	0	0	0	0

NOTE: Available activity (> 10 μCi) is from a single worst-case fuel element as a function of time after reactor operation. Data are derived from ORIGEN 2.1 calculations [CCC-371] as summarized in Appendices A and B. Data are raw computational results and the number of significant figures exceeds the precision of the calculation.

Table 13.11, Reference Case Particulate Activities Potentially Released in a Maximum Hypothetical Accident at the KSUTRIGA Mk. II Nuclear Reactor.

		Available activity (μCi) at time in days after reactor shutdown						
Element	Nuclide	0	1	2	3	7	14	28
BA	139	1605	0	0	0	0	0	0
BA	140	128	120	115	108	88	60	28
BA	141	1480	0	0	0	0	0	0
BA	142	1470	0	0	0	0	0	0
CE	141	48	53	53	50	45	40	30
CE	143	553	340	205	125	18	0	0
CS	138	1713	0	0	0	0	0	0
LA	140	68	88	98	103	95	68	33
LA	141	1140	18	0	0	0	0	0
LA	142	1455	0	0	0	0	0	0
LA	143	1493	0	0	0	0	0	0
MO	99	395	308	240	185	68	13	0
MO	101	1273	0	0	0	0	0	0
NB	97	598	248	93	35	0	0	0
NB	98	1463	0	0	0	0	0	0
ND	147	55	50	48	45	35	23	10
ND	149	263	0	0	0	0	0	0
ND	151	105	0	0	0	0	0	0
PM	151	40	23	13	8	0	0	0
PR	143	70	88	98	100	90	65	33
PR	145	638	40	3	0	0	0	0
PR	147	573	0	0	0	0	0	0
RB	88	800	3	0	0	0	0	0
RB	89	1218	0	0	0	0	0	0
RH	105	78	65	40	25	5	0	0
RH	107	40	0	0	0	0	0	0
RU	105	188	5	0	0	0	0	0
SE	81	53	0	0	0	0	0	0
SE	83	50	0	0	0	0	0	0
SN	128	83	0	0	0	0	0	0
SR	89	28	28	28	28	25	23	20
SR	91	795	138	25	5	0	0	0
SR	92	1325	3	0	0	0	0	0
TC	101	1273	0	0	0	0	0	0
TC	104	460	0	0	0	0	0	0
TE	129	95	3	0	0	0	0	0
TE	131	628	5	3	3	0	0	0
TE	132	250	203	165	133	58	13	0
TE	133	965	0	0	0	0	0	0
TE	134	1713	0	0	0	0	0	0
Y	92	848	38	0	0	0	0	0
Y	93	863	170	33	8	0	0	0
Y	94	1583	0	0	0	0	0	0
Y	95	1613	0	0	0	0	0	0
Y	91M	423	88	15	3	0	0	0
ZR	95	30	30	30	28	28	25	23
ZR	97	658	245	93	35	0	0	0

Available activity (> 10 μCi) is for a single worst-case fuel element as a function of time after reactor operation. Data are derived from ORIGEN 2.1 calculations [CCC-371] as summarized in Appendices A and B. The table includes only those nuclides with activities in excess of 10 μCi. Data are raw computational results and the number of significant figures exceeds the precision of the calculation.

f. Comparison with the DAC and the ALI

The ALI is the activity that, if ingested or inhaled, would lead to either (a) the maximum permissible committed effective dose equivalent incurred annually in the workplace, nominally 5 rem, or (b) the maximum permissible dose to any one organ or tissue, nominally 50 rem. The DAC is the air concentration that, if breathed by reference man for one work year (2000 h), would result in the intake of the ALI. ALI does not apply to noble-gas radionuclides.

Potential activity releases are compared to ALIs, and air concentrations in the reactor bay are compared to DACs in Tables 13.12 and 13.13. Only for radioiodine does the available activity exceed the ALI. However, there is no credible scenario for accidental inhalation or ingestion of the undiluted radioiodine released from a fuel element.

When one compares with DACs the potential airborne concentration of radionuclides in the reactor bay, only the ^{131}I, ^{133}I, and ^{135}I isotopes plus ^{87}Kr and ^{88}Kr are of potential consequence. However, annual dose limits could be attained only with a constant air concentration over a long period of time. The ^{131}I released in the failure of a single element, for example, would decay with a half-life of 8.04 days. Thus, even the undetected failure of a fuel element would not be expected to lead to violations of the occupational dose limits expressed in 10CFR20 or in other federal guidance.

g. Comparison with the effluent concentration

Effluent concentration, listed in the last columns of Tables 13.12 and 13.13, are defined in continuous exposure (8760 hours per year) rather than 2000 hours per year occupational exposure. Exposure to a constant airborne concentration equal to the effluent concentration for one full year results in the annual dose limit of 100 mrem to members of the public. As is apparent from Tables 13.12 and 13.13, the reactor bay average concentrations immediately after fuel element failure exceed the effluent concentrations for several radionuclides. Thus, only for these radionuclides is it necessary to consider radioactive decay and atmospheric dispersal after release in estimating potential risk to members of the public. For posting purposes, concentrations relative to DACs are additive. For dosimetry purposes, products of concentrations and times, relative to DAC-hours, are additive.

h. Potential downwind dose to a member of the public

In this dose assessment, it is assumed that the available activity in a failed fuel element is released instantaneously and immediately after reactor shutdown. It is further assumed that a member of the public is positioned directly downwind from the Nuclear Reactor Facility and remains in place during the entire passage of the airborne radioactivity. The very conservative approximations of Hawley and Kathren [1982] are adopted in the assessment, namely, that the atmospheric dispersion (χ/Q) factor is 0.01 s/m^3 (2.78×10^{-6} h/m^3) and the breathing rate V is 1.2 m^3/h. No credit is taken for partial containment, plateout, or other potential mitigating mechanisms, however realistic and probable.
Let the activity of nuclide i released be A (μCi) as given in Tables 13.12 and 13.13. If one neglects radioactive decay, the activity inhaled during passage of the airborne

activity is $A\ V(\times\ \chi/Q)$ [Faw and Shultis, 1993]. The product of the activity inhaled and the dose conversion factor $\Re$ (mrem/μCi) [Eckerman et al., 1988] yields D_i (mrem), the more critical of the organ dose or the effective dose equivalent to the total body. Results of such calculations are presented in Table 13.14. As is apparent from the table, individual organ doses as well as the total committed effective dose equivalent are well below any regulatory limits. Entries are shown only for doses of 0.001 mrem or greater.

Table 13.12, Comparison of Halogen and Noble Gas Available Activities Immediately After Reactor Shutdown with ALIs and Reactor Bay Concentrations with DACs and Effluent Concentrations.

Element	Nuclide	Half-life	Available activity (μCi)	Inhalation ALI (μCi)	DAC (μCi/cm³)	Reactor bay conc. (μCi/cm³)	Effluent conc. (μCi/cm³)	Ratio to DAC	Ratio to Eff Limit
Br	83	2.39 h	12100	6.E+04	3.E-5	3.00E-06	9E-08	0.1	33.3
Br	84	31.8 m	25375	6.E+04	2.E-5	6.25E-06	8E-08	0.3	78.1
I	131	8.04 d	8500	5.E+01	2.E-8	2.08E-06	2E-10	103.8	10375.0
I	132	2.30 h	23275	8.E+03	3.E-6	5.75E-06	2E-08	1.9	287.5
I	133	20.8 h	69950	3.E+02	1.E-7	1.73E-05	1E-09	172.5	17250.0
I	134	52.6 m	192225	5.E+04	2.E-5	4.75E-05	6E-08	2.4	791.7
I	135	6.61 h	98750	2.E+03	7.E-7	2.43E-05	6E-09	34.6	4041.7
Kr	83m	1.83 h	9950		1.E-2	2.45E-06	5E-05	0.0	0.0
Kr	85m	4.48 h	23375		2.E-5	5.75E-06	1E-07	0.3	57.5
Kr	85	10.7 y	50		1.E-4	1.25E-08	7E-07	0.0	0.0
Kr	87	76.3 m	64500		5.E-6	1.58E-05	2E-08	3.2	787.5
Kr	88	2.84 h	79050		2.E-6	1.95E-05	9E-09	9.8	2166.7
Xe	133m	2.19 d	0		1.E-4	1.00E-08	6E-07	0.0	0.0
Xe	133	5.25 d	20350		1.E-4	5.00E-06	5E-07	0.1	10.0
Xe	135m	15.3 m	25		9.E-6	6.25E-09	4E-08	0.0	0.2
Xe	135	9.09 h	58950		1.E-5	1.45E-05	7E-08	1.5	207.1

[a]Bay concentration exceeds DAC.
[b]Bay concentration exceeds effluent concentration.

Table 13.13, Comparison of Particulate Available Activities (> 100 μCi) with ALIs and Reactor Bay Concentrations with DACs and Effluent Concentrations.

Element	Nuclide	Half-life	Available activity (μCi)	Inhalation ALI (μCi)	DAC (μCi/cm³)	Reactor bay conc. (μCi/cm³)	Effluent conc. (μCi/cm³)	Ratio to DAC	Ratio to Eff Limit
Ba	139	82.7 m	1600	75000	0.00001	4E-07	2E-09	4.00E-02	2.00E+02
Ba	140	12.7 d	125	2500	6E-07	3E-08	2E-09	5.00E-02	1.50E+01
Ba	141		1475	175000	0.00003	3.5E-07	1E-07	1.17E-02	3.50E+00
Ce	141		50	1500	2E-07	1.23E-08	8E-10	6.13E-02	1.53E+01
Ce	143	33.0 h	550	5000	7E-07	1.35E-07	2E-09	1.93E-01	6.75E+01
Cs	138	32.2 m	1700	150000	0.00002	4.25E-07	8E-08	2.13E-02	5.31E+00
La	140		75	2500	5E-07	1.85E-08	2E-09	3.70E-02	9.25E+00
La	141	3.93 h	1150	22500	0.000004	2.75E-07	1E-08	6.88E-02	2.75E+01
La	142	92.5 m	1450	50000	0.000009	3.5E-07	3E-08	3.89E-02	1.17E+01
La	143		1500	225000	0.00004	3.75E-07	1E-07	9.38E-03	3.75E+00
Mo	99	66.0 h	400	2500	6E-07	9.75E-08	2E-09	1.63E-01	4.88E+01
Nb	98		1475	125000	0.00002	3.5E-07	7E-08	1.75E-02	5.00E+00
Nd	147	1.73 h	50	2000	4E-07	1.23E-08	1E-09	3.06E-02	1.23E+01
Pm	151		50	7500	0.000001	1.23E-08	4E-09	1.23E-02	3.06E+00
Pr	143		75	1750	3E-07	1.85E-08	9E-10	6.17E-02	2.06E+01
Pr	145	5.98 h	650	20000	0.000003	1.6E-07	1E-08	5.33E-02	1.60E+01
Rb	88	17.8 m	800	150000	0.00003	1.95E-07	9E-08	6.50E-03	2.17E+00
Rb	89		1225	250000	0.00006	3E-07	2E-07	5.00E-03	1.50E+00
Ru	105	4.44 h	200	25000	0.000005	5E-08	2E-08	1.00E-02	2.50E+00
Se	81	18.5 m	50	500000	0.00009	1.23E-08	3E-07	1.36E-04	4.08E-02
Sn	128	59.1 m	75	75000	0.00001	1.85E-08	4E-08	1.85E-03	4.63E-01
Sr	89		25	250	6E-08	6.25E-09	2E-10	1.04E-01	3.13E+01
Sr	91	9.5 h	800	10000	0.000001	1.95E-07	5E-09	1.95E-01	3.90E+01
Sr	92	2.71 h	1325	17500	0.000003	3.25E-07	9E-09	1.08E-01	3.61E+01
Te	129	69.6 m	100	150000	0.00003	2.45E-08	9E-08	8.17E-04	2.72E-01
Te	131	25.0 m	625	12500	0.000002	1.53E-07	1E-09	7.63E-02	1.53E+02
Te	132	78.2 h	250	500	9E-08	6.25E-08	9E-10	6.94E-01	6.94E+01
Te	133	12.5 m	975	50000	0.000009	2.4E-07	8E-08	2.67E-02	3.00E+00
Te	134	41.8 m	1725	50000	0.00001	4.25E-07	7E-08	4.25E-02	6.07E+00
Y	91m	49.7 m	425	500000	0.00007	1.05E-07	2E-07	1.50E-03	5.25E-01
Sr	91	9.5 h	800	10000	0.000001	1.95E-07	5E-09	1.95E-01	3.90E+01
Sr	92	2.71 h	1325	17500	0.000003	3.25E-07	9E-09	1.08E-01	3.61E+01
Te	129	69.6 m	100	150000	0.00003	2.45E-08	9E-08	8.17E-04	2.72E-01
Te	131	25.0 m	625	12500	0.000002	1.53E-07	1E-09	7.63E-02	1.53E+02
Te	132	78.2 h	250	500	9E-08	6.25E-08	9E-10	6.94E-01	6.94E+01
Te	133	12.5 m	975	50000	0.000009	2.4E-07	8E-08	2.67E-02	3.00E+00

Table 13.13, *Comparison of Particulate Available Activities (> 100 μCi) with ALIs and Reactor Bay Concentrations with DACs and Effluent Concentrations.*

Element	Nuclide	Half-life	Available activity (μCi)	Inhalation ALI (μCi)	DAC (μCi/cm³)	Reactor bay conc. (μCi/cm³)	Effluent conc. (μCi/cm³)	Ratio to DAC	Ratio to Eff Limit
Te	134	41.8 m	1725	50000	0.00001	4.25E-07	7E-08	4.25E-02	6.07E+00
Y	91m	49.7 m	425	500000	0.00007	1.05E-07	2E-07	1.50E-03	5.25E-01
Y	92	3.54 h	850	20000	0.000003	2.08E-07	1E-08	6.92E-02	2.08E+01
Y	93	10.1 h	850	5000	0.000001	2.08E-07	3E-09	2.08E-01	6.92E+01
Zr	95		25	250	5E-08	6.25E-09	4E-10	1.25E-01	1.56E+01
Zr	97	16.9 h	650	2500	5E-07	1.6E-07	2E-09	3.20E-01	8.00E+01

i. Residual Activity from Fuel Utilization Prior to Receipt

All but a few instrumented Mark-II fuel elements in the original 1962 core loading were replaced by Mark-III elements on July 10, 1973. The replacement elements had seen considerable use prior to their installation at Kansas State University. The two most heavily used elements, with serial numbers 4078 and 4079, had experienced, respectively, consumption of 11.27 and 10.33 g of ^{235}U. Even after about 25 years of subsequent use, considerable ^{137}Cs, ^{90}Sr, and ^{85}Kr remain from fission during the pre-1973 use. However, the ^{85}Kr atmospheric concentration inside the reactor bay immediately after release would be orders of magnitude lower than the DAC. Therefore only ^{37}Cs and ^{90}Sr offer a potential for occupational or public risk. In the absence of knowledge about the pattern of early fuel utilization, it is assumed that all the generation of fission products took place in 1973 and that fission product decay took place over the period of 28 years from 1973 until 2001.

If Y is the fission yield, λ is the decay constant (s^{-1}), and N_a is Avogadro's number, the activity A (Bq) of any one radionuclide immediately after fissioning of mass m (g) of ^{235}U is

$$A = \frac{N_a}{235} m Y \lambda. \qquad (13.2.4\text{-}3)$$

Activity calculations using this formula and consequences, as computed in 13.14 are reported in Table 13.15.

Table 13.15. Worst Case Source Terms and Consequence Calculations for a Single TRIGA Fuel Element Experiencing 11.27 g of ^{235}U Consumption 28 Years Prior to Element Failure.

FACTOR	RADIONUCLIDE	
	^{90}Sr	^{137}Cs
Half life (y)	29.12	30.00
Decay constant λ (s^{-1})	7.54×10^{-10}	7.32×10^{-10}
Fission yield Y	0.0577	0.0615
Release fraction	1.00×10^{-06}	1.00×10^{-06}
Initial Bq/g contained in element	$1.12\times 10^{+11}$	$1.15\times 10^{+11}$
Initial µCi available for release	34.0	35.2
µCi available for release in 28 y	17.4	18.4
ALI (µCi)	4	200
Reactor bay concentration (µCi/cm^3)	4.3×10^{-09}	4.5×10^{-09}
DAC (µCi/cm^3)	$2.\times 10^{-09}$	$6.\times 10^{-08}$
Tissue at risk	Bone surface	Total body
Dose conversion factor (mrem/µCi)	2690	32
Maximum downwind dose (mrem)	0.16	0.0020

Whereas the ^{90}Sr activity available for release would exceed the occupational ALI and, if dispersed within the reactor bay, would have a concentration in excess of the DAC, credible mechanisms for ingestion or inhalation of the full available activity or even its full dispersion are not apparent. Thus, neither the ^{90}Sr nor the ^{137}Cs would pose a significant occupational threat. Even if the total available activity were somehow dispersed to the free atmosphere, no person downwind of the accidental release would receive doses even approaching regulatory limits.

j. Conclusions

Fission product inventories in TRIGA fuel elements were calculated with the ORIGEN code, using very conservative approximations. Then, potential radionuclide releases from worst-case fuel elements were computed, again using very conservative approximations. Even if it were assumed that releases took place immediately after reactor operation, and that radionuclides were immediately dispersed inside the reactor bay workplace, few radionuclide concentrations would be in excess of occupational derived air concentrations, and then only for a matter of hours or days. Only for certain nuclides of iodine would the potential release be in excess of the annual limit of intake. However, there is no credible scenario for accidental inhalation or ingestion of the undiluted radioiodine that might be released from a damaged fuel element.

For the residual ^{90}Sr and ^{137}Cs remaining in fuel elements from consumption of ^{235}U prior to receipt of the fuel at Kansas State University, only the former would pose any conceivable occupational threat. However, the total ^{90}Sr activity available for release is

estimated to be at most about 4 times the ALI and there is no credible scenario for its consumption by a worker.

As far as potential consequences to the general public are concerned, only for the few radionuclides listed in Table 13.14, are maximum concentrations inside the reactor facility in excess of effluent concentrations listed in 10CFR20 and potential doses 0.001 mrem or greater. However, even in the extremely unlikely event that radionuclides released from a damaged fuel element were immediately released to the outside atmosphere, very conservative calculations reveal that radionuclides inhaled by persons downwind from the release would lead to organ doses or effective doses very far below regulatory limits. As is shown in Table 13.15, the same is true for residual ^{90}Sr and ^{137}Cs remaining in fuel elements from early operations.

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APPENDICES TO CHAPTER 13

A Origen 2.1 input file for ^{235}U fission at 1 W thermal power for 40 years.

B Origen 2.1 input file for ^{235}U fission at 1 W thermal power 8 hours per day for 5 days.

C Origen 2.1 output file extracts for ^{235}U fission at 1 W thermal power for 40 years.

D Origen 2.1 output file extracts for ^{235}U fission at 1 W thermal power 8 hours per day for 5 Days.

E Maximum activity available for release from a single TRIGA fuel element as a function of time after shutdown for a ^{235}U-fueled thermal reactor operating at 3.5 kW thermal power for 40 years, based on one element of 81 at 86.42 W.

