Texas A&M University
Nuclear Science Center
License No. R-83
Docket No. 05000128

Safety Analysis Report

Redacted Version*

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Subject: Response to NRC Requests for Additional Information Questions 1 through 37 (Non-Financial) for the Nuclear Science Center Reactor (NSCR, License No. R-83, Docket 50-128)

To Whom It May Concern:

The Texas A&M University System, Texas Engineering Experiment Station (TEES), Nuclear Science Center (NSC, License No. R-83) operates a LEU, 1MW, TRIGA reactor under timely renewal. In December, 2003 the NSC submitted a Safety Analysis Report (SAR) as part of the license renewal process. In December, 2005 a conversion SAR (Chapter 18) was submitted resulting in an order to convert from the NRC. In July 2009, the NSC submitted an updated SAR, dated June 2009, to the Nuclear Regulatory Commission (NRC). This updated 2009 version of our SAR incorporated the information from the conversion SAR and the startup of the new LEU reactor core. On November 13, 2009 the NRC submitted a Request for Additional Information as a part of the review process. This request included 37 questions about the technical aspects of the NSC's SAR submittal. Attached to this letter is the NSC's updated SAR which includes changes required by the responses to Questions 1-37 of the NRC's non-financial RAIs.

If you have any questions, please contact Jim Remlinger, Jerry Newhouse, or W. Dan Reece at 979-845-7551.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 09, 2011.

Jerry Newhouse
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NRR

Texas Engineering Experiment Station
Texas A&M University

Safety Analysis Report

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

Docket Number 50-128
License Number R-83

May 2011

Abstract

The Texas A&M University System (TAMUS), Texas Engineering Experiment Station (TEES) holds the license for the operation 1.0 MW TRIGA nuclear reactor at the Nuclear Science Center (NSC) in College Station, Texas.

This document supports the renewal of License R-83 and supersedes all previous submittals in Docket 50-128. The purpose of this Safety Analysis Report (SAR) is to provide a description and safety analysis of structures, systems and components in terms of their ability to provide proper operational performance implemented since the initial operation of the Nuclear Science Center Reactor (NSCR) has improved reactor safety and prevented the need for restrictions on reactor operations due to age of structures or equipment.

This SAR is a consolidated and updated safety analysis for the continued operation of the Nuclear Science Center Reactor using LEU 30/20 TRIGA fuel and contains previously reviewed material from the August 1967 and June 1979 SAR and their supplements, as well as the December 2005 HEU to LEU Conversion Report.

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Acronyms

AEC	Atomic Energy Commission
ALARA	As Low as Reasonably Achievable
ARM	Area Radiation Monitor
BOL	Beginning-of-Life
BP	Beam Port
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CHR	Critical Heat Rate
CRDM	Control Rod Drive Mechanism
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
DOE	Department of Energy
EOL	End-of-Life
FAM	Facility Air Monitor
FLIP	Fuel Lifetime Improvement Program
GA	General Atomics
HEU	Highly-enriched Uranium
HVAC	Heating, Ventilation and Air Conditioning
IFE	Instrumented Fuel Element
LEU	Low-enriched Uranium
LOCA	Loss of Coolant Accident
LSSS	Limiting Safety System Setting
MCNP	Monte Carlo N-Particle Code
MHA	Maximum Hypothetical Accident
MTR	Materials Test Reactor
MW	Megawatt
NRC	Nuclear Regulatory Commission
NSC	Nuclear Science Center
NSCR	Nuclear Science Center Reactor
NSF	National Science Foundation
NUREG	Nuclear Regulation
NWS	National Weather Service
PRNC	Puerto Rico Nuclear Center
QA	Quality Assurance
RERTR	Reduced Enrichment for Research and Test Reactors
RSB	Reactor Safety Board
SAR	Safety Analysis Report
SbBe	Antimony Beryllium
SOP	Standard Operating Procedures

SRO	Senior Reactor Operator
TAMU	Texas A&M University
TAMUS	Texas A&M University System
TEDE	Total Effective Dose Equivalent
TEES	Texas Engineering Experiment Station
TLD	Thermoluminescent Dosimetry
TRIGA	Training, Research, and Isotope Production, General Atomics

1 THE FACILITY

1.1 Introduction

The Texas Engineering Experiment Station (TEES) owns and operates a nuclear reactor facility located in College Station, Texas, at Texas A&M University (TAMU) named the Nuclear Science Center (NSC). The facility contains a university-operated research reactor designed to provide a center for the university's students from various disciplines, university researchers, other academic and non-academic research, and commercial users.

The facility, located on the western end of Texas A&M University campus, houses a TRIGA (Training, Research and Isotopes, General Atomic) reactor. The NSCR utilizes low-enriched uranium (LEU) 30/20; TRIGA fuel normally operates at 1.0 MW.

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

1.2 Summary and Conclusions on Principal Safety Considerations

The primary safety feature of a TRIGA-type reactor is the use of a fuel that has a strong, negative, prompt temperature coefficient of reactivity. This limits the excursions from reactivity insertions preventing fuel damage from all credible reactivity accidents. Ejection of the transient rod from the core, while 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.

An operating power level of 1.0 MW results in a decay heat potential in the fuel small enough that a loss of reactor coolant will not result in fuel damage or the release of fission products.

The only case where significant radiation exposure occurs requires the simultaneous failure of fuel 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 one 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 (DBA), which is defined as the loss of integrity of the fuel cladding for one fuel element and the simultaneous loss of pool water resulting in fission product release.

1.3 General Description of the Facility

The TAMU NSC houses the TRIGA reactor in a dedicated building apart from the main university campus. Figure 1-1 is a photograph of the front of the reactor building and adjacent laboratory building.

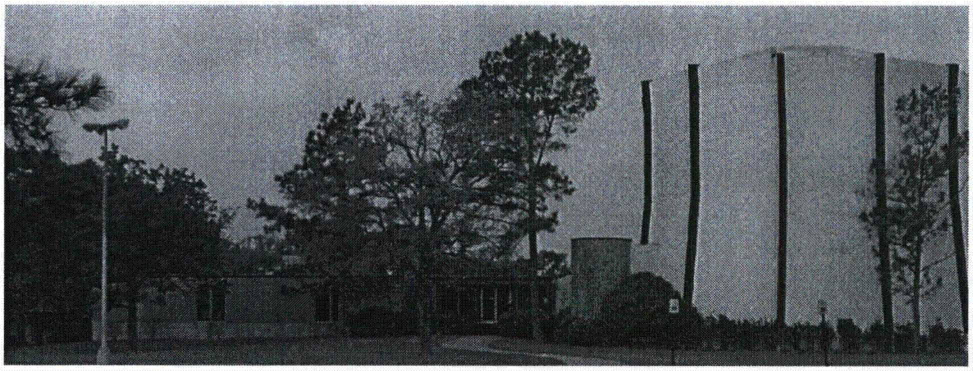

Figure 1-1: Nuclear Science Center Reactor and Laboratory Buildings

The reactor is a 1.0 MW pool-type nuclear reactor. The current configuration uses four-element bundles of TRIGA-LEU 30/20 (30 percent uranium by weight and around 20 percent enriched) fuel. Light water flows through the reactor by natural convection. Graphite serves as a neutron 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 fuel-followed shim safety control rods, a transient control rod and a regulating rod control the reactor's reactivity.

Table 1-1: Summary of Reactor Data

GENERAL REACTOR DATA	
Responsible Organization	Texas Engineering Experiment Station
Location	College Station, TX
Purpose	Training, Research, and Isotope Production
FUEL	
Type	TRIGA LEU
Number of elements (nominal)	[redacted] (Including Fuel-Followed Control Rods)
CONTROL	
Safety Elements	4 Shim Safety Control Rods (Fuel-followed)
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 facility shares a common wall and electrical distribution system with Nuclear Science Center auxiliary shops. This building is external to the confinement building. The Nuclear Science Center shares no other facilities or equipment.

1.5 Comparison with Similar Facilities

No four-rod TRIGA reactor using Fuel Lifetime Improvement Program (FLIP) highly enriched uranium (HEU) fuel had been previously converted to TRIGA LEU 30/20. However, the NSC reactor operated with a TRIGA standard, TRIGA FLIP or FLIP/Standard mixed core between 1968 and 2006. This provides the greatest operational history with which to compare the current LEU core.

Although no empirical data regarding a reactor of this type is available, General Atomics produced safety and accident analyses for the fuel based on the computational methods as described in Section 4. The Puerto Rico Nuclear Center (PRNC) reactor was used to benchmark the computational technique used for evaluating the TRIGA LEU 30/20 fuel in the NSC TRIGA four-rod cluster core.

The reactor instrumentation and control systems are identical to those used in operating the previous NSC reactors with standard and FLIP fuel. These systems have a demonstrated history of reliability and satisfactory safety performance.

1.6 Summary of Operation

The NSCR provides the following:

Laboratory exercises for undergraduate and graduate students at TAMU,

1. Neutron activation analysis facilities for numerous departments at TAMU and other educational institutions without a research reactor,
2. A source of radioisotopes for various research and educational projects,
3. Radioisotopes for several commercial organizations, and
4. 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 (DOE) to the U.S. Nuclear Regulatory Commission dated May 3, 1983, it was determined that all universities operating non-power reactors have entered into a contract with the DOE. The contract provides that DOE retain title to the fuel and the DOE is obligated to take the spent fuel and/or high-level waste for storage or reprocessing. Because TAMU entered into such a contract with the DOE, the TAMU NSC has satisfied the applicable requirements of the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

The initial planning for the NSC began in 1957. At that time, the university embarked on a program of expanding graduate education and research programs. TAMU administrators recognized that a research reactor capable of serving many departments and supporting a large variety of research activities would significantly contribute to this program of expansion.

In March 1958, the application for a construction permit and operating license was submitted 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 Materials Test Reactor (MTR) swimming-pool type reactor at 100 kW.

The reactor first achieved criticality on December 18, 1961. By January 1965, increased use of the facility necessitated operation on a two-shift basis three days per week and one shift operation for two days per week. Only four years passed from initial reactor operation before TAMU implemented a comprehensive upgrade program. In December 1965, the NSC submitted proposals 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, which are briefly described below:

Phase I: Pool Modification and Liner

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

Phase II: Cooling System

> To allow steady state operation at power levels up to 1.0 MW, the NSC provided a cooling system for the reactor. The installed cooling system has a rated capacity of 2.0 MW to facilitate possible future needs.

Phase III: Conversion of the Reactor Core

> The NSC converted the reactor core to utilize standard TRIGA fuel elements, and on July 31, 1968, an amended facility license allowed the NSCR to operate at a maximum steady state power level of 1,000 kW and to pulse 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. The 1.0 MW reactor power improved a number of existing research programs and encouraged the initiation of new projects.

Phase IV: Laboratory Building

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

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. From initiation, the plan to convert the fuel 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. During this time, the reactor routinely operated two shifts for five days per week.

In 1968, the reactor was converted to TRIGA fuel and the maximum steady state power level was increased to 1,000 kW (1.0 MW).[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.

Operating experience with standard TRIGA fuel revealed a high fuel burnup rate resulting in fuel additions to maintain sufficient reactivity. In August 1970, the installation of fuel-followed control rods lead to a gain in excess reactivity helping 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] Modifying the reactor grid plate extended core life by approximately 1 ½ years and provided for the installation of fuel-followed control rods. This modification achieved an average $1.10 increase per fueled follower. The high fuel burnup rate of standard TRIGA cores continued to be an operational problem for the NSCR.

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, TRIGA FLIP fuel.[5] It is almost identical to the standard TRIGA fuel except that the uranium 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 has a calculated lifetime of approximately nine 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 performed by the NSC 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 provided the additional satisfaction of producing substantially greater burnup in the standard fuel used in the mixed core.

In June 1973, the NRC licensed the NSCR to operate full standard, mixed, or full FLIP TRIGA cores. The license for the mixed core permitted the NSC to operate at a maximum steady state power of 1.0 MW with maximum pulse reactivity insertions of $2.00.

Sufficient funds provided for a partial loading of FLIP fuel in a 98-element core with a 35-element FLIP region and 63 standard elements. This configuration achieved criticality in July 1973.[8] The burnup data indicated that the burn up rate was initially 0.5¢ per MW-day and after samarium buildup, the rate dropped to 0.2¢ per MW-day.

The NSCR operated with two mixed core loadings one contained 35 FLIP elements and one contained 59 FLIP elements, since initial approval in June 1973. Thus, the incorporation of FLIP fuel increased the lifetime of the core by a factor of three (Core III). The reactor was not fully converted to a full FLIP core, with [redacted] including four fuel-followed control rods, until 1979. However, in the time between the full core conversions, there were changes made to the allowable maximum pulse rates.

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 gauge test for transverse bend. Steady state hydrogen migration followed by rapid hydrogen pressurization during reactor pulses caused the damage. The reactor was not pulsed again until a complete analysis was submitted to the NRC in 1981. The maximum pulse allowed by the NSCR was reduced to that amount which would not cause the peak reactor temperature to exceed a temperature limit of 830°C (1525°F).

The FLIP core provided a significant improvement in core lifetime. In 2004, the FLIP core began experiencing xenon-precluded startups due to significant fuel depletion. In 2005, the NSC submitted a Safety and Accident Analysis Report prepared in conjunction with General Atomics for a core conversion to TRIGA LEU 30/20 fuel.

In 2006, the NRC issued a conversion order to perform the requested conversion. The core conversion occurred in late 2006. The reactor configuration following the conversion was identical to the configuration used for the previous TRIGA FLIP core. The reactor was declared steady state operational in November of 2006 and pulse-operational in December 2006.

While first bringing the LEU core to full power during startup testing, it was noted that indicated fuel temperature was higher than expected, though still within operational safety limits. The NSC and General Atomics conducted an investigation of the anomaly and determined that the instrumented fuel element providing reactor fuel temperature did not exhibit the expected fuel-cladding gap closure. Although the anomaly presented no safety-related concerns, the NSC prudently replaced the instrumented element. The NSC has operated to date with the installed TRIGA LEU 30/20 fuel without incident.[9]

[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.

[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.

[9] J. Hernandez, "Texas A&M University Nuclear Science Center Reactor Startup Report, TRIGA® 1.0 MW LEU Conversion Reactor", Nuclear Science Center, Texas A&M University, April 2007.

2 SITE CHARACTERISTICS

2.1 Introduction

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

2.2 Geography and Demography

2.2.1 Site Location and Description

2.2.1.1 Specification and Location

The Texas A&M University Nuclear Science Center is located 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), 3 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].

Figure 2-1: NSC and Easterwood Airport

The NSC is surrounded by land owned and controlled by Texas A&M University and Easterwood Airport. A chain-link steel fence, which provides reasonable restriction of access to the site, defines the 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.

2.2.1.2 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.

