REDACTED VERSION

[redacted]

INFORMATION

REDACTED VERSION

TEXAS ENGINEERING EXPERIMENT STATION

TEXAS A&M UNIVERSITY

3575 TAMU
COLLEGE STATION, TEXAS 77843-3575

NUCLEAR SCIENCE CENTER
979/845-7551
FAX 979/862-2667

February 27, 2003

2003-0024

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: Application for Renewal of R-83 Docket Number 50-128

This memo and the enclosed Safety Analysis Report (SAR) serve as application for 20-year renewal of the Texas A&M Nuclear Science Center with the listed license and docket number. The combination of this letter and the enclosed SAR meet the requirements of 10CFR54.

The Nuclear Science Center Emergency Plan, Physical Security Plan and Operator Requalification Plan will have no associated changes. The enclosed SAR addresses the financial and environmental issues concerning the continued operation of the Nuclear Science Center.

Sincerely,

Warren D. Reece, Director
TAMU Nuclear Science Center

WDR/ym

xc: 2.11/Central File
Marvin Mendonca, USNRC
USNRC, Region IV

A053
A020
A00

1/14

SAFETY ANALYSIS REPORT

for the

Nuclear Science Center Reactor
Texas A&M University
Texas Engineering Experiment Station

February 2003

ABSTRACT

This document supports of the renewal of License R-83 and supercedes all previous submittals in Docket 50-128. This Safety Analysis Report (SAR) is a consolidated and updated safety analysis for the continued operation of the Nuclear Science Center Reactor (NSCR) using standard TRIGA and/or FLIP TRIGA fuel and contains previously reviewed material from the August 1967 and June 1979 SARs and their supplements

The purpose of this SAR is to provide a description and safety analysis of structures, systems and components in terms of their ability to provide proper operational performance and functions for the twenty-year term of the license renewal The continual upgrading that has been implemented since the initial operation of the NSCR has improved reactor safety and prevented the need for restrictions on reactor operations due to age of structures or equipment.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 THE FACILITY

1.1 Introduction

The Texas Engineering Experiment Station owns and operates the nuclear reactor facility located at Texas A&M University named the "Nuclear Science Center" The facility is a university-operated research reactor designed to provide a center for the university's students in various disciplines and for outside research and commercial users.

The facility, located on the West end of Texas A&M University campus in College Station, Texas, houses a TRIGA (Teaching, Research, and Isotopes, General Atomic) reactor utilizing FLIP, LEU and/or Standard TRIGA fuel with a maximum operating power level of 1.0 MW.

Principal and inherent safety features include passive shutdown (SCRAM) capability and negative temperature/power feedback. This Safety Analysis Report contains documentation and basic information as well as considerations to support the conclusion that the Nuclear Science Center (NSC) can operate safely. This document supports the renewal of NRC License R-83.

1.2 Summary and Conclusions on Principal Safety Considerations

The primary safety features of a TRIGA-type reactor are from the use of a fuel with a strong negative prompt temperature coefficient of reactivity, which limits the excursions from reactivity insertions, thus preventing fuel damage from credible reactivity accidents. Ejection of the transient rod from the core, when the core is operating at the power-level scram point will result in no fuel damage. Since experiments are limited to less reactivity worth than the transient rod, experiment failure cannot cause a more severe accident

The operating power level of 1 MW (limited to 70 hours per week) results in decay heat potential in the fuel small enough that a loss of reactor coolant does not result in fuel damage or release of fission products

The only case where significant exposure occurs requires the simultaneous failure of the fuel element cladding, catastrophic failure of the pool and liner and a failure of the ventilation system with personnel remaining within the reactor facility for a period of 1 hour after release. This would result in a maximum thyroid exposure of 49R. Thus, no realistic hazard of consequence will result from the Design Basis Accident.

1.3 General Description of the Facility

The Texas A&M Nuclear Science Center houses the TRIGA pool-type reactor in a dedicated building off the main part of campus. Figure 1-1 is a photograph of the front of the Reactor Building and adjacent Laboratory Building.

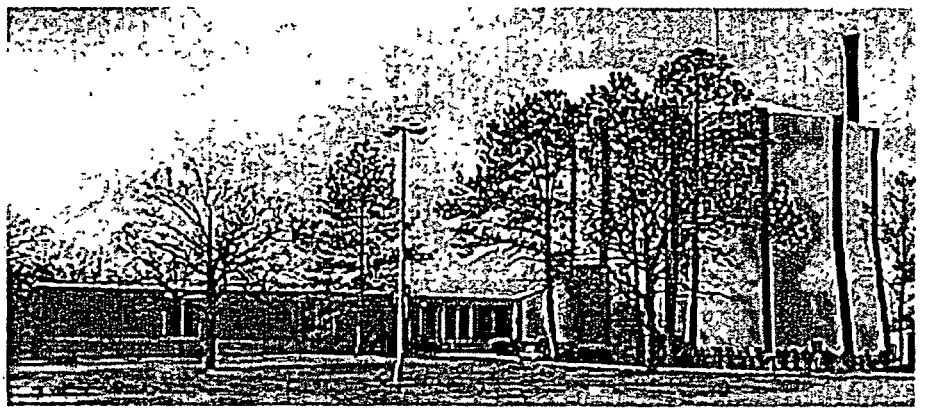


Figure 1-1: Nuclear Science Center Reactor and Laboratory Buildings

The reactor is a 1-MW pool-type nuclear reactor The current configuration uses 4-element bundles with TRIGA-FLIP fuel Light water flows through the reactor by natural convection Graphite serves as a reflector.

A suspension frame supported by a bridge that spans the pool supports a grid block which in turn supports the fuel, reflector, control rods, samples and any other in-core material Four shim-safety control rods, a transient control rod and regulating rod control reactivity. Table 1-1 is a summary of reactor data

Table 1-1: Summary of Reactor Data

Responsible Organization	Texas Engineering Experiment Station
Location	College Station, TX
Purpose	Teaching and Research
Fuel	
Type	TRIGA-Standard, TRIGA FLIP, and TRIGA LEU
Number of elements (nominal)	[redacted] (Including Fuel-Followed Control Rods)
Control	
Safety Elements	4 Shim-Safety Control Rods
Regulating Element	1 Servo-controlled Control Rod
Transient Control	1 Pneumatic Operated Control Rod

1.4 Shared Facilities and Equipment

The Nuclear Science Center shares utilities with an accelerator facility that is part of the Nuclear Engineering Department. This accelerator shares a building with Nuclear Science Center auxiliary shops and thus shares a wall and electrical distribution system. This building is external to the confinement building. The Nuclear Science Center shares no other facilities or equipment.

1.5 Comparison with Similar Facilities

The reactor has operated with a full FLIP core or FLIP/Standard mixed core since 1973. This provides the greatest operational history with which to compare.

The University of Wisconsin and Washington State University operate 1-MW reactors using TRIGA FLIP and Standard fuel Unlike the Texas A&M reactor, General Electric built both of these reactor pools and so the pool sizes and configurations differ from between the three reactors Nevertheless, the reactor behavior and accident analysis between the reactors is similar.

1.6 Summary of Operation

The NSCR provides the following:

- Laboratory exercises for undergrad and graduate students at TAMU,
- Neutron activation analysis facilities for numerous departments at TAMU,
- Neutron activation analysis facilities for educational institutions without a research reactor,
- A source of radioisotopes for various research and educational projects,
- Radioisotopes for the medical industry,
- Radioisotopes for several commercial organizations, and
- A neutron radiography facility for research and commercial use

1.7 Compliance with the Nuclear Waste Policy Act of 1982

In accordance with a letter from the U.S. Department of Energy (R. L. Morgan) to the U.S Nuclear Regulatory Commission (H. Denton) dated May 3, 1983, it has been determined that all universities operating non-power reactors have entered into a contract with the Department of Energy (DOE). The contract provides that DOE retain title to the fuel and DOE is obligated to take the spent fuel and/or high-level waste for storage or reprocessing.

Because Texas A&M University has entered into such a contract with DOE, the Texas A&M Nuclear Science Center has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

The initial planning for the NSCR began in 1957. At that time, the University was embarking on a program of expanding graduate education and research programs Texas A&M Administration recognized that a research reactor that would be able to serve many departments and support a large variety of research activities would significantly contribute to this development

The application for a construction permit and operating license was submitted in March 1958 along with the Hazard Summary Report. Supplement I to the Hazard Summary Report was submitted in 1959. The construction permit, Number CPRR-38, was issued in August 1959. This permit was converted to operating license R-83 which authorized operation of a MTR swimming-pool type reactor at 100 kW.

The reactor first went critical on December 18, 1961. Since that time, the use of the facility has increased steadily to its current position of supporting an active nuclear engineering educational program and various other research endeavors. The facility serves many campus departments, other universities and colleges, several city and state agencies, and other industrial and research organizations By January 1965, the use of the facility had increased to necessitate the operation of a two-shift basis three days per week and one shift operation for two days After July 1966, the reactor routinely operated two shifts for five days per week In 1968, the reactor was converted to TRIGA fuel and the power level was increased to 1,000 kW. Only three years had elapsed from initial reactor operations before a comprehensive upgrading program was implemented In December 1965, proposals were submitted to the National Science Foundation (NSF) and the Atomic Energy Commission (AEC) for funds to support a long-range expansion program.

The expansion of the facility included four separate phases and are described briefly below:

Phase I: Pool Modification and Liner

The large reactor pool was modified by installing a multipurpose irradiation cell. This facility allows exposure of large animals or other objects to the radiation from the reactor core. A permanent stainless steel liner was installed as part of the pool modification to eliminate problems of pool leakage, a source of previous significant operational problems.

Phase II: Cooling System

To allow steady-state operation at power levels up to 1.0 MW, a cooling system was provided for the reactor. The 1 0 MW reactor power improved a number of existing research programs and encouraged the initiation of new projects.

Phase III: Conversion of the Reactor Core

The reactor core was converted to employ standard TRIGA fuel elements, and on July 31, 1968, an amended facility license allowed the NSCR to be operated at a maximum steady-state power level of 1,000 kW and pulsing up to $3.00 reactivity insertion The inherent safety of the TRIGA fuel allowed increased flexibility and utilization of the reactor Pulsing was possible due to the prompt negative temperature coefficient of reactivity and the integrity of TRIGA fuel at pulse peak temperatures

Phase IV: Laboratory Building

The original research space within the Nuclear Science Center confinement building was quite limited. A laboratory building was constructed to adequately accommodate the increased research load and to allow for anticipated expansion of programs

From initiation, the plan covered a period of 3 ½ years to completion in mid-1969. The plan not only changed the initial facility physical plant but also established a new reactor program

Operating experience with standard TRIGA fuel revealed a high fuel burnup rate resulting in fuel additions to maintain sufficient reactivity. Modifying the reactor grid plate in late 1970 extended Core life by approximately 1 ½ years and provided for the installation of fuel-followed control rods Subsequent operation, however, eventually required the addition to the core of all standard TRIGA fuel on hand, which led to dramatically reduced fluxes available for experiments. The solution to this problem was the initiation of a program to provide a core loading utilizing TRIGA FLIP (Fuel Lifetime Improvement Program) fuel. In June 1973, the NSCR licensed to operate full standard, mixed, or full FLIP TRIGA cores. The mixed cores licensed to operate at a maximum steady-state power of 1,000 kW with maximum pulse reactivity insertions of $2.00. In July 1973, the first NSCR mixed TRIGA core, containing [redacted] FLIP and [redacted] standard elements, went into service (Core III) In July 1975, the maximum permissible pulse reactivity insertion increased to $2.70

On September 27, 1976, during a loading operation, four "lead" elements failed to pass through a "go/no-go" gauge. Steady state hydrogen migration followed by rapid hydrogen pressurization during reactor pulses caused the damage (GA-A16613, 1981) The reactor was not pulsed again until a complete analysis was submitted to the NRC in 1981. The maximum pulse allowed by the NSCR reduced to that amount which would not cause the reactor to exceed a temperature limit of 830°C (1525°F).

The Nuclear Science Center Reactor (NSCR) operated from 1962 until 1967 with MTR-type curved aluminum plate elements. During this time, the reactor operated extensively at a maximum power level at 100 kW. In 1968, the reactor began using TRIGA fuel at power level of 1,000 kW.[1,2] The initial core loading was quite satisfactory, but fuel burnup and samarium buildup soon affected experimental capability.[3] To restore excess reactivity, the NSC periodically added additional fuel to the core and graphite reflectors to all core faces This eventually led to a 126-element core with a resultant decrease in the flux of almost 40% and the elimination of most of the irradiation facilities.

In August 1970, the installation of fuel followed control rods lead to a gain in excess reactivity and helped solve the problem of maintaining excess reactivity. This installation required modification of the grid plate to allow passage of the fueled portion of the control rod through the grid plate.[4] Specifically this modification achieved an average $1.10 increase per fueled follower, which extended the core life nearly two years The high fuel burnup rate of standard TRIGA cores continued to be an operational problem for the NSCR The NSCR has operated approximately [redacted]-days per year since 1969.

It was obvious that a solution was needed that could fit within the constraints of a university budget and limited federal support. Replacement of the core with new fuel would have lead to considerable expense with a very short effective life of a standard core. Since the average core-burnup was only 10% and a reasonable amount of fuel would only provide small reactivity increases, Cycling new fuel into the core was no more attractive. The solution to the problem was in a new fuel developed and marketed by General Atomic [5] It is almost identical to the standard

---

[1]J.D. Randall, "Power Uprating Experience Following Conversion of a Pool Reactor From Plate-Type to TRIGA Fuel Elements," Nuclear Safety, Vol. 10, No. 6, December 1969.

[2]W.B. Wilson, et al., The Installation and Operating Characteristics of the Texas A&M University Reactor, Technical Report 23, Nuclear Science Center, Texas A&M University, August 1969.

[3]D.R. Schad & J.D. Randall, "Operational Reactivity Considerations of the Texas A&M TRIGA," TRIGA Seminar, Denver, Colorado, 1970.

[4]D.E. Feltz, P.M. Mason, J.D. Randall, "Modification of a BSR-MTR Type Grid Plate to Accept Fueled Followers," Conference on Reactor Operating Experience, American Nuclear Society, Denver, Colorado, August 8-11, 1971.

[5]F. Foushee, J R. Shoptaugh, G B. West, W.L Whittemore, "TRIGA FLIP-A Unique Long-Lived Version of the TRIGA Reactor," Trans. Amer. Nucl Soc., Vol. 14, No. 2, Miami Beach, October 1971.

TRIGA fuel except that the enrichment was 70% rather than 20%. The hydrogen to zirconium ratio decreased from approximately 1.7 to 1.6, and 1.5-weight percent natural erbium was added as a burnable poison The fuel designated as FLIP (Fuel Life Improvement Program) has a calculated lifetime of approximately 9 MW-years This contrasts with experience for a standard core, where it was possible to operate only six months (approximately one-seventh of a MW-year) without a fuel addition.

Inasmuch as funds for a complete FLIP core were not available, it was necessary to consider operation with a core comprised of a mixture of FLIP and standard TRIGA fuel A precedent for this had been established by General Atomic when they operated a standard core loaded with eighteen centrally located FLIP elements in a fuel test program.[6] Calculations at Texas A&M led to the conclusion that satisfactory core arrangements were possible with a mixed core.[7] As funds became available, the amount of FLIP fuel could increase until the core was completely FLIP fuel. This concept provides the additional satisfaction of producing substantially greater burnup in the standard fuel used in the mixed core.

Sufficient funds provided for a partial loading of FLIP fuel in a [redacted] element core with a [redacted]-element FLIP region. This configuration achieved criticality in July 1973.[8] The burnup data indicated that the burnup rate was initially 0.5¢ per MW-day and after samarium buildup the rate dropped to 0.2¢ per MW-day. Thus, the incorporation of FLIP fuel had increased the lifetime of the core by a factor of three.

The NSCR has operated with two mixed core loadings containing [redacted] FLIP and [redacted] FLIP, elements each since initial approval in June 1973 Since the late 1970, the core has operated with all FLIP fuel.

---

[6]G.B. West and J.R. Shoptaugh, "Experimental Results From Tests of 18 TRIGA FLIP Fuel Elements in the Torrey Pines Mark F. Reactor," GA-9350, September 1969.

[7]D.E. Feltz, M. Hardt & J.D Randall, "Feasibility Studies of a Mixed Core Using Standard TRIGA and FLIP Fuel," TRIGA Owners Conf. II, College Station, 1972.

[8]D.E. Feltz, T.A Godsey, M. Hardt & J.D Randall, "Performance Testing of a Mixed Core Utilizing TRIGA-Standard and TRIGA FLIP Fuel," Trans. of Amer. Nucl Soc., Vol 17-1, Myrtle Beach, August 1973.

2 SITE CHARACTERISTICS

This chapter covers the geographical, geological, seismological, hydrological and meteorological characteristics of the NSCR site and its vicinity

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

The Texas A&M Nuclear Science Center (NSC) is on a rectangular six-acre site on the Texas A&M University campus 1,500 feet from the North-South runway of Easterwood Airport. Figure 2-1 shows the relationship between the Nuclear Science Center and the Easterwood Airport runways. The facility is six miles south of the city-center of Bryan (pop 65,660), three miles southwest of the main campus of Texas A&M and two and one-half miles west-southwest of the city of College Station (pop 67,890) in Brazos County, Texas The facility location is [redacted], [redacted].

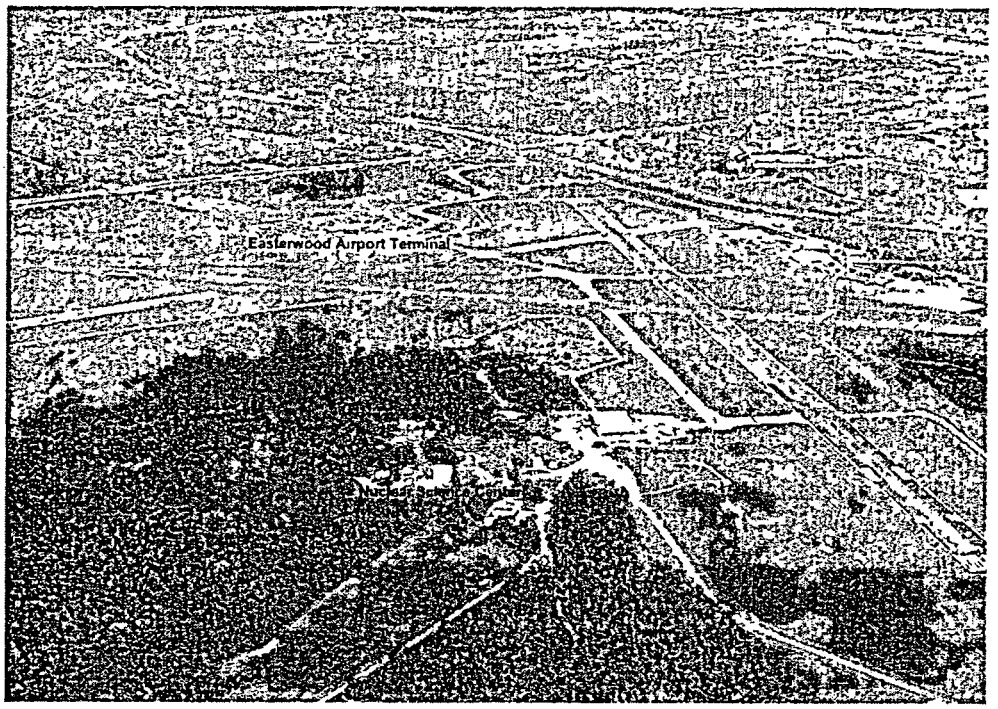


Figure 2-1: NSC and Easterwood Airport

Land owned and controlled by Texas A&M University and Easterwood Airport surrounds the site. A chain-link steel fence that provides reasonable restriction of access to the site defines the indemnity confines of the site. The main entrance into the site is through an electrically operated chain-link steel gate at the east end of the site. The entire area inside the perimeter fence of the NSC is a "Restricted Area". Located within the boundaries of the site are the reactor confinement building, reception room, laboratory building, mechanical equipment room, cooling system equipment, holding tanks and other storage and support buildings.

2.1.1.2 Boundary and Zone Area Maps

The Nuclear Science Center Site (Figure 2-2) defines the operation boundary for the NSC. In addition to the NSC, the site contains a linear accelerator and associated laboratory.

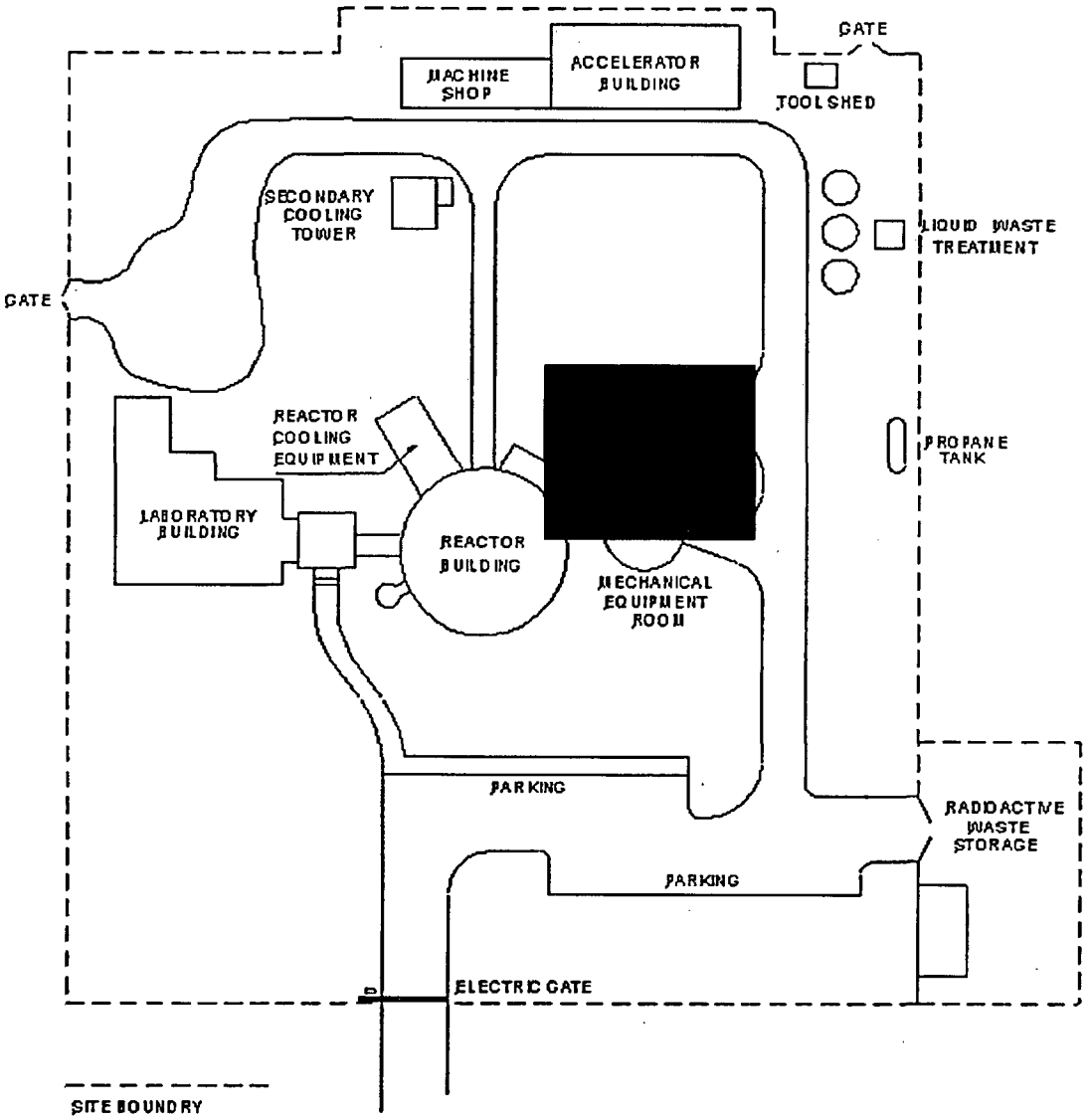


Figure 2-2: NSC Site

A map of the area surrounding the NSC (Figure 2-3) shows the major roads in the area up to a distance of 8 km from the NSC The figure includes the area for the Texas A&M University campus as well as College Station and most of Bryan. Figure 2-4, Figure 2-5, Figure 2-6 and Figure 2-7 show the floor plans for the Reactor Building and Laboratory Building.

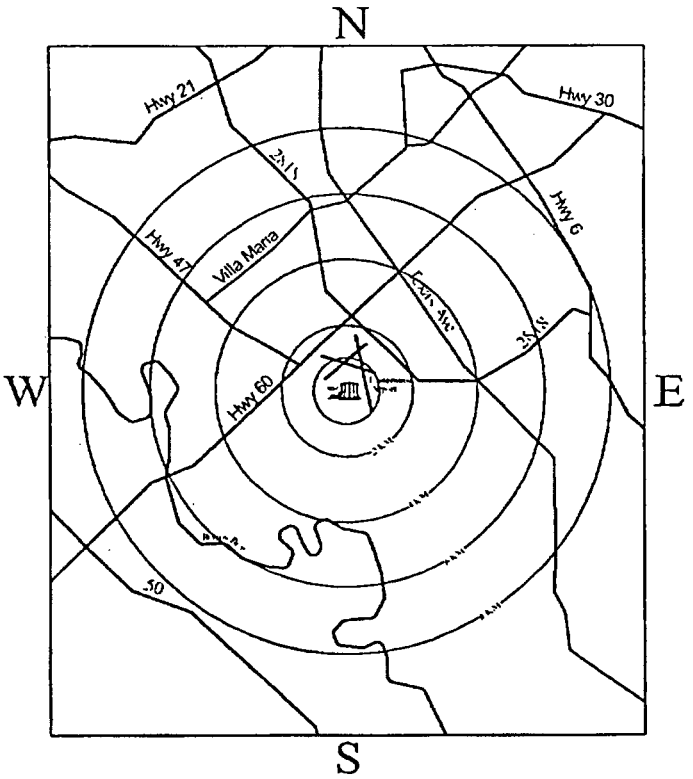


Figure 2-3: Major Roads around NSC and Easterwood

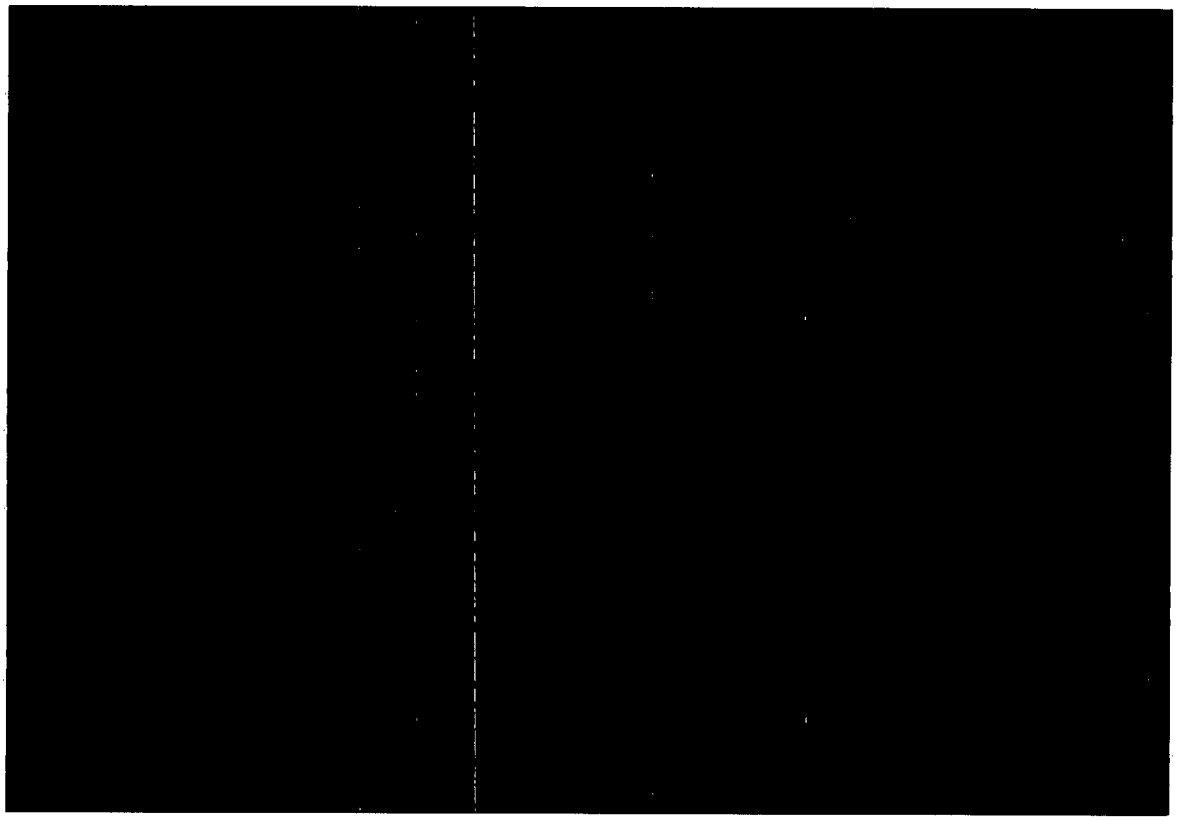


Figure 2-4: NSCR Building Cross Section

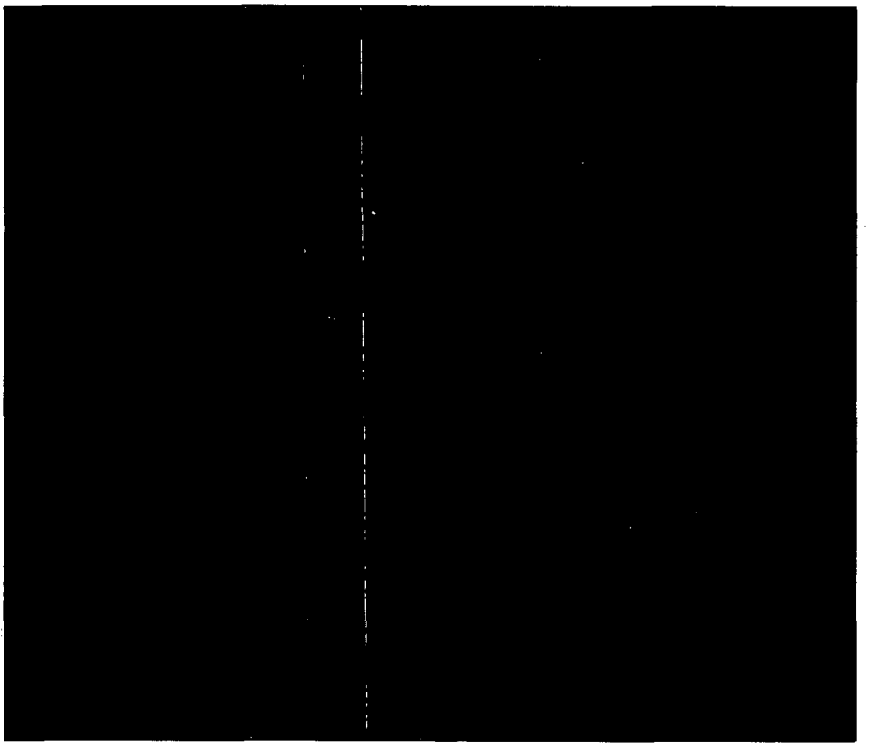


Figure 2-5: Upper Research Level

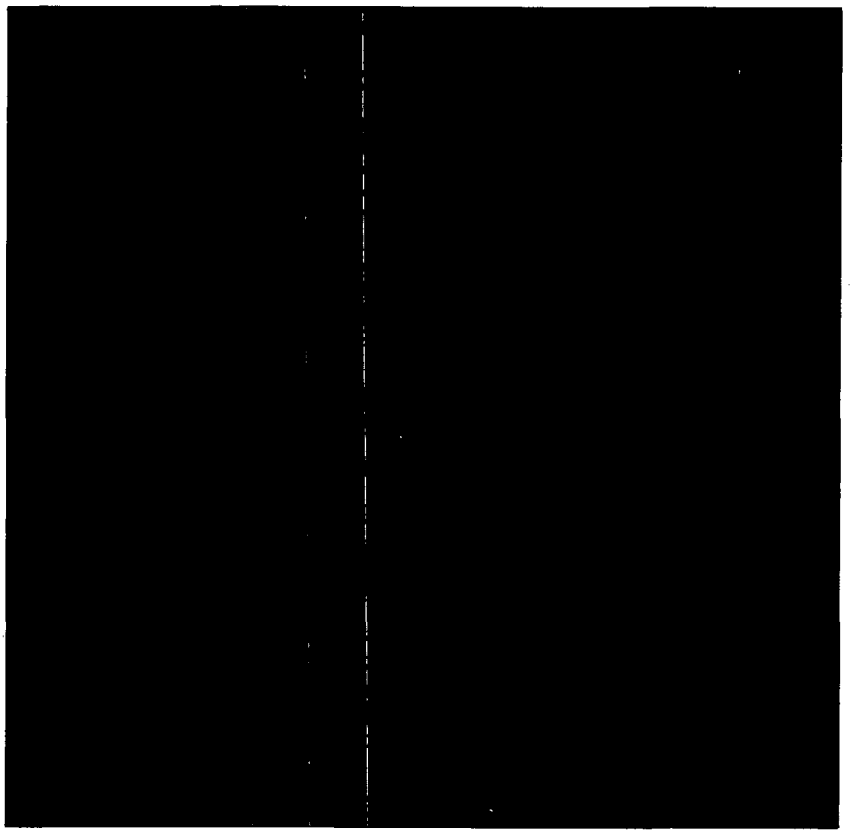


Figure 2-6: Lower Research Level

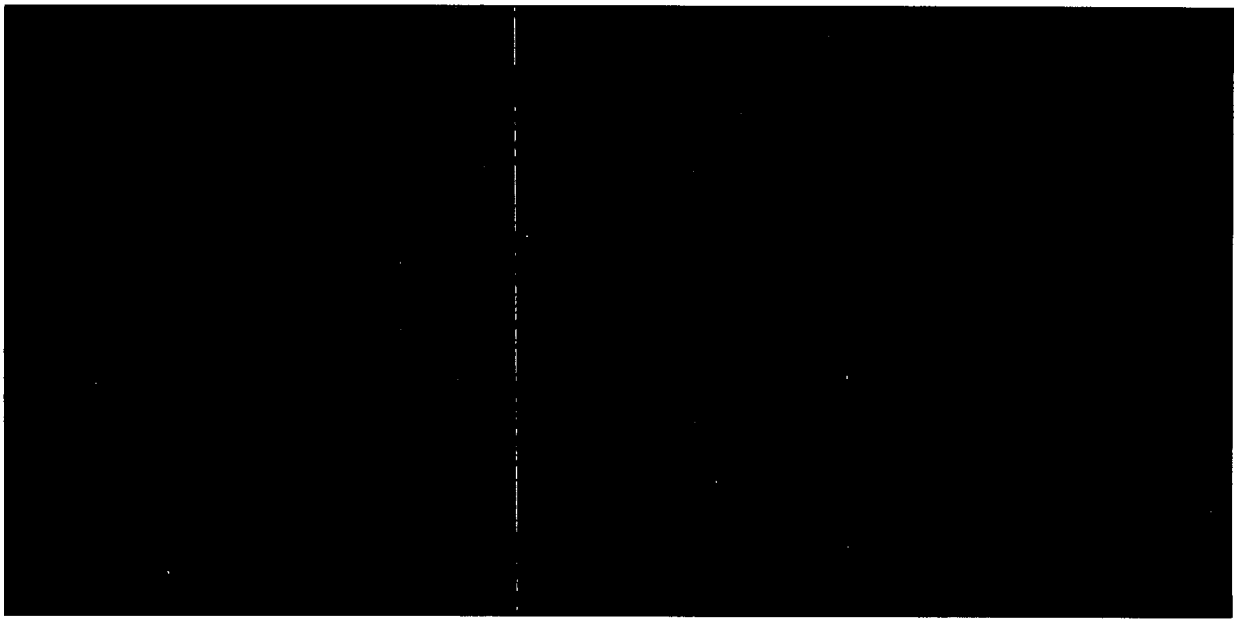


Figure 2-7: Laboratory and Reactor Building with Pneumatic Interconnection

2.1.2 Population Distribution

Table 2-1 shows estimated population maximums. The dormitories for Texas A&M are greater than 4 km and less than 6 km from the NSC. Therefore, the estimated population for 6 km will increase by approximately 10,000 during the fall and spring semesters

No residences exist or are likely within 1 km of the NSC as this property is owned by Texas A&M and Easterwood Airport A firefighter training school and a few other small facilities, within 1 km, employ dozens of people. As many as a few hundred firefighters attend training a few weeks each year. The area within 2 km of the NSC is mostly owned by Texas A&M and houses Easterwood Airport. The population in this area is low and quite stable. The area within 8 km includes much of College Station, part of Bryan and surrounding areas. The population of the 8 km area may continue to increase.

Table 2-1: Population Distribution in the NSC Vicinity

Distance from Facility (kilometers)	Estimated 2000 population
1	0
2	<1,000
4	<25,000
6	<75,000
8	<150,000

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

No industrial facilities are near the NSC.

The nearest railroad runs through the main campus and comes to within 3.5 km of the NSC.

The NSC is located approximately 1.5 km West of FM 2818 and 2 km South of Highway 60 Highway 6 is approximately 7 km East of the NSC. There is no interstate highway in the area. Refer to Figure 2-3

Easterwood Airport is in the immediate vicinity of the NSC The nearest runway is 300 meters from the NSC at its closest point The NSC is approximately 700 meters from the private and non-passenger commercial terminal and is over 1 km from he main terminal. Refer to Figure 2-1

There are no military facilities in Bryan/College Station area with the exception of the National Guard Facility

2.2.2 Air Traffic

Easterwood Airport, the only commercial airport near the NSC, is immediately adjacent to the site. Although one of the three runways is close to the NSC, none have trajectories that take commercial traffic directly over the reactor. Although commercial, private and military training flights use Easterwood Airport, arrival frequency is low and the local control tower rarely places inbound traffic in holding patterns

2.2.3 Analysis of Potential Accidents at Facilities

No industrial, transportation or military facilities within the vicinity of the NSC pose sufficient risk to the reactor to render the site unusable for operation of the reactor facility Although an airport is nearby, the construction of the NSC (the reactor is below the surface of the ground and protected by thick pool walls) and the trajectory of the runways make the magnitude of a casualty resulting from an aircraft collision and the probability of such an event low.

2.3 Meteorology

2.3.1 General and Local Climate

The Bryan/College Station area is located approximately 160 kilometers (100 miles) inland from the Texas Gulf Coast Largely, the high-pressure areas that are predominant over the Gulf of Mexico determine the local weather. As a result, warm southeasterly winds occur a large majority of the time on an annual basis (Figure 2-8). Average annual rainfall is between 76 and 89 cm (30-35 inches). Snow occurs only rarely and temperature reaches sub-freezing infrequently for brief periods during the winter. Northwest winds normally accompany the passage of the frontal systems Calms occur an average of 10% of the time, and wind speeds above 38 kph (21 knots) rarely occur.

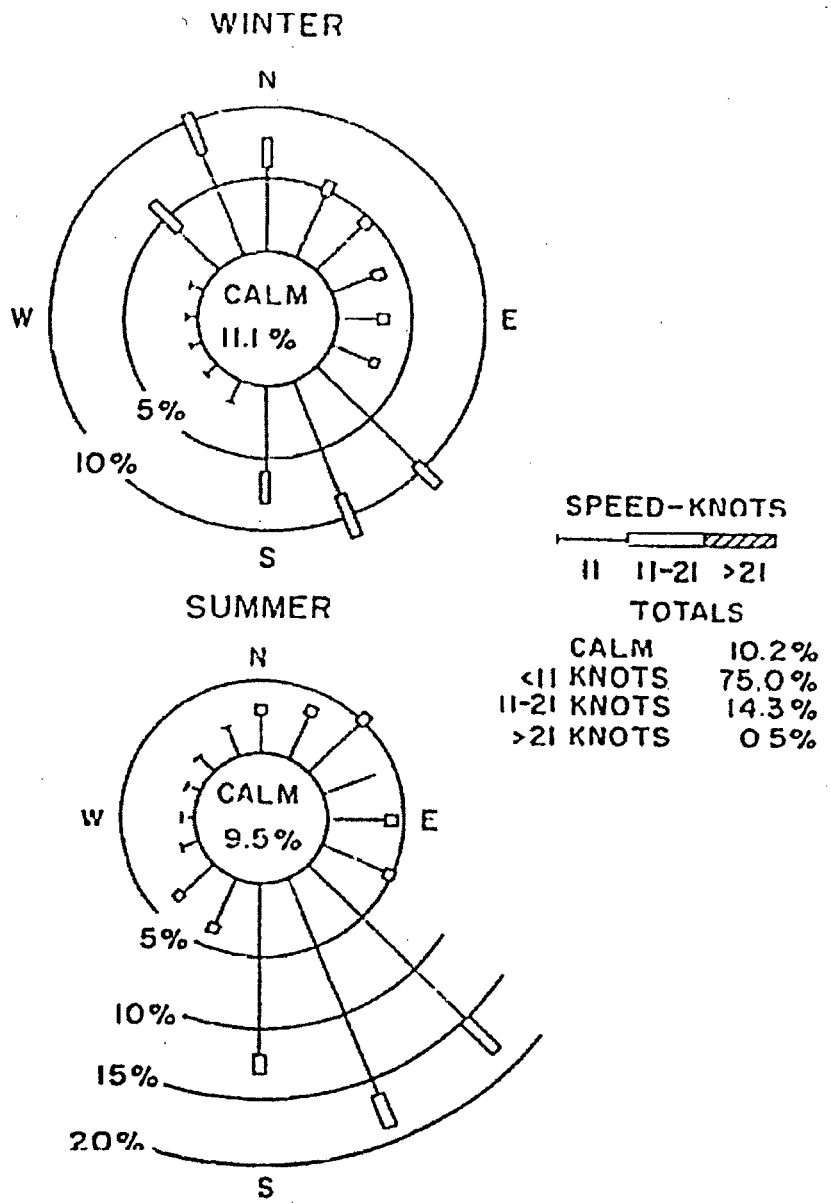


Figure 2-8: Average Wind Frequency Distribution

Tornadoes are common in Texas. Data on tornado frequency between 1950 and 2002 indicates fifteen tornadoes in Brazos County Texas during that fifty-two year period [1] Sixty percent of those occurred in May and over half occurred in the afternoon between 2:00 and 6 00 pm The season usually starts in March and reaches its peak in May.

[1]National Oceanic and Atmospheric Administration, National Climatic Data Center, http://lwf.ncdc.noaa gov/oa/ncdc.html

The reactor building design requires that it withstand 207 kPa (30 psi) over pressure with the exception of the domed roof. It will withstand only 2344 Pa (50 psf or 0 34 psi) In the case of a tornado passing nearby, the roof would probably act as a pressure relief mechanism The basic steel structure in the roof would probably remain intact unless a tornado made direct contact on the building. The reactor building design requires that it withstand a straight 145 Kph (90 mph) wind The reinforced concrete construction and round shape of the building provide a considerable strength to withstand high winds.

In the event of a tornado within an 8 km (five mile) radius of TAMU, the radio operator at the TAMU Communications Center will notify the NSC or the first available person on the NSC emergency notification roster. The radio room receives notification of tornadoes from both the TAMU weather radar and the Brazos County, Bryan/College Station Disaster Emergency Planning Organization. The method of tornado detection is by TAMU radar, area spotters and the National Weather Service (NWS)

2.4 Hydrology

Drainage of the site is by way of White Creek to the Brazos River three miles to the southwest. The facility is on high ground and the entire area drains well via a number of tributaries of White Creek Based on history, the site, which is approximately 92 meters (304 feet) above sea level, is not in flood area The highest recorded crest on the Brazos River at Bryan (December 1913) was 16 meters (54 feet) above flood stage or 75 meters (246 feet) above sea level.

The probability of contaminating drinking water supplies is low since the Brazos River is not a source of water and there are no open reservoirs in the surrounding area. The public water supply comes from deep wells several miles from the Nuclear Science Center.

Ground water should not present any problems. The NSC is on a formation known as the Easterwood Shale. The thickness of the formation is between 3 and 90 meters (10 and 300 feet). The buildings in College Station and those on the campus have this shale as a foundation. The shallowest aquifer is the Bryan Sandstone, which underlies the Easterwood Shale. It is well below the depths required for building excavation

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional Geology

Texas lies on the North American tectonic plate, several hundred miles from the nearest edge. In addition, there are no active faults in Texas.

2.5.2 Seismicity

Texas lies in a region of minor seismic activity. Extreme west Texas, over 965 kilometers (600 miles) west of College Station, is the closest to the active belt along the west coast of Mexico and the United States. There are occasional minor shocks of very small magnitude in the state. There is only record of one earthquake of any significance in Texas; this shock was at 30.6 N and 104.2 W on August 16, 1931, near El Paso in extreme west Texas and was a Class C (6 4 in magnitude) shock.

Reinforced concrete structures provide good protection against earthquakes. The heavily reinforced NSC wall structure and reactor pool walls would withstand any minor shock that may occur.

2.5.3 Maximum Earthquake Potential

Figure 2-9 shows the relative probability for seismic activity in the United States Central Texas has the lowest hazard Earthquake potential at the NSC is unlikely.

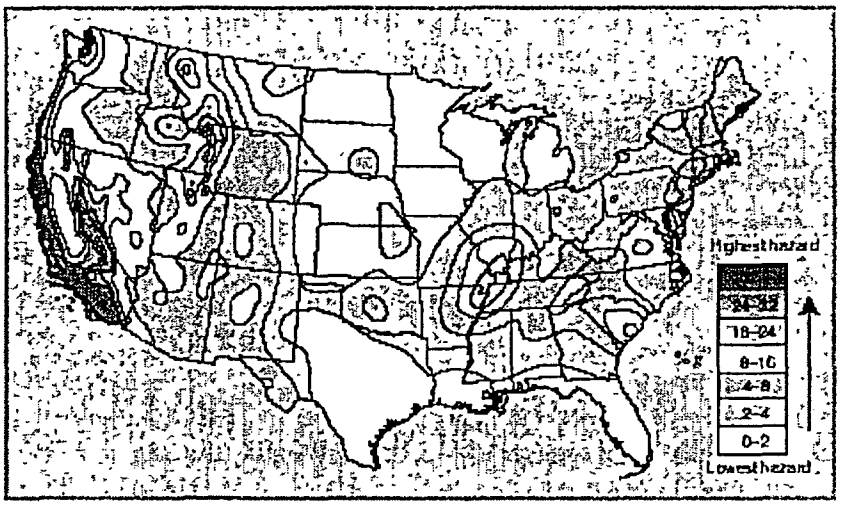


Figure 2-9: Earthquake Probability Map[2]

[2] Probabilistic Seismic Hazard Maps of Alaska by Robert L. Wesson, Arthur D. Frankel, Charles S. Mueller, and Stephen C. Harmsen, U.S. Geological Survey Open-File Report 99-36, 43 p.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The primary design criteria for the safe operation of a TRIGA reactor is that the facility is able to withstand any credible accident with negligible hazard to the public, without relying on active safety systems. TRIGA FLIP, Standard and LEU fuel with stainless steel cladding meets this criterion TRIGA Standard and FLIP fuel exhibit a prompt negative temperature coefficient responsible for reactor shutdown for all credible temperature excursions Many references not specifically listed here document the characteristics of TRIGA fuel. Chapter 13 deals in detail with the most credible loss of coolant accident. The fuel and cladding construction, rather than structure or control systems, meet the design criteria that the Texas A&M Nuclear Science Center Reactor (NSCR) can withstand credible accidents with negligible hazard to the public.