F Maximum activity available for release from a single TRIGA fuel element as a function of time after shutdown for a ^{235}U-fueled thermal reactor operating at 500 kW kW thermal power for 8 hours per day for 5 days, based on one element of 83 at 12.05 kW

CHAPTER 13 APPENDIX A

```
        ORIGEN Input file for 1 tonne U-235 at 1 watt for 40 years
-1
-1
-1
  RDA     ORIGEN2, VERSION 2.1 (8-1-91)  TRIGA REACTOR REFERENCE PROBLEM
  RDA     UPDATED BY: Richard E. Faw, Kansas State University
  BAS     ONE TONNE OF U-235
  RDA     Continuous operation for 40 years at 1 watt
  RDA     WARNING:  VECTORS ARE OFTEN CHANGED WITH RESPECT TO THEIR CONTENT.
  RDA               THESE CHANGES WILL BE NOTED ON RDA CARDS.
  CUT     -1
  RDA     LIBRARY PRINT (1=PRINT,0=DON'T PRINT)
  LIP     0 0 0
  RDA     DECAY LIBRARY CHOICES (0=PRINT; 1 2 3 DECAY LIBRARIES; 601 ...
  RDA     CROSS SECTIONS; ETC, SEE P. 47)
  LIB     0  1 2 3  201  202  203   9   3   0  1   38
  PHO     101 102 103 10   <<< PHOTON LIBRARIES, P. 47
  TIT     INITIAL COMPOSITIONS OF UNIT AMOUNTS OF FUEL AND STRUCT MAT'LS
  RDA     READ FUEL COMPOSITION INCLUDING IMPURITIES (1 G)
  INP     -1   1  -1  -1  1  1
  TIT     IRRADIATION OF ONE TONNE U-235
  MOV     -1   1   0   1.0
  HED      1                                                  CHARGE
  BUP
  IRP      5 .000001   1   2   5 2  1 W/Tonne FOR 5 YEARS
  IRP     10 .000001   2   1   5 0  1 W/Tonne FOR 5 YEARS
  IRP     15 .000001   1   2   5 0  1 W/Tonne FOR 5 YEARS
  IRP     20 .000001   2   1   5 0  1 W/Tonne FOR 5 YEARS
  IRP     25 .000001   1   2   5 0  1 W/Tonne FOR 5 YEARS
  IRP     30 .000001   2   1   5 0  1 W/Tonne FOR 5 YEARS
  IRP     35 .000001   1   2   5 0  1 W/Tonne FOR 5 YEARS
  IRP     40 .000001   2   1   5 0  1 W/Tonne FOR 5 YEARS
  BUP
  OPTL      24*8          ACTIVATION PRODUCT OUTPUT OPTS P. 56
  OPTA      24*8          ACTINIDE OUTPUT OPTIONS P. 59
  OPTF       6*8 5   17*8  FISSION PRODUCT OUTPUT OPTIONS P. 59
  RDA     DECAY TO 28 DAYS
  DEC      1      1   2   4   2
  DEC      2      2   3   4   0
  DEC      3      3   4   4   0
  DEC      7      4   5   4   0
  DEC     14      5   6   4   0
  DEC     28      6   7   4   0
  OUT      -7     1     -1    0
  OUT       7     1     -1    0
  END
  2   922340    0.0  922350 1.E06 922380  0.      0    0.0   PURE U-235
  0
```

CHAPTER 13 APPENDIX B

ORIGEN Input File for 1 tonne U-235 at 1 watt
8 hours per day for 5 days

```
-1
-1
-1
  RDA     ORIGEN2, VERSION 2.1 (8-1-91)  TRIGA REACTOR REFERENCE PROBLEM
  RDA     UPDATED BY: Richard E. Faw, Kansas State University
  BAS     One tonne U-235
  RDA     -1 = Fuel composition
  CUT     -1
  RDA     LIBRARY PRINT (1=PRINT,0=DON'T PRINT)
  LIP     0 0 0
  RDA     DECAY LIBRARY CHOICES (0=PRINT; 1 2 3 DECAY LIBRARIES; 601 ...
  RDA     CROSS SECTIONS; ETC, SEE P. 47)
  LIB     0  1 2 3  201  202  203   9   3   0  1   38
  PHO     101 102 103 10   <<< PHOTON LIBRARIES, P. 47
  TIT     INITIAL COMPOSITIONS OF UNIT AMOUNTS OF FUEL AND STRUCT MAT'LS
  RDA     READ FUEL COMPOSITION INCLUDING IMPURITIES
  INP     -1   1  -1  -1  1  1
  TIT     One tonne U-235 8 h/d for 5 days at 1 kWt
  MOV     -1   1   0   1.0
  HED      1                                                CHARGE
  BUP
  IRP      8.0   0.001    1   2   3 2  OPERATE FOR 8 HR AT 1 kW
  DEC     24.0            2   3   3 0  COOL FOR 16 HOURS
  IRP     32.0   0.001    3   4   3 0  OPERATE FOR 8 HR
  DEC     48.0            4   5   3 0  COOL FOR 16 HOURS
  IRP     56.0   0.001    5   6   3 0  OPERATE FOR 8 HR
  DEC     72.0            6   7   3 0  COOL FOR 16 HOURS
  IRP     80.0   0.001    7   8   3 0  OPERATE FOR 8 HR
  DEC     96.0            8   9   3 0  COOL FOR 16 HOURS
  IRP    104.0   0.001    9  10   3 0  OPERATE FOR 8 HR
  OPTL      24*8           ACTIVATION PRODUCT OUTPUT OPTS P. 56
  OPTA      24*8           ACTINIDE OUTPUT OPTIONS P. 59
  OPTF       6*8 5   17*8  FISSION PRODUCT OUTPUT OPTIONS P. 59
  RDA     MOVE COMPOSITION VECTOR FROM 10 TO 1
  MOV       10   1   0  1.0
  RDA     DECAY TO 0.1 UNITS (2=MINUTES) FROM COMP VEC 1 TO VEC 3
  DEC      1     1  2   4   2
  DEC      2     2  3   4   0
  DEC      3     3  4   4   0
  DEC      7     4  5   4   0
  DEC      14    5  6   4   0
  DEC      28    6  7   4   0
  OUT      -7     1     -1    0
  OUT       7     1     -1    0
  END
  2   922340    0.0  922350  1.E6 922380    0.00  0    0.0   1 g U-235
  0
```

CHAPTER 13 APPENDIX C

ORIGEN Output File Extracts for 1 tonne U-235 at 1 watt for 40 Years
NUCLIDE TABLE: RADIOACTIVITY, CURIES

		Time post discharge						
		0	1.0 D	2.0 D	3.0 D	7.0 D	14.0 D	28.0 D
AG	111	1.65E-04	1.51E-04	1.37E-04	1.25E-04	8.62E-05	4.50E-05	1.22E-05
BA	140	5.23E-02	4.96E-02	4.70E-02	4.45E-02	3.58E-02	2.45E-02	1.15E-02
BA	137	2.93E-02	2.93E-02	2.93E-02	2.93E-02	2.93E-02	2.93E-02	2.92E-02
CE	141	4.93E-02	4.85E-02	4.75E-02	4.65E-02	4.27E-02	3.68E-02	2.73E-02
CE	143	4.98E-02	3.03E-02	1.83E-02	1.11E-02	1.47E-03	4.32E-05	3.72E-08
CE	144	4.56E-02	4.55E-02	4.54E-02	4.53E-02	4.48E-02	4.41E-02	4.26E-02
CS	137	3.10E-02	3.10E-02	3.09E-02	3.09E-02	3.09E-02	3.09E-02	3.09E-02
EU	155	2.75E-04	2.75E-04	2.75E-04	2.75E-04	2.75E-04	2.74E-04	2.72E-04
EU	156	1.13E-04	1.10E-04	1.06E-04	1.01E-04	8.43E-05	6.13E-05	3.23E-05
I	131	2.37E-02	2.20E-02	2.03E-02	1.87E-02	1.33E-02	7.28E-03	2.18E-03
I	132	3.56E-02	2.95E-02	2.39E-02	1.93E-02	8.23E-03	1.86E-03	9.45E-05
I	133	5.68E-02	2.62E-02	1.18E-02	5.29E-03	2.16E-04	8.00E-07	1.10E-11
I	135	5.31E-02	4.29E-03	3.46E-04	2.80E-05	1.19E-09	2.66E-17	1.34E-32
KR	85	2.10E-03	2.10E-03	2.10E-03	2.10E-03	2.10E-03	2.10E-03	2.09E-03
KR	85M	1.07E-02	2.64E-04	6.44E-06	1.57E-07	5.58E-14	2.88E-25	0.00E+00
LA	140	5.24E-02	5.19E-02	5.06E-02	4.89E-02	4.08E-02	2.82E-02	1.32E-02
LA	141	4.93E-02	7.77E-04	1.13E-05	1.64E-07	7.27E-15	9.89E-28	0.00E+00
MO	99	5.06E-02	3.94E-02	3.06E-02	2.38E-02	8.68E-03	1.49E-03	4.36E-05
NB	95	5.38E-02	5.38E-02	5.38E-02	5.37E-02	5.35E-02	5.28E-02	5.04E-02
NB	97	4.93E-02	1.85E-02	6.90E-03	2.58E-03	5.03E-05	5.14E-08	5.70E-14
NB	95M	3.78E-04	3.77E-04	3.76E-04	3.74E-04	3.64E-04	3.41E-04	2.95E-04
NB	97M	4.66E-02	1.74E-02	6.51E-03	2.43E-03	4.74E-05	4.84E-08	5.01E-14
ND	147	1.90E-02	1.79E-02	1.68E-02	1.58E-02	1.23E-02	7.91E-03	3.29E-03
PM	147	1.91E-02	1.91E-02	1.91E-02	1.91E-02	1.91E-02	1.90E-02	1.89E-02
PM	149	9.11E-03	6.89E-03	5.04E-03	3.68E-03	1.05E-03	1.17E-04	1.46E-06
PM	151	3.52E-03	1.97E-03	1.10E-03	6.11E-04	5.86E-05	9.70E-07	2.65E-10
PR	143	4.98E-02	4.93E-02	4.80E-02	4.64E-02	3.86E-02	2.71E-02	1.33E-02
PR	144	4.56E-02	4.55E-02	4.54E-02	4.53E-02	4.48E-02	4.41E-02	4.26E-02
PR	145	3.29E-02	2.06E-03	1.27E-04	7.89E-06	1.16E-10	4.06E-19	5.14E-36
PR	144	5.48E-04	5.46E-04	5.45E-04	5.43E-04	5.38E-04	5.29E-04	5.11E-04
RH	105	8.53E-03	6.09E-03	3.83E-03	2.39E-03	3.64E-04	1.35E-05	1.86E-08
RH	106	3.27E-03	3.26E-03	3.26E-03	3.25E-03	3.23E-03	3.18E-03	3.10E-03
RH	103	2.37E-02	2.33E-02	2.29E-02	2.25E-02	2.09E-02	1.85E-02	1.45E-02
RU	103	2.63E-02	2.58E-02	2.54E-02	2.49E-02	2.32E-02	2.05E-02	1.60E-02
RU	105	8.53E-03	2.08E-04	4.90E-06	1.16E-07	3.57E-14	1.44E-25	0.00E+00
RU	106	3.27E-03	3.26E-03	3.26E-03	3.25E-03	3.23E-03	3.18E-03	3.10E-03
SB	125	2.49E-04	2.49E-04	2.49E-04	2.49E-04	2.48E-04	2.47E-04	2.45E-04
SB	127	1.10E-03	9.28E-04	7.75E-04	6.47E-04	3.15E-04	8.93E-05	7.18E-06
SB	129	5.33E-03	1.15E-04	2.45E-06	5.20E-08	1.06E-14	2.08E-26	0.00E+00
SM	151	9.34E-04	9.34E-04	9.34E-04	9.34E-04	9.34E-04	9.33E-04	9.33E-04
SM	153	1.36E-03	9.55E-04	6.69E-04	4.68E-04	1.13E-04	9.30E-06	6.34E-08
SN	125	1.13E-04	1.05E-04	9.75E-05	9.07E-05	6.80E-05	4.11E-05	1.50E-05
SR	89	4.05E-02	3.99E-02	3.94E-02	3.89E-02	3.68E-02	3.34E-02	2.76E-02
SR	90	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02
SR	91	4.94E-02	8.59E-03	1.49E-03	2.59E-04	2.35E-07	1.12E-12	2.51E-23
SR	92	5.04E-02	1.09E-04	2.35E-07	5.07E-10	1.10E-20	0.00E+00	0.00E+00
TC	99M	4.43E-02	3.76E-02	2.95E-02	2.29E-02	8.36E-03	1.43E-03	4.20E-05
TE	127	1.09E-03	1.02E-03	8.88E-04	7.68E-04	4.49E-04	2.27E-04	1.37E-04
TE	129	5.25E-03	6.43E-04	4.99E-04	4.86E-04	4.48E-04	3.88E-04	2.90E-04
TE	131	2.13E-02	3.95E-04	2.27E-04	1.30E-04	1.42E-05	2.92E-07	1.24E-10
TE	132	3.54E-02	2.86E-02	2.32E-02	1.87E-02	7.99E-03	1.80E-03	9.17E-05
TE	127	1.53E-04	1.53E-04	1.52E-04	1.52E-04	1.50E-04	1.44E-04	1.33E-04
TE	129	7.91E-04	7.78E-04	7.63E-04	7.47E-04	6.88E-04	5.95E-04	4.46E-04
TE	131	3.04E-03	1.75E-03	1.01E-03	5.79E-04	6.30E-05	1.30E-06	5.52E-10
XE	133	5.68E-02	5.47E-02	5.03E-02	4.52E-02	2.74E-02	1.09E-02	1.72E-03
XE	135	5.52E-02	2.02E-02	4.15E-03	7.39E-04	5.35E-07	1.47E-12	1.09E-23
XE	131	2.63E-04	2.62E-04	2.61E-04	2.58E-04	2.41E-04	1.97E-04	1.11E-04
XE	133	1.65E-03	1.51E-03	1.24E-03	9.65E-04	2.96E-04	3.27E-05	3.90E-07
XE	135	9.52E-03	6.87E-04	5.55E-05	4.48E-06	1.91E-10	4.27E-18	2.14E-33
Y	90	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02	2.97E-02
Y	91	4.94E-02	4.91E-02	4.86E-02	4.80E-02	4.58E-02	4.21E-02	3.57E-02
Y	92	5.05E-02	1.60E-03	1.70E-05	1.60E-07	1.10E-15	5.65E-30	0.00E+00
Y	93	5.44E-02	1.06E-02	2.04E-03	3.93E-04	5.41E-07	5.32E-12	5.15E-22
Y	91M	2.86E-02	5.46E-03	9.47E-04	1.64E-04	1.49E-07	7.09E-13	1.60E-23
ZR	95	5.38E-02	5.32E-02	5.26E-02	5.21E-02	4.99E-02	4.62E-02	3.97E-02
ZR	97	4.92E-02	1.84E-02	6.87E-03	2.57E-03	5.00E-05	5.11E-08	5.29E-14

CHAPTER 13 APPENDIX D

ORIGEN Output File Extracts for 1 tonne U-235 at 1 W for 8 h/d, 5 Days
NUCLIDE TABLE: RADIOACTIVITY, CURIES

		Time post discharge						
		0	1.0 D	2.0 D	3.0 D	7.0 D	14.0 D	28.0 D
AS	78	1.24E-01	2.17E-05	5.66E-10	1.16E-14	1.11E-33	0.00E+00	0.00E+00
BA	139	5.33E+01	3.47E-04	1.99E-09	1.14E-14	1.28E-35	0.00E+00	0.00E+00
BA	140	4.22E+00	4.00E+00	3.79E+00	3.59E+00	2.89E+00	1.98E+00	9.27E-01
BA	141	4.91E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BA	142	4.88E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BR	83	4.02E+00	4.13E-03	3.92E-06	3.72E-09	3.01E-21	0.00E+00	0.00E+00
BR	84	8.43E+00	2.19E-13	5.13E-27	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CE	141	1.57E+00	1.74E+00	1.71E+00	1.67E+00	1.54E+00	1.32E+00	9.81E-01
CE	143	1.84E+01	1.13E+01	6.83E+00	4.13E+00	5.49E-01	1.61E-02	1.39E-05
CE	144	1.85E-01	1.84E-01	1.84E-01	1.83E-01	1.82E-01	1.79E-01	1.73E-01
CS	138	5.68E+01	3.40E-12	1.17E-25	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	131	2.82E+00	2.74E+00	2.55E+00	2.37E+00	1.70E+00	9.33E-01	2.79E-01
I	132	7.73E+00	6.95E+00	5.62E+00	4.54E+00	1.94E+00	4.37E-01	2.22E-02
I	133	2.32E+01	1.11E+01	5.00E+00	2.25E+00	9.17E-02	3.40E-04	4.66E-09
I	134	6.38E+01	1.62E-06	9.36E-15	5.37E-23	0.00E+00	0.00E+00	0.00E+00
I	135	3.28E+01	2.65E+00	2.14E-01	1.73E-02	7.35E-07	1.65E-14	8.26E-30
KR	87	2.14E+01	4.51E-05	9.39E-11	1.96E-16	0.00E+00	0.00E+00	0.00E+00
KR	88	2.62E+01	7.49E-02	2.14E-04	6.09E-07	4.03E-17	6.19E-35	0.00E+00
KR	83M	3.30E+00	1.59E-02	1.65E-05	1.58E-08	1.29E-20	0.00E+00	0.00E+00
KR	85M	7.76E+00	1.93E-01	4.70E-03	1.15E-04	4.07E-11	2.10E-22	0.00E+00
LA	140	2.25E+00	2.88E+00	3.22E+00	3.38E+00	3.19E+00	2.27E+00	1.07E+00
LA	141	3.79E+01	6.11E-01	8.86E-03	1.29E-04	5.72E-12	7.77E-25	0.00E+00
LA	142	4.83E+01	1.15E-03	2.43E-08	5.12E-13	1.01E-31	0.00E+00	0.00E+00
LA	143	4.96E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MO	99	1.31E+01	1.02E+01	7.93E+00	6.17E+00	2.25E+00	3.85E-01	1.13E-02
MO	101	4.23E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NB	97	1.98E+01	8.19E+00	3.06E+00	1.14E+00	2.23E-02	2.28E-05	2.53E-11
NB	98	4.86E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ND	147	1.79E+00	1.69E+00	1.59E+00	1.49E+00	1.16E+00	7.50E-01	3.12E-01
ND	149	8.74E+00	5.96E-04	3.98E-08	2.65E-12	5.23E-29	0.00E+00	0.00E+00
ND	151	3.50E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PD	109	1.19E-01	3.50E-02	1.02E-02	2.95E-03	2.11E-05	3.69E-09	1.13E-16
PM	151	1.36E+00	7.72E-01	4.30E-01	2.39E-01	2.30E-02	3.80E-04	1.04E-07
PR	143	2.31E+00	2.92E+00	3.22E+00	3.33E+00	3.03E+00	2.16E+00	1.06E+00
PR	144	1.86E-01	1.84E-01	1.84E-01	1.83E-01	1.82E-01	1.79E-01	1.73E-01
PR	145	2.12E+01	1.33E+00	8.25E-02	5.11E-03	7.52E-08	2.63E-16	3.22E-33
PR	147	1.90E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RB	88	2.65E+01	8.37E-02	2.39E-04	6.80E-07	4.50E-17	7.40E-35	0.00E+00
RH	105	2.57E+00	2.17E+00	1.37E+00	8.55E-01	1.30E-01	4.84E-03	6.68E-06
RH	107	1.37E+00	1.84E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RU	103	7.45E-01	7.33E-01	7.20E-01	7.08E-01	6.59E-01	5.83E-01	4.55E-01
RU	105	6.23E+00	1.54E-01	3.62E-03	8.54E-05	2.64E-11	1.07E-22	0.00E+00
SB	127	2.19E-01	1.96E-01	1.63E-01	1.36E-01	6.64E-02	1.88E-02	1.51E-03
SE	81	1.76E+00	2.49E-09	6.77E-17	1.84E-24	0.00E+00	0.00E+00	0.00E+00
SE	83	1.64E+00	8.98E-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SM	153	4.27E-01	3.01E-01	2.11E-01	1.48E-01	3.55E-02	2.93E-03	2.00E-05
SM	155	2.76E-01	8.53E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SN	127	6.52E-01	2.37E-04	8.59E-08	3.12E-11	5.40E-25	0.00E+00	0.00E+00
SN	128	2.78E+00	1.25E-07	5.62E-15	2.53E-22	0.00E+00	0.00E+00	0.00E+00
SR	89	9.30E-01	9.27E-01	9.15E-01	9.02E-01	8.54E-01	7.76E-01	6.40E-01
SR	91	2.64E+01	4.60E+00	7.99E-01	1.39E-01	1.26E-04	5.98E-10	1.35E-20
SR	92	4.40E+01	9.50E-02	2.05E-04	4.42E-07	9.59E-18	1.62E-36	0.00E+00
TC	101	4.23E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TC	104	1.53E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE	127	1.61E-01	1.77E-01	1.55E-01	1.31E-01	6.49E-02	1.94E-02	2.81E-03
TE	129	3.11E+00	1.18E-01	1.87E-02	1.62E-02	1.49E-02	1.29E-02	9.65E-03
TE	131	2.09E+01	1.51E-01	8.68E-02	4.99E-02	5.43E-03	1.12E-04	4.76E-08
TE	132	8.32E+00	6.74E+00	5.45E+00	4.41E+00	1.88E+00	4.24E-01	2.16E-02
TE	133	3.20E+01	6.42E-08	9.61E-16	1.44E-23	0.00E+00	0.00E+00	0.00E+00
TE	134	5.69E+01	2.42E-09	1.03E-19	4.40E-30	0.00E+00	0.00E+00	0.00E+00
XE	133	6.76E+00	8.00E+00	7.97E+00	7.43E+00	4.68E+00	1.88E+00	2.97E-01
XE	135	1.96E+01	1.01E+01	2.19E+00	3.97E-01	2.91E-04	7.99E-10	5.94E-21
XE	138	5.26E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y	91	7.71E-01	9.16E-01	9.32E-01	9.25E-01	8.84E-01	8.13E-01	6.89E-01
Y	92	2.81E+01	1.25E+00	1.35E-02	1.28E-04	8.81E-13	4.51E-27	0.00E+00
Y	93	2.86E+01	5.64E+00	1.09E+00	2.09E-01	2.88E-04	2.83E-09	2.74E-19
Y	94	5.26E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y	95	5.35E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Y	91M	1.41E+01	2.93E+00	5.08E-01	8.81E-02	8.00E-05	3.80E-10	8.56E-21
ZR	95	9.72E-01	9.68E-01	9.58E-01	9.47E-01	9.07E-01	8.41E-01	7.23E-01
ZR	97	2.18E+01	8.15E+00	3.05E+00	1.14E+00	2.22E-02	2.27E-05	2.34E-11

CHAPTER 13 APPENDIX E

Maximum Activity Available for Release
One TRIGA Element at 86.42 W for 40 Years
(Release fractions: 1E-04 for halogens and noble gases, 1E-06 for particulates)