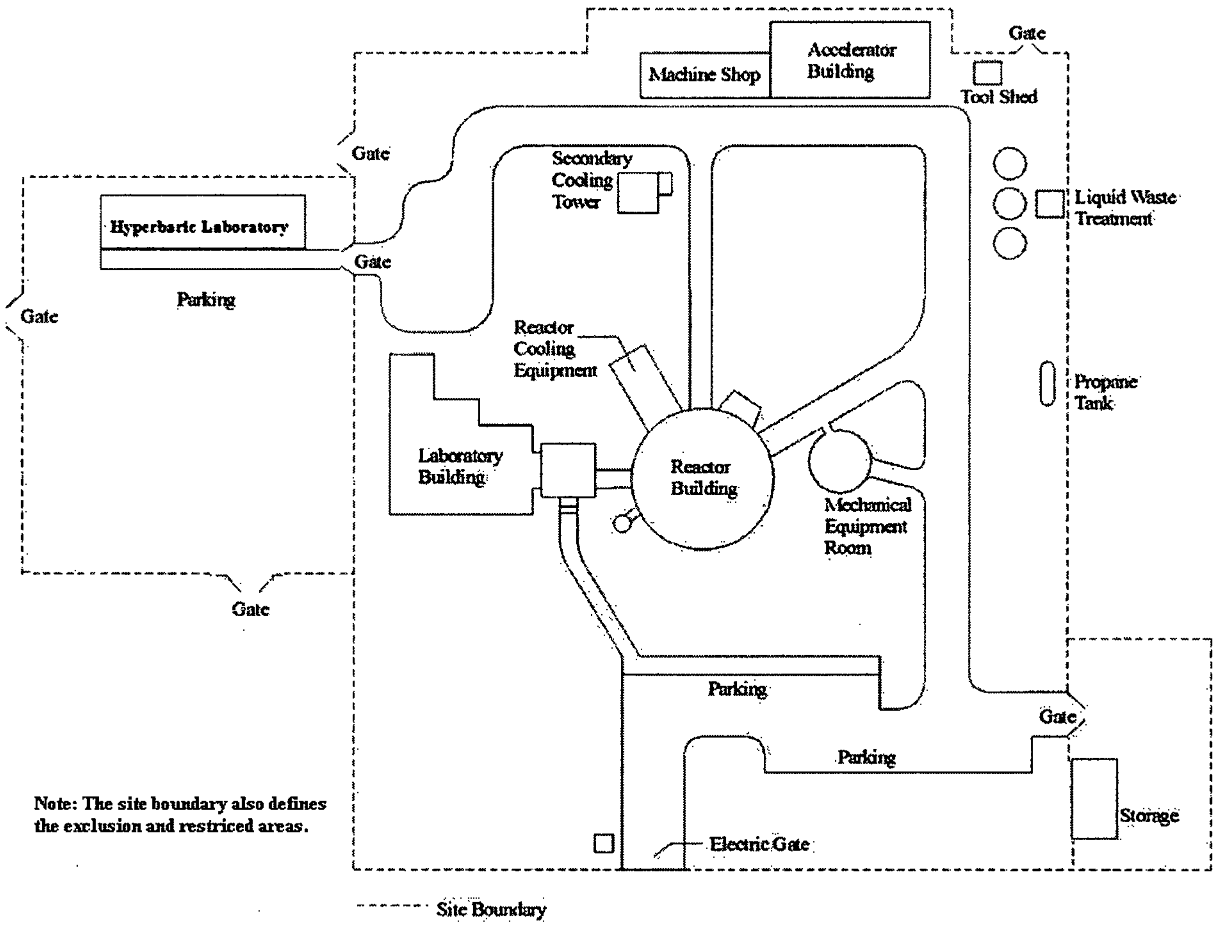

Figure 2-2: NSC Site

A map of the area surrounding the NSC (Figure 2-3) shows the major roads in the area up to a distance of eight 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.

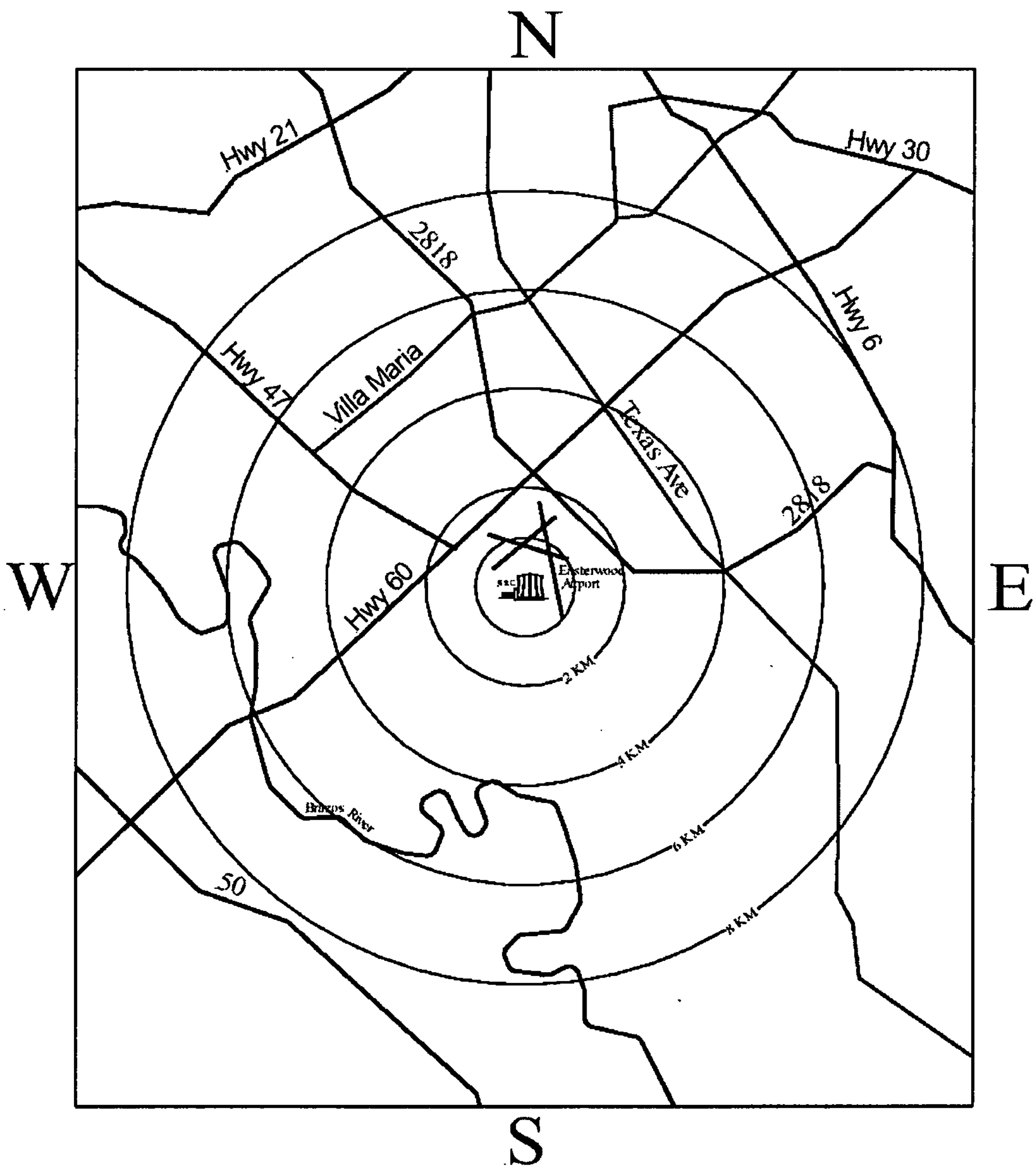

Figure 2-3: Major Roads around NSC and Easterwood

[redacted]

Figure 2-4: NSCR Building Cross Section

[redacted]

Figure 2-5: Upper Research Level

[redacted]

Figure 2-6: Lower Research Level

[redacted]

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

2.2.2 Population Distribution

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

No residences exist or are likely within one km of the NSC, as this property is owned by Texas A&M and Easterwood Airport. Within one kilometer are a firefighter training school and a few other small facilities which together employ dozens of people. Several thousand firefighters attend training throughout the year. The area within two 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 eight km includes much of College Station, part of Bryan and surrounding areas. The population of the eight 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

2.3 Nearby Industrial, Transportation, and Military Facilities

2.3.1 Locations and Routes

No industrial facilities are near the NSC.

The nearest railroad runs through the main campus and at its nearest approach is 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 the main terminal. Refer to Figure 2-1. Easterwood Airport stores up to 44,000 gallons of Jet A fuel in tanks located 1.5 km from the NSC. The tanks are located across two runways and are between several buildings and the NSC. The distance to the tanks and the barrier provided by the runways and buildings help mitigate any fire or concussion effects that an explosion might have on the NSC. Therefore, these tanks do not represent a new type of accident or external event, or change the consequences of an accident already discussed in Chapter 13 of this document.

The only military facility in the Bryan/College Station area is a National Guard facility located in Bryan. The National Guard Facility is located 9.2 km from the Nuclear Science Center and does not represent a new type of accident or external event, or change the consequences of an accident already discussed in Chapter 13.

2.3.2 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 has 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.

2.3.3 Analysis of Potential Accidents at Facilities

No industrial, transportation, or military facilities within the vicinity of the NSC pose sufficient risk to the reactor of rendering 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 and probability of a casualty resulting from an aircraft collision by such an event low.

2.4 Meteorology

2.4.1 General and Local Climate

The Bryan/College Station area is located approximately 160 kilometers (100 miles) inland from the Texas Gulf Coast. The high-pressure areas that are predominant over the Gulf of Mexico largely 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. Calm winds occur an average of 10% of the time, and wind speeds above 38 kph (21 knots) rarely occur.

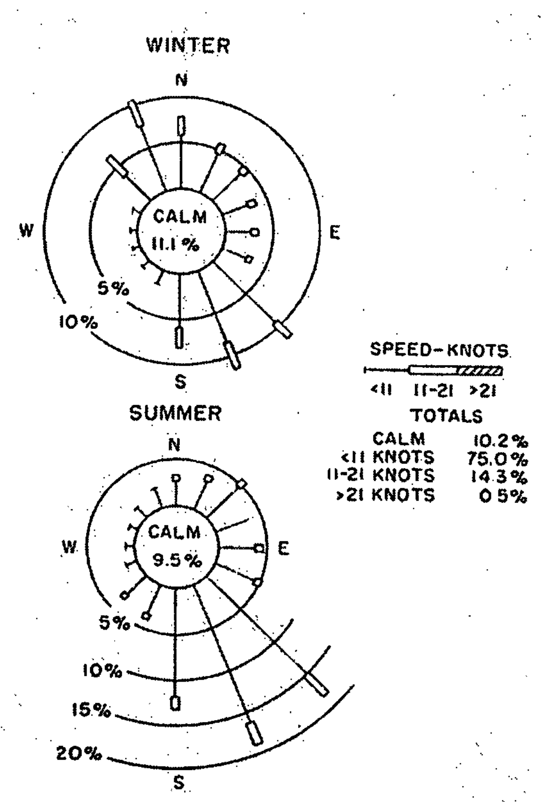

Figure 2-8: Average Wind Frequency Distribution

Tornadoes are common in Texas. Data on tornado frequency between 1950 and 2007 indicates 22 tornadoes in Brazos County Texas during that 57 year period.[1] Fifty percent of those occurred between March and May, and over half occurred in the afternoon between 2:00 and 6:00 pm. The tornado season usually starts in March and reaches its peak in May.

The reactor building design requires that it withstand 207 kPa (30 psi) over pressure with the exception of the domed roof. The roof withstands 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 (72 knots) 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 (5 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).

Since 1950 only two tropical storms and no hurricanes have affected Brazos County and the Nuclear Science Center. The tropical storms occurred in 1998 and 2008.

2.5 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 drinking 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.

2.6 Geology, Seismology, and Geotechnical Engineering

2.6.1 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.

2.6.2 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.

2.6.3 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.[2]

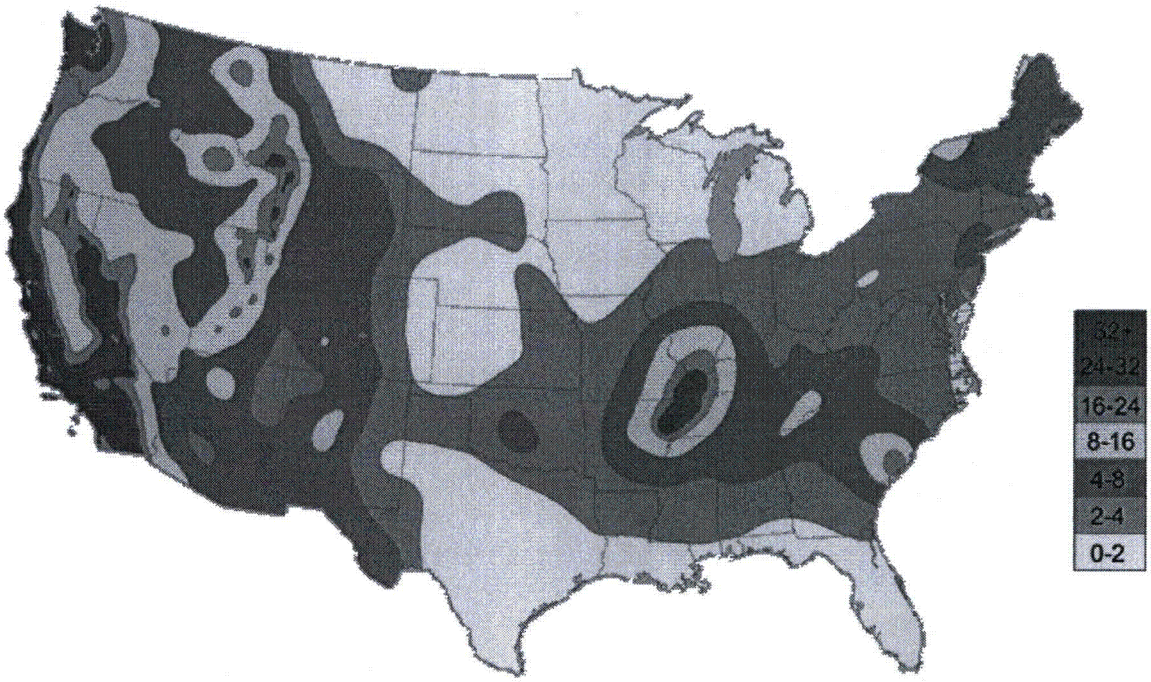

Figure 2-9: Earthquake Probability Map

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

[2] Documentation for the 2002 Update of the National Seismic Hazard Maps, United State Geological Survey, Arthur D. Frankel, Director, Open File Report 02-420, 2002

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 LEU 30/20 fuel with stainless steel cladding meets this criterion. TRIGA LEU 30/20 fuel exhibits 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 can withstand credible accidents with negligible hazard to the public.

The building which houses the NSCR was completed in 1959. The following is criteria for the design required control of the airflow into and out of the NSC. Air enters through ground level suctions and exits through 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 2.0 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. Therefore, there are no criteria for safe operation based on possible weather related damage. The NSC building meets all local meteorological codes and has stood since 1959 without suffering any such damage.

3.3 Water Damage

The accident analysis in Chapter 13 does not use criteria for the condition of the building or equipment 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 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 and Section 2.5 addresses the minimal potential for seismic activity for the region. Therefore, there are no criteria for safe operation based on seismology. This safety analysis report (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 TRIGA-LEU 30/20. Pool water cools the reactor via natural convection, serves as a biological shield, and moderator. The NSC uses graphite moderators on two sides of the core and primarily uses the other two sides for sample irradiation.

The reactor is supported on a nine by six reactor grid. Each location in the grid supports a fuel bundle with four 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 while it is moved 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.

As mentioned in Section 1.5, the Texas A&M University NSC TRIGA LEU 30/20 is the first conversion core of its kind, and no empirical data for a reactor of this type was available prior to core construction. General Atomics (GA) provided safety and accident analysis for the fuel produced using computational methods described in this chapter. The Puerto Rico Nuclear Center (PRNC) TRIGA HEU FLIP core provided an opportunity to benchmark the computational technique to be used for evaluating the TRIGA LEU 30/20 fuel in the NSC TRIGA four-rod cluster core. In addition, empirical data collected during the TAMU NSC core conversion was used to further validate the computational methods used.

The computations produced operational parameters to be compared with the actual measured values from the commissioning tests conducted by General Atomics (GA) for the PRNC TRIGA core loaded with HEU FLIP fuel. The experimentally measured parameters included the 1/M approach to critical tests, the reactivity for the fully loaded core [redacted] plus 1 stainless steel dummy with full water reflection), the control rod calibration values and the reactivity loss and peak fuel temperatures as a function of reactor power. Also included was pulsing performance testing including peak power, peak fuel temperature and energy production all as a function of prompt reactivity insertion. In addition, the computation produced a plot of the prompt negative temperature coefficient of reactivity (Δk/k-°C) versus reactor fuel temperature.

The steady state parameters for the NSC LEU 30/20 core were calculated using the same computational procedures adapted to the NSC four-rod configuration to provide for safety and accident analysis of the NSC core. Finally, calculated results from General Atomics analysis were compared to actual data for the same reactor parameters collected following the NSC core conversion.