The building that houses the NSCR was completed in 1958. The criteria for the design allowed control of the airflow into and out of the NSC through ground level suctions and an exhaust stack of a specific height Section 1.8 of this report describes the history of modifications to the NSC Among these are the additions of a larger 1 MW cooling system to accommodate higher power levels and lining the reactor pool with stainless steel to reduce the loss through leakage

3.2 Meteorological Damage

The accident analysis in Chapter 13 does not use criteria for the condition of the building or equipment for accident analysis. Therefore, there are no criteria for safe operation based on possible meteorological damage. The NSC building has stood since 1959 without suffering meteorological damage. The building meets all local codes for withstanding meteorological damage

3.3 Water Damage

The accident analysis in Chapter 13 does not use criteria for the condition of the building or equipment for accident analysis and Section 2.4 addresses hydrology. There are no criteria for safe operation based on possible water damage. The NSC building is equipped with a sump pumping system. This keeps the lower level of the building dry during heavy rains In any case, if the sump system fails and the lower level becomes flooded, water damage cannot affect the safe operation of the NSCR The building meets all local codes for withstanding flooding.

3.4 Seismic Damage

The accident analysis in Chapter 13 does not use criteria for the condition of the building or equipment for accident analysis and section 2.5 addresses seismic activity for the region. Therefore, there are no criteria for safe operation based on seismology. This SAR only assumes the fuel and cladding are operable and intact for accident mitigation.

3.5 Systems and Components

The accident analysis in Chapter 13 does not use criteria for the condition of the systems, components or other equipment for accident analysis. Therefore, there are no criteria for safe operation based on systems or components. This SAR only assumes the fuel and cladding are operable and intact for accident mitigation

4 REACTOR DESCRIPTION

4.1 Summary Description

4.1.1 Introduction

The NSCR is a pool-type TRIGA reactor with Standard TRIGA, TRIGA-LEU and/or TRIGA-FLIP fuel. The NSC has used 70%-enriched FLIP fuel since the 1970s but may use 20% enriched or a combination of the three types. Pool water cools the reactor via natural convection and serves as a biological shield and moderator As of 2003, the NSC uses graphite moderators on two sides of the core and primarily uses the other two sides for sample irradiation.

The reactor support is a 7X9 grid Each location in the grid supports a fuel bundle with 4 positions for either fuel elements, control rods or other non-fueled elements. The Reactor Bridge, mounted on rails along the top of the pool, supports the frame that in turn supports the reactor grid. When the reactor is critical, the frame rests on the floor of the pool However, when the reactor is shutdown the bridge can support the reactor and move the reactor to any location along the centerline of the pool

In addition to supporting the reactor, the grid provides a guide for the fuel-followed Shim-Safety control rods When fully inserted the fueled portion extends through guide holes in the grid. A safety plate below the grid prevents the rods from falling out of the core should they become detached Guide tubes attached to the fuel bundles guide the transient rod and regulating rod

4.1.2 Summary of Reactor Data

Table 4-1: Summary of Reactor Data

Responsible Organization	Texas Engineering Experiment Station
Location	College Station, Texas A&M University
Purpose	Teaching, Research and Isotope Production
Fuel Type	TRIGA FLIP, Standard and/or TRIGA-LEU
Control	
Safety Control Rods	Four Fuel-Followed Shim-safety SCRAM rods
Regulating Rod	Non-followed Control Non-SCRAM Rod
Transient Control Rod	Void Followed SCRAM Rod
Experiment Facilities	
Beam Ports	Five permanently installed beam ports in reactor pool
Pneumatic Tubes	Pneumatic receiver various and changing core locations and periphery locations
Reactor Materials	
Fuel-Moderator	U-ZrH
^{235}U Enrichment	70% (FLIP), 20% (Standard) or 20% (LEU 20-20)

4.2 Reactor Core

General Atomic has successfully operated Mark III standard fuel elements and FLIP elements in TRIGA cores at steady-state power levels of up to 1.5 MW. The arrangement of fuel in the NSCR is such that the minimum nominal spacing between the fuel rods provides adequate convection cooling of cores up to 2.0 MW. This spacing and the extra cooling holes at the corners of the bundle enhance core cooling. Cooling of the NSCR is also improved due to the increased depth of pool

4.2.1 Reactor Fuel

The fuel bundles are in three or four-element bundles that allowed conversion to TRIGA fuel with the existing grid The four-element fuel element assemblies of the TRIGA core provide easy passage of cooling water between the elements. Water flows by natural convection through the two inch diameter hole in the grid plate adapter, passes through the large cruciform opening and then over the entire element until it leaves the core through the numerous openings in the aluminum handle at the top of the bundle In addition to the coolant passages through the grid plate adapters, the NSCR grid plate has additional coolant holes one-half inch in diameter located at the corner of each four-rod bundle

The four-element assembly (See Figure 4-1) consists of an aluminum bottom adapter, four [redacted] clad TRIGA fuel rods or non-fueled elements, and an aluminum top handle. The bottom adapter fits into the NSCR grid The four elements screw into four tapped holes on the topside of the bottom adapter. A TRIGA element threads into the bottom adapter until a flange on the element sits firmly on the adapter providing a cantilever support.

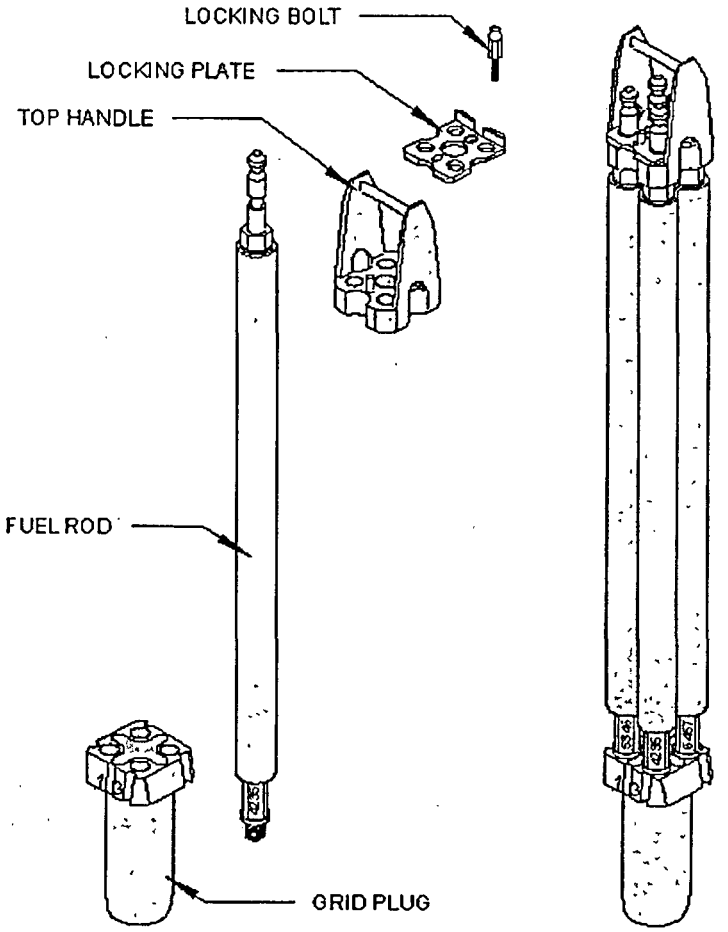


Figure 4-1: Four-Element Bundle

A three-element fuel assembly (See Figure 4-2) holds three fuel elements and a control rod, an instrumented fuel element or an experiment The NSCR utilizes two separate types of three-element fuel assemblies for housing control rods with and without guide tubes.

Figure 4-2 shows the three-element assembly that substitutes one fuel element with a fuel-followed shim-safety control rod. In this instance, control rods do not have a guide tube as the fueled follower portion of the rod remains in contact with the assembly. Since the fueled follower must pass through the fuel assembly base, it was necessary to design a base that serves as a guide for the fueled follower portion that extends through the bottom of the grid plate. Figure 4-3 depicts the position in the grid plate of a fueled follower assembly base. The top handle of the bundle serves as the upper guide for the fuel-followed control rod Figure 4-4 illustrates a similar view of the positioning of the top handle of a fueled follower assembly In the event the transient rod has a follower, it uses a specially designed control-rod guide-tube and must have a base assembly as in Figure 4-5.

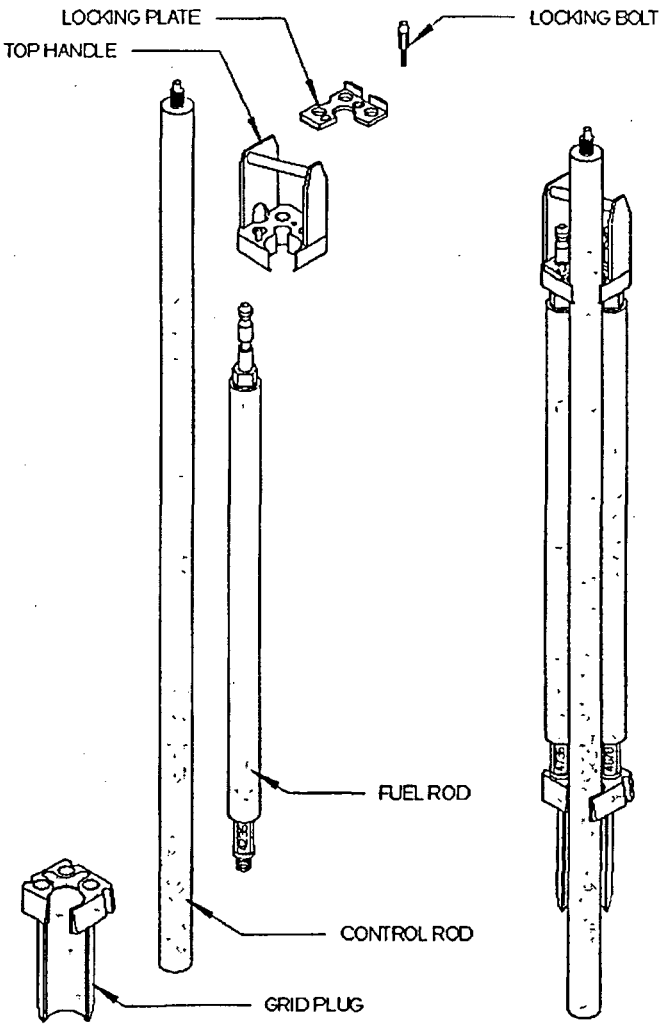


Figure 4-2 Three-Element Bundle with Fuel Followed Control-Rod

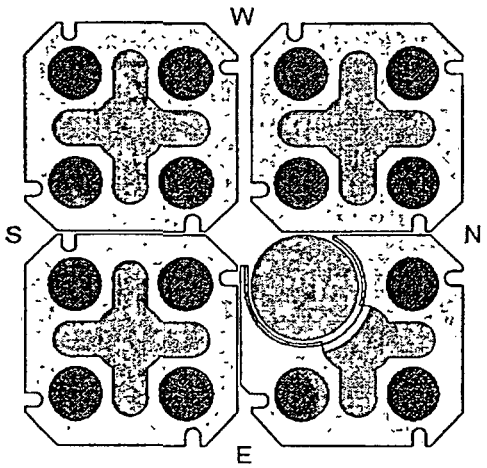


Figure 4-3: Fuel Follower Installation Bottom

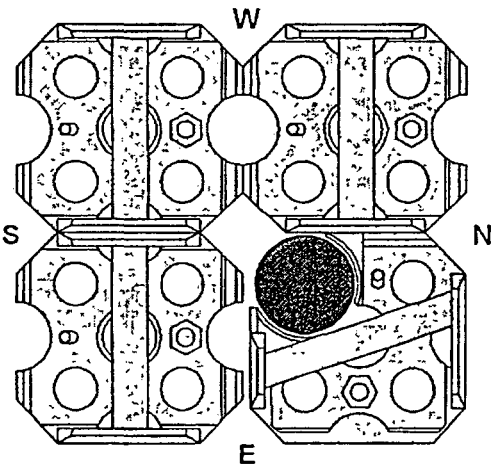


Figure 4-4: Fuel Follower Installation Top

Figure 4-5 shows the three-element assembly that substitutes one fuel element in an assembly with a guide tube that has an outside diameter of one and one-half inches. Like the bundles with fuel-followed control rods, the top handles on control rod assemblies of this type are at an angle to accommodate the guide tube The regulating rod and the transient rod without a follower utilize this type of assembly
Three-element assemblies also support instrumented fuel elements Not being an integral part of the bundle, the instrumented fuel element slides into the bundle after it is in the grid plate and fits into the bottom adapter. The instrumented fuel element fits into the bundle the same as a fuel-followed control rod

The NSCR can utilize TRIGA Standard, TRIGA FLIP and TRIGA LEU 20-20 type self-moderated elements. Zirconium hydride homogeneously combined with partially enriched uranium fuel, provides moderation Table 4-2 shows the physical characteristics of the three fuel types

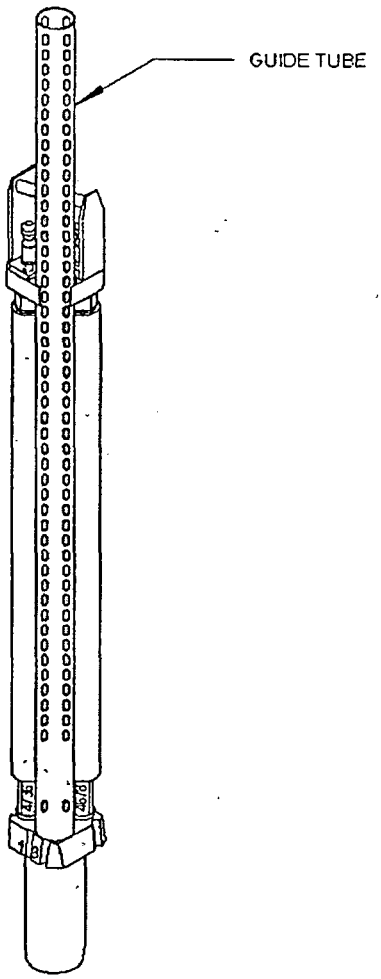


Figure 4-5: Control Rod Bundle with Guide Tube

Table 4-2: Principal Fuel Element Design Parameters

	Fuel Element Type		
	FLIP	**LEU 20-20**	**STANDARD**
Fuel-moderator material	U-ZrH	U-ZrH	U-ZrH
Uranium content	8.5 Wt-%	20 Wt-%	8.5 Wt-%
^{235}U enrichment	70%	20%	20%
Average ^{235}U content per element	[redacted]		
Burnable poison	Natural erbium (1.5 weight %)	Natural erbium (0.59 weight %)	None
Shape	Cylindrical	Cylindrical	Cylindrical
Length of fuel meat	[redacted]		
Diameter of fuel meat			
Cladding material			
Cladding thickness			

A 0 18-inch hole in the center of the active section of the Standard, FLIP and LEU 20-20 elements facilitates hydriding during fabrication A zirconium rod, inserted after hydriding, fills the hole As shown in Figure 4-6, graphite slugs, three and one-half inches in length, act as top and bottom reflectors.

Serial numbers on the bottom end fittings identify individual fuel rods A machined flat tip on the top fitting of FLIP fuel elements helps differentiate these from Standard elements

Figure 4-6: Detailed Drawing of FLIP Fuel Rod

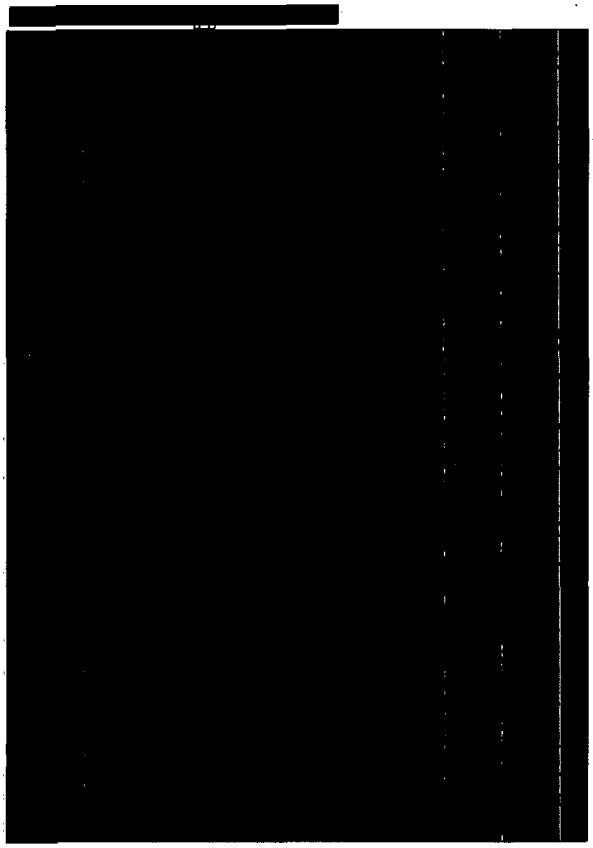


Figure 4-7: Integrated Filament Thermocouple Fuel Rod

Three thermocouples embedded in the fuel of specially fabricated instrumented elements measure fuel temperature during reactor operation. As shown in Figure 4-7, the sensing tips of the fuel rod thermocouples are located halfway from the vertical centerline at the center of the fuel section and one inch above and below the center. The thermocouple lead wires pass through a seal contained in a stainless steel tube welded to the upper end fixture. This tube projects about three inches above the upper end of the element and connects to extended tubing by SwageLock® unions to provide a watertight conduit protecting the lead wires up to the pool surface.

4.2.2 Control Rods

Six motor-driven control rods (four shim-safety rods, a regulating rod, and a transient rod) control the reactor and provide SCRAM and shutdown capability. The shim-safety and transient control rods provide scram capability. They fall into the core whenever power is lost to a valve solenoid or electromagnets. The regulating rod maintains constant power during steady state operation and does not have SCRAM capabilities. Section 7.3 details these functions.

The shim-safety control rods are fuel followed. Each consists of a fueled region and a poison region. The poison region is borated graphite with the same cladding as the fuel elements. The fueled region is [redacted] of active fuel-moderator identical to the other fuel elements. When fully inserted, the fueled portion of the control rod extends through the grid plate below the reactor core with the poison section in the core. Figure 4-8 shows the fuel-followed

control rod. The fuel-followed control rods do not have guide tubes, as a guide tube would limit cooling to the fueled section

In the absence of a guide tube. a hold-down foot fits over the top handle cross bar. This foot prevents the bundle from moving with the control rod should it bind The blade attaches to the side of the tube that houses the control rod extension When the rod drive unit is secured to the reactor support structure, there is a one-eighth inch clearance between the foot and fuel element top handle cross bar. This clearance permits small thermal expansion of the fuel without vertical restriction.

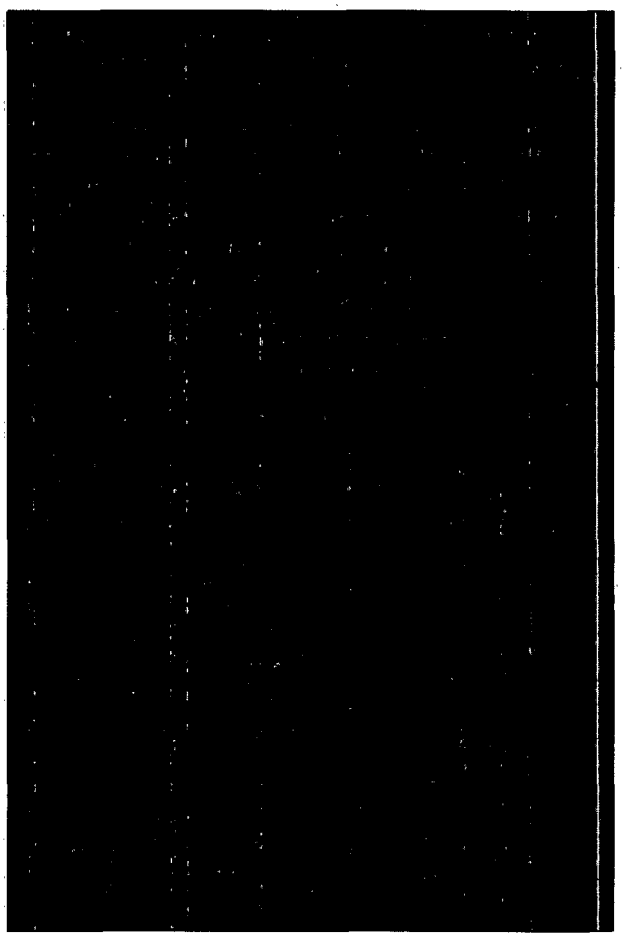


Figure 4-8: Control Rod Detailed

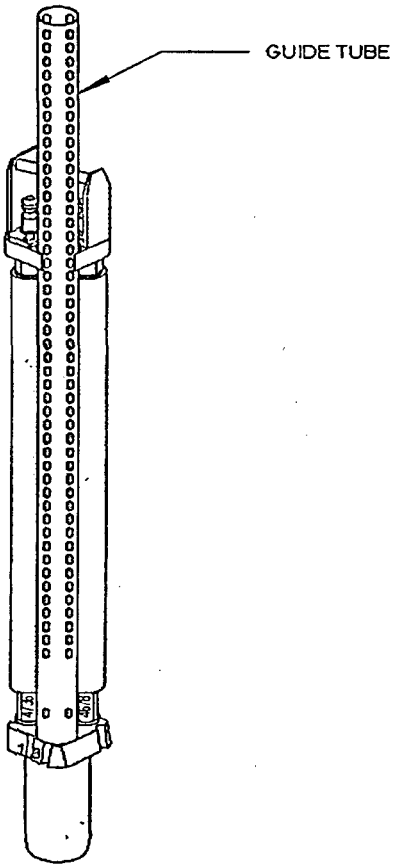


Figure 4-9: Fuel Bundle With Guide Tube

The transient control rod is void followed. It consists of a poison section and an evacuated section. The poison section is borated graphite with aluminum cladding The follower portion is also aluminum clad. Figure 4-9 shows a bundle with a guide tube that keeps the transient rod in place. The guide tube surrounds the rod and has holes for proper cooling.

The regulating rod is water followed (no physical follower section). The poison section of the regulating rod is a B_4C powder. The regulating rod uses a guide tube similar to the transient rod

4.2.3 Neutron Moderator and Reflector

Reflectors, excluding experiments and experimental facilities, are water or a combination of graphite and water. Graphite elements are machined to fit flush against a machined spacer fit into the grid.

4.2.4 Neutron Startup Source

The neutron startup source is located in the core to provide good multiplication data on the startup channel The source strength is such that the startup channel will change by greater than 2 cps upon the insertion or removal of the source from the core at initial startup

4.2.5 Core Support Structure

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

Quick disconnect valves are mounted just below the grating on the upper research level at each end of the pool to facilitate water and air connections upon movement of the reactor from the stall position to the large pool section or irradiation cell operating position. Flexible quick disconnect hoses for the diffuser system and the transient rod air allow operation at any location in the pool. Quick disconnects for the pneumatic sample transfer system allow operation of the system in a limited area near the East end of the pool.

An adjustable frame on the west side of the bridge called the bridge yoke serves as the mounting for the reactor suspension system A large crank wheel and jack mechanism mechanically raises or lowers the yoke and allows approximately a six-inch vertical adjustment of core position A seven-inch I-beam on the yoke frame insures that the reactor frame can support the weight of the transient rod mechanism

An aluminum suspension frame supports the reactor grid plate (Figure 4-10). The suspension frame is a welded structure of three-eighths by two by two inch aluminum angle. The west side of the frame is open toward the large section of the pool. This angle construction allows unrestricted flow of the cooling water. An aluminum stabilizer frame, bolted to the bottom of the grid plate, provides for vertical support. Stainless steel guides on the bottom of the stabilizer fit between tracks on the pool floor. This allows accurate repositioning of the reactor core, which is essential for numerous experiments The stabilizer also allows lowering the core until it rests on the bottom of the pool. This prevents swaying that could introduce reactivity variations.

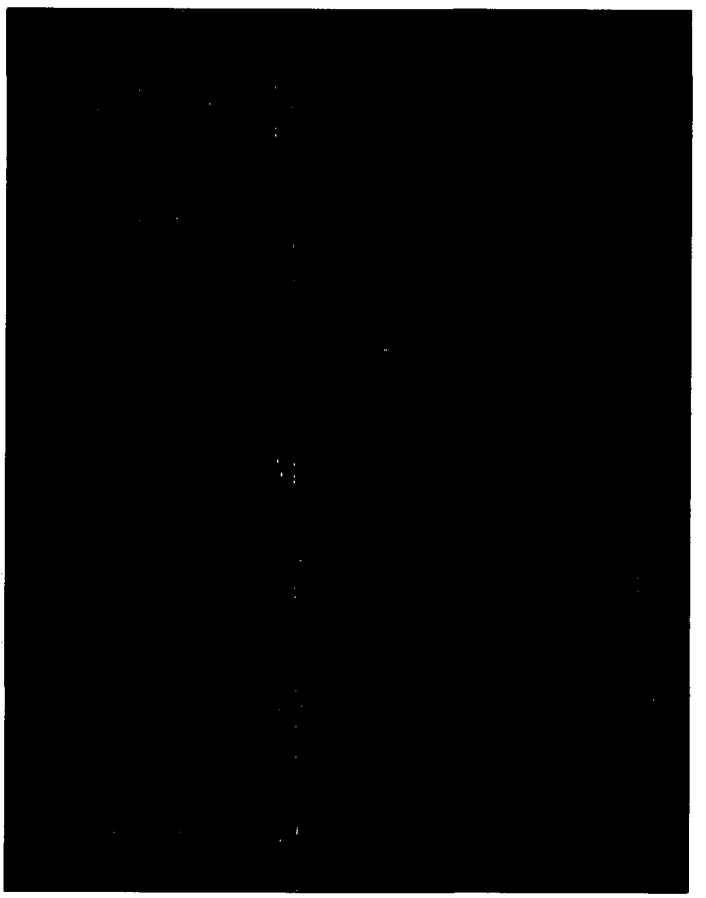


Figure 4-10: Core Grid Plate

One-quarter inch stainless steel pins attaches the aluminum frame to the grid plate on all four corners This grid plate supports the TRIGA fuel elements TRIGA elements are considerably heavier than the aluminum-plate-type fuel elements the grid plate initially supported

The grid plate contains 54 holes arranged in a nine by six array to accommodate fuel bundle assemblies, graphite, instruments, and experiment locations A reactor core loading could have several options for location in the grid plate. Figure 4-11 shows a typical core loading containing [redacted] elements and graphite reflectors. In this loading, the 'A' row of the grid plate is available for positioning experiments To accommodate a fuel followed control rod, a one and three-quarter inch diameter clearance hole through the grid plate allows passage of the fueled section of the rod (Figure 4-10) Twelve clearance holes are compatible with the four-rod TRIGA assembly design. Each hole is located at the southwest corner of the four-rod fuel assembly.

A safety plate assembly beneath the reactor grid plate stops a control rod follower two inches below its normal down position should it become detached from its mounting

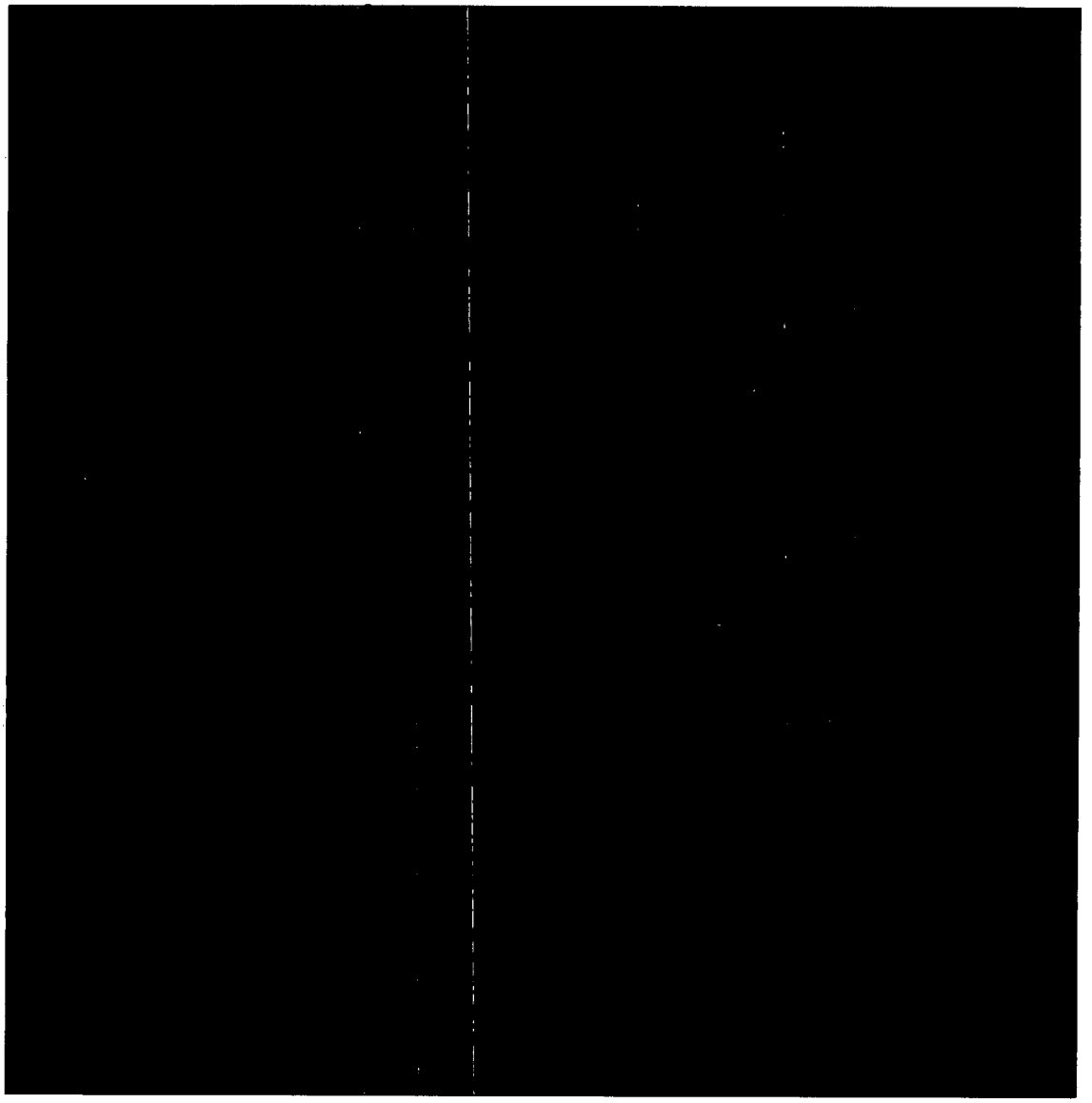


Figure 4-11: Core Configuration VIII-A

4.3 Reactor Pool

The concrete pool structure and the pool water provide shielding of the reactor The shield capacity is for a reactor operating at 5 MW, which is well above the current 1 MW TRIGA maximum operating level The movable reactor bridge permits operation of the reactor at any position on the pool centerline, which runs approximately east to west. The pool has a stall section and a main pool section (See Figure 5-3). An aluminum gate can isolate these sections to allow draining only one section The pool is thirty-three feet deep and eighteen feet wide in the main pool. The stall section is nine feet across and has an 180°-curved surface with a four and one-half foot radius

The upper seventeen feet of the pool wall is standard concrete. The lower portion of the pool wall is barites concrete and light concrete. The Irradiation Cell is a shielded structure adjoining the main pool (Figure 5-3). The Irradiation Cell may support reactor experiments or serve as pool water storage. An irradiation window, located in the shield wall, separates the reactor pool and irradiation cell The reactor can operate any desired distance from the window for irradiation of experiments in the cell.

Stainless steel (Type 304) lines the reactor pool for maximum water containment and water purity. The pool walls are 10 gauge and the floor is one-quarter inch thick. A [redacted] provides a drainage system [redacted] draining possible liner leakage into the sump of the valve pit. This leakage ultimately goes to the liquid-radioactive waste storage tanks.

Experimental penetrations consist of the thermal column, pneumatic tubes, beam ports and the irradiation cell window (Figure 5-3) The removable ends of these penetrations have bolted flanges with mechanical seals for water tightness.

The 250-gpm Transfer pump interconnects the two pool sections, the irradiation cell and the demineralizer room. The Transfer pump and associated piping are in the valve pit of the cooling equipment room. The system can transfer water to and from pool sections for storage, to the waste sump for disposal, or to the demineralizer room for purification. A pump switch is located at the pump and in the reactor control room on the water system control panel for operation of the system A single three-inch crossover line connects the demineralizer system and water transfer-storage system for flexibility of operation. Figure 5-5 shows the pool-water transfer-storage system.

4.4 Biological Shield

Concrete and water serve as a biological shield to protect personnel and visitors from the intense radiation that the reactor produces. Normally, 26 feet of water covers the NSCR core. The core normally operates in the stall area of the reactor pool. In the stall, five feet of high-density concrete provides most of the shielding for personnel in the Lower Research Level near the stall area (approximately the same level as the reactor).

When the reactor is operating in the large part of the pool, approximately eight feet of water and [redacted] high-density concrete provide shielding to personnel closest to the reactor in the Lower Research Level.

The Diffuser System pulls water from the main area of the pool and discharges it through a nozzle above the core. This forces the ^{16}N in the coolant flowing directly out of the core into the deeper part of the pool, thereby allowing most of the ^{16}N to decay before it has a chance to reach the surface. The result is lower radiation levels at the surface of the pool.

Radiation levels on the reactor bridge, which is directly above the reactor, are less than 10 mR/hr with the reactor operating at 1 MW. Radiation levels in the Lower Research Level, with the reactor operating at 1 MW in the center of the stall are less than 0.5 mR/hr. Levels are higher in the immediate vicinity of the beam ports when extracting a beam or operating the reactor adjacent to the graphite coupler box.

The reactor operator can monitor the reactor bridge at all times, thus limiting access to that radiation area. If an individual enters an area in the Lower Research Level that could be a radiation area, an alarm will alert both the reactor operator and the individual. Other devices flood the beam ports and/or SCRAM the reactor to reduce the radiation levels if personnel enter the area around a neutron beam

4.5 Nuclear Design

4.5.1 Normal Operating Conditions

Note· The Normal Operating Conditions section is included in the Reactor Core Physics Parameters section below.

4.5.2 Reactor Core Physics Parameters

4.5.2.1 Standard TRIGA Cores

This class of TRIGA reactors has well known operating characteristics and inherent safety characteristics.[1] The first NSCR Standard TRIGA core loading reached criticality in August 1968. The following were the operating characteristics from the initial operational standard TRIGA core for the NSCR:

Table 4-3: NSCR Standard TRIGA Core Characteristics

Steady-state Power Level	1 MW
Critical Mass	[redacted] grams ^{235}U
Core Mass	[redacted] grams ^{235}U
Maximum Excess Reactivity	$3.77
Prompt Negative Coefficient of Reactivity	-1 2 x 10^{-4} Δk/k-°C
Power Coefficient (1 MW)	$3.60
Maximum Pulse Energy ($2.00 insertion)	14.7 MW-sec
Total Control Rod Worth	$11.23
Maximum Pulse Reactivity Insertion	$3.00

The pulsing of standard TRIGA cores at a pulsing limit of $2.35 resulted in safe conditions since operational NSCR cores were regularly pulsed at $3.00 insertions The pulsing characteristics for a NSCR operational standard TRIGA core are in Figure 4-12. The pulsing analysis for standard TRIGA fuel is different from that of FLIP fuel due to a rather constant negative temperature coefficient (~1.2 x 10^{-4} Δk/k-°C) for standard TRIGA as compared to a variable temperature coefficient for FLIP fuel. More than 50% of the temperature coefficient for standard TRIGA cores comes from the "cell effect" or dependent disadvantage factor, and ~20% each from Doppler broadening of the ^{238}U resonances and temperature dependent leakage from the core.

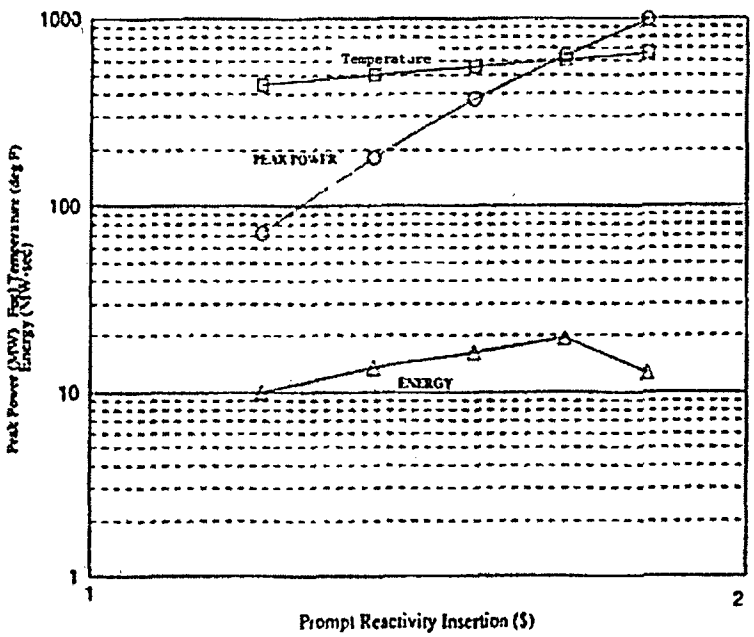


Figure 4-12: FLIP Pulsing Characteristics

Extensive measurements were made of the various parameters relating to the pulsing operation of the General Atomic Prototype reactor. The most important of these are below for step insertions of reactivity up to 2.1% δk/k ($3 00)

[1]GA-3886 (Rev. A), TRIGA Mark III Reactor Hazards Analysis, Feb. 1965.

During pulsing operation, the reactor is super-prompt-critical The asymptotic period exhibits an inverse relationship to the prompt reactivity insertion Figure 4-13 shows the results of plotting the reciprocal of the measured period versus the prompt reactivity insertion The minimum period for reactivity insertions of $3 00 ($2 00 prompt) is approximately three milliseconds Figure 4-13 also shows a plot of the reciprocal of the measured width at half maximum power versus prompt reactivity insertion.

Figure 4-13, Figure 4-14, Figure 4-15, Figure 4-16 and Figure 4-17 show the interrelationship between maximum transient power, pulse widths and period. When considered together, these plots serve to describe the general features of the TRIGA Mark III core performance in the pulsing mode. For a given core configuration, the amount of reactivity inserted determines the peak power, integral power in the prompt burst, and width of the pulse The plots show that the peak power is controllable over a wide range since this parameter is very nearly proportional to the square of δk/k minus $1.00 Pulse width and integral power, on the other hand, are approximately linear functions of reactivity insertions above prompt critical, so, their range is more limited

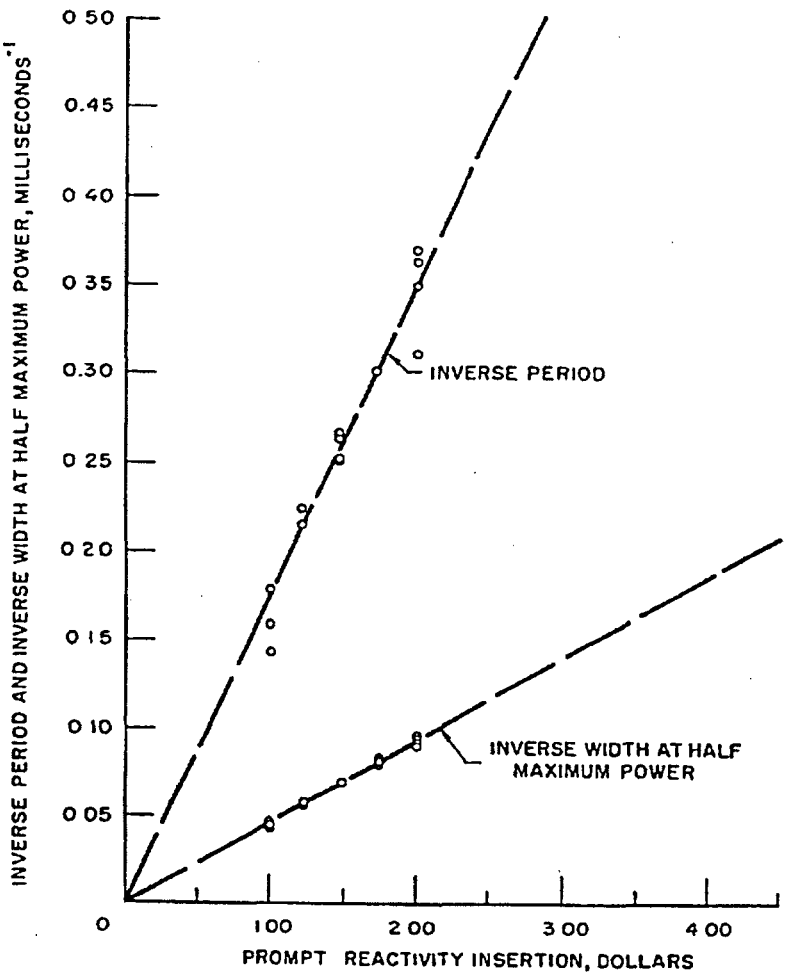


Figure 4-13: Inverse Period and Inverse Width at Half Maximum Power Versus Prompt Reactivity Insertion

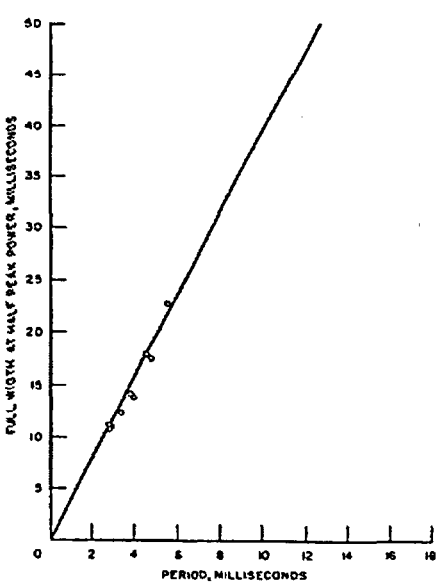


Figure 4-14: Full Width at Half Peak Power Versus Period

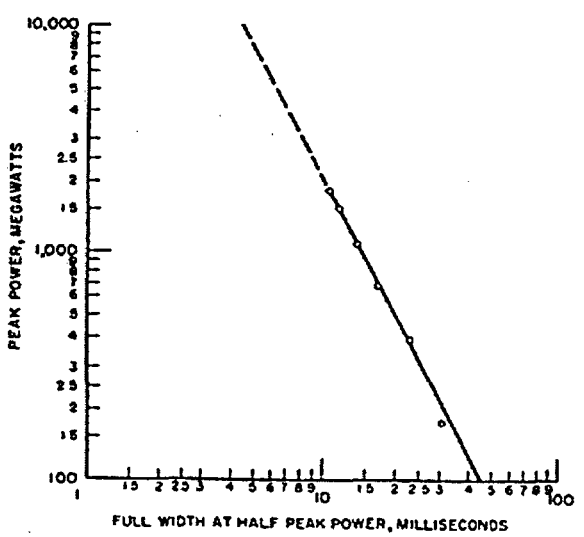


Figure 4-15: Peak Power Versus Full Width at Half Peak Power

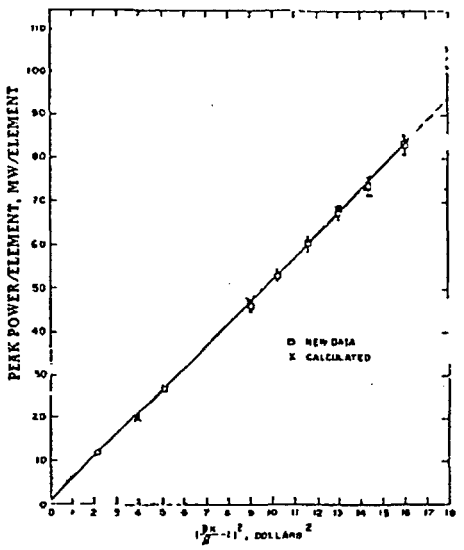


Figure 4-17: Peak Pulse Power vs. Prompt Reactivity Squared

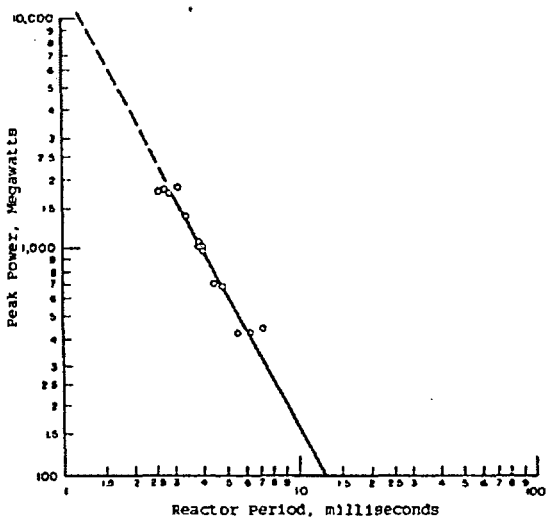


Figure 4-16: Peak Power vs. Reactor Period

4.5.2.2 Mixed Standard-FLIP and Full FLIP Cores

The NSCR has previously operated a mixed core reactor composed of both TRIGA standard and FLIP elements (Figure 4-18) Considerable information from General Atomic on the TRIGA and FLIP systems reinforced the decision to convert to a mixed core from a standard core. Studies made by the NSC for a variety of cores from all standard fuel to all FLIP fuel indicated that a core with a mixed loading would safely satisfy all operational requirements. FLIP fuel elements are located in a contiguous central region of the core. Future additions of FLIP fuel were made such that the FLIP region grew outward as the outside core dimensions remained essentially the same.