		Potential Activity Release (µCi) Time post discharge (days)							Inhalation ALI	DAC	Initial Bay Concentration
		0	1	2	3	7	14	28	µCi	µCi/cm³	µCi/cm³
AG	111	0.0	0.0	0.0	0.0	0.0	0.0	0.0	9.E+02	4.E-07	3.5E-12
BA	140	4.5	4.3	4.1	3.8	3.1	2.1	1.0	1.E+03	6.E-07	1.1E-09
BA	137	2.5	2.5	2.5	2.5	2.5	2.5	2.5	na	na	6.2E-10
CE	141	4.3	4.2	4.1	4.0	3.7	3.2	2.4	6.E+02	2.E-07	1.0E-09
CE	143	4.3	2.6	1.6	1.0	0.1	0.0	0.0	2.E+03	7.E-07	1.1E-09
CE	144	3.9	3.9	3.9	3.9	3.9	3.8	3.7	1.E+01	6.E-09	9.7E-10
CS	137	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.E+02	6.E-08	6.6E-10
EU	155	0.0	0.0	0.0	0.0	0.0	0.0	0.0	9.E+01	4.E-08	5.8E-12
EU	156	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.E+02	2.E-07	2.4E-12
I	131	204.6	190.1	175.3	161.4	114.9	62.9	18.8	5.E+01	2.E-08	5.0E-08
I	132	307.3	255.3	206.4	166.8	71.2	16.0	0.8	8.E+03	3.E-06	7.5E-08
I	133	490.5	226.5	101.8	45.7	1.9	0.0	0.0	3.E+02	1.E-07	1.2E-07
I	135	458.6	37.0	3.0	0.2	0.0	0.0	0.0	2.E+03	7.E-07	1.1E-07
KR	85	18.1	18.1	18.1	18.1	18.1	18.1	18.1		1.E-04	4.5E-09
KR	85M	92.2	2.3	0.1	0.0	0.0	0.0	0.0		2.E-05	2.3E-08
LA	140	4.5	4.5	4.4	4.2	3.5	2.4	1.1	1.E+03	5.E-07	1.1E-09
LA	141	4.3	0.1	0.0	0.0	0.0	0.0	0.0	9.E+03	4.E-06	1.0E-09
MO	99	4.4	3.4	2.6	2.1	0.7	0.1	0.0	1.E+03	6.E-07	1.1E-09
NB	95	4.6	4.6	4.6	4.6	4.6	4.6	4.4	1.E+03	5.E-07	1.1E-09
NB	97	4.3	1.6	0.6	0.2	0.0	0.0	0.0	7.E+04	3.E-05	1.0E-09
NB	95M	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.E+03	9.E-07	8.0E-12
NB	97M	4.0	1.5	0.6	0.2	0.0	0.0	0.0	na	na	9.9E-10
ND	147	1.6	1.5	1.4	1.4	1.1	0.7	0.3	8.E+02	4.E-07	4.0E-10
PM	147	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.E+02	5.E-08	4.0E-10
PM	149	0.8	0.6	0.4	0.3	0.1	0.0	0.0	2.E+03	8.E-07	1.9E-10
PM	151	0.3	0.2	0.1	0.1	0.0	0.0	0.0	3.E+03	1.E-06	7.5E-11
PR	143	4.3	4.3	4.2	4.0	3.3	2.3	1.1	7.E+02	3.E-07	1.1E-09
PR	144	3.9	3.9	3.9	3.9	3.9	3.8	3.7	1.E+05	5.E-05	9.7E-10
PR	145	2.8	0.2	0.0	0.0	0.0	0.0	0.0	8.E+03	3.E-06	7.0E-10
PR	144	0.0	0.0	0.0	0.0	0.0	0.0	0.0	na	na	1.2E-11
RH	105	0.7	0.5	0.3	0.2	0.0	0.0	0.0	6.E+03	2.E-06	1.8E-10
RH	106	0.3	0.3	0.3	0.3	0.3	0.3	0.3	na	na	6.9E-11
RH	103	2.0	2.0	2.0	1.9	1.8	1.6	1.2	1.E+06	5.E-04	5.0E-10
RU	103	2.3	2.2	2.2	2.2	2.0	1.8	1.4	6.E+02	3.E-07	5.6E-10
RU	105	0.7	0.0	0.0	0.0	0.0	0.0	0.0	1.E+04	5.E-06	1.8E-10
RU	106	0.3	0.3	0.3	0.3	0.3	0.3	0.3	1.E+01	5.E-09	6.9E-11
SB	125	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.E+02	2.E-07	5.3E-12
SB	127	0.1	0.1	0.1	0.1	0.0	0.0	0.0	9.E+02	4.E-07	2.3E-11
SB	129	0.5	0.0	0.0	0.0	0.0	0.0	0.0	9.E+03	4.E-06	1.1E-10
SM	151	0.1	0.1	0.1	0.1	0.1	0.1	0.1	1.E+02	4.E-08	2.0E-11
SM	153	0.1	0.1	0.1	0.0	0.0	0.0	0.0	3.E+03	1.E-06	2.9E-11
SN	125	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.E+02	1.E-07	2.4E-12
SR	89	3.5	3.5	3.4	3.4	3.2	2.9	2.4	1.E+02	6.E-08	8.6E-10
SR	90	2.6	2.6	2.6	2.6	2.6	2.6	2.6	4.E+00	2.E-09	6.3E-10
SR	91	4.3	0.7	0.1	0.0	0.0	0.0	0.0	4.E+03	1.E-06	1.0E-09
SR	92	4.4	0.0	0.0	0.0	0.0	0.0	0.0	7.E+03	3.E-06	1.1E-09
TC	99M	3.8	3.3	2.5	2.0	0.7	0.1	0.0	2.E+05	6.E-05	9.4E-10
TE	127	0.1	0.1	0.1	0.1	0.0	0.0	0.0	2.E+04	7.E-06	2.3E-11
TE	129	0.5	0.1	0.0	0.0	0.0	0.0	0.0	6.E+04	3.E-05	1.1E-10
TE	131	1.8	0.0	0.0	0.0	0.0	0.0	0.0	5.E+03	2.E-06	4.5E-10
TE	132	3.1	2.5	2.0	1.6	0.7	0.2	0.0	2.E+02	9.E-08	7.5E-10
TE	127	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.E+02	1.E-07	3.2E-12
TE	129	0.1	0.1	0.1	0.1	0.1	0.1	0.0	2.E+02	1.E-07	1.7E-11
TE	131	0.3	0.2	0.1	0.1	0.0	0.0	0.0	4.E+02	2.E-07	6.4E-11
XE	133	490.7	473.1	434.5	390.3	236.8	94.5	14.9		1.E-04	1.2E-07
XE	135	477.0	174.2	35.8	6.4	0.0	0.0	0.0		1.E-05	1.2E-07
XE	131	2.3	2.3	2.3	2.2	2.1	1.7	1.0		4.E-04	5.6E-10
XE	133	14.3	13.1	10.7	8.3	2.6	0.3	0.0		1.E-04	3.5E-09
XE	135	82.3	5.9	0.5	0.0	0.0	0.0	0.0		9.E-06	2.0E-08
Y	90	2.6	2.6	2.6	2.6	2.6	2.6	2.6	1.E+02	5.E-08	6.3E-10
Y	91	4.3	4.2	4.2	4.1	4.0	3.6	3.1	8.E+03	3.E-06	1.0E-09
Y	92	4.4	0.1	0.0	0.0	0.0	0.0	0.0	2.E+03	1.E-06	1.1E-09
Y	93	4.7	0.9	0.2	0.0	0.0	0.0	0.0	2.E+03	1.E-06	1.2E-09
Y	91M	2.5	0.5	0.1	0.0	0.0	0.0	0.0	2.E+05	7.E-05	6.1E-10
ZR	95	4.6	4.6	4.5	4.5	4.3	4.0	3.4	1.E+02	5.E-08	1.1E-09
ZR	97	4.3	1.6	0.6	0.2	0.0	0.0	0.0	1.E+03	5.E-07	1.0E-09

CHAPTER 13 APPENDIX F

Maximum Activity Available for Release
One TRIGA Element at 31.125 kW, 8 h/d, 5 Days
(Release fractions: 1E-04 for halogens and noble gases, 1E-06 for particulates)

	Potential Activity Release (µCi) Time post discharge							ALI	DAC	Initial Bay Concentration	
	0	1 d	2 d	3 d	7 d	14 d	28 d	µCi	µCi/cm³	µCi/cm³	Issue
AS 78	3.75	0	0	0	0	0	0	2.E+04	9.E-06	9.25E-10	NA
BA 139	1604.3	0	0	0	0	0	0	3.E+04	1.E-05	4.00E-07	NA
BA 140	127.25	120.5	114.3	108.3	87.25	59.5	28	1.E+03	6.E-07	3.00E-08	NA
BA 141	1480.3	0	0	0	0	0	0	7.E+04	3.E-05	3.75E-07	NA
BA 142	1470.5	0	0	0	0	0	0	1.E+05	6.E-05	3.50E-07	NA
BR 83	12110	12.5	0	0	0	0	0	6.E+04	3.E-05	3.00E-06	NA
BR 84	25386	0	0	0	0	0	0	6.E+04	2.E-05	6.25E-06	NA
CE 141	47.25	52.5	51.5	50.25	46.25	39.75	29.5	6.E+02	2.E-07	1.15E-08	NA
CE 143	553.5	340.75	205.8	124.3	16.5	0.5	0	2.E+03	7.E-07	1.35E-07	NA
CE 144	5.5	5.5	5.5	5.5	5.5	5.5	5.25	1.E+01	6.E-09	1.38E-09	NA
CS 138	1711.8	0	0	0	0	0	0	6.E+04	2.E-05	4.25E-07	NA
I 131	8489.3	8245.3	7694	7131	5124	2810	840.5	5.E+01	2.E-08	2.08E-06	103.8
I 132	23281	20943	16930	13686	5838	1317	67	8.E+03	3.E-06	5.75E-06	1.9
I 133	69950	33529	15072	6772	276.3	1	0	3.E+02	1.E-07	1.73E-05	172.5
I 134	192228	0	0	0	0	0	0	5.E+04	2.E-05	4.75E-05	2.4
I 135	98750	7980	644.3	52	0	0	0	2.E+03	7.E-07	2.43E-05	34.6
KR 87	64498	0.25	0	0	0	0	0	na	5.E-06	1.58E-05	3.2
KR 88	79048	225.75	0.75	0	0	0	0	na	2.E-06	1.95E-05	9.8
KR 83M	9953.3	47.75	0	0	0	0	0	na	1.E-02	2.45E-06	NA
KR 85M	23368	580.25	14.25	0.25	0	0	0	na	2.E-05	5.75E-06	NA
LA 140	67.75	86.75	97	101.8	96	68.5	32.25	1.E+03	5.E-07	1.68E-08	NA
LA 141	1140.3	18.5	0.25	0	0	0	0	9.E+03	4.E-06	2.75E-07	NA
LA 142	1454.5	0	0	0	0	0	0	2.E+04	9.E-06	3.50E-07	NA
LA 143	1493.3	0	0	0	0	0	0	9.E+04	4.E-05	3.75E-07	NA
MO 99	395.5	307.5	239	185.8	67.75	11.5	0.25	1.E+03	6.E-07	9.75E-08	NA
MO 101	1272.8	0	0	0	0	0	0	1.E+05	6.E-05	3.00E-07	NA
NB 97	596.75	246.75	92.25	34.5	0.75	0	0	7.E+04	3.E-05	1.48E-07	NA
NB 98	1463.5	0	0	0	0	0	0	5.E+04	2.E-05	3.50E-07	NA
ND 147	53.75	51	48	45	35	22.5	9.5	8.E+02	4.E-07	1.33E-08	NA
ND 149	263.25	0	0	0	0	0	0	2.E+04	1.E-05	6.50E-08	NA
ND 151	105.5	0	0	0	0	0	0	2.E+05	8.E-05	2.50E-08	NA
PD 109	3.5	1	0.25	0	0	0	0	5.E+03	2.E-06	8.75E-10	NA
PM 151	41	23.25	13	7.25	0.75	0	0	3.E+03	1.E-06	1.00E-08	NA
PR 143	69.5	88	97	100.3	91.25	65	31.75	7.E+02	3.E-07	1.70E-08	NA
PR 144	5.5	5.5	5.5	5.5	5.5	5.5	5.25	1.E+05	5.E-05	1.38E-09	NA
PR 145	638	40.25	2.5	0.25	0	0	0	8.E+03	3.E-06	1.58E-07	NA
PR 147	572	0	0	0	0	0	0	2.E+05	8.E-05	1.40E-07	NA
RB 88	799	2.5	0	0	0	0	0	6.E+04	3.E-05	1.95E-07	NA
RH 105	77.5	65.25	41.25	25.75	4	0.25	0	6.E+03	2.E-06	1.90E-08	NA
RH 107	41	0	0	0	0	0	0	2.E+05	1.E-04	1.00E-08	NA
RU 103	22.5	22	21.75	21.25	19.75	17.5	13.75	6.E+02	3.E-07	5.50E-09	NA
RU 105	187.5	4.75	0	0	0	0	0	1.E+04	5.E-06	4.50E-08	NA
SB 127	6.5	6	5	4	2	0.5	0	9.E+02	4.E-07	1.63E-09	NA
SE 81	53	0	0	0	0	0	0	2.E+05	1.E-04	1.30E-08	NA
SE 83	49.5	0	0	0	0	0	0	1.E+05	5.E-05	1.23E-08	NA
SM 153	12.75	9	6.25	4.5	1	0	0	3.E+03	1.E-06	3.25E-09	NA
SM 155	8.25	0	0	0	0	0	0	2.E+05	9.E-05	2.05E-09	NA
SN 127	19.75	0	0	0	0	0	0	2.E+04	8.E-06	4.75E-09	NA
SN 128	83.75	0	0	0	0	0	0	3.E+04	1.E-05	2.05E-08	NA
SR 89	28	28	27.5	27.25	25.75	23.25	19.25	1.E+02	6.E-08	6.75E-09	NA
SR 91	796.25	138.75	24	4.25	0	0	0	4.E+03	1.E-06	1.95E-07	NA
SR 92	1325.8	2.75	0	0	0	0	0	7.E+03	3.E-06	3.25E-07	NA
TC 101	1273	0	0	0	0	0	0	3.E+05	1.E-04	3.00E-07	NA
TC 104	459.75	0	0	0	0	0	0	7.E+04	3.E-05	1.13E-07	NA
TE 127	4.75	5.25	4.75	4	2	0.5	0	2.E+04	7.E-06	1.18E-09	NA
TE 129	93.75	3.5	0.5	0.5	0.5	0.5	0.25	6.E+04	3.E-05	2.30E-08	NA
TE 131	628.75	4.5	2.5	1.5	0.25	0	0	5.E+03	2.E-06	1.55E-07	NA

	Potential Activity Release (µCi)							ALI	DAC	Initial Bay Concentration	
	Time post discharge										
	0	1 d	2 d	3 d	7 d	14 d	28 d	µCi	µCi/cm	µCi/cm	issue
TE 132	250.75	203	164.3	132.8	56.75	12.75	0.75	2.E+02	9.E-08	6.25E-08	NA
TE 133	964.25	0	0	0	0	0	0	2.E+04	9.E-06	2.38E-07	NA
TE 134	1712.5	0	0	0	0	0	0	2.E+04	1.E-05	4.25E-07	NA
XE 133	20362	24106	24010	22389	14111	5673	894.75	na	1.E-04	5.00E-06	NA
XE 135	58955	30517	6594	1195	1	0	0	na	1.E-05	1.45E-05	1.5
XE 138	158548	0	0	0	0	0	0	na	4.E-06	4.00E-05	10.0
Y 91	23.25	27.5	28	28	26.5	24.5	20.75	1.E+02	5.E-08	5.75E-09	NA
Y 92	846.75	37.75	0.5	0	0	0	0	8.E+03	3.E-06	2.08E-07	NA
Y 93	861.25	169.75	32.75	6.25	0	0	0	2.E+03	1.E-06	2.10E-07	NA
Y 94	1583.8	0	0	0	0	0	0	8.E+04	3.E-05	4.00E-07	NA
Y 95	1613	0	0	0	0	0	0	1.E+05	6.E-05	4.00E-07	NA
Y 91M	423.5	88	15.25	2.75	0	0	0	2.E+05	7.E-05	1.05E-07	NA
ZR 95	29.25	29.25	28.75	28.5	27.25	25.25	21.75	1.E+02	5.E-08	7.25E-09	NA
ZR 97	657	245.5	91.75	34.25	0.75	0	0	1.E+03	5.E-07	1.60E-07	NA

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CHAPTER 14

TECHNICAL SPECIFICATIONS

SEPARATELY PUBLISHED AS REPORT KSUNE-00-05
DEPARTMENT OF MECHANICAL AND NUCLEAR
ENGINEERING, KANSAS STATE UNIVERSITY

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15. Financial Qualifications

15.1 Financial Ability to Operate the Reactor

The KSU TRIGA Reactor has been in continuous operation since 1962. From 1962 until 1997, the Department of Nuclear Engineering operated the Reactor at the University. In 1997, the Departments of Mechanical and Nuclear Engineering merged to become a single department. The budget for the reactor is integrated into the department budget. The department has substantial resources, in that 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.

Appendix A contains information related to demonstrating financial ability of Kansas State University to operate the K-State reactor:

- Fuel Cycle Assistance Contract (including compliance with the Nuclear Waste Policy Act of 1982) and Amendments thereto
- Support and Responsibility for Decommissioning
- Nuclear Liability Insurance Indemnity Agreement and Amendments thereto

15.2 Financial Ability to Decommission the Facility

As indicated in Appendix C, the university administration intends to support extension of the operating license of the TRIGA reactor. Whenever a decision is reached to decommission the reactor, the university will request legislative appropriation of funds, or otherwise provide funds sufficiently in advance of decommissioning to prevent delay of required activities.

15.3 Bibliography

NUREG/CR-1756 "Technology, Safety, and Costs of Decommissioning Reference Nuclear Research and Test Reactors," U.S. Nuclear Regulatory Commission,, March 1982; Addendum, July 1983.

Appendix 15.A, Financial Statements

INEL
Idaho National Engineering Laboratory

July 6, 1988

Mr. Warren Strauss, Comptroller
Kansas State University
Manhattan, Kansas 66506

TASK ORDER NO. 1 UNDER SUBCONTRACT NO. C88-101856 (DE-AC07-76ER02082) WITH THE KANSAS STATE UNIVERSITY FOR REACTOR FUEL ASSISTANCE - AR-69-88

Dear Mr. Strauss:

1. This Task Order No. 1, effective July 1, 1988, is in accordance with:

   A. The terms of Subcontract No. C88-101856.

   B. The Subcontractor's letter to continue the Reactor Fuel Assistance Program.

2. As a result of this Task Order, the Subcontractor will provide for utilization of the reactor owned by the Subcontractor in a program of education and training of students in nuclear science and engineering, and for faculty and student research. The contract provides for the continued possession and use of DOE-owned nuclear materials, including: enriched uranium in reactor fuel without charge of use, burn-up, or reprocessing while used for research, education and training purposes. A report shall be submitted annually on the use of the leased material during the past year.

3. EG&G Idaho Administration.

   A. Contractual responsibilities under this Task Order shall be administered by Ann Rydalch.

   B. All work to be performed under this Task Order shall be under the technical jurisdiction and direction of Keith Brown.

4. Subcontractor Administration.

   A. The Subcontractor's contract responsibilities under this Task Order shall be administered by Warren Strauss.

EG&G P.O. Box 1625 Idaho Falls, ID 83415

Mr. Warren Strauss
July 6, 1988
AR-69-88
Page 2

B. Technical Administration: All work to be performed under this Task Order shall be under the technical jurisdiction and direction of the Director of the Reactor Fuel Assistance Program.

5. Subject to the terms and conditions of Subcontract No. C88-101856, the Subcontractor shall use all reasonable efforts to complete the Scope of Work by October 30, 1993.

6. The total estimated cost and ceiling amount for this Task Order is -0-. Funds may be provided at any time under this agreement, if, and as needed.

   The amount currently obligated for performance of this Task Order is the maximum amount obligated and must not be exceeded without the prior approval of the Subcontract Administrator designated herein. If at any time the Subcontractor has reason to believe that the total cost to the Contractor will be greater than the obligated amount and/or ceiling amount shown on this Task Order, the Subcontractor shall notify the Contractor in writing to that effect, giving the revised estimate of such total cost for the performance of the Task Order.

7. Title to all special nuclear material loaned to the Subcontractor under this Task Order shall at all times be and remain in the United States Government.

8. The Contractor will not charge the Subcontractor for such amount of material as is (1) consumed in the operation of the facility until expiration of this Task Order, and (2) not recovered in reprocessing subsequent to the ultimate return of the special nuclear material, will waive all use charges on the material until expiration of this Task Order; and will, at the Contractor's option, either reimburse the Subcontractor for cost of shipping spent fuel for chemical reprocessing or accept return of the material at the Subcontractor's facility. The Subcontractor's obligation is to return material in the form defined above, as affected by use for the contract purposes, and, therefore, the Subcontractor has no responsibility for reprocessing of such material.

   Except as otherwise provided herein, the Subcontractor is responsible for and will pay the Contractor any charges imposed by the Contractor for material delivered to the Subcontractor and not ultimately returned to the Contractor.