As a result of the methods described, the NSC conversion core was declared steady state operational on November 1, 2006. The core was declared pulse operational on December 12, 2006. Comparing all core data to calculated and predicted values, the conversion core parameters were consistent with expectations. All measured values were within 5.0% of calculated values with the exception of rod worth and excess reactivity values for the operational core. These values however, were still within the calculation tolerances as presented in the remainder of this chapter.

Table 4-1 provides a summary of reactor data as calculated by General Atomics for the [redacted] (including four fuel-followed control rods) core configuration currently utilized by the TAMU NSC.

Table 4-1: Summary of Reactor Data, Calculated by GA

DESIGN DATA	
Number of Fuel Rods	[redacted]
Fuel Type	UZrH-Er
Uranium Enrichment, wt %	19.75
Erbium, wt %	0.90
Zirconium Rod Outer Diameter, mm	[redacted]
Fuel Meat Outer Diameter, mm	[redacted]
Fuel Meat Length, mm	[redacted]
Clad Thickness, mm	[redacted]
Clad Material	[redacted]
REACTOR PARAMETERS	
Routine Steady State Power, MW	1.0
Testing, Maintenance, MW	1.3
Maximum Fuel Temperature at 1.0 MW, °C	373
Maximum Pulsing Power with Peak Fuel Temperature ($\hat{T}$) limited to 830 °C, MW	1325
Cold Clean Excess Reactivity, $\Delta k/k\beta$ ($)	7.73
Prompt Negative Temperature Coefficient of Reactivity, $-\Delta k/k°C$, 23-700°C	0.53 to 1.31×10^{-4}
Maximum Rod Power at 1.0 MW, kW/element	17.4
Average Rod Power at 1.0 MW1.0 MW, kW/element	11.1
Maximum Rod Power at DNB = 1.0, kW/element	42
DNB Ratio at Operating Power	2.42
Prompt Neutron Lifetime, µsec	26.3
Effective Delayed Neutron Fraction	0.0070
Shutdown Margin $\Delta k/k\beta$ ($) with Most Reactive Rod and Regulating Rod Stuck Out	-1.15
SAFETY LIMITS	
Limiting Safety System Setting, Measured Fuel Temperature, °C	525
Minimum DNB Ratio at 1.0 MW	2.42
Maximum Positive Pulsed Reactivity Insertion to Reach $\hat{T}$ = 830°C, $\Delta k/k\beta$ ($)	2.10
Peak Pulsed Power, MW	1325
Peak Pulsed Fuel Temperature, °C	830

4.2 Reactor Core

The NSC conversion reactor is a primarily homogeneous, light water moderated and cooled, pool-type reactor with a full core of TRIGA-LEU 30/20 fuel in a four-rod cluster configuration. A five-inch thick aluminum grid plate supports the fuel clusters. Figure 4-1 shows the NSC movable reactor core against the graphite coupler box in the stall location. At the NSC, the two most frequently used reactor locations are:

1. Against the coupler box or
2. 50 to 75 cm removed from the coupler box with light water intervening.

A typical NSC core configuration is shown in Figure 4-2 that contains [redacted], [redacted] control rods, a water-followed regulating rod, and an air-followed transient rod. Graphite reflector blocks are used in-core for reflection in addition to the graphite in the coupler box and thermal column.

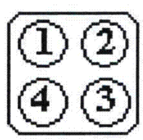
[redacted]

Figure 4-1: Stall Beam Port Installation with Graphite Coupler Box and Thermal Column Extension

[redacted]

Figure 4-2: Core Configuration

Note: Each core location is designated by cluster location (e.g., 5C) and by one of four positions in the cluster. See the following example:



The reactor is supported from the top of the pool wall and is moveable along the central axis of the reactor pool. The reactor is controlled by four fuel-followed control rods plus one water-followed regulating rod and one air-followed transient rod. All six control rods are supported from the bridge structure at the top of the reactor pool wall.

4.2.1 Reactor Fuel

The fuel is arranged 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 4-3) consists of an aluminum bottom adapter, also called a grid plug, a total of four stainless steel clad elements (either fueled or non-fueled elements) and an aluminum top handle. The four elements screw into four tapped holes on the top of the grid plug which fits into the NSC reactor grid. A TRIGA element threads into the grid plug until a flange on the element sits firmly on the adapter providing a cantilever support.

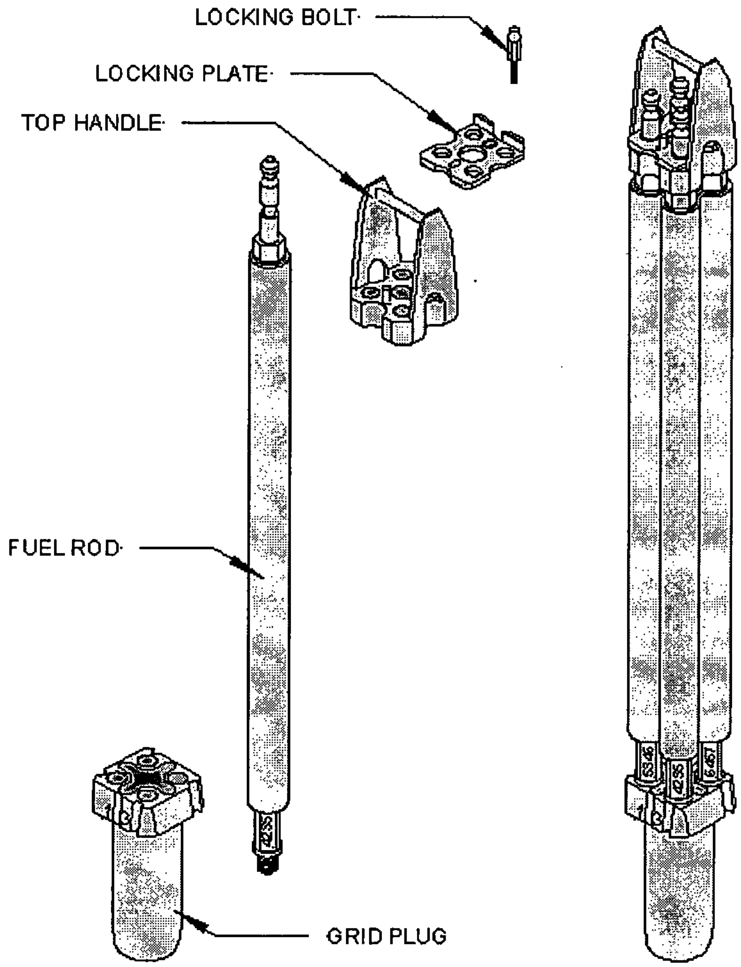


Figure 4-3: Four-Element Bundle

A three-element fuel assembly (Figure 4-4) 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 or without guide tubes.

Figure 4-4 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 because 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 4-5 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 4-6 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 4-7.

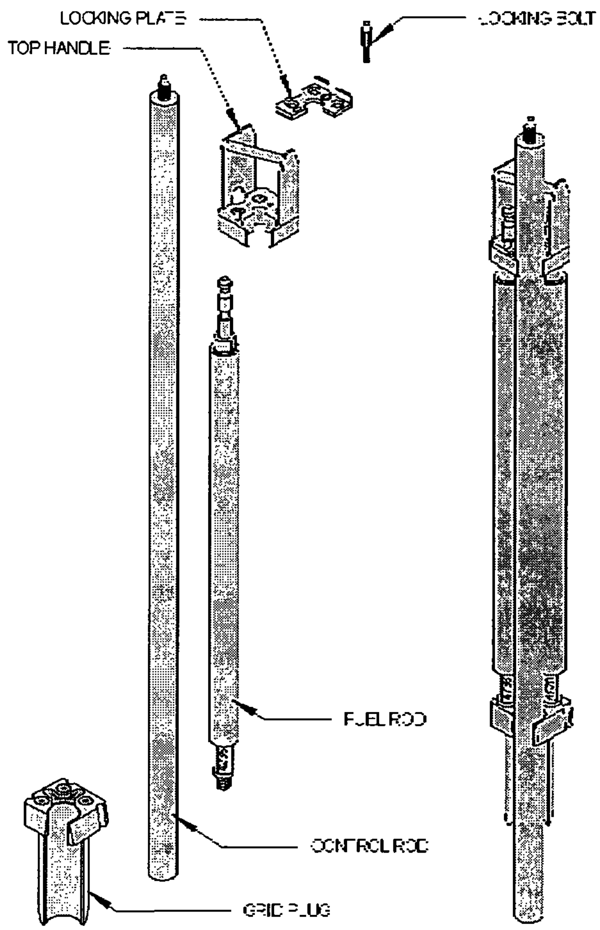


Figure 4-4: Three-Element Bundle with Fuel-Followed Control Rod

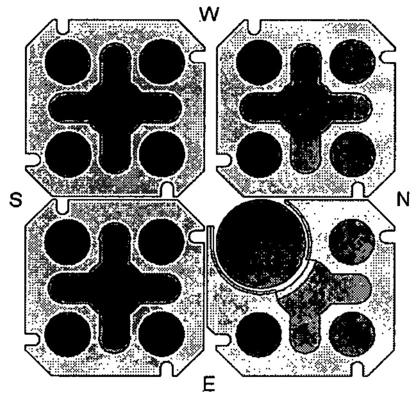


Figure 4-5: Fuel Follower Installation Bottom

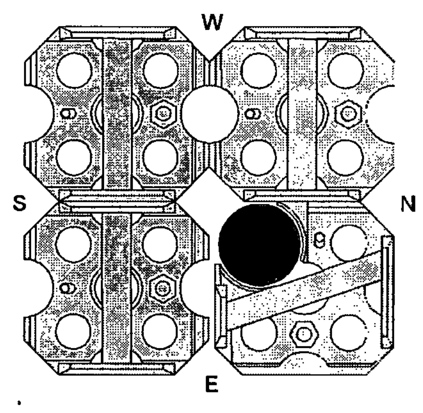


Figure 4-6: Fuel Follower Installation Top

Figure 4-7 shows the three-element assembly that substitutes one fuel element 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 utilize this type of assembly.

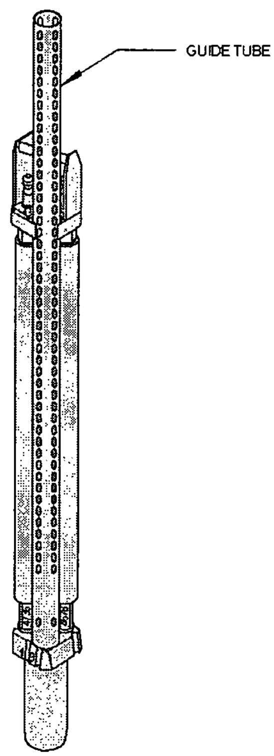


Figure 4-7: Control Rod Bundle with Guide Tube

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 way as a fuel-followed control rod.

The NSCR utilizes TRIGA LEU 30/20 type self-moderated elements. Zirconium hydride, homogeneously combined with partially enriched uranium fuel, provides moderation. Table 4-2 shows the physical characteristics of the fuel along with characteristics for HEU fuel utilized by the PRNC, which was used as a benchmark for computational techniques, utilized in this analysis.

Table 4-2: Description of LEU and HEU Fuel Elements, as Calculated by GA

DESIGN DATA	NSC (LEU)	PRNC (HEU)
Number of Fuel Elements		
Critical Test	[redacted]	[redacted]
Full Load	[redacted]	[redacted]
Fuel Type	U-ZrH 30/20	U-ZrH FLIP
Enrichment, wt %	19.75	70.00
Uranium Density		
g/cm^3	2.14	0.50
wt-%	30	8.42
Number of Fuel Elements per Bundle	[redacted]	[redacted]
^{235}U per Fuel Bundle, g	[redacted]	[redacted]
^{235}U per Fuel Element, g	[redacted]	[redacted]
^{166}Er per Fuel Element, g	7.46	10.27
^{167}Er per Fuel Element, g	5.15	7.09
Erbium, wt-%	0.90	1.48
Zirconium Rod Outer Diameter, mm	[redacted]	[redacted]
Fuel Meat Outer Diameter, mm	[redacted]	[redacted]
Fuel Meat Length, mm	[redacted]	[redacted]
Cladding Thickness, mm	[redacted]	[redacted]
Cladding Material	[redacted]	[redacted]

A 0.18-inch hole in the center of the active section of the fuel elements facilitates hydriding during fabrication. A zirconium rod, inserted after hydriding, fills the hole. As shown in Figure 4-8, 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.

Three thermocouples embedded in the fuel of specially fabricated instrumented elements measure fuel temperature during reactor operation. As shown in Figure 4-9, 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 the extended tubing by Swagelok® unions to provide a watertight conduit protecting the lead wires up to the pool surface.

[redacted]

Figure 4-8: Detailed Drawing of TRIGA LEU 30/20 Fuel Rod

[redacted]

Figure 4-9: Integrated Filament Thermocouple Fuel Rod

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 regulates reactor power during steady state operations and does not have scram capabilities. Section 7.3 details these functions.

The four 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 compositionally 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 4-10 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.

[redacted]

Figure 4-10: Fuel Followed Control Rod

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.

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 4-7 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 B4C 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, which are machined to fit flush against a machined spacer, fit into the grid.

4.2.4 Neutron Startup Source

The neutron startup source (SbBe) is located in the core to provide reasonable multiplication data on the startup channel (Log Power Channel, see 7.2.3.1). The source strength is such that the startup channel will indicate at least two cps prior to startup with the source installed in the core.

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. A cable bundle that lies in a covered trough, which is parallel to the south wall of the reactor pool, has sufficient slack for bridge movement. The bridge accommodates electric power, control-circuit wiring, and compressed air.

Quick disconnect valves are mounted just below floor level 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, facility air monitor (FAM)-2, 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 4-11). 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.

One-quarter inch stainless steel pins attach 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 9 by 6 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.

[redacted]

Figure 4-11: Core Grid Plate **Figure 4-12: Core Configuration VIII-A**

Figure 4-12 shows the core configuration (VIII-A) loading containing [redacted] 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 4-11). 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.

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 5.0 MW, which is well above the current 1.0 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 (Figure 5-3). An aluminum gate can isolate these sections to allow draining only one section. The pool is [redacted] in the main pool. The stall section is 9 feet across and has a 180°curved end with a 4.5 foot radius.

The upper 17 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 ten gauge and the floor is one-quarter inch thick. A ten-inch line provides a drainage system beneath the liner 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 connects 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 system for flexibility of operation. Figure 5-8 shows the pool-water transfer system.

4.4 Biological Shield

Concrete and water serve as the biological shields 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 three feet of 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 diffuser above the core. This forces the 16N in the coolant flowing directly out of the core into the deeper part of the pool, thereby allowing most of the 16N 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 directly above the reactor are less than 10 mR/hr with the reactor operating at 1.0 MW. Radiation levels in the lower research level, with the reactor operating at 1.0 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

The Texas A&M University conversion TRIGA LEU 30/20 reactor is the first four-rod TRIGA reactor previously using HEU FLIP fuel to have been converted to low-enriched uranium. As such, no historical data is available from an operating reactor to provide for comparison of operating and safety characteristics.

The evaluation of the PRNC TRIGA HEU FLIP core provided an opportunity to benchmark the computational technique to be used for evaluating the TRIGA LEU 30/20 fuel in the Texas A&M University TRIGA four-rod cluster core. It also provided information required for the performance comparison of the fresh HEU FLIP and fresh LEU 30/20 fuel for the HEU-to-LEU conversion.

Several factors were considered in deciding to use the PRNC core to validate the model and code inputs, versus the NSC HEU core. NUREG-1538 Chapter 18 requires a performance review of the fresh initial fuel. Fresh HEU FLIP fuel was never loaded into NSC but was transferred as partially depleted fuel from PRNC to the NSC. A complete startup with fresh HEU fuel was conducted for PRNC.

The two cores (PRNC and NSC) were essentially identical reactors, using grid plate layouts, control rods, etc. in parallel configurations. The only essential design difference lies in the reflector region. In order to eliminate any difference due to reflector design, both cores were evaluated with full water reflectors.