Texas A&M investigators performed an analysis using Exterminator-2, a two-dimensional multigroup diffusion theory code. The code employs the technique of variable dimensioning, therefore the limitation on problem size was actual machine storage space available. An optional total thermal flux printout and/or computer plot helped facilitated the core studies An additional program calculated the average power generation

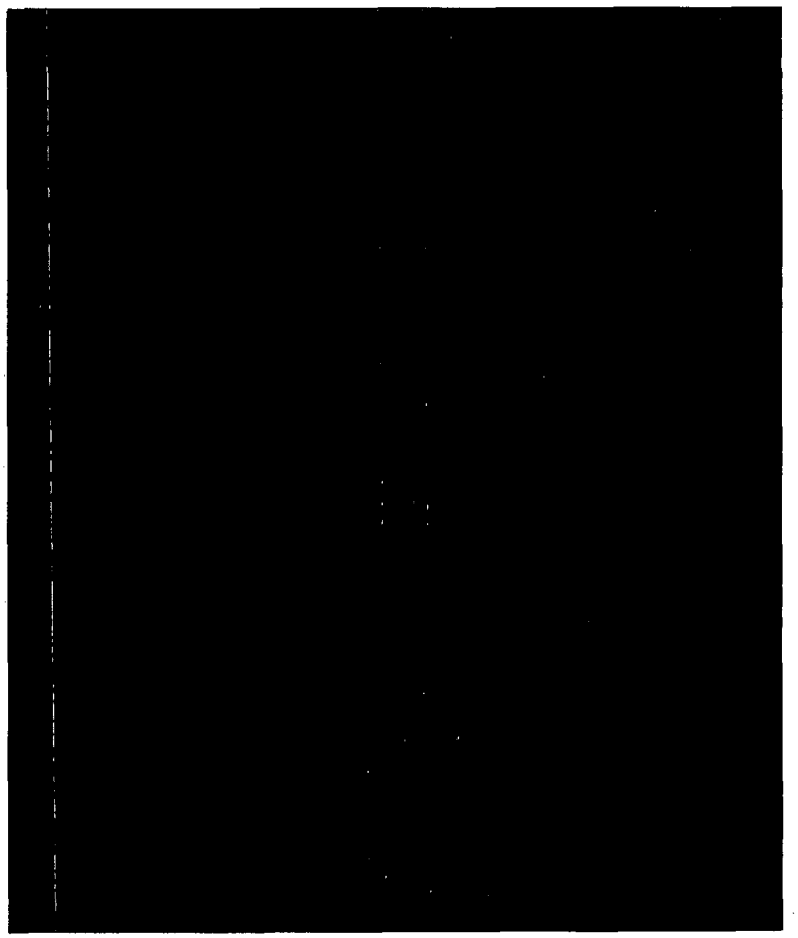


Figure 4-18: NSCR Core III-A

in each fuel element for the cores under investigation General Atomic provided basic-proven input data such as cross sections, number densities and buckling

Homogenized calculations, in which each fuel element and its associated moderator were a cell and each control rod and its associated moderator were a cell, were necessary because the spacing between fuel elements is small and diffusion theory is not valid for the heterogeneous problem Fuel cell dimensions for NSCR cores were 3 854 centimeters by 4 050 centimeters

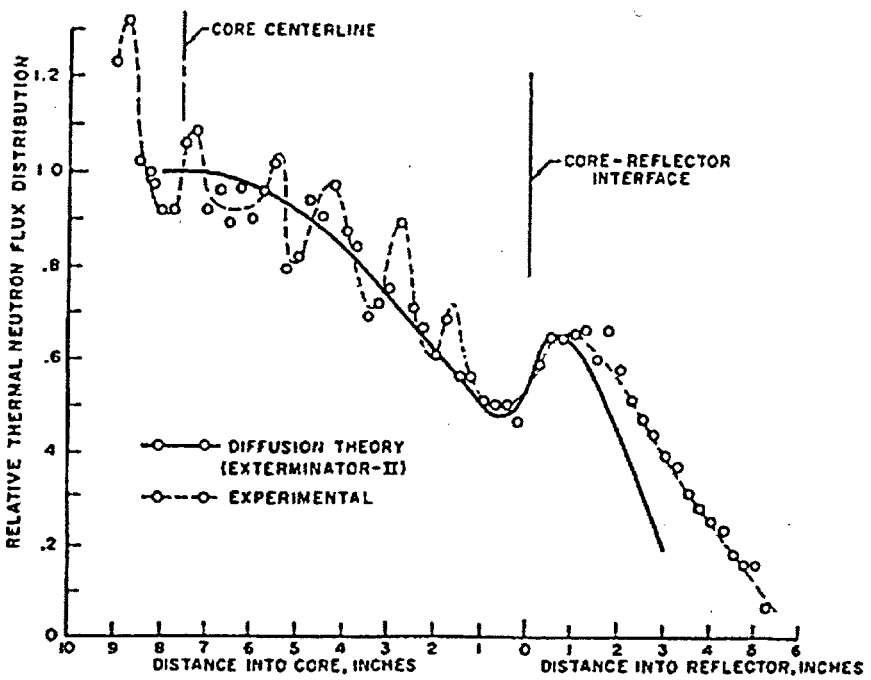


Figure 4-19: Experiment Versus Calculated Flux Using Exterminator-2

Comparisons between calculated and experimental results tested the accuracy of the input data and insured meaningful results using Exterminator-2 Three cases in particular show the validity of the code. The first test of the Exterminator-2 program was to repeat some of General Atomics' calculations for a FLIP core. Both programs gave essentially identical results The second test was a comparison of the calculated and measured values of a relative thermal flux distribution in an operational NSCR TRIGA core. The results, Figure 4-19, show good agreement considering diffusion theory calculation gives only flux values averaged over a cell. The third test was a comparison of measured and calculated values (Table 4-4) of k_{eff} for several TRIGA cores.

Table 4-4: Comparison of Measured and Calculated Values of K_{eff}

	k_{eff}	
Core	Calculated	Measured
NSCR Core IA (1) (all standard)	1.032	1.026
Puerto Rico (all FLIP)	1.068	1.059

The cores for the comparisons were both 5 x 5 array (100 cell) cores. The NSCR IA (1) core ha[redacted] standard TRIGA fuel elements, 3 shim-safety control rods, 1 transient rod and 1 regulating rod. The Puerto Rico all-FLIP core ha[redacted] FLIP fuel elements, 3 shim-safety rods, 1 transient rod, one regulating rod, and one air void experiment hole. In both cases, the calculated values were slightly higher than the measured values, and for safety consideration, the results were conservative.

These results established confidence in the Exterminator-2 code and the GA cross-sections. The conclusion is that any safety calculations obtained from this program would yield realistic and reliable results.

As mentioned earlier, Exterminator-2 computed the power generation in a given core for all fuel cells. However, the computed peak adiabatic fuel temperature during a pulse was on a selective basis following examination of calculated cell power values Weighting the average power with the flux distribution, for the cells exhibiting the highest average power, provided the distribution of power density in each The flux distribution was the product of the average flux from Exterminator-2 and the intra-cell flux from General Atomic. This method was in good

agreement with two dimensional cell calculations of higher order, which account separately for cell effects A knowledge of the temperature as a function of the volumetric heat content allowed calculating the maximum fuel temperature for a pulse of given energy.

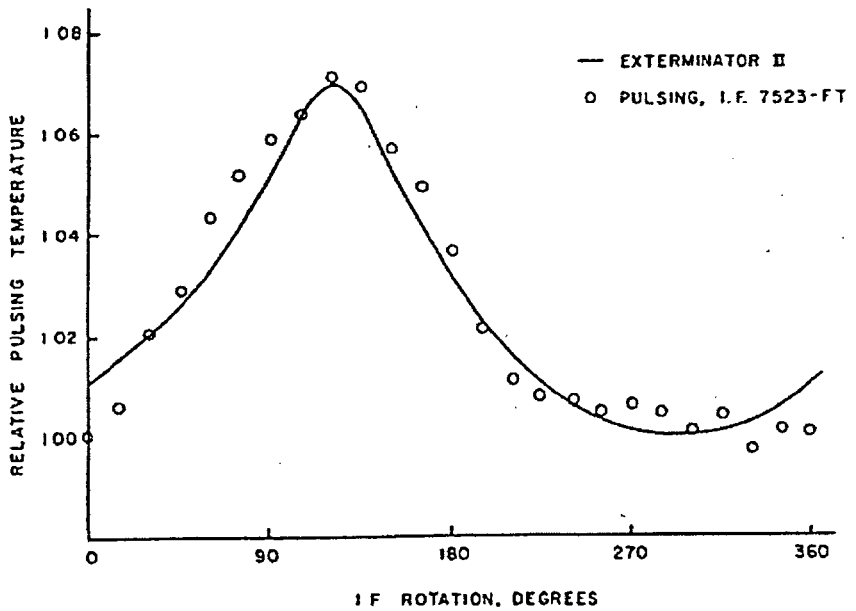
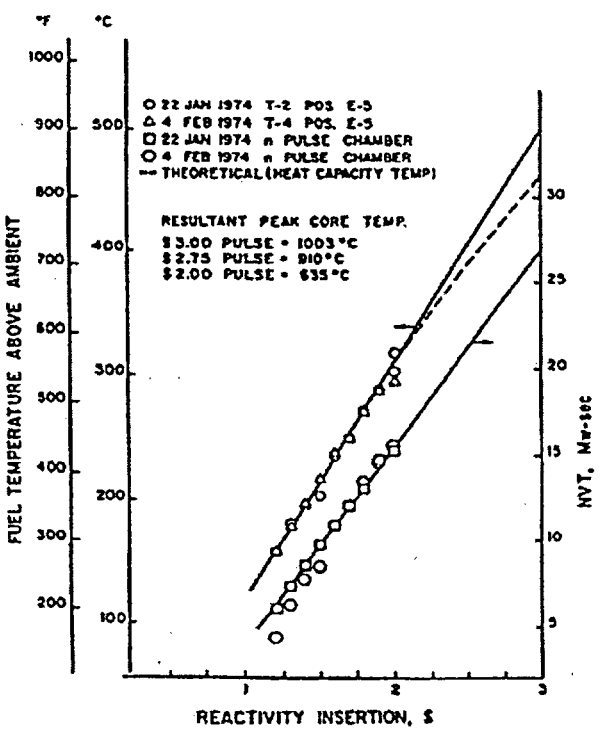


Figure 4-20: Experimental vs. calculated pulse temperatures

Figure 4-20 shows a comparison of measured and calculated pulse temperatures for singe thermocouple locations in mixed TRIGA cores. Pulse temperatures were for a $2 04 pulse insertion following rotation of the instrumented element in fifteen increments The relative values for the pulse temperatures determined by the code and the experimental values agree very well.

The next comparison was the experimental and calculated ratios for $2.00 pulse temperatures for two different instrumented element locations. The experiment, repeated four times, used two instrumented FLIP fuel elements and the same thermocouple for each set of values obtained For [redacted] FLIP element mixed core, the observed and calculated ratios agree to within 4%. The design core for the first NSCR loading was [redacted] FLIP and [redacted] standard TRIGA elements. This core went Critical on July 1, 1973, and pulsed on July 4, 1973. Figure 4-18 shows the core loading designated as III-A. Table 4-5 shows the operating characteristics for core III-A.



(SE)

Table 4-5 Operating Characteristics of NSCR Core III-A

Steady-state Power Level	1 MW
Critical Mass	[redacted] grams ^{235}U
Core Mass	[redacted] grams ^{235}U
Maximum Excess Reactivity	$6 09
Power Coefficient (1 M@)	$2.50
Maximum Pulse Energy ($2.00 insertion)	15.7 MW-sec
Total Control Rod Worth	$15.57
Maximum Pulse Reactivity Insertion	$2.00

Reactor core studies using Exterminator-2 predicted a core excess of $6.20 for Core III-A as compared to the measured value of $6 09. Pulsing characteristics observed for III-A are in Figure 4-21.

The design considerations for pulsing FLIP fuel as compared to standard TRIGA fuel are different due to a temperature-dependent negative temperature coefficient for FLIP fuel (refer to Figure 4-23). For a TRIGA FLIP fuel element, the uranium loading is about three and one-half times that of a standard TRIGA element, which makes the neutron mean free path FLIP element much shorter. For this reason, heating the fuel-moderator does not greatly reduce the

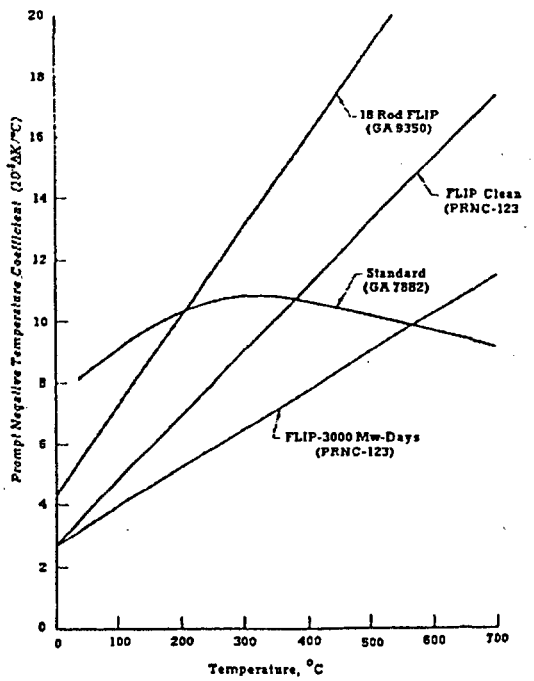


Figure 4-22: Temperature Coefficients of TRIGA Fuels

resonance escape probability for neutrons (The longer slowing down length of Standard fuel increases the probability that a neutron will be absorbed will slowing down in a U-238 resonance) In the TRIGA FLIP fuel, the temperature-hardened spectrum decreases reactivity through its interaction with a low energy resonance material. Thus erbium, with its double resonance at ~0 5 eV, serves both as a burnable poison and enhances the prompt negative temperature coefficient in the TRIGA FLIP fuel. The neutron spectrum shift pushes more of the thermal neutrons into the ^{167}Er resonance as the fuel temperature increases In TRIGA fuel, the temperature coefficient is prompt because there is a lager portion of moderator in the fuel-moderator material (This effect is greater for Standard than for FLIP) The fuel and moderator temperatures rise simultaneously producing the temperature dependent spectrum shift.

In a TRIGA FLIP core, the results of cell structure on the temperature coefficient are small. Almost the entire coefficient comes from temperature dependent changes in η_f within the core and ~80% of this effect is independent of the cell structure. The calculated temperature coefficients are in Figure 4-22 for standard, mixed and FLIP cores. The temperature dependence of the temperature coefficient of a TRIGA FLIP core is advantageous in that a minimum reactivity loss is incurred in reaching normal operating temperatures, but any sizable increases in the average core temperature result in a sizably increased prompt negative temperature coefficient to act as a shutdown mechanism. The burnup calculations indicate that after 3000 MW days of operation, the ^{235}U concentration averaged over the core is ~67% and the ^{167}Er concentration is ~33% of the beginning-of-life values. The end-of-life coefficient for a FLIP core is less temperature dependent than the beginning-of-life coefficient since the sizable loss of ^{167}Er and the resulting increased transparency of the ~0.5 eV resonance region to thermal neutrons.

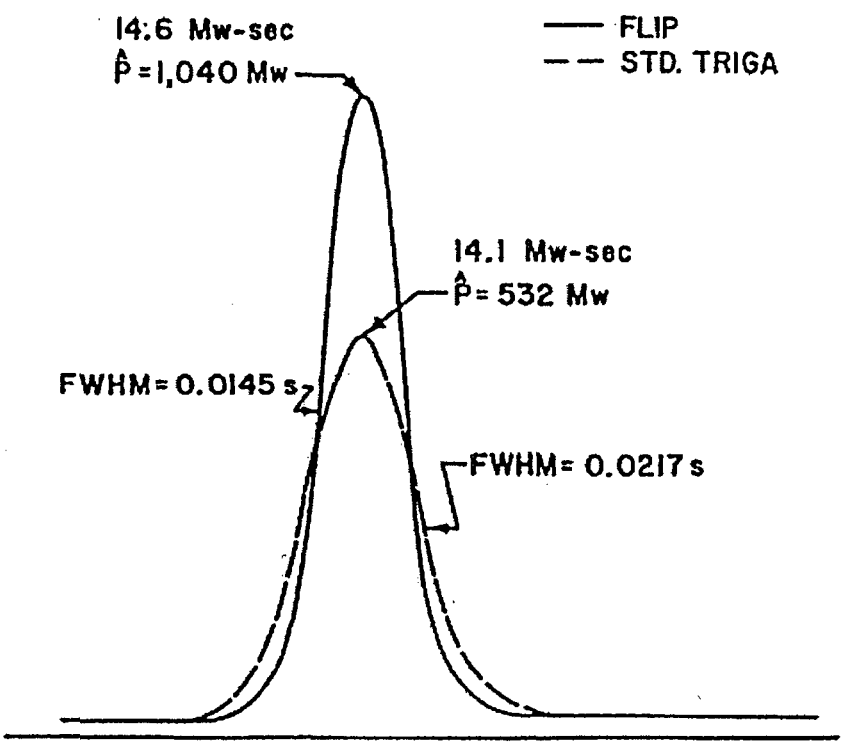


Figure 4-23: Comparison Of FLIP To Std. TRIGA Pulse For Similar Reactivity Insertions

Figure 4-23 demonstrates the effect of the temperature coefficient for standard and FLIP cores upon the pulse shape. Note that the FLIP pulse peak power is considerably higher than that in standard fuel, thus higher fuel temperatures for the same reactivity insertions. The time span for full width at half maximum (FWHM) for a FLIP pulse is considerably less than that for the standard core pulse, yet the total pulse energy is approximately the same.

4.5.2.3 FLIP and LEU Cores

Standard and FLIP TRIGA cores have well known inherent safety characteristics[2]. The pulsing of TRIGA-FLIP cores with $2 00 reactivity insertions results in very safe conditions. Earlier operational NSCR cores were regularly pulsed with $3 00 insertions The pulsing characteristics for a NSCR operational TRIGA-FLIP core are in Figure 4-12. The pulsing analysis of LEU cores indicates performance similar to that of FLIP cores.

In TRIGA FLIP and LEU fuel, the temperature hardened thermal neutron spectrum adds negative reactivity through its interaction with erbium For TRIGA LEU fuel, the ^{235}U loading is less than that of TRIGA FLIP fuel This results in an increased concentration of ^{238}U. This allows the higher ^{238}U loading to generate a similar prompt negative coefficient of reactivity with a lower erbium concentration Another important design feature of TRIGA fuel is the ZrH moderator present in the fuel structure. The ZrH moderates neutrons in the fuel rod. As the fuel temperature increases, the density of the material decreases, resulting in a negative feedback effect

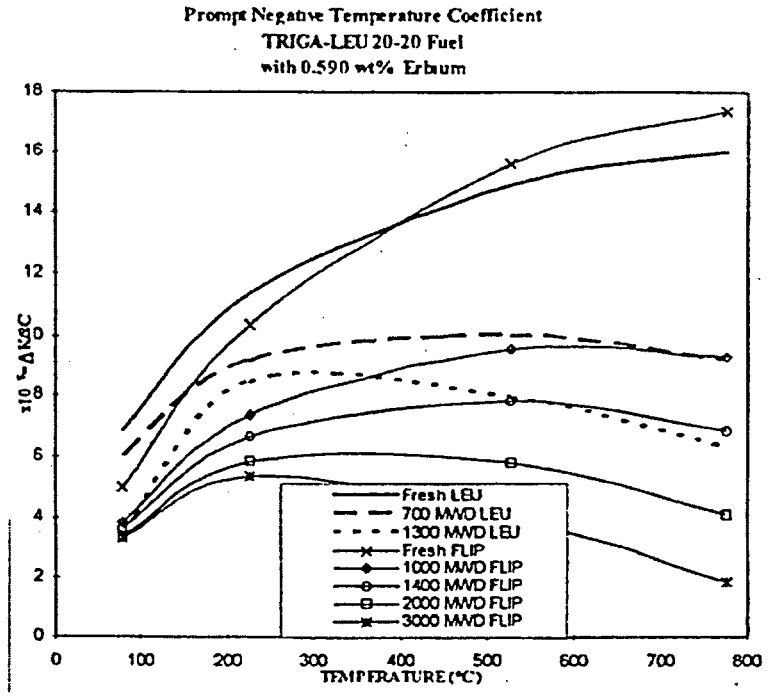


Figure 4-24: Prompt Negative Temperature Coefficient TRIGA–LEU 20-20 Fuel

Figure 4-24 shows the calculated temperature coefficients in for both FLIP and LEU cores. The temperature dependent character of the temperature coefficient of a TRIGA-FLIP or LEU core is advantageous because reaching normal operating temperatures provides an acceptable negative reactivity addition Further increases in the average core temperature result in a sizably increased prompt negative temperature coefficient to act as a shutdown mechanism. The burnup calculations indicate that after 3000 MWD of operation for FLIP fuel, the ^{235}U concentration averaged over the core is ~67% and the ^{167}Er concentration is ~33% of the beginning-of-life values. For LEU fuel at 1300 MWD, the ^{235}U concentration is ~47% and the ^{167}Er concentration is ~11% of the beginning-of-life values. The end-of-life coefficient for both fuel types is less temperature dependent than the beginning-of-life coefficient because of the sizable loss of ^{167}Er and the resulting increased transparency of the 0 5 eV resonance to thermal neutrons.

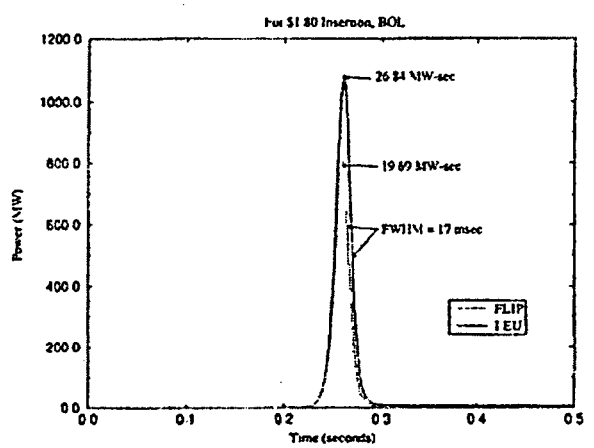


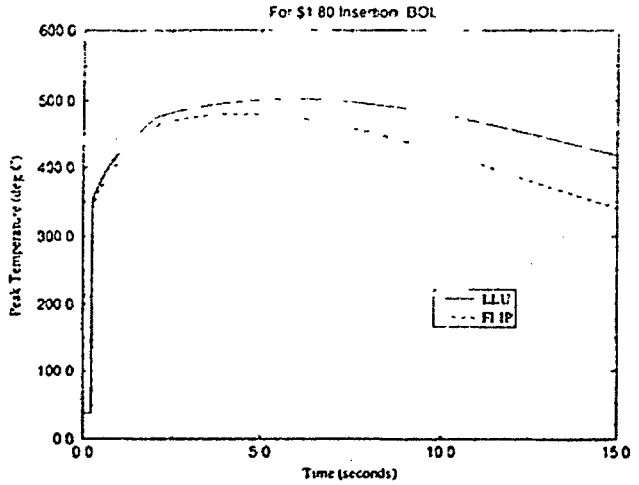
Figure 4-25: Comparison of FLIP And LEU Pulses (BOL)

Figure 4-25 demonstrates the effects of the temperature coefficient for FLIP and LEU cores upon the pulse shape at beginning of life. Note that the LEU pulse peaks at a power higher than does the FLIP pulse; thus, higher fuel temperatures will result during the LEU pulse as Figure 4-26 shows. This relationship between the two fuel types will change as the core life progresses By the end of core life, the FLIP pulse will generate higher temperatures than the LEU pulse. Figure 4-27 and Figure 4-28 show this behavior.

---

[2] GA - 3886 (Rev A) TRIGA Mark III Reactor Hazards Analysis. Feb. 1965

During pulsing operation, the reactor is in a super-prompt-critical condition The asymptotic period is inversely related to the prompt reactivity insertion. Figure 4-13 shows the results of plotting the reciprocal of the measured period versus the prompt reactivity insertion The minimum period obtained for reactivity insertions of $3.00 ($2.00 prompt) is 3 msec Figure 4-13 Also shows a plot of the reciprocal of the measured width at half maximum power versus prompt reactivity insertion See Figure 4-14, Figure 4-15, Figure 4-16 and the accompanying text for a discussion of pulse response.



The NSCR may operate a core composed of TRIGA LEU fuel elements (Table 4-2). The LEU fuel will have essentially the same properties as the TRIGA FLIP fuel.

The investigation of the design of a core composed of this type of fuel required the use of a computer code capable of producing all the necessary design parameters during both steady state operation and during accident scenarios Accuracy of computations was of the utmost importance. Performing benchmark calculations for FLIP cores and comparing results for the LEU core to those for the FLIP core assured Accuracy.

The code selected for the generation of few group neutron cross sections representative of various sub-regions in the core and its surroundings is WIMSd4/m, a one-dimensional neutron transport code capable of solving for flux distributions using many fine energy groups. WIMSd4/m created spatially and energetically averaged sets of cross sections in the seven standard energy groups used in TRIGA analyses for each core sub-region. Since neutron cross sections are highly temperature dependent, WIMSd4/m calculated many cases covering the entire operational temperature range for both normal and accident conditions for each type of core sub-region. A sub-region can consist of a fuel pin and its surrounding water, a water hole, a control rod and its surrounding water, a section of the graphite reflector or any other material present in the core or surrounding experimental irradiation areas of interest.

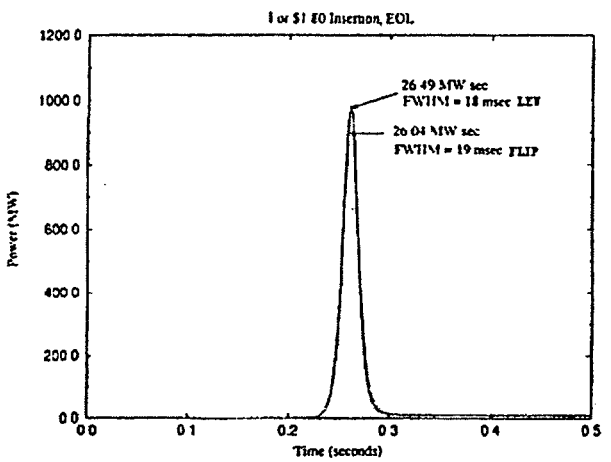
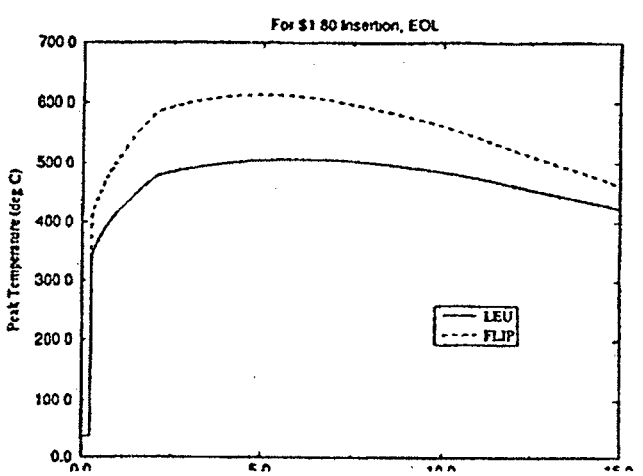


Figure 4-27: Comparison of FLIP And LEU Pulses For $1.80 Insertion (EOL)

The code DIF3D, using three dimensional neutron diffusion calculations, modeled the core, the reflector and the irradiation facilities. DIF3D used the generated temperature and burnup dependent library of homogenized cross sections.

This code is capable of computing the k_{eff} for different fuel and control rod configurations, as well as the spatially dependent power and neutron fluxes for each energy group. The code allows the investigation of different control rod positions and fuel loading arrangements and generate the prompt temperature coefficients of reactivity necessary for the transient calculations on a full core basis.



To prove the accuracy of these codes for modeling the NSCR, investigators at Texas A&M University recreated a test case run by General Atomics on a two-megawatt TRIGA core[3] This test case models a core containing fuel similar to the TRIGA-LEU fuel proposed for use at the NSC and thus served as a benchmark calculation. The fuel in both the NSCR and in the test case has essentially the same properties for each element The major difference is the addition of a shroud around each four-rod cluster in the GA test case. In addition, the GA test case core has a modeled power level of 2 MW while the NSCR core will operate at 1 MW. This does not adversely affect the validity of the test case Table 4-6 shows a summary of the core parameters for this test case

Figure 4-28: Pulsing Temperatures for $1.80 Insertion (EOL)

Table 4-6: Core Parameters of GA Test Case

Fuel Cluster:	TRIGA-LEU 20 wt% U in U-ZrH	
	Fuel rods per cluster:	
	Standard Cluster:	4
	Control Cluster:	3
	Nominal Fuel Rod Dimension	
	Fuel O.D ·	[redacted] mm
	Clad O.D :	[redacted] mm
	Fuel Height:	[redacted] mm
Fuel Loading:	[redacted] U (20% enriched)/rod	
	[redacted] U (20% enriched)/cluster	
	[redacted] U-235/std cluster	
	~0 59 wt% Erbium as burnable absorber	
Number of fuel clusters in core:	[redacted]	
	Standard Clusters	[redacted]
	Control Clusters:	[redacted]
Reflector:	Water	
Core size:	[redacted] liters	
U-235 content/core:	[redacted]	
Core Geometry:	[redacted] arrangement	
Grid Plate:	[redacted] positions (normal conversion)	
Burnup Status of the core	Equilibrium core	
Average core burnup:	~20%	
Thermal-Hydraulic data:		
	Average Power Density:	26 Kw/liter
	Coolant Flow Rate:	1000 gpm
	Core inlet temperature:	38°C

[3] General Atomics, "Generic Enrichment Reduction Calculations for Rod-Type Reactors" Research Reactor Core Conversion From the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels Guidebook. International Atomic Energy Agency, Vienna. 1980

Table 4-7 lists The peaking factors generated and reported by GA for the test and the corresponding values from the NSC calculations using WIMSD4/m and DIF3D

Table 4-7: Peaking Factors From GA Test Case

Tape of Peaking	$\hat{P}/\bar{P}$ General Atomics	$\hat{P}/\bar{P}$ TAMU 0 MWD	$\hat{P}/\bar{P}$ TAMU 900 MWD
Core Radial	1 57	1.61	1.62
Core Axial	1.36	1.30	1.31
1D Cell (23°C)	1.48	1.41	1.29
1D Cell (310°C)	1 52	1.44	1 31
1D Cell (700°C)	1.61	1.49	1.34

The core radial and core axial values agree quite well, but there is a small discrepancy in the 1D cell values. The GA test case is at a burnup of approximately 20%, which corresponded to approximately 900 MWD. TAMU values would be in better agreement with the no burnup on the core.

Table 4-8 compares of the peak thermal flux values in the core and the water reflector. The sum of the values of groups 5, 6 and 7 in the 7-group model provided the thermal flux values. These flux values are in close agreement.

Table 4-8: Estimated Peak Thermal Flux at 2 MW 4-Rod Cluster TRIGA-LEU Fuel

	General Atomics	TAMU
Core	**1.5×10^{13}**	**1.5×10^{13}**
Reflector (water)	**2×10^{13}**	**1.9×10^{13}**

The final parameter that GA generated in their report was the prompt negative temperature coefficient. Figure 4-29 shows the Texas A&M values along with the GA values for this parameter.

These are the only parameters verified against General Atomics data for the benchmark case. However, they show that the TAMU models are capable of producing accurate results as compared to the GA approved models DIF3D is also capable of producing output in much more detail than this document shows. With the aid of an external plotting program (TECPLOT), investigators generated three dimensional flux maps that might prove useful in future operations at the NSCR

A steady state thermal hydraulic analysis, using the DIF3D power distribution and NCTRIGA, determined the maximum fuel and clad temperature during operation. NCTRIGA is a one-dimensional thermal hydraulic code that calculates temperatures at several nodes along a single fuel rod channel and determines the natural convection induced coolant flow rate. NCTRIGA uses the power distribution

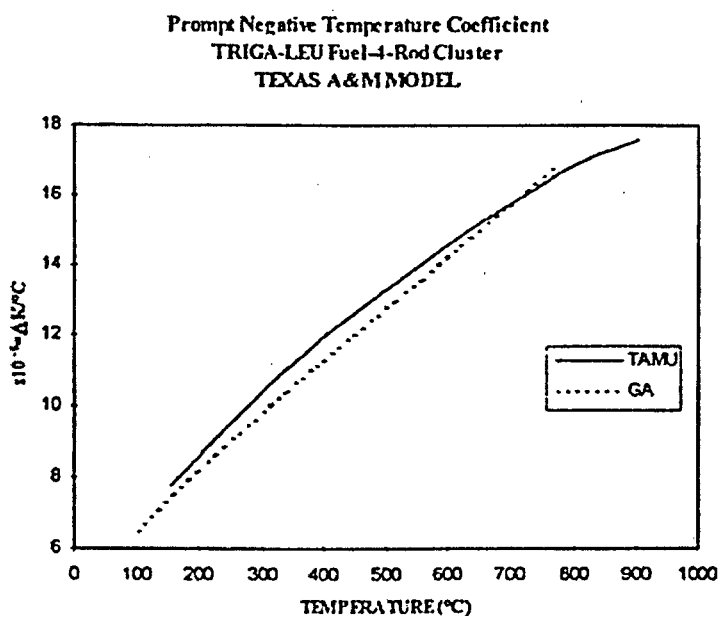


Figure 4-29: Prompt negative temperature coefficient TRIGA LEU fuel 4-rod cluster

generated from the neutronic analysis and information on the geometry and material composition of the flow channel to produce temperatures in the fuel and to predict the coolant flow rate.

Investigators used the code PARET from Argonne National Laboratory to perform calculations to determine peak fuel and clad temperatures during normal pulsing and accident transients for the both the FLIP and LEU cores. This code is capable of performing 1-D radial heat transfer calculations under non-steady state conditions at several axial nodes along a fuel rod. They used NCTRIGA to generate the initial temperatures and flow rates needed for input to PARET and DIM (with temperature dependent cross sections taken from WIMSd4/m) to generate the temperature coefficients of reactivity to input to PARET. Figure 4-24 shows these for FLIP and LEU fuels at various values of burnup. To acquire initial power distribution data for pulsing, investigators ran a 'DIM job' at 300 watts steady state power with the transient rod fully inserted. Compiling this data into a PARET deck produced a model for predicting performance of the peak fuel element and core during pulsing

Comparisons of peak pulsing power data from experiments on the Texas A&M Nuclear Science Center Reactor and data from the model provide a benchmark for the method. Figure 4-30, Figure 4-32, Figure 4-31 and Figure 4-33 show the relationship between predicted and measured pulse values and that the predicted values are reliable.

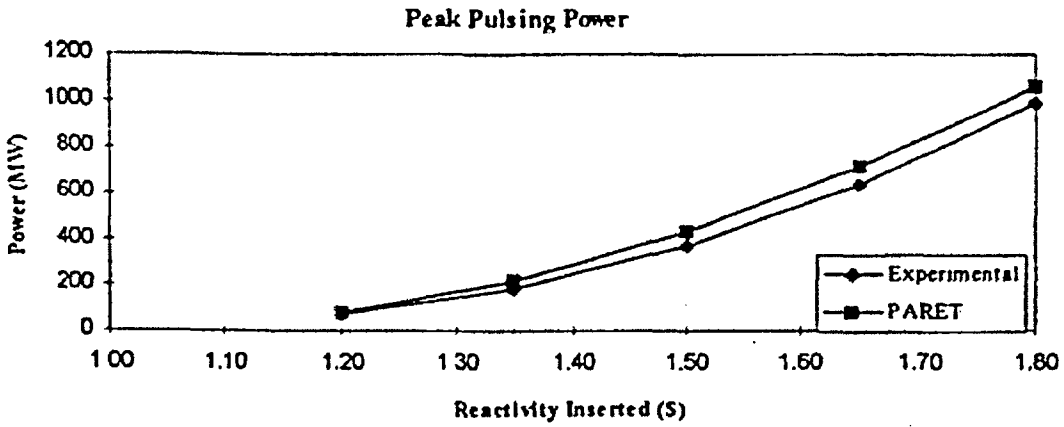


Figure 4-30: Peak Pulsing Power

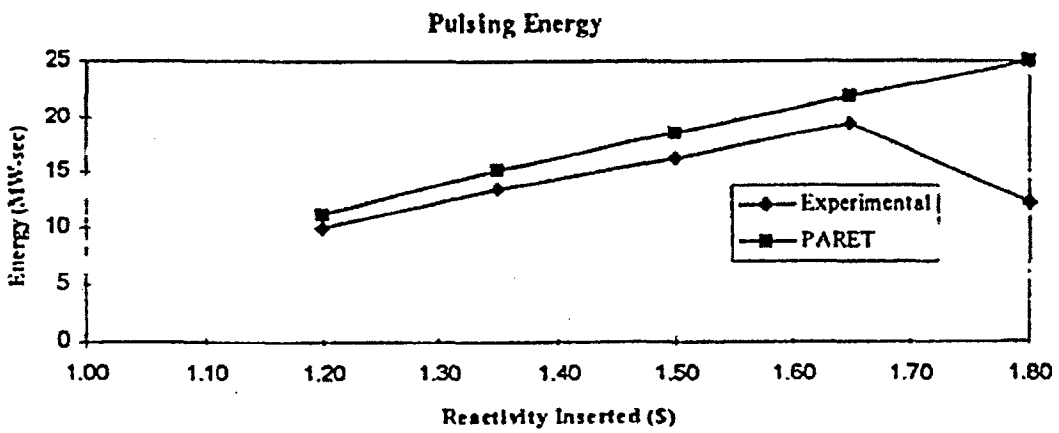


Figure 4-31: Pulsing Energy

The pulse energy mismatch at $1.80 (Figure 4-31) is most likely because the experimental value is too small caused by the detector overloading due to the excessive amount of energy generated in this pulse.

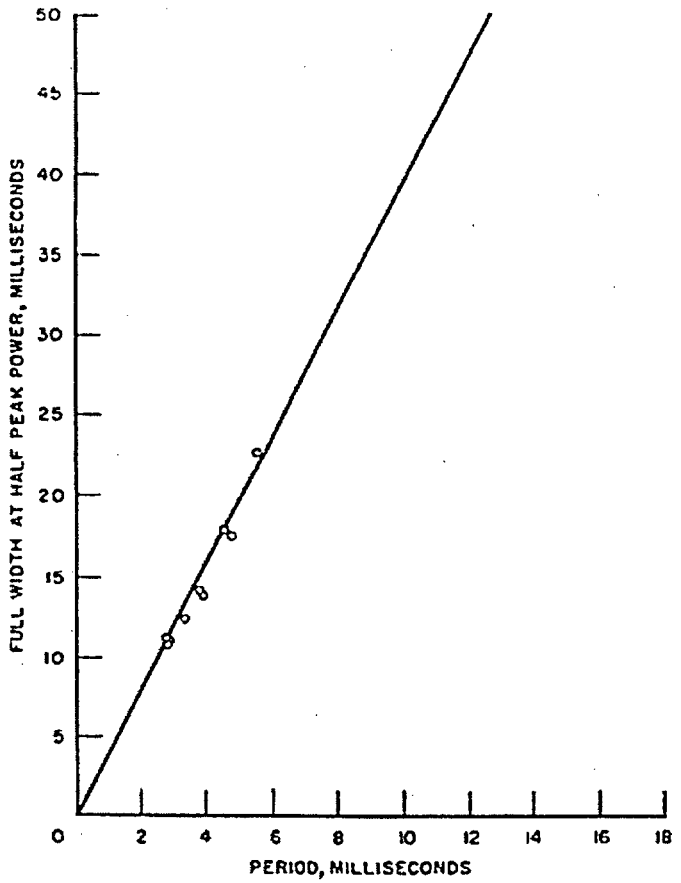


Figure 4-32: Full Width at Half Peak Power

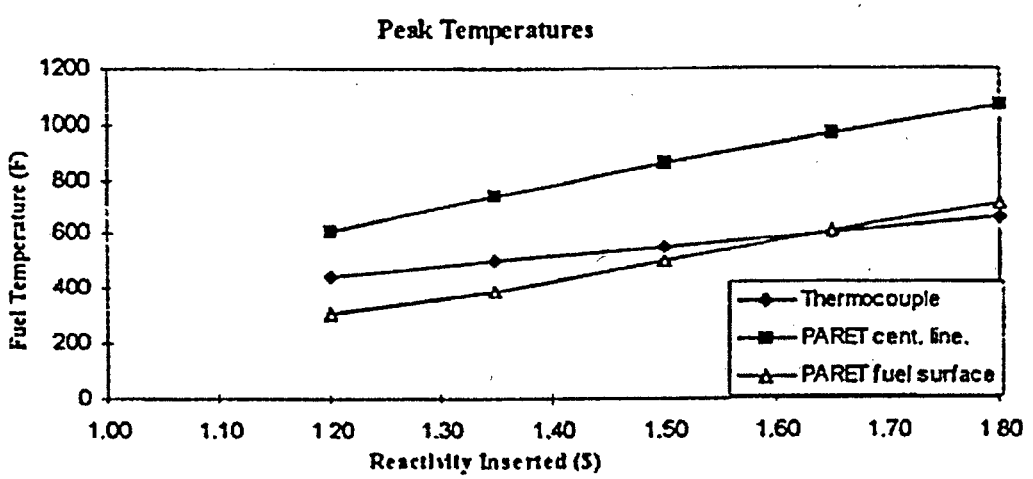


Figure 4-33: Peak Temperatures

This peak fuel temperature is difficult to exactly quantify due to the nature of the experimental data. The experimental temperature is from a thermocouple mid-way between the fuel surface and the centerline of the instrumented element (Figure 4-7). Since the measured temperature falls between the predicted centerline and fuel surface temperatures, the results are reasonable

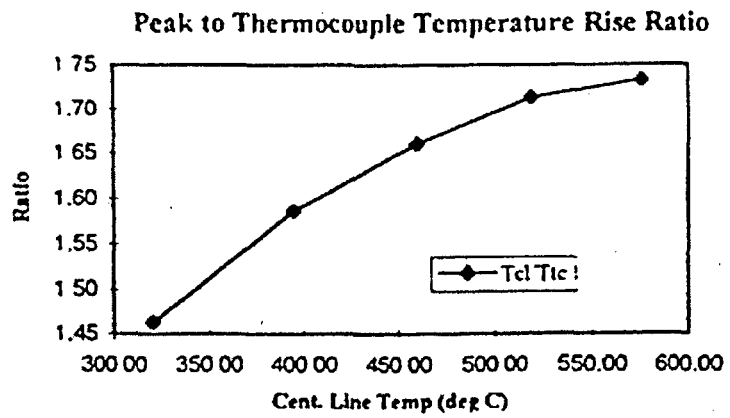


Figure 4-34: Peak to thermocouple Temperature rise ratio

Investigators used a curve fit to develop a functional relationship for the ratio of the peak fuel temperature to peak thermocouple temperature as a function of peak centerline temperature This related the thermocouple temperature to the predicted peak temperature in the element Figure 4-34 shows a plot of this for the data.

A second order polynomial fit to this data results in:

$$R(T_{cC}) = 0.5066 + 4.046\times10^{-3}T_{CL} - 3.32667\times10^{-6}T_{CL}^{2}$$

Where:
$R(T_{cL})$ is the ratio of the peak centerline temperature to the peak thermocouple temperature
T_{CL}, is PARET calculated peak centerline temperature of the instrumented element.

Since LEU and FLIP fuel heat transfer properties are similar, this ratio should hold for both fuel types

4.5.2.4 2. LEU Cores

Designing a core for the NSCR from the methods described above, a LEU core for the NSCR has the properties in Table 4-9.

Table 4-9: NSCR LEU Core Properties

Steady State Power Level:	1 MW
Number of Fuel Elements:	[redacted]
Critical Mass:	[redacted] grams ^{235}U
Core Mass:	[redacted] grams ^{235}U
Maximum Excess Reactivity:	$3.43
Total Control Rod Worth:	$16 91
SS1:	$2.78
SS2:	$1.85
SS3:	$2 99
SS4:	$5.10
RR:	$1.39
TR:	$2.80
Minimum Shutdown Margin:	$0 91

The core map is the same as in Figure 4-11 All of the design parameters meet the requirements of the Technical Specifications for the NSCR. The procedure indicated in the Technical Specifications provides the method to calculate the shutdown margin The designed core exceeds the required limit of $0.25.

4.5.3 Operating Limits

The Technical Specifications (Section 14 of this SAR) specify the operating limits for the NSCR

4.6 Thermal-Hydraulic Design

The NSCR operates at 1 MW steady state with natural convection cooling The NSCR can operate anywhere along the pool centerline except in the gateway between the stall and large pool section. Pool water constantly surrounds the reactor core. This water flows freely from the bottom and sides of the core during the convection cooling process.

Figure 4-1 shows that the four-rod fuel element assembly provides easy passage of cooling water through the assembly. Water flows by natural convection through the 2" diameter hole in the grid plate adapter. It passes through the large cruciform opening and then over the entire element until it leaves the core through the numerous openings in the aluminum handle In addition to the coolant passages through the grid plate adapters, the NSCR grid plate has additional coolant holes 1/2" in diameter located at the corner of each four-rod element.

General Atomics has successfully operated Mark III standard fuel elements and FLIP elements operated in TRIGA cores at steady state power levels of up to 1.5 Megawatts. The arrangement of fuel in the NSCR is such that the minimum nominal spacing between the fuel rods provides adequate convection cooling of cores up to 2.0 MW. Figure 4-35 shows the nominal spacing of the fuel rods in the NSCR core. This increased spacing and the extra cooling holes at the corners of the bundle considerably enhance Core cooling. .

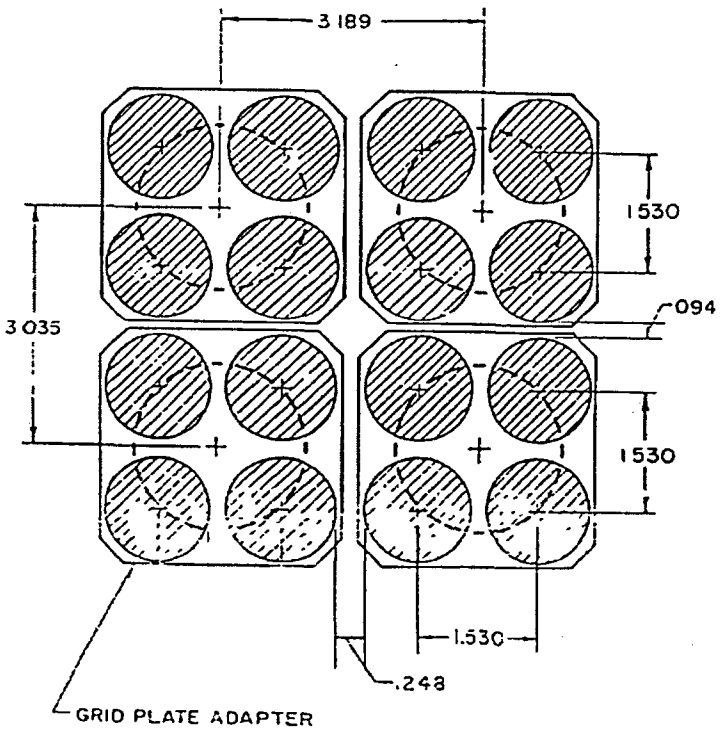


Figure 4-35: Nominal Fuel Rod Spacing in the NSCR Core

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The various pool water systems accomplish heat removal, purification, recirculation, transfer, storage, make-up water addition, pool surface cleaning and liquid waste disposal. System components and piping handling pool water are stainless steel, aluminum, and plastic to maintain maximum pool purity. Welded piping systems with mechanical seals insure minimum leakage.

A heat exchanger system with pool water on the primary side and cooling tower water on the secondary side cools the reactor pool The entire cooling system consists of the pool, heat exchanger, cooling tower, primary and secondary pumps and associated piping

The maximum operating water pressure occurs in the heat exchanger tubes. The maximum pressure for other pool water systems corresponds to the reactor pool depth of thirty-three feet. The maximum heat exchanger tube pressure of approximately twenty-two psi is well below the design pressure of 150 psi for all systems.