Mr. Warren Strauss
July 6, 1988
AR-89-88
Page 3

9. Upon expiration of the terms of this Task Order, the Subcontractor may continue to use the material under its license and in such event, the Contractor has the option to require the Subcontractor to be responsible for all charges required by the Contractor relating to the material from the date of expiration, including charges for use and consumption.

10. Notwithstanding any other provision of this contract or Task Order, the Government shall not be responsible for or have any obligation to the Subcontractor for decontamination or decommissioning (D&D) of any of the Subcontractor's facilities.

11. The Subcontractor is responsible for verification and accountability of any DOE-Owned nuclear materials in its possession. A report shall be submitted annually on the use of the loaned material during the past year. In the event the terms and conditions of this contract or Task Order are not in agreement with NRC rules and regulations, the NRC requirements will take precedence.

12. Invoices for services provided under this Task Order where applicable, may be accompanied by a certified statement of costs in the format set forth in Appendix A; however, must accompany the final invoice.

The original and two copies of this Task Order are forwarded. Please execute the original and one copy and return them to EG&G Idaho, Inc. The remaining copy, executed by EG&G Idaho, Inc., may be retained for your records.

EG&G IDAHO, INC.	KANSAS STATE UNIVERSITY
By Ann Rydalch	By Warren Strauss
Title Ann Rydalch Subcontract Administrator	Title Warren Strauss Controller JUL 27 1988

Js

cc: Keith Brown

INEL
Idaho National Engineering Laboratory

November 29, 1993

Mr. Richard E. Faw
Kansas State University
Manhattan, KS 66506

MODIFICATION NO. 1 TO TASK ORDER NO. 1 UNDER SUBCONTRACT NO. C88-101856 (DE-AC07-76ER02426) WITH KANSAS STATE UNIVERSITY FOR REACTOR FUEL ASSISTANCE - KF-201-93 2052 95

Dear Mr. Faw:

This Modification No. 1 to Task Order No. 1 under Subcontract No. C88-101856, effective November 1, 1993 is to:

1. Expand the scope of work to include additional compliance requirements,
2. Modify the EG&G Idaho Administration,
3. Extend the term five years.

NOW, THEREFORE, the parties mutually agree to the following:

1. Item 2 - is expanded to include the Attachment A - UNIVERSITY REACTOR FUEL ASSISTANCE PROGRAM - Subcontract Materials Management Requirements.
2. Item 3 - EG&G Idaho Administration is modified to change the contractual representative and the technical representative as follows:
   A. Contractual Responsibilities under this task order modification shall be administered by Ken Feliciano.
   B. All work to be performed under this task order modification shall be under the technical jurisdiction and direction of Anthony Vinnola.
3. Item 5 is modified to extend the term five years. Therefore, Item 5 is modified to read as follows:

   Subject to the terms and conditions of Subcontract No. C88-101856, the subcontractor shall use all reasonable efforts to complete the Scope of Work by October 30, 1998.

RECEIVED
DEC 07 1993

EG&G Idaho, Inc. P.O. Box 1625 Idaho Falls, ID 83415 -3932 Engineering Experiment Station

Mr. Richard E. Faw
November 29, 1993
KF-201-93
Page 2

Except only as changed by this Modification No. 1, all of the terms and conditions of Task Order No. 1 Under Subcontract No. C88-101856 shall remain in full force and effect.

Two copies of this Modification No. 1 to Task Order No. 1, executed by EG&G Idaho, Inc., are forwarded. Please execute one copy and return it to EG&G Idaho, Inc. One copy may be retained for your records.

KANSAS STATE UNIVERSITY	EG&G IDAHO, INC.
By James Shanteau	By Ken Feliciano
pp James Shanteau	Ken Feliciano
Title Interim Assoc. Vice Provost for Research	Title Subcontract Administrator
Date	Date 11-30-93

KSU
KANSAS STATE UNIVERSITY

Office of the Provost

Anderson Hall
Manhattan, Kansas 66506
913-532-6224

July 17, 1990

United States Nuclear
Regulatory Commission, Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76012

RE: License R-88
Docket 50-188

Dear Sirs:

This concerns the ultimate decommissioning of the Kansas State University TRIGA Nuclear Reactor Facility, currently licensed for operation by the University until August 15, 2001. Pursuant to Federal Regulations, Title 10, Part 50, this is to assure that the University, an entity of the State of Kansas, will obtain the funds for decommissioning when necessary.

It is our intention, at the appropriate time, to request an extension of the Reactor Facility operating license beyond the August 2001 termination of the current license. Nevertheless, whenever a decision is reached to decommission the facility, the University will request legislative appropriation of funds, or otherwise provide funds sufficiently in advance of decommissioning to prevent delay of required activities.

Enclosed is a preliminary cost estimate for decommissioning of the Reactor Facility. The cost estimate was prepared on the basis of the methods described in the May 1989 Nuclear Regulatory Commission Draft Regulatory Guide on "Assuring the Availability of Funds for Decommissioning Nuclear Reactors." Cost escalation factors other than those prescribed in publications cited by the Guide are based on U.S. Department of Labor statistics.

With assurance that I have the authority to sign this statement of intent, I am

Yours truly,

James R. Coffman

A.K. James R. Coffman
Provost

Encl.

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

NOV 27 1989

Docket No. 50-188

Dr. Richard E. Faw, Director
Nuclear Reactor Facility
Department of Nuclear Engineering
Ward Hall
Kansas State University
Manhattan, Kansas 66506

Dear Dr. Faw:

SUBJECT: KANSAS STATE UNIVERSITY - AMENDMENT TO INDEMNITY AGREEMENT

We are enclosing herewith an amendment to your indemnity agreement reflecting the changes to 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," effective July 1, 1989. The amendments to Part 140 reflect the increase from $160 million to $200 million in the primary layer of nuclear energy liability insurance provided by American Nuclear Insurers and Mutual Atomic Energy Liability Underwriters. The amendment also conforms to changes made to the Price-Anderson Act by "The Price-Anderson Amendments Act of 1988" which was enacted on August 20, 1988.

Please signify your acceptance of the amendment to your indemnity agreement in the space provided and return one signed copy to Ira Dinitz, Senior Insurance/Indemnity Specialist, U. S. Nuclear Regulatory Commission, Mail Stop 12E-4, Washington, D.C. 20555. If you have any questions about the foregoing, please contact Mr. Dinitz at 301-492-1289.

Sincerely,

Theodore S. Michaels

Theodore S. Michaels, Project Manager
Non-Power Reactor, Decommissioning and
Environmental Project Directorate
Division of Reactor Projects - III, IV,
V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Amendment to Indemnity
Agreement

cc w/enclosure:
See next page

Recd
18 Jan 90

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Kansas State University

Docket No. 50-188

cc: Office of the Governor
State of Kansas
Topeka, Kansas 66612

Mayor of Manhattan
P. O. Box 748
Manhattan, Kansas 66502

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

Docket No. 50-188

Amendment to Indemnity Agreement No. E-1
Amendment No. 13

Effective July 1, 1989, Indemnity Agreement No. E-1, between Kansas State University of Argriculture and Applied Science, and the Atomic Energy Commission, dated June 18, 1962, as amended, is hereby further amended as follows:

The amount "$160,000,000" is deleted wherever it appears and the amount "$200,000,000" is substituted therefor.

The amount "$124,000,000" is deleted wherever it appears and the amount "$155,000,000" is substituted therefor.

The amount "$36,000,000" is deleted wherever it appears and the amount "$45,000,000" is substituted therefor.

Paragraph 1, Article I is modified to read as follows:

1. "Nuclear reactor," "byproduct material," "person," "source material," "special nuclear material," and "precautionary evacuation" shall have the meanings given them in the Atomic Energy Act of 1954, as amended, and the regulations issued by the Commission.

The definition of "public liability" in paragraph 7, Article I is deleted, and the following is substituted therefor:

"Public liability" means any legal liability arising out of or resulting from a nuclear incident or precautionary evacuation (including all reasonable additional costs incurred by a State or a political subdivision of a State, in the course or responding to a nuclear incident or precautionary evacuation), except (1) claims under State or Federal Workmen's Compensation Acts of employees of persons indemnified who are employed (a) at the location or, if the nuclear incident occurs in the course of transportation of the radioactive material, on the transporting vehicle, and (b) in connection with the licensee's possession, use or transfer of the radioactive material; (2) claims arising out of an act of war; and (3) claims for loss of, or damage to, or loss of use of (a) property which is located at the location and used in connection with the licensee's possession, use, or transfer of the radioactive material, and (b) if the nuclear incident occurs in the course of transportation of the radioactive material, the transporting vehicle, containers used in such transportation, and the radioactive material.

Paragraph 4(c), Article II is revised to read as follows:

(c) Any issue or defense based on any statute of limitations if suit is instituted within three years from the date on which the claimant first knew, or reasonably could have known, of his injury or damage and the cause thereof.

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Paragraph 1, Article IV is revised to read as follows:

1. When the Commission determines that the United States will probably be required to make indemnity payments under the provisions of this agreement, the Commission shall have the right to collaborate with the licensee and other persons indemnified in the settlement and defense of any claim (including such legal costs of the licensee as are approved by the Commission) and shall have the right (a) to require the prior approval of the Commission for the settlement or payment of any claim or action asserted against the licensee or other person indemnified for public liability or damage to property of persons legally liable for the nuclear incident which claim or action the licensee or the Commission may be required to indemnify under this agreement; and (b) to appear through the Attorney General of the United States on behalf of the licensee or other person indemnified, take charge of such action and settle or defend any such action. If the settlement or defense of any such action or claim is undertaken by the Commission, the licensee shall furnish all reasonable assistance in effecting a settlement or asserting a defense.

In paragraph 1, Article VIII, the amount "$5,000,000" is deleted and the amount "$63,000,000" is substituted therefor.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O. Thomas, Chief
Policy Development and Technical Support Branch
Program Management Policy Development
and Analysis Staff
Office Nuclear Reactor Regulation

Accepted 7/11/, 19~~89~~ 90

By James R. Coleman
Kansas State University of
Agriculture and Applied Science

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

Docket No. 50-188

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Amendment No. 13

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Paragraph 1, Article I is modified to read as follows:

1. "Nuclear reactor," "byproduct material," "person," "source material," "special nuclear material," and "precautionary evacuation" shall have the meanings given them in the Atomic Energy Act of 1954, as amended, and the regulations issued by the Commission.

The definition of "public liability" in paragraph 7, Article I is deleted, and the following is substituted therefor:

"Public liability" means any legal liability arising out of or resulting from a nuclear incident or precautionary evacuation (including all reasonable additional costs incurred by a State or a political subdivision of a State, in the course or responding to a nuclear incident or precautionary evacuation), except (1) claims under State or Federal Workmen's Compensation Acts of employees of persons indemnified who are employed (a) at the location or, if the nuclear incident occurs in the course of transportation of the radioactive material, on the transporting vehicle, and (b) in connection with the licensee's possession, use or transfer of the radioactive material; (2) claims arising out of an act of war; and (3) claims for loss of, or damage to, or loss of use of (a) property which is located at the location and used in connection with the licensee's possession, use, or transfer of the radioactive material, and (b) if the nuclear incident occurs in the course of transportation of the radioactive material, the transporting vehicle, containers used in such transportation, and the radioactive material.

Paragraph 4(c), Article II is revised to read as follows:

(c) Any issue or defense based on any statute of limitations if suit is instituted within three years from the date on which the claimant first knew, or reasonably could have known, of his injury or damage and the cause thereof.

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2

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1. When the Commission determines that the United States will probably be required to make indemnity payments under the provisions of this agreement, the Commission shall have the right to collaborate with the licensee and other persons indemnified in the settlement and defense of any claim (including such legal costs of the licensee as are approved by the Commission) and shall have the right (a) to require the prior approval of the Commission for the settlement or payment of any claim or action asserted against the licensee or other person indemnified for public liability or damage to property of persons legally liable for the nuclear incident which claim or action the licensee or the Commission may be required to indemnify under this agreement; and (b) to appear through the Attorney General of the United States on behalf of the licensee or other person indemnified, take charge of such action and settle or defend any such action. If the settlement or defense of any such action or claim is undertaken by the Commission, the licensee shall furnish all reasonable assistance in effecting a settlement or asserting a defense.

In paragraph 1, Article VIII, the amount "$5,000,000" is deleted and the amount "$63,000,000" is substituted therefor.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas, Chief
Policy Development and Technical Support Branch
Program Management Policy Development
and Analysis Staff
Office Nuclear Reactor Regulation

Accepted 7/11/, 1989

By
Kansas State University of
Agriculture and Applied Science

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Safety Analysis Report: Additional Information

Kansas State University

TRIGA Mark II Nuclear Reactor Facility

License R-88

Docket 50-188

21 December 2004

Department of Mechanical and Nuclear Engineering
Kansas State University
302 Rathbone Hall
Manhattan, KS 66506

K-State Nuclear Reactor Facility
110 Ward Hall
Manhattan, KS 66506

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
CHAPTER 1						
1-1	1.1	This report is based on the *Kansas State University TRIGA Mark II Hazards Summary Report* (1961) for the initial operation of the reactor at 100 kW thermal power, the 1968 Safety Analysis Report and Safety Evaluation Report for license amendment to allow 250 kW steady state thermal power (250 MW pulsing capability), and subsequent analyses supporting steady state operations at 500 kW (pulsing to a nominal $3.00 reactivity insertion.	1-1	1.1	This report is based on the *Kansas State University TRIGA Mark II Hazards Summary Report* (1961) for the initial operation of the reactor at 100 kW thermal power, the 1968 Safety Analysis Report and Safety Evaluation Report for license amendment to allow 250 kW steady state thermal power (250 MW pulsing capability), and subsequent analyses supporting steady state operations at 1,250 kW (pulsing to a nominal $3.00 reactivity insertion. A $3.00 reactivity insertion is expected to result in a peak thermal power of approximately 1,340 MW). Based on proposed reactivity limits, the KSU reactor will only be able to achieve about ½ the proposed maximum power level for steady state operation; therefore thermal-hydraulic and source term calculations are conservative by a factor of 2 in analysis.	Clarification
1-1	1.1	This report addresses safety issues associated with operation of the reactor at 500 kW, including increased pulsing capabilities. This report reflects the as-built condition of the facility, and includes experience with the operation and performance of the reactor, radiation surveys, and personnel exposure histories related to operation of the reactor at 250 kW steady-state power. Where appropriate, radiological characteristics have been extrapolated to reflect operation at 500 KW. The consequence of routine generation of radioactive effluent and other waste products from steady state operation at 500 kW is addressed in Chapter 11. Radiation worker and public doses from radiation associated with routine operations are well within the limits of Title 10, Code of Federal Regulations, even under unrealistically conservative scenarios. The consequence of accident scenarios from operation at 500 KW steady-state power and pulsing is presented in Chapter 13.	1-1	1.1	This report addresses safety issues associated with operation of the reactor at 1,250 kW, including increased pulsing capabilities. The maximum excess reactivity permitted by Technical Specifications cannot achieve a continuous steady state power level greater than about 500 kW; therefore analysis performed for steady state operations at 1,250 kW is extremely conservative in evaluating consequences and characteristics of normal and accident scenarios. This report reflects the as-built condition of the facility, and includes experience with the operation and performance of the reactor, radiation surveys, and personnel exposure histories related to operation of the reactor at 250 kW steady-state power. Where appropriate, radiological characteristics have been extrapolated to reflect operation at 1,250 KW. The consequence of routine generation of radioactive effluent and other waste products from steady state operation at 1,250 kW is addressed in Chapter 11. Radiation worker and public doses from radiation associated with routine operations are well within the limits of Title 10, Code of Federal Regulations, even under unrealistically conservative scenarios. The consequence of accident scenarios from operation at 1,250 KW steady-state power and pulsing is presented in Chapter 13.	Power level for bounding analysis
1-1	1.1	The description of the reactor core and thermal hydraulic analysis presented in Chapter 4, the Secondary Cooling System in Chapter 5, and the Reactor Control System in Chapter 7 are based on 500 KW operations.	1-2	1.1	The description of the reactor core and thermal hydraulic analysis presented in Chapter 4, the Secondary Cooling System in Chapter 5, and the Reactor Control System in Chapter 7 are based on 1,250 KW operations.	Power level for bounding analysis

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
1-2	1.2.1	As of July, 1999, there were over 66 TRIGA reactors in use or under construction at universities, government and industrial laboratories, and medical centers in 24 countries. Historically, analysis and testing of TRIGA fuel has demonstrated that fuel cladding integrity is not challenged as long as stress on the cladding remains within yield strength at cladding temperature. Elevated TRIGA fuel temperatures evolve hydrogen from the zirconium matrix, with concomitant pressure buildup in the cladding. Therefore, the strength of the clad as a function of temperature establishes the upper limit on fuel temperature. Fuel temperature less than limiting values will ensure clad integrity (as evaluated in NUREG 1282) and therefore contain radioactive materials produced by fission in the reactor core.	1-2	1.2	As of July, 1999, there were over 70 TRIGA reactors in use or under construction at universities, government and industrial laboratories, and medical centers in 24 countries. Historically, analysis and testing of TRIGA fuel has demonstrated that fuel cladding integrity is not challenged as long as stress on the cladding remains within yield strength for the cladding temperature. Elevated TRIGA fuel temperatures evolve hydrogen from the zirconium matrix, with concomitant pressure buildup in the cladding. Therefore, the strength of the clad as a function of temperature establishes the upper limit on fuel temperature. Fuel temperature less than limiting values will ensure clad integrity (as evaluated in NUREG 1282) and therefore contain radioactive materials produced by fission in the reactor core.	Power level for bounding analysis
1-2	1.2.1 Para 2	1,000	1-2	1.2.1 Para 2	1,250	Power level for bounding analysis
1-2	1.2.1 para 3	.... Consequently, maximum possible power using TRIGA fuel is controlled by the amount of fuel loading. A minimum of eighty-three elements is required to allow operation at 500 kW. ...			.... Consequently, maximum possible power using TRIGA fuel is controlled by limiting the amount of fuel loading. A minimum of eighty-three elements is required to allow operation at 1,250 kW. ...	Power level for bounding analysis
1-2	1.2.2	As indicated in Chapter 11, radiation sources are discharged from the reactor facility in gaseous (airborne), liquid or solid form. These forms are treated individually in subsections of Chapter 11. Airborne radiation sources consist mainly of Argon-41, Nitrogen-16 and Tritium. Argon 41 is the major contributor to off site dose.	1-2	1.2.2	As indicated in Chapter 11, radiation sources are discharged from the reactor facility in gaseous (airborne), liquid or solid form. These forms are treated individually in subsections of Chapter 11. Airborne radiation sources consist mainly of Argon-41, Nitrogen-16 and Tritium, with Argon 41 the major contributor to off site dose. Limits on Argon-41 and Tritium are tabulated below, with Cesium 137, the other significant isotope of interest for the KSU reactor.	Clarification
1-3	1.2.2	NA	1-3	1.2.2	Added chart of 10CFR20 App B values & related information	Clarification

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
1-3	1.2.2	Argon 41 is the major contributor to radiation exposure incident to the operation of the K-State reactor. Argon 41 is attributed to neutron activation of natural argon (in air) in the reactor bay atmosphere, rotary specimen rack adjacent to the core, and dissolved in primary coolant. Argon 41 has 1.8 h half-life. Even under unrealistically conservative assumptions of operational conditions, the off site dose from Argon 41 is well within limits, and doses in the reactor bay are below the levels requiring controls of an airborne radioactivity area. Chapter 11 Appendix A shows peak off-site activity concentration during normal operations would be about 6×10^{-4} pCi/mL at 135 m downwind under slightly unstable atmospheric conditions, less than the effluent limit of 0.01 pCi/mL. A full year exposure at the maximum concentration would lead to an effective dose of only about 3 mrem, well within applicable limits	1-3	1.2.2	Argon 41 is the major contributor to radiation exposure incident to the operation of the K-State reactor. Argon 41 is attributed to neutron activation of natural argon (in air) in the reactor bay atmosphere, rotary specimen rack adjacent to the core, and dissolved in primary coolant. Argon 41 has 1.8 h half-life. Calculations based on 1,250 kW steady state continuous operations show that doses in the reactor bay remain below inhalation DAC. Using extremely conservative assumptions of operational conditions in concert with the worst-case wind stability class, the off site dose from Argon 41 is slightly less than 10% of the 10 mrem/year limit. A summation of all relative frequencies for winds under Pasquill stability category A (Table B-3) indicates frequency less than 0.6%, i.e., the contribution to off site doses from Argon 41 produced during a year of full power, steady state operations accounts for less than 0.6% of the total dose. All other atmospheric dispersion calculations show that the off site dose from Argon 41 is well within limits, and doses in the reactor bay are below the levels requiring controls of an airborne radioactivity area. Chapter 11 Appendix A shows peak off-site activity concentration during normal operations would be about 4.5×10^{-3} pCi/mL at 53 m downwind under extremely unstable atmospheric conditions, less than the effluent limit of 0.01 pCi/mL. A full year exposure to equilibrium argon concentration for 1,250 kW operations under normal atmospheric conditions would lead to an effective dose of less than 7 mrem, well within applicable limits.	