Both the PRNC and NSC reactor models were developed by the same modeler using the same level of modeling detail, which assures model fidelity. The MCNP-5 models of both the PRNC and the NSC models were independently verified.

The NSC core uses fuel-followed control rods unlike the PRNC core. Fuel-followed control rods have a higher reactivity worth, which was noted in the calculation. The NSC core has a transient rod with an air follower while the PRNC transient rod has no follower so that water replaces the transient rod when it is pulsed or removed. Peak power density during pulsing occurs next to the transient rod in PRNC but next to the water-filled experiment location at the NSC.

The cores' arrangements are similar, and in fact, the core arrangement presently used at NSC is not the same as the core arrangement when FLIP fuel was first added. FLIP fuel was added to NSC to form a mixed core with 35 FLIP fuel elements. Neutronically, a mixed core of FLIP and standard fuel is different from a core with uniform loading of either FLIP or LEU fuel. The PRNC core is neutronically more similar to a uniform core of LEU fuel than a mixed core of FLIP and standard TRIGA fuel.

A comparison of the FLIP and LEU 30/20 fuel is presented in Table 4-2 which shows a large increase in U-238 due to the decreased enrichment but all other parameters are similar or identical between the two fuel types.

The computations produced operational parameters to be compared with the actual measured values from the commissioning tests conducted by General Atomics for the PRNC TRIGA core loaded with FLIP (HEU) fuel. The experimentally measured parameters included the 1/M approach to critical tests, the reactivity for the fully loaded core [redacted] plus 1 stainless steel dummy with full water reflection), the control rod calibration values, the reactivity loss and peak fuel temperatures as a function of reactor power, and pulsing performance including peak power, peak fuel temperature, and energy production all as a function of prompt reactivity insertion. In addition, the computation produced a plot of the prompt, negative temperature coefficient of reactivity ($\Delta k/k$-°C) versus reactor fuel temperature.

The steady state parameters for the NSC LEU 30/20 core were calculated using the same computational procedures adapted to the NSC four-rod configuration.

4.5.1 Computational Models and Nuclear Analysis Codes

Two-dimensional and three-dimensional calculations were performed using both diffusion theory and Monte Carlo codes. In general, multi-group diffusion theory is used for design calculations since it gives adequate results for systems of this kind and its multi-group fluxes and cross sections are easily utilized in nuclide burnup calculations. The Monte Carlo calculations are used to evaluate the facilities around the core and to compute the worth of core components and different core configurations.

The diffusion theory code is DIF3D,[1,2] a multi-group code which solves the neutron diffusion equations with arbitrary group scattering.

The Monte Carlo code is MCNP-5,[3] which contains its own cross section library.

This section discusses the MCNP-5 models developed for these analyses and the benchmark calculations for the HEU core, and determines a reference critical LEU core. Reactor calculations were performed in three dimensions for the initial criticality and the full core loading of the PRNC HEU core and the NSC LEU core using the MCNP-5, Version 1.3, continuous energy Monte Carlo code. The nuclide cross sections were based on ENDF/B VI data included in the MCNP-5 data libraries.

The PRNC FLIP and NSC LEU 30/20 fuel meat nuclide densities used in the two models are shown in Table 4-3. The other materials beside the fuel used in the PRNC and NSC MCNP-5 models are listed in Table 4-4.

Table 4-3: Nuclide Densities of Fuel Meat Used in the PRNC (FLIP) and NSC (LEU 30/20) MCNP-5 Models

		FLIP		LEU 30/20	
Nuclide	**Atomic Mass**	**Nuclide Density (atoms/b-cm)**	**Mass (g)**	**Nuclear Density (atoms/b-cm)**	**Mass (g)**
H	1.0079	0.05459307	32.05	0.04915763	28.86
C	12.011	0.00149606	10.47	0.00178701	12.50
Zr	91.224	0.03529874	1875.81	0.03227955	1715.37
Er-166	165.93	0.00010624	10.27	0.00007717	7.46
Er-167	166.932	0.00007295	7.09	0.00005299	5.15
U-234	234.041	0.00000659	[redacted]	0.00000715	[redacted]
U-235	235.0439	0.00090116	[redacted]	0.00108821	[redacted]
U-236	236.0456	0.00000423	[redacted]	0.00000627	[redacted]
U-238	238.0508	0.00037065	[redacted]	0.00432194	[redacted]
Hf	178.49	2.11792E-06	0.22	1.93677E-06	0.20
Total		**0.09285181**	[redacted]	**0.08877792**	[redacted]

Table 4-4: Material Composition used in the MCNP-5 Models

Material	Nuclide	Nuclide Density (atoms/b-cm)	Physical Density (g/cc)
[redacted]clad)	Cr-50	0.000778	7.98
	Cr-52	0.015003	
	Cr-53	0.001701	
	Fe-56	0.056730	
	Ni-58	0.007939	
	Mn-55	0.001697	
Graphite (TC)	C		1.7
Graphite (reflector in fuel)	C		1.75
Zirconium (rod)	Zr		6.51
6061 Al (grid plate and control rod clad)	Al-27	0.058693	
	Fe-56	0.000502	
90% B_4C (control rod)	B-10	0.020950	
	B-11	0.084310	
	C	0.026320	
Boral (35wt% B_4C, 65wt% Al), (detector channel)	B-10	0.06031	2.64
	B-11	0.24489	
	C	0.08725	
	Al-27	0.63581	
Al+Water Mix 1 (2" lower cluster adapter)	H	0.028748	
	O	0.014374	
	Al-27	0.033455	
	Fe-56	0.000286	
Al+Water Mix 2 (5" grid plate)	H	0.030788	
	O	0.015394	
	Al-27	0.031663	
	Fe-56	0.000271	
Water			1.0
Air			0.000123

4.5.1.1 Geometric Models

Each fuel rod was explicitly modeled such that fifteen cells and six surface cards were constructed to properly represent one fuel rod. A total of 930 cells and 372 surface cards were made for the [redacted] and a similar number of cells and surface cards for th[redacted] in the approach-to-critical core.

For the full core, the model for PRNC with [redacted] has a total of 1425 cells and 570 surface cards. The full core NSC core model ha[redacted] with a total of 1350 cells and 540 surface cards.

4.5.1.2 PRNC Reactor Model, Approach-to-Critical

A detailed MCNP-5 model of the reactor was made including [redacted], 5 water-followed control rods, 1 rabbit system, the 2 inches thick lower cluster adapter, and the 5 inches thick grid plate below the fuel rods. This number of FLIP fuel elements was chosen to match the number and geometry used by GA in the approach-to-critical tests at PRNC in 1972. The reactor went critical on [redacted].

Figure 4-13 and Figure 4-14 are the XY (top down) and XZ (side view) plots of the MCNP-5 model of the PRNC cold critical case. In this case, the whole core model was infinite water reflected since the initial approach to critical measurements during commissioning used this configuration.

[redacted]

Figure 4-13: PRNC Radial Model for Monte Carlo (MCNP-5 XY Plot)

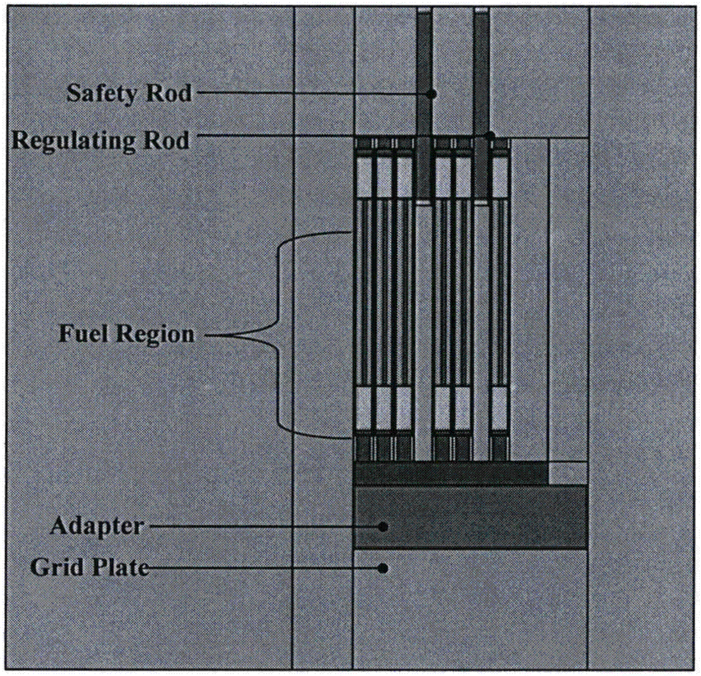


Figure 4-14: PRNC Axial Model for Monte Carlo (MCNP-5 XZ Plot) Transport Calculations - [redacted]

[redacted]

Figure 4-15: NSC Radial Model for Monte Carlo (MCNP-5 XY Plot) Transport Calculation – [redacted]

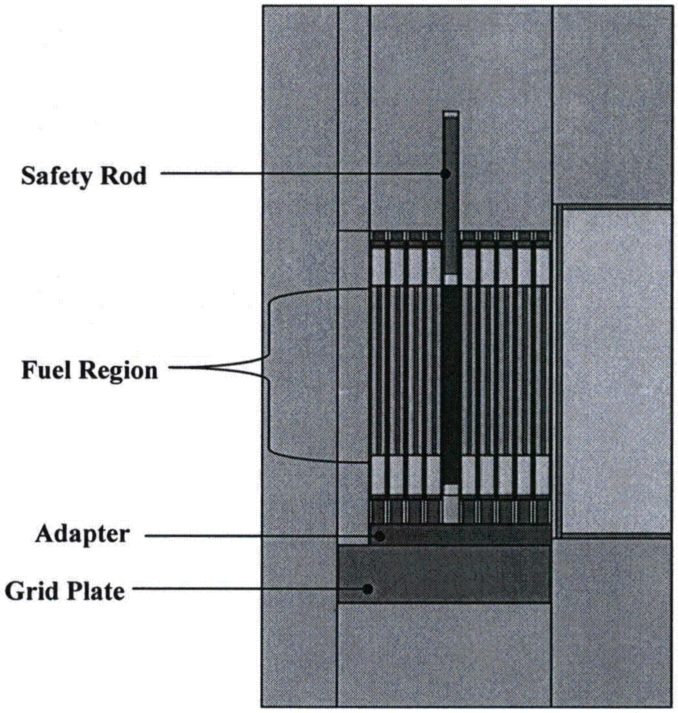


Figure 4-16: NSC Axial Model for Monte Carlo (MCNP-5 XZ Plot) Transport Calculation –[redacted]

4.5.1.3 NSC Reactor Model, Approach-to-Critical

A detailed MCNP-5 model of the NSC reactor was also made including [redacted], 4 fuel-followed rods, 1 void followed transient rod, 1 water-followed regulating rod, 11 graphite blocks around the core, and 4 detector assemblies. This number of fuel rods was chosen in order to be close to the [redacted] in the critical configuration at PRNC.

For the initial critical case, the NSC core was modeled to be close to the coupler box, 0.5 inch of water gap between the core, and the coupler box. This core positioning was chosen since this is the most reactive arrangement. Figure 4-15 and Figure 4-16 are the XY and XZ plots of the MCNP-5 model of the NSC cold critical case.

4.5.1.4 PRNC Full Core

The full core for the PRNC reactor ha[redacted] and was fully water reflected. A detailed MCNP-5 model included [redacted], 5 water-followed control rods, 3 shim safety rods, 1 transient rod, and 1 regulating rod), 1 rabbit system, 2 inches thick lower cluster adapter, and the 5 inches thick aluminum grid plate below the fuel rods.

Figure 4-17 and Figure 4-18 are the XY and XZ plots of the MCNP-5 model of the full un-rodded core PRNC reactor with infinite water reflector. Figure 4-19 and Figure 4-20 are the XY and XZ plots of the MCNP-5 model of the full core PRNC reactor with all control rods inserted. The reactor has full water reflector.

4.5.1.5 NSC Full Core

The full core for the NSC reactor has [redacted] and will be adjacent to the coupler box (graphite). A detailed MCNP-5 model includes [redacted], 4 fuel-followed control rods, 1 void followed transient rod, 1 water-followed regulating rod, 11 graphite blocks around the core, a coupler box (graphite), 2 inches thick lower cluster adapter, and the 5 inches thick aluminum grid plate below the fuel rods.

Figure 4-21 and Figure 4-22 are the XY and XZ plots of the MCNP-5 model of the full un-rodded core NSC reactor. Figure 4-23 and Figure 4-24 are the XY and XZ plots of the MCNP-5 model of the full core NSC reactor with all fuel-followed rods inserted.

Note: An un-rodded core means the control rods are 100% withdrawn.

[redacted]

Figure 4-17: PRNC Radial Model Monte Carlo (MCNP-5 XY Plot) Transport Calculations – [redacted], Unrodded

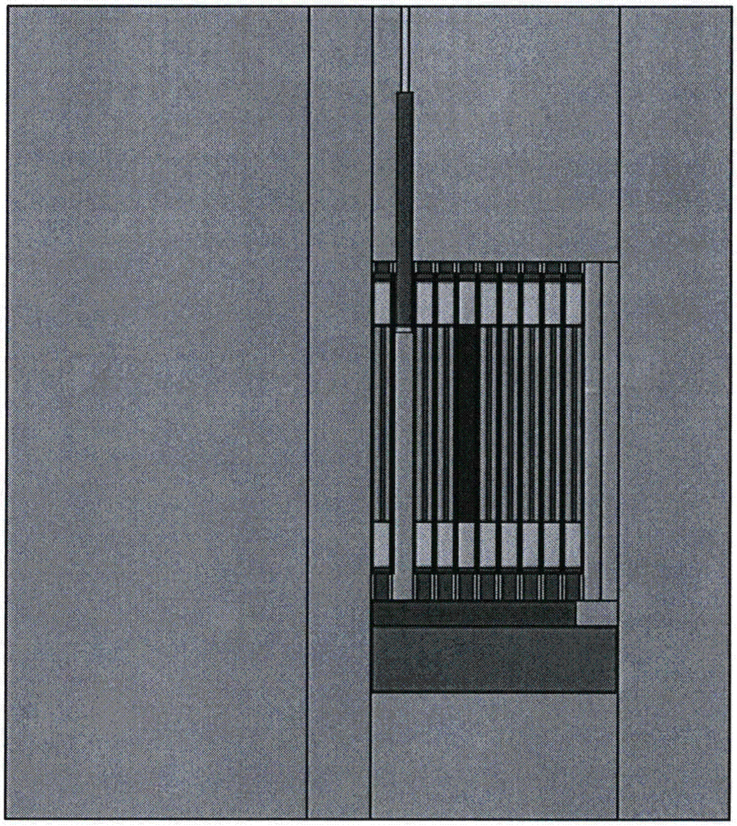


Figure 4-18: PRNC Axial Model Monte Carlo (MCNP-5 XZ Plot) Transport Calculations – [redacted], Unrodded

[redacted]

Figure 4-19: PRNC Radial Model Monte Carlo (MCNP-5 XY Plot) Transport Calculations – [redacted], All Control Rods Inserted