The Control Room is the remote operating station of the pumping components of the pool water systems. Figure 5-1 is a schematic of the pool water systems

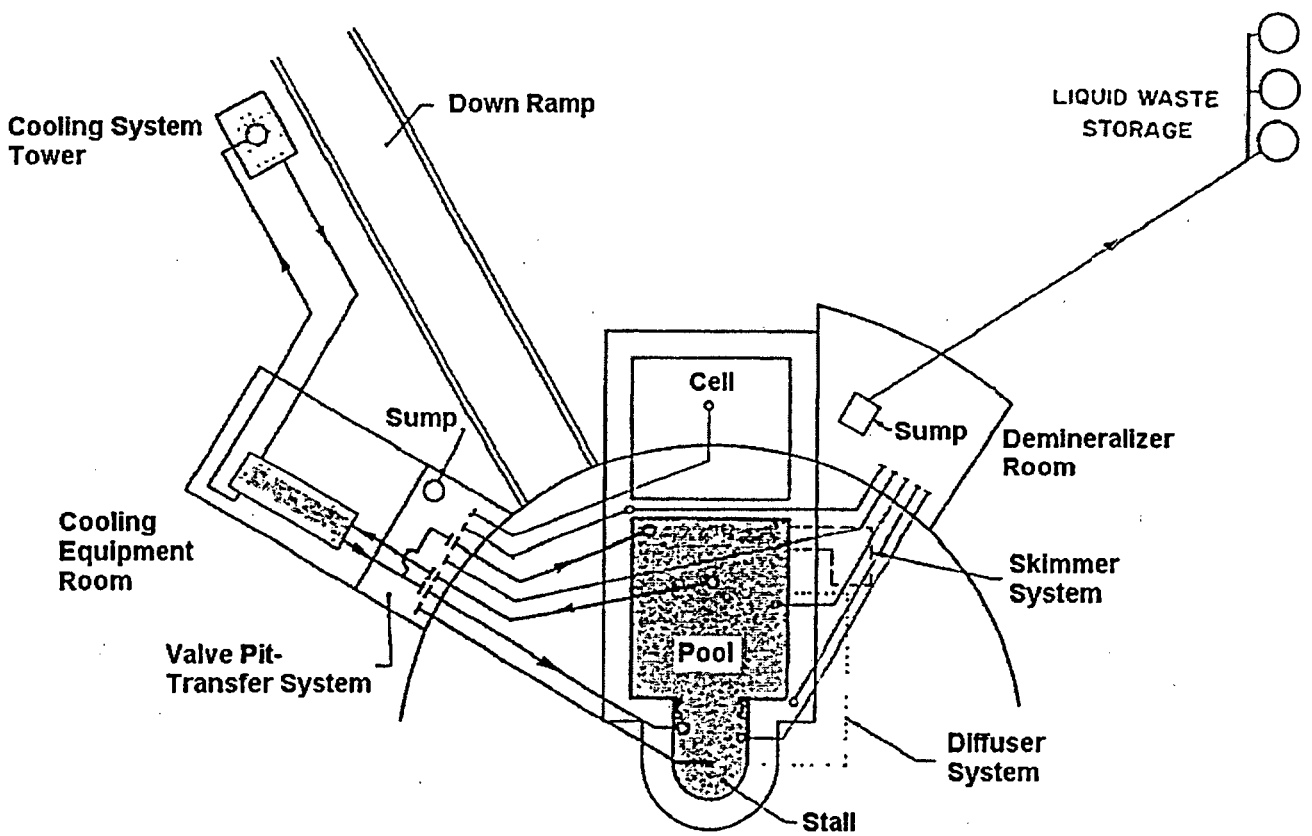


Figure 5-1: Pool Water Systems

Figure 5-2 shows the elevations of the water systems.

[redacted]

Figure 5-2: NSCR Pool Water Elevation

Two [redacted] terminate in the demineralizer room. These lines are for drainage and recirculation. Two three-inch demineralizer recirculation and fill lines are located near the top of the pool The pool surface skimmer system has two one and one-half inch lines at the top of the pool for operation A [redacted] The irradiation cell floor has a three-inch drain line. Figure 5-3 shows the piping penetrations in the reactor pool

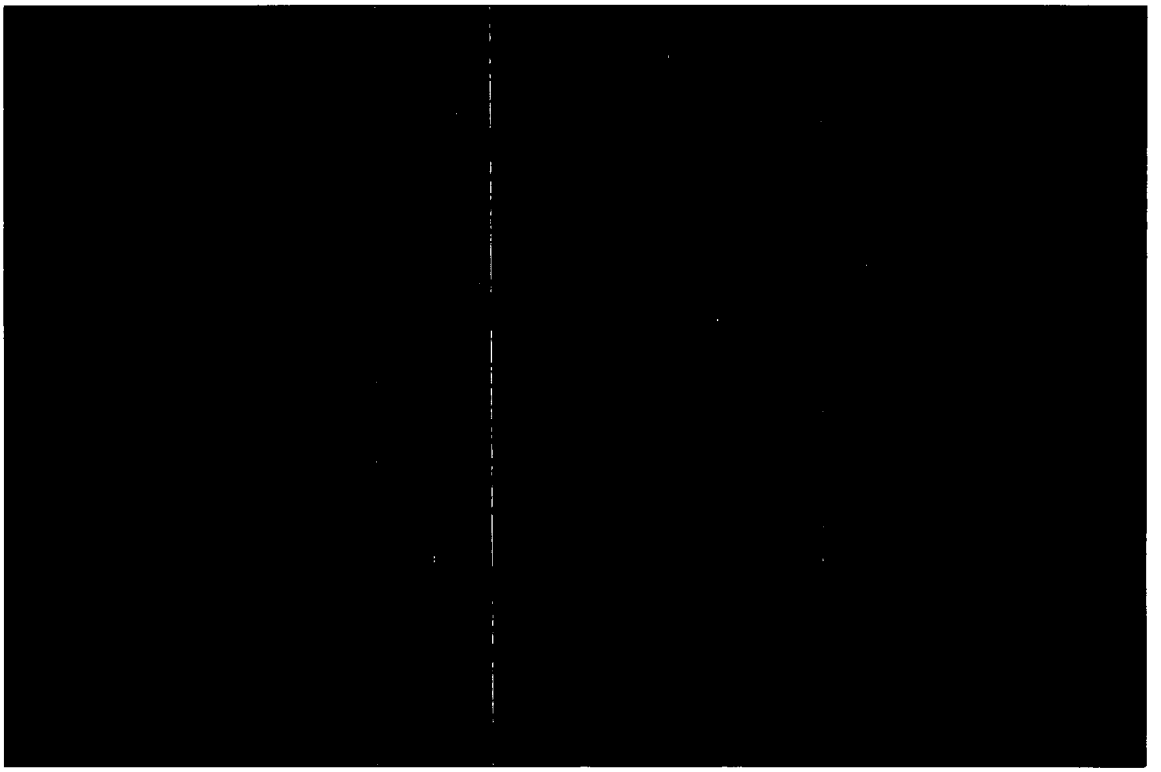


Figure 5-3: Reactor Pool Sections and Penetrations

5.2 Primary Coolant System

Three [redacted] are located [redacted] of the reactor pool. The Primary Cooling pump takes suction on the single ten-inch line located on the centerline of the main pool. The pump discharges through the heat exchanger and back to the pool through one of two ten-inch lines (these discharge in the stall and main pool). Diffusers are on the discharge of the two return lines.

The pool cooling system (Figure 5-4) has a design capacity of 2 MW with nominal pool operating temperature between 70°F and 80°F. Reactor pool water flows through the tube side of the heat exchanger for cooling and then returned to the reactor pool. This primary system is a closed loop with a design flow rate of 1,000 gpm

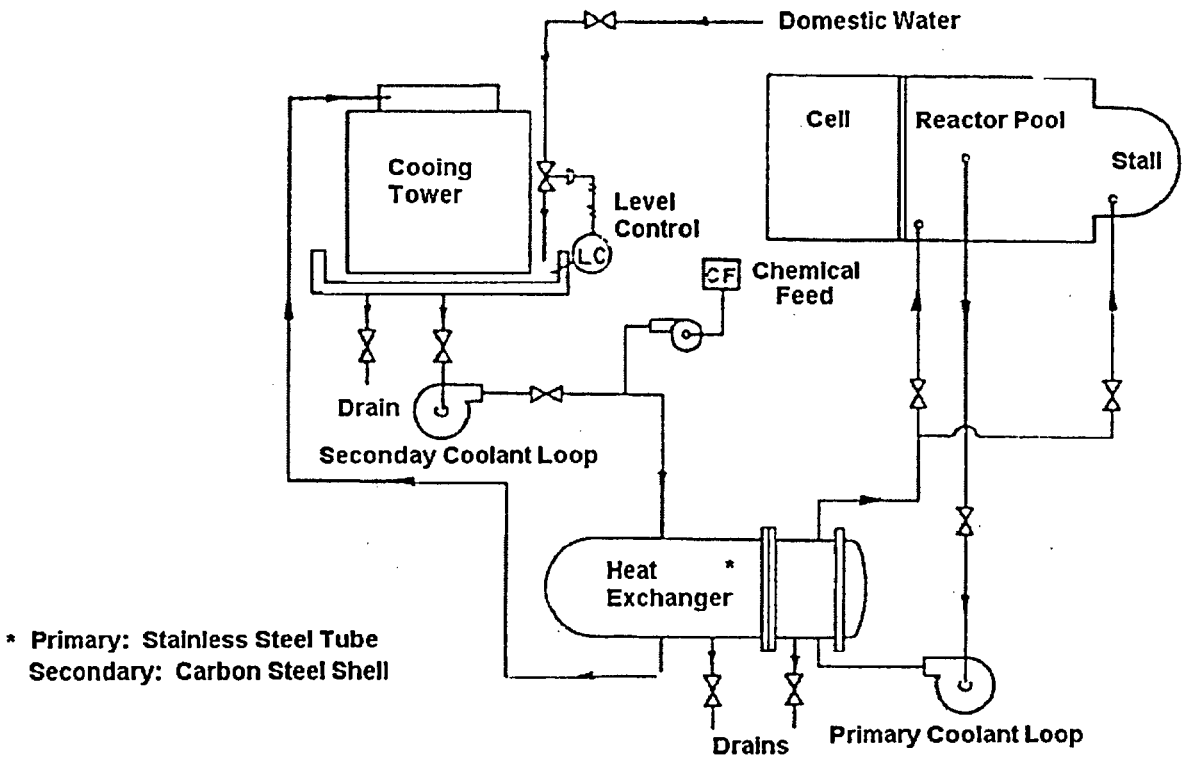


Figure 5-4: Reactor Pool Cooling System

The secondary cooling water flows from the basin of the cooling tower through the shell side of the heat exchanger and back to the cooling tower. The cooling tower uses evaporative cooling to remove heat from the secondary water to the atmosphere at the cooling tower. The secondary loop has a nominal flow rate of 1575 gpm The cooling tower will deliver 83°F water at 78°F wet bulb air temperatures

The primary loop components are stainless steel This helps to preserve pool water purity during the cooling process Components for the primary cooling loop are located in the cooling equipment room on the lower research level

The tubes, tube sheet and header of the heat exchanger are stainless steel and the shell is carbon steel. Design operating inlet pressures of the heat exchanger are 30 psi for the primary side and 22 psi for the secondary side.

The convection cooled TRIGA core does not present a problem of fuel melt down and resultant fission product release when there is a loss of coolant flow through the heat exchanger. Loss of the cooling system with the reactor in operation would result in a gradual pool temperature increase. Therefore, ample time is available before it would be necessary to terminate reactor operations due to a high pool temperature. It follows that loss of electrical power to all coolant systems would not result in a hazardous condition

5.3 Secondary Coolant System

The Secondary Cooling System consists of a pump, secondary side of the heat exchanger cooling tower and associated piping "Auto-Off-Hand" switches allow local operation and are located at the secondary pump and cooling tower. When the local switch is in the "Auto" position "On-Off" switches in the control room operate the components.

The Auxiliary Alarm Panel in the Control Room provides alarms in for primary and secondary pump power failures and for secondary loss of flow. Local detectors monitor heat-exchanger inlet and outlet temperatures A computer or electronic system displays temperatures for Control Room operators.

Chemical treatment of the secondary loop extends the life of the components and reduces scale deposits in the heat exchanger. A system to control secondary chemistry samples the water and activates chemical injection or initiates a blow down

5.4 Primary Coolant Cleanup System

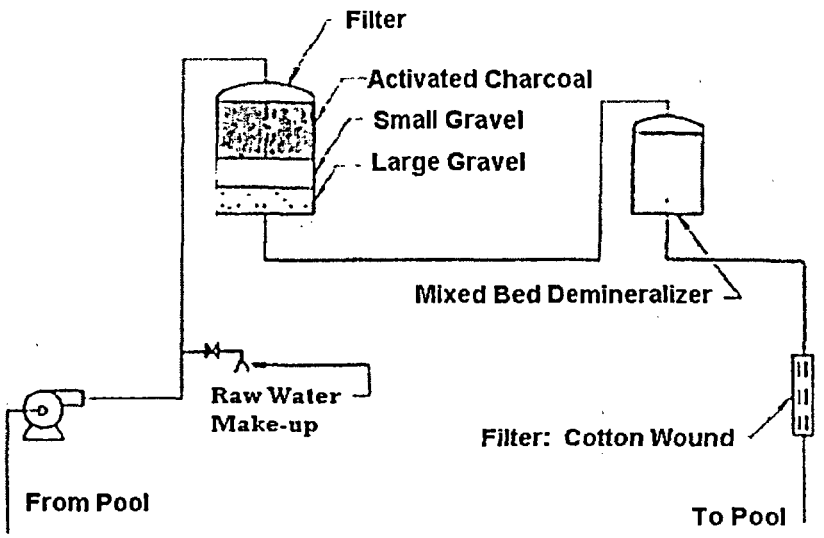
5.4.1 Demineralizer/Recirculation System

The purposes of the Demineralizer/Recirculation System are:

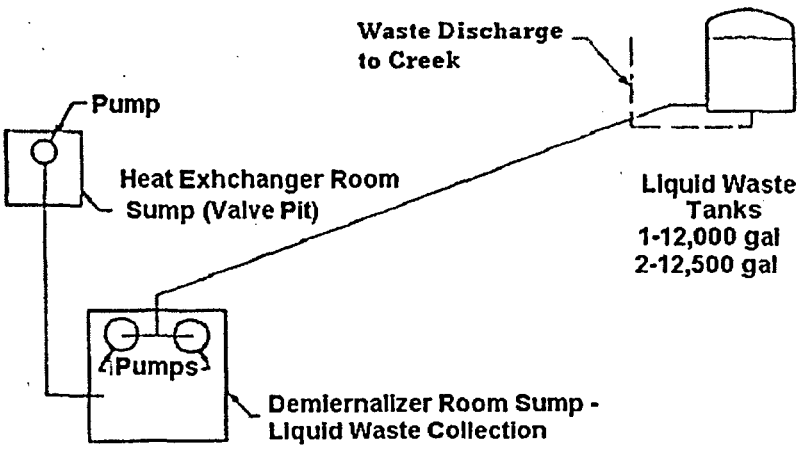
1) Maintain pool water purity,
2) Provide a filtering mechanism for makeup water and
3) Provide a path for makeup water for filling the pool during both normal and emergency pool fills.

The Demineralizer/Recirculation system uses regenerative mixed bed demineralizer unit in conjunction with micron filters, activated charcoal and gravel filters The Recirculation pump takes its suction in the Southwest corner of the pool and discharges to the Northeast corner of the main pool area.

This system (Figure 5-5) is located in the demineralizer room on the lower research level It has a design flow rate of 75 gpm with an output conductivity of less than one microseimen per cm^3



Water Purification System



Liquid Waste Disposal System

Figure 5-5: Water Purification and Disposal System

A remote "on-off" switch is located in the reactor control room for operation of the demineralizer recirculation pump. The local "Auto-Off-Hand" switch is located next to the pump in the Demineralizer room If the local switch is in the "Auto" position, the remote switch in the Control Room controls the pump. "Off" prevents the remote switch from starting the pump and "Hand" starts the pump regardless of the remote switch position.

Regular maintenance includes manual regeneration of the demineralizer as required.

5.4.2 Skimmer System

The purpose of the Skimmer System is to maintain the surface of the pool free of dust and debris. This system has little effect on the actual purity of the pool water The Skimmer pump takes suction from a surface suction filter and discharges in the Northwest corner of the pool The pump and filter are located in the chase level

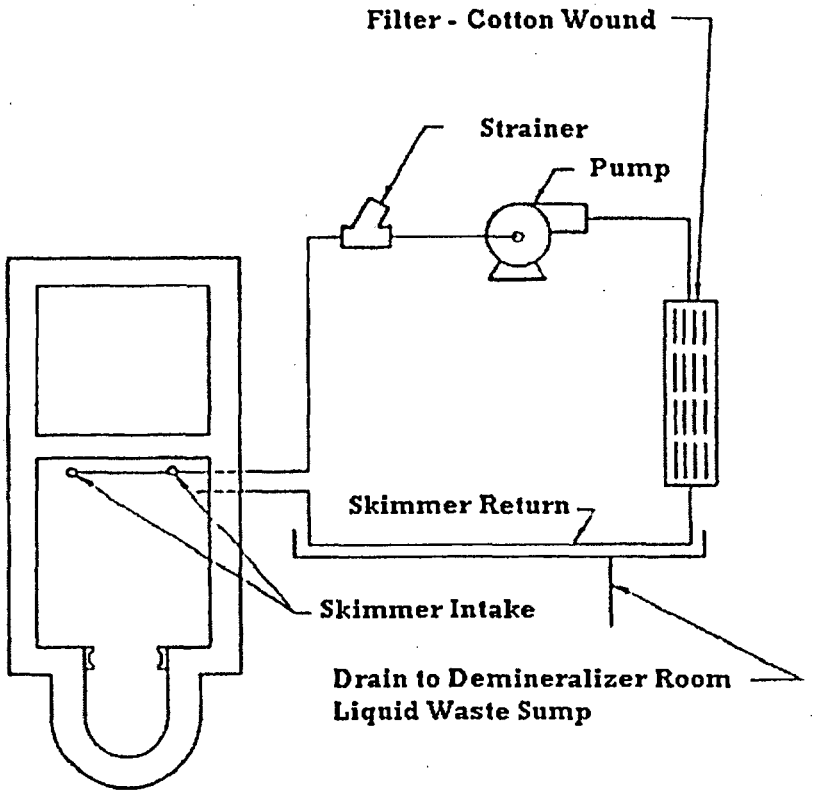


Figure 5-6: Skimmer system

5.5 Primary Coolant Makeup Water System

The Demineralizer system provides makeup water by processing raw water before it enters the pool. Raw water enters the Demineralizer system downstream of the Recirculation pump. The flow path to the pool is through the filter, charcoal bed, gravel bed and demineralizer. Figure 5-5 shows the system with the Raw Water Connection.

5.6 Nitrogen-16 Control System

The NSCR core diffuser system draws water from the pool and discharges it through a nozzle above the core. The resulting circulation pattern reduces the dose rate at the pool surface from ^{16}N and ^{41}Ar produced in the coolant water as it passes through the core. The diffuser pump and associated piping is located in the mechanical chase as in Figure 4-10. Two outlets permit operation of the system when the reactor is in the large pool or stall section. A flexible, quick disconnect water hose connects the bridge piping to the diffuser outlets. The On-Off switch for the system is in the Control Room on the water systems control panel

5.7 Auxiliary Systems Using Primary Coolant

Primary coolant provides cooling and shielding. It has no auxiliary uses

6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

Section 13 2.1 states that "no realistic hazard of consequence will result from the Design Basis Accident" As a result, the NSCR does not require engineered safety features. This analysis considered simultaneous failures of the reactor pool integrity, fuel cladding and the ventilation system operability.

6.2 Detailed Descriptions

6.2.1 Confinement

The reactor confinement building is a cylindrical steel reinforced concrete structure, approximately seventy feet in diameter and seventy feet high. Approximately fifty-five feet of the structure is above grade. An exhaust blower and fresh air inlet louvers maintain a negative pressure (relative to atmospheric pressure) inside the building Three major floor levels exist within the confinement building (Figure 2-4) A stairwell adjacent to the primary building provides access within the confinement building from one level to another.

The upper research level (Figure 2-5) is the largest by volume of the three levels. The exterior walls of this level and those of the central mechanical chase are reinforced concrete slabs between concrete-encased steel columns The columns and slabs are slanted approximately 6 degrees toward the center of the building and joined to a structural steel frame The roof deck is of poured gypsum construction Access to the upper research level is through the main stairwell, a personnel door from the reception room and a large truck door at the west end of the reactor pool. Surrounding the reactor pool are the reactor control room and various workspaces The roof for these rooms provides a floor for a mezzanine area

The next level down and approximately at grade level is the central mechanical chase. The building air ducts and blowers, electrical conduits and utility piping are located on this level. Access to the chase is either from the main stairwell or from a utility tunnel leading from the fuel storage room. The reactor pool walls take up a major portion of the available space on this level. Signal and power cables, which connect the reactor to the control room, pass through trays attached to the ceiling of the chase.

The lower research level is the lowest level of the confinement building (Figure 2-6). The floor and outer walls of this level are reinforced concrete. Access to this level is through the main stairwell and the lower level double-doors. Facilities located at this level are the cooling system equipment room, research laboratories, the demineralizer room, beam ports and thermal column. Several steel tubes extend into the east wall of this level and provide storage facilities for beam port plugs

The reception room is located outside the south side of the confinement structure A master control panel for operation of exhaust and air conditioning systems in the confinement structure is located on the north wall of the reception room.

A laboratory building, on the south end of the reception room and outside the confinement building, contains pneumatic receivers (Figure 2-7). Each pneumatic system to the laboratory building lies within a large airtight tube and air within this tube flows through the existing exhaust system and Facility Air Monitors before release from the stack. This design allows for monitoring and controlled release of radioactive gases associated with operation of the pneumatic system

Four air handling units and an exhaust fan control pressure, temperature and humidity within the confinement building. The confinement building has three zones of negative pressure for effective isolation of possible contaminated areas. The zone of least negative pressure includes the control room, locker areas and the building entry where contamination lest likely. Air recirculates in this zone but exhausts through the monitoring system and the stack. An intermediate zone of negative pressure includes the upper and lower research levels where infrequent contamination might occur. Air also recirculates in this zone. The third zone of maximum negative pressure includes areas where contamination of activation is most likely to occur, i.e., beam ports, thermal column, and

through tubes Also included in this zone of maximum negative pressure is Laboratory 1. Air does not recirculate in this zone and exhausts directly to the stack and monitoring system.

Air-handling units provide and circulate fresh air in the building All four units have controls on the control panel in the reception room and will shutdown simultaneously with the central exhaust fan when airborne radioactivity reaches alarm levels on the exhaust particulate monitor, the Xe-125 monitor or the fission product monitor

Dampers are located at the air inlet to all handling units, the fresh air bypass to the exhaust fan, and in the 84-feet high exhaust stack The Air Handler Shutdown button in the Control Room simultaneously closes the dampers and shuts off the air handlers, thus isolating the building. An emergency exhaust filter-bank is in-line between the exhaust fan and building stack. The emergency filter system consists of two particulate filter banks and one bank of activated carbon filters. The controls for putting the filter bank on and off-line are in the reception room.

The air handling system is comprised of two sections. One section handles fresh air, controls temperature and humidity, and recirculates building air. The second section controls building pressure and exhaust A control panel is located in the reception room for operation of the system The air-handling units, exhaust-fan and associated dampers can be operated from this panel. Emergency air handling operations are performed at this panel.

6.2.2 Containment

The Central Exhaust Fan and the Central Exhaust Bypass Dampers maintain the building at a negative pressure To prevent or minimize the uncontrolled release of radioactivity to the environment surrounding the Nuclear Science Center. In addition, the Facility Air Monitors continuously monitor air discharged from the building If a spill or a release increases radioactivity levels above a pre-selected set point, the Facility Air Monitors generate a signal to shutdown the air-handlers and close the inlet dampers This isolates the building. Operators can control the air handling system, including the Emergency Exhaust Filter-bank, and monitor the activity in the air from the control panel in the reception room

6.2.3 Emergency Core Cooling System

Emergency cooling for the NSCR is the 106,000 gallons of water contained in the pool and stall portion of the reactor pool. The large heat capacity of this amount of water can cool the reactor for several hours at 1 MW in the event of failure of the cooling system.

In the event of gross leakage of water from the primary system, a float switch in the reactor pool shuts down the pool-water recirculation pump at a preset pool water level. This switch also energizes an alarm in the University Communications Room Communications operator response to this alarm is to notify the first available person on the NSC Emergency Notification Roster. The capacity of the pool is so large that even a major leak is unlikely to uncover the core before NSC personnel arrive.

The two coolant return lines and the coolant extraction line [redacted] have manual closures to isolate the pool in the event of cooling system component failure.

Two emergency raw water fill lines are installed adjacent to the reactor pool which can supply approximately 400 gallons per minute to the reactor pool in the event of loss of beam port integrity, pump housing failure, coolant line breakage, or other catastrophic accident.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The reactor operates in two modes:

- Steady State Mode – steady power levels up to 1 MW
- Pulse Mode – a rapid Transient Rod withdrawal (the technical specifications set the limit) causes a large power excursion

A Mode Selector Switch at the Reactor Console allows operators to select between Steady State and Pulse modes

All reactor operations are at the Reactor Console, which provides for reactor and reactor systems controls and indication of reactor and reactor systems parameters

7.2 Design of Instrumentation and Control Systems

Five radiation-based instruments provide indication of reactor power from intrinsic source range levels to full power. Two of these instruments, A Wide Range Linear Drawer (with multiple scales) and a Log Drawer (with multiple instruments), provide indication over the entire range. Two, the Safety Drawers, provide indication only above 10kW. One, the Pulse Drawer, provides indication above 10kW, and provides indication of peak power and energy during reactor pulsing. The Fuel Temperature Instrument provides indication of fuel temperature and records maximum temperature during pulses The Safety Drawers, Log Drawer and Fuel Temperature Instrument provide SCRAM capability to the Reactor Safety System.

The two Steady State methods of controlling the reactor are Manual and Automatic In Automatic (i e. Servo), the Wide Range Linear Drawer provides the power level input to a servo controller. The servo controller generates a signal to drive the Regulating Control Rod (Reg Rod) as required to maintain a constant-preset power level.

Various experiment and manual scrams exist as interlocks that will automatically shut down the reactor.

7.2.1 Design Criteria

The instrumentation and control system provides the following:

- Information on the status of the reactor
- The means for insertion and withdrawal of control rods
- Automatic control of reactor power level
- The means for detecting over-power, fuel over-temperature or loss of detector voltage and automatically shutting down the reactor to terminate operation

7.2.2 Design-Basis Requirements

The primary design basis for TRIGA reactor safety is the limit on fuel temperature. In the pulse mode, the reactor SCRAMS when temperature reaches the Limiting Safety System Set point for a fuel temperature. In Steady State mode, fuel temperature trip and the power level trips independently prevent exceeding a safe operating temperature.

7.2.3 System Description

7.2.3.1 Log Power Channel

The log power channel consists of a fission chamber, preamp, amplifier and rate meter.

The log power channel performs the following:

- Provides reactor power indication over a of range of ten decades of reactor power
- Provides the following:
  - Low Count Interlock
    - Prevents withdrawing control rods without at least two counts per second

- Period Scram
  - Provides a scram signal to the Safety Amplifier when reactor period is three seconds or less and the Period Scram is not bypassed

The Log Drawer contains two overlapping instruments. The low-range instrument converts pulses from the fission chamber to a logarithmic power indication The high-range instrument converts detector current to a logarithmic power indication The instruments overlap to provide continuous indication Figure 7-1 shows a simplified diagram of the log power channel

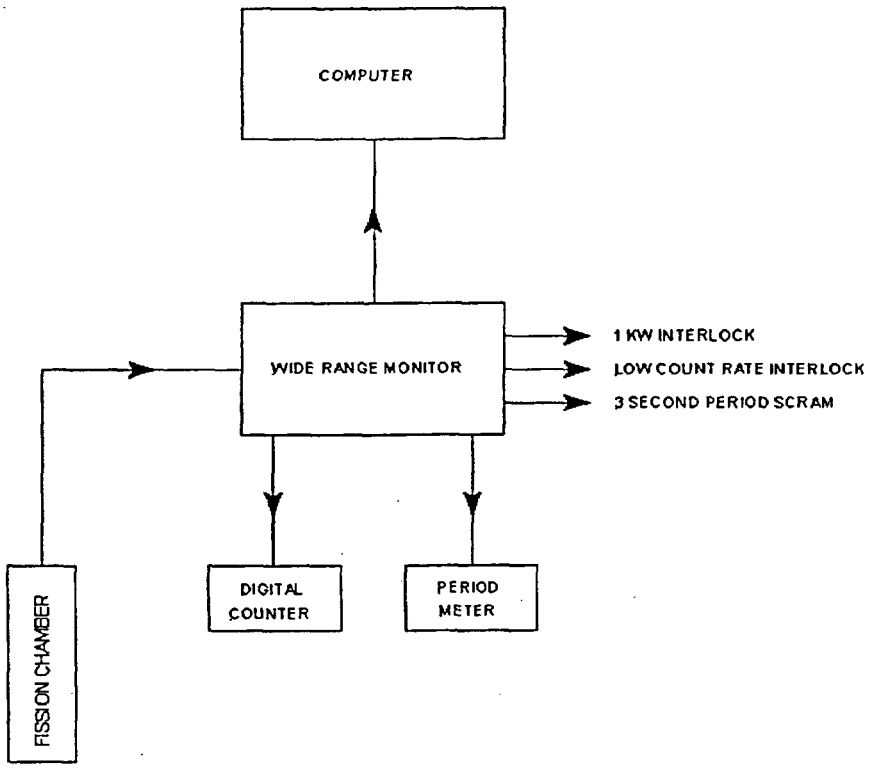


Figure 7-1: Log Power Channel

7.2.3.2 Pulse Channel

The Pulse Channel consists of an uncompensated ion chamber and the Pulse Drawer.

The Pulse Drawer provides the following indications in the associated mode

- Steady State Mode
  - Percent Power
- Pulse Mode
  - Percent Power
  - Peak Power
  - Energy (Mw-Sec)

7.2.3.3 Wide Range Linear Channel

The linear power channel consists of a compensated ion chamber and a Linear Wide Range Drawer.

The Linear Wide Range Drawer provides the following:

- Power indication over the entire range of operating and shutdown levels

- Input to the servo controller for automatic power control

The detector is above a tapered graphite reflector element that scatters the neutron flux from the core face This configuration provides excellent linearity and significantly reduces the contribution due to gamma rays so that the system is sensitive and accurate at low power levels even after extended operation at 1MW.[1]

7.2.3.4 Servo Control System

In automatic control, the servo controller compares the signal from the Linear Wide Range Drawer to a preset signal. It provides a shim-in or shim-out signal to the Regulating Rod Drive Controller to adjust power, as indicated by the Wide Range Linear Drawer, to the preset level. Regulating rod control automatically shifts back to manual if the actual level drifts excessively from the preset level The Regulating Rod Drive Controller receives a signal from both the servo controller and the manual control switch on the Rod Control Module. Figure 7-2 shows the linear power channel

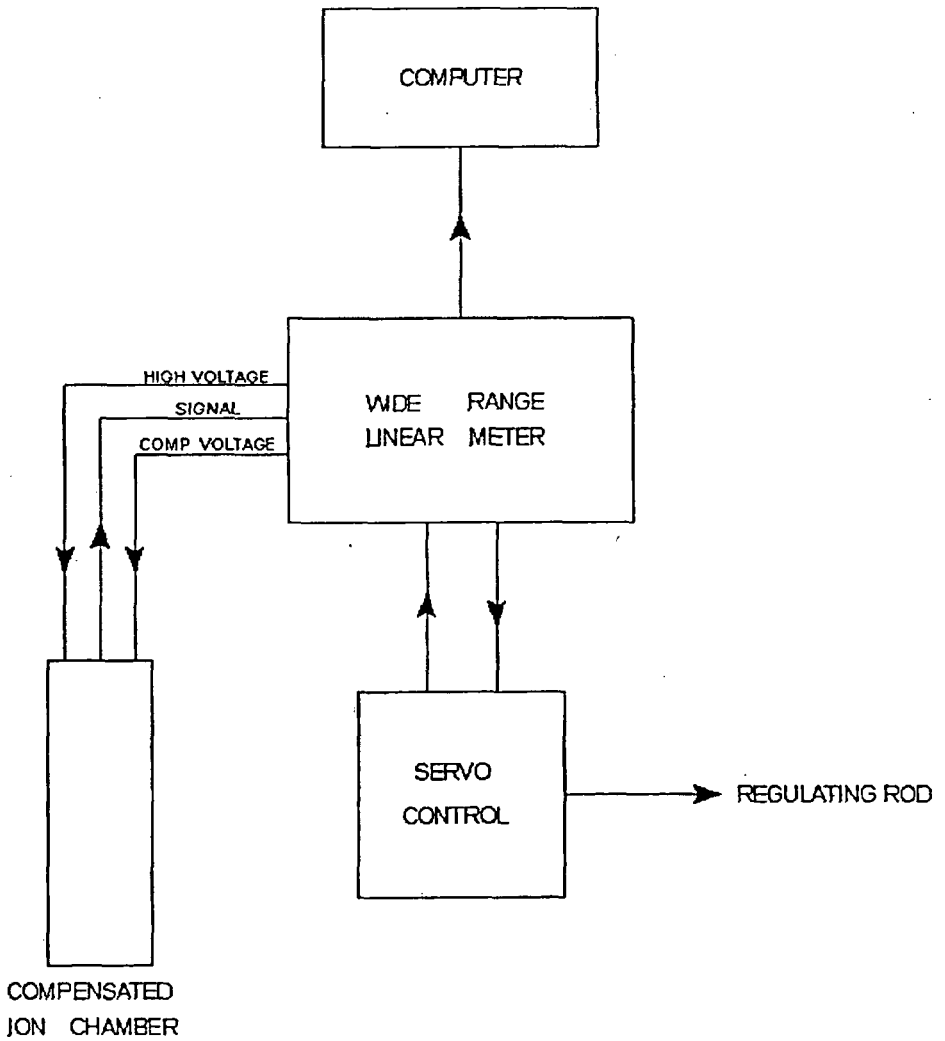


Figure 7-2: Wide-Range Linear Drawer

7.2.3.5 Safety Power Channels

Each of the two Safety Power Channels consists of an uncompensated ion chamber, a Safety Drawer and an external high-voltage power supply The two channels are identical and isolated from each other. Each channel provides an independent scram input to the Safety Amplifier located between the Safety Drawers

The Safety Power Channels can be the Limiting Safety System with the LSSS at 125% Full Power in Steady State if the Fuel Temperature is not available

There are two scrams associated with the Safety Drawer.

- High power scram
  - Provides a scram when the reactor power reaches 125% of full power
- Loss of high voltage scram
  - Provides a scram signal when the detector voltage drops below 150V

7.2.3.6 Safety Amplifier

The Safety Amplifier supplies current to the control rod magnets providing the mechanism for scramming the reactor. When the Safety Amplifier receives a scram signal, it stops supplying current to the electromagnets that hold the control rods in position. Without the magnets, the control rods gravity-fall to the fully inserted position.

The Safety Amplifier also receives scrams that are not internal to the Safety Power Drawer. Following are the scrams provided to the Safety Amplifier·

- High Power (125% Full Power)
- Low Voltage (150V)
- Period (<3 sec)
- Fuel Temperature (975°)
- Manual (Console)
- Bridge lock scram
- Various Experiment Scrams allow experimenters to independently and locally scram the reactor. These are manual scram buttons located as follows:
  - Beam Port Areas
  - Irradiation Cell
  - Reactor Bridge
  - Pool Side
- Interlocked scrams that ensure the reactor is shutdown when:
  - Beam Port 4 Cave Door is open and the reactor is near the Thermal Column Graphite Coupler Box
  - Cell Door is open and the reactor is within 8 feet of the cell window

7.2.3.7 Fuel Temperature Channel

The Fuel Temperature Channel consists of a thermocouple embedded in an instrumented fuel element and a Fuel Temperature Instrument. A second temperature indicator on the Reactor Console, with a thermocouple selector switch is available to read out the temperature of thermocouples in the fuel, the pool water and the irradiation cell. Fuel Temperature is also available in the Reception Room normally from a lower reading thermocouple. Figure 7-4 shows a diagram of the Fuel Temperature Channel.

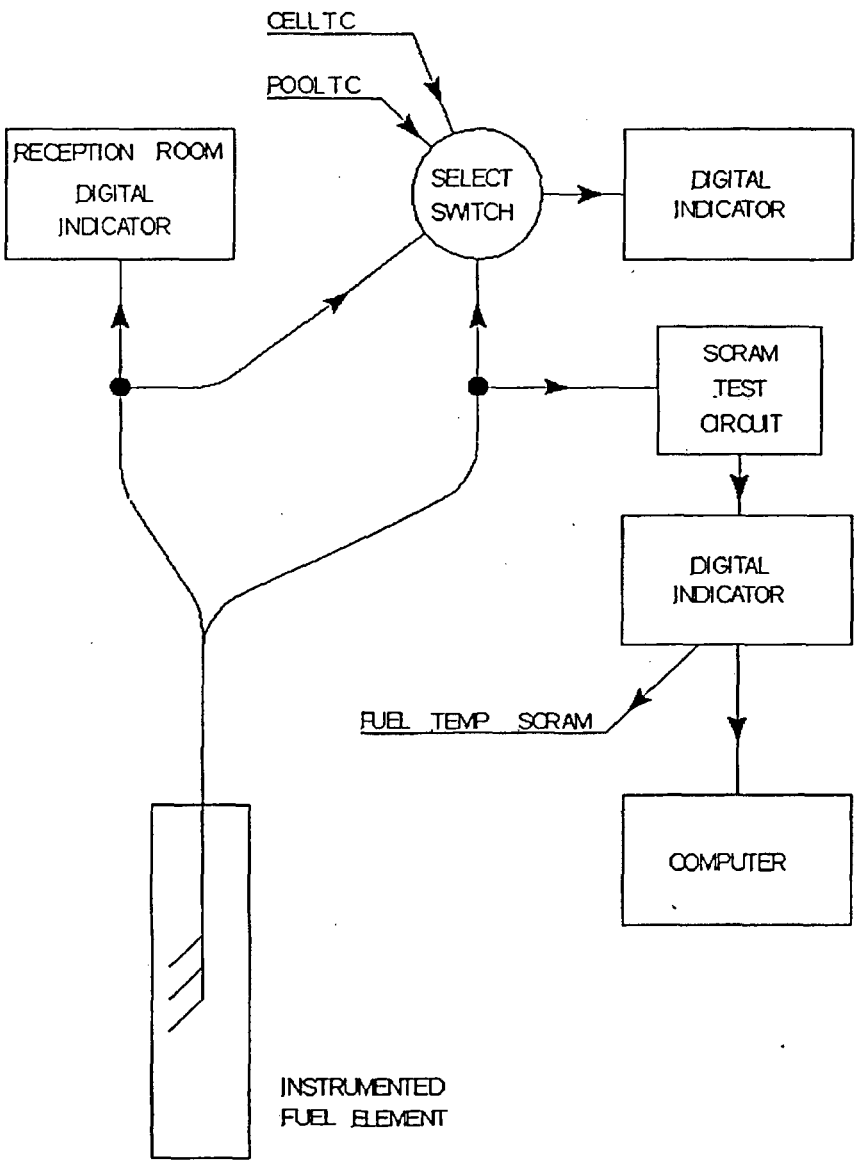


Figure 7-3: Fuel Temperature Channel

The Fuel Temperature Channel is the Limiting Safety System with the LSSS at 975°F. If the Fuel Temperature Channel is not available, the Safety Channels can act as the Limiting Safety System.

The Fuel Temperature Instrument performs the following functions:

- Scrams the reactor if the thermocouple temperature reaches 975°F
- Captures peak fuel temperature during pulse

The Fuel Temperature Instrument normally operates in continuous indication mode. It captures and displays peak pulse temperature only when operating in the peak mode.

The instrumented fuel element (IF) is located adjacent to the central bundle excluding the corner positions and observed temperatures are proportional to maximum fuel temperature experienced by the fuel. This IF can be in any of eight locations in the core. Three chromel-alumel thermocouples, embedded in the IF, are located at the vertical center and one inch above and below the vertical center and 0.3 inches from the center (Figure 4-7).

7.2.3.8 Preset Timer

The Preset Timer scrams the Transient Rod 15 seconds or less after a pulse The design of the Preset Timer is such that the actual time is adjustable, but the maximum setting is 15 seconds.

The purpose of the Preset Timer is to prevent the reactor from restarting following a pulse

7.2.4 System Performance Analysis

The instrumentation and control systems have been in routine operation for over 40 years Solid-state electronics have replaced nearly all measuring instruments The result is improved reliability.

Limiting Safety System Setting, Limiting Conditions for operation, surveillance requirements and action statements concerning the control and instrumentation systems are in the Technical Specifications

7.2.5 Conclusion

The Safety Power Channels and the Fuel Temperature prevent exceeding the operating limits for fuel temperature and reactor power. The operating limits for temperature and power independently protect from exceeding the Limiting Safety System Setting

Other scram conditions including loss of ac power, Loss of Safety detector voltage and Manual Scram ensure that the safety equipment will operate as planned.

7.3 Reactor Control System

The Reactor Control System consists of Rod Control Modules, Control Rod Drive Mechanisms and external inputs (i.e. shim in/shim out signal and interlocks) to the Rod Control Modules for Steady State and Pulse modes. The Rod Control Systems for each Control Rod consist of a hold-down device, Control Rod barrel, electromechanical Control Rod Drive Mechanism (CRDM), Rod Control Module and associated control circuits.

The Rod Control System performs the following functions:

- Provides method for controlled addition of reactivity
- Provides scram capability
- Holds Control Rods and Control Rod bundles in position
- Provide indication of Control Rod position
- Provides Rod Withdrawal interlocks

Rod Control Modules in the Reactor Console:

- Provide signal for individual rod motion
- Provide Logic interlocks
- Provide rod and carriage position indication

Each Control Rod Drive Mechanism (CRDM) motor drives either a lead screw (Shim Safeties and Regulating Rod) or a chain-driven externally threaded cylinder (Transient Rod). The CRDM couples the rod extension to the carriage to move the control rod. The hold-down assembly assures control rod bundle remains in place.

The CRDM receives a signal that controls carriage motion from its associated Control Rod Module. This signal controls the motor, which is the source of rod motion. When the rod is coupled to the carriage, the motor controls rod motion; when it is not coupled, the rod remains fully inserted regardless of carriage position The CRDM also has various switches that provide information to the Rod Control Module about the status of the rod (the following sections explain these in detail).

7.3.1.1 Shim-Safety Rod Control

The rod control system for the Shim Safety control rods allows the operator to control these four rods individually or as a group. Each rod drive has a Rod Control Module, Control Rod Drive Mechanism (CRDM), control rod barrel

with offset and hold-down tube In addition, the Shim Safety Rod Drive Systems share control circuitry for interlocks and group motion and a Power Supply for rod motion.