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
1-3	1.2.2	Nitrogen 16 is the major contributor to radiation fields directly over the reactor pool during operation. Nitrogen 16 is produced by a fast neutron reaction with oxygen (as a natural component of water in the core). Nitrogen 16 has a 7.1 second half-life, and consequently does not contribute to off-site dose. Chapter 11 shows very conservative calculations lead to an expected exposure rate of 40 mR/h at one meter above the center of the reactor tank during sustained operation at 500 kW thermal power. Measured exposure rates are about 20 mR/h at 250 kW operation. Therefore, radiation dose rates directly above the reactor pool during operation are within required levels for a radiation area.	1-3	1.2.2	Nitrogen 16 is the major contributor to radiation fields directly over the reactor pool during operation. Nitrogen 16 is produced by a fast neutron reaction with oxygen (as a natural component of water in the core). Nitrogen 16 has a 7.1 second half-life, and consequently does not remain at concentrations capable of contributing significantly to off-site dose. Chapter 11 shows very conservative calculations lead to an expected exposure rate of slightly less than 100 mrem/hr at one meter above the center of the reactor tank during sustained operation at 1,250 kW thermal power. The 22-foot level has radiation monitors directly above the pool and at the rail surrounding access to the pool. Measured exposure rates directly above the pool surface are about 20-30 mR/h at 250 kW operations, and measurements at the rail approach 2 mR/hr. During normal, steady state 500 kW operations dose rate can be expected to achieve 40-60 mrem/hr, and during steady state operations at 1,250 kW the area directly above the pool surface may become a high-radiation area. Therefore, radiation dose rates directly above the reactor pool during expected operations at levels up to 500 kW are within required levels for a radiation area as defined in 10CFR20, and additional administrative controls for access to the area directly above the reactor pool 500 kW to the maximum license power level of 1,250 kW may be required. Installed monitoring systems provide information necessary to identify appropriate access controls.	Power level for bounding analysis
1-3	1.2.2	Tritium is generated by sequential activation of hydrogen (in water) in the core area. Measured tritium specific activity in primary coolant is less than 10^{-3} µCi/g. If the reactor bay atmosphere were saturated with this water at 30°C, the water concentration in the air would be less than 3×10^{-5} g/mL and the activity concentration in the atmosphere, well below the 10CFR20 Appendix-B DAC and the atmospheric effluent limit. Even under the unrealistic assumption that the complete tritium inventory of the reactor pool is released into the reactor bay atmosphere, the tritium concentration would be within limits for an unrestricted area.	1-3	1.2.2	Tritium is generated by sequential activation of hydrogen (in water) in the core area. Measured tritium specific activity in primary coolant is less than 5×10^{-3} µCi/g. If the reactor bay atmosphere were saturated with this water at 30°C, the water concentration in the air would be less than 3×10^{-5} g/mL and the activity concentration in the atmosphere 1.5×10^{-7} µCi/ml, well below the DAC limit and well below the atmospheric effluent limit with the dilution factor of 200 for discharge from the top of the reactor bay. Even under the extremely conservative assumption that the complete tritium inventory of the reactor pool is released into the reactor bay atmosphere, the tritium concentration would be within limits for an unrestricted area.	Power level for bounding analysis

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
1-4	1.2.3.b p 1	Although total loss of reactor pool water is considered to be an extremely improbable event, calculations have been made to determine the maximum fuel temperature rise that could be expected to result from such an event taking place after long-term operation at full power of 500 kW under extraordinarily conservative assumptions and approximations, the maximum fuel temperature reached in a loss of coolant accident is less than 290°C, well below any safety limit for TRIGA reactor fuel.	1-6	1.2.3.b p 1	Although total loss of reactor pool water is considered to be an extremely improbable event, calculations have been made to determine the maximum fuel temperature rise that could be expected to result from such an event taking place after long-term operation at full power of 1,250 kW. Even under extraordinarily conservative assumptions and approximations, the maximum fuel temperature reached in a loss of coolant accident is less than 300°C, well below any safety limit for TRIGA reactor fuel.	Power level for bounding analysis
1-5	1.2.3.b p2	Radiation doses from loss of coolant accident under extremely conservative assumptions are computed and have been tabulated in Chapter 13. Water injection into the reactor pool is accomplished by operating valves and pumps on the 12-foot and 0-foot levels. Radiation levels calculated under these assumptions are calculated to be less than 500 mR/hr on the 0 and 12 foot levels at the time *of water loss...*		1.2.3.b p2	Radiation doses from loss of coolant accident under extremely conservative assumptions are computed and have been tabulated in Chapter 13. Water injection into the reactor pool is accomplished by operating valves and pumps on the 12-foot and 0-foot levels. Radiation levels calculated under these assumptions are high, but valve operation can be accomplished in a time period which will ensure doses do not exceed 10CFR20 limits	Power level for bounding analysis
1-5	1.2.3.c	Two reactivity accident scenarios are presented. The first is the insertion of 2.1% ($3.00) reactivity at zero power by sudden removal of a control rod. The second is the sudden removal of the same reactivity with the core operating at a power level equivalent to the balance of the core excess reactivity. Movements of control rods for the first case are controlled, in part, administratively, while movements for the second are prevented by control circuit design. Analysis shows that (in neither scenario) peak fuel temperatures reach limits, with a maximum of 869°C for conditions where initial steady state power level is regulated only by the balance of core excess reactivity (a condition prevented by interlocks).	1-5	1.2.3.c	Two reactivity accident scenarios are presented. The first is the insertion of 2.1% ($3.00) reactivity at zero power by sudden removal of a control rod. The second is the sudden removal of the same reactivity with the core operating at a power level equivalent to the remainder of the core excess reactivity. Analysis shows that peak fuel temperatures in the first case does not reach fuel temperature limits, with a maximum temperature less than 750°C at the peak in the hot channel for conditions where initial steady state power level is regulated only by the balance of core excess reactivity, while cladding temperature remains below 500°C. In the second case, maximum fuel temperature is calculated at a maximum of less than 870°C at the peak in the hot channel, again with cladding temperature less than 500°C. Although the two scenarios meet criteria required to ensure fuel integrity, movements of control rods for the first case are controlled (in part) administratively, while movements for the second case are prevented by control circuit design.	Clarification

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
1-6	1.3.3	The KSU TRIGA reactor is a water-moderated, water-cooled thermal reactor operated in an open pool. The reactor is fueled with heterogeneous elements clad with stainless steel, consisting of nominally [redacted] percent enriched uranium in a zirconium hydride matrix. In 1968, the KSUTMII was licensed to operate at a steady-state thermal power of 250 kW with a pulsing thermal power limit of 250 MW. Application is made concurrently with license renewal to operate at 500 kW steady-state thermal power and nominal 1000 MW pulsing maximum power.	1-6	1.3.3	The KSU TRIGA reactor is a water-moderated, water-cooled thermal reactor operated in an open pool. The reactor is fueled with heterogeneous elements clad with stainless steel, consisting of nominally [redacted] percent enriched uranium in a zirconium hydride matrix. In 1968, the KSUTMII was licensed to operate at a steady-state thermal power of 250 kW with a pulsing thermal power limit of 250 MW. Application is made concurrently with license renewal to operate up to a maximum steady state power level of 1,250 kW steady-state thermal powers and pulsing to $3.00 (nominal 1,340 MW peak power). Reactor cooling is by natural convection. The 250-kW core consists typically of 80 fuel elements (a minimum of [redacted] planned for the 1,250-kW core), each containing as much as [redacted] grams of ^{235}U. The reactor core is in the form of a right circular cylinder about 9 in. (23 cm) radius and 15 in. (38 cm) depth, positioned with axis vertical near the base of a cylindrical water tank 1.98 m (6.5 ft.) diameter and 6.25 m (16 ft.) depth. Criticality is controlled and shutdown margin assured by control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite. The 250 kW core originally used three control rods, the 1,250-kW core will be controlled by four. The reactor tank is surrounded on the side and at the base by a biological shield of reinforced concrete at least 8.2 ft (2.5 m) thick. The tank and shield are in a 4078 m^3 (144,000 ft^3) dynamic confinement building made of reinforced concrete and structural steel, with composite sheathing and aluminum siding. Sectional views of the reactor are shown in Figures 1.1 and 1.2, with a floor layout in Figure 1.3 showing the 0-foot, 12-foot and 22-foot levels of the facility.	Clarify Power level for bounding analysis
1-6	1.3.4	Reference to 500 kW	1-8	1.3.4	Changed to 1,250 kW	Power level for bounding analysis

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
1-7	1.3.5.a	The reactor control system includes the mechanical and electrical systems for control rod drives, and instruments that monitor control rod position. Each control rod can be independently manipulated by pushbutton console controls. One control rod can be operated in an automatic mode to regulate reactor power according to a manual setpoint, indicated power on the linear power level monitoring channel and a wide range power level monitoring channel (period) feedback. The automatic portion of the reactor control system is interconnected thorough the automatic mode. The wide range power level monitoring channel of the reactor protection system provides interlock signals and actions to the reactor control system. The reactor control system is also interconnected to the reactor protection system through a manual scram bar above the control rod drive switches, allowing the reactor protection system to be actuated manually.	1-8	1.3.5.a	The reactor control system includes the mechanical and electrical systems for control rod drives, and instruments that monitor control rod position. Each control rod can be independently manipulated by pushbutton console controls. One control rod can be operated in an automatic mode to regulate reactor power according to a manual setpoint, indicated power on the linear power level monitoring channel and a wide range power level monitoring channel (period) feedback. The wide range power level monitoring channel of the reactor protection system provides interlock signals and actions to the reactor control system. The reactor control system is also interconnected to the reactor protection system through a manual scram bar above the control rod drive switches (allowing the reactor protection system to be actuated manually) and the automatic mode control (as described above).	Editorial
1-8	1.3.5.b	Primary water temperature is measured by a resistance temperature detector (RTD) in the water box and displayed on the console. A manometer indicates flow rate through the cleanup loop locally.	1-9	1.3.5.b	Primary water temperature is measured in the water box and displayed on the console. A manometer indicates flow rate through the cleanup loop locally. Changed "level" to "levels."	Equipment modifications
1-8	1.5.3.c	The reactor protection system is designed to ensure reactor and personnel safety by limiting parameters to operation within analyzed operating ranges. Parameters that can automatically initiate reactor protection system include neutron level, rate of rise (period) and fuel temperature.	1-9	1.5.3.c	The reactor protection system is designed to ensure reactor and personnel safety by limiting parameters to operation within analyzed operating ranges. Process parameters that can automatically initiate reactor protection system actions include neutron level, rate of rise (period) and fuel temperature.	Editorial
1-8	1.5.3.C	"....Additionally,"	1-9	1.5.3.c	Deleted "Additionally," introduced "although" to identify location of the temperature scram switch	Editorial
1-8	1.5.3.d	A system of fixed and moveable radiation monitors are installed to monitor...			Radiation monitors are installed to monitor radiological conditions at the facility. One monitor is stationed on the top of the reactor, with a local, high range indicator and alarm (at 5 R/hr) to initiate evacuation of the reactor bay. One monitor is stationed at the control room door to the reactor bay, with a 2.5-mrem/hr-alarm setpoint. Electrical connections are installed near each beam port, permitting control room and local indication of radiation levels near an open beam port.	Modification in progress
1-9	1.3.5.e	"...subcritical, secured..." and reference to backup power for evacuation alarm	1-10	1.3.5.e	Removed secured; reserved term; evacuation alarm does not have backup power supply at this time	Editorial
1-10	1.3.6	..."Potential concerns over these systems are address3ed in this chapter of the Safety Analysis Report...."...equipment an operaton..."	1-11	1.3.6	Removed sentence; typo, should be and vice an	Editorial

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
1-11	1.3.6.b	Water returns to an open surge tank by gravity.	1-12	1.3.6.c	Water returns to an open surge tank (located in the reactor bay) by gravity.	Clarification
1-11	1.3.6.d	Reference to use of water transfer system to recirculate			Deleted	System changes
1-11	1.3.7.b	Liquid sources are limited principally to tritium-bearing condensate water from the facility air handling system, and occasional releases of tritium-bearing primary coolant from level adjustments in the reactor tank or bulk-shield tank. All reactor bay floor drains and the HVAC condensate drains discharge to a reactor bay sump. Discharges form the reactor bay sump are sampled and assayed to assure limits for discharge are met prior to discharge. A recirculation system filters sump water to allow discharge to campus sewerage (when radiological requirements are met). Liquid wastes are released through the sanitary sewerage system after filtration and assay for beta, gamma, and alpha activity.	1-11	13.7.b	Liquid sources are limited generally to tritium-bearing condensate water from the facility air handling system, and occasional releases of tritium-bearing primary coolant from level adjustments in the reactor tank or bulk-shield tank. All reactor bay floor drains and the HVAC condensate drains discharge to a reactor bay sump. Contents of the reactor bay sump are sampled and assayed to assure limits for discharge are met prior to discharge. Sump effluent is filtered prior to discharge to meet NPDES requirements for discharge to campus sewerage. Liquid wastes are released through the sanitary sewerage system after filtration and assay for beta, gamma, and alpha activity.	Clarification and system changes
1-12	1.3.8	Left justified	1-12	1.3.8	Full justification	Editorial
1-12	1.3.8.a	...Although the shield water may be removed to allow extraction of a vertical thermal-neutron and gamma-ray beam (not done at the KSU facility), four 0.25-in (6.3-mm) holes are located in the tube at the top of the core to prevent expulsion of water from the section of the tube within the reactor core.	1-13	1.3.8.a	...Although the shield water may be removed to allow extraction of a vertical thermal-neutron and gamma-ray beam (not currently done at the KSU facility at the time this report was completed), four 0.25-in (6.3-mm) holes are located in the tube at the top of the core to prevent expulsion of water from the section of the tube within the reactor core.	Prevent inhibiting the experiment
1-12	1.3.8.b	A rotary 40-position rotary specimen rack (RSR) is located in a well in the top of the graphite radial reflector.	1-13	1.3.8.b	A 40-position rotary specimen rack (RSR) is located in a well in the top of the graphite radial reflector.	Editorial
1-13	1.5	The design of the fuel for the KSU TRIGA is similar to that for fuels used in 70 reactors in 24 nations (General Atomics July 1999 data). Of total number of reactors, 45 are currently in operation or under construction with 63 rated for steady-state thermal powers of 250 kW or greater, 36 at 500 kW or greater. There have been 35 TRIGA reactors in the U.S., with 21 currently in operation. As indicated in Table 1.1, 11 United States TRIGA installations operate at power levels of at least 500 kW. .... Principal design parameters for the KSU TRIGA are given in Table 1.3.	1-14	1.5	The design of the fuel for the KSU TRIGA is similar to that for fuels used in 70 reactors in 24 nations (General Atomics July 1999 data). Of total number of reactors, 45 are currently in operation or under construction with 40 rated for steady-state thermal powers of 250 kW or greater, 22 at 500 kW or greater, and 20 at 1 MW or greater. Nine of the larger power reactors are TRIGA Mark II. The TRIGA Mark II design is a substantial fraction of the 70 reactors using TRIGA fuel world-wide. In the United States, there have been 26 TRIGA reactors built, with 19 currently in operation (5 TRIGA facilities and 3 non-TRIGA reactors converted to operate with TRIGA fuel at power levels greater than 1,000 kW, as indicated in Table 1.1)..... Major design parameters for the KSU TRIGA are given in Table 1.3.	Clarification, correction for use of tables
1-14	Table 1.1	Table number & placement	1-15	Table 1.2	Table number & placement	Clarification, correction for use of tables

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
1-14	Table 1.1	Table 1.1, U.S. MARK II TRIGA REACTORS.	1-15	Table 1.2	Table 1.2, U.S. MARK II TRIGA REACTORS. <inserted bottom line in table>	Clarification, correction for use of tables, editorial
1-15	Table 1.2	Table 1.2, U.S. TRIGA REACTORS AT 500 kW OR GREATER.	1-14	Table 1.1	Table 1.1, U.S. TRIGA REACTORS AT 500 kW OR GREATER.	Clarification, correction for use of tables
1-16	Table 1.3	Table 1.3, KSU TRIGA Reactor Principal Design Parameters at 500 kW Steady-State Power	1-15	Table 1.3	Deleted footnotes, added line at bottom of table, reformatted table	RAI # 1 & editorial
1-16	1.6	List of outside users	1-17	1.6	Added University of Chicago & University of Nebraska at Lincoln	Update user base
1-17	Table 1.4	1998 value for MWh of thermal energy: 154	1-17	Table 1.4	1998 value for MWh of thermal energy: 26	Correction
1-18	Table 1.5	Year : 2002 Scheduled activities	1-17	Table 1.5	Revised to reflect accomplishment & current estimates	Overtaken by events
CHAPTER 2: No Changes						
CHAPTER 3						
3-12	3.5.1	Original printing showed "is" as "i" superimposed on "s"	3-12	3.5.1	Production of hydrogen from a high temperature zirconium-water reaction is a well-known phenomenon. Zirconium hydride does not exhibit the same chemical reactivity as zirconium, and tests demonstrate this reaction is not an issue for TRIGA fuel.	Editorial
3-13	3.5.1	No corresponding text	3-13	3.5.1	Fuel growth and deformation can occur during normal operations, as described in General Atomics technical report E-117-833. Damage mechanisms include fission recoils and fission gases, strongly influenced by thermal gradients. Operating with maximum long-term, steady state fuel temperature of 750°C does not have significant time- and temperature-dependent fuel growth. Since the KSU reactor will not be operated in the regime vulnerable to this degradation, the damage mechanism is not applicable.	Add information
3-13	3.5.1.n	Therefore, temperatures and chemical reactivity of TRIGA fuel matrix ensure that a zirconium water reactor will not occur at magnitudes that could cause hazard to the reactor.	3-14	3.5.1.n	Therefore, temperatures and chemical reactivity of TRIGA fuel matrix ensure that a zirconium water reaction will not occur at magnitudes that could cause hazard to the reactor.	Typographical error
CHAPTER 4						
4-1	4.1	List of experimental facilities, "Tangential (thermal neutron) (2)"	4-1	4.1	"Tangential (thermal neutron) (1)"	Typographical error

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
4-1	4.1	The reactor was licensed in 1962 to operate at a steady-state thermal power of 100 kilowatts (kW). The reactor has been licensed since 1968 to operate at a steady-state thermal power of 250 kW and a pulsing maximum thermal power of 250 MW. Application is made concurrently with license renewal to operate 500 kW steady state thermal power with pulsing to $3.00 reactivity insertion. All cooling is by natural convection. The 250-kW core consists of 81 fuel elements typically (at least [redacted] planned for the 500-kW core), each containing as much as [redacted] grams of ^{235}U. The reactor core is in the form of a right circular cylinder about 23 cm radius and 38 cm depth, positioned with axis vertical near the base of a cylindrical water tank 1.98 m diameter and 6.25 m depth.....	4-1	4.1	The reactor was licensed in 1962 to operate at a steady-state thermal power of 100 kilowatts (kW). The reactor has been licensed since 1968 to operate at a steady-state thermal power of 250 kW and a pulsing maximum thermal power of 250 MW. Application is made concurrently with license renewal to operate at a maximum of 1,250 kW, with fuel loading to support 500 kW steady state thermal power with pulsing to $3.00 reactivity insertion. All cooling is by natural convection. The 250-kW core consists of 81 fuel elements typically (at least [redacted] planned for the 1,250-kW core), each containing as much as [redacted] grams of ^{235}U. The reactor core is in the form of a right circular cylinder about 23 cm (approximately 9 in.) radius and 38 cm (14.96 in.) depth, positioned with axis vertical near the base of a cylindrical water tank 1.98 m (6.5 ft.) diameter and 6.25 m (20.5 ft.) depth. Criticality is controlled and shutdown margin assured by control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite	Power level for bounding analysis
4-1	4.1	....Criticality is controlled and shutdown margin assured by three control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite.	4-1	4.1	....Criticality is controlled and shutdown margin assured by control rods in the form of aluminum or stainless-steel clad boron carbide or borated graphite. . Reactivity requirements (i.e., minimum shutdown margin with the most reactive rod fully withdrawn and maximum excess reactivity) can be met for 250 kW with three control rods, but reactivity required to compensate for fuel temperature and fission products for operations at power levels of 500 kW requires four control rods to meet reactivity requirements.	Clarification
4-1	4.1	The tank and shield are in a 4078-m3-containment building....	4-2	4.1	The tank and shield are in a 4078-m^3 (144,000 $ft.^3$) confinement building	Correct terminology
4-2	4.1	In 1968 pulsing...	4-2	4.1	In 1968, pulsing...	Editorial
4-3	4.2	It is more convenient to set a power level limit that is based on temperature. The design bases analysis indicates that operation at up to 1900 kW (with an [redacted] element core and 120°F inlet water temperature) with natural convective flow will not allow film boiling therefore high fuel and clad temperatures which could cause loss of clad integrity could not occur. An [redacted] element core distributes the power over a larger volume of heat generating elements, and therefore results in a more favorable, more conservative thermal hydraulic response.	4-4	4.2	It is convenient to set a power level limit that is based on temperature. The design bases analysis indicates that operation at up to 1900 kW (with an [redacted] element core and 120°F inlet water temperature) with natural convective flow will not allow film boiling; therefore high fuel and clad temperatures capable of causing loss of clad integrity cannot occur. An [redacted] element core distributes the power over a larger volume of heat generating elements, and therefore using [redacted] elements in analysis results in a less favorable, more conservative thermal hydraulic response.	Editorial
4-4	4.2.1.a	Various	4-5	4.2.1.a	Added SI or SAE units, as appropriate	Add inforiation
4-5	4.2.1.b	Reference to problems of density changes	4-5	4.2.1.b	Added "in earlier designs."	Clarification