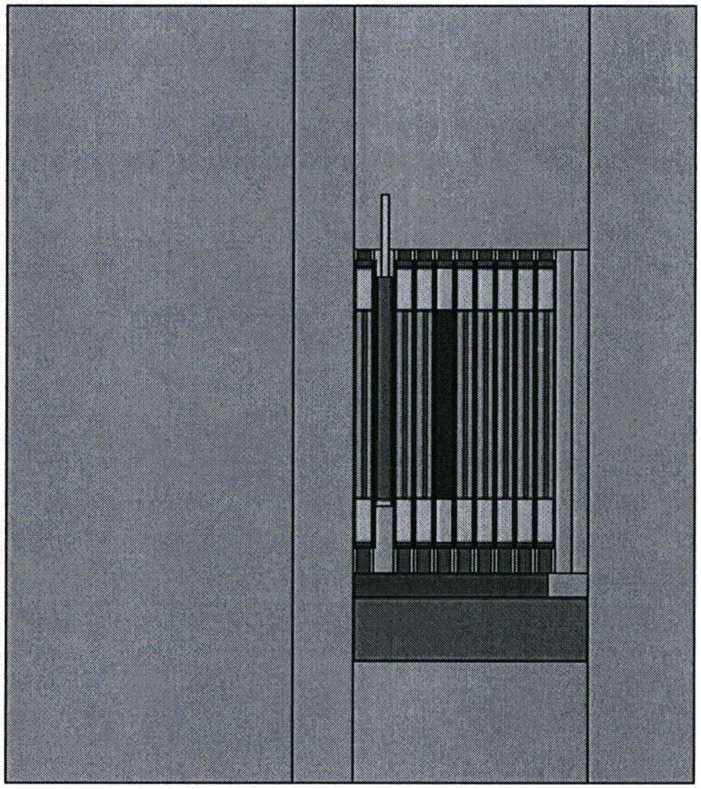


Figure 4-20: PRNC Axial Model Monte Carlo (MCNP-5 XZ Plot) Transport Calculations – [redacted], All Control Rods Inserted

[redacted]

Figure 4-21: NSC Radial Model Monte Carlo (MCNP-5 XY Plot) Transport Calculations – [redacted], Unrodded

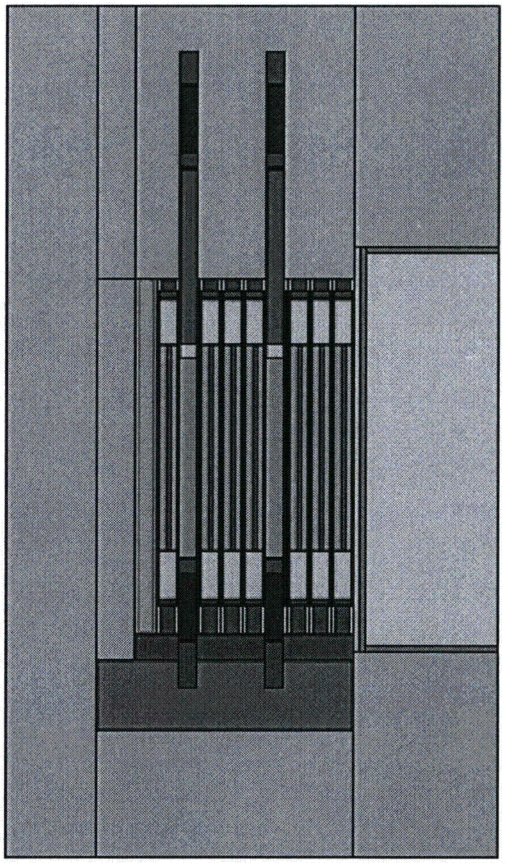


Figure 4-22: NSC Axial Model Monte Carlo (MCNP-5 XZ Plot) Transport Calculations – [redacted], Unrodded

[redacted]

Figure 4-23: NSC Radial Model Monte Carlo (MCNP-5 XY Plot) Transport Calculations – [redacted], All Control Rods Inserted

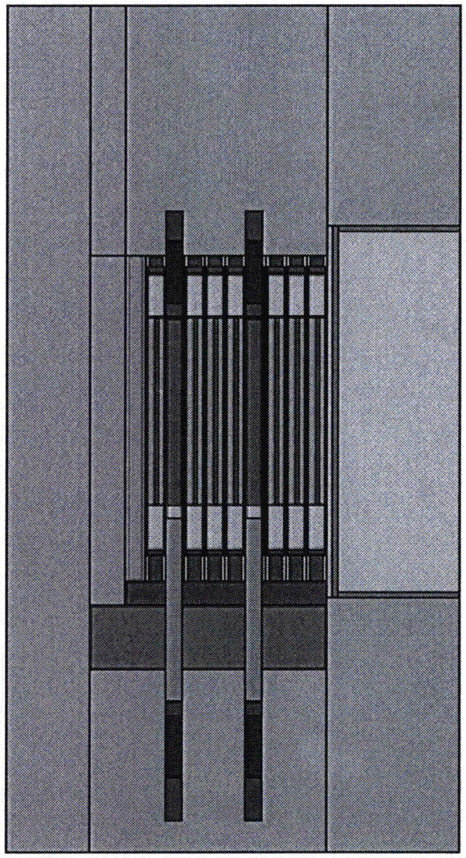


Figure 4-24: NSC Radial Model Monte Carlo (MCNP-5 XZ Plot) Transport Calculations – [redacted], All Control Rods Inserted

4.5.2 Approach-to-Critical

The evaluation of the Puerto Rico Nuclear Center TRIGA HEU FLIP core provided an opportunity to benchmark the computational techniques to be used for evaluating the TRIGA LEU 30/20 fuel in the Texas A&M University TRIGA 4-rod cluster core. It also provided the information required for the performance comparison of the fresh HEU FLIP and fresh LEU 30/20 fuel for the HEU-to-LEU conversion.

4.5.2.1 Approach-to-Critical – PRNC

The calculated k_{eff} with 1 sigma uncertainty value for the PRNC core with [redacted] fuel elements is:

$$k_{eff} = 1.00097 \pm 0.00018$$

This computed value is consistent with just critical ($k_{eff} = 1.0$) with [redacted] with infinite water reflector. The present value gives a slight excess reactivity of about \$0.14.

The initial approach to critical conducted by GA in 1972 using the 1/M approach indicated that the PRNC reactor went critical with [redacted] in the same infinite water reflector configuration; the [redacted] in this configuration gave a measured excess reactivity of $0.78. Thus, the MCNP-5 code predicts that three more fuel elements are required for initial criticality, an error of about 5.0%.

The small difference (5.0%) between calculated and measured just critical fuel loading may possibly be due to a slightly different actual U-235 and Erbium content in the PRNC core. Impurities in actual materials could also contribute to the reactivity difference. The values used as input for the MCNP-5 code were the best values available from the fuel manufacturing records.

4.5.2.2 Approach-to-Critical – NSC

The loading of fuel elements to obtain criticality was accomplished using the standard inverse multiplication curve (1/M) approach. This approach is based on the fact that subcritical multiplication is given as:

$$M = 1/(1-k)$$

from which one obtains:

$$1/M = 1-k$$

where k ranges from 0 (no fuel) to 1 (at criticality). The experimental values for 1/M subcritical multiplication are given by the count rate with no fuel C_0 divided by C_n for loading step n. However, for the present 1/M application for approach to critical, the value C0 can start at any convenient loading point.

Using two fission chambers, the approach to criticality was monitored using the inverse multiplication (1/M) method. For practical purposes, the core configuration deviated from the modeled core configuration during the approach to criticality, but every effort was made to maintain a rectangular configuration similar to the model. The shim safety control rods and the transient rod were inserted in the core at the earliest practicable time; however, the regulating rod remained out of the core until criticality was achieved.

The value for C_0 for the 1/M Method was determined with [redacted]. As further count rates were determined, a half-step approach to criticality was performed where practicable, however, the bundle-style NSC core did not allow for a single-element approach to criticality. Criticality was achieved with [redacted] installed in the core. This was within the predicted range of [redacted]. The core configuration at time of initial criticality is shown in Figure 4-25. Core loading at time of initial criticality was [redacted] of Uranium-235. The core excess reactivity was determined to be $0.147 based on the asymptotic period of the reactor with all rods fully withdrawn. This is equivalent to:

[redacted]

[redacted]

4.5.3 Worth of Control Rods

4.5.3.1 Control Rod Worth, Full Core Loading – PRNC

The full core loading in the PRNC water reflected core contained [redacted] and 1 in-core stainless steel dummy. The MCNP-5 calculation with all control rods inserted gave a k_{eff} value with one (1) sigma uncertainty:

$$k_{eff} = 0.96079 \pm 0.00018$$

This is equivalent to a reactivity shutdown of -\$5.75. The 5 control rods have a calculated worth of \$12.00. The total worth of the experimentally determined individual 5 control rods was \$12.07.

4.5.3.2 Control Rod Worth, Full Core Loading – NSC

The full core loading in the NSC reactor contains [redacted] control rods. The reactor core is positioned against the coupler box (graphite). The MCNP-5 calculation with all control rods inserted gives a k_{eff} value with one (1) sigma uncertainty:

$$k_{eff} = 0.94314 \pm 0.00017$$

This is equivalent to reactivity shutdown of -\$8.61. The 6 control rods have a combined calculated reactivity worth of \$16.34.

4.5.3.3 Control Rod Worth, Full Core Loading, Measured—NSC

After criticality was achieved, the reactor was fully loaded to the desired final configuration. Rod worth determination was performed on control rods using the Positive Period-Differential Worth Method.

For this method, a TAMU developed computer program (CRCAL), was used to determine the reactor period, amount of reactivity inserted, and the reactivity per percent of control rod withdrawal. CRCAL was used in the previous NSC HEU core and uses an input signal from the linear drawer, which is proportional to reactor power. The program measures the rate of change of this signal, computes the log reactor power, and displays a printable plot of reactor power versus time on the PC monitor. A linear fit of this curve is taken to generate a reactor period, which is used to calculate the amount of reactivity inserted and the reactivity per percent of control rod withdrawal. This program can also determine and print out the control rod position and corresponding differential worth.

To establish reactor period, the reactor is brought critical with the rod being calibrated at the desired initial rod height and all other rods positioned as necessary to achieve criticality. The rod being calibrated is then withdrawn to establish a steady positive period. The CRCAL program collects power data until reactor power approaches the point of adding heat. This process is repeated until the entire length of the rod is calibrated.

The accuracy of the procedure is dependent on the number of iterations performed per rod. Each iteration was performed in a manner to establish a reactor period of approximately 15 – 20 seconds where possible. Significantly larger reactor periods sometimes occurred due to the low worth of the associated rod. Table 4-5 contains the results of the control rod calibrations including the number of iterations performed. The determined value of $17.197 was $0.857 greater (5.2%) than calculated values.

Table 4-5: NSC Measured Control Rod Worths

Control Rod	Number of Iterations	Control Rod Worth ($)
Shim Safety #1	13	3.174
Shim Safety #2	9	2.035
Shim Safety #3	14	2.901
Shim Safety #4	19	4.609
Regulating Rod	7	1.021
Transient Rod	15	3.457
Total Control Rod Worth		**17.197**

4.5.4 Excess Reactivity

For comparison with the proposed LEU 30/20 core, the critical core configuration was evaluated for the clean HEU FLIP fuel in the PRNC reactor. There were [redacted], including 4 fuel follower control rods. The proposed core loading for the NSC LEU core was [redacted] control rods.

4.5.4.1 Excess Reactivity, Full Core Loading – PRNC

The full core loading in the PRNC water reflected core contained [redacted] and 1 in-core stainless steel dummy. The MCNP-5 calculation gave an un-rodded k_{eff} value with one (1) sigma uncertainty:

$$k_{eff} = 1.04649 \pm 0.00017$$

This is equivalent to a reactivity of $6.26. The experimentally determined measured value was $7.12.

4.5.4.2 Excess Reactivity, Full Core Loading, Calculated – NSC

The full core loading in the NSC core contains [redacted] including [redacted]. With the core assembly moved a distance of about 250 mm from the coupler box (graphite), the MCNP-5 calculation gives an un-rodded k_{eff} value with 1 sigma uncertainty of the embedded k_{eff} value with one (1) sigma uncertainty is:

$$k_{eff} = 1.04553 \pm 0.00017$$

This corresponds to a core reactivity of $6.22.

4.5.4.3 Excess Reactivity, Full Core Loading, Measured – NSC

The results of Section 4.5.3.3 were used in determining excess reactivity. The results were measured with the reactor in a cold, xenon-free condition (reactor shutdown > 72 hours). This procedure was performed with the reactor > 250 mm away from the thermal column.
To determine excess reactivity, the reactor was made critical at 300 W following an extended shutdown. Three hundred watts is high enough to ensure that source neutrons are not making a significant contribution to neutron population, but low enough to ensure that the reactor is below the point of adding heat and therefore adding negative reactivity.
With the reactor critical, rod heights were recorded. Using the rod worth data from Section 4.5.3.3, the rod worth at the recorded rod height was subtracted from total rod worth to determine excess reactivity. With the reactor away from the thermal column, excess reactivity was determined to be $7.483 (Table 4-6). This value was greater than the calculated value by $1.263. Approximately $0.86 of this value is attributed to the 5.2% deviation between calculated and measured control rod worth.

Table 4-6: NSC Measured Excess Reactivity Away from Thermal Column

Control Rod	300 W Position (%)	Control Rod Excess ($)
Shim Safety #1	49.0	1.700
Shim Safety #2	49.0	1.159
Shim Safety #3	49.0	1.631
Shim Safety #4	49.0	2.453
Regulating Rod	45.6	0.540
Transient Rod	100.0	0.000
Total Core Excess		**7.483**

4.5.5 Shutdown Margin

4.5.5.1 Shutdown Margin, PRNC Core

As stated in the applicable technical specifications, the reactor shall not be operated unless the shutdown margin provided by the control rods is greater than $0.25 with:

1. The highest worth non-secured experiment in its most reactive state,
2. The highest worth control rod fully withdrawn, and
3. The reactor in the cold condition without xenon.

The full core loading in the PRNC water reflected core contained [redacted] and 1 in-core stainless steel dummy. The MCNP-5 calculation with all control rods inserted gave a k_{eff} value with one (1) sigma uncertainty:

$$k_{eff} = 0.96079 \pm 0.00018$$

This is equivalent to a reactivity shutdown of -$5.75.

The MCNP-5 code with appropriate input files was also used to calculate the shutdown reactivity with the most reactive rod fully withdrawn and all other rods in the core. The k_{eff} with one (1) sigma uncertainty for the shutdown core was:

$$k_{eff} = 0.98599 \pm 0.00018$$

This corresponds to a reactivity of -$2.00. This is considerably more negative than -$0.25. It agrees well with the -$2.12 determined from the calibrated control rod reactivity values measured during startup.

4.5.5.2 Shutdown Margin, Calculated, NSC Core

As stated in the applicable NSCR Technical Specifications, the reactor shall not be operated unless the shutdown margin provided by the control rods is greater than $0.25 with:

1. The highest worth non-secured experiment in its most reactive state,
2. The highest worth control rod fully withdrawn, and
3. The reactor in the cold condition without xenon.

Reactivity shutdown was calculated with the reactor core positioned against the thermal column. The MCNP-5 calculation with all control rods inserted gives a k_{eff} value with one (1) sigma uncertainty:

$$k_{eff} = 0.94314 \pm 0.00017$$

This is equivalent to reactivity shutdown of -$8.61. No reactivity was calculated for the reactor away from the thermal column.

To determine shutdown margin as defined by technical specifications, the MCNP-5 code was run for the case with [redacted] and the most reactive rod plus the non-scrammable regulating rod up, with three control rods with fuel followers and the transient rod inserted in the core. The MCNP-5 results gave a k_{eff} with one (1) sigma uncertainty of:

$$k_{eff} = 0.99198 \pm 0.00017$$

This corresponds to a shutdown margin of -$1.15, which is clearly more negative than -$0.25.

4.5.5.3 Shutdown Margin, Measured, NSC Core

Shutdown reactivity was determined by subtracting total rod worth from excess reactivity. The measured value for reactivity shutdown was $9.707 away from the thermal column.
To measure shutdown margin as defined by technical specifications, the worth of the most reactive control rod, the worth of the regulating rod and $1.00 (the worth of the highest worth experiment in its most reactive state. The measured value for shutdown margin was -$3.077 away from the thermal column. Approximately $0.86 of this value is attributed to the 5.2% deviation between calculated and measured control rod worth. Calculations were performed with the reactor against the thermal column while the measured value was attained with the reactor away from the thermal column. This further caused $1.16 of this value to be attributed to the worth of operating against the thermal column. The net difference between calculated and measured shutdown margin was $1.057.