Figure 7-4 shows the control rod magnet and armature and rod assembly-dampening device for the shim-safeties The piston action provides dampening of the control rod towards the end of its fall into the core. Water relief slots in the barrel allow the rod to drop freely until the rod begins the last six inches of travel. At this point, the piston ring forces the water out of the bottom of the control rod barrel. When the piston enters the piston receiver, the reduced clearance dampens the rod's fall Stainless steel and aluminum components provide smooth movement and reduced wear of sliding parts

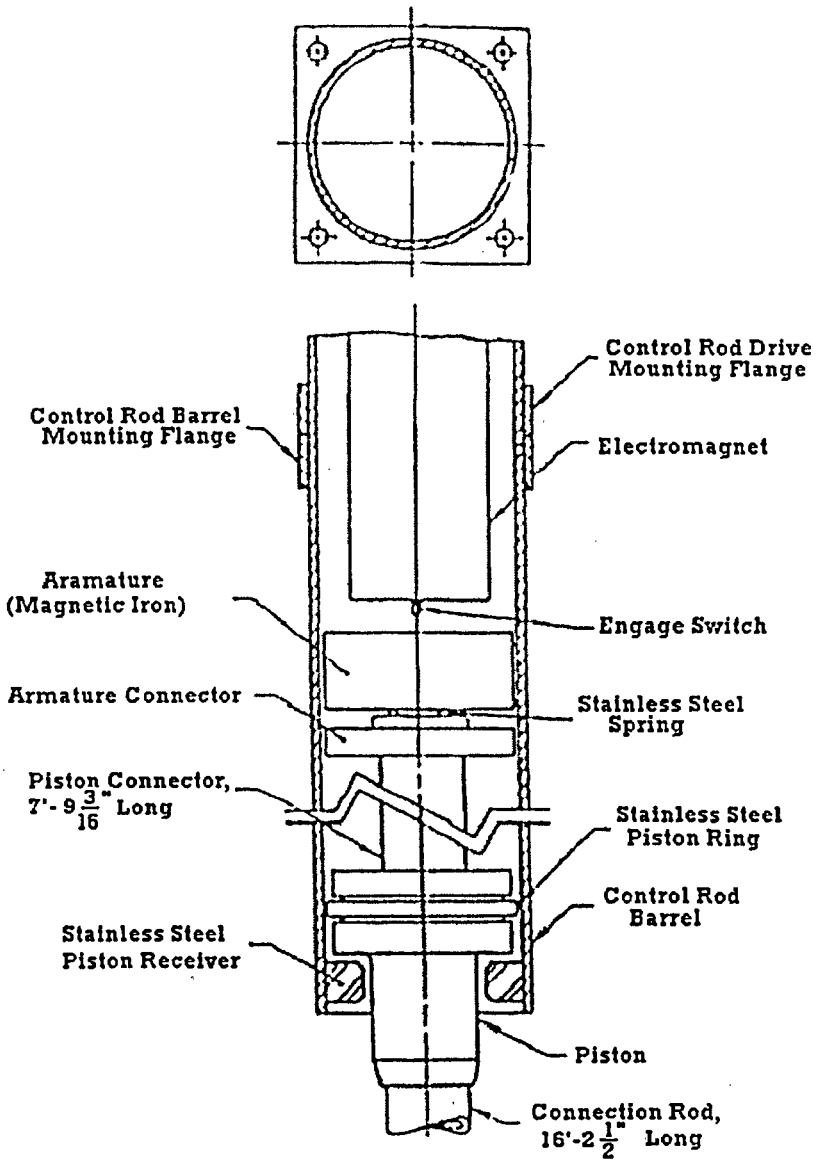


Figure 7-4: Shim Safety Armature and Dampening Device Assembly

The shim-safety control rods have eight optional control rod positions The offset assembly functions similarly to the bolt action of a rifle. Spacers, which separate the two piston rods, are in slots that allow vertical movement but restrict lateral movement. The offset barrel can rotate in 45° increments. The hold-down assembly provides a means to enclose the control rod extension and prevent accidental lifting of a fuel element The hold-down tube extends downward to the reactor core and fits over the upper end the cross bar of the fuel bundle

The control rods attach to a horizontal plate on the upper portion of the reactor frame structure with machined slots and clamps to hold the rod drives in position (Figure 7-5). A support ring holds the Shim Safety Control Rod assembly. This assembly permits removal of the associated control drives for maintenance without moving the associated control rod from the core. Figure 7-6 shows the installation of a shim-safety control rod

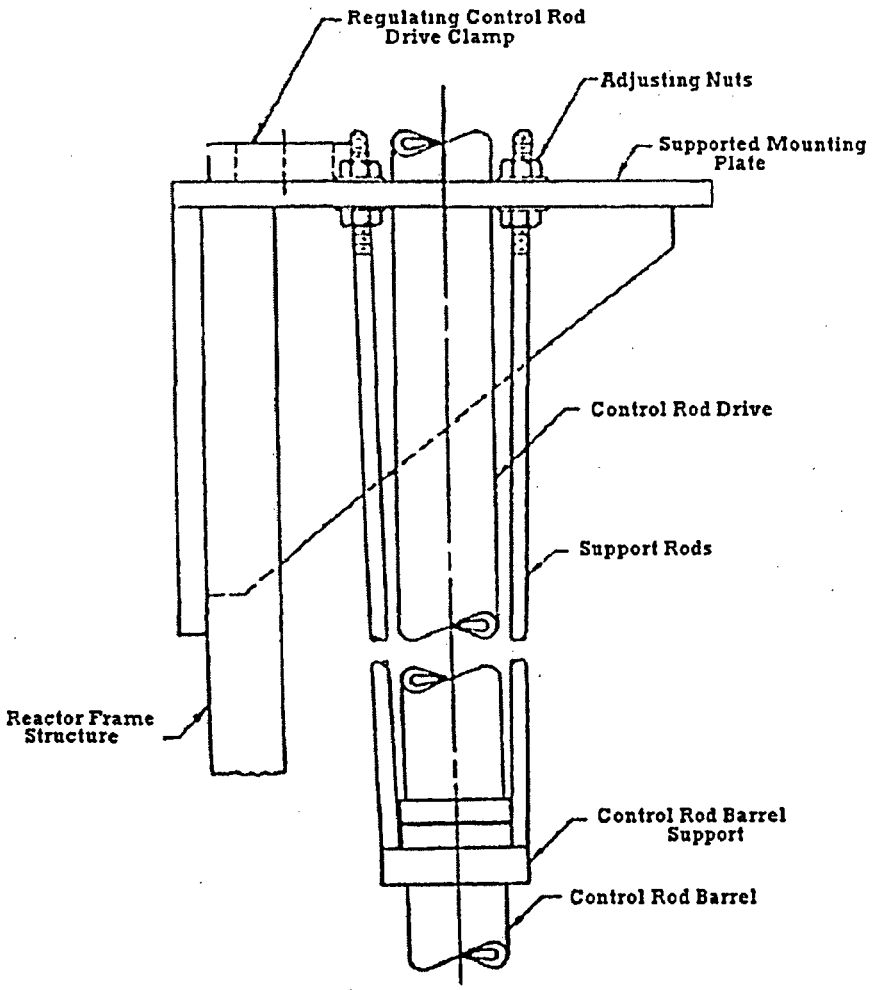


Figure 7-5: Control Rod Assembly Support Mechanism

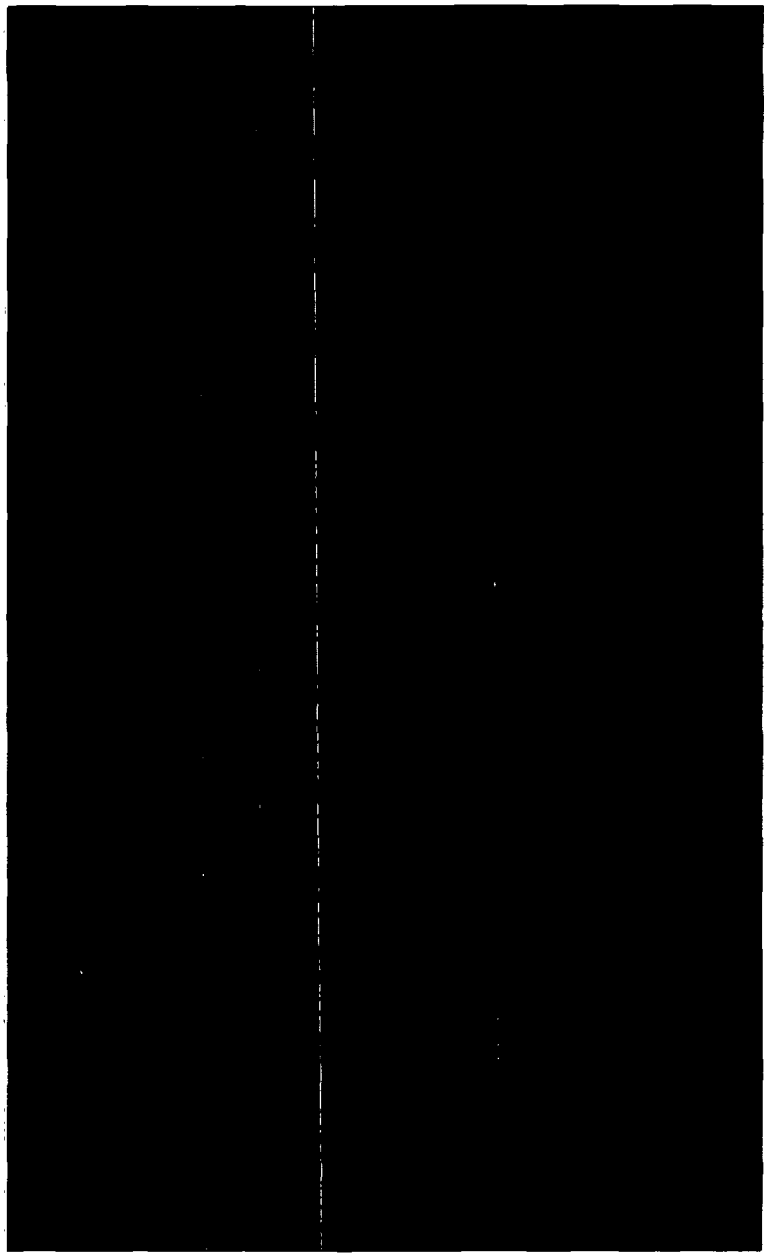


Figure 7-6: Control Rod Installation

Each shim-safety rod mechanism connects to a Rod Control Module at the reactor console. Push buttons permit operation of each rod drive independently of the other control rods. A Gang Switch, located on the Reactor Console near the Modules, permits operating all Shim Safety Control Rod drives simultaneously. The Gang Switch does not necessarily override the individual rod drive buttons on the individual Modules. Rather, an IN signal -from either the individual or Gang switch- overrides an OUT signal.

Travel speed for the shim-safety rod drives is 11.3 centimeters per minute.

To provide rod height indication, the Module receives a signal from a digital encoder that rotates with the lead screw through the stepping motor drive shaft The Module uses this signal to provide Carriage Height indication in units of percent withdrawn and to provide logic interlocks for carriage full out (100%) and carriage full in (0%). If the rod is coupled to the carriage (as indicated by the Engaged light), rod height and carriage height are the same.

The *Rod Control Modules* for the Shim Safety Control Rods perform the following *functions:*

- Provides digital indication of carriage height
- Provides Rod In/Rod Out signal to CRDM
- Provides indication of the following
    - Rod Engaged
    - Rod Down
    - Rod Jammed
    - Carriage Down
    - Carriage Up
- Resets the rod position indication to 1 0% when:
    - The engaged switch changes from disengaged to engaged

      -while-
    - Carriage is driving in

      -and-
    - The Rod Down switch is made
- Provides the following logic interlocks
    - Prevents rod insertion for jammed rod
    - Prevents rod insertion if
        - Carriage height is 0 0

          -and-
        - Rod is engaged
    - Prevents rod withdrawal if rod height is 100.0
    - Prevents rod withdrawal if the gang switch is in the Gang Down position
- Provides individual rod withdrawal and insertion capability

Interlocks associated with the *Shim Safety Control Rods* are as follows.

- Rod Jammed
    - Prevents driving carriage down when lead screw presses the Jam switch
- Rod Down
    - Prevent driving carriage down when:
        - Carriage height indication at 0.0%

          -and-
        - Rod engaged
- Rod Out Interlock
    - Prevents rod withdrawal if rod height is 100.0%
- Rod In Override
    - If Gang Switch is in the Gang Down position or the individual Rod Down button is pushed, the rod will drive in regardless of the position of the other switch
- Shim Safety Pulse Interlock
    - Prevents withdrawing Control Rods in the Pulse Mode
- Low Count Interlock
    - Prevents withdrawing Control Rods with < 4mW on the Log Power Channel

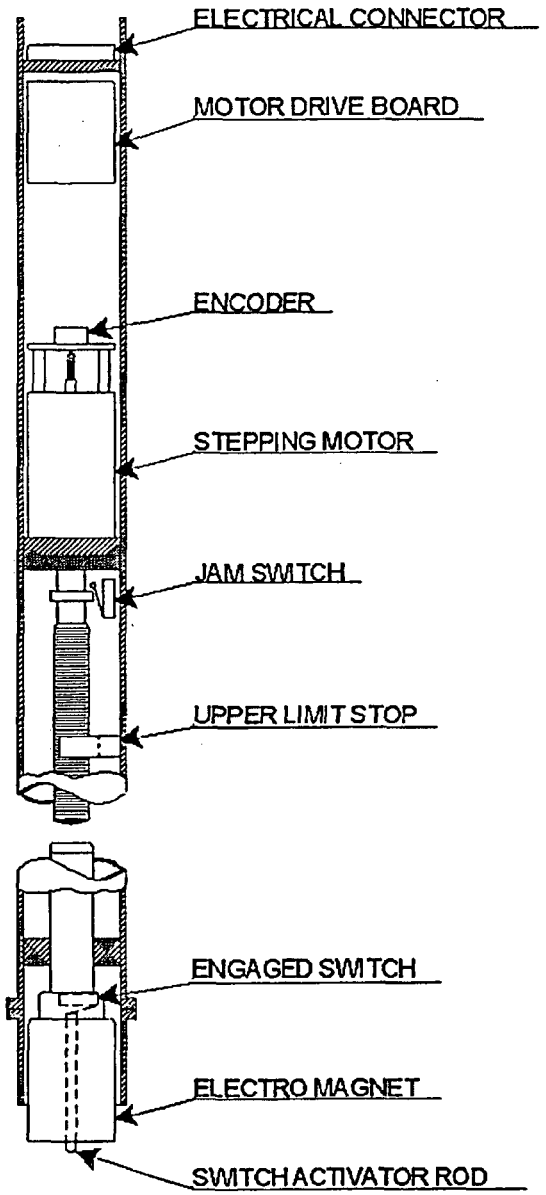


Figure 7-7: Control Rod Drive Mechanism for Shim Safety Control Rods

The Control Rod Drive Mechanisms (CRDM's) (Figure 7-7) are electromechanical assemblies that provide the motive force to move control rods and hold them in position. The motor provides the movement, while an electromagnet (attached to the bottom of the lead screw) holds the rod. When energized, the electromagnet holds the iron armature against the Engaged switch

Each Shim Safety CRDM has an Engaged, Jam and Rod Down switch. These three physical switches perform the following and have the following physical descriptions:

- Engaged Switch
  - Functions
    - Lights Engaged light on Rod Control Module

        - Provides input to rod position reset circuit to reset indication at 1 0% when the control rod extension presses the Engaged switch.
        - Provides input to Rod Down Interlock (prevents further insertion if rod at 0 0% and rod engaged)
    - Description
        - Push button on the bottom face of the electromagnet
- Rod Down Switch
    - Functions
        - Lights Rod Down light when control rod is <5% withdrawn
        - Provides input to rod position reset circuit
    - Description
        - Magnetic reed switch external to the CRDM and adjacent to the armature on top of the control rod extension assembly
- Rod Jammed
    - Functions
        - Lights Rod Jammed light
        - Provides input to rod jammed interlock, which prevents rod insertion if rod is jammed
    - Description
        - A micro-switch pressed when the lead screw drives in without the carriage lowering

7.3.1.2 Transient Rod Control

The rod control system for the Transient control rod (Transient Rod) allows the operator to control this rod individually in both the Steady State and Pulse Mode. The rod drive system has a Rod Control Module, Control Rod Drive Mechanism (CRDM), control rod barrel with hold-down tube. In addition, the Transient Rod Drive System shares control circuitry for interlocks with neutron detection instruments and other Rod Control Modules.

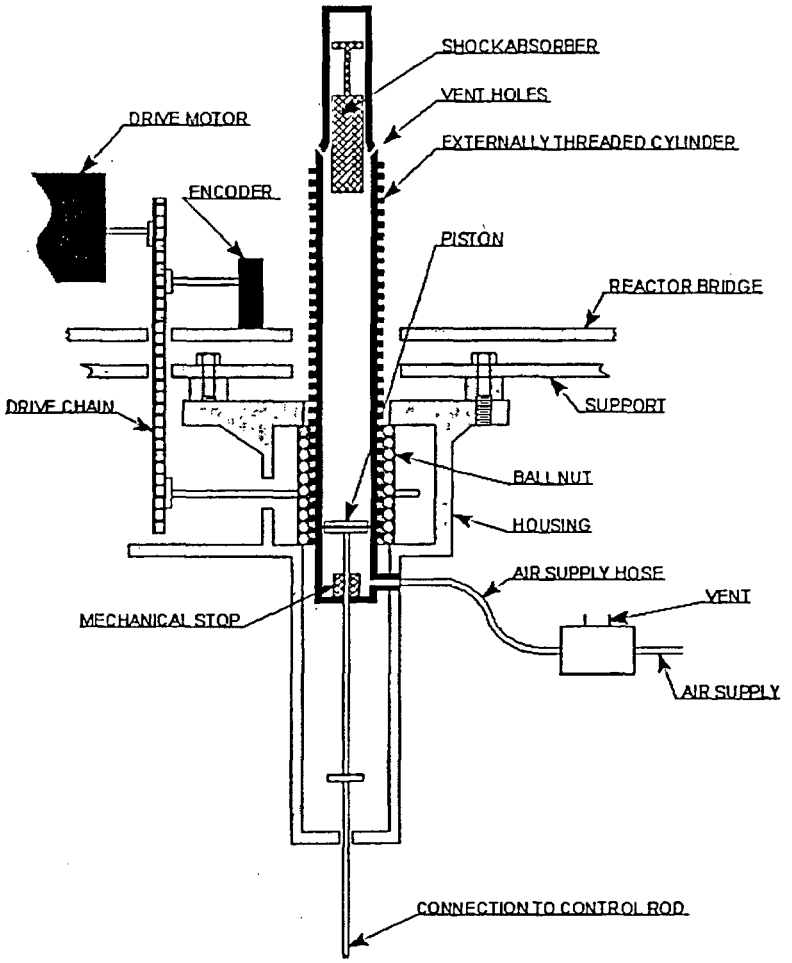


Figure 7-8: Transient Rod Drive

Figure 7-8 shows the pneumatic-electromechanical Control Rod Drive Mechanism for the Transient rod. The pneumatic portion of the CRDM is a single acting pneumatic cylinder. A piston within the cylinder attaches to the transient rod by means of a connecting rod. The piston rod passes through an air seal at the lower end of the cylinder. Compressed air, admitted at the lower end of the cylinder, drives the piston upward. As the piston rises, the compressed air above the piston exhausts through vents at the upper end of the cylinder. During the final inch of travel, the shock absorber slows the rod to minimize mechanical shock when the piston reaches its upper limit stop.

An accumulator tank mounted on the reactor bridge stores compressed air for operating the pneumatic portion of the CRDM. A three-way solenoid valve controls the air. De-energizing the solenoid valve interrupts the air supply and relieves the pressure in the cylinder so that the piston drops to its lower limit by gravity.

Following describes the operation of the Transient Rod for both modes of operation.

- Steady State
  - High-pressure air acting on a piston holds the transient rod against its rod drive carriage (specifically the shock absorber)
- Pulse
  - High-pressure air pushes the transient rod rapidly to the carriage position
  - Air vents when the Preset Timer times out

The transient rod drive attaches to a support frame that bolts to the reactor bridge. The Transient control rod is in the center of the core The hold-down assembly provides a means to enclose the control rod extension and prevent accidental lifting of a fuel element The hold-down tube extends downward to the reactor core and fits over the upper end of a control rod-guide tube.

The Transient rod mechanism connects to a Rod Control Module at the reactor console. This module is similar to those for the Shim-Safety control rods. Push buttons permit operation of the rod drive.

Travel speed for the Transient rod drive is 11 3 centimeters per minute

To provide rod height indication, the Module receives a signal from a chain driven digital encoder that rotates with the motor and worm assembly. The Module uses this signal to provide Carriage Height indication in units of percent withdrawn and to provide logic interlocks for carriage full out (100%) and carriage full in (0%). If the rod is coupled to the carriage (as indicated by the Air Applied light), rod height and carriage height are the same.

The *Rod Control Module* for the Transient Control Rod performs the following *functions*:

- Provides digital indication of carriage height
- Provides Rod In/Rod Out signal to CRDM
- Provides indication of the following
    - Air Applied
    - Rod Down
    - Carriage Down
    - Carriage Up
    - TR Fire Ready
- Resets the rod position indication to 1.0% when
    - The Carriage Down switch makes
      -while-
    - Carriage is driving in
      -and-
    - The Rod Down switch is made
- Provides the following logic interlocks
    - Prevents rod insertion if:
        - Carriage height is 0 0
          -and-
        - Carriage Down switch is made
    - Prevents rod withdrawal if rod height is 100 0
- Provides signal for pulsing

For steady-state reactor operations, the electromechanical portion of the transient rod drive controls the transient rod position. The pneumatic cylinder must be in the fully inserted position in order to apply air to the piston for steady-state operation of the rod. Once air is applied, the pneumatic cylinder movement controls the transient rod at a rate of approximately 11.3 centimeters per minute.

Interlocks associated with the *Transient Control Rods* are as follows:

- Rod Down
    - Prevent driving carriage down when:
        - Carriage height indication at 0 0%
          -and-
        - Rod Carriage is Down
- Rod Out Interlock
    - Prevents rod withdrawal if rod height is 100.0%
- Air Applied interlock
    - Allows applying air when:
        - Mode Selector switch in Pulse
          -and-

- Power is less than 1kW

-or-

- Mode Selector Switch is in Steady State

-and-

- Carriage is down

-or-

- Air is applied (Having air applied satisfies the electronic logic interlock and allows air to continue to be applied when rod is withdrawn in steady state. If air is vented, air cannot be re-applied until the carriage is down)

- TR Withdrawal
  - Prevents withdrawing Control Rods in the Pulse Mode
- Low Count Interlock
  - Prevents withdrawing Control Rods with < 2cps on the Log Power Channel

The Control Rod Drive Mechanisms (CRDM) (Figure 7-8) is an electromechanical assembly that provides the motive force to move the Transient Rod and hold it in position The motor provides the movement, while high-pressure air holds the rod

The Transient Rod CRDM has a Carriage Down and Rod Down switch. These physical switches perform the following and have the following physical descriptions:

- Carriage Down Switch
  - Functions
    - Provides input to rod position reset circuit to reset indication at 1 0% when
      - The carriage presses the Carriage Down switch

        -while-
      - Rod is driving in

        -and-
      - Rod is down.
    - Provides input to Rod Down Interlock (prevents further insertion if rod at 0.0% and carriage is down)
  - Description
    - Micro switch activated by carriage
- Rod Down Switch
  - Functions
    - Lights Rod Down light when control rod is <5% withdrawn
    - Provides input to rod position reset circuit
  - Description
    - Micro switch activated by the piston rod

7.3.1.3 Regulating Rod Control

The rod control system for the Regulating control rod (Reg Rod) allows the operators to control the rod manually or automatically via the servo controller. The Reg rod has a Rod Control Module, Control Rod Drive Mechanism (CRDM), control rod barrel and hold-down tube. In addition, the Reg Rod shares control circuitry for interlocks and a Power Supply for rod motion with other control Rods.

The regulating rod control assembly is similar to the Shim Safety control rods in Figure 7-7 except that the barrel contains a lower guide piece with no piston action since the control rod extension bolts to the lead screw and does not scram

The control rods attach to a horizontal plate on the upper portion of the reactor frame structure with machined slots and clamps to hold the rod drives in position (similar to the Shim Safety Control Rods). A support ring holds the Control Rod assembly. This assembly permits removal of the associated control drives for maintenance without moving the associated control rod from the core; however, since the lead screw for the Reg Rod physically attaches to the connector rod, the CRDM must be disassembled

The Reg Rod CRDM connects to a Rod Control Module at the reactor console Push buttons permit operation of the rod drive.

Travel speed for the Reg Rod is 11.3 centimeters per minute.

To provide rod height indication, the Module receives a signal from a digital encoder that rotates with the lead screw through the stepping motor drive shaft The Module uses this signal to provide Carriage Height indication in units of percent withdrawn and to provide logic interlocks for carriage full out (100%) and carriage full in (0%) Rod height and carriage height are the same

The *Rod Control Module* for the Regulating Rod performs the following *functions.*

- Provides digital indication of carriage height
- Provides Rod In/Rod Out signal to CRDM
- Provides indication of the following
  - Carriage <20%
  - Carriage >80%
  - Rod Jammed
  - Carriage Down
  - Carriage Up
- Resets the rod position indication to 1.0% when:
  - Carriage is driving in
    -and-
  - The Rod Down switch is made
- Provides the following logic interlocks
  - Prevents rod insertion for jammed rod
  - Prevents rod insertion if:
    - Carriage height is 0.0
      -and-
    - Rod Down Switch is made
  - Prevents rod withdrawal if rod height is 100.0
- Provides signal for Shimming Required alarm
  - Alarms when Red Rod is <20% for >80% fully withdrawn

Interlocks associated with the *Regulating Rod* are as follows:

- Rod Jammed
  - Prevents driving carriage down when lead screw presses the Jam switch
- Rod Down
  - Prevent driving carriage down when:
    - Carriage height indication at 0.0%
      -and-
    - Rod Down switch is made
- Rod Out Interlock
  - Prevents rod withdrawal if rod height is 100.0%
- Shim Safety Pulse Interlock
  - Prevents withdrawing Control Rods in the Pulse Mode
- Low Count Interlock
  - Prevents withdrawing Control Rods with < 4mW on the Log Power Channel

The Control Rod Drive Mechanism (CRDM's) for the Regulating Rod operates exactly like the Shim Safety Control Rod Drive Mechanisms except that there is no electromagnet.

The Reg Rod CRDM has a Jam and Rod Down switch. These physical switches perform the following and have the following physical descriptions·

- Rod Down Switch
  - Functions

        - Provides input to rod position reset circuit
    - Description
        - Micro switch activated by carriage
- Rod Jammed
    - Functions
        - Lights Rod Jammed light
        - Provides input to rod jammed interlock, which prevents rod insertion if rod is jammed
    - Description
        - A micro-switch pressed when the lead screw drives in without the carriage lowering

7.3.1.4 Mode Selector Switch

The Mode Selector Switch selects between Steady State and Pulse modes It ensures the appropriate interlocks for both pulse and steady state and prevents pulsing in Steady State Mode.

The Mode Selector Switch has two positions that provide the following functions·

- Steady State
    - Prevents applying Transient Rod air unless the following conditions are met·
        - Transient Rod Carriage Down

          -or-
        - Transient Rod Air applied (Having air applied satisfies the electronic logic interlock and allows air to continue to be applied when rod is withdrawn in steady state. If air is vented, air cannot be re-applied until the carriage is down)
- Pulse
    - Allows Transient Rod Air applied
    - Activates Preset Timer
        - Preset Timer scrams the Transient Rod less than 15 seconds after applying air
    - Prevents control rod withdrawal
    - Disables Safety Drawer amplifiers
    - Bypasses Period Scram

7.4 Reactor Protection System

Table 7-1 indicates the minimum reactor safety circuits and interlocks that are necessary for reactor operation. Failure to comply with any of the safety criteria will result in an immediate reactor scram

Table 7-1: Minimum Reactor Safety Channels

Safety Channel	Function	Number Operable	Effective Mode	
			Steady-State	Pulse
Fuel Element Temperature	SCRAM @ LSSS	1		X
Safety Power	SCRAM @ 125%	2	X	
	SCRAM on loss of supply voltage to detector power supply	2	X	
Console SCRAM Button	SCRAM	1	X	X
Preset Timer	Transient rod SCRAM less than 15seconds after pulse	1		X
Log Power	Prevent shim-safety withdrawal at less than 4×10^{-3} W	1	X	
Transient Rod Air Apply	Prevent application of air unless fully inserted	1	X	
Shim-safety & Regulating Rod Pulse Interlock	Prevent withdrawal in Pulse Mode	1		X

7.5 Engineered Safety Features Actuation System

There are no engineered safety features actuation systems

7.6 Control Console and Display Instruments

The NSCR operates in two standard modes: Steady State and pulse. Steady-state mode is for operation at power levels up to 1000 kW (thermal). Pulsed mode is for the condition resulting from the rapid withdrawal of the transient rod, which introduces a step insertion of reactivity that results in peak powers of up to about 1,600,000 kW. The reactor console displays all pertinent reactor-operating conditions and allows for reactor control. The console also displays information about the cooling system, environmental monitoring and experimental facilities. The control system consists of five power measuring channels utilizing three uncompensated ion chambers, one compensated ion chamber and one fission counter. Descriptions for the specific controls for the reactor and associated water systems are in those sections

At all times when the console is turned on, a licensed reactor operator or licensed senior reactor operator will be in the control room. Reactor operators in training may operate the reactor in the presence of a licensed reactor operator or licensed senior reactor operator in the control room All fuel additions to the reactor core and critical experiments require the presence of a member of management as designated by the director.

Table 7-2 lists indications and controls on the reactor console,

Table 7-3 lists alarms displayed on the main reactor console.

Table 7-2: Summary of Information Displayed and Recorded on Reactor Console

Reactor Safety Systems	Control	Indication	Record
Log Power: Power Indication		X	X
Log Power: Period Indication		X	
Linear Power		X	X
Safety Amplifier		X	
Pulse Power (Integrated)		X	
Fuel Temperature		X	X
Rod Drives	X	X	
Manual SCRAM	X	X	
Other SCRAMs		X	
Facility & Reactor Conditional Alarms		X	
Water Systems			
Pool Water Cooling System	X	X	
Pool Recirculation System	X	X	
Pool Skimmer System	X	X	
Diffuser System	X	X	
Transfer System	X	X	
Secondary Treatment System	X	X	
Personnel Control & Radiation Protection			
Area Radiation Monitors		X	
Facility Air Monitors		X	X
Air Handling System Shutdown	X	X	
Emergency Evacuation Horn	X	X	
Irradiation Cell Exhaust	X		
Television Monitors		X	
Facility "Door Open" Alarms		X	
Experimental Facilities			
Pneumatic System	X	X	
Sample Rotisserie Motor	X	X	
"C-2" Experiment Personnel Control Alarm	X	X	

Table 7-3: Summary of Alarms Displayed on Reactor Console

Alarms
Bridge Unlocked
Fuel Temperature Scram
Period Scram
Safety Amplifier Scram
Manual Scram
Experiment Scram
Manual Scram
Servo Fault
Period Scram Bypass
Bridge Interlock
Air Handler shutdown Bypass
Regulating Rod Shimming Required
Area Radiation alarm
Facility Air Monitoring
Emergency shutdown air Handling System
Building Pressure System Failure
Cell Door Open
Pool Level Alarm

7.7 Radiation Monitoring Systems

Two systems for monitoring radiation in the facility are Area Radiation Monitors (ARM) and Facility Air Monitors (FAM)

7.7.1 Area Radiation Monitors (ARM's)

Area Radiation Monitors are located throughout the facility to monitor levels in areas where radiation levels could exceed normal levels One ARM above the reactor provides radiation levels in the reactor bay area. Other ARM's are by the beam ports accesses, demineralizer room and radioactive sample handing areas.

The ARM's provide audible and visual for Alert and Alarm. Operators can adjust these alarm settings in the Control Room The indicators are in the Reception Room (Emergency Support Center), Control Room and locally for each ARM.

7.7.2 Facility Air Monitors

Six Facility Air Monitors (FAM's) detect airborne activity in both gaseous and particle form. They monitor air in the building and leaving the building. Following is a list of the detectors by channel including their functions and sample points.

1) Channel 1 – Stack Particulate
   a Monitors for radioactive particles in the air entering the exhaust stack
   b. Automatically shuts down the air handling system
2) Channel 2 – Fission Product
   a. Monitors for radioactive particulate above the reactor core
   b. Automatically shuts down air handling system
3) Channel 3 – Stack Gas
   a Monitors for Ar-41 entering the exhaust stack

4) Channel 4 – Building Particulate
    a Monitors for radioactive particles in the confinement building
5) Channel 5 – Xenon Monitor (Note Shares a detector with Channel 3)
    a. Monitors for Xe-125 entering the exhaust stack
    b Automatically shuts down air handling system
6) Channel 6 – Building Gas
    a. Monitors for Ar-41 in the confinement building

Each FAM channel provides indication in the FAM Equipment Room, the Control Room and in the Reception room Each FAM channel also provides an audible Facility Air Monitoring alarm in the control room and an alarm light in the Reception Room. The FAM channels that shut down the air handlers also provide an Emergency Shutdown Air Handling System alarm in the control room.

---

[1] T.A. Godsey and J.D. Randall, "A Solution to the Varying Response of the Linear Power Monitor Induced by Xenon Poisoning," Presented at TRIGA Owners Conference III, Albuquerque, NM, 1974

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

No electrical power supplies are critical for maintaining the facility in a safe shutdown condition

Figure 8-1 shows the electrical distribution system for the Nuclear Science Center. Texas A&M plant services supplies electrical service to the facility from the distribution system through power poles on the NSC site Emergency disconnects are in place at the transformer stations on the NSC site

480 VAC 3 Phase Electrical Power

Power panels MCC'MA' and MCC'MB', in the mechanical equipment building, supply the majority of the loads in the reactor and laboratory buildings MCC'PA' is located in the heat exchanger room in the lower research level and supplies power to the reactor cooling system equipment motors MCC'RA' is located on the chase level of the confinement building and supplies the majority of the loads on the chase.

120/208 VAC Electrical Power

Various distribution panels receive 450 VAC for loads in the facility. Reactor building panels, labeled RB, are inside the confinement building. RB'A' and RB'B' are located in the electrical shop. RB'C' is on the chase level. RB'B' supplies RB'D' which is in the control room on the wall behind the main control panel.

LB panels LB'A' and LB'B', located in laboratory seven Laboratory building, supply loads for the laboratory area

8.2 Emergency Electrical Power Systems

Rechargeable, battery-operated emergency floodlights are located throughout the building. In the event of a power failure, these lights, which are normally off, provide sufficient lighting to permit evacuation of the reactor building or the performance of emergency activities in the building

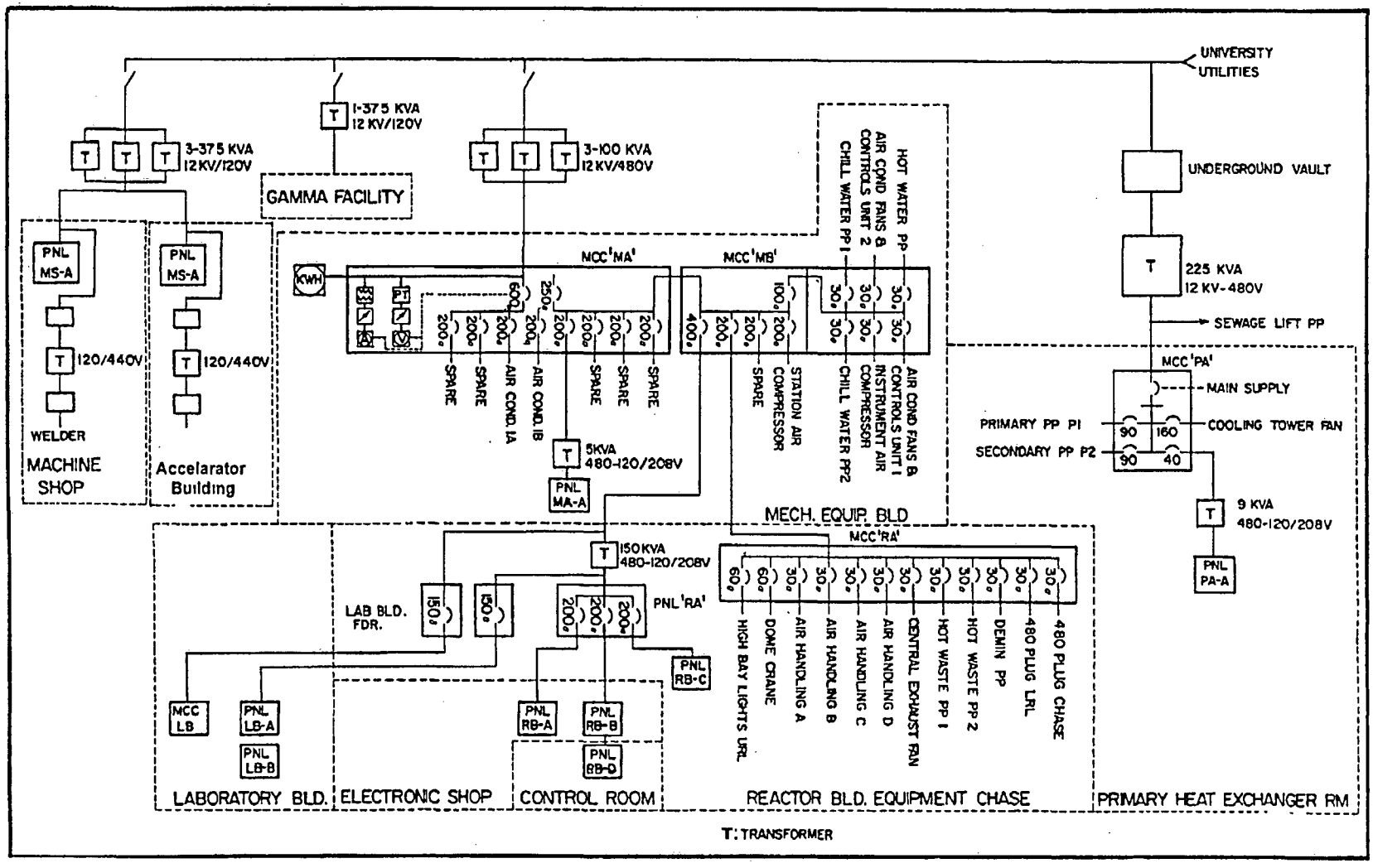


Figure 8-1: NSCR Electrical distribution

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning Systems

9.1.1 Heating and Air Conditioning

Chillers and a furnace provide chilled water and hot water for the ventilation system. As air enters the building, it flows through a heat exchanger with both hot and cold water from the furnace and chiller. This heating and cooling system removes humidity from the air and maintains the building at a comfortable temperature

9.1.2 Air Handling Units

Four air handling units and an exhaust fan control airflow, pressure, temperature and humidity within the reactor building. The facility has three zones of negative pressure for effective isolation of possible airborne radioactive material. Following is a list of the three zones and the areas they cover.

1) Least negative pressure zone
    a Control Room and locker areas where contamination is least likely
2) Intermediate zone of negative pressure
    a The main research areas where infrequent contamination might occur
3) Maximum negative pressure includes areas where radioactive contamination is likely
    a Beam ports
    b Thermal column
    c. Through tubes

The Air Handling units supply air to the following areas of the building:

1) Air handler A
    a Upper research level
2) Air Handler B
    a. Lower research level
3) Air Handler C
    a. Control room
4) Air Handler D
    a Restrooms
    b Electronics shop

The Central Exhaust Fan takes suction on all areas and discharges directly to the stack or through an Emergency Filter Bank. The height of the exhaust stack above ground level is 85 feet.

A Bypass Damper, at the suction of the Central Exhaust Fan, controls pressure in the building

Controls for all four units are in the control panel in the reception room. An interlock prevents running units A, B, C or D unless the Central Exhaust fan is running. This ensures the building will not be at a positive pressure.

The Central Exhaust Fan will shut down and the inlet and exhaust dampers close when:

- The FAM's generate an Emergency Shutdown signal
- The Air Handling Shutdown button is pressed on the Reactor Console
- The High Temperature Sensor above the Emergency Filter Bank exceeds its set point

When the Exhaust Fan shuts down, the rest of the air handlers also shut down

9.1.3 Dampers and Filters

Dampers are located at the air inlet to all air-handling units, the fresh air bypass to the exhaust fan. and in the exhaust stack. In cases of emergency, a switch in the reactor control room can close these dampers and simultaneously secure the air handlers to isolate the building and stop airflow (see 9 1.2). An Emergency Exhaust Air Filter Bank is between the exhaust fan and building stack. The Emergency Filter Bank consists of two particulate filter banks and one bank of activated carbon filters Controls for the filter bank are in the Reception Room

9.1.4 Emergency Operation

The Emergency Control Panel in the Reception Room provides all the controls for operating the ventilation system for both emergency and normal operations

9.2 Handling and Storage of Reactor Fuel

Technical specifications require stored fuel to be in a configuration with k_{eff} to be less than 0 8 for all conditions of moderation. The storage arrays for irradiated fuel permit sufficient natural convection cooling by water or air such that the fuel elements or fueled device temperature will not exceed design values.

Fuel elements are stored and handled in [redacted] areas at the NSC. These areas are the [redacted] torage room and the reacto[redacted] areas Unirradiated fuel can be temporarily stored in approved shipping containers used by the fuel manufacturers for shipment in the reactor.

9.2.1 Fuel Handling

Two bundle handling tools (one rigid, one flexible) provide a means for moving three- and four-element bundles. Each of these tools has a 'C' hook with a locking mechanism. Once the 'C' is around the Top Handle, a locking mechanism prevents the bundle from slipping out. The operator controls the locking mechanism via a handle at the top of the tool. Each of these has an additional control point in about the middle of the length, which allows the operators to control the locking mechanism when putting the fuel in a wall storage rack.

An Individual Fuel Element Tool allows operators to move individual elements. This General Atomics tool locks onto the top fitting of an individual fuel element. An operator can release the element using the handle at the top of the tool. The tool will only release the element when the handle is in the Open position and there is no weight on the tool. This prevents the accidental release of a fuel element when the fuel element tool is supporting it

9.2.2 Fuel Storage

9.2.2.1 Fuel Storage Room or Fuel Vault

The fuel storage room (Fuel Vault) [redacted] provides a storage location for unirradiated fuel. Elements stored in this room are in an environment of dry, cool air and are under lock and key and intrusion security. Fuel bundles (four elements maximum per bundle) and individual elements are in cadmium lined aluminum tubes secured to an aluminum frame mounted to the concrete walls of the fuel storage room.

9.2.2.2 Reactor Pool Storage Areas

Both irradiated and unirradiated fuel elements are stored under water in racks mounted on the reactor pool walls or in a storage rack on the floor of the reactor pool. Wall storage racks have aluminum tube positions for six fuel bundles and twelve individual elements. The fuel element rack positioned on the floor of the pool is constructed of aluminum angle and tubing and has facilities for twenty-four fuel bundles. This facility has a large aluminum lid that covers the tubes and protects stored fuel.

Special storage facilities provide for instrumented fuel and fuel-followed control-rod storage.

9.2.3 Fuel Bundle Maintenance and Measurements

The Maintenance Jig supports the entire fuel bundle and prevents any individual element from falling when it is unscrewed from the lower guide of the bundle This jig provides access to the lower end of the fuel elements This allows operators to completely disassemble and reassemble a bundle in the jig.

Once an element is unscrewed from the lower guide, the single element tool holds the fuel element for visual inspection.

The Fuel Measuring Device provides a method of a go/no-go test for transverse bend and length measurement for element elongation. An element will not fit into the device if the transverse bend exceeds 0.125 inches over the length of the cladding. The device holds one fuel element and allows operators to measure the length difference between a given fuel element and the standard. The difference between these two changes over the life of the element and provides the elongation information

9.3 Fire Protections Systems and Programs

Smoke detectors that alarm off-site and numerous fire extinguishers throughout the facility provide fire protection at the Nuclear Science Center Additionally, the College Station Fire Department provides the NSC with fire protection services and is on call twenty-four hours a day. Fire department personnel receive training in radiological hazards and NSC site familiarization.

9.4 Communication Systems

The NSC is equipped with several commercial telephone lines, all of which are available in the Control Room, Emergency Control Center (Reception Room) and several other locations within and outside the Reactor Building. The system allows public address from any telephone.

Two-way radios provide additional communication between the NSC and the 24-hr staffed Communications Center at Texas A&M.

Finally, the Communications Center maintains an emergency recall roster listing home phone and pager numbers for key personnel

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

The NSC receives, possesses and uses, in amounts as required, any byproduct material without restriction to chemical or physical form that has a definite research, development or education purpose. It may also have any byproduct material generated by the licensed activities, but may not separate such fueled byproduct material.

All activities covered by the NSC license take place on the NSC site and adjacent NSC controlled facilities.

9.6 Cover Gas Control in Closed Primary Coolant Systems

The NSCR does not have a closed primary loop; ordinary light water at atmospheric pressure in an open pool is the primary coolant. Therefore, no cover gas control is necessary.

9.7 Other Auxiliary Systems

There are no other auxiliary systems required for safe reactor operation

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

10.2 Experimental Facilities

10.2.1 Beam Ports

Five permanent beam ports of Type 304 stainless steel are cast into the pool wall at the lower research level. Beam port 5 is in the north wall of the main pool (Figure 10-1); the other four beam ports are in the stall end of the pool. The thermal column modification accommodates a film irradiation system. The film irradiation system displaces beam ports 6, 7 and 8 in Figure 10-2. Figure 10-2 shows the stall section of the pool with an unmodified thermal column Normally, Beam Ports 2 and 3 are flanged at the inner pool wall See the Film Irradiation section for specifics on the thermal column Figure 10-3 shows the arrangement with the Thermal Column extension and Graphite Coupler Box.

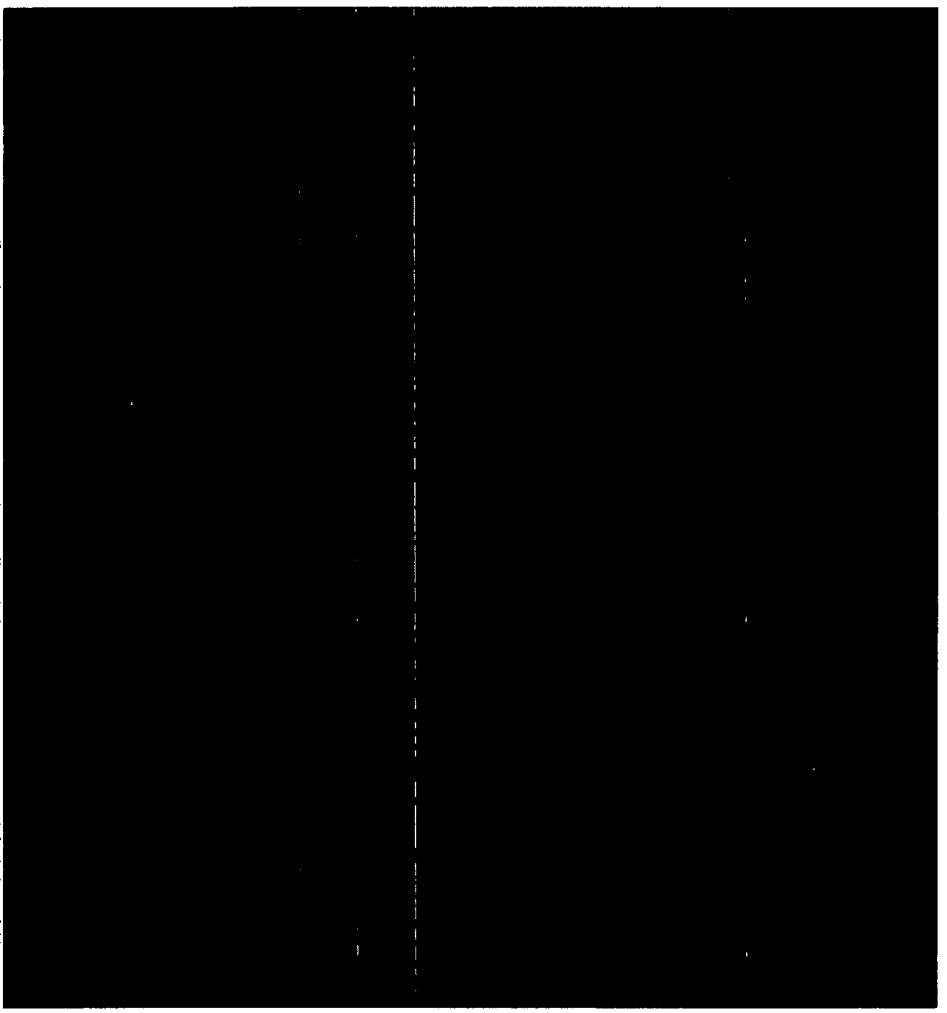


Figure 10-1: Pool Experiment Facilities

Figure 10-2: Stall Beam Port Installations with Bismuth Trough

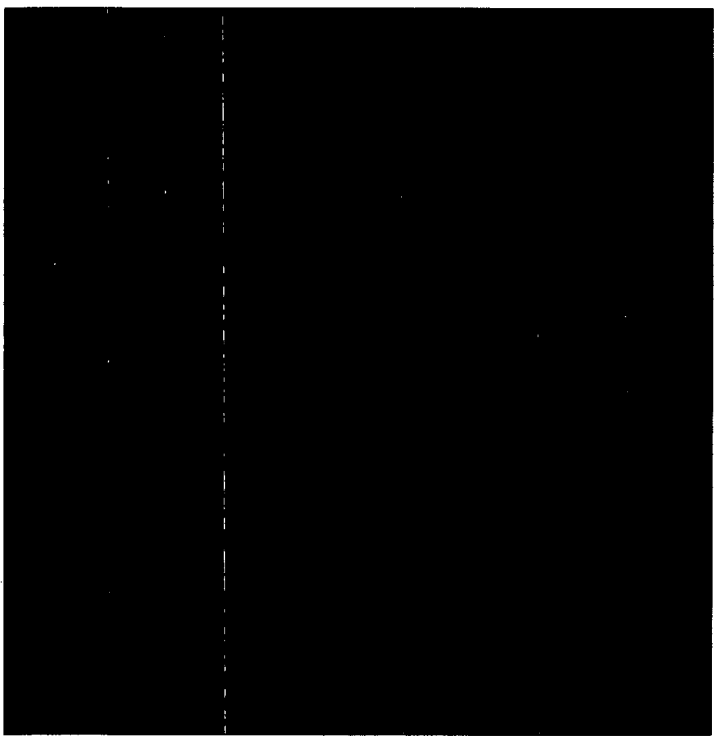


Figure 10-3: Stall Beam Port Installations with Graphite Coupler Box and Thermal Column Extension

The beam ports are stainless steel in sections of six, ten, and nineteen inches in diameter divided longitudinally into three, two and one-half, and one foot segments, respectively This design prevents neutron streaming when concrete shield plugs are in place. The six and ten inch sections have one-quarter inch Boral lining with the exception of the six-inch section of beam port 4 Each of the above ports ends flush with the external face of the pool wall and is sealed by a hinged two-foot square, four-inch thick, carbon steel clad lead door. The doors are equipped with an O-ring seal and tightening lug to provide a water barrier in the event of port flooding. A micro switch actuates an enunciator light on the console when these doors open.