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
4-7	4.2.1.c	The layout ofr the 250-kW core is illustrated in Figure 4.4. The layout for the 500-kW core is expected to be very similar, with about 4 graphite elements replacing three fuel elements and one control rod.		4.2.1.c	A typical layout for a KSU TRIGA II 250-kW core (Core II-18) is illustrated in Figure 4.4. The layout for the 1,250-kW core is expected to be similar, except that the graphite elements will be replaced by fuel elements, one additional control rod will be added, and control rod positions will be adjusted..	Correct error
4-7	4.2.1.c	No corresponding text	4-7	4.2.1.c	The additional fuel elements are required to compensate for higher operating temperatures from the higher maximum steady state power level. The additional control rod is required to meet reactivity control requirements at higher core reactivity associated with the additional fuel. The control rod positions will be different to allow a higher worth pulse rod (the 250 kW pulse rod reactivity worth is $2.00, the 1,250 kW core pulse rod reactivity worth is $3.00), balancing the remaining control rods worth's to meet minimum shutdown margin requirements, and meeting physical constraints imposed by the dimensions of the pool bridge	Clarification
4-7	4.2.2	The pulse rod is 1.25 in. diameter. Other rods are 7/8 in. diameter. Control rods are 20 in. long boron carbide or borated graphite, clad with a 30-mil aluminum sheath.	4-8	4.2.2	With exception of initial paragraph, extensive re-write.	RAI # 2
4-7	4.2.2.a	Reactivity of the KSU reactor is controlled by up to four standard control rods plus one transient (pulsing) control rod.	4-10	4.2.2.a	While three control rods were adequate to meet Technical Specification requirements for reactivity, operation at 500 kW requires control by four control rods (three standard and one transient/pulsing control rod)..... Interlocks ensure operation of the control rods remains within analyzed conditions for reactivity control or limit potential for accident scenarios, while scrams operate at limiting safety system settings. A detailed description of the control-rod system is provided in Chapter 7; a summary of interlocks and scrams is provided below in Table 4.2 and 4.3. Note that (1) the high fuel temperature and period scrams are not required, (2) the fuel temperature scram limiting setpoint depends on core location for the sensor, and (3) the period scram can be prevented by an installed bypass switch.	Correct error & clarify use of interlocks
4-8	4.2.2.a	No corresponding text	4-11	4.2.2.a	Table 4.2 & 4.3 added	Add information
4-8	4.2.2.b	Reference to 500 kW operation	4-12	4.2.2.b	Added approximately to qualify 500 kW	Clarification
4-8	4.2.2.b	Nominal speed of the standard control rods is about 12 in. per minute of the transient rod is about 24 in. per minute, with a total travel about 15 in. Maximum rate of reactivity change for standard control rods is specified in Technical Specifications.	4-12	4.2.2.b	Nominal speed of the standard control rods is about 12 in. (30.5 cm) per minute (with the stepper motor specifically adjusted to this value), of the transient rod is about 24 in. (61 cm) per minute, with a total travel about 15 in. (38.1 cm). Maximum rate of reactivity change for standard control rods is specified in Technical Specifications.	Add information

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
4-8	4.2.3	No corresponding text	4-12	4.2.3	Hydrogen in the Zr-H fuel serves as a neutron moderator. Demineralized light water in the reactor pool also provides neutron moderation (serving also to remove heat from operation of the reactor and as a radiation shield). Water occupies approximately 35% of the core volume. A graphite reflector surrounds the core, except for a cutout containing the rotary specimen rack (described in Chapter 10). Each fuel element contains graphite plugs above and below fuel approximately 3.4 in. in length, acting as top and bottom reflectors.	Add information consistent with review standard
4-8	4.2.4	SI and SAE units used	4-12	4.2.4	Added SI or SAE units parenthetically	Add information
4-9	4.2.5	The fuel elements are spaced and supported by two 0.75-in. thick aluminum grid plates. The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures. Space is provided for the passage of cooling water around the sides of the bottom grid plate and through 36 special holes in it. The 1.5-in. diameter...	4-13	4.2.5	The fuel elements are spaced and supported by two 0.75-in. (1.9 cm) thick aluminum grid plates. The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures. Space is provided for the passage of cooling water around the sides of the bottom grid plate and through 36 experiment penetrations. The 1.5-in. (3.8 cm) diameter...	Editorial

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
4-9	4.3	The KSU TRIGA reactor core support structure rests on the base of a continuous, cylindrical aluminum tank surrounded by a reinforced, standard concrete structure (with a minimum thickness of 8 ft 2 in.), as illustrated in Figures 4.1 and 4.2. The tank is a welded aluminum structure with 1/4-in. thick walls. The tank is 6.5-ft in diameter and 20.5-ft in depth. The minimum thickness of concrete shielding at the core level is 8 ft. 2 in.	4-13	4.3	The KSU TRIGA reactor core support structure rests on the base of a continuous, cylindrical aluminum tank surrounded by a reinforced, standard concrete structure (with a minimum concrete thickness of approximately 249 cm. or 8 ft 2 in), as illustrated in Figures 4.1 and 4.2. The tank is a welded aluminum structure with 0.635 cm. (1/4-in.) thick walls. The tank is approximately 198 cm (6.5-ft) in diameter and approximately 625 cm (20.5-ft) in depth. The exterior of the tank was coated with bituminous material prior to pouring concrete to retard corrosion. Each experiment facility penetration in the tank wall (described below) has a water collection plenum at the penetration. All collection plenums are connected to a leak-off volume through individual lines with isolation valves, with the leak-off volumes monitored by a pressure gauge. The bulk shield tank wall is known to have a small leak into the concrete at the thermalizing column plenum, therefore a separate individual leak-off volume (and pressure gauge) is installed for the bulk shield tank; all other plenums drain to a common volume. In the event of a leak from the pool through an experiment facility, pressure in the volume will increase. Isolating individual lines allows identification of the specific facility with the leak. A bridge of steel plates mounted on two rails of structural steel provides support for control rod drives, central thimble, the rotary specimen rack, and instrumentation. The bridge is mounted directly over the core area, and spans the tank. Aluminum grating with clear plastic attached to the bottom is installed that can be lowered over the pool. The grating normally remains up to reduce humidity at electro-mechanical components of the control rod drive system and to prevent the buildup of radioactive gasses at the pool surface during operations. The grating can be lowered during activities that could cause objects or material to fall into the reactor pool. Four beam tubes extend from the reactor wall to the outside of the concrete biological shield in the outward direction. Tubes welded to the inside of the wall extend toward the reactor core. Three of the tubes (NW, SW, and SE) end at the radial reflector. The NE beam tube penetrates the radial reflector, extending to the outside of the core. Two penetrations in the tank allow neutron extraction into a thermal column and a thermalizing column (described in Chapter 10).	Add information consistent with review standard
4-9	4.4	SI and SAE units used	4-13	4.4	Added SI or SAE units parenthetically	Add information

TABULATION OF CHANGES TO DRAFT KSU TRIGA II SAFETY ANALYSIS REPORT						
Original			Revised			Reason
Page	Paragraph	Text	Page	Paragraph	Text	
4-9	4.5	....For fuels with [redacted] U, [redacted] enrichment, the value is nearly constant at 0.01% Δk/k per °C, and varies only weakly dependent on geometry and temperature.	4-14	4.5	....For fuels with [redacted] U, [redacted] enrichment, the value is nearly constant at 0.01% Δk/k per °C, only weakly dependent on geometry and temperature.	Editorial
4-10	4.5	...The design bases analysis indicates that operation at 500 kW thermal power with an [redacted] element across a broad range of core and coolant inlet temperatures with natural convective flow will not allow film boiling leading to high fuel and clad temperatures that could cause loss of clad integrity. Increase in maximum thermal power from 250 to 500 kW kW does not affect fundamental...	4-14	4.5	The design bases analysis indicates that operation at 1,250 kW thermal power with an [redacted] element across a broad range of core and coolant inlet temperatures with natural convective flow will not allow film boiling that could lead to high fuel and clad temperatures that could cause loss of clad integrity. Increase in maximum thermal power from 250 to 1,250 kW does not affect fundamental...	Power level for bounding analysis
4-10	4.5.1	The limiting core configuration for this analysis is a compact core defined by the TRIGA Mk II grid plates (Section 4.2.5), namely ..... The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and graphite dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures.	4-15	4.5.1	The limiting core configuration for this analysis is a compact core defined by the TRIGA Mk II grid plates (Section 4.2.5). The grid plates have a total of 91 spaces, up to 85 of which are filled with fuel-moderator elements and graphite dummy elements, and the remaining spaces with control rods, the central thimble, the pneumatic transfer tube, the neutron source holder, and one or more voids in the E or F (outermost two rings) as required to support experiment operations or limit excess reactivity. The bottom grid plate, which supports the weight of the fuel elements, has holes for receiving the lower end fixtures.	Clarification
4-10	Table 4.2	Position change	4-15	Table 4.2	Position change	Editorial
4-12	4.5.3.a	No corresponding text	4-16	4.5.3.a	Substantial rewrite, incorporating steady state calculations	Add information
4-12	4.5.3.b	Substantial re-write	4-20	4.5.3.b	Substantial re-write	Power level for bounding analysis; RAI # 3 & 21
4-14	4.5.3.c	Substantial re-write	4-23	4.5.3.c	Substantial re-write	Power level for bounding analysis
4-17	Table 4.6	Upper power level 500 kW	4-26	4.11	Table extended to higher maximum power	Power level for bounding analysis

4-17	4.7	As described in 3.5.1 (Fuel System) and NUREG 1282, fuel temperature limits both steady-state and pulse-mode operation. The fuel temperature limit stems from potential hydrogen outgassing from the fuel and the subsequent stress produced in the fuel element clad material. The maximum temperature limits of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) have been set to limit internal fuel cladding stresses that might lead to clad integrity (NUREG 1282)...		4.7	As described in 3.5.1 (Fuel System) and NUREG 1282, fuel temperature limits both steady-state and pulse-mode operation. The fuel temperature limit stems from potential hydrogen outgassing from the fuel and the subsequent stress produced in the fuel element clad material by heated hydrogen gas. Yield strength of cladding material decreases at a temperature of 500°C; consequently, limits on fuel temperature change for cladding temperatures greater than 500°C. A maximum temperature of 1150°C (with clad < 500°C) and 950°C (with clad > 500°C) for U-ZrH ($H/Zr_{1.65}$) will limit internal fuel cladding stresses that might lead to clad integrity (NUREG 1282) challenges.	RAI # 4 & 5, Clarification
4-18	4.8.2	Substantial re-write	4-26	4.8.2	Substantial re-write	RAI # 6-10
4-18	4.8.3	Substantial re-write	4-27	4.8.3	Substantial re-write	RAI # 6, 7
4-18	Table 4.10	Deleted	4-27	NA	Removed table	RAI # 8
CHAPTER 5						
5-2	5.1	This cooling system combination provides enough heat removal for continuous full-power operation. In addition to the cooling system, the reactor is provided with a bulk-shielding tank. This 6500-gallon (25 kL) tank contains distilled water, and can be used to supplement make-up water for the primary tank or provide temporary fuel storage. Makeup water for both systems is provided by a steam-powered still.... Although the cooling system is not required to be operating during reactor operation, there is an administrative requirement that the system be capable of operation (operable). In normal...	5-2	5.1	This cooling system combination provides enough heat removal for continuous full-power operation. In addition to the cooling system, the reactor is provided with a bulk-shielding tank. This 6500-gallon (25 kL) tank contains distilled water, and can be used to supplement make-up water for the primary tank (using a makeup water system independent of the above drawing) or provide temporary fuel storage. Makeup water for both systems is provided by a steam-powered still....In...	Clarification
5-3	5.2	Principal functional requirements of the primary coolant system are to (1) transfer heat from the reactor core to the secondary cooling system, and (2) provide radiation shielding directly above the reactor core. Although natural convection cools the reactor core, proposed Technical Specifications require primary bulk water temperature be kept below 130°F (48.9 °C).	5-3	5.2	Principal functional requirements of the primary coolant system are to (1) transfer heat from the reactor core to the secondary cooling system, and (2) provide radiation shielding directly above the reactor core. Although natural convection cools the reactor core, primary bulk water temperature should be kept below 130°F (48.9 °C).	Correction
5-4	5.2	A plate-type compact heat exchanger is used to remove heat from the primary coolant (see Figure 5.2). The heat exchanger consists of sandwiched stainless steel plates alternately carrying primary and secondary cooling water. The heat exchanger has a transfer capacity of (2,327,080. BUT h^{-1}) (500 kW) under normal conditions.	5-3	5.2	A plate-type compact heat exchanger is used to remove heat from the primary coolant (see Figure 5.2). The heat exchanger consists of sandwiched stainless steel plates alternately carrying primary and secondary cooling water. The heat exchanger has a transfer capacity of 682 kW (1,709,000. BUT h^{-1}) under normal conditions.	Modification
5-4	5.2	If a major loss of coolant were to occur, there are two level sensors that would illuminate a light on the control panel. One sensor is located in the reactor bay sump, activating when the sump id full. Since all floor drains in the reactor bay connect to the sump, any leaks would accumulate there. A second sensor is located at the top of	5-4	5.2	If a major loss of coolant were to occur, there are three level sensors that would illuminate lights on the control panel. Two sensors are located in the reactor bay sump, activating when the sump level is high. Since all floor drains in the reactor bay connect to the sump, any leaks would accumulate there. A third sensor is located at the top of the tank....	Equipment modification

		the tank....				
5-4	5.3	Cooling tower capacity was upgraded	5-4	5.3	The secondary cooling system is designed for continuous operation at approximately 723 kW.	Clarification
5-6	5.3.1	The water from the surge tank is then drawn into the system by a direct-coupled, self-priming, centrifugal pump (see Figure 5.3). When the controls are in a normal configuration, the pump is energized (with the primary pump and cooling tower fans) from a backlit push button switch in the control room. Normal flow rate through the system is 176 gallons per minute (11.1 $L \cdot s^{-1}$).	5-6	5.3.1	The water from the surge tank is then drawn into the system by a direct-coupled, self-priming, centrifugal pump (see Figure 5.3). When the controls are in a normal configuration, the pump is energized (with the primary pump and cooling tower fans) from a backlit push button switch in the control room. Normal flow rate through the system is 250 gallons per minute (15.8 $L \cdot s^{-1}$).	Equipment modification
5-6	Fig 5.3	New picture	5-6	Fig 5.3	New picture	Equipment modification
5-8	5.3.3	To detect possible leaks in the heat exchanger, the secondary water is tested monthly for radioactivity. Since the primary tank has 22-feet (6.7 m) of static head, as opposed to 3 to 9-feet (0.9 to 2.7 m) in the secondary (depends on surge tank level), a breach in the heat exchanger would result in flow from the primary to secondary cooling system when the cooling system is secured. Leaks as a result of system operating pressure are unlikely test pressure was 150 psig while since the maximum pressure differential across the heat exchanger (primary pump running, secondary pump off) is 30 psig. A small breach in the heat exchanger would be evidenced by tritium contamination of the secondary water. A larger breach would be indicated by loss of primary coolant from the reactor tank. Regardless, primary water typically remains within 10CFR20 limits for release to sanitary sewers, and is not seen as an eminent hazard.	5-8	5.3.3	To detect possible leaks in the heat exchanger, the secondary water is tested monthly for radioactivity. Since the primary tank has 22-feet (6.7 m) of static head, as opposed to 3 to 9-feet (0.9 to 2.7 m) in the secondary (depends on surge tank level), a breach in the heat exchanger would result in flow from the primary to secondary cooling system when the cooling system is secured. Leaks while the cooling system is operating are unlikely; the heat exchanger was pressure tested to 150 psig, and the maximum pressure differential across the heat exchanger (primary pump running, secondary pump off) is 30 psig. A small breach in the heat exchanger would be evidenced by tritium contamination of the secondary water. A larger breach would be indicated by loss of primary coolant from the reactor tank. Primary water typically remains within 10CFR20 limits for release to sanitary sewers, and is not a hazard even if a leak were to occur.	Clarification
5-10	5.6	The cooling systems return line to the reactor pool enters the pool through a diffuser. The diffuser is constructed to induce a helical flow pattern in the reactor tank. This extends transport time of the convection flow of water from the core to allow much of the nitrogen-16 generated during operation to decay before reaching the pool surface. A radiation monitor directly above the pool surface provides the control room operator with information to prompt exposure controls (generally energizing the primary cooling pump to initiate the helical flow for decay, or limiting access to the area directly over the pool). Waist-level radiation measurements at full licensed power directly above the pool surface are typically 5 to 10 $mR \cdot h^{-1}$ (10 $\mu Gy \cdot h^{-1}$) form all sources.	5-9	5.6	The cooling systems return line to the reactor pool enters the pool through a diffuser. The diffuser is constructed to induce a helical flow pattern in the reactor tank. This extends transport time of the convection flow of water from the core to allow much of the nitrogen-16 generated during operation to decay before reaching the pool surface. A radiation monitor directly above the pool surface provides the control room operator with information to prompt exposure controls (generally energizing the primary cooling pump to initiate the helical flow for decay, or limiting access to the area directly over the pool). Pool surface monitor radiation measurements at 250 kW directly above the pool surface are typically 30 to 40 $mR \cdot h^{-1}$ from all sources, and is expected to be 60-80 $mR \cdot h^{-1}$ at 500 kW operation. A radiation monitor at the rail around the pool provides the control room operator with information to prompt exposure controls for personnel on the 22-foot level but not directly over the reactor pool. At 250 kW, radiation levels at the rail are less than 2 mR/hr.	Clarification
CHAPTER 6						

CHAPTER 7						
7-2	7.1	The control console and display instruments are primarily housed in a control console, with auxiliary instruments located in a rack next to the console. At the console, the reactor operator has direct control over mode of operation, control rod drive positions, cooling system operation, opening of reactor bay doors, and manual scram of the reactor. Display instruments located in the control console provide measurements of reactor power, control rod positions, primary water temperature, and fuel temperature. Indicators in the console display scram information, low air pressure, low primary water level, high reactor sump water level, sump full, secondary surge tank level low, source interlock status, reactor bay upper door open, reactor bay lower door open, thermal column door open, person on stairway, and rod drive status.	7-2	7.1	The control console and display instruments are primarily housed in a control console, with auxiliary instruments located in a rack next to the console. At the console, the reactor operator has direct control over mode of operation, control rod drive positions, cooling system operation, opening of reactor bay doors, and manual scram of the reactor. Display instruments located in the control console provide measurements of reactor power, control rod positions, primary water temperature, and fuel temperature. Indicators in the console display scram information, low air pressure, low primary water level, high reactor sump water level, sump high water level, sump overflow water level] secondary surge tank level low, source interlock status, reactor bay upper door open, reactor bay lower door open, thermal column door open, person on stairway, and rod drive status. Secondary surge tank makeup is controlled with a backlit pushbutton that indicates surge tank low level and surge tank makeup valve operation.	Equipment modifications
7-3	7.2.2	The primary function of the RCS is to govern the manner in which reactivity is varied in the reactor core. The RCS system should prevent the reactor operator from unintentionally inserting large amounts of reactivity, through various interlock systems. The operator should only be able to remove one rod at a time from the reactor core, preventing large insertion rates. The pulse rod must not be able to be rapidly ejected from the core while in steady-state operation. Furthermore, the pulse rod should be the only rod that can be can be moved in pulse mode, preventing supercritical pulses.	7-3	7.2.2	The primary function of the RCS is to govern the manner in which reactivity is varied in the reactor core. The RCS system should prevent the reactor operator from unintentionally inserting large amounts of reactivity, through various interlock systems. The operator should only be able to remove one rod at a time from the reactor core, preventing large insertion rates. The pulse rod must not be able to be rapidly ejected from the core while in steady-state operation. Furthermore, the pulse rod should be the only rod that can be withdrawn in pulse mode, preventing supercritical pulses.	Correct wording
7-4	7.2.2	The primary function of the RPS is to automatically insert the control rods into the reactor core when certain parameters deviate from limited safety system settings. Several scrams involve the neutronic channels in the RCS. If 110% rated power level is exceeded in steady state mode, one of two trip-points will scram the reactor. Failure of the high voltage power supplies for operating neutronic channels will also cause a scram. For pulsing operations, a scram will be actuated when the fuel temperature is in excess of 450°C.	7-4	7.2.2	The primary function of the RPS is to automatically insert the control rods into the reactor core when certain parameters deviate from limited safety system settings. Several scrams involve the neutronic channels in the RCS. If 110% rated power level is exceeded in steady state mode, one of two trip-points will scram the reactor. Failure of the high voltage power supplies for operating neutronic channels will also cause a scram.	SCRAM removed form Technical Specifications
7-5	7.2.3	The overall system layout is depicted in Figure 7.2. The majority of the RCS is housed in a General Atomics (GA) console originally manufactured for the USGS reactor, which is shown with modifications in Figure 7.2. A detailed description of this figure is provided	7-5	7.2.3	The overall system layout is depicted in Figure 7.2. The majority of the RCS is housed in a General Atomics (GA) console originally manufactured for the USGS reactor, which is shown with modifications in Figure 7.2. A detailed description of this figure is provided in Table 7.1] Figure 7.3 shows a representative layout of the auxiliary instrumentation rack.	Clarification
7-6	Fig 7.3	Labels changed	7-8	Fig 7.3	Timer & Pool Light controls removed	Equipment modifications
7-7	Table 7.1, No. 13	Scram Status, Source Interlock, Low Air Pressure, Person on Stairway, Upper & Lower Doors, and Cooling System Power	7-7	Table 7.1, No. 13	Scram Status, Source Interlock, Low Air Pressure, Hi & Hi-Hi sump level, surge tank level & makeup, Upper & Lower Doors, and Cooling System Power	Equipment modifications