4.5.6 Additional Core Physics Parameters for HEU and LEU Cores

4.5.6.1 Effective Delayed Neutron Fraction, β_{eff} for PRNC Core

The effective delayed neutron fraction, β_{eff}, was derived from diffusion theory reactor calculations where the reactivity is first computed with the prompt fission spectrum alone and then recalculated with the fission spectrum of both prompt and delayed neutrons. Seventeen groups in the fast energy range and the standard four thermal groups were used for these 3-D calculations in order to represent the two fission spectra in greater detail than is possible with only three fast groups. The results of previous GA SAR calculations indicate that detail in the group structure is of much greater importance than geometric detail. This is not unexpected since the calculation of β_{eff} is directly related to neutron energy effects.

The prompt fission spectrum is obtained from the GGC-5 spectrum calculation[4]. The delayed fission spectrum is obtained by integrating over the broad energy groups.

The prompt and total fission spectra for each of the broad energy groups in the calculation are given in Table 4-7.

The computed values of K_t and K_p are used in the following expression to obtain β_{eff},[5,6]:

$$\beta_{eff} = [\ K_t(1+\beta_o)/K_p\ ]-1$$

where:

- K_t = core reactivity using prompt and delayed fission spectrum,
- K_p = core reactivity using prompt fission spectrum,
- β_o = intrinsic delayed neutron fraction for U-235 (0.0065)

The 3-D model used 21 total groups and very tight convergence criteria (1.0x10-8 on k_{eff}, 1.0 x 10^{-6} point flux). The cases were run cold (23°C) with fresh fuel. The result for PRNC FLIP fuel was:

$$\beta_{eff} = 0.0071$$

Table 4-7: Fission Spectra Used for Calculation of β_{eff}

Group	Energy Interval eV	Prompt χ_p	Delayed χ_t
1	$15.0 \times 10^6 - 10.0 \times 10^6$	0.00104	0.00103
2	$10.0 \times 10^6 - 8.19 \times 10^6$	0.00345	0.00343
3	$8.19 \times 10^6 - 6.70 \times 10^6$	0.00979	0.00973
4	$6.70 \times 10^6 - 5.49 \times 10^6$	0.02149	0.02135
5	$5.49 \times 10^6 - 4.49 \times 10^6$	0.03825	0.03800
6	$4.49 \times 10^6 - 3.68 \times 10^6$	0.05745	0.05708
7	$3.68 \times 10^6 - 3.01 \times 10^6$	0.07525	0.07476
8	$3.01 \times 10^6 - 2.02 \times 10^6$	0.18292	0.18180
9	$2.02 \times 10^6 - 1.50 \times 10^6$	0.14006	0.13943
10	$1.50 \times 10^6 - 1.00 \times 10^6$	0.16006	0.15972
11	$1.00 \times 10^6 - 6.08 \times 10^5$	0.14012	0.14046
12	$6.08 \times 10^5 - 3.02 \times 10^5$	0.10357	0.10463
13	$3.02 \times 10^5 - 1.11 \times 10^5$	0.05119	0.05235
14	$1.11 \times 10^5 - 4.09 \times 10^4$	0.01248	0.01305
15	$4.09 \times 10^4 - 9.12 \times 10^3$	0.00288	0.00317
16	$9.12 \times 10^3 - 4.54 \times 10^2$	0	0
17	$4.54 \times 10^2 - 1.125$	0	0
TOTAL		1.00000	1.00000

4.5.6.2 Effective Delayed Neutron Fraction, β_{eff}, for NSC Core

The effective delayed neutron fraction, β_{eff}, for the NSC core was calculated exactly for the same way that the PRNC core above but with the NSC input parameters.

The 3-D model used 21 total groups and very tight convergence (1.0 x 10-8 on k_{eff}, 1.0 x 10^{-6} point flux). The cases were run cold (23°C) with fresh 30/20 fuel.

The result for TRIGA LEU 30/20 fuel was:

$$\beta_{eff} = 0.0070$$

4.5.6.3 ***Prompt Neutron Life (ℓ) for PRNC Core***

The prompt neutron lifetime, ℓ, was computed by the 1/v absorber method where a very small amount of boron is distributed homogeneously throughout the system and the resulting change in reactivity is related to the neutron lifetime. This calculation was done using the 3-D diffusion theory model for the core to allow very tight convergence of the problems. The boron cross sections used in the core were generated over a homogenized core spectrum. Boron cross sections used in all other zones were generated over a water spectrum.

The neutron lifetime, ℓ, is defined as follows:

$$= \Delta k_{eff}/\omega$$

where Δk_{eff} is the change in reactivity due to the addition of boron and ω is related to the boron atom density and:

$$NB = \omega/\delta_o\ v_o = 6.0205 \times 10^{-7}$$

where:

1. N_B = boron density (atoms/b-cm)
2. ω = integer = 100 (the calculation is insensitive to changes in ω between 1 and 100),
3. v_o = 2200 m/sec,
4. δ_o = 755 barns = $\delta_a{}^B$ at 2200 m/sec

As described in the β_{eff} section above, the 3-D model used very tight convergence criteria (1.0 x 10^{-8} of k_{eff}, 1.0 x 10^{-6} point flux). The cases were run cold (23°C) with fresh FLIP fuel. The result for prompt neutron lifetime in the un-rodded core is the following:

$$\ell = 22.5\ \mu\text{sec}$$

4.5.6.4 ***Prompt Neutron Life (ℓ) for NSC Core, Calculated***

Using the same 1/v absorber method described above for the PRNC core, the prompt neutron life, ℓ, has been evaluated for the TRIGA LEU 30/20 fuel in the NSC core.

The result for the prompt neutron life (ℓ), at beginning-of-life (BOL), in NSC core is the following:

$$\ell = 26.3\ \mu\text{sec}$$

4.5.6.5 *Prompt Neutron Life (ℓ) for NSC Core, Measured*

Prompt neutron (ℓ) lifetime can be calculated from reactor pulse data using the following equation:

$$\ell = \frac{FWHM \times (\Delta k_p) \times \beta_{eff}}{4\cosh^{-1}\sqrt{2}} = \frac{FWHM \times (\Delta k_p) \times \beta_{eff}}{3.524}$$

A series of pulses were performed to obtain the FWHM (full width half maximum) values for a range of reactivity insertions. For each pulse, prompt neutron lifetime was calculated using the above equation. These values were then averaged to derive the NSC prompt neutron lifetime. Table 4-8 contains the results of the reactor pulse measurements and the calculated prompt neutron lifetime. The measured value for ℓ was consistent with the calculated value.

Table 4-8: Prompt Neutron Lifetime from Pulses at Various Reactivity's

Δk_p	FWHM	ℓ
($)	**msec**	**msec**
0.22	63.6	0.027793
0.24	64.2	0.030606
0.42	30.2	0.025195
0.52	24.1	0.024893
0.33	40.5	0.026548
0.47	25.9	0.02418
0.56	21.6	0.024027
Average		0.026178

4.5.6.6 *Prompt Negative Temperature Coefficient of Reactivity, α, for PRNC Core*

The definition of the prompt negative temperature coefficient of reactivity, α, is given as

$$\alpha = \frac{d\rho}{dT}$$

where:

1. 1. ρ = reactivity = (k-1)/k
2. T = reactor temperature (°C)

To evaluate ($\Delta\ \rho$) from reactivity as a function of reactor core temperature, the finite difference equation can be written as follows:

$$\Delta\rho_{1,2} = \frac{k_2 - 1}{k_2} - \frac{k_1 - 1}{k_1}$$

$$= \frac{k_2 - k_1}{k_1 k_2}$$

Thus,

$$\alpha_{1,2} \cong \frac{k_2 - k_1}{k_1 k_2} \text{ x } \frac{1}{\Delta T_{1,2}}$$

The data in Table 4-9 were produced by DIF3D for the listed core temperatures.

Figure 4-26 is a histogram plot of the computed values of α as a function of reactor temperature. Also shown in the same figure is the representations of α given in the 1969 SAR. It can be seen that the presently computed values of α are somewhat smaller for each core temperature.

Table 4-9: Reactivity Change with Temperature, PRNC

Average Core Temperature (°C)	k_{eff}	Δk_{eff}	$\frac{k_2 - k_1}{k_1 k_2}$	$\alpha_{1,2}$
23	1.04171			
		0.00909	0.008450	4.77 x 10-5
200	1.03262			
		0.00898	0.008495	10.6 x 10-5
280	1.02364			
		0.0942	0.009073	7.56 x 10-5
400	1.01422			
		0.04168	0.042256	14.1 x 10-5
700	0.97254			
		0.04822	0.053641	17.9 x 10-5
1000	0.92432			

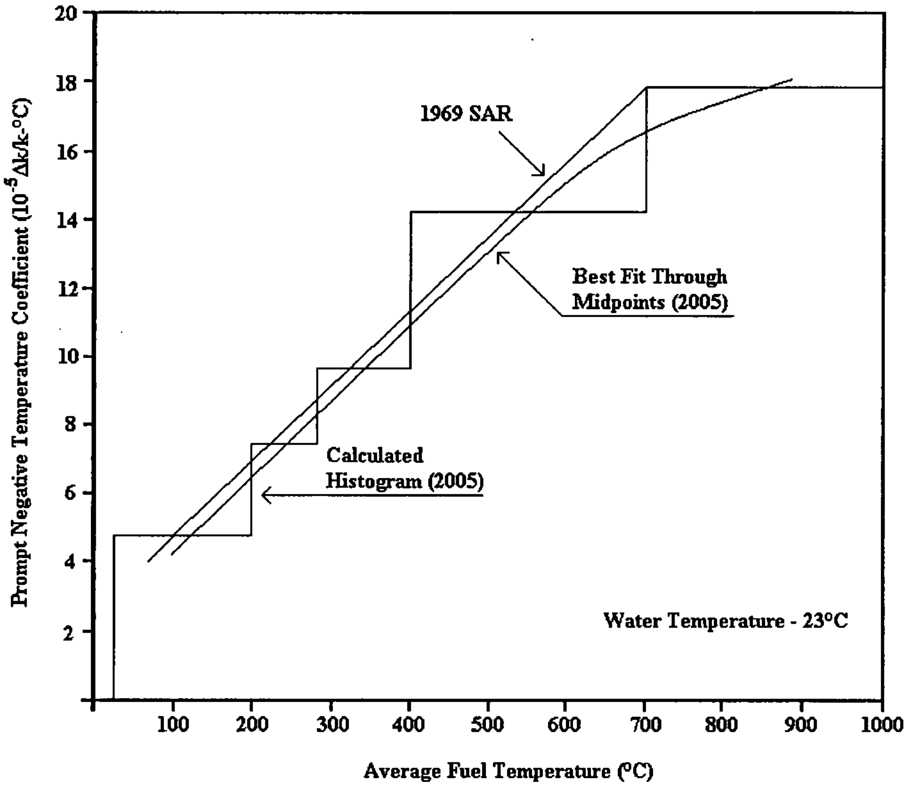


Figure 4-26: Calculated Prompt Negative Temperature Coefficient vs. Fuel Temperature at Beginning-of-Life for FLIP Fuel. The Straight line is included from the 1969 SAR for PRNC for comparison.

4.5.6.7 Prompt Negative Temperature Coefficient of Reactivity, α, for NSC Core

Following the procedure set forth in the Section 4.5.6.6 for the PRNC core, the computer results from DIF3D for the NSC reactor reactivity change with temperature are listed in Table 4-10 for Beginning-of-Life (BOL) and in Table 4-11 for End-of-Life (EOL).

Figure 4-27 is a histogram plot of the computed values for α in Table 4-10 and Table 4-11 as a function of core temperature for both BOL and EOL.

In Figure 4-27, it can be seen that the prompt negative temperature coefficient, α, for LEU 30/20 fuel has only a modest decrease in values at 2000 MWD burnup (EOL) (e.g., 13.1 x 10^{-5} to 9.9 x 10^{-5} Δk/°C at 700-1000°C). As illustrated in the NSC SAR, the corresponding decrease for FLIP fuel is much larger, as an example, 17 x 10^{-5} to 4 x 10^{-5} Δk/°C at 800°C for 2000 MW burnup.

The relatively small change in α for LEU 30/20 fuel is expected due to the 80 wt-% U-238 in LEU 30/20 fuel compared to 30 wt-% U-238 in FLIP fuel. It will be remembered that it is the 80 wt-% U-238 in standard TRIGA fuel that is responsible for the nearly temperature independent α value of about 10 x 10^{-5} Δk/°C.

Table 4-10: Reactivity Change with Temperature, NSC, BOL

Average Core Temperature (°C)	k_{eff}	Δk_{eff}	$\frac{k_2 - k_1}{k_1 k_2}$	$\alpha_{1,2}$
23	1.04496			
		0.01015	0.009387	5.30×10^{-5}
200	1.03481			
		0.00585	0.005494	6.87×10^{-5}
280	1.02896			
		0.01039	0.009913	8.26×10^{-5}
400	1.01857			
		0.03271	0.032574	10.9×10^{-5}
700	0.98586			
		0.03673	0.039254	13.1×10^{-5}
1000	0.94913			

Table 4-11: Reactivity Change with Temperature, NSC, EOL

Average Core Temperature (°C)	k_{eff}	Δk_{eff}	$\frac{k_a - k_b}{k_a k_b}$	$\alpha_{a,b}$
23	1.01876			
		0.00901	0.008579	4.95×10^{-5}
200	1.00975			
		0.00496	0.004886	6.11×10^{-5}
280	1.00479			
		0.08440	0.008431	7.03×10^{-5}
400	0.99635			
		0.02497	0.025800	8.60×10^{-5}
700	0.97138			
		0.02746	0.029949	9.98×10^{-5}
1000	0.94392			

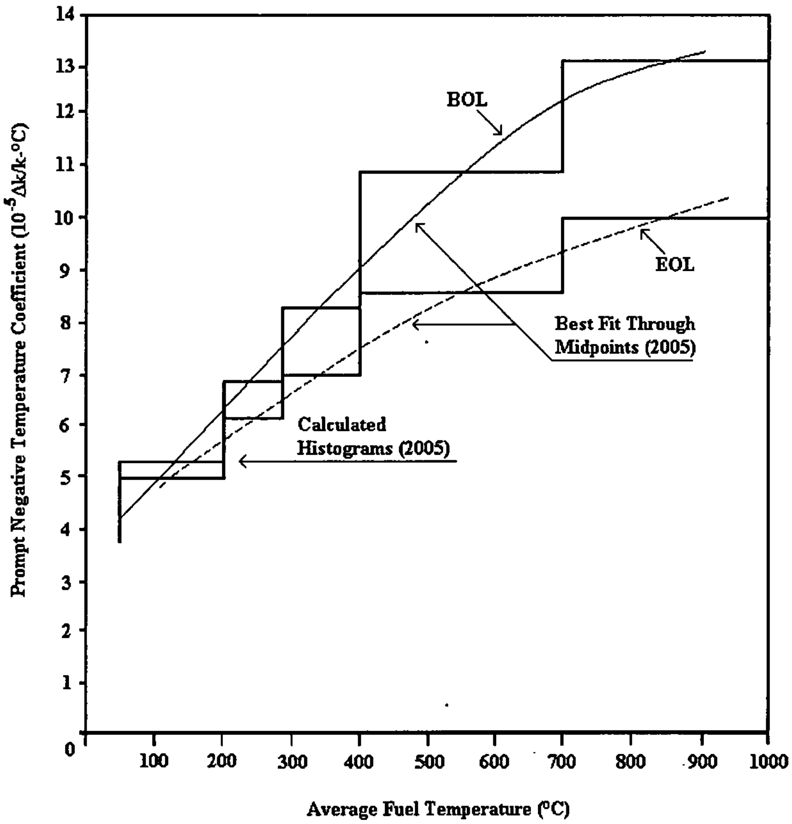


Figure 4-27: Prompt Negative Temperature Coefficient for TRIGA LEU 30/20 Fuel, Beginning-of-Life (BOL) and End-of-Life (EOL), NSC Core

4.5.6.8 Void Coefficient – PRNC Core

The void coefficient of reactivity is defined for a TRIGA reactor as the negative reactivity per 1.0% void in the reactor core water. For the PRNC reactor with FLIP fuel, the calculated void coefficient is -0.107% Δk/k per 1% water void. This void coefficient is not normally considered a safety concern for TRIGA reactors. The reason is the relatively small size of this coefficient and the fact that all TRIGA reactors are significantly under-moderated. Therefore, if a portion of the core water is replaced with a low density material (i.e., steam, gas including air, etc.), a negative reactivity will occur. An example would be a dry, experimental tube introduced into the core [e.g., (4B1) see Figure 4-2] with a volume of 242 cc in the [redacted]. The calculated loss in core reactivity would be about $0.15.