Beam port plugs are aluminum cylinders filled with barites concrete, each about one foot long with a handle recessed in the exposed end for ease of handling Three of these plugs can fit into the six-inch diameter beam port-section and two can fit into the ten-inch diameter section when the port is not in use. A nineteen-inch diameter, one-foot thick section is available to plug the final recessed section of the beam port.

Each beam port has a two-inch diameter pipe connecting it to the central exhaust system, which maintains a constant negative pressure in the tube The vent connection to the tube is nearer the inner pool wall to ensure the removal of any gases before they can reach the external end of the tube

These beam ports enable a variety of experiments such as the extraction of a well-collimated beam of neutrons and/or gamma rays from the reactor. Varieties of extensions can attach to the beam ports as required by different experiments The extensions prevent interference with the movement of the reactor frame and grid plate when the graphite coupler box is not in place. A short extension suspended between the tips of beam ports 1 and 4 and the graphite coupler box removes water from between the coupler box and the beam port. Beam ports 2 and 3 are radial ports with extensions removed A "C-2" alarm device indicates to the Control Room when personnel enter into the beam port areas A system to magnetically lock doors to the lower research level is available for activation during beam port usage

Two separated segments of a single through tube, constructed of 304 stainless steel, penetrate the stall section of the pool. Their construction is essentially identical to that of the beam ports except that they have no boral liners or outer doors. Since the tubes sit along a collinear axis, a straight six-inch diameter connecting tube can be bolted to the flanged pool ends of the tubes providing a continuous six-inch diameter passage completely through the pool Concrete plugs similar to those described above provide the necessary shielding in this tube to prevent streaming of radiation The through tube also vents to the central exhaust system.

The through tubes can facilitate transit experiments that pass through this tube or fixed experiments Each segment may be used as a separate beam port by fitting an extension tube between the reactor and the end of the through tube segment

10.2.2 Thermal Column

The stainless steel and aluminum thermal column is located in the east end of the stall portion of the pool. It consists of a three and one-half-foot square section on the inside of the pool that enlarges to a four-foot square opening on the experimenter's side that penetrates the pool wall at core level. The walls of the thermal column are welded to the stainless steel pool liner. An aluminum cover plate with gasket, bolted to the inside flange of the cavity, provides the water seal.

A graphite coupler box, adjacent to the thermal column couples the thermal neutron flux from the reactor. The reactor can operate with its east face adjacent to the coupler box for maximum thermal neutron density to the thermal column and the beam ports.

A vent line from the thermal column cavity extends directly to the central exhaust system, where the air goes through the Facility Air Monitoring system before leaving the building. A movable thermal column door, constructed of lead and concrete shielding material, is on tracks embedded in the lower research level floor.

The current thermal column arrangement makes it a film irradiation facility. The film irradiation system extends into the thermal column penetration into the pool so that film comes close to the graphite coupler box.

10.2.3 Pneumatic System

The NSC pneumatic system consists of an electronic control, experiment receivers, in-core receivers, a gas supply system and interconnecting tubing The control system allows the experimenter to control the length of irradiation and the control room operator to provide or prevent permit to the experimenter. The experiment receivers provide a means for the experimenter to load and retrieve the samples The in-core receivers receive and support the sample in the core. The gas supply system provides the pressure to 'shoot' the sample into the core and return the sample from the core

The pneumatic tube itself consists of a core receiver, polyethylene tubing, protective metal sheathing at the reactor bridge and a receiver in any of several laboratories (Figure 10-4) The pool wall pneumatic penetrations in Figure 10-1 are unused because of inconvenience in maintaining the system within the reactor pool. At present, the pneumatic system lines enter the pool at the reactor bridge and pass over the top of the pool walls.

The pneumatic tubes are for the production of short-lived radioisotopes primarily to support neutron activation analysis.

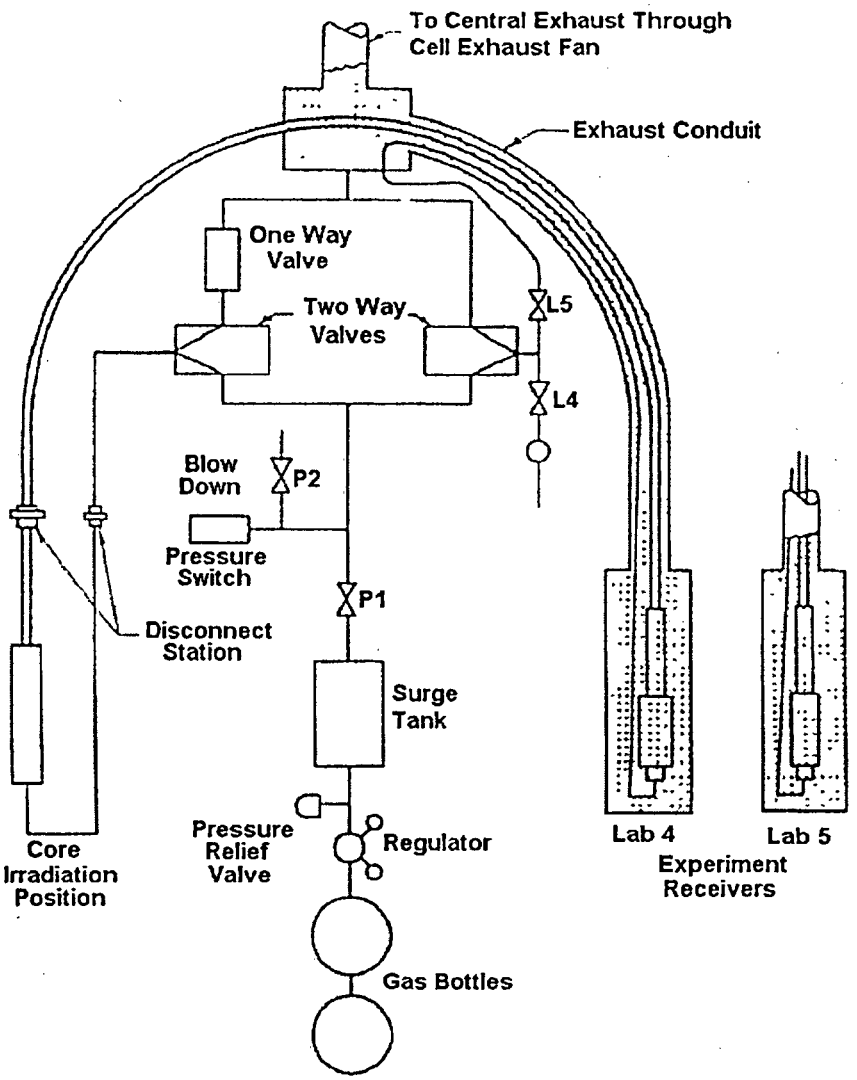


Figure 10-4: Pneumatic System

10.2.4 Irradiation Cell

The irradiation cell is located at the west end of the reactor pool. This cell is approximately eighteen feet wide by sixteen feet deep by ten feet high. The frame for the concrete roof is an eight by eight inch steel I-beam column connected with six by fifteen inch steel I-beam joists. An overlay of four by six inch timbers provide decking for the concrete blocks which are two by two by four feet. The blocks are stacked four feet high with an opening of approximately five by five feet left directly over the cell window. A motor driven concrete shield covers the opening the opening (Figure 10-5). The concrete roof of the lower irradiation cell provides the floor for the upper irradiation cell.

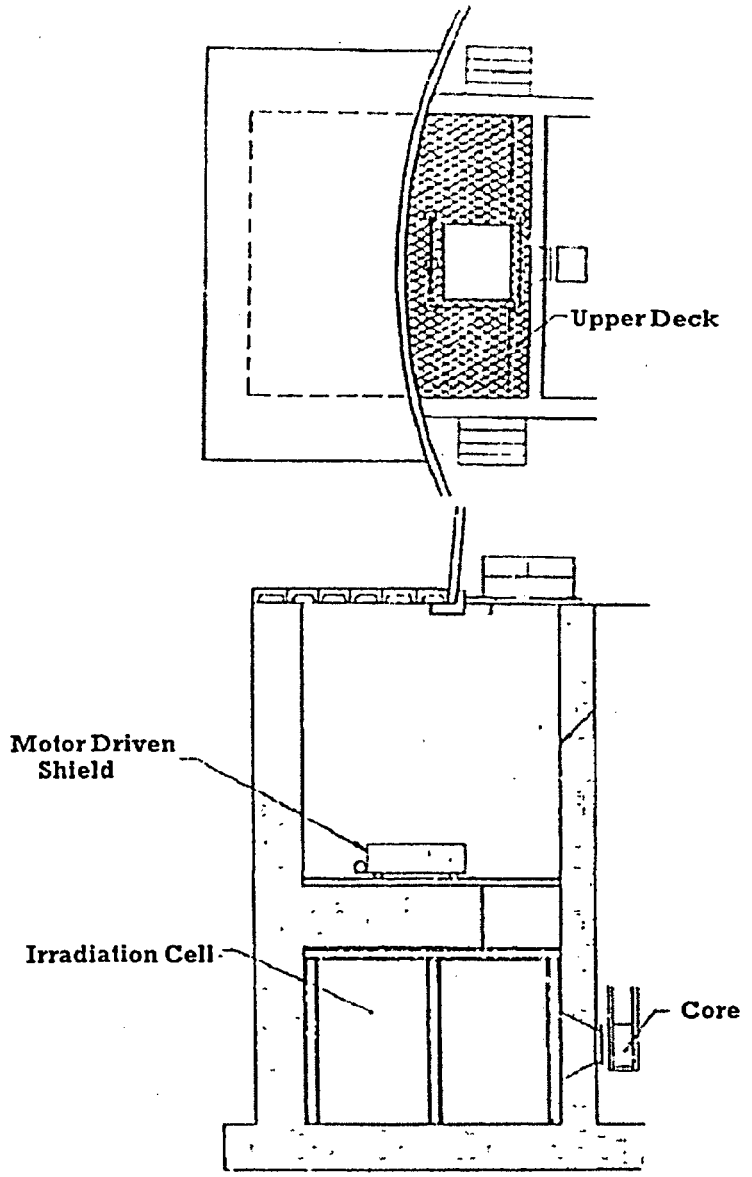


Figure 10-5: Irradiation Cell

A steel ladder that extends from the observation deck to the upper irradiation cell provides access to the upper irradiation cell. An extendable ladder or an electro-mechanical lift provides access to the lower irradiation cell. With the exception of an opening to accommodate a sample platform to carry samples and people to the lower irradiation cell, the observation deck over the cell is steel plate. A small-hinged section of the deck plate provides access to the ladder that runs from the upper level to the top of the concrete shield.

Concrete steps lead up to the observation deck on the south side of the pool. This area provides an excellent vantage point for facility visitors. The irradiation cell window is in the two feet thick wall that separates the cell from the reactor pool. The window is two feet square on the pool side and flares out to four feet square on the cell side. A one-half inch aluminum plate bolts to the pool side of the cell window to provide a watertight barrier. The pool side flange is large enough to prevent the cell window from projecting inside the reactor frame. A boral plate can be hung over the window to shield samples in the irradiation cell from excessive neutron flux; a box can also be hung over the window to accommodate various sources for gamma irradiations.

The breaker that supplies electrical power for the motor driven shield serves as a manual interlock for personnel safety. Locking the breaker open prevents opening or shutting the shield door. Mechanical stops on the rail prevent inadvertent movement of the reactor closer than eight feet away from the irradiation cell window. In addition, a bridge interlock provides a scram in the event the irradiation cell door is open and the reactor is within eight feet of the cell

To handle removal of ^{41}Ar due to activation in the cell, an exhaust duct extends to the bottom of the cell for continuous removal of air from the cell. The duct discharges to the central building exhaust ahead of the stack gas monitor. The Facility Air Monitors monitor the cell air before release to the environment. The controls for the cell air exhaust are located on the reactor console. An experimenter SCRAM button and an intercom are located inside the cell.

10.2.5 Neutron Radiography Cave

The neutron radiography facility is a concrete block structure on the lower research level located adjacent to the pool shield wall. It contains and shields a thermal neutron beam extracted from beam port 4 (Figure 10-6). The cave structure surrounding the beam port provides for remote positioning of samples with the beam port in operation A hydraulic shutter at the beam port exit can shield the neutron beam between exposures. A sample preparation room and a dark room are available for loading and unloading cassettes and film processing.

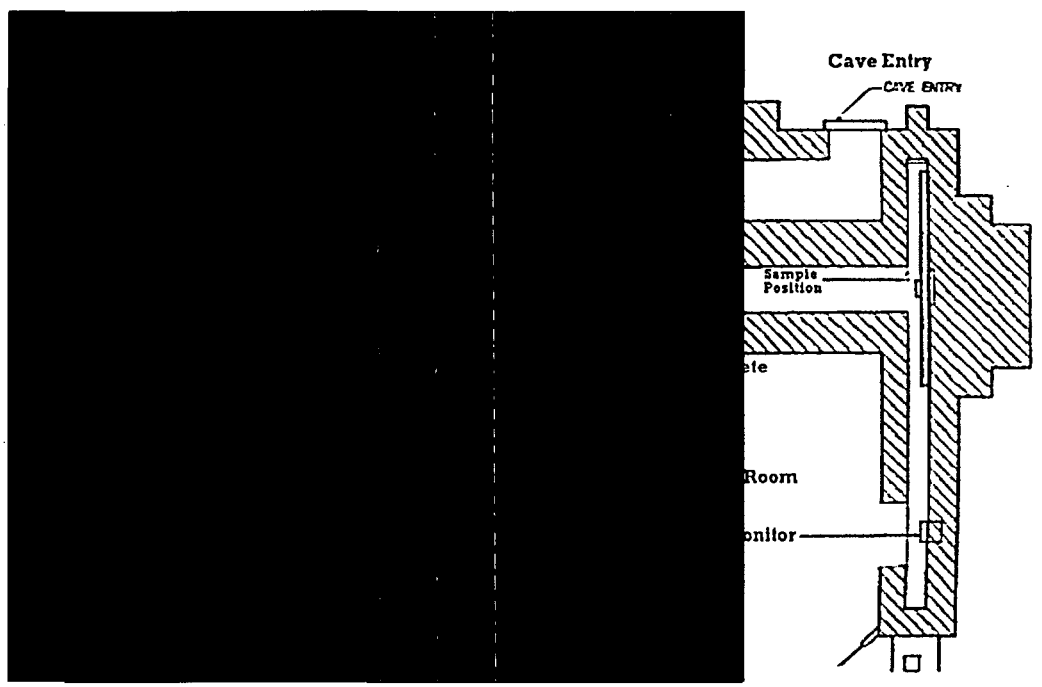


Figure 10-6: BP-4/Radiography Cave

An alarm in the reactor control room will alert the operator upon personnel entry into the sample preparation room film loading access area or the cave and a "C-2" device is visible to the person entering. An entry device on the cave door will cause a SCRAM if the cave door opens when the reactor is against the radiography reflector (graphite coupler box). The bridge rail stop restricts moving the reactor any closer than eighteen inches from the reflector when in place for cave entry.

10.3 Experiment Review

The NSC Standard Operating Procedures (SOP's) give guidelines for review and approval of any new experiment or class of experiments. In addition, the Technical Specifications provide specific review requirements for the Reactor Safety Board

The Senior Reactor Operator on duty can authorize the conduct of routine experiments.

All experiments are subject to the limitations in the Technical Specifications.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

Activities at the Nuclear Science Center will comply with 10 CFR Part 20 "Standards for Protection Against Radiation". The NSC will assess and control exposure of individuals and release of radioactivity to the environment to maintain compliance with all applicable sections of the regulations. The following sections outline methods and instrumentation that establish compliance.

11.1 Radiation Protection

The intent of the radiation protection program at the reactor facility is to maintain radiation exposures as low as reasonably achievable (ALARA) given current technologies The design of the experimental facilities, reactor pool and the reactor shield includes protective measures and devices that limit radiation exposure and radioactive material release. The Standard Operating Procedures govern general requirements such as dosimeter use and records, certification of training, survey frequency, leak testing of sources, and ALARA program

11.1.1 Radiation Sources

11.1.1.1 Airborne Radiation Sources

- Releases from abnormal reactor operations

The fuel retains its fission products, with releases to the environment only during a breach of fuel cladding. This possibility is one of the accidents considered in Chapter 13 of this report in the analysis of the design basis accident.

- Releases form normal reactor operations

Production of radioactive gases, primarily ^{41}Ar, result from the irradiation of air and dissolved gases in the cooling system, open beam port tubes, the dry tube, pneumatic irradiation systems, and the irradiation cell.

Nuclear Science Center Technical Report Number 32[Error! Bookmark not defined.] documents that the NSC releases approximately 4 7 Ci of ^{41}Ar on an annual basis. Applying a dilution factor of 5×10^{-3}, the releases produce approximately 0 8% of the effluent concentration of Argon-41 as specified in 10 CFR Part 20, "Standards for Protection Against Radiation". The result of 4.7 Ci assumed 100 MW-day operation of the NSCR The EPA COMPLY[1] program indicates that the maximally exposed receptor would receive a dose of 0 2 mrem/y for continuous discharge.

There are several important findings concerning ^{41}Ar production at the NSC.

1) On a long-term basis, the pool accounts for more than 95% of the facility's production.

2) The measured peak ^{41}Ar concentration in the Irradiation Cell for individual four-hour runs were:

   a. 6.7×10^{-5} pCi/cc with the cell exhaust fan on and,
   b. 1.2×10^{-4} pCi/cc with the exhaust fan off.

3) The maximum ^{41}Ar release from firing the pneumatics system was 208 pCi; this occurred after the reactor operated at one MW for six hours. Firing the pneumatic system with the reactor shutdown for a long time resulted in a 6.8 pCi ^{41}Ar release. Firing 5 times always purged the pneumatics system of argon.

4) As expected, the dry tube did not contribute to ^{41}Ar release, but the beam port measurements showed a level of 2.15×10^{-3} pCi/cc at 1 MW in beam port 1 closest to the core.

Although the pool is the major production source over a long period, the other sources can rival the pool release rate on occasion

The building central exhaust pulls the ^{41}Ar in the beam ports and in the irradiation cell directly to the exhaust stack. The ^{41}Ar from air in the pool water goes from the pool into the building and then through the building ventilation system to the central exhaust

The central exhaust disposes of the gaseous waste into the environment through the 85-feet high building stack.

The equations for developing the dilution factors below are from F.A. Gifford, Jr. [2,3]. These calculations are based on release at ground level and utilize the building dilution factor ($D_B = cAu$), where A is the cross sectional area of the building normal to the wind and u is wind speed in meters/second The estimated value of c is 0.5. The cross sectional area of the Nuclear Science Center is 357 m^2

The equation for the atmospheric dilution factor is:

$$X = \frac{Q}{\pi\sigma_y\sigma_z u}\exp\left(-\frac{1}{2}\left[\frac{y_2^2}{\sigma_y}+\frac{h_2^2}{\sigma_z}\right]\right), \qquad \textbf{Equation 11-1}$$

where

X is the concentration in grams or curies per cubic meter;
Q is the original source strength in grams or curies per second;
u is the mean wind speed in meters per second;
y is the crosswind in meters from the plume axis;
h is the source height in meters; and
σ_y and σ_z are the dispersion coefficients in m^2.

By combining the building dilution factor, D_B, with the atmospheric dilution factor and in the downwind direction $(y = 0)$, the formula becomes:

$$X = \frac{Q}{(\pi\sigma_y\sigma_z + cA)\cdot u} \qquad \textbf{Equation 11-2}$$

The average wind speed as determined from U S. Weather Bureau data for this location is 10 mph. The following calculation utilizes dispersion coefficients of σ_y and σ_z for stable conditions and a wind speed of 1 m/sec (2 mph) to determine the dilution factor available under pessimistic conditions $(Q = 1)$ at a distance of 100 meters from the point of release.

$$X = \frac{1}{(\pi\sigma_y\sigma_z + cA)\cdot u} \qquad \textbf{Equation 11-3}$$

$$X = \frac{1}{\pi\cdot 4\cdot 2 + 0.5\cdot 357} \qquad \textbf{Equation 11-4}$$

X= 1/203 **Equation 11-5**

This calculation indicates that the minimum dilution at 100 meters is 200 under the most adverse conditions. From the wind rose diagram shown in Figure 2-8, these conditions are indicated approximately 10% of the time; however,

most calm conditions occur at night while the majority of operations occur during the daylight hours Assuming an average wind velocity of ten mph. the dilution factor becomes

$$X = 1/903 \qquad \textbf{Equation 11-6}$$

Again, this is a pessimistic approach since the dilution is at only 100 meters (approximate boundary of the NSC site). The calculation at 1500 meters under stable conditions and with a wind speed of ten mph yields a dilution factor of 6,920 With a wind speed of only 2 mph, the dilution factor is still 1,570

The calculations presented in this section clearly show that the Nuclear Science Center can use a dilution factor of 200 for stack release without endangering the public health and safety.

Section 7.7.2, Facility Air Monitors, lists and summarizes the functions of the instruments that monitor airborne radioactivity in the NSC facility and in the exhaust leaving the facility.

Section 7.7.1, Area Radiation Monitors (ARM's), describes the system for monitoring radiation level in the workspaces at the NSC. .

Section 10.2, Experimental Facilities, describes the Pneumatics System, Irradiation Cell and Beam Ports.

11.1.1.2 Liquid Radioactive Sources

Nitrogen-16 (^{16}N) is the only isotope in the pool from normal reactor operations capable of exposing personnel to hazardous levels of radiation The reactor produces a significant amount of ^{16}N when coolant passes through the core at power levels greater than 400 kW. The diffuser system controls the ^{16}N exposure (section 5 6) and reduces the dose rate at the pool surface to 2 to 3 mrem /h during full power operation. If the diffuser system fails during full power operation, the dose rate at the pool surface is less than 100 mrem/h.

Radiation levels from the liquid radioactive waste are extremely low and do not present radiation exposure hazards. Section 11.2.2 addresses disposal of liquid waste.

11.1.1.3 Solid Radioactive Sources

The major source of radiation and radioactivity is the fission product in the reactor fuel. Typical four-element fuel bundles will generate fields of 100 to more than 1000 R/h in air at 3 ft if removed from the reactor pool. As long as the fuel is contained within the pool filled with water, this source of radiation presents no personnel hazard. Chapter 13 considers a loss of pool water. The pool design makes a complete loss of pool water highly unlikely.

Other possible sources of significant radiation exposure from solid radioactive material are the fission gammas from the operating TRIGA core, samples irradiated for isotopic production, reactor components which have spent a long time in or near the core and the reactor startup source. The non-fuel sources are all small compared to fission product activity in the operating core. Activity produced during irradiations is significant; and the NSC estimates final activities of samples before the irradiation Equipment and procedures are in place to deal with the activity after the irradiation is completed

11.1.2 Radiation Protection Program

All personnel entering the facility will have appropriate personnel monitoring devices. Personnel monitoring devices may include beta-gamma and neutron film badges and pocket ionization chambers.

Protective clothing including coveralls, boots, shoe covers and gloves are available for use at the NSC. Use of protective clothing will be as prescribed by the health physics staff.

At least one shower connected to the "hot" drain on the lower research level provides for decontamination of personnel The laundry also has "hot" drains for cleaning contaminated clothing.

A radioactive material handling area, located adjacent to the reactor on the upper research level, provides a space for processing and packaging radioactive materials. Protective clothing and equipment are available for use in this area. 10 CFR Part 20 governs access control and posting requirements for this area.

A standard radiochemistry laboratory on the lower research level is available for research experiments and health physics use.

Texas A&M University and the Texas Department of Health have agreed to an environmental monitoring program Through this program, NSC staff collects sediment samples from the NSC Creek and milk samples from a downstream Dairy. The NSC analyzes these samples for radioactivity and radioisotope identification. Data from these samples have remained unchanged since 1974. None of the results shows a significant impact on the environment.

NSC staff use portable survey meters to survey operations in restricted areas and during potentially hazardous experimental activities to assure personnel safety and compliance with 10 CFR Part 20 limits and local ALARA limits

The NSC maintains appropriate counting equipment to survey for surface contamination on equipment removed from the building, to determine extent of contamination in the event of a radioactive spill, to conduct a routine radiological safety surveillance program, and to conduct analyses of liquid waste and other samples

11.1.3 ALARA Program

The NSCR standard operating procedures include an ALARA plan and procedures that reflect the management's commitment to ALARA principles. A Reactor Safety Board member, Ex-officio member or designee conducts an annual ALARA review and the Board reviews the results

11.1.4 Radiation Monitoring and Surveying

NSC standard operating procedures dictate the requirements for periodic radiation and contamination surveys. Section 7.7 of this report describes installed radiation detectors and facility air monitors. In general, NSC Health Physics conducts regular surveys for both radiation levels and contamination, and the procedures and forms set requirements for radiation monitors when removing samples from irradiation devices.

11.1.5 Radiation Exposure Control and Dosimetry

NSCR standard operating procedures specify requirements on radiation control and dosimetry. NSC Health Physics staff administers the dosimetry program. TLD, film-badge or equivalent dosimeters detect exposures for operating personnel and students using the NSC on a regular basis, while pocket ion chambers monitor exposures for tour groups and visitors

All Radiation, High Radiation and Very High Radiation areas are subject to the requirements of 10CFR20.

Visitors and tour groups receive very low radiation doses; and tour groups do not have access in any area with dose rate exceeding 2 mrem/hr. No student has ever received a measurable exposure from reactor operation. Occupational exposures of operations and maintenance personnel have been low, seldom exceeding 1 Rem TEDE in a year.

11.1.6 Contamination Control

NSCR standard operating procedures specify requirements on Contamination Control. NSC Health Physics monitor for radioactive contamination as noted in section 11.1.4.

11.1.7 Environmental Monitoring

Environmental TLD on the fence surrounding the NSC provide indication of environmental radiation levels.

11.2 Radioactive Waste Management

NSCR standard operating procedures specify requirements for dealing with radioactive waste

11.2.1 Radioactive Waste Control

Liquid waste from radioactive laboratory floor drains, laboratory sinks, decontamination showers, demineralizer regeneration collects in a sump in the Demineralizer Room Sump pumps transfer the waste into one of three 12,000-gallon liquid waste hold up tanks.

11.2.2 Release of Radioactive Waste

A normal operation associated with the Nuclear Science Center Reactor generates solid radioactive waste in the form of gloves, paper towels, used laboratory equipment, sample containers, aluminum, and used experimental hardware.

The NSC accumulates Low-level solid waste in plastic-lined waste containers located at strategic points throughout the facility. When filled, these plastic containers remain sealed in the radioactive waste storage building (Figure 2-2· NSC Site) Short-lived radioactive waste decays and ends up as non-radioactive waste in a local landfill Long-lived waste stays at the NSC awaiting final disposal.

Activated equipment normally stays in the high-level waste storage area adjacent to the outside wall of the irradiation cell If equipment is not reusable, it becomes either short-lived or long-lived waste.

Low-level liquid waste originates from four primary sources at the Nuclear Science Center. These sources are: floor drains, laundry, showers, and laboratories on the lower research level; the demineralizer room filter and ion bed; condensate from air handling units on mechanical chase; and the valve pit sump in cooling equipment room.

Liquid waste flows through common headers to a liquid waste sump located below the grade of the lower research level. A sump pump transfers waste to one of three storage tanks located above grade 200 feet northwest of the building. These tanks have a total storage capacity of 34,000 gallons. Each tank is equipped with an inlet valve, outlet valve, volume indicator and sampling line. There is a locked valve on the master outflow line. Fresh water is available to the master outflow line for dilution.

The NSC discharges liquid waste from the hold up tanks to the environment as approved by the Texas Natural Resource Conservation Commission (TNRCC) discharge permit.

Liquid waste from the reactor building drains to the hot waste sump in the demineralizer room. Two 100-gpm-sump pumps lift the liquid waste for storage in collection tanks located on the northwest corner of the reactor site. The sump pump is below the base elevation of the reactor pool. Liquid waste from the pool liner and cooling equipment room drains to the valve pit sump. The valve-pit sump pump transfers the water to the demineralizer sump. Figure 5-5 shows the liquid waste disposal system.

11.1 Bibliography

---

[1] U.S. Environmental Protection Agency, COMPLY Program Rev. 2, October 1989
[2] F.A. Gifford, Jr., Nuclear Safety, December 1960
[3] F.A Gifford, Jr., Nuclear Safety, July 1961

12 CONDUCT OF OPERATIONS

All operations involving the reactor will be conducted in compliance with the regulations specified in 10 CFR Part 50 and 10 CFR Part 55. The reactor will be operated within the limits of the license and technical specifications.

12.1 Organization

The Nuclear Science Center is operated by the Texas Engineering Experiment Station (TEES) The Director of the Nuclear Science Center is responsible to the Director of the TEES for the administration and the proper and safe operation of the facility. Figure 10-1 shows the administration chart for the Nuclear Science Center.

The Reactor Safety Board advises the Director of the NSC on all matters or policy pertaining to safety.

The NSC Radiological Safety Officer provides "onsite" advice concerning personnel and radiological safety and provides technical assistance and review in the area of radiation protection

12.1.1 Structure

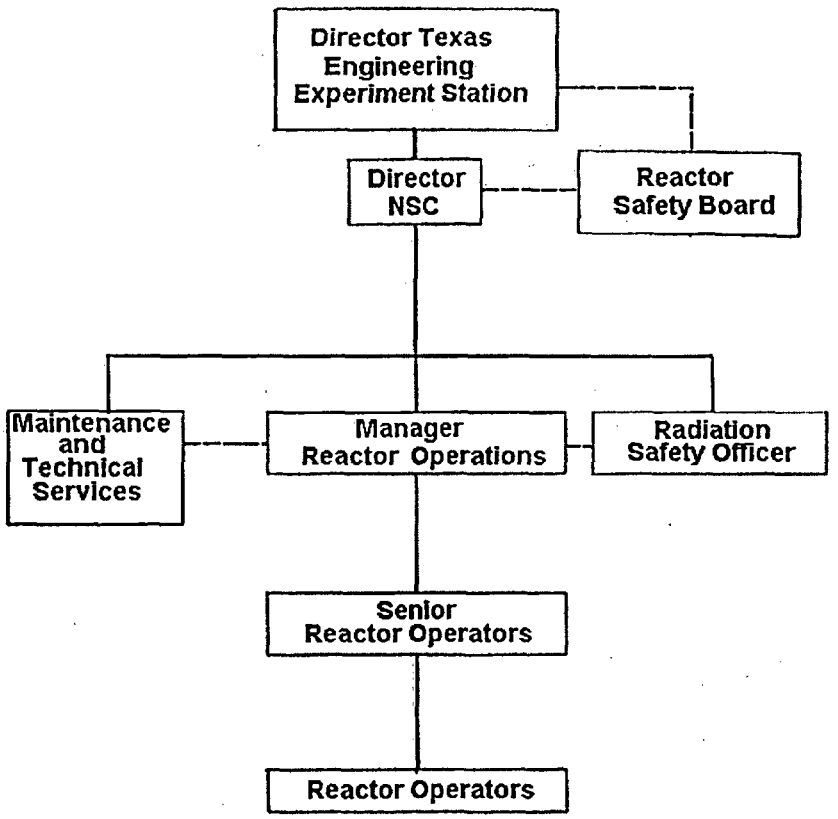


Figure 12-1: Organization Chart for Reactor Administration

A line management organizational structure provides administration and operation of the reactor facility.

The Deputy Director of the Texas Engineering Experiment Station (TEES) and the Director of the Nuclear Science Center (NSC) have line management responsibility for adhering to the terms and conditions of the Nuclear Science Center Reactor (NSCR) license and technical specifications and for safeguarding the public and facility personnel from undue radiation exposure. The facility shall be under the direct control of the Director (NSC) or a licensed senior reactor operator.

12.1.1.1 Management Levels

Level 1· Deputy Director TEES (Licensee) Responsible for the NSCR facility license

Level 2. Director (NSC) Responsible for reactor facility operation and shall report to Level 1.

Level 3: Senior Reactor Operator on Duty: Responsible for the day-to-day operation of the NSCR or shift operation and shall report to Level 2.

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

12.1.1.2 Radiation Safety

A qualified, health physicist has the responsibility for implementation of the radiation protection program at the NSCR. The individual reports to Level 2 management.

12.1.1.3 Reactor Safety Board (RSB)

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

12.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established above.

12.1.3 Staffing

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

1) A licensed reactor operator will be in the Control Room (if senior operator licensed, may also be the senior reactor operator below)
2) A designated senior reactor operator shall be readily available at the facility or on call (i.e., capable of getting to the reactor facility within a reasonable time)
3) A second designated person present at the site able to carry out prescribed written instructions.
4) The Director (NSC) or his designated alternate is readily available for emergencies or on call (i.e , capable of getting to the reactor facility within a reasonable time).
5) At least one member of the health physics support group will be readily available at the facility or on call (i.e., capable of getting to the reactor facility within a reasonable time).

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

1) Administrative personnel
2) Radiation safety personnel
3) Other operations personnel

12.1.3.3 The following designated individuals shall direct the events listed:

1) The Director (NSC) or his designated alternate shall direct any loading of fuel or control rods within the reactor core region.
2) The Director (NSC) or his designated alternate shall direct any loading of an in-core experiment with a reactivity worth greater than one dollar.
3) The senior reactor operator on duty shall direct the recovery from an unplanned or unscheduled shutdown other than a safety limit violation.

12.1.4 Selection and Training of Personnel

A training program for reactor operations personnel exists to prepare personnel for the USNRC Operator or Senior Operator examination This training program normally contains twenty hours of lecture, outside study, and requires several reactor startups

12.1.4.1 The selection and training of operations personnel shall be in accordance with the following:

1) Responsibility:

   a) The Director (NSC) or his designated alternate is responsible for the training and requalification of the facility reactor operators and senior reactor operators

2) Requalification Program

   a) Purpose.
      i) To insure that all operating personnel maintain proficiency at a level equal to or greater than that required for initial licensing

   b) Scope:
      i) Scheduled lectures, written examinations and evaluated console manipulations insure operator proficiency.

12.1.5 Radiation Safety

Members of the health physics staff routinely perform radiation safety aspects of facility operations, including routine surveying for radiation and contamination and sampling water and air. Chapter 11 details the radiation safety program for this license.

12.2 Reactor Safety Board (RSB) Review and Audit Activities

A Reactor Safety Board (RSB) acts as a review panel for new reactor experiments, procedural changes and facility modifications The RSB thus provides an independent audit of the operations of the Nuclear Science Center. Issues concerning nuclear safety are immediately brought to the attention of the RSB. The University Radiological Safety Office provides Health Physics assistance for the Nuclear Science Center. This organizational arrangement thus provides another independent review of reactor operations (Figure 10-1).

12.2.1 RSB Composition and Qualifications

The Reactor Safety Board (RSB) shall consist of at least three voting members knowledgeable in fields that relate to nuclear safety. The RSB shall review, evaluate and make recommendations on safety standards associated with the operational use of the facility. Members of NSC operations and health physics shall be ex-officio members on the RSB. The review and advisory functions of the RSB shall include NSCR operations, radiation protection and the facility license. The Chairman of the Reactor Safety Board under the direction of the Deputy Director of TEES shall appoint the board members.

12.2.2 RSB Charter and Rules

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

1) Meeting frequency: not less than once per calendar year and as frequent as circumstances warrant consistent with effective monitoring of facility activities.

2) Voting rules

3) Quorums

4) Use of subcommittees

5) Review, approval and dissemination of minutes

12.2.3 RSB Review Function

The review responsibilities of the Reactor Safety Committee shall include, but are not limited to the following.

1) Review and approval of new experiments utilizing the reactor facilities;

2) Review and approval of all proposed changes to the facility, procedures, license and technical specifications,

3) Determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in Technical Specification;

4) Review of abnormal performance of plant equipment and operating anomalies having safety significance;

5) Review of unusual or reportable occurrences and incidents that are reportable under 10CFR20 and 10CFR50;

6) Review of audit reports; and

7) Review of violations of technical specifications, license, or procedures and orders having safety significance

12.2.4 RSB Audit Function

The RSB or a subcommittee thereof shall audit reactor operations and radiation protection programs at least quarterly, but at intervals not to exceed four months. Audits shall include but are not limited to the following.

1) Facility operations, including radiation protection, for conformance to the technical specifications, applicable license conditions, and standard operating procedures at least once per calendar year (interval between audits not to exceed 15 months),

2) The retraining and requalification program for the operating staff at least once per calendar year (interval between audits not to exceed 15 months);

3) The facility security plan and records at least once per calendar year (interval between audits not to exceed 15 months);

4) The reactor facility emergency plan and implementing procedures at least once per calendar year (interval between audits not to exceed 15 months).

The licensee or his designated alternate (excluding anyone whose normal job function is within the NSCR) shall conduct an audit of the reactor facility ALARA program at least once per calendar year (interval between audits not to exceed 15 months). The results of the audit shall be transmitted by the licensee to the RSB at the next scheduled meeting.

12.3 Procedures

The philosophy of nuclear safety at the Nuclear Science Center assumes that all operations utilizing the reactor will be carried out in such a manner as to protect the health and safety of the public. This philosophy is augmented in practice by detailed, written procedures. All personnel using the facilities of the Nuclear Science Center follow the procedures. The loading or unloading of any core is performed according to detailed written procedures. Startup and operation of the reactor is also performed according to detailed written procedures.

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

1) Startup, operation, and shutdown of the reactor,

2) Fuel and experiment loading, unloading, and movement within the reactor,

3) Control rod removal or replacement,

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

5) Testing and calibration of reactor instrumentation and controls, control rod drives, area radiation monitors, and facility air monitors;

6) Civil disturbances on or near the facility site;

7) Implementation of required plans such as emergency or security plans, and

8) Actions to be taken to correct specific and foreseen potential malfunctions of systems, including responses to alarms and abnormal reactivity changes

The Director (NSC) and the Reactor Safety Board shall make substantive changes to the above procedures effective only after documented review and approval. The Director (NSC) or his designated alternate may make only minor modifications or temporary changes to the original procedures that do not change their original intent. All such temporary changes shall be documented and subsequently reviewed by the Reactor Safety Board

12.4 Required Actions

12.4.1 Action to be Taken in the Event a Safety Limit is Exceeded

In the event a safety limit is exceeded:

1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.

2) An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Board, and reports shall be made to the NRC in accordance with Section 6.6.2 of these specifications, and

3) A report shall be prepared which shall include an analysis of the cause and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Board for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

12.4.2 Action to be Taken in the Event of a Reportable Occurrence

In the event of a reportable occurrence, the following action shall be taken:

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

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

3) The Director (NSC) or his designated alternate shall notify the Chairman of the Reactor Safety Board

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

5) A report shall be made to the NRC in accordance with Section 6 6 2 of these specifications

6) Occurrence shall be reviewed by the RSB at their next scheduled meeting

12.5 Reports

12.5.1 Annual Report

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

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

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

c) The number of emergency shutdowns and inadvertent scrams, including reasons thereof,

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

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

f) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient.

i) Liquid Waste (summarized on a monthly basis)

(1) Radioactivity discharged during the reporting period.

(a) Total radioactivity released (in Curies)

(b) The Effluent Concentration used and the isotopic composition if greater than 1×10^{-7} µCuries/cc for fission and activation products.

(c) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.

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

(2) Total volume (in gallons) of effluent water (including dilutent) during periods of release.

ii) Gaseous Waste (summarized on a monthly basis)

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

(a) Argon-41

(b) Particulates with half-lives greater than eight days

iii) Solid Waste

(1) The total amount of solid waste transferred (in cubic feet)

(2) The total activity involved (in Curies).

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

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

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

12.5.2 Special Reports

In addition to the requirements of applicable regulations, reports shall be made to the NRC Document Control Desk and special telephone reports of events should be made to the Operations Center as follows:

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

a) Violation of safety limits (See Required Actions).

b) Any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;

c) Any reportable occurrences as defined in the Specifications. The written report (and, to the extent possible, the preliminary telephone or telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent reoccurrence of the event;

2) A written report within 30 days of:

a) Personnel changes in the facility organization involving Level 1 and Level 2.

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

12.6 Records

A daily reactor operations log is maintained by the reactor operator, and contains such information as core loading, experiments in the reactor, time of insertion and removal of experiments, power levels, time of startup and shutdown, core excess reactivity, fuel changes, and reactor instrumentation records.

Records are maintained which indicate the review, approval and conditions necessary for the production of radioisotopes or performance of irradiation experiments.

Records of facility operations in the form of logs, data sheets or other suitable forms are retained for the period indicated in the following sections:

12.6.1 Records to be retained for a Period of at Least Five Years or for the Life of the Component Involved

1) Normal reactor facility operation

2) Principal maintenance operations

3) Reportable occurrences

4) Surveillance activities required by the Technical Specifications

5) Reactor facility radiation and contamination surveys where required by applicable regulations

6) Experiments performed with the reactor

7) Fuel inventories, receipts, and shipments

8) Approved changes in operating procedures

9) Records of meeting and audit reports of the RSB

12.6.2 Records to be retained for at Least One Training Cycle

1) Retraining and Requalification of certified operations personnel

2) Records of the most recent complete cycle shall be maintained for individuals employed

12.6.3 Records to be retained for the Lifetime of the Reactor Facility

1) Gaseous and liquid radioactive effluents released to the environs.

2) Off-site environmental monitoring surveys required by the Technical Specifications

3) Radiation exposure for all personnel monitored

4) Drawings of the reactor facility.

12.7 Emergency Planning

The Emergency Plan for the Texas A&M Nuclear Science Center was prepared to meet the requirements of ANSI/ANS 15.16-1978 as amplified by Nureg-0849 The NSC submitted this plan to the NRC for review in August of 1994. The plan has subsequent revisions in September 1995 and December 1999. This version is the current version in use at the facility.

The Emergency Plan applies to Texas A&M University System Texas Engineering Experiment Station Nuclear Science Center (NSC) facility.

Texas A&M University has a campus wide radiological emergency plan which is intended to integrate radiological emergency planning at all campus facilities using radioactive materials or radiation producing devices. The NSC Emergency Plan is an integral part of the Texas A&M University emergency plan and specifies the objectives and implementing procedures to be followed for emergencies occurring at the NSCR.

The Emergency Plan indicates response capabilities for emergency conditions arising in connection with operation of the reactor. It includes identification of various precursor conditions (loss of electrical power, fires, reactor pool leaks, etc.) and the consequences for various independent or simultaneous precursors. The plan includes the event classification system. Detailed emergency implementing procedures have been developed and are referenced in the plan

The NSC Director has primary responsibility for emergency planning and response. The plan specifies delegation of responsibility and authority in the absence of the Director. The Emergency Plan and implementing procedures are reviews annually to assure that any required changes are incorporated into the plan

12.8 Security Planning

The Nuclear Science Center Security Plan indicates the measures provided to protect special nuclear material, including details of the protective equipment and police agencies As result, it is not for public access. The NSC Director is responsible for administering the security program and assuring that it is current

The Physical Security Plan provides the NSC with specific criteria and direction to protect the NSC from acts of sabotage and theft, which might endanger the health and safety of the public or the integrity of the facility

The plan fulfills the applicable security planning requirements of 10CFR50 and 10CFR70 Specifically, the NSC maintains a non-power reactor license and implements physical protection based on 10CFR73.60, "Additional requirements for the physical protection of special nuclear material at non-power reactors "

12.9 Quality Assurance

Since the NSC is not seeking a construction permit, this SAR does not include a description of a quality assurance program for the design and construction of the structures, systems and components of the facility This section describes the Quality Assurance program that is in place to govern safe operation and modification of the facility. This program meets the applicable requirements of Regulatory Guide 2.5 and ANSI/ANS-15 8-1995.

The NSC Director has responsibility for the quality assurance activities, and thus has the authority to identify problems, to initiate corrective actions and to insure that corrective actions are complete. He exercises a QA oversight by assuring that operating and maintenance procedures include specific requirements to assure that modification, maintenance and calibration of safety-related systems maintain the quality and reliability of equipment. Further, experiment reviews use written requirements to assure that installation and operation of the experiment does not degrade the performance of safety equipment. Planning and reviewing modification of safety-related equipment using formal written checklist-type procedures assure equipment continues to meet NSC specifications. Most of the reactor equipment in use in the facility does not have a formal QA documentation. The provisions of section 4 of ANSI/ANS-15.9 cover this equipment. After-maintenance checks, alignment and calibration of the replacement equipment assure that equipment meets the original equipment specifications.

Procedures include schedules of equipment maintenance and calibration, and provide records that such functions are completed. Calibration procedures include requirements that critical equipment and instruments used in the calibrations are themselves currently calibrated

12.10 Operator Training and Requalification

The Standard Operating Procedures covers the detailed requirements for Requalification of licensed operators. The NSC is committed to maintaining the highest level of operator qualification The program consists of lectures followed by detailed and in-depth exams to verify level of knowledge, a number or reactor manipulations to ensure proficiency and operator evaluations to ensure an effective skill level

12.11 Startup Plan

This Safety Analysis Report does not include a startup plan because the facility has been in routine operation for many years.

12.12 Environmental Reports

The Atomic Energy Commission concluded "that there will be no significant environmental impact associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 Mwt or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities." This is from a letter date January 23, 1974 from D. R. Miller.

The NSC expects no change in land and water use because of extending the NSC license for an additional 20 years. Emissions of radioactive materials or other effluents will not change because of extending the license term.

13 ACCIDENT ANALYSIS

13.1 Accident-Initiating Events and Scenarios

This section is included in the specific subsections in 13 2

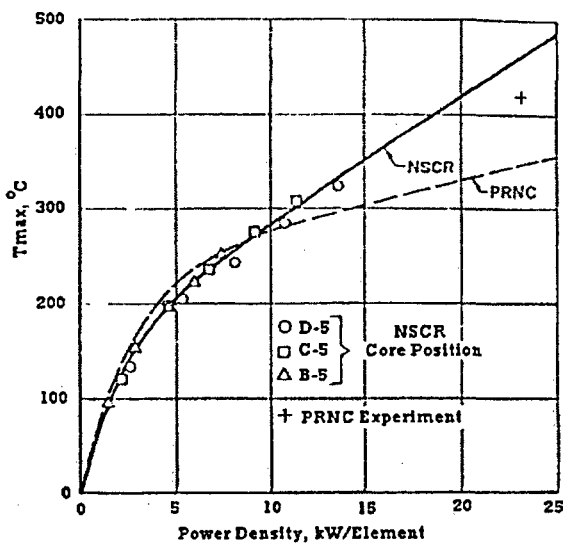
13.2 Accident Analysis and Determination of Consequences

13.2.1 Maximum Hypothetical Accident

The Design Basis Accident is the loss of fuel cladding integrity for one fuel element and the simultaneous pool-water loss resulting in fission product release.