7-9	7.3.1	The remainder of the channel circuitry is located in the NLW-1000 unit in the central console. The NLW-1000 unit supplies the high voltage for the detector and power for the preamplifier. The instrument switches from pulse mode operation to current mode as reactor power increases out of the source range, allowing the instrument to measure reactor power in the upper ranges. Three displays indicate reactor power, high voltage, and reactor period. The power signal is permanently recorded via an opto-isolated output to a strip-chart recorder located in the instrumentation rack. The period meter has a scram at 3 sec and there is a high voltage scram, both of which are bypassed in pulse mode. This channel also provides a protective interlock which prevents rod withdrawal when indicated neutron flux is < 2 cps, which is also activated in pulse mode to prevent removal of the shim, safety and regulating rods. Another interlock prevents pulsing when reactor power is above 1 kW.	7-8	7.3.1	The remainder of the channel circuitry is located in the NLW-1000 unit in the central console. The NLW-1000 unit supplies the high voltage for the detector and power for the preamplifier. The instrument switches from pulse mode operation to current mode as reactor power increases out of the source range, allowing the instrument to measure reactor power in the upper ranges. Three displays indicate reactor power, high voltage, and reactor period. The power signal is permanently recorded via an opto-isolated output to a strip-chart recorder located in the instrumentation rack. The period meter has a scram at 3 sec and there is a high voltage scram, both of which are bypassed in pulse mode. This channel also provides a protective interlock which prevents rod withdrawal when indicated neutron flux is < 2 cps, which is also activated in pulse mode to prevent removal of the shim, safety and regulating rods. Another interlock prevents pulsing when reactor power is above 10 kW (normally set at 1 kW). The unit has two calibration checks in pulse mode, two in current mode, and checks for the period and high voltage scrams.	Clarification
7-11	7.3.1	The original instruments that the N-1000 series units replaced are still housed in the control console for backup use. These older analog devices have all of the same measurement and RPS features, except that they lack opto-isolated outputs for computer acquisition of reactor data. They also require more manual input as the linear channel does not possess auto-ranging features. These instruments were used for many years at USGS and for one year at K-State until the N-1000 units arrived, and provide adequate backup for an interim time while the N-1000 series units are serviced. Wiring diagrams and calibration procedures for these instruments are located in the maintenance manual for the USGS console.	7-9	7.3.1	Deleted	Not needed
7-11	7.3.2	Temperature indications for the primary water and specific B-Ring fuel elements are provided on the front section of the control panel and in the instrumentation rack. The instrumented fuel elements have three chromel-alumel thermocouples in the fuel element that are used for temperature indication on the console or in the instrumentation rack. Since all three thermocouples are located 0.76 cm/0.3-in below the fuel surface and with two spaced only 2.5 cm from the third at the midpoint of the element, an averaged value from all three thermocouples for a single element is typically used for instrument readings.	7-10	7.3.2	Temperature indications for the primary water and specific B-Ring fuel elements are provided on the front section of the control panel and in the instrumentation rack. The instrumented fuel elements have three chromel-alumel thermocouples in the fuel element that are used for temperature indication on the console or in the instrumentation rack. The thermocouples are located 0.76 cm (0.3-in) below the fuel surface spaced at the midpoint of the element and at ±, 2.5 cm from the midpoint; an averaged value from all three thermocouples is typically used for instrument readings.	Clarification

7-12	7.3.3	Four control rods are required for reactor operations at 500 kW: a shim rod, a regulating rod, a transient rod, and a safety rod. The shim, regulating and safety rods share identical and circuitry and provide coarse and fine power control. Two of the rod drives are original, analog systems. One of the rod drives uses a stepper motor. The pulse rod is designed so that it can be rapidly ejected from the core to a preset height to initiate a reactor pulse. However, it still functions as a normal control rod in steady state mode. All rods can be individually scrammed without shutting down the reactor.	7-11	7.3.4	Four control rods are required for reactor operations at 1,250 kW to meet reactivity control requirements: a shim rod, a regulating rod, a transient rod, and a safety rod. The shim, regulating and safety rods share identical control circuitry (Figure 7.7) and provide coarse and fine power control. Two of the rod drives are original, analog systems. One of the rod drives uses a stepper motor. Drive position is determined by voltage drop across a potentiometer that is adjusted as the control rod drive is moved. The position indicator for the analog motors is attached to a shaft coupled to the drive motor shaft with a setscrew, while the stepper motor is connected to the position indicator with a chain drive.	RAI # 11, 12
7-12	7.3.4.a	The rod drive motor is dynamically braked and held by an electrically locked motor. In the static condition, both windings are energized with the same phase (see Figure 7.7), electrically locking the motor. Clockwise (up) or counter-clockwise (down) rotation is enabled by shifting the phase between the windings with a 1-μF capacitor; motor control switches allow the appropriate phase shift.	7-11	7.3.4.a	The analog rod drive motor is dynamically braked and held by an electrically locked motor. In the static condition, both windings are energized with the same phase (see Figure 7.7), electrically locking the motor. Clockwise (up) or counter-clockwise (down) rotation is enabled by shifting the phase between the windings with a 1-μF capacitor; motor control switches allow the appropriate phase shift. The stepper motor operates using phase switched direct current power. The motor shaft advances 200 steps per revolution (1.8 degrees per step). Since current is maintained on the motor winding when the motor is not being stepped, high holding torque is maintained. A translator module drives the stepping motor.	RAI # 12, 13
7-17	7.3.4.c.4	The pulse rod is the only control rod that can be moved when the reactor is in the PULSE mode (this does not prevent the scramming of any control rod).		7.3.4.c	4. The pulse rod is the only control rod that can be withdrawn when the reactor is in the PULSE mode (this does not prevent the scramming of any control rod).	Correct wording
7-17	7.3.4	Additionally, there is an interlock that prevents reactor pulses from being fired if the reactor power is above 1 kW. There is also a key switch for bypassing the source interlock during fuel loading operations to check for criticality.	7-16	7.3.4	Additionally, there is an interlock that prevents reactor pulses from being fired if the reactor power is above 10 kW (normally set at 1 kW). There is also a key switch for bypassing the source interlock during fuel loading operations to check for criticality.	Clarification
7-18	7.4	In pulse mode, the mode selector switch is set to the HI PULSE position, interrupting detector signal to the linear channel. When the pulse interlock is activated (to initiate the source interlock) to prevent movement of the shim, safety and regulating rods, the detector signal to the logarithmic wide range detector is interrupted.	7-17	7.4	In pulse mode, the mode selector switch is set to the HI PULSE position, interrupting detector signal to the linear channel. When the pulse interlock is activated (to initiate the source interlock) to prevent withdrawal of the shim, safety and regulating rods, the detector signal to the logarithmic wide range detector is interrupted.	Correct wording
7-18	7.6	There are several additional pieces of equipment in the control room. Directly behind the operator are the circuit breakers to interrupt power to electrical devices in the control room and reactor bay (see Figure 7.13). A halon fire extinguisher is located next to the breakers for use in fighting electrical fires. Current core and facility configuration is shown in a display cabinet. A wall-mounted box in the control room has illuminated switches to indicate personnel in the reactor bay. A radiation area monitor is located above the door in the control room to the reactor bay.	7-17	7.6	There are several additional pieces of equipment in the control room. Directly behind the operator are the circuit breakers to interrupt power to electrical devices in the control room and reactor bay (see Figure 7.13). A halon fire extinguisher is located next to the breakers for use in fighting electrical fires. Current core and facility configuration is shown in a display cabinet (Figure 7.13.b). A wall-mounted box in the control room has illuminated switches to indicate personnel in the reactor bay. A local radiation area monitor (including indicator and alarm) is located above the door in the control room to the reactor bay.	Clarification

9-6	9.7.3.d	The pneumatic transfer system or rabbit is used to rapidly transport samples between an in-core location and the Neutron Activation Analysis Laboratory. From commercial cylinders, compressed helium fills small tanks at either end of the system. Pressure is limited by release valves at 275 kPa (40 psi). The system is operated from the control room, where the reactor operator sets the direction of motion by positioning vent valves and applies the helium by another valve. Indicator lights show position of valves.	9-5	9.7.3.d	The pneumatic transfer system or rabbit is used to rapidly transport samples between an in-core location and the Neutron Activation Analysis Laboratory. From commercial cylinders, compressed helium fills small tanks at either end of the system. Pressure is limited by release valves at 275 kPa (40 psi). The system is operated from the instrumentation rack (Fig 7.3) in the control room, where the reactor operator sets the direction of motion by positioning vent valves and applies the helium by another valve. Indicator lights show position of valves.	Clarification
CHAPTER 10						
10-1	10.1.3.a	The interface of experimental facilities (beam ports, thermal column and thermalizing column) and the reactor pool liner contains an open plenum with piping connected to a leak off volume. The leak detection piping is connected to a single volume, except that a separate leak off volume and pressure gauge has been installed for the thermalizing column. If the pool leaks into the experiment facilities, the water will overflow the plenum and fill the leak off volume. Pressure monitors in the leak off volume indicate when the volume is partially or fully filled.	10-1	10.1.3.a	The interface of experimental facilities (beam ports, thermal column and thermalizing column) and the reactor pool liner contains an open plenum with piping connected to a leak off volume. The leak detection piping is connected to a single volume, except that a separate leak off volume and pressure gauge has been installed for the thermalizing column. If the pool leaks into the experiment facilities, the water will overflow the plenum and fill the leak off volume. Pressure monitors in the leak off volume indicate when the volume is partially or fully filled. The leak off volume and the pressure monitors are located on the north wall of the biological shielding near the northwest (radial) beam port.	Add information
Various	Various	Included both SI and SAE units	Various	Various	Included both SI and SAE units	Add informaiton
CHAPTER 11						
11-1	11.1.1, P2	....Because of its short half-life, ^{16}N contributes negligibly to off-site radiation exposure.	11-1	11.1.1 P2	. Because of its short half-life, ^{16}N contributes negligibly to off-site radiation exposure, but is the major source of radiation dose to the area above the reactor pool.	Clarification
11-2	11.1.1.a, Tritium in the Reactor Bay	Measured tritium specific activity in primary coolant is less than 10^{-3} µCi g^{-1}.	11-2	11.1.1.a, Tritium in the Reactor Bay	A 5-year average of tritium assay (performed monthly) indicates specific activity in the primary coolant of 228 pCi/ml for 250 kW operations. If the reactor bay atmosphere were saturated with this water at 30°C, the water concentration in the air would be less than 3×10^{-5} g mL^{-1} and the activity concentration in the atmosphere would be less than 6.84E-09 µCi/ml. Based on history, tritium concentration at 500 kW would be less than 1.37×10^{-8} µCi/ml, and tritium concentration at 1,250 kW would be less than 3.42×10^{-8} µCi/ml. In all cases, tritium concentrations are	New informaiton
11-2	11.1.1.a	During normal operation of the reactor facility, there are three airborne sources, ^{3}H, ^{16}N, and ^{41}Ar.	11-3	11.1.1.a	During normal operation of the reactor facility, there are three major potential airborne sources, ^{3}H, ^{16}N, and ^{41}Ar.	Clarification
11-2	11.1.1.a	Exhaust of the rotary specimen rack:[1] As shown in Chapter 11 Appendix A, the equilibrium activity of ^{41}Ar in the RSR is 0.56 Ci for sustained operation at 500 kW thermal power. If this activity were instantly dispersed into the reactor bay atmosphere, under normal ventilation conditions, and a worker were continuously exposed thereafter, the cumulative exposure would be 1.8×10^{-4} µCi h mL^{-1}, well below 6×10^{-4} µCi h mL^{-1} for 2000 hours occupational exposure at the DAC.	11-2	11.1.1.a	Exhaust of the rotary specimen rack:[1] As shown in Chapter 11 Appendix A, the equilibrium activity of ^{41}Ar in the RSR is 0.56 Ci for sustained operation at 500 kW thermal power, 1.4 Ci at 1,250 kW. If this activity were instantly dispersed into the reactor bay atmosphere, under normal ventilation conditions, and a worker were continuously exposed thereafter, the cumulative exposure would be at 4.4×10^{-10} µCi h mL^{-1}, well below the occupational exposure limit for 2000 hours at the DAC, 3×10^{-6} µCi h mL^{-1}.	RAI # 14 & Power level for bounding analysis

7-19	7.7	Radiation monitoring systems are employed throughout the reactor facility: G-M detectors at the reactor pool surface and cleanup loop, 5 remote area monitor channels (3 permanent, 1 mobile), a....	7-18	7.7	Radiation monitoring systems are employed throughout the reactor facility: G-M detectors at the reactor pool surface and cleanup loop, 7 remote area monitor channels (3 general area or process monitors, 4 channels for beam ports – 1 beam port channel is currently instrumented, with the remainder scheduled for instrumentation near term), a.....	Modification in progress
7-20	Fig 7.13	Changed figures	7-20	Fig 7.13a & b	Split into 2 separate pictures	Configuration change
7-21	7-7	The remote area monitors utilize G-M detectors located throughout the reactor bay (typical unit illustrated in Figure 7.14). Permanent locations are: at the top of the reactor tank, above the bulk shield tank, and near the ion exchanger in the primary coolant system, and directly over the primary water tank. A movable detector can be located at any beamport or the thermal column. The detectors feature an analog readout in both the control room and locally with visual indicators for normal, alert, and alarm conditions. The control room alarm has an audible signal as well.	7-20	7-7	The remote area monitors utilize G-M detectors located throughout the reactor bay (typical unit illustrated in Figure 7.14). Permanent locations are: at the top of the reactor tank, above the bulk shield tank, and near the ion exchanger in the primary coolant system, and directly over the primary water tank. Each beam port has signal and power lines to support installing a beam port monitor. The detectors feature an analog readout in both the control room and locally with visual indicators for normal, alert, and alarm conditions. The control room alarm has an audible signal as well.	Modification in progress
7-22	Fig 7.15	Changed figures	7-21	Fig 7.15	Changed figures	New location for air monitoring system
CHAPTER 8						
8-3	8.2	Backup batteries supply the evacuation alarm; a 12 V deep-cycle battery for the air horn and a 6 V battery for the detector. These batteries are tested annually and a charger is permanently incorporated into the system. As mentioned in Chapter 7, the evacuation alarm is located at the upper level of the reactor and signals a need to evacuate the reactor bay.	8-2	8.2	Deleted	Configuraiton change
CHAPTER 9						
9-5	9.7.2	A polar crane in the reactor bay is used to manipulate loads of up to 3630 kg (8000 lb). Various lifting straps and attachments are available for handling varied loads. A breaker in the reactor control room supplies power to the crane. A basket is attached to the outside edge of the crane for changing light bulbs in the ceiling of the reactor bay. Due to safety concerns, lighting was installed around the periphery of the reactor bay, eliminating the need for the overhead lights and the basket.	9-4	9.7.2	A polar crane in the reactor bay is used to manipulate loads of up to 3630 kg (8000 lb). Various lifting straps and attachments are available for handling varied loads. A breaker in the reactor control room supplies power to the crane. A lockable breaker on the west wall of the reactor bay permits securing power to the crane. A disconnect on the crane permits personnel servicing the crane a local and positive control over power to the crane. A basket is attached to the outside edge of the crane for changing light bulbs in the ceiling of the reactor bay. Due to safety concerns, lighting was installed around the periphery of the reactor bay, reducing the need for the overhead lights and the basket.	Equipment modification

11-3	11.1.1.a	Release from primary coolant: As shown in Chapter 11 Appendix A, even with extremely conservative assumptions, during sustained operation at full power with ventilation, the steady-state activity concentration of ^{41}Ar in the reactor-bay atmosphere would be 3×10^{-6} µCi mL^{-1}, less than the occupational DAC.	11-3	11.1.1.a	Release from primary coolant: As shown in Chapter 11 Appendix A, even with extremely conservative assumptions, during sustained operation at full power with ventilation, the steady-state activity concentration of ^{41}Ar in the reactor-bay atmosphere would be 7.2×10^{-7} µCi mL^{-1}, less than the occupational DAC.	Power level for bounding analysis
11-3	11.1.1.a	Offsite Impact of ^{41}Ar. As shown in Chapter 11 Appendix A, the peak off-site activity concentration during normal operations would be about 6×10^{-4} pCi mL^{-1} at 135 m downwind under slightly unstable atmospheric conditions. This concentration is less than the effluent limit of 0.01 pCi mL^{-1}. A full year of exposure at the maximum concentration would lead to an effective dose of only about 3 mrem, well within applicable limits.	11-3	11.1.1.a	Offsite Impact of ^{41}Ar. As shown in Chapter 11 Appendix A, the peak off-site activity concentration during normal operations would be about 0.003903 pCi mL^{-1} at 135 m downwind under slightly unstable atmospheric conditions, occurring 0.6% of total time. This concentration is less than the effluent limit of 0.01 pCi mL^{-1}. A full year of operation at the maximum power level maximum concentration would result in an effective dose at the receptor with the maximum concentration of only about 0.16 mrem, well within applicable limits. The highest dose to a location occurs at 2140 meters with a dose of 3.8 mrem, well below the maximum allowed 10 mrem from effluents.	Power level for bounding analysis
11-3	11.1.1.a	As shown in chapter 11 appendix A, very conservative calculations lead to an expected exposures rate of 40 mR h^{-1} at one meter above the center of the reactor tank during sustained operations are 500 kW thermal power. Measured exposure rates are about 20 mR h^{-1} at 250 kW operations.	11-3	11.1.1.a	As shown in Chapter 11 Appendix A, conservative calculations lead to an expected exposure rate of approximately 25 mR h^{-1} at one meter above the center of the reactor tank during sustained operation at 500 kW thermal power, increasing to nearly 100 mR h^{-1} at 1,250 kW.	Power level for bounding analysis
11-4	Table 11.1b	Gamma Cell location as room 3	11-5	Table 11.1b	Gamma Cell location as room 14	Change in location
11-11	11.1.5.b, Access Control During Ops	No corresponding text	11-11	11.1.5.b, Access Control During Ops	The 22-foot level access has a line of sight to the control room, and has radiation monitoring positioned directly over the pool surface and mounted on the rail surrounding the pool. The operator at the controls is responsible for appropriately controlling access to the 22-foot level based on radiological conditions.	Power level for bounding analysis
11-13	11.1.6	Acceptable surface contamination levels for unconditional release are given in Table. 11.5, Limits on average contamination levels for unconditional release are calculated based on survey areas smaller than 1 m^2. Limits on maximum contamination levels for unconditional release are calculated based on survey areas smaller than 100 cm^2.	11-13	11.1.6	Acceptable surface contamination levels for unconditional release are given in Table. 11.5, as provided in the approved *Radiation Protection Program, KSU TRIGA Mark II Nuclear Reactor Facility*. Limits on average contamination levels for unconditional release are calculated based on survey areas smaller than 1 m^2. Limits on maximum contamination levels for unconditional release are calculated based on survey areas smaller than 100 cm^2.	RAI # 15