A safety effect of rapid reactivity insertion to be considered is the effect of accidental flooding of an in-core dry experimental tube such as postulated above. In this case, the rapid reactivity insertion would be only about $0.15. The insertion of $0.15 reactivity is far less than $1.00 (prompt critical).
The conclusion is that the very small void coefficient of reactivity is not a source of safety concern.

4.5.6.9 Void Coefficient – NSC

The void coefficient for the NSC reactor is -0.130% Δk/k per 1% water void. If a dry experimental region (of 281cc in the [redacted]) is inserted near core center (replacing a fuel rod) and is accidentally flooded with water, the prompt gain in reactivity is about $0.26. This reactivity addition is far less than $1.00 required for prompt critical.

The conclusion is that the very small void coefficient is not a source of safety concern.

4.5.7 Core Burn-Up – NSC LEU Fuel

Burnup analyses were performed using the DIF3D multi-dimensional diffusion theory code along with the BURP[7] depletion code. All burnup analyses used the cross-sections generated for BOL concentrations at the approximate average fuel temperature of 200°C, the closest nuclear data available. The burnup curve was then adjusted to 237°C using α.

Figure 4-28 shows the results from design calculations for core k_{eff} as a function of burnup of the initial core. The time steps used for the burnup calculation started with 3 days (to evaluate equilibrium xenon poisoning) and then 50 day intervals from time 0 at full power (1.0 MW). The LEU burnup curve in Figure 4-28 gives a lifetime of the initial core (with no fuel shuffling) of about 2000 MWD at 1.0 MW, full equilibrium xenon poisoning, and about $0.60 reactivity left for burnup or experiments. For comparison, a FLIP burnup curve for [redacted] core was calculated and is also shown in Figure 4-28. The FLIP burnup curve gives a lifetime of the initial core (with no fuel shuffling) of about 2350 MWD at 1.0 MW, full equilibrium xenon poisoning, and about $0.60 reactivity remaining.

The data on burnup at the 3-day interval indicates a xenon equilibrium poison value of about $1.31. The magnitude of this xenon poison may seem small for a 1.0 MW reactor ($\sim 10^{13}$ n/cm$^2 \bullet$ s). However, xenon is produced interior to the TRIGA fuel elements where the thermal neutron flux is severely depressed due to 30 wt% uranium and erbium burnable poison. At the end of core life, an independent calculation gives an equilibrium xenon poison value of $1.51. This value is larger at EOL because the thermal flux in the fuel is larger due to burnup of a portion of the U-235 and erbium.

With NSC operation for 35-70 hours/week, full equilibrium xenon will not be built into the core (a process taking more than 60 continuous hours). Consequently, this increased core excess reactivity will permit operation to about 2600 MWD (an additional 600 MWD). The burnup curve in Figure 4-28 assumes no reactivity requirement for experiments but full equilibrium xenon poison. In view of the fact that far less xenon poison will be present, one can assume that a portion of the $1.31 reactivity for xenon poison could be used as needed to cover reactivity loss due to experiments with the same 2000 MWD core life.

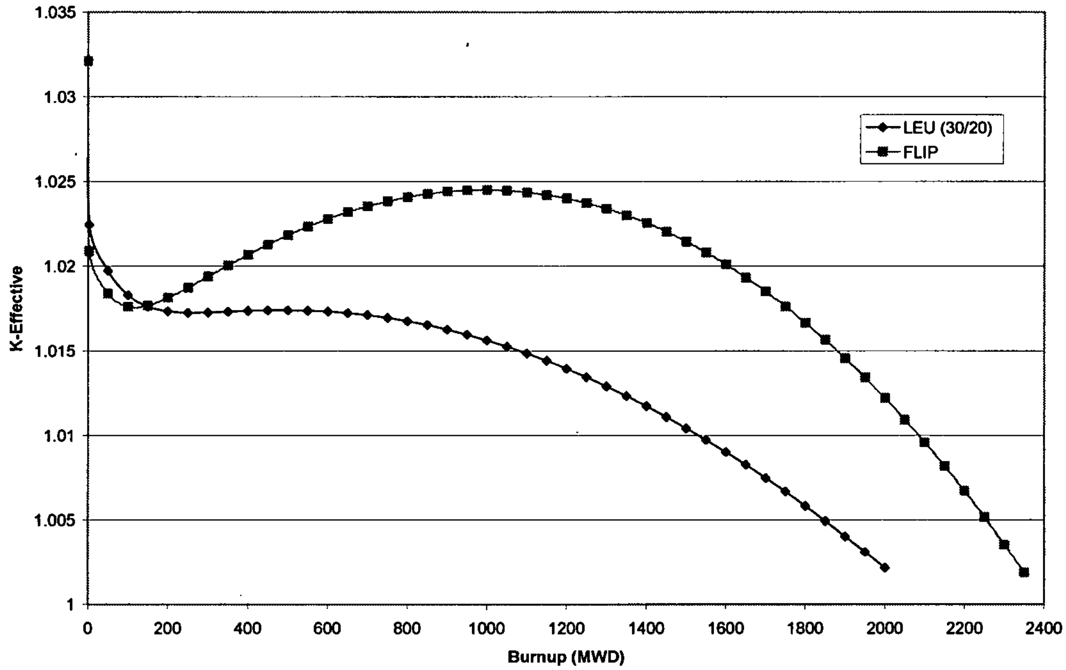


Figure 4-28: k-effective vs. Burn-up for 1.0 MW TRIGA FLIP and LEU 30/20 Cores

The above comments lend credence to the assertion that the TRIGA LEU 30/20 core is typically a "lifetime core". Since NSC operates the core between 35 and 70 MW hrs per week, this corresponds to 1.46 to 2.9 MWD/week, or about 73 to 146 MWD/year. With only partial xenon poisoning expected during the planned operating schedule (≤70 MWH/wk), the initial core loading is expected to provide operation at 1.0 MW as required for a time period ranging from 18 to 35 years.

With the initial LEU core expected to provide full power operation on demand for an extended period of time (18 to 35 years), a detailed reload schedule would not be useful at this time. However, the addition of either a fresh fuel rod or a full cluster of fresh fuel rods to the center of a heavily burned core raises concern for excessive power peaking. Unless a detailed calculation is performed to evaluate the magnitude of the power peaking, the addition of fresh fuel must be limited to the outer perimeter of the core. However, it may be noted that shuffling the outer, less burned fuel clusters to the core center during the 2000 MWD burnup will add significantly to the core life without concern for excessive power peaking. Typically, experience has shown that a shuffled core can add an additional 20% to the life of an un-shuffled core.

4.5.7.1 Reactor Parameters at 2000 MWD Burn-up

Using the procedure previously described for the delayed neutron fraction, the effective delayed neutron fraction has been evaluated for the LEU 30/20 core at 2000 MWD burnup. The value obtained is

$$\beta = 0.0070,$$

unchanged from the beginning of core life.

Similarly, using the procedure previously described for the prompt neutron life, the prompt neutron life, ℓ, at 2000 MWD burnup has been evaluated for the LEU 30/20 core. The value obtained is

$$\ell = 27.3 \ \mu\text{sec},$$

slightly greater than the beginning of core life value of 26.3 µsec.

The values of the prompt negative temperature coefficient for TRIGA LEU 30/20 fuel at 2000 MWD burnup have already been presented in Table 4-10 and Table 4-11 and shown in Figure 4-27 compared with the values at beginning-of-life.

4.5.8 Reactivity Loss at Reactor Power

4.5.8.1 Reactivity Loss – PRNC

The prompt negative temperature coefficient of reactivity affects all reactor operations for which the fuel temperature is elevated above ambient. Consequently, core reactivity is lost at any power above a few kilowatts when fuel temperatures begin to rise. Calculations of k_{eff} were made for reactor powers of 0.5, 1.0, and 1.4 MW. During the commissioning tests of the PRNC FLIP core, measurements were made to evaluate the reactivity loss at various reactor power levels up to 1.4 MW. These calculated and measured values of reactivity losses are tabulated in Table 4-12.

Table 4-12: Calculated and Measured Reactivity Loss, PRNC

P(MW)	$\rho = \Delta k / k\beta$	$\Delta\rho$ (\$)$_{calc}$	$\Delta\rho$ (\$)$_{meas}$
0	5.639	NA	NA
0.5	4.432	1.21	1.20
1.0	4.118	1.52	1.84
1.4	3.928	1.71	2.39

As can be seen from a comparison of measured and calculated reactivity loss, the agreement is good at 0.5 MW, but for powers of 1.0 and 1.4 MW, the measured reactivity loss is greater than the calculated value. The magnitude of the reactivity losses is related directly to the calculated temperatures. In Table 4-13, the measured temperatures ($\hat{T}_{0.3}$) agree well with the calculated $\hat{T}_{0.3}$ temperatures at the 3 power levels, 0.5, 1.0, and 1.4 MW. Thus, the lack of agreement for reactivity losses at the higher power levels, 1.0 and 1.4 MW is unexpected. This lack of agreement could be attributed to a 30-60°C error in the actual average core temperature at 1.0 and 1.4 MW.

Table 4-13: Calculated and Measured Fuel Temperatures, PRNC

P(MW)	$\hat{T}_{meas}$ (°C)	$\hat{T}_{calc}$ (°C)		
		$\hat{T}$	$\hat{T}_{0.3}$	$\overline{T}_{avg\ core}$
0.5	221	275	266	202
1.0	342	362	344	234
1.4	421	449	423	252

4.5.8.2 Reactivity Loss – NSC, Calculated

Calculations of core reactivity were made for operating power levels up to 1.3 MW. From these calculated values of k_{eff}, the loss in reactivity has been computed and is shown in Figure 4-29. The reactivity loss at 1.0 MW is \$1.69 (cold-hot reactivity swing). After burnup to 2000 MWD, the reactivity loss at 1.0 MW is \$1.51.

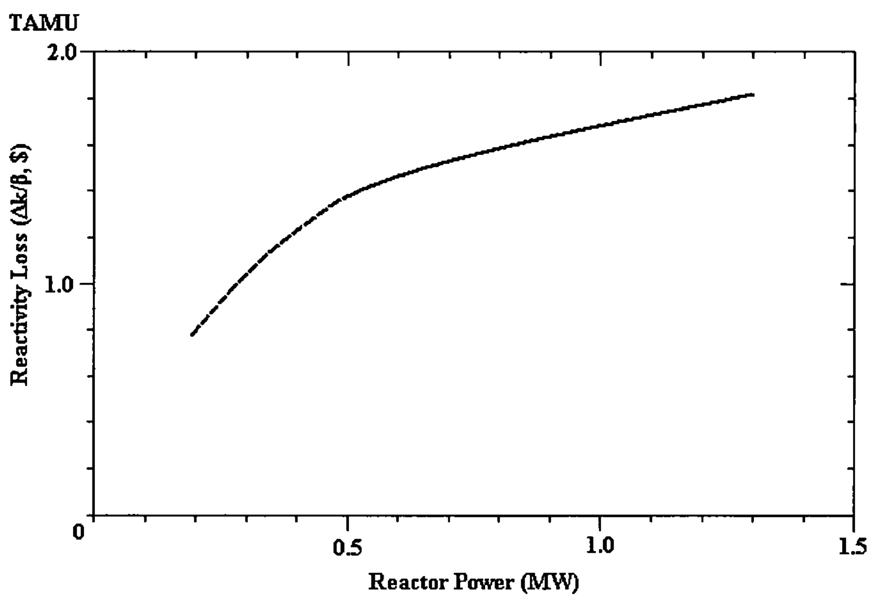


Figure 4-29: Calculated Reactivity Loss at Various Reactor Power Levels

4.5.8.3 Reactivity Loss – NSC, Measured

To determine reactivity loss at reactor power, the reactor was first made critical at 300 W (well below the point of adding heat). Control rod reactivity was calculated at their criticality positions to serve as a reference point.

The intent was then to raise reactor power incrementally to full power collecting rod height data and calculating added control rod reactivity. The change in reactivity from the initial reference point represented the reactivity loss at the associated reactor power level. The power levels to be used were 300 W, 100 kW, 250 kW, 500 kW, and 1.0 MW.

During the performance of this measurement, reactor temperature was observed to be greater than expected at 750 kW. The measurement at 1.0 MW was not performed for this procedure due to the abnormally high temperature indication. The high temperature indication was determined to be caused by a defective instrumented fuel element, which was subsequently replaced, as described in Section 1.8.

Using the data for reactor power up to 750 kW, the reactivity losses of Table 4-14 were calculated. The shape of the plot of measured reactivity loss vs. reactor power (See Figure 4-30) is consistent with the calculated plot; however the value of measured reactivity loss differs from the calculated value by $0.80. Much of this difference is attributable to the measured difference (measured vs calculated) in control rod worth as described in Section 4.5.3.3. Figure 4-29 shows the calculated reactivity loss at the various reactor powers.

Table 4-14: Reactivity Loss at Reactor Power, NSC

Linear Power (kW)	Total Reactivity Added by Control Rods ($)	Change in Reactivity ($)
0.3	$9.704	$0.000
100	$10.449	$0.745
250	$11.048	$1.344
500	$11.652	$1.948
750	$12.062	$2.358

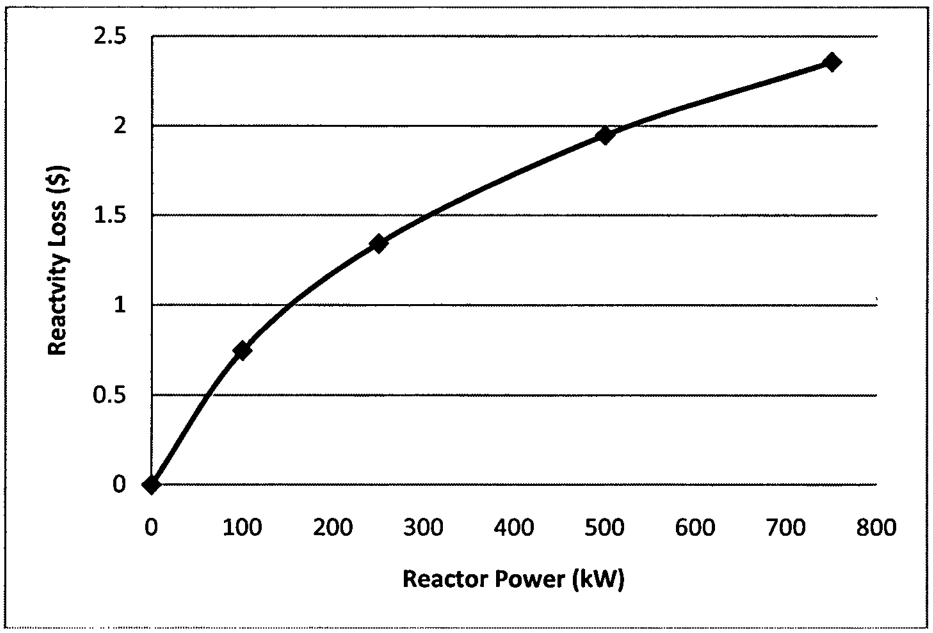


Figure 4-30: Reactivity Loss vs. Reactor Power

4.5.9 Power Peaking; Temperature Peaking – PRNC Core

Power peaking in the core is analyzed based on the following component values:

1. $\overline{P}_{rod} / \overline{P}_{core}$: rod power factor, the power generation in a fuel rod (element) relative to the core averaged rod power generation
2. $(\hat{P} / \overline{P})_{axial}$: axial peak-to-average power ratio within a fuel rod
3. $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$; rod peaking factor, the peak-to-average power in a radial plane within a fuel rod

Since maximum fuel temperature is the limiting operational parameter for the core, the peaking factor of greatest importance for steady state operation is $\overline{P}_{rod} / \overline{P}_{core}$. The maximum value of this factor for the hottest rod, the hot-rod factor, [($\overline{P}_{rod} / \overline{P}_{core}$) = hot-rod factor], determines the power generation in the hottest fuel element. When combined with the axial power distribution, the hot-rod factor is used in the thermal analysis for determination of the maximum fuel temperature. The radial power distribution within the element has only a small effect on the peak temperature but is also used in the steady state thermal analysis.