In an effort to estimate the total release of fission product in the case of fuel-cladding failure, Texas A&M conducted a series of experiments to predict the maximum fuel temperature The experimenters measured temperatures in a few core locations Experiments used this data to estimate the actual peak temperature inn those locations

To find the actual power density (PD) in these locations, the experimenters used the program Exterminator-2. With this information, along with the peak temperatures, they were able to graph the relationship between PD and peak fuel temperature (Figure 13-1). The results show higher peak fuel temperature than predicted for the Puerto Rico Nuclear Center The Puerto Rico Nuclear Center also obtained one measurement for a PD of 23 kW per fuel element The Texas A&M relationship shows a higher temperature for that PD



[redacted]

Figure 13-1: Steady State Fuel Temperature as a Function of Power Generation

Finally, this analysis assumes a maximum temperature of 535°C for a PD of 28 kW, the maximum allowed for the NSCR With this maximum temperature and PD, the surface or "minimum" temperature would be 150°C.

To estimate the fission product release, this analysis uses the fission-product release fraction of 2 6x10^{-5}. General Atomics experimentally determined this fraction averaged over the fuel volume

Using the above assumptions, the saturated activities of the significant fission products at 1 Mw in a single fuel element are

[redacted]

Applying the release fraction of 2.6 x 10^{-5} to the total inventory in a single element operating at 1 Mw yields the following activities that would be released in a cladding failure.

[redacted]

If the release accident occurred with water in the pool, the halogens will remain in the water. The resulting concentration would be 3 65x10^{-4} μc/cm^3. Within 24 hours, this value would decay to 8.34 x 10^{-5} μc/cm^2. The Demineralizer system would be removed these soluble fission products and they would eventually go into the liquid waste system

Texas A&M calculated the results of the release of fission products from a single fuel element with and without water in the reactor pool, and with and without the ventilation system in operation.. Table 13-1 shows the calculated exposure to population outside the building and exposure to operating personnel inside the facility. The only case where significant exposure occurs requires the simultaneous failure of the fuel element clad, catastrophic failure of the pool and liner, and a failure of the ventilation system with personnel remaining within the reactor facility for a period of 1 hour after release. The maximum exposure is 49 R to the thyroid. Thus, no realistic hazard of consequence will result from the Design Basis Accident.

Table 13-1: Summary of Radiation Exposures Following Cladding Failure of the highest Power Density FLIP Fuel element

	WBGD*	Thyroid Dose
A. Building Ventilation Operating:		
1. Maximum Exposure to Population Outside Building		
Pool Water Remaining	3.5x10^{-3}	--
Pool Water Drained	1.4x10^{-2}	3.7 mR
2. Exposure to Operating Personnel in One Hour After Release		
Pool Water Remaining	0.84 mR	--
Pool Water Drained	1.75 mR	10.5 mR
B. Building Ventilation Shut Down		
1. Maximum Exposure to Population Outside Building (12 hours)	WBGD*	Thyroid Dose
Pool Water Remaining	3.6x10^{-3}	--
Pool Water Drained	2.1x10^{-2}	18 mR
2. Exposure to Operating Personnel in One Hour After Release		
Pool Water Remaining	1.75 mR	--
Pool Water Drained	4.2 mR	49 R

* WBGD = Whole Body Gamma Dose

13.2.2 Insertion of Excess Reactivity

13.2.2.1 Accidental Pulsing From Full Power

This analysis is for the accidental prompt reactivity addition for the NSCR operating at power. Normally these pulses are from below the point of adding heat ,It is necessary to examine this situation in spite of the interlocks that will prevent this from happening

General Atomic performed the calculations using the BLOOST 2 code assuming adiabatic processes. The details of the calculations are below Texas A&M supplied the experimental parameters and core power distributions The study was to find the reactivity insertion from power that would produce a peak core temperature of 950° C. Figure 13-2 shows the results from both 1 MW and 300 watts as a function of the number of FLIP elements in the mixed core at the beginning-of-life. Figure 13-2 also shows the calculated end-of-life case for a full FLIP core. As Figure 13-2 shows, if the reactor is pulsed from 1 MW, considerably more prompt reactivity is required to obtain 950° C Therefore, it is pulsing from low power that limits the amount of insertion and not the "accident" situation

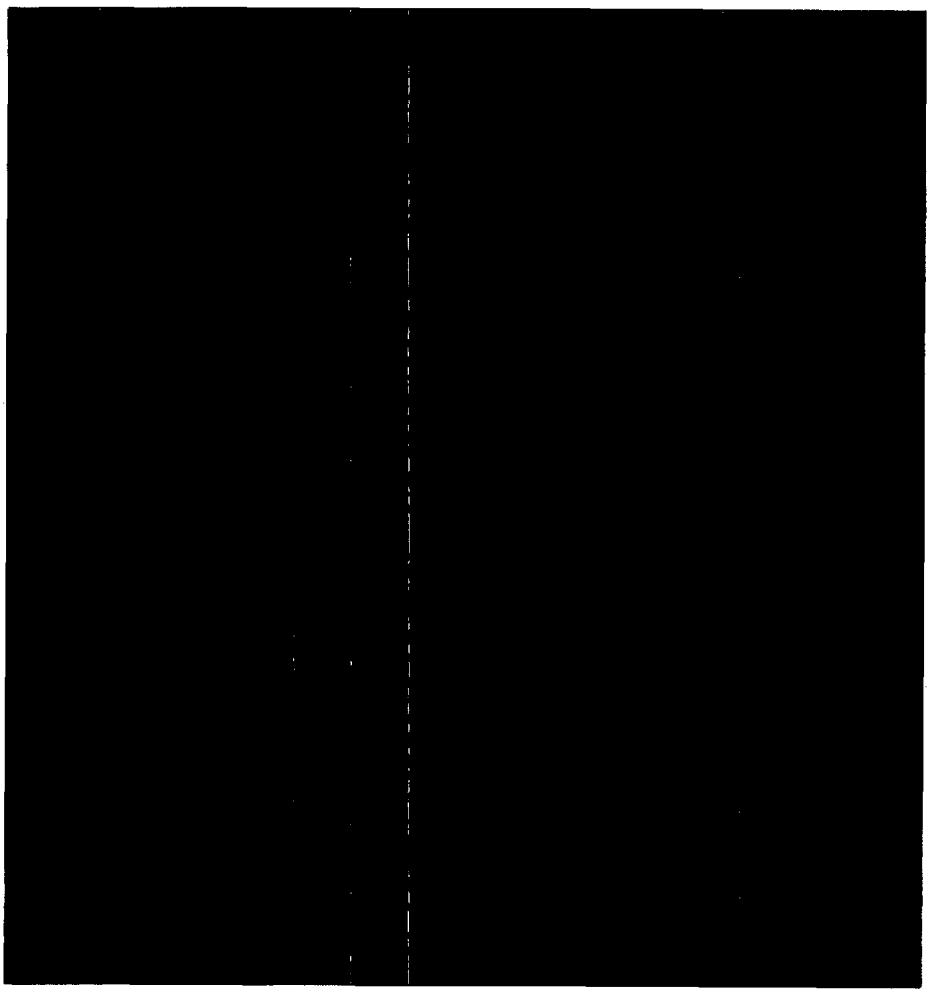


Figure 13-2: Pulse to Produce 950°C Peak Temperature

13.2.2.2 The Pulsing Accident in Mixed and Flip Cores

The scope of this study performed by General Atomic was:

1) Determine the size of the pulse producing 950°C peak temperature as a function of the number of FLIP elements in the core and the burnup by assuming adiabatic processes, and

2) If the results of 1) do not allow pulses of an acceptable magnitude, to refine the calculations by including the effects of heat transfer which would give a more realistic assessment of the effects of the reactivity addition

The results of 1) indicated that 2) was not required Figure 13-2 shows the size of the pulse that would produce maximum temperatures of 950°C as a function of the number of FLIP elements in the core. The results are for pulses from 1 Mw and from 300w steady state power. The curves represent beginning-of-life (BOL) conditions for the prompt neutron lifetime and the temperature coefficient General Atomics used the end-of-life (EOL) values for these parameters to calculate the two points for the full FLIP core

Following is a summary of the process involved in acquiring this information:

1) The calculations were made using BLOOST 2 The input parameters that were common to all problems were.
2) No of elements = [redacted]
3) Delayed neutron fraction = 0 007
4) Fuel specific heat = 720 0 + 1.48 T (w-sec/°C-element)
5) Water specific heat = 860 (W-sec/°C-element)
6) Fuel thermal resistance = 10000 (°C/MW/element)
7) Coolant thermal resistance = 1175 (°C/MW/element)
8) Initial average fuel temperature at 1 MW = 238° C
9) Initial average coolant temperature at 1 MW = 45° C
10) Pulse insertion time = 100 msec
11) Scram delay time = 15 msec
12) Rod drop time = 0.985 sec

For the pulses from 300 W, General Atomics assumed that the system had an initial temperature of 25°C and that there was no scram

Figure 13-3 shows the measured transient rod integral worth The "ramp" table assumes this rod uniformly accelerates from its initial position to 605 units (i e., $3.25) The initial position provides the desired worth of the pulse. The $3 25 position for the upper end of the insertion sharpens the pulse as the integral worth curve flattens out drastically above that point Using the full out position as the final point would tend to clip the pulse. Figure 13-4 shows the ramp as a function of time for $2 00, $2.25 and $2.75 pulse. These curves are plots of the ramp table.

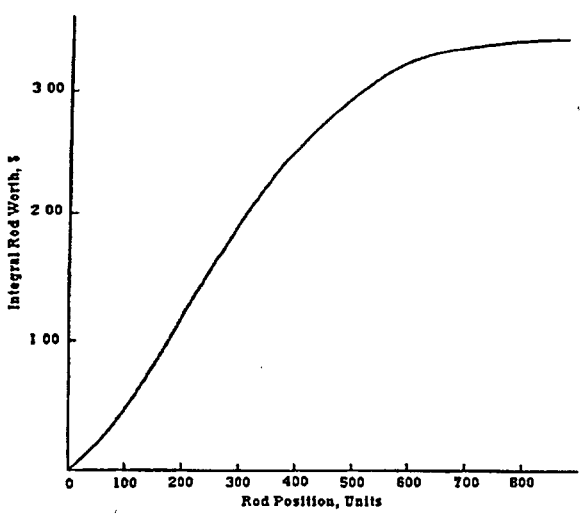


Figure 13-3: Transient Rod Integral Rod Worth

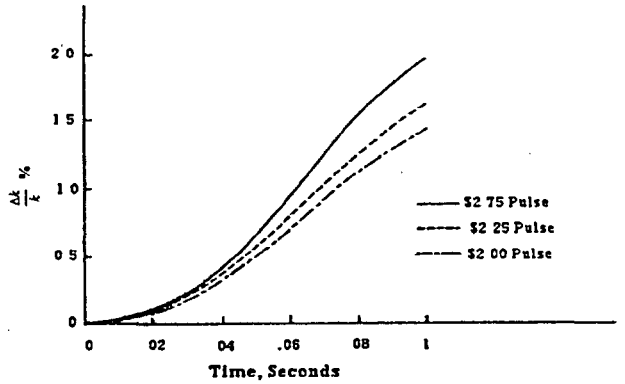


Figure 13-4: Time Dependent Pulse Reactivity Insertion Used to Obtain Ramp Table

Construction of the scram table assumed the total excess available was $7.00, that $3.50 was held down by the 1 Mw temperature, total worth of the rods available for the scram was $12.00, the shape of the rod worth is represented by the General Atomic "standard" shape and the rods fall with uniform acceleration Figure 13-5 is a plot of the scram table.

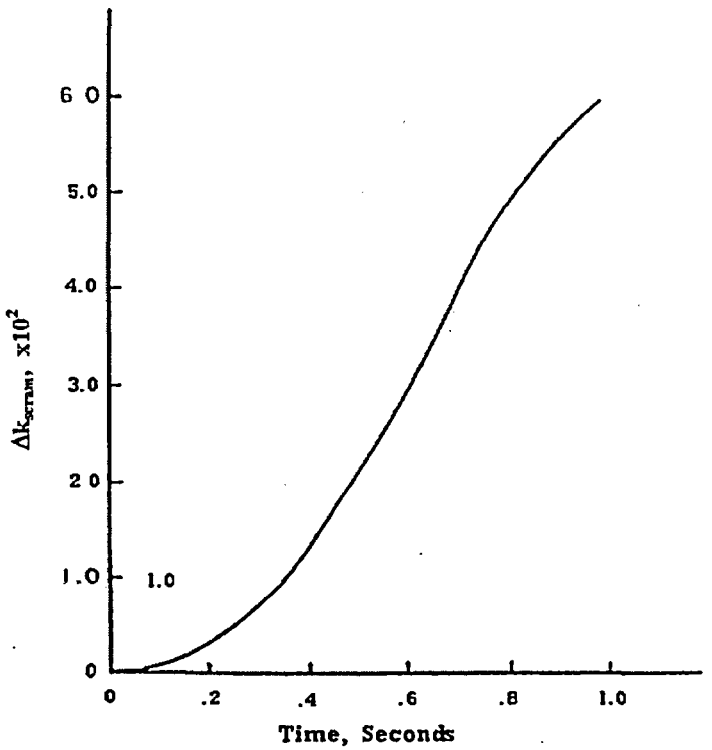


Figure 13-5: Time Dependent Reactivity Insertion Used to Generate Scram Table

General Atomics included three different core configurations to determine the effect of adding FLIP fuel. All assumed the FLIP fuel was in the central region surrounded by standard fuel (if any). All the cores consisted of 98 fuel elements The values of the several parameters that are dependent on configurations were estimated by interpolation between data points already acquired.

Peaking factors for the power distribution were

[redacted] elements: axial = 1.36; radial = 2.00; cell = 1.95

[redacted] elements: axial = 1 36, radial = 1.91, cell = 1 95
[redacted] elements: axial = 1 36; radial = 1 56, cell = 1 95

The following calculated data provided the input to estimate the prompt neutron lifetimes·

BOL - [redacted] l= 28 0 μsec
BOL [redacted] l= 17.5 μsec
EOL- [redacted] l= 21.0 μsec

Linear interpolation, based on the fraction of FLIP elements in the core, gives the following

[redacted] BOL l = 26.5 μsec
[redacted] BOL l = 23.1 μsec
[redacted] BOL l =17.5 μsec
[redacted] EOL l = 21 0 μsec

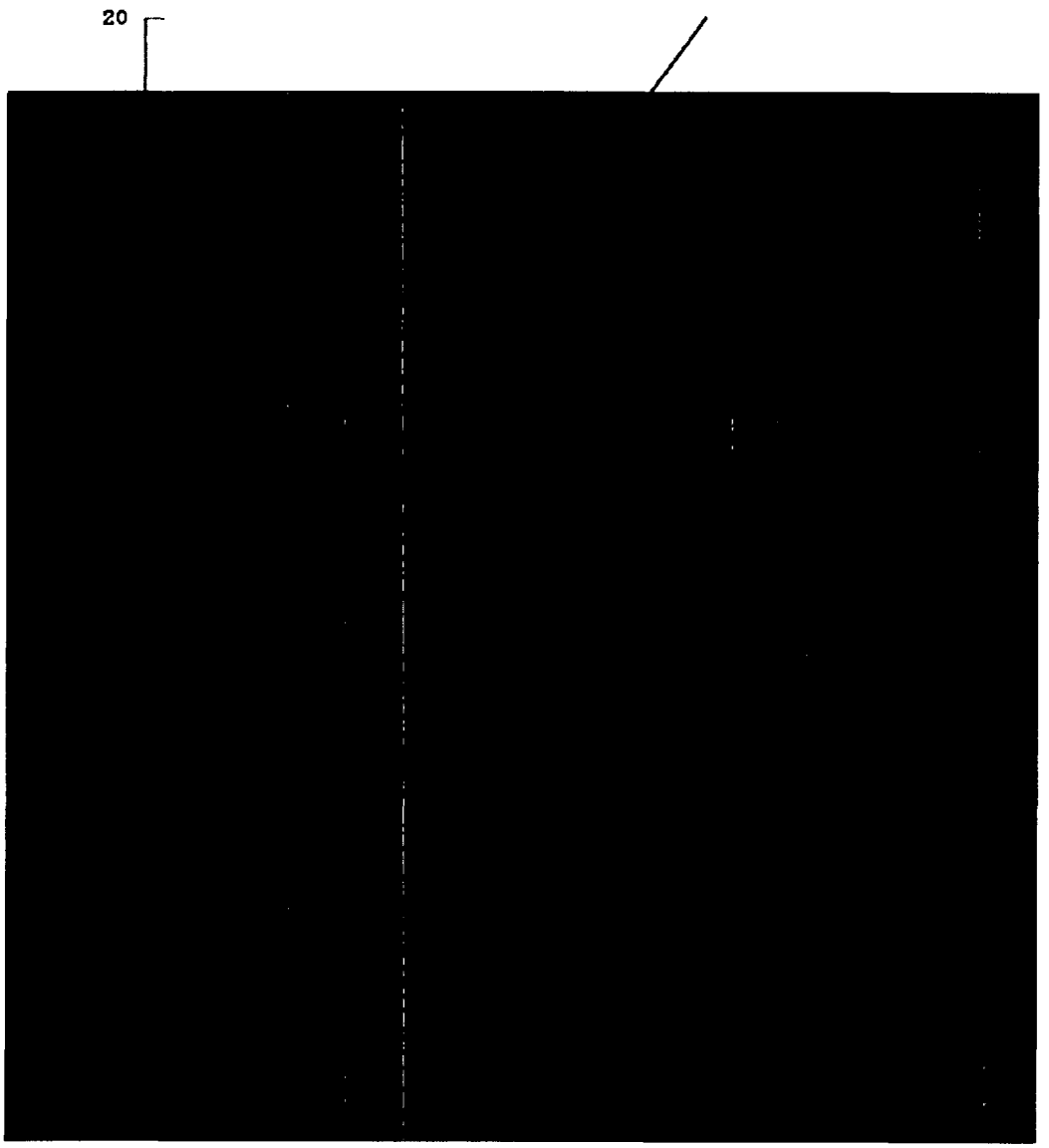


Figure 13-6: Temperature Coefficients of TRIGA Fuels

Figure 13-6 shows the calculated prompt negative temperature coefficients for the 18-rod FLIP at BOL and the full FLIP at BOL and EOL The BOL coefficient for the mixed cores between [redacted] and full FLIP was estimated by making a linear interpolation between the two curves with the ratio of FLIP to total as the proportionality constant Figure 13-7 provides plots of the integral temperature coefficients, used in BLOOST 2, for the four core configurations

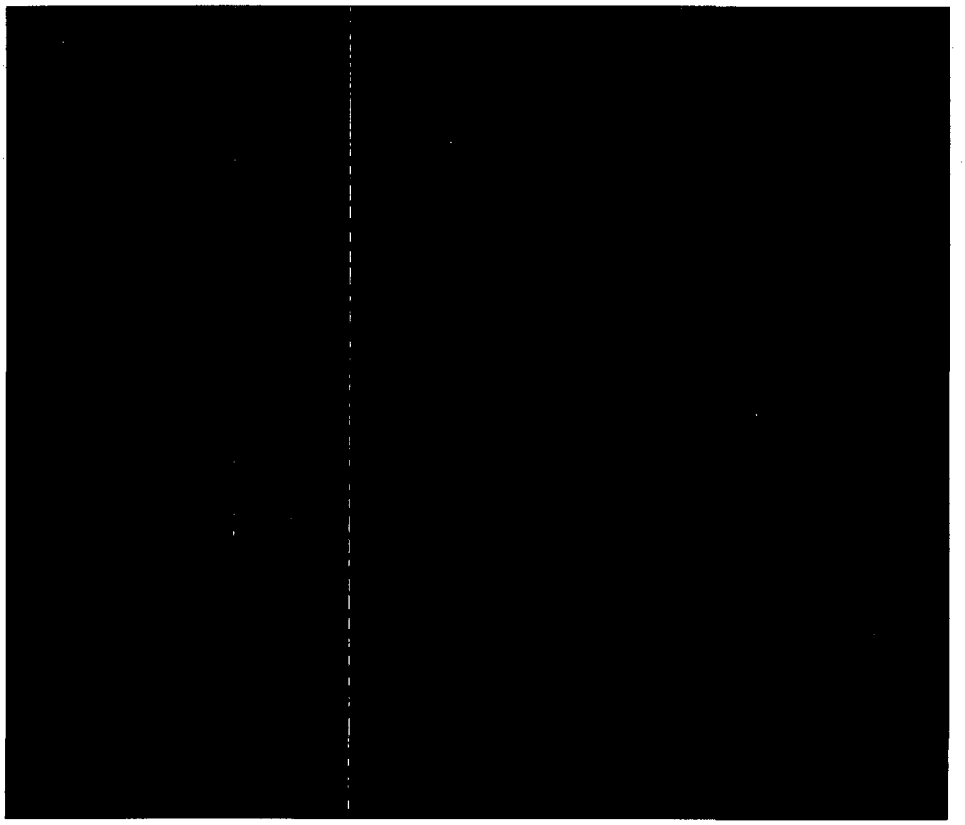


Figure 13-7: Integral Temperature Coefficient

The integral water temperature coefficient is the standard FLIP coefficient in the inset to Figure 13-7.

Since the BLOOST 2 does not calculate the peak temperature properly when the pulse is from power, hand calculations provided these values. The following steps describe the calculation procedure.

1) Determine the average power density in the "hottest" element at 1 MW.

2) From this power density, determine the steady state temperature at the periphery of the fuel at the axial centerline.

3) Calculate the energy content at the point at 1 MW.

4) Calculate the energy added at that point in the pulse

Determine the adiabatic temperature that would result from that energy content. The peak edge temperature in the fuel before the pulse is:

[redacted]	SS edge temp = 165°C	Energy content = 0.139 MW/element
	SS edge temp = 170°C	Energy content = 0.144 MW/element
	SS edge temp = 180°C	Energy content = 0.154 MW/element

The following tables give the principle results of the BLOOST 2 calculations

Table 13-2: BLOOST Results for Pulse from 1 MW

PULSE FROM 1 Mw							
Prob #	$	No. Flip	Burnup	P Max (MW)	Prompt Energy (MW-sec)	t	T
11	2 00	[redacted]	BOL	677	12 4	345	667.3
12	2 75	[redacted]	BOL	1383	18 4	394	844 1
13	2.00	[redacted]	BOL	817	13 5	354	685 9
14	2.75	[redacted]	BOL	1579	19 6	403	855.2
15	2.00	[redacted]	BOL	1110	15.8	373	673.7
16	2.25	[redacted]	BOL	1420	18.1	392	728.9
17	2.00	[redacted]	BOL	1502	22.4	425	826 2
18	2.25	[redacted]	BOL	1951	26 4	455	910.0

Table 13-3: BLOOST Results for Pulse from 300 W

PULSE FROM 300W								
Prob #	$	No. FLIP	Burnup	Pmax	Prompt Energy	Energy to 2s	t	T
21	2.00	[redacted]	BOL	1263Mw	17.7Mw	24 0	225	767
22	2.75	[redacted]	BOL	3625Mw	28.5Mw	34.4	324	1057
23	2.00	[redacted]	BOL	1619Mw	19.6Nw	26.5	244	802
24	2.75	[redacted]	BOL	4631Mw	31.6Mw	38.0	349	1094
25	2.00	[redacted]	BOL	2645Mw	24.1Mw	32.5	286	802
26	2.25	[redacted]	BOL	4016Mw	28.9Mw	36.8	328	907
27	2.00	[redacted]	BOL	2561Mw	29.0Mw	39 4	329	911
28	2.25	[redacted]	BOL	3905Mw	35 0Mw	45.7	379	1031

To find the values of the reactivity insertion that would result in peak fuel temperatures of 950° C, an interpolated value was found from the data in the tables above. These values are in the next table It was felt that these results were sufficient to allow effective operation so it was not necessary to do the second part of the program that would consider heat flow from the fuel after the pulse.

Table 13-4: Pulse Values to Exceed Temperature

REACTIVITY TO GIVE 950° C PEAK TEMPERATURE			
No. FLIP Mw	Burnup	Pulse from 300w	Pulse from 1Mw
[redacted]	BOL	$2.45	$3.32
[redacted]	BOL	2.35	3.29
[redacted]	BOL	2.36	3.65
[redacted]	BOL	2.08	2.42

13.2.3 Loss of Coolant

The strength of the fuel element clad is a function of its temperature. The stress imposed on the clad is a function of the fuel temperature as well as the hydrogen-to-zirconium ratio, the fuel burnup, and the free gas volume within the element. In the analysis of the stress imposed on the clad and strength of the clad uses the following assumptions:

1) The fuel and clad are at the same temperature.
2) The hydrogen-to-zirconium ratio is 1.7 for standard fuel and 1.6 for FLIP fuel
3) A space one-eighth inch high within the clad represents the free volume within the element
4) The reactor contains fuel that has experienced burnup equivalent to 7700 MW-days.
5) Maximum operating temperature of the fuel is 600° C.

The fuel element internal pressure P is given by:

$$P = P_h + P_{fp} + P_{air} \qquad \text{Eq. 13-1}$$

where.

P_h is the hydrogen pressure,

P_{fp} is the pressure exerted by volatile fission products, and

P_{air} is the pressure exerted by trapped air

For hydrogen-to-zirconium ratios greater than about 1.58, the equilibrium hydrogen pressure can be approximated by:

$$P_h = \exp\left(1.76 + 10.3014x - \frac{19740.37}{T_k}\right) \qquad \text{Eq. 13-2}$$

where

x is the ratio of hydrogen atoms to zirconium atoms, and

T_k is the fuel temperature (K)

The pressure exerted by the fission product gases is given by.

$$P_{fp} = f\frac{n}{E}\frac{RT_k}{V}E \qquad \text{Eq. 13-3}$$

where.

f is the fission product release fraction;

$\frac{n}{E}$ is the number of moles of gas evolved per unit of energy produced (mol/MW-day),

R is the gas constant (8.206 x 10^{-2} L-atm/mol-K);

V is the free volume occupied by the gasses (L), and

E is the total energy produced in the element (MW-day)

The fission product release fraction is given by:

$$f = 1.5\times10^{-5} + 3.6\times10^{3}\exp\left(\frac{-1.34\times10^{4}}{T_o}\right) \qquad \text{Eq. 13-4}$$

where:

T_o is the maximum fuel temperature in the element during normal operation (K).

The fission product gas production rate, $\frac{n}{E}$, varies slightly with the power density. The value 1.19 x 10^{-3} mol/MW-day is accurate to within a few percent over the range from a few kilowatts per element to well over 40 kW per element. The free volume occupied by the gases is assumed to be a space one-eighth inch (0.3175 cm) high at the top of the fuel so that

$$V = 0.3175\pi \cdot r_i^2 \qquad \text{Eq. 13-5}$$

where.

r_i is the inside radius of the clad (1.745 cm).

For standard TRIGA fuel, the maximum burnup is about 4 5 MW-days per element, but the TRIGA-FLIP fuel is capable of burnup to about 77 MW-days per element As the fission product gas pressure is proportional to the energy released, assume that the FLIP fuel in the reactor has experienced maximum burnup

Finally, the air trapped within the fuel element clad will exert a pressure

$$P_{air} = \frac{RT_k}{24} \qquad \text{Eq 13-6}$$

where it is assumed that the initial specific volume of the air is 22.4 L/mol. Actually, the air forms oxides and nitrides with the zirconium, so that after relatively short operation the air is no longer present in the free volume inside the fuel element clad. The results of the stress imposed on the clad for standard and FLIP fuels are in Figure 13-8.

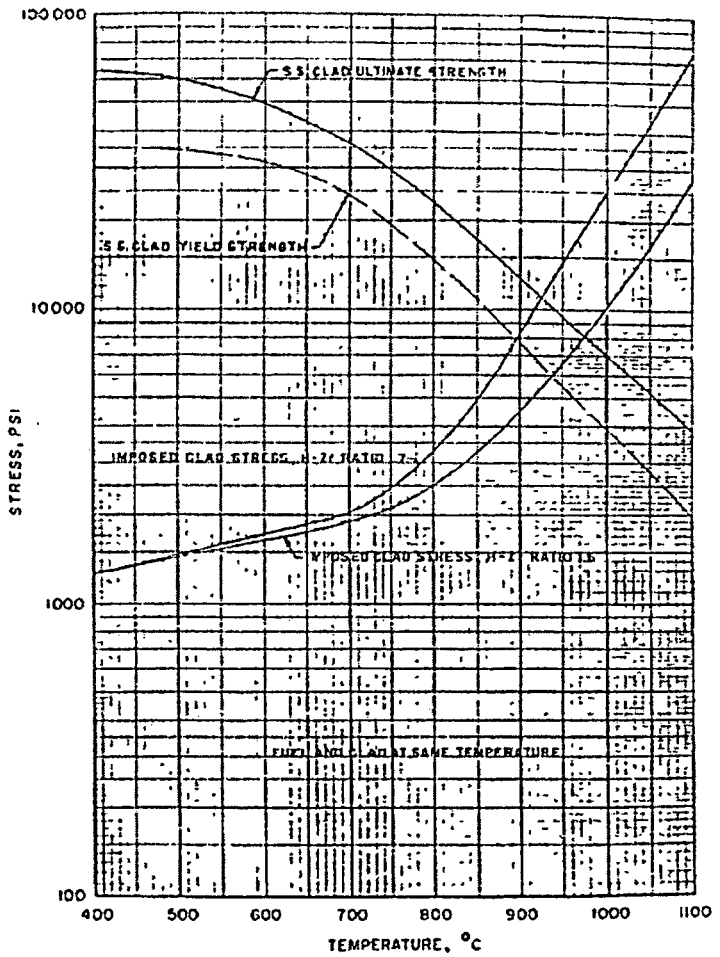


Figure 13-8: Strength and Applied Stress as a Function of Temperature for 1.7 and 1.6 H-Zr TRIGA Fuel

General Atomic developed A two dimensional transient-heat transport computer code for calculating the system temperatures after the loss of pool water. They were derived the heat removal parameters for the calculations. The assumption was that the reactor was shutdown [redacted] efore the core was uncovered (the time between the actuation of the pool level alarm and the uncovering of the fuel fo[redacted]

If the reactor operates for seventy MW-hours or less per week, power generation per element values approximately 20% higher are sufficient Thus, 25 kW/element for standard and 28 kW/element for FLIP fuel are adequate power densities A comparison of decay heat generation versus time following loss of coolant for infinite reactor operations and 70 MW-hours per week cycle operation are in Figure 13-9.

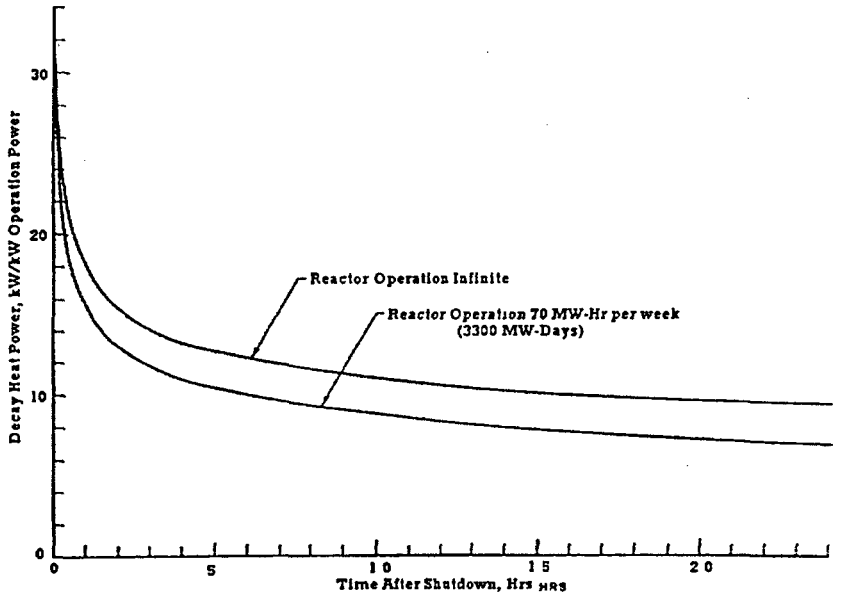


Figure 13-9: Decay Heat Power Generation Following Loss of Coolant for Infinite Reactor Operation and Periodic Reactor Operation

13.2.4 Loss of Coolant Flow

The NSC Reactor uses natural convection flow; therefore, this casualty is not applicable.

13.2.5 Mishandling or Malfunction of Fuel

This accident is included in the Maximum Hypothetical Accident. The worst case would be mishandling or malfunction leading to cladding failure. The Maximum Hypothetical Accident addresses this with a catastrophic failure of the pool and a failure of the ventilation system

13.2.6 Experiment Malfunction

The Reactor Safety Board must approve any new experiment involving the reactor. This committee reviews all experiments for safety and for compliance with the operating license and NRC regulations. The Senior Reactor Operator controls loading, unloading or moving Experiments affecting the reactivity of the core.

The reactivity effects of experimental facilities used with the present core present no significant problems The values reported for similar experimental facilities at other TRJGA installations appear to be comparable and therefore no hazard exists. The reactivity worth of any non-secured experiment shall have reactivity wroth less than $1.00. This specification provides assurance that the worth of a single unfastened experiment will be limited to a value such that suddenly inserting the positive worth of the experiment will not exceeded the safety limit. Removal of experiments of $0.30 worth or more from the reactor at full power often requires a power decrease by the operator to prevent high power levels

13.2.7 Loss of Normal Electrical Power

The NSCR will shutdown upon loss of electrical power, as electricity is required to maintain magnet current in the control rod drives to keep the control rods raised out of the core. Once electrical power is lost, magnet current is lost

and control rods drop into the core. The reactor will have sufficient shutdown margin even with the most reactive control rod stuck in its fully withdrawn position No emergency power supply is available.

13.2.8 External Events

Floods, hurricanes, earthquakes and tornados are the three areas of concern for naturally occurring external events In the case of floods and hurricanes, a complete flooding of the Lower Research Level will have no effect on reactor safety or the safety of the public. Nor is it likely that hurricane winds will affect the integrity of the building (a fallout shelter) or the pool As chapter 2 of this report states, earthquakes are extremely unlikely in this area. Finally, tornados occur near the NSC. However, damage to the reactor pool, which protects the reactor, is not credible

While aircraft collisions are unlikely, they are possible given the proximity of Easterwood airport. Impacts by aircraft will not breach the concrete shield at core level. Penetrating the outer walls of the building will not result in the release of radiation

[redacted]

The NSC reaches the same conclusion for the NSC Reactor.

13.2.9 Mishandling or Malfunction of Equipment

Potential Hazards Considered

1. Fuel Bundle Rotation

To achieve symmetry with the central FLIP region it is often necessary to load both FLIP and standard elements in a single bundle. Since the fuel bundles can be physically rotated 180° and still fit the grid plate, the inadvertent rotation of a four-element fuel bundle containing one FLIP and three standard elements was considered In the event of such a rotation, standard fuel would surround the FLIP element by. The analysis showed no appreciable increase in average power generated either in the FLIP element, or in the standard elements. The maximum fuel temperature that would be obtained during a 30 Mw-sec pulse with the rotated bundle is 1450°F (788°C) which is well below the safety limit. Table 13-5 shows the results of the calculations of peak average power per cell during 1 Mw steady-state operation and the maximum fuel temperature due to a 30 Mw-sec pulse with and without rotation of one bundle. There is no safety problem due to inadvertent rotation of a fuel bundle.

Table 13-5: Fuel Bundle Rotation Study for Maximum Power and Maximum Temperature

Limiting Criterion	Core with unrotated bundle	One bundle rotation
Maximum power per element	18.18 kW	17.94 kW
Maximum fuel temperature due to a 30MW-sec pulse	1400°F (761°C)	1450°F (788°C)

2 Control Rod Run-Out

The magnitude of the result of the withdrawal of all control rods at maximum speed was considered for the PRNC reactor The magnitude of the effect of this accident is dependent primarily on the speed of rod withdrawal and the value of the temperature coefficient Since proposals for the Texas A&M reactor included core loadings that varied from all, standard to all FLIP TRIGA fuel the temperature coefficient had to be examined for these variations. General Atomic performed calculations for a variety of cores with the results shown in Figure 13-6 As can be seen, the standard fuel with no poison had a temperature coefficient that was relatively constant with temperature The addition of FLIP fuel resulted in temperature coefficient that increased linearly with temperature Most of the core loadings anticipated for the NSCR will lie between [redacted] It is doubtful that an entire core would ever achieve the 3000 MW/day burnup indicated for the lowest curve due to the planned loading sequence. Even for this limiting case, however, the magnitude of the temperature coefficient was large enough to allow safe operation of the reactor.

For the calculation of the PRNC control rod run-out accident, the withdrawal time used was 16.2 seconds The withdrawal time of the shim-safeties of the NSCR reactor is 347 minutes. The NSCR will be set to scram at 1.25 Mw (or less) as opposed to 2.2 Mw for the PRNC reactor. Since the temperature coefficient will be the same or larger and the control rod removal rate is so much slower, the reactor power level will follow the rod insertion so that the excess reactivity will be near zero. Thus, when the trip occurs the core temperature will nearly correspond to the case of a reactor operating at steady state. The maximum power generated in any cell will be approximately 25 kW, which is well below the maximum permitted

13.3 Summary and Conclusions

None of the accidents here will result in consequences to the public health and safety.

14 TECHNICAL SPECIFICATIONS FOR FACILITY LICENSE NO. R-83

Included in this section are the Technical Specifications and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for informational purposes only. They are not part of Technical Specifications and they do not constitute limitations or requirements to which the licensee must adhere.

14.1 Definitions

14.1.1 Abnormal Occurrence

An abnormal occurrence is an unscheduled incident or event that the Nuclear Regulatory Commission determines is significant from the standpoint of public health or safety

14.1.2 ALARA

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

14.1.3 Channel

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

14.1.3.1 Channel Test

A channel test is the introduction of a signal into the channel for verification that it is operable

14.1.3.2 Channel Calibration

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

14.1.3.3 Channel Check

A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

14.1.4 Confinement

Confinement means a closure of the overall facility that results in the control of the movement of air into it and out of the facility through a defined path.

14.1.5 Core Lattice Position

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

14.1.6 Experiment

An operation, hardware, or target (excluding devices such as detectors, foils etc.) which is designed to investigate non-routine reactor characteristics, or which is intended for irradiation within the pool, on or in a beam port or irradiation facility, and which is not rigidly secured to a core or shield structure so as to be a part of their design.

14.1.7 Experimental Facilities

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

14.1.8 Experiment Safety Systems

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

14.1.9 FLIP Core

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

14.1.10 Fuel Bundle

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

14.1.11 Fuel Element

A fuel element is a single TRIGA fuel rod of either standard, FLIP, or LEU type.

14.1.12 Instrumented Element

An instrumented element is a special fuel element in which a sheathed chromal-alumel or equivalent thermocouple is embedded in the fuel near the horizontal center plane of the fuel element at a point near the center of the fuel body.

14.1.13 LEU Core

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

14.1.14 Limiting Safety System Setting

The limiting safety system setting is the setting for automatic protective devices related to those variables having significant safety functions

14.1.15 Measuring Channel

A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device that are connected for the purpose of measuring the value of a variable.

14.1.16 Measured Value

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

14.1.17 Mixed Core

A mixed core is an arrangement of standard TRIGA fuel elements with at least 35 TRIGA-FLIP and/or LEU fuel elements located in a central contiguous region of the core.

14.1.18 Movable Experiment

A movable experiment is one for which it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating

14.1.19 Operable

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

14.1.20 Operating

Operating means a component or system is performing its required function

14.1.21 Steady State Operational Core

A steady state operational core shall be a standard core, FLIP core, LEU core, or mixed core for which the core parameters of shutdown margin, fuel temperature and power calibration have been determined

14.1.22 Pulse Operational Core

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

14.1.23 Pulse Mode

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

14.1.24 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter the experiment position or configuration

14.1.25 Reactor Console Secured

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

14.1.26 Reactor Operating

The reactor is operating whenever it is not secured.

14.1.27 Reactor Safety Systems

Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action Manual protective action is considered part of the reactor safety system

14.1.28 Reactor Secured

A reactor is secured when:

1) It contains insufficient fissile material or moderator present in the reactor and adjacent experiments to attain criticality under optimum available conditions of moderation and reflection, or

2) The reactor console is secured <u>and</u> no work is in progress involving core fuel, core reflector material, installed control rods, or control rod drives unless they are physically decoupled from the control rods.

14.1.29 Reactor Shutdown

The reactor is shut down if it is subcritical by at least one dollar with the reactor at ambient temperature, xenon-free and with the reactivity worth of all experiments included.

14.1.30 Reportable Occurrence

A reportable occurrence is any of the following that occurs during reactor operation:

1) Operation with actual safety system settings for required systems less conservative than the limiting safety-system settings specified in the Technical Specifications 2.2.

2) Operation in violation of limiting conditions for operation established in the technical specifications.

3) A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (Note: Where components or systems are

provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

4) An unanticipated or uncontrolled change in reactivity greater than one dollar

5) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both

6) An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations

14.1.31 Rod-Control

A control rod is a device fabricated from neutron absorbing material and/or fuel that is moved up or down to control the rate of a nuclear reaction It may be coupled to its drive unit allowing it to perform a safety function (scram) when the coupling is disengaged.

14.1.32 Rod-Regulating

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

14.1.33 Rod-Shim Safety

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

14.1.34 Rod-Transient

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

14.1.35 Safety Channel

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

14.1.36 Safety Limit

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

14.1.37 Scram Time

Scram time is the time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control rod reaches its fully inserted position.

14.1.38 Secured Experiment

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

14.1.39 Shall, Should and May

The word "shall" is used to denote a requirement, the word "should" to denote a recommendation, and the word "may" to denote permission, neither a requirement nor a recommendation In order to conform to this standard, the user shall conform to its requirements but not necessarily to its recommendations.

14.1.40 Shutdown Margin

Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition. It assumes that the most reactive scrammable rod and any non-scrammable rods are fully withdrawn, and that the reactor will remain subcritical without any further operator action

14.1.41 Standard Core

A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.

14.1.42 Steady State Mode

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

14.1.43 True Value

The true value is the actual value of a parameter.

14.1.44 Unscheduled Shutdown

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

14.2 Safety Limit and Limiting Safety System Setting

14.2.1 Safety Limit - Fuel Element Temperature

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element cladding will result.

Specifications

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

b) The temperature in a stainless steel-clad Standard TRIGA fuel element shall not exceed 1830°F (1000°C) under any conditions of operation.

Bases

The important parameter for a TRIGA reactor is fuel element temperature. This parameter is well suited as a single specification because it can be measured directly with a thermocouple or inferred indirectly through reactor power. A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure within the fuel element if the fuel element temperature exceeds the temperature safety limit The fuel element temperature and the ratio of hydrogen to zirconium in the fuel-moderator material determine the magnitude of the pressure buildup. The

mechanism for the pressure buildup is the dissociation of hydrogen from the zirconium hydride moderator that has been blended with uranium to form the fuel mixture encased within the fuel element cladding

The temperature safety limit for the TRIGA-FLIP or LEU fuel element is based on data which indicates that the internal stresses within the fuel element due to hydrogen pressure from the dissociation of the zirconium hydride will not result in compromise of the stainless steel cladding if the fuel temperature is not allowed to exceed 2100°F (1150°C) and the fuel element cladding is water cooled

The temperature safety limit for the Standard TRIGA fuel element is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel, which indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will not result in compromise of the stainless steel cladding if the fuel temperature is not allowed to exceed 1830°F (1000°C) and the fuel element cladding is water cooled.

14.2.2 Limiting Safety System Setting

Applicability

This specification applies to the scram settings that prevent the fuel element temperature from reaching the safety limit

Objective

The objective is to prevent the fuel element temperature safety limits from being reached

Specification

a) For steady state operation

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

2) The limiting safety system setting shall be 125% of 1MW as measured on the two high power level safety channels. This safety channel LSSS is more conservative than the temperature LSSS.

b) For pulsing operation

The limiting safety system setting shall be 975°F (525°C) as measured in an instrumented fuel element. The instrument element shall be located adjacent to the central bundle with the exception of the corner positions Pulsing is not allowed if this limiting safety system channel is not operable.

Basis

The limiting safety system setting (LSSS) is a temperature or reactor power setting that, if exceeded, will cause a reactor scram to be initiated preventing the safety limit from being exceeded.

The temperature safety limit for TRIGA-FLIP or LEU fuel is 2100°F (1150°C), while the limit for Standard TRIGA fuel is 1830°F (1000°C) Due to various errors in measuring temperature in the core, it is necessary to arrive at a Limiting Safety System Setting (LSSS) for the fuel element safety limit that takes into account these measurement errors. In doing this analysis, the NSC picked the Standard fuel element temperature limit as the base case to derive the LSSS for all fuel types. One category of error between the true temperature value and the measured temperature value is due to the accuracy of the fuel element channel and any overshoot in reactor power resulting from a reactor transient during steady state mode of operation Although a lesser contributor to error, a minimum safety margin of 10% was applied, on an absolute temperature basis. Adjusting the Standard fuel temperature safety limit to degrees Kelvin, °K, and applying a 10% safety margin results in a safety limit reduction of 150°C. Applying this first margin of safety, the safety setting would be 850°C for Standard fuel and 1000°C for FLIP or LEU fuel. However,

to arrive at the final LSSS it is necessary to allow for the major difference between the measured temperature value and the true temperature value (peak core temperature), which is a function of the location of the thermocouple within the core. For example, if the thermocouple element were located in the hottest position in the core, the difference between the true and measured temperatures would be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. However, at the NSC this core position is not available due to the location of the transient rod For the NSC the location of the instrumented elements is therefore restricted to the positions closest to the central element Calculations indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than 40%. When applying this 40% worse case measurement scenario to Standard fuel and considering the previously mentioned sources of error between the true and measured values a final LSSS temperature of 975°F (525°C) is imposed on operation with all types of fuel. Viewed on an absolute temperature scale, °K, this represents a 37% safety margin in the FLIP/LEU fuel safety limit and a 44% reduction in the Standard fuel safety limit.

In the pulse mode of operation, the temperature limiting safety system setting will apply. However, the temperature channel will have no effect on limiting peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting the "tail" off the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

The reactor high power level safety limit for TRIGA-FLIP, LEU or Standard fuel during steady state operation will be a measured power of 125% (1.25 MW) on either of the two power safety channels. The high power safety drawers are aligned with the linear power monitor during annual calibration for nominal 1.0 MW operation These safety channels independently measure reactor power and have been a part of Texas A&M University reactor operation safety systems for over 30 years During the years of 1MW operation of the TAMU TRIGA reactor the LSSS temperature limit of 975°F (525°C) has never been reached although several scrams of the safety channels have been recorded indicating that the LSSS of 125% for the high power level safety channels is more conservative than the temperature setting.

14.3 Limiting Conditions for Operation

14.3.1 Reactor Core Parameters

14.3.1.1 Steady State Operation

Applicability

This specification applies to the energy generated in the reactor during steady state operation.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded during steady state operation.

Specifications

The reactor power level shall not exceed 1.3 megawatts (MW) under any condition of operation. The normal steady state operating power level of the reactor shall be 1.0–1.2 MW. However, for purposes of testing and calibration, the reactor may be operated at higher power levels not to exceed 1.3 MW during the testing period.