11-14	11.1.7.d	No corresponding text	11-14	11.1.7.d	As required by 10 CFR 20.1501, contamination surveys are conducted to ensure compliance with regulations reasonable under the circumstances to evaluate the magnitude and extent of radiation levels; concentrations or quantities of radioactive material; and potential radiological hazards. Guidance has been promulgated in IE Circular No. 81-07 (Control of Radioactively Contaminated Materials) for releasing materials from restricted to unrestricted areas: *Based on the studies of residual radioactivity limits for decommissioning (NUREG-06132 and NUREG-07073), it can be concluded that surfaces uniformly contaminated at levels of 5000 dpm/100cm2 (beta-gamma activity from nuclear power reactors) would result in potential doses that total less than 5 mrem/yr. Therefore, it can be concluded that for the potentially undetected contamination of discrete items and materials at levels below 5000 dpm/100cm2, the potential dose to any individual will be significantly less than 5 mrem/yr even if the accumulation of numerous items contaminated at this level is considered.* The contamination monitoring using portable survey instruments or laboratory measurements should be performed with instrumentation and techniques (survey scanning speed, counting times, background radiation levels) necessary to detect 5000 dpm/100 cm2 total and 1000 dpm/100 cm2 removable beta/gamma contamination. Instruments should be calibrated with radiation sources having consistent energy spectrum and instrument response with the radionuclides being measured. If alpha contamination is suspected appropriate surveys and/or laboratory measurements capable of detecting 100 dpm/100 cm2 fixed and 20 dpm/100 cm2 removable alpha activity should be performed.	RAI # 15
11.A-1	A.1	No corresponding text			Normal operation of the KSU reactor results in two potential source terms for radioactive gaseous effluent, ^{41}Ar and ^{16}N. There are variations in experimental configuration and possible scenarios where the production of ^{41}Ar may be different than the routine operations; these scenarios do not produce not long term, routine radioactive effluent but need to be assed to determine if the amount of radioactive effluent is so high as to impact the annual exposure that might result from routine operations.	Clarification
11.A-1	A.1.2	The concentration to dose rate (effective dose equivalent) conversion factor for submersion in an infinite atmosphere of ^{41}Ar is as follows: 2.3×10^{-10} Sv h^{-1} per Bq m^{-3} (EPA 1993).	11.A-2	A.1.2	The concentration to dose rate (effective dose equivalent) conversion factor for submersion in an infinite atmosphere of ^{41}Ar is as follows: 2.17×10^{-10} Sv h^{-1} per Bq m^{-3}, or 0.803 mrem/h per pCi/ml (EPA 1993).	Power level for bounding analysis
11-A-2	Table A.1	Parameters for 500 kW	11-A-2	Table A.1	Changed parameters for 1,250 kW	Power level for bounding analysis

Various	Various	In some cases, SI or SAE units used; in some cases, 500 kW used	Various	Various	Included both sets of units for fundamental measurements; used 1,250 kW for maximum power	Power level for bounding analysis & Add infomraiton
11.A-4	A.2.2	No corresponding text	11.A-4	A.2.2	Operation with a fully open beam port is not a routine operational condition. Beam port operations normally have shielding, collimation and beam stops that prevent a full beam from penetrating the column defined by the beam port into air volume between the reactor and the reactor bay wall.	Clarification
11.A-4	A.2.1	or 3.42×10^{-9} µCi mL^{-1} in conventional units. Operations at maximum power are not performed for radiography, and radiography is not performed long enough to achieve equilibrium ^{41}Ar. Therefore, scaling the calculation for sustained operations at 500 kW provides an extremely conservative bound on ^{41}Ar production. Fifty times the 10 kW ^{41}Ar production value results in 1.7×10^{-7} µCi mL^{-1} which meets the submersion DAC for occupational exposure with no further consideration. This value is slightly higher than the effluent limit for continuous release; meeting the effluent release limit id assured at K-State through conservatisms in the calculations and because the reactor is not operated continuously.	11.A-4	A.2.2	or 3.42×10^{-9} µCi mL^{-1} in conventional units. Operations at maximum power are not performed for radiography, and radiography is not performed long enough to achieve equilibrium ^{41}Ar. Therefore, scaling the calculation for sustained operations at 1,250 kW provides an extremely conservative bound on ^{41}Ar production. Scaling the 10 kW ^{41}Ar production value to 1,250 kW results in 4.28×10^{-7} µCi mL^{-1} which is slightly above submersion DAC for occupational exposure; however, conditions for the source term are related to a very unusual set of conditions (open beam port with no shielding) that are not continuous in two respects. Shielding for radiography external to the bema port limits the beam to less than ½ of the analyzed volume. Radiography configuration is implemented only for radiography operations, a small fraction of all operations. Typically radiography occurs less than 1 day per month. Radiography operations are inherently discontinuous as the purposes of individual operations are met when the image is obtained. Typically a day of radiography operations involves less than 4 hours of operation at full power. These conservatisms assure DAC and effluent limits are met with no further consideration.	Power level for bounding analysis
11.A-4	A.2.2	No corresponding text	11.A-4	A.2.2	The air volume in the rotary specimen rack does not freely exchange with the air in the reactor bay; there is no motive force for circulation and the rotary specimen rack opening is routinely covered during operation. If the rotary specimen rack were to flood, water would force the air volume in the RSR into the reactor bay.	Clarification
11.A-5	A.2.2	This is well below the 6×10^{-3} µCi h mL^{-1} annual limit of 2000 DAC hours specified in 10CFR20.	A.2.2	A.2.2	The value 1.6×10^{-6} µCi s mL^{-1}, or 4.4×10^{-10} µCi h mL^{-1}, is well below the 3×10^{-6} µCi h mL^{-1} annual limit of 2000 DAC hours specified in 10CFR20/EPA-520/1-88-020.	RAI # 16
11.A-5	A.2.3	No corresponding text	11.A-5	A.2.3	The reactor tank water surface is open to the reactor bay. Radioactive ^{41}Ar is circulated in the pool by convection heating, and freely exchanges with the reactor bay atmosphere during normal operation.	Clarification
NA	NA	NA	11.A-6	Table A.2	Added	Clarification

11.A-6	A.2.3	See previous version	11.A-8	A.2.3	The equilibrium ^{41}Ar concentration during full power steady state operation at 1,250 kW in the reactor bay would be 0.072 Bq cm^{-3} (1.9×10^{-6} µCi mL^{-1}) without ventilation and 0.027 Bq cm^{-3} (7.2×10^{-7} µCi mL^{-1}) with ventilation. Environmental Protection Agency, Federal Guidance Report No. 11 (EPA FG 11 -- Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion) lists the DAC for ^{41}Ar as (Table 1.b) $3X10^{-6}$ µCi/ml, or 3 pCi/ml. Therefore, equilibrium ^{41}Ar concentration during full power steady state operation at 1,250 kW is less than DAC and there are no restrictions on activities in the reactor bay imposed by the normal Ar^{41} production mode.	Power level for bounding analysis
11.A-6	A.2.4	No corresponding text	11.A-6	A.2.4	Although there are three modes of ^{41}Ar production, only the release of radioactive argon dissolved in water occurs routinely.	Clarification
11.A-6	Table A.3	Concentration values for 500 kW	11.A-7	Table A.3	Concentration values for 1,250 kW	Power level for bounding analysis
11.A-6	A.2.4	No corresponding text	A.2.4	11.A-7	The dose conversion factor provided by EPA FG 11 for ^{41}Ar (Table 2.3) is 2.17×10^{-10} Sv/hr per Bq/m^3, or (using the provided conversion factor of 3.7×10^{10}) 8.03×10^{3} mrem/hr per µCi/cm^3, 8.03×10^{-1} mrem/hr per pCi/cm^3. Using the highest maximum concentration of Table A.3 (0.003903 pCi cm^{-3}) at steady state full power operation for a full year (8760 hours) with observed frequency of class A stability (see Appendix 2.C) would result in a dose less than 1 mrem/year. Frequency of occurrence and the concentration at the maximum dose will occur from class C conditions, with a maximum annual dose of 1.7 mrem. The maximum concentration at the highest frequency (class G) is 0.001013 pCi cm^{-3}, with a dose of 3.8 mrem. The assumed 24-7 operating history is not feasible for the KSU reactor, which has an average operating time for two decades of about 8 hours/week. Additionally, a full power, continuous operation would require a significant quantity of new fuel. Note that over the full range of conditions examined in Table 2.3, the peak downwind concentration is substantially below the DAC of 3 pCi cm^{-3} established in 10CFR20 Appendix B, and less than the permissible effluent concentration of 1×10^{-8} µCi/cm^3 for all meteorological conditions except the set of conditions with the lowest frequency of occurrence; for that stability classification, the instantaneous effluent concentration is slightly higher than the DAC.	Clarification, add information

11.A-7	A.2.5	Substantial rewrite	11.A-8	A.2.5	Substantial rewrite	RAI # 17, Clarification, add information
CHAPTER 5						
12-6	12.1.2.b	The Radiation Safety Officer reports to the Manager of the Department of Environmental Health and Safety. The Radiation Safety Officer, or an authorized representative, shall be available (upon due notice) for advice and consultation regarding radiation surveys and radiation safety in connection with isotope production and radiation streaming problems as might arise in connection with reactor operation or experimentation. The Radiation Safety Officer is *ex officio* a member of the Kansas State University Radiation Safety Committee. The Radiation Safety Officer serves *ex officio* as a member of the Reactor Safeguards Committee, with any action of the Committee requiring approval of the Radiation Safety Officer.	12-5	12.1.2.b	The Radiation Safety Officer reports to the Manager of the Department of Environmental Health and Safety. The Radiation Safety Officer, or an authorized representative, shall be available (upon due notice) for advice and consultation regarding radiation surveys and radiation safety in connection with isotope production and radiation streaming problems as might arise in connection with reactor operation or experimentation. The Radiation Safety Officer is *ex officio* a member of the Kansas State University Radiation Safety Committee. The Radiation Safety Officer serves *ex officio* as a member of the Reactor Safeguards Committee, with any action (i.e., concerning potential radiation exposure or radioactive effluents) of the Committee requiring approval of the Radiation Safety Officer.	RAI # 18
12-7	12.1.3	Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator who is designated as Reactor Supervisor. The Supervisor shall be on call, within ten minutes travel time to the facility, and cognizant of reactor operations.	12-7	12.1.3	Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator who is designated as Reactor Supervisor. The Supervisor shall be on call, within twenty minutes travel time to the facility, and cognizant of reactor operations.	Change
12-13	12.5.2	A report shall be made within 10 days in writing to the NRC Operation Center for any violation of safety limit or reportable occurrence	12-7	12.5.2	A report shall be made within 14 days in writing to the NRC Operation Center for any violation of safety limit or reportable occurrence	RAI # 19
12-15	12.6.3	No corresponding text	12-15	12.6.3.f	Corrected and as-built facility drawings	RAI # 20
12-15	12.8	A physical security plan for protection of reactor plant shall be established and followed in accordance with NRC regulations.	12-15	12.8	Administrative controls for protection of the reactor plant shall be established and followed in accordance with NRC regulations.	Correction
CHAPTER 13						
13-1	13.1	These are the three conditions considered in the initial licensing of the Reactor Facility in 1962 for 100-kW steady-state operation and in the 1968 upgrade of the license permitting 250-kW steady state operation and 250-MW pulsing operation. The analysis presented here treats the same conditions, but for steady-state operation at 500 kW and pulsing operation at 1,000 MW.	13-1	13.1	These are the three conditions considered in the initial licensing of the Reactor Facility in 1962 for 100-kW steady-state operation and in the 1968 upgrade of the license permitting 250-kW steady state operation and 250-MW pulsing operation. The analysis presented here treats the same conditions, but for steady-state operation at 1,250 kW and pulsing operation to a $3.00 reactivity insertion, estimated peak power of 1,340 MW.	Power level for bounding analysis
various	various	500 kW	various	various	1,250 kW	Power level for bounding analysis
13-8	13.2.2.e	Fuel and cladding temperatures are reported in Table 13.2 and illustrated in Figure 13.6 for the case of zero time post accident.	13-8	13.2.2.e	Fuel and cladding temperatures are reported in Table 13.6 and illustrated in Figure 13.2 for the case of zero time post accident.	Correct references
13-8	Table 13.6	Values for 500 kW	13-8	Table 13.6	Values for 1,250 kW	Power level for bounding analysis

13-13	13.2.2.g	Although a loss of pool water is considered to be an extremely improbable event, calculations show the maximum fuel temperature that could be expected to result from such an event (after long-term operation at full power of 500 kW) is 261°C, well below any safety limit for TRIGA reactor fuel.	13-13	13.2.2.g	Although a loss of pool water is considered to be an extremely improbable event, calculations show the maximum fuel temperature that could be expected to result from such an event (after long-term operation at full power of 1,250 kW) is 294°C, well below any safety limit for TRIGA reactor fuel.	Correction
13-13	Table 13.9	4 table values in error	13-13	Table 13.9	R/hr values, 1,250 kW	Correction
13-14	13.2.3	Rapid compensation of a reactivity insertion is the distinguishing design feature of the TRIGA reactor.	13-14	13.2.3	Rapid compensation of a reactivity insertion is the distinguishing design feature of the TRIGA reactor. Characteristics of a slow (ramp) reactivity insertion are less severe than a rapid transient since temperature feedback will occur rapidly enough to limit the maximum power achieved during the transient....	RAI # 24
13-15	13.2.3	No corresponding text	13-15	13.2.3	• A control rod interlock preventing pulsing operations from power levels greater than a maximum of 10 kW is not credited • Conservative hot channel factors as calculated in 4.5.3 are used	Clarification
13-15	13.2.3.b	If T_o is the average core temperature at the start of the excursion, the maximum temperature rise (°K) is given by	13-15	13.2.3.b	A maximum pulse of $3.00 would result in a power rise of approximately 1430 MW(t). If T_o is the average core temperature at the start of the excursion, the maximum temperature rise (°K) is given by	Power level for bounding analysis
13-15	13.2.3.b	Extensive rewrite	13-15	13.2.3.b	Extensive rewrite	Clarification
13-16	13.2.3.c	**CASE I: Analysis of a 2.1% ($3.00) Reactivity Insertion at Zero Power** For this case, Eqs. (13.2.3-2) and (13.2.3-3) yield a power rise of 1430 MW and a core-average fuel temperature rise of temperature of 229°K. Peak temperature rise would be the core average multiplied by the overall peak-to average ratio of π, resulting in a hot spot temperature of 27 + 719 = 746°C. Therefore, insertion of the maximum possible reactivity without initial temperature feedback (i.e., fuel temperature is too low to limit core available reactivity) results in fuel temperatures well below the safety limit. **CASE II: Analysis of a 0.7% ($1.00) Reactivity Insertion at 94 kW Power** For this case, Eqs. (13.2.3-2) and (13.2.3-3) also yield a power rise of 1430 MW and a core-average fuel temperature rise of temperature of 229°K. In this case, the hot spot temperature is 150 + 719 = 869°C. Therefore, insertion of the maximum possible reactivity with initial temperature feedback (i.e., fuel temperature limits available) results in fuel temperatures well below the safety limit.	13-16	13.2.3.c	Insertion of the maximum possible reactivity without initial temperature feedback (i.e., fuel temperature is too low to limit core available reactivity) results in a peak hot spot fuel temperature of 746°C, well below the safety limit. Insertion of the maximum possible reactivity with initial temperature feedback (i.e., fuel temperature limits available) results in a peak hot spot fuel temperature of 869°C, well below the safety limit.	RAI # 22, 23

13.24.a.2	13-17	For short-lived radionuclides, calculations of radionuclide inventory in fuel are based on operation at the full thermal power of 500 kW for eight hours per day, for five successive days prior to fuel failure.	13-17	13.24.a.2	For short-lived radionuclides, calculations of radionuclide inventory in fuel are based on operation at the full thermal power of 1,250 kW for eight hours per day, for five successive days prior to fuel failure, an average of 31.25 kW-hr/day.	Power level for bounding analysis
13-17	13.2.4.b	For times much greater than the half-life of the radionuclide, $A \approx A_{\infty}$, and for times much less than the half-life, $A(t) = A_{\infty} * ? * t$.	13-17	13.2.4.b	For times much greater than the half-life of the radionuclide, $A \approx A_{\infty}$, and for times much less than the half-life, $A(t) = A_{\infty} * \lambda * t$.	Correction/typographical error
13-19	Table 13.10	Values for 500 kW	13-20	Table 13.10	Values for 1,250 kW	Power level for bounding analysis
13-20	Table 13.11	Values for 500 kW	13-21	Table 13.11	Values for 1,250 kW	Power level for bounding analysis
13-22	13.24.h	Reference to table 12.14 incorrect	13-22	13.24.h	Changed to 13.22	Incorrect table no.
13-22	Table 13.12	Values for 500 kW	13-23	Table 13.12	Values for 1,250 kW	Power level for bounding analysis
13-23	Table 13.13	Values for 500 kW	13-24	Table 13.13	Values for 1,250 kW	Power level for bounding analysis
App 13.F	App 13.F	Values for 500 kW	App 13.F	App 13.F	Values for 1,250 kW	Power level for bounding analysis
CHAPTER 14: Addressed Separately						
CHAPTER 15: No changes						

KSU Research Reactor Docket No. 50-188 Responses to Request for Additional Information				
No.	SAR Page	SAR Section/Table/Figure	NRC Request for Information (or Question)	Response
1	1-16	Table 1.3	The bottom of this table is missing. In particular footnote 1 is missing. Please provide complete table.	Table corrected, footnote added
2	4-7	Section 4.2.2	Please discuss the construction of the control rods. Are they located within guide tubes? Arte the control rods constructed with followers?	Section 4.2.2, Control rods, rewritten
3	4-13	Section 4.5.3.b (Bullet 6)	The value of $\pi/2$ appears to be in error	Documented use of $\pi/2$ as conservative in text
4	4-17	Section 4.7	Please discuss how arbitrarily declaring the fuel temperature safety limit as 1000°C "...ensures that the maximum temperature limits indicated in analysis cannot be achieved.	Section 4.7 rewritten
5	4-17	Section 4.7	Please discuss the need for two different fuel temperature safety limits (i.e., 1150°C with the clad <500°C and 950°C with the clad temperature equal to the fuel temperature). Under what conditions would the latter limit be valid?	Section 4.7 rewritten
6	4-18	Section 4.8.2	If power level is the Limiting Safety System setting (LSSS) during steady state (SS) operation and peak fuel temperature is the Safety Limit parameter please provide the correlation between power level and peak fuel temperature SS operation. What is the expected highest peak measured fuel temperature during 500 kW steady state (SS) operation?	Section 4.8.2 rewritten
7	4-18	Section 4.8.2, Table 4.5	Please discuss how the Limiting Safety System Setting (LSSS) was determined. Please clarify the correlation between the licensed power level, and the LSSS	Section 4.8.2 rewritten
8	4-18	Section 4.8.2, Table 4.5	Is the instrumented fuel element used with the Limited Safety System Setting (LSSS) always located in the B ring? If not, please discuss how the radial location for the thermocouple is in the instrumented element affects the ratio of measured peak fuel temperature/actual peak fuel temperature during SS vs. pulse operation	Removed LSSS on fuel temperature

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No.	SAR Page	SAR Section/Table/Figure	NRC Request for Information (or Question)	Response
9	4-18	Section 4.8.2, Table 4.5	Please discuss how the LSSS of 600°C provides assurance that the safety limit is not exceeded. How was the LSSS value of 600°C determined?	Removed LSSS on fuel temperature
10	7-12	Table 4.5	This table indicates that the LSSS is for the steady state mode, however, the Technical Specification (TS) section 2.2.1, Applicability, states that the LSSS applies only in the pulse mode. In addition TS, Section 3.4.3, Table 1 indicates that the fuel element temperature measuring channel is not applicable during SS mode. Please discuss.	Removed LSSS on fuel temperature
11	7-12	7.3.4	This section indicates that four control rods are required for 500 kW operation. Is this also the case for operation up to 250 kW? Please justify not having an LCO for the number of control rods.	Discussed reason for 4 versus three as related to reactivity control
12	7-12	7.3.4	The second sentence in this section states that three control rods have identical circuitry. In the next sentence there is a statement that two rod drives are original analog systems. Then it is state that one drive uses a stepper motor. Please provide a simple, circuit diagram showing the stepper motor circuit and interface.	7.3.4 (and associated material) rewritten
13	7-12	7.3.4	Please describe the drive that utilizes the stepper motor. In particular describe the limitations on rate of speed, position indication of the control rod to the operator, and failure modes and effects of the drive.	7.3.4 (and associated material) rewritten
14	11-3	1st line of text	The value of $6X10^{-4}$ µci ml^{-1} appears to be incorrect	Corrected
15	11-13	Table 11.5	What is the reference for this table? Please discuss how the information in this table will be used in radiation and waste management	Inserted reference as approved RPP; included information from Generic Communications
16	11.A-5	2nd sentence	The value of $6X10^{-3}$ µci ml^{-1} appears to be incorrect	Corrected
17	11.A-7	1st sentence	The calculation of dose appears to be in error low	Corrected

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18	12-6	Section 12.1.2.b	The RSO has veto power in the Reactor Safeguards Committee. IS there a process established to override the veto?	Revised to indicate veto power over radiation control issues
19	12-13	Section 12.5.2	What is the reason to restrict the 14-day report to within 10 days	Corrected
20	12-15	Section 12.6.3	Please include the update, corrected, and as-built d\facility drawings in this section as indicated in TS Section 6.10.b)6	Included
21	13-16	Section 13.2.3(3) Case 1 and Table 13.4	Please discuss the calculation of the peak to average core temperature ratio equal to π or provide a reference. π appears to be too high a value for this parameter for your reactor.	Use of the factor discussed in rewritten material of Chapter 4 & referenced
22	13-16	Section 13.2.c Case II	Please discuss the limitation of the initial power for Case II to 94 kW. Please discuss the possibility of an experiment reactivity change while at power greater than that analyzed. Please correct the inconsistency between this analysis and TS Section 3.1.5 with regard to the initial power.	Rewritten
23	13-16	Section 13.2.3(3) Case I & II	Please discuss the conclusion that the core power rise will be the same for 2.1% and 0.7% insertion of reactivity.	Rewritten
24	13-16	Also TS Section 5.3	Justify not analyzing a ramp accident and using the results as bases for the LSSS and the reactivity change rate limits for moveable experiments and control rod motion. How are the consequences of such accidents limited?	Discussed ramp as being bounded by pulse