Basis

Thermal and hydraulic calculations indicate the TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with natural convection cooling.

14.3.1.2 Pulse Mode Operation

Applicability

This specification applies to the peak temperature generated in the fuel as the result of a pulse insertion of reactivity.

Objective

The objective is to assure that respective pulsing will not induce damage to the reactor fuel

Specification

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

b) Until any new fuel core has been calibrated, maximum pulse shall be limited to $2.00 After calibration, a new maximum pulse insertion value shall be adhered to.

Basis

TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1 6 for FLIP fuel and 1.65 for standard. This yields delta phase zirconium hydride that has high creep strength and undergoes no phase changes at temperatures over 1000°C. However, after extensive steady state operation at 1 MW the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of the expected for $ZrH_{1.6}$ If the pulse insertion is such that the temperature of the fuel exceeds 874°C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grows with each pulse The pulsing limit of 830°C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature. The decrease in temperature from 874°C to 830°C reduces hydrogen pressure by a factor of two, which is an acceptable safety factor. This phenomenon does not alter the safety limit since the total hydrogen in a fuel element does not change Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the element.

In practice, the pulsing limit of 830°C will be translated to a reactivity insertion limit for each specific core. The peaking factors from the thermocouple element to the hottest spot in the core must be calculated for each core configuration that is to be used. Temperature would then be measured for small pulse insertions

For new uncalibrated cores, the pulse insertions shall be increased by small increments to a maximum of $2.00 to allow an extrapolation of peak temperatures, thereby establishing the maximum allowed pulse insertion for a given core. Following approval by the NRC staff of the calibration of the new core, the $2.00 restriction shall be removed.

14.3.1.3 Shutdown Margin

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. They apply for all modes of operation.

Objective

The objective is to assure that the reactor can be shutdown at all times and to assure that the fuel temperature safety limit will not be exceeded.

Specifications

The reactor shall not be operated unless the shutdown margin provided by control rods is greater than $0.25 with·

a) The highest worth non-secured experiment in its most reactive state,

b) The highest worth control rod and the regulating rod (if not scrammable) fully withdrawn. and

c) The reactor in the cold condition without xenon

Basis

The value of the shutdown margin assures that the reactor can be shut down from any operating condition even if the highest worth control rod should remain in the fully withdrawn position If the regulating rod is not scrammable, its worth is not used in determining the shutdown reactivity.

14.3.1.4 Core Configuration Limitation

Applicability

This specification applies to mixed cores of FLIP, LEU and standard types of fuel and to full FLIP, LEU or standard cores.

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded due to power peaking effects in full FLIP, LEU, or Standard cores and in mixed fuel cores

Specifications

a) The TRIGA core assembly may be Standard, FLIP, LEU or a combination thereof (mixed core) provided that any FLIP and/or LEU fuel core be comprised of at least thirty-five (35) FLIP and/or LEU fuel elements, located in a contiguous, central region

b) The instrumented element, if present and serving as the Limiting Safety System, shall be located adjacent to the central bundle with the exception of the corner positions

Bases

a) In mixed cores, it is necessary to specify the minimum number of FLIP and/or LEU elements and arrange them in a contiguous, central region of the core to control flux peaking and power generation values in individual elements.

b) Reference· 14.2.2 Limiting Safety System Setting

14.3.1.5 Maximum Excess Reactivity

Applicability

This specification applies to the maximum excess reactivity, above cold critical, which may be loaded into the reactor core at any time.

Objective

The objective is to ensure that the core analyzed in the safety analysis report approximates the operational core within reasonable limits.

Specifications

The maximum reactivity in excess of cold, xenon-free critical shall not exceed 5.5% $\Delta k/k$ ($7.85)

Basis

Although maintaining a minimum shutdown margin at all times ensures that the reactor can be shut down, that specification does not address the total reactivity available within the core This specification, although over-constraining the reactor system, helps ensure that the licensee's operational power densities, fuel temperatures and temperature peaks are maintained within the evaluated safety limits The specified excess reactivity makes up for negative reactivity due to power coefficients, xenon poisoning, experiments and fuel depletion

14.3.2 Reactor Control and Safety Systems

14.3.2.1 Reactor Control Systems

Applicability

This specification applies to the information that must be available to the reactor operator during reactor operation

Objective

The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor

Specifications

The reactor shall not be operated unless the measuring channels listed in the following table are operable

Measuring Channel	*Min. No. Operable*	*Operating Mode*	
		S.S	*Pulse*
High Power Level	2	X	
Fuel Element Temperature	1		X
Linear Power Level	1	X	
Log Power Level	1	X	
Integrated Pulse Power	1		X

Bases

Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level monitors assure that the reactor power level is adequately monitored for both steady state and pulsing modes of operation. Monitoring of the high power level channel is important since it is used to ensure the temperature safety limit is not reached, since the power level is related to the fuel temperature.

14.3.2.2 Reactor Safety Systems

Applicability

This specification applies to the reactor safety system circuits.

Objective

The objective is to specify the minimum number of reactor safety system channels that must be operable for safe operation

Specifications

The reactor shall not be operated unless the safety circuits described in the following table are operable

Safety Channel	*Number Operable*	*Function*	*Effective Mode*	
			S.S	*Pulse*
Fuel Element Temperature	1	SCRAM @ LSSS (975°F)		X
High Power Level	2	SCRAM @ LSSS (125%)	X	
High Power Level Detector Power Supply	2	SCRAM on loss of supply voltage, or low power supply	X	
Console Scram Button	1	SCRAM at operator's discretion.	X	X
Preset Timer	1	Transient rod scram time to be 15 seconds or less after pulse.		X
Log Power	1	Prevent withdrawal of shim safeties at <4 $\times 10^{-3}$ W (Low count interlock).	X	
Transient Rod position	1	Prevent application of air in steady state mode unless transient rod is fully inserted.	X	
Shim Safeties & Regulating Rod Position	1	Prevent withdrawal of shim safeties and regulating rod while in pulse mode		X

Bases

The fuel temperature and high power level scrams provide protection to assure that the reactor can be shutdown before the safety limit on fuel element temperature will be exceeded.

In the event of failure of the power supply for a high power level safety channel, operation of the reactor without adequate instrumentation is prevented.

The manual console scram allows the operator to shut down the system if an unsafe or abnormal condition occurs.

The preset timer insures that the reactor power level will reduce to a low level after pulsing

The interlock to prevent startup of the reactor at power levels less than 4×10^{-3} W which corresponds to approximately 2 cps assures that sufficient neutrons are available for proper startup

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

The interlock to prevent the withdrawal of the shim safeties or regulating rod in the pulse mode is to prevent the reactor from being pulsed while on a positive period.

14.3.2.3 Scram Time

Applicability

This specification applies to the time required for the scrammable control rods to be fully inserted from the instant that the fuel temperature safety channel or the high power level safety channel variable reaches their respective Limiting Safety System Setting.

Objective

The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification

The scram time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 1 2 seconds

Basis

This specification assures that the reactor will be promptly shutdown when a scram signal is initiated Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor

14.3.3 Confinement

14.3.3.1 Operations that Require Confinement

Applicability

This specification applies to confinement requirements during operation of the reactor and the handling of radioactive materials.

Objective

To maintain normal or emergency airflow into and out of the reactor building during operations that produce or could potentially produce airborne radioactivity.

Specification

Confinement of the reactor building will be required during the following operations.

a) Reactor operating.

b) Handling of radioactive materials with the potential for airborne release

Note: For periods of maintenance to the central exhaust fan, entry doors to the reactor building will remain closed except for momentary opening for personnel entry or exit.

Bases

a) This basis applies during the conduct of those activities defined as reactor operations. Argon-41 is produced during operation of the reactor in experimental facilities and in the reactor pool; thus, air control within the building and the exhaust system in necessary to maintain proper airborne radiation levels in the reactor building and release levels in the exhaust stack. Other radioactivity releases to the reactor building must be considered during reactor operation, such as fission product release from a leaking fuel element or a release from fixed experiments in or near the core

b) The handling of radioactive materials can result in the accidental or controlled release of airborne radioactivity to the reactor building environment or direct release to the building exhaust system. In these cases, the control of air into and out of the reactor building is necessary.

14.3.3.2 Equipment to Achieve Confinement

Applicability

This specification applies to the equipment and controls needed to provide confinement of the reactor building.

Objective

The objective is to assure that a minimum of equipment is in operation to achieve confinement as specified in 3.3.1 and that the control panel for this equipment is available for normal and emergency situations.

Specifications

a) The minimum equipment required to be in operation to achieve confinement of the reactor building shall be the central exhaust fan

b) Controls for establishing the operation of the ventilation system during normal and emergency conditions shall be located in the reception room

Note: During periods of maintenance to the central exhaust fan, entry doors to the reactor building will remain closed except for momentary opening for personnel entry or exit

Bases

a) Operation of the central exhaust fan will achieve confinement of the reactor building during normal and emergency conditions when the controls for air input are set such that the central exhaust fan capacity remains greater than the amount of air being delivered to the reactor building. The exhaust fan has sufficient capacity to handle extra air intake to the building during momentary opening of doors.

b) The control panel for the ventilation system provides for manual selection of air input to the reactor building and the automatic or manual selection of air removal. The air-supply and exhaust systems work together to maintain a small negative pressure in the reactor building. These controls are located in the reception room for accessibility during emergency conditions.

14.3.4 Ventilation System

Applicability

This specification applies to the operation of the facility ventilation system.

Objective

The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation

Specification

The reactor shall not be operated unless the facility ventilation system is operable, except for periods necessary to permit repair of the system. In the event of a substantial release of airborne radioactivity, the ventilation system will be secured automatically by signals from the appropriate facility air monitor.

Basis

During normal operation of the ventilation system, the concentration of Argon-41 in unrestricted areas is below the effluent concentration (Section 11). In the event of a substantial release of airborne radioactivity, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system shutdown for short periods of time to make repairs insures the same degree of control of release of radioactive materials. Moreover, facility air monitors within the building independent of those in the ventilation system will give warning of high levels of radiation that might occur during operation with the ventilation system secured

14.3.5 Radiation Monitoring Systems and Effluents

14.3.5.1 Radiation Monitoring

Applicability

This specification applies to the radiation monitoring information that must be available to the reactor operator during reactor operation.

Objective

The objective is to assure that sufficient radiation monitoring information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the radiation monitoring channels listed in the following table are operable.

Radiation Monitoring Channels	*Function*	*Number*	*Channel Backup During Maintenance*
Area Radiation Monitor (ARM) – Reactor Bridge	Monitor radiation levels within the reactor bay	1	Area Radiation Monitor (ARM) – Material Handling Area
Facility Air Monitor (FAM) – Fission Products	Monitor radiation levels within the reactor bay	1	Facility Air Monitor (FAM) – Building Particulates
Facility Air Monitor (FAM) – Stack Exhaust Gas	Monitor radiation levels of the stack exhaust gases	1	Facility Air Monitor (FAM) – Building Gas
Facility Air Monitor (FAM) – Stack Exhaust Particulates	Monitor radiation levels of the stack exhaust particulates	1	Facility Air Monitor (FAM) – Building Particulates

Note: For periods of maintenance to the Radiation Monitoring Channels, the intent of this specification will be satisfied if they are replaced with the Channel Backups listed in the preceding table. If two of the above Radiation Monitor Channels are not operating, the reactor shall be shutdown.

Bases

The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

14.3.5.2 Argon-41 Discharge Limit

Applicability

This specification applies to the concentration of Argon-41 that may be discharged from the TRIGA reactor facility.

Objective

To insure that the health and safety of the public is not endangered by the discharge of Argon-41 from the TRIGA reactor facility.

Specification

The concentration of Argon-41 in the effluent gas from the facility as diluted by atmospheric air in the lee of the facility due to the turbulent wake effect shall not exceed 1.0×10^{-8} µCi/ml averaged over one year.

Basis

The maximum allowable concentration of Argon-41 (^{41}Ar) in air in unrestricted areas as specified in Appendix B, Table II of 10CFR20 is 1.0×10^{-8} µCi/ml Section 11 of the SAR for the NSCR substantiates a 5.0×10^{-3} atmospheric dilution factor for a 2.0 mph wind speed. This dilution factor represents the conditions at the site building for a wind speed of 2.0 mph, which occurs less than 10% of the time on an annual basis.

14.3.5.3 Xenon and Iodine Monitoring

Applicability

This specification applies to the radiation monitoring systems necessary to monitor and control the concentration of any effluent releases during the production of ^{125}I from the radioactive decay of ^{125}Xe

Objective

The objective is to assure that sufficient radiation monitoring information is available to the operator to insure that the health and safety of the general public is not endangered during the production of ^{125}I.

Specification

No experiment that involves active handling of ^{125}Xe and/or ^{125}I may be performed unless their respective radiation monitoring system is operable. No experiment may be performed, except decay of ^{125}Xe, unless the ^{125}Xe FAM channel is operable.

Radiation Monitoring Channels	*Function*	*Number*
Facility Air Monitor (FAM) – Stack Exhaust Gas (^{125}Xe)	Monitors ^{125}Xe radiation level in stack exhaust gases	1
^{125}I Air Monitor	Monitors work area ^{125}I radiation levels	1

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

Basis

The required facility air monitors (14.3.5.1) are not calibrated for ^{125}I The ^{125}Xe FAM channel provides information to operators in the event there is a significant release of ^{125}Xe during the production of ^{125}I The ^{125}I air monitor measures ^{125}I radiation levels in the immediate area where the iodine handling is being performed

14.3.6 Limitations on Experiments

14.3.6.1 Reactivity Limits

Applicability

This specification applies to the reactivity limits on experiments installed in the reactor and its experimental facilities.

Objective

The objective is to assure control of the reactor during the handling of experiments adjacent to or in the reactor core.

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist

a) The reactivity worth of any single, non-secured experiments shall be less than one dollar.

b) The reactivity worth of any single experiment shall be less than two dollars

Bases

a) This specification is intended to provide assurance that the worth of a single unfastened experiment will be limited to a value such that the safety limit will not be exceeded if the positive worth of the experiment were suddenly inserted This does not restrict the number of non-secured experiments adjacent to or in the reactor core.

b) The maximum worth of a single experiment is limited so that its removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit Since experiments of such worth must be fastened in place, its removal from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained.

14.3.6.2 Material Limitations

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities

Objective

The objective is to prevent damage to the reactor or excessive release of radioactivity by limiting materials quantity and radioactive material inventory of the experiment.

Specifications

a) Explosive materials in quantities greater than 5 pounds shall not be allowed within the reactor building. Irradiation of explosive materials shall be restricted as follows:

1) Explosive materials in quantities greater than 25 milligrams shall not be irradiated in the reactor pool. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container.

2) Explosive materials in quantities greater than 25 milligrams shall be restricted from the reactor pool, the upper research level, the demineralizer room, cooling equipment room and the interior of the pool containment structure.

3) Explosive materials in quantities greater than 5 pounds shall not be irradiated in experimental facilities.

4) Cumulative exposures for explosive materials in quantities greater than 25 milligrams shall not exceed 10^{12} n/cm^2 for neutron or 25 Roentgen for gamma exposures.

b) Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 10 Ci.

Bases

a) This specification is intended to prevent damage to the reactor or reactor safety systems resulting from failure of an experiment involving explosive materials.

1) This specification is intended to prevent damage to the reactor core and safety related reactor components located within the reactor pool in the event of failure of an experiment involving the irradiation of explosive materials. Limited quantities of less than 25 milligrams and proper containment of such experiment provide the required safety for in-pool irradiation.

2) This specification is intended to prevent damage to vital equipment by restricting the quantity and location of explosive materials within the reactor building. Explosives in quantities exceeding 25

milligrams are restricted from areas containing the reactor bridge, reactor console, pool water coolant, and purification systems and reactor safety related equipment

3) The failure of an experiment involving the irradiation of up to 5 pounds of explosive material in an experimental facility located external to the reactor pool structure will not result in damage to the reactor or the reactor pool containment structure

4) This specification is intended to prevent any increase in the sensitivity of explosive materials due to radiation damage during exposures

b) The 10 Ci limitation on Iodine 131 through 135 assures that in the event of failure of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR 20 for an unrestricted area.

14.3.6.3 Failures and Malfunctions

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

a) Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor building or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10CFR20.

b) In calculations pursuant to a) above, the following assumptions shall be used·

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

2) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0 3 micron particles at least 10% of these vapors can escape

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

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

Bases

a) This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B of 10 CFR 20 will be released to the atmosphere outside the facility boundary of the NSC.

b) These assumptions are used to evaluate the potential airborne radioactivity release due to an experiment failure.

c) Operation of the reactor with reactor fuel or structure damage is prohibited to avoid release of fission products. Potential damage to reactor fuel or structure must be brought to the attention of the Director (NSC) or his designated alternate for review to assure safe operation of the reactor.

14.3.6.4 Xenon Irradiation for Iodine Production

Applicability

This specification applies to the experiments that produce I-125 from the activation of enriched ^{124}Xe and the decay of ^{125}Xe.

Objective

The objective is to prevent excessive release of radioactivity by limiting the quantity and radioactive material inventory of the experiment.

Specifications

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

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

Bases

a) The 2000 Ci limitation on Xenon-125 produced in any one experiment assures that in the event of a failure of an experiment leading to the accidental release of xenon, the exposure to the general public would be less than 0.05 rem per year (10 CFR 20 1301).

b) Xenon-125 production in excess of this limit is not necessary.

14.3.7 As Low As Reasonably Achievable (ALARA) Radioactive Effluents Released

Applicability

This specification applies to the measures required to ensure that the radioactive effluents released from the facility are in accordance with ALARA criteria.

Objective

The objective is to limit the annual radiation exposure to the general public resulting from operation of the reactor to a level as low are reasonably achievable below the limits listed in 10 CFR 20.1301.

Specifications

1) In addition to the radiation monitoring specified in Section 14.5.4, an environmental radiation-monitoring program shall be conducted to measure the integrated radiation exposure in and around the environs of the facility on a quarterly basis

2) The annual radiation exposure (dose) to the public due to reactor operation shall not exceed the limits defined in 10 CFR 20 1301. The facility perimeter shall be monitored to ensure this specification is being met.

3) The total annual discharge of Argon-41 into the environment may not exceed 30 Ci per year unless permitted by the RSB.

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

5) Before discharge, the facility liquid effluents collected in the holdup tanks shall be analyzed for the nature and concentration of radioactive effluents The total annual quantity of radioactivity in liquid effluents (above background) shall not exceed 1 Ci per year.

Basis

The simplest and most reliable method of ensuring that ALARA release limits are accomplishing their objective of minimal facility-caused radiation exposure to the general public is to actually measure the integrated radiation exposure in the environment on and off the site

14.3.8 Primary Coolant Conditions

Applicability

This specification applies to the quality of the primary coolant in contact with the fuel cladding

Objective

The objectives are (1) to minimize the possibility for corrosion of the cladding on the fuel elements and (2) to minimize neutron activation of dissolved materials.

Specifications

1) Conductivity of the bulk pool water shall be no higher than 5×10^{-6} mhos/cm (5 μsiemens/cm) for a period not to exceed two weeks

2) The pH of the bulk pool water shall be in the range 5.5 and 8.0 (inclusive). Deviations of pH values outside this range shall not exceed a period of two weeks

Bases

A small rate of corrosion continuously occurs in a water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposure during maintenance and operations.

14.4 Surveillance Requirements

14.4.1 General

Applicability

This specification, applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related to reactor safety.

Specifications

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

Basis

This specification relates to changes in reactor systems, which could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, then it can be assumed that they meet the presently accepted operating criteria

14.4.2 Reactor Core Parameters

14.4.2.1 Steady State Operation

Applicability

This specification applies to the surveillance requirement of the power level monitoring channels

Objective

The objective is to verify that the maximum power level of the reactor meets the license requirements

Specification

A channel calibration shall be made of the power level monitoring channels by the calorimetric method annually but at intervals not to exceed 15 months.

Basis

The power level channel calibration will assure that the reactor will be operated at the proper power level.

14.4.2.2 Pulse Mode Operation

Applicability

This specification applies to the surveillance requirements for operation of the reactor in the pulse mode.

Objective

The objective is to verify that operation of the reactor in the pulse mode is proper and safe and to determine if any significant changes in fuel characteristics have occurred.

Specification

The reactor shall be pulsed semiannually to compare fuel temperature measurements and core pulse energy with those of previous pulses of the same reactivity value or the reactor shall not be declared operational for pulsing until such pulse measurements are performed.

Basis

The reactor is pulsed at suitable intervals to make a comparison with previous similar pulses and to determine if changes in fuel or core characteristics are taking place.

14.4.2.3 Shutdown Margin

Applicability

This specification applies to the surveillance requirement of control rod calibrations and shutdown margin

Objective

The objective is to verify that the requirements for shutdown margins are met for operational cores.

Specification

The reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months.

Basis

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worth of experiments inserted in the core. Experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis is adequate to insure no significant changes in the shutdown margin

14.4.2.4 Reactor Fuel Elements

Applicability

This specification applies to the surveillance requirements for the fuel elements.

Objective

The objective is to verify the continuing integrity of the fuel element cladding and to ensure that no fuel damage has occurred

Specification

a) All fuel elements will be inspected visually for damage or deterioration and measured for length and bend within a 5-year period

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

c) The reactor shall not be operated knowingly with damaged fuel.

d) A fuel element shall be considered damaged and must be removed from the core if:

    1) In measuring the transverse bend, the bend exceeds 0.125 inch over the length of the cladding.

    2) In measuring the elongation, its length exceeds its original length by 0.125 inch, or

    3) A clad defect exists as indicated by release of fission products.

Bases

The frequency of inspection and measurement schedule is based on over 30 years of operating experience and on the parameters most likely to affect the fuel cladding of a pulsing reactor operated at moderate pulsing levels and utilizing fuel elements whose characteristics are well known.

The limit of transverse bend has been shown to result in no difficulty in disassembling fuel bundles. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in

touching has occurred without deleterious effects The elongation limit has been specified to assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow

14.4.3 Reactor Control and Safety Systems

14.4.3.1 Reactor Control Systems

Applicability

These specifications apply to the surveillance requirements for reactor control systems

Objective

The objective is to verify the condition and operability of system components affecting safe and proper control of the reactor

Specifications

The control rods shall be visually inspected for deterioration at intervals not to exceed 5 years.

Basis

The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation of the reactor.

14.4.3.2 Reactor Safety Systems

Applicability

These specifications apply to the surveillance requirements for measurements, tests and calibrations of the control and safety systems.

Objective

The objective is to verify the performance and operability of the systems and components that are directly related to reactor safety.

Specifications

a) A channel test of each of the reactor safety system channels for the intended mode of operation shall be performed before each day's operation or before each operation extending more than one day, except for the pool level channel which shall be tested weekly.

b) Whenever a reactor scram caused by high power level or high fuel element temperature occurs, an evaluation shall be conducted to determine whether the fuel element temperature safety limit was exceeded.

c) A calibration of the temperature measuring channels shall be performed semiannually but at intervals not to exceed 8 months.

d) A channel check of the fuel element temperature measuring channel for pulse mode operation and the high level power channels for steady state operation shall be made daily whenever the reactor is operated by recording a measured value of a meaningful temperature or high power level indication.

Basis

Channel tests will assure that the safety system channels are operable on a daily basis or prior to an extended run. Operational experience with the TRIGA system gives assurance that the thermocouple measurements of fuel

element temperatures and the high power level channels have been sufficiently reliable to assure accurate indication of these parameters

14.4.3.3 Scram Time

Applicability

This specification applies to the surveillance of control rod scram times.

Objective

The objective is to verify that all scrammable control rods meet the scram time requirement

Specification

The scram time shall be measured annually but at intervals not to exceed 15 months

Basis

Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly.

14.4.4 Equipment to Achieve Confinement: Ventilation System

Applicability

This specification applies to the building confinement ventilation system

Objective

The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the uncontrolled environment

Specification

During periods of extended operation, or radioactive material handling, the ventilation system shall be verified operable weekly. This specification is not required during periods of non-operation, e.g., holidays, extended maintenance outages.

Basis

Experience accumulated over several years of operation has demonstrated that the tests of the ventilation system on a weekly basis are sufficient to assure the proper operation of the system and control of the release of radioactive material.

14.4.5 Radiation Monitoring Systems and Effluents

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the continuous facility air monitoring (FAM) system

Objective

The objective is to assure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings

Specification

The area radiation monitoring system (ARM) and the facility air monitoring system (FAM) shall be calibrated annually but at intervals not to exceed 15 months and shall be verified to be operable at weekly intervals.

Basis

Experience has shown that weekly verification of area radiation and air monitoring system operations in conjunction with annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span

14.4.6 Experiments

Applicability

This specification applies to the surveillance requirements for experiments installed in the reactor and its experimental facilities and for irradiations performed in the irradiation facilities

Objective

The objective is to prevent the conduct of experiments or irradiations that may damage the reactor or release excessive amounts of radioactive materials as a result of failure.

Specifications

a) A new experiment shall not be installed in the reactor or its experimental facilities until a hazard analysis has been performed and reviewed for compliance with Section 14 6 of the Technical Specifications. Minor modifications to a reviewed and approved experiment may be made at the discretion of the senior reactor operator responsible for the operation with concurrence from a person qualified in health physics. The senior reactor operator and health physicist must review the hazards associated with the modifications and determine that the modifications do not create a significantly different, a new, or a greater safety risk than the original approved experiment

b) The performance of an experiment classified as an approved experiment shall not be performed until a licensed senior operator and a person qualified in health physics has reviewed it for compliance.

c) The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.

Basis

It has been demonstrated over a number of years of experience that experiments and irradiations reviewed by the Reactor Staff and the Reactor Safety Board as appropriate can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

14.5 Design Features

14.5.1 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

a) TRIGA-FLIP Fuel

1) The individual unirradiated FLIP fuel elements shall have the following characteristics

2) Uranium content· maximum of 9 Wt% enriched to nominal 70% Uranium-235.

3) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1 6 H atoms to 1.0 Zr atoms

4) Natural erbium content (homogeneously distributed): nominal 1.5 Wt%.

5) Cladding. 304 stainless steel, nominal 0.020 inch thick.

6) Identification: Top pieces of FLIP elements will have characteristic markings to allow visual identification of FLIP elements employed in mixed cores.

b) Standard TRIGA fuel

The individual unirradiated Standard TRIGA fuel elements shall have the following characteristics:

1) Uranium content: maximum of 9.0 Wt% enriched to a nominal 20% Uranium-235.

2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.7 H atoms to 1.0 Zr atoms.

3) Cladding 304 stainless steel, nominal 0.020 inch thick.

c) TRIGA-LEU Fuel

1) The individual unirradiated LEU fuel elements shall have the following characteristics:

2) Uranium content maximum of 20 Wt% enriched to nominal 20% Uranium-235

3) Hydrogen-to-zirconium atom ratio (in the ZrH_x). nominal 1 6 H atoms to 1 0 Zr atoms.

4) Natural erbium content (homogeneously distributed): nominal 0 59 Wt%.

5) Cladding: 304 stainless steel, nominal 0 020 inch thick.

Bases

a) A maximum uranium content of 9 Wt% in a TRIGA-FLIP element is about 6% greater than the design value of 8.5 Wt%. Such an increase in loading would result in an increase in power density of about 2%. Similarly, a minimum erbium content of 1.1% in an element is about 30% less than the design value. This variation would result in an increase in power density of only about 6%. An increase in local power density of 6% reduces the safety margin by at most ten percent. The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than the value resulting from a hydrogen-to-zirconium ratio of 1 60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad.

When standard and FLIP and/or LEU fuel elements are used in mixed cores, visual identification of types of elements is necessary to verify correct fuel loadings. The accidental rotation of fuel bundles containing standard and FLIP and/or LEU elements can be detected by visual inspection. Should this occur, however, studies of a single LEU element accidentally rotated into a standard fuel region indicate an insubstantial increase in power generation in the LEU element.

b) A maximum uranium content of 9 Wt% in a standard TRIGA element is about 6% greater than the design value of 8 5 Wt%. Such an increase in loading would result in an increase in power density of less than 6%. An increase in local power density of 6% reduces the safety margin by at most 10%. The maximum hydrogen-to-zirconium ratio of 1 8 will produce a maximum pressure within the clad during an accident well below the rupture strength of the clad

c) The Department of Energy (DOE) is facilitating the conversion of HEU reactors such as the Texas A&M University NSC TRIGA reactor. The DOE controls the timetable for conversion to LEU fuel

14.5.2 Reactor Core

Applicability

This specification applies to the configuration of fuel and in core experiments

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments to provide assurance that excessive power densities will not be produced

Specifications

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

b) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water or D_2O.

Bases

a) Standard TRIGA cores have been in use for years and their characteristics are well documented. FLIP cores have been operated at General Atomics and the Puerto Rico Nuclear Center and their operational characteristics are available. General Atomics has also performed a series of experiments using standard and FLIP fuel in mixed cores In addition, studies performed at Texas A&M for a variety of mixed core arrangements and operational experience with mixed cores indicate that such loadings would safely satisfy all operational requirements General Atomics and Texas A&M has done a series of studies documenting the viability of using LEU fuel in TRIGA reactors.

b) The core will be assembled in the reactor grid plate that is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements

14.5.3 Control Rods

Applicability

This specification applies to the control rods used in the reactor core.

Objective

The objective is to assure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

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

b) The regulating control rod need not have scram capability and shall be a stainless rod or contain the materials as specified for shim-safety control rods. This rod may incorporate a fueled follower.

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

Bases

Using neutron absorbing borated graphite, B4C powder or boron and its compounds, satisfies the poison requirements for the control rods. Since the regulating rod normally is a low worth rod, using a solid stainless steel rod could satisfy its function. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to insure mechanical stability during movement and to isolate the poison from the pool water environment. Control rods that are fuel followed provide additional reactivity to the core and increase the worth of the control rod. The use of fueled followers in the FLIP region has the additional advantage of reducing flux peaking in the water filled regions vacated by the withdrawal of the control rods Scram capabilities are provided for rapid insertion of the control rods, which is the primary safety feature of the reactor. The transient control rod is designed for a reactor pulse The nuclear behavior of the air or aluminum follower that may be incorporated into the transient rod is similar to a void. A voided follower may be required in certain core loadings to reduce flux peaking values.

14.5.4 Radiation Monitoring System

Applicability

This specification describes the functions and essential components of the area radiation monitoring (ARM) equipment and the facility air monitoring (FAM) system for continuously monitoring airborne radioactivity.

Objective

The objective is to describe the radiation monitoring equipment available to the operator to assure safe operation of the reactor.

Specification

The radiation monitoring equipment listed in the following table will have these characteristics.

Radiation Monitoring Channel	**14.5.4.1 Detector Type**	**14.5.4.2 Function**
Area Radiation Monitor (ARM)	Gamma sensitive instruments.	Monitor radiation fields in key locations. Alarm and readout in the control room and readout in the reception room.
Facility Air Monitor (FAM) - Particulates	Beta-Gamma sensitive detector.	Monitors concentration of airborne radioactive particulate activity. Alarm and readout in the control room and readout in the reception room
Facility Air Monitor (FAM) – Gases	Gamma sensitive detector.	Monitors concentration of radioactive gases. Alarm and readout in the control room and readout in the reception room

Basis

The radiation monitoring system is intended to provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

14.5.5 Fuel Storage

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core

Objective

The objective is to assure that fuel that is being stored will not become critical and will not reach an unsafe temperature.

Specifications

a) All fuel elements shall be stored in a geometrical array for which the k-effective is less than 0 8 for all conditions of moderation

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

Basis

The limits imposed by Specifications 14 5.5 a and 14 5 5.b are conservative and assure safe storage.

14.5.6 Reactor Building and Ventilation System

Applicability

This specification applies to the building that houses the reactor.

Objective

The objective is to assure that provisions are made to restrict the amount of release of radioactivity into the environment

Specifications

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

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

c) Emergency shutdown controls for the ventilation system shall be located in the reception room and the system shall be designed to shut down in the event of a substantial release of fission products.

Bases

The facility is designed such that the ventilation system will normally maintain a negative pressure with respect to the atmosphere so that there will be no significant uncontrolled leakage to the environment. The free air volume within the reactor building is confined when there is an emergency shutdown of the ventilation system. Controls for startup, emergency filtering, and normal operation of the ventilation system are located in the reception room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reception room minimizing exposure to operating personnel

14.5.7 Reactor Pool Water Systems

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding

Specifications

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

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

c) Pool water inlet and outlet pipes to the heat exchanger shall have emergency covers within the reactor pool for manual shut off in case of pool water loss due to external pipe system failure.

d) A pool level alarm shall indicate loss of primary coolant before or equal to the pool level dropping to 10% below the normal operating level.

Bases

a) This specification is based on thermal and hydraulic calculations which show that the TRIGA-FLIP core can operate continuously in a safe manner at power levels up to 2,700 kW with natural convection flow and sufficient bulk pool cooling. A comparison of operation of the TRIGA-FLIP and standard TRIGA Mark III has shown them to be safe for the above power level Thermal and hydraulic characteristics of mixed cores are essentially the same as that for TRIGA-FLIP and standard cores

b) In the event of accidental siphoning of pool water through inlet and outlet pipes of the demineralizer, skimmer or diffuser systems, the pool water level will drop to no more than 15 feet from the top of the pool.

c) Inlet and outlet coolant lines to the pool heat exchanger terminate at the bottom of the pool. In the event of pipe failure, these lines must be manually sealed from within the reactor pool. Covers for these lines will be stored in the reactor pool. The time required to uncover the reactor core due to failure of a single pool coolant pipe system is 17 minutes.

d) Coolant water loss of 10% or more requires corrective action. This alarm is observed in the reactor control room and in the reception room.

14.6 Administrative Controls

14.6.1 Organization

The Nuclear Science Center is operated by the Texas Engineering Experiment Station (TEES). The Director of the Nuclear Science Center is responsible to the Director of the TEES for the administration and the proper and safe operation of the facility. Figure 10-1 shows the administration chart for the Nuclear Science Center.

The Reactor Safety Board advises the Director of the NSC on all matters or policy pertaining to safety.

The NSC Radiological Safety Officer provides "onsite" advice concerning personnel and radiological safety and provides technical assistance and review in the area of radiation protection.

14.6.1.1 Structure

A line management organizational structure provides administration and operation of the reactor facility.

The Deputy Director of the Texas Engineering Experiment Station (TEES) and the Director of the Nuclear Science Center (NSC) have line management responsibility for adhering to the terms and conditions of the Nuclear Science Center Reactor (NSCR) license and technical specifications and for safeguarding the public and facility personnel from undue radiation exposure The facility shall be under the direct control of the Director (NSC) or a licensed senior reactor operator

14.6.1.1.1 Management Levels

Level 1: Deputy Director TEES (Licensee)· Responsible for the NSCR facility license.

Level 2: Director (NSC)· Responsible for reactor facility operation and shall report to Level 1

Level 3: Senior Reactor Operator on Duty: Responsible for the day-to-day operation of the NSCR or shift operation and shall report to Level 2.

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

14.6.1.1.2 Radiation Safety

A qualified, health physicist has the responsibility for implementation of the radiation protection program at the NSCR. The individual reports to Level 2 management.

14.6.1.1.3 Reactor Safety Board (RSB)

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

14.6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established above.

14.6.1.3 Staffing

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

1) A licensed senior reactor operator and either a licensed reactor operator or trainee shall be present at the facility.

2) A licensed reactor operator or senior reactor operator will be in the Control Room

3) The Director (NSC) or his designated alternate is readily available for emergencies or on call (i.e., capable of getting to the reactor facility within a reasonable time).

4) At least one individual qualified in health physics will be readily available at the facility or on call (i.e., capable of getting to the reactor facility within a reasonable time)

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

1) Administrative personnel

2) Radiation safety personnel

3) Other operations personnel

14.6.1.3 3 The following designated individuals shall direct the events listed:

1) The Director (NSC) or his designated alternate shall direct any loading of fuel or control rods within the reactor core region

2) The Director (NSC) or his designated alternate shall direct any loading of an in-core experiment with a reactivity worth greater than one dollar

3) The senior reactor operator on duty shall direct the recovery from an unplanned or unscheduled shutdown other than a safety limit violation.

14.6.1.4 Selection and Training of Personnel

A training program for reactor operations personnel exists to prepare personnel for the USNRC Operator or Senior Operator examination This training program normally contains twenty hours of lecture, outside study, and requires several reactor startups

14.6.1.4.1 The selection and training of operations personnel shall be in accordance with the following·

1) Responsibility:

a) The Director (NSC) or his designated alternate is responsible for the training and requalification of the facility reactor operators and senior reactor operators.

2) Requalification Program

a) Purpose:
To insure that all operating personnel maintain proficiency at a level equal to or greater than that required for initial licensing

b) Scope.
Scheduled lectures, written examinations and evaluated console manipulations insure operator proficiency.

14.6.1.5 Radiation Safety

Members of the health physics staff routinely perform radiation safety aspects of facility operations, including routine radiation and contamination surveys, and air and water sampling. Chapter 11 details the radiation safety program for this license.

14.6.2 Reactor Safety Board (RSB) Review and Audit Activities

A Reactor Safety Board (RSB) acts as a review panel for new reactor experiments, procedural changes and facility modifications. The RSB thus provides an independent audit of the operations of the Nuclear Science Center. Issues concerning nuclear safety are immediately brought to the attention of the RSB. The University Radiological Safety Office provides Health Physics assistance for the Nuclear Science Center. This organizational arrangement thus provides another independent review of reactor operations (Figure 10-1).

14.6.2.1 RSB Composition and Qualifications

The Reactor Safety Board (RSB) shall consist of at least three voting members knowledgeable in fields that relate to nuclear safety. The RSB shall review, evaluate and make recommendations on safety standards associated with the operational use of the facility. Members of NSC operations and health physics may be ex-officio members on the RSB. The review and advisory functions of the RSB shall include NSCR operations, radiation protection and the facility license. The Chairman of the Reactor Safety Board under the direction of the Deputy Director of TEES shall appoint the board members.

14.6.2.2 RSB Charter and Rules

The operations of the RSB shall be in accordance with a written charter, including provisions for·

1) Meeting frequency: not less than once per calendar year and as frequent as circumstances warrant consistent with effective monitoring of facility activities.

2) Voting rules

3) Quorums

4) Use of subcommittees

5) Review, approval and dissemination of minutes

14.6.2.3 RSB Review Function

The review responsibilities of the Reactor Safety Committee shall include, but are not limited to the following.

1) Review and approval of new experiments utilizing the reactor facilities;

2) Review and approval of all proposed changes to the facility, procedures, license and technical specifications,

3) Determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in Technical Specification;

4) Review of abnormal performance of plant equipment and operating anomalies having safety significance,

5) Review of unusual or reportable occurrences and incidents that are reportable under 10CFR20 and 10CFR50,

6) Review of audit reports; and

7) Review of violations of technical specifications, license, or procedures and orders having safety significance

14.6.2.4 RSB Audit Function

The RSB or a subcommittee thereof shall audit reactor operations and radiation protection programs at least quarterly, but at intervals not to exceed four months Audits shall include but are not limited to the following:

1) Facility operations, including radiation protection, for conformance to the technical specifications, applicable license conditions, and standard operating procedures at least once per calendar year (interval between audits not to exceed 15 months);

2) The retraining and requalification program for the operating staff at least once per calendar year (interval between audits not to exceed 15 months);

3) The facility security plan and records at least once per calendar year (interval between audits not to exceed 15 months);

4) The reactor facility emergency plan and implementing procedures at least once per calendar year (interval between audits not to exceed 15 months).

The licensee or his designated alternate (excluding anyone whose normal job function is within the NSCR) shall conduct an audit of the reactor facility ALARA program at least once per calendar year (interval between audits not to exceed 15 months). The licensee shall transmit the results of the audit to the RSB at the next scheduled meeting.

14.6.3 Procedures

The philosophy of nuclear safety at the Nuclear Science Center assumes that all operations utilizing the reactor will be carried out in such a manner as to protect the health and safety of the public. This philosophy is augmented in practice by detailed, written procedures. All personnel using the facilities of the Nuclear Science Center follow the procedures. The loading or unloading of any core is performed according to detailed written procedures. Startup

and operation of the reactor is also performed according to detailed written procedures

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

1) Startup, operation, and shutdown of the reactor,

2) Fuel and experiment loading, unloading, and movement within the reactor;

3) Control rod removal or replacement,

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

5) Testing and calibration of reactor instrumentation and controls, control rod drives, area radiation monitors, and facility air monitors;

6) Civil disturbances on or near the facility site;

7) Implementation of required plans such as emergency or security plans; and

8) Actions to be taken to correct specific and foreseen potential malfunctions of systems, including responses to alarms and abnormal reactivity changes

The Director (NSC) and the Reactor Safety Board shall make substantive changes to the above procedures effective only after documented review and approval. The Director (NSC) or his designated alternate may make only minor modifications or temporary changes to the original procedures that do not change their original intent All such temporary changes shall be documented and subsequently reviewed by the Reactor Safety Board

14.6.4 Experiment Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures

1) All new experiments or class of experiments shall be reviewed by the RSB and implementation approved in writing by the Director (NSC) or his designated alternate.

2) Substantive changes to previously approved experiments shall be made only after review by the RSB and implementation approved in writing by the Director (NSC) or his designated alternate. The Director (NSC) or his designated alternate may approve minor changes that do not significantly alter the experiment.

14.6.5 Required Actions

14.6.5.1 Action to be Taken in the Event a Safety Limit is Exceeded

In the event a safety limit is exceeded:

1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.

2) An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Board, and reports shall be made to the NRC in accordance with Section 14.6 6.2 of these specifications, and

3) A report shall be prepared which shall include an analysis of the cause and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Board for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

14.6.5.2 Action to be Taken in the Event of a Reportable Occurrence

In the event of a reportable occurrence, the following action shall be taken:

1) NSC staff shall return the reactor to normal operating or shut down conditions If it is necessary to shut down the reactor to correct the occurrence. operations shall not be resumed unless authorized by the Director (NSC) or his designated alternate.

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

3) The Director (NSC) or his designated alternate shall notify the Chairman of the Reactor Safety Board.

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

5) A report shall be made to the NRC in accordance with Section 6.6 2 of these specifications.

6) Occurrence shall be reviewed by the RSB at their next scheduled meeting

14.6.6 Reporting Requirements

14.6.6.1 Annual Report

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

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

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

C) The number of emergency shutdowns and inadvertent scrams, including reasons thereof;

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

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

F) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient.

1) Liquid Waste (summarized on a monthly basis)

a) Radioactivity discharged during the reporting period.

(i) Total radioactivity released (in Curies)

(ii) The Effluent Concentration used and the isotopic composition if greater than 1×10^{-7} μCi/cc for fission and activation products

(iii) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis

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

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

2) Airborne Waste (summarized on a monthly basis)

a) Radioactivity discharged during the reporting period (in Curies) for·

(i) Argon-41

(ii) Particulates with half-lives greater than eight days.

3) Solid Waste

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

b) The total activity involved (in Curies)

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

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

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

14.6.6.2 Special Reports

In addition to the requirements of applicable regulations, reports shall be made to the NRC Document Control Desk and special telephone reports of events should be made to the Operations Center as follows:

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

a) Violation of safety limits (See Required Actions).

b) Any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;

c) Any reportable occurrences as defined in the Specifications. The written report (and, to the extent possible, the preliminary telephone or telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent reoccurrence of the event;

2) A written report within 30 days of.

a) Personnel changes in the facility organization involving Level 1 and Level 2.

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

14.6.7 Records

A daily reactor operations log is maintained by the reactor operator, and contains such information as core loading, experiments in the reactor, time of insertion and removal of experiments, power levels, time of startup and shutdown, core excess reactivity, fuel changes, and reactor instrumentation records.

Records are maintained which indicate the review, approval and conditions necessary for the production of radioisotopes or performance of irradiation experiments

Records of facility operations in the form of logs, data sheets or other suitable forms are retained for the period indicated in the following sections

14.6.7.1 Records to be retained for a Period of at Least Five Years or for the Life of the Component Involved

1) Normal reactor facility operation

2) Principal maintenance operations

3) Reportable occurrences

4) Surveillance activities required by the Technical Specifications

5) Reactor facility radiation and contamination surveys where required by applicable regulations

6) Experiments performed with the reactor

7) Fuel inventories, receipts, and shipments

8) Approved changes in operating procedures

9) Records of meeting and audit reports of the RSB

14.6.7.2 Records to be retained for at Least One Training Cycle

1) Retraining and Requalification of certified operations personnel.

2) Records of the most recent complete cycle shall be maintained for individuals employed

14.6.7.3 Records to be retained for the Lifetime of the Reactor Facility

1) Gaseous and liquid radioactive effluents released to the environs.

2) Off-site environmental monitoring surveys required by the Technical Specifications.

3) Radiation exposure for all personnel monitored.

4) Drawings of the reactor facility.

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Construct a Non-Power Reactor

Texas A&M is not seeking to construct.

15.2 Financial Ability to Operate a Non-Power Reactor

The Nuclear Science Center is part of the Texas A&M Engineering Experiment Station (TEES). TEES provides an annual budget to the NSC designed to maintain the facility available for both academics and research. While the NSC has its own independent budget, it does share personnel with other departments, primarily the Nuclear Engineering Department in the College of Engineering For example, the NSC Director is a professor of Nuclear Engineering and so the NSC pays a portion of his salary while the Nuclear Engineering Department pays the remainder. In this information, there is no effort to take into account the total cost of the shared personnel This information reflects the actual independent budget of the Nuclear Science Center. As a result, this information is subject to change from year to year depending on the needs and resources of the Nuclear Science Center and other departments.

The operating budget for the Nuclear Science Center is not fixed. While TEES provides a fixed amount to the NSC budget, most of the operating budget comes from services provided to commercial users, researchers and educators The information here is for the 2001-2002 fiscal year. These values have not changed significantly for several years

The total operating budget for the 2002 fiscal year was $773,000 Of this, earned commercial income covered to $415,000, State Funding $250,000, earned income through academic services $50,000, earned income through Texas A&M Experiment Engineering Station's Research Enhancement fund $22,000; fees for services paid through Department of Energy's Reactor Sharing program $24,000, and fees for services provided to other Texas A&M Departments $13,000

The largest part of our expenses is salaries and varies with demand for services

TEES administration has been supportive of the reactor facility and continuation of the Nuclear Science Center to support various curricula including Nuclear Engineering curriculum.

Much of the capital equipment funding in recent years has come from the DOE program to update the instrumentation and experimental equipment for non-power reactors.

15.3 Financial Ability to Decommission the Facility

The University of Wisconsin estimated the cost of decommissioning a similar facility in twenty years to be between $3 million and $8 million. TEES is a state agency and will be able to obtain the funding when necessary.

16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

The Texas A&M Nuclear Science Center uses FLIP fuel from a research reactor in Puerto Rico. Chapter 4 of this SAR addresses the measured characteristics of this fuel. Chapter 13 considers this fuel for accident analysis The NSC has had this fuel in operation for over 20 years; the characteristics and operating parameters are well known

16.2 Medical Use of Non-Power Reactors

Texas A&M Nuclear Science Center does not engage in nor is it licensed to engage in any activities for medical use of the facility.