The rod peaking factor $(\hat{P}_{rod} / \overline{P}_{rod})$ radial is of importance in the transient analysis for calculating maximum fuel temperatures in the time range where heat transfer is not yet significant. It is used in the safety analysis to calculate the peak fuel temperature under adiabatic conditions, where temperature distribution is the same as power distribution.

Peaking factors calculated for the PRNC TRIGA FLIP core are shown in Table 4-15. The axial power distribution shown in Figure 4-31 is relatively independent of fuel temperature or radial position in the core.
The fuel temperatures for selected reactor power levels have been calculated for the hottest and average fuel rods. These results are presented in Table 4-13 together with the experimentally measured fuel temperatures for the same reactor power levels.

Note that the instrumented fuel element was in fact located in the hottest core location. Thus, the fuel temperature $\hat{T}_{meas}$ is the measured fuel temperature in the hottest fuel element. Since the sensing tip of the thermocouple was 0.30 inches from the axial centerline of the fuel element, the temperatures reported in Table 4-13 were calculated for the hottest radial position ($\hat{T}$) and for a position 0.30 inches from the center line ($\hat{T}_{0.3}$). Finally, the average core temperature ($\overline{T}_{avg\ core}$) was calculated.

Table 4-15: Power Peaking Factors - PRNC

Type of Peaking	Beginning of Core Life 95 Fuel Rods and 1 Stainless Steel Dummy
$(\overline{P}_{rod} / \overline{P}_{core})_{max}$	1.55
$(\hat{P} / \overline{P})_{axial}$	1.27
Average $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	1.52
Peak $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	1.967

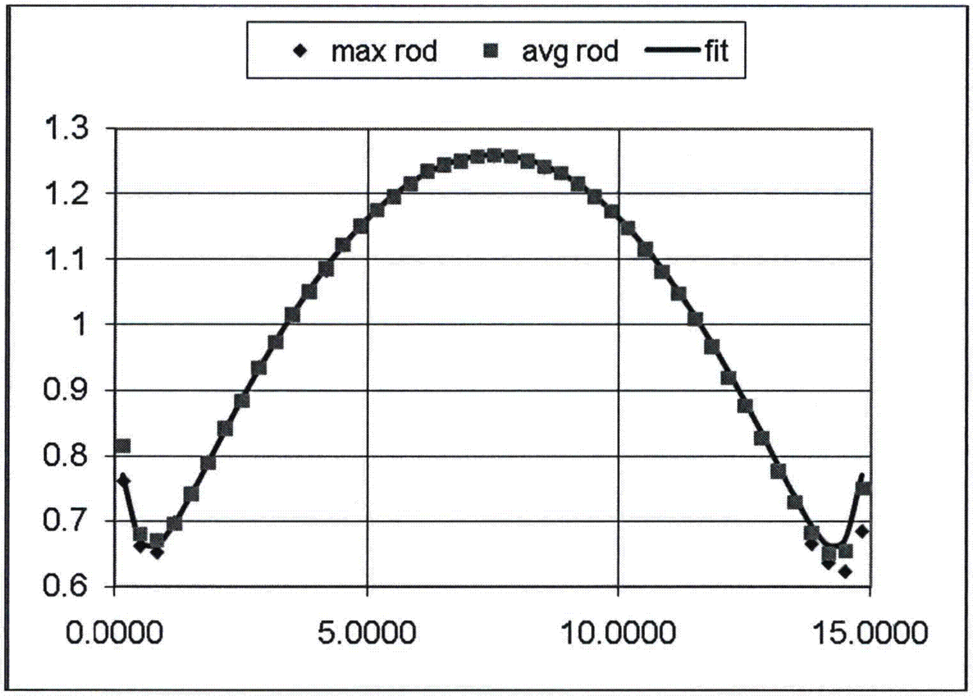


Figure 4-31: PRNC Axial Power Profile vs. Distance from Bottom of Fueled Section (inches)

4.5.10 Power Peaking; Temperature Peaking – NSC Core

The peaking factors calculated for the NSC LEU 30/20 core at both BOL and EOL are shown in Table 4-16 and Table 4-17. The axial power distribution shown in Figure 4-32 is relatively independent of fuel temperatures or radial position in the core. Unlike PRNC, the peak power density does not occur in the fuel element with the maximum rod factor power. Therefore, it is necessary to know the rod factor in both the maximum power rod, $(\overline{P}_{rod} / \overline{P}_{core})_{max}$ and the peak power density rod, $(\overline{P}_{rod} / \overline{P}_{core})_{peak}$.

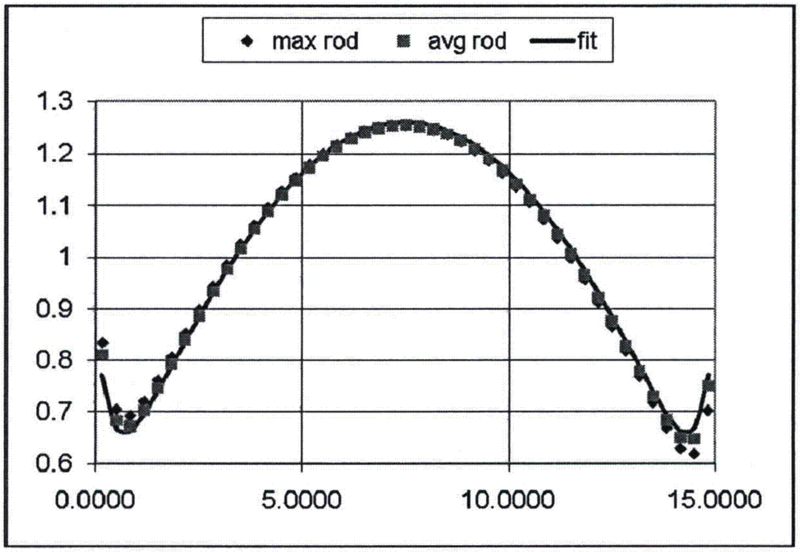


Figure 4-32: NSC Axial Power Profile vs. Distance from Bottom if Fueled Section (inches)

Table 4-16: Power Peaking Factors – NSC – BOL

Type of Peaking	Beginning of Core Life 90 Fuel Rods
$(\overline{P}_{rod} / \overline{P}_{core})_{max}$	1.565
$(\hat{P} / \overline{P})_{axial}$	1.26
Average $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	1.57
$(\overline{P}_{rod} / \overline{P}_{core})_{peak}$	1.446
Peak $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	2.297

Table 4-17: Power Peaking Factors – NSC - EOL

Type of Peaking	End of Core Life (2000 MWD) 90 Fuel Rods
$(\overline{P}_{rod} / \overline{P}_{core})_{max}$	1.511
$(\hat{P} / \overline{P})_{axial}$	1.26
Average $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	1.530
$(\overline{P}_{rod} / \overline{P}_{core})_{peak}$	1.352
Peak $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$	2.200

The fuel temperatures for NSC steady state operation have been calculated for the hottest, measured, and average fuel rods. These results are presented in Table 4-18. The value for $\hat{T}_{0.3}$ is the thermocouple temperature that is located 0.3 inch from the fuel centerline.

Table 4-18: Calculated Fuel Temperatures (°C), NSC

P (MW)	$\hat{T}$ (°C)	$\hat{T}_{0.3}$(°C)	$\bar{T}_{core}$ (°C)
0.5	289	260	205
1.0	373	329	237
1.3	440	384	249

For the NSC reactor, note that the instrumented fuel element (IFE) is not located at the hottest core position. The IFEs are located in 5E4 and 6D4; the hottest fuel element is calculated to be 5D3 (see Figure 4-2 for location). Other positions can be chosen for the IFEs. The sensing tip of the thermocouple is 0.3 inch (7.62 mm) from the fuel axial center line, just outside the 0.25 in (6.35 mm) diameter zirconium rod positioned along the axial center of the fuel. The results reported in Table 4-18 give the peak fuel temperature $\hat{T}$ in the hottest fuel element, the computed temperature $\hat{T}_{0.3}$ in the instrumented fuel element (IFE) that can be compared with future measured temperatures, and the average core temperature, $\bar{T}_{core}$.

The average power per fuel element in the NSC core operating at 1.0 MW with [redacted]. For the following locations, refer to Figure 4-2. The fuel element immediately adjacent to the Transient Rod (5D3) produces the greatest power (17.4 kW per element). An ideal location for one of the IFE would be 5D3; however, this location is within the cluster that has the transient rod. A three-rod locking plate on this cluster precludes placing an IFE within the cluster. The IFE is located at the next nearest location, 5E4, where the power generated is 15.4 kW per element.

The water in position 3D leads to significant power peaking, especially in fuel elements such as 4D3 and 4D4. An IFE in 4D4 would generate a slightly lower power (15.3 kW/element), but has a peak power density closer to the maximum peak power density of the whole core, even larger than in the fuel element immediately adjacent to the Transient Rod (5D3). Consequently, a thermocouple element ($T_{0.3}$) in this location would give the highest measured fuel temperature during pulsing.

The core configuration in Figure 4-2 indicates that a second IFE is located at 6D4. In this location the power generated is 14.0 kW per element with a low-power density.

4.5.11 Pulsing Operation – PRNC

Most of the 65 TRIGA reactors have pulsing capability. Thousands of TRIGA reactor pulses have been safely performed. A computational procedure, (TRIGA-BLOOST) based on a space-independent kinetics model[8] has been developed for predicting pulse performance of TRIGA reactors.

The following simplified relationships are given to show qualitatively how pulsing performance is influenced by the important reactor parameters:

$$\tau \doteq \ell / \Delta k_p = \text{Reactor period}$$

$$\overline{\Delta T} = \frac{2\Delta k_p}{\alpha} = E / C$$

$$\hat{P} = \frac{C(\Delta k_p)^2}{2\alpha\ell} = \text{ Peak pulsed power}$$

$$E = \frac{2C\Delta k_p}{\alpha} = \text{Total energy release in prompt burst}$$

where

ℓ	= prompt neutron life
α	= prompt negative temperature coefficient
C	= total heat capacity of the core available to the prompt pulse energy release
$\overline{\Delta T}$	= change in average core temperature produced by the prompt pulse
Δk_p	= that portion of the step reactivity insertion which is above prompt critical

Water filled regions within the core promote flux peaking and result in increased power peaking and peak fuel temperatures, especially during a reactivity pulse. The PRNC core was a compact core with no in-core experimental regions that could be water filled. However, all five control rods had water-followed regions and thus constituted regions of enhanced power peaking. These regions were correctly modeled in the codes that are used. The instrumented fuel element gave the peak fuel temperature because it was positioned beside a control rod with its water follower.

The BLOOST pulsing performance results have been prepared for reactivity insertions of $1.45, $1.95, and $2.30. Measured data from pulses performed during the commissioning tests are available for direct comparisons. This data is also presented in Table 4-19.

Table 4-19: Pulse Performance: Calculated and Measured, PRNC 95 TRIGA HEU FLIP Fuel

Parameter	\$1.45 Pulse	\$1.95 Pulse	\$2.30 Pulse
Measured Data			
$\hat{P}\ (MW)$	308	1163	1954
$E\ (MW - \sec)$ *	11.5	17.4	20.5
$\hat{T}_{0.3}$ (°C)	306	411	466
BLOOST Calculations			
$\hat{P}\ (MW)$	436	1814	3262
$E\ (MW - \sec)$	16.8	27.2	33.5
$\hat{T}$ (°C)	488	766	909
$\overline{T}_{core}$ (°C)	177	286	347
$\hat{T}_{0.3}$ (°C)	327	515	618

* All measured values of E contain a small "tail" contribution (until the 1.0 sec scram) estimated at 25-33% of total value

The values for pulsing during the commissioning of the PRNC FLIP core were significantly smaller than those predicted in the PRNC SAR. These experimentally measured values are also smaller than the values currently calculated. It may be noted that the peak thermocouple reading ($\hat{T}_{0.3}$) in the hottest fuel rod has also been calculated for a distance of 0.3 inch from the fuel vertical centerline, which is the same location as the sensing tip of the thermocouple in the instrumented fuel rod.

The results for energy from the BLOOST-calculations are reported at a time of about one second after the pulse initiation (well after the peak pulsed power) and at about the time a control rod scram was initiated. The pulse is completed about 0.3 seconds after pulse initiation at which time the peak fuel temperature is computed. Thereafter, the peak fuel temperature (at the outer surface of the fuel rods) decreases as energy flows both to the center of the fuel rod and to the cooling water. However, the average core temperature continues to rise as energy is accumulated from the "tail" of the pulse until at about one second, when all control rods were scrammed, shutting down the pulsed reactor. Account has been taken for the power peaking in the water filled region when the transient rod is pulsed. The hottest fuel rod is the instrumented fuel element and is adjacent to the transient rod, reflecting this power peaking.

4.5.12 Pulse Operation – NSC – BOL, Calculated

The NSC reactor has extensive experience with pulsing performance using fuel with a strong, prompt negative temperature coefficient of reactivity (α). The new LEU 30/20 fuel also produces a similarly strong temperature dependence on α. Comparing the curves for α for FLIP fuel (Figure 4-26) and for LEU 30/20 fuel (Figure 4-27), one notes similar temperature dependences; however, the magnitude of α is somewhat smaller for the LEU fuel. The BOL neutron lifetime is 26.3 μsec; the EOL neutron lifetime is 27.3 μsec.

Table 4-20 presents the beginning of core life pulsing parameters for the NSC LEU 30/20 core. Results for the FLIP core are included for easy comparison.

Figure 4-33 shows a plot of the pulsed fuel temperatures as a function of reactivity insertion for NSC LEU 30/20 fuel at BOL. Results for reactivity insertions of $1.45, $1.95, $2.30, and $2.50 are shown. The peak fuel temperature in the core (4D3) is listed. Not only are measured temperature results ($\hat{T}_{0.3}$) shown for the planned instrumented fuel element in position 5E4, but also for positions 4D4 and 6D4. The position 4D4 was included since a fuel element in this location experiences the highest peak power density (though not the highest power per element) and hence the highest measured temperature, $\hat{T}_{0.3}$. Since it will be shown in a later section that the peak pulsed fuel temperature will need to be limited to 830°C, the reactivity insertion ($2.10) to produce this temperature is indicated in Figure 4-33. Table 4-20 also shows that the maximum reactivity insertion of the transient rod must be limited to $2.95 in order to limit peak fuel temperatures to less than the 1150°C safety limit. An electro-mechanical interlock on the UP position of the transient rod and pulsing circuitry shall prevent accidental pulsing of more than $2.95 thereby preventing a pulse, which could exceed the safety limit based on the analysis presented in Table 4-20.

At NSC, it has been standard practice to place one of the two IFEs in 5E4 and a second one at 6D4. This analysis suggests that 4D4 would be another suitable location for the IFE.

Table 4-20: Calculated Pulse Performance for [redacted] – BOL

Parameter	Flip Fuel [redacted]			LEU 30/20 Fuel [redacted]					
	$1.45	$1.95	$2.30	$1.45	$1.95	$2.30	$2.50	$2.95	$3.21
$\hat{P}$ (MW)	436	1814	3262	227	1008	1873	2468	3434	3775
E (MW-sec)	16.8	27.2	33.5	11.3	19.8	25.5	28.2	29.6	35.5
$\hat{T}$ (°C) (4D3)	488	766	909	459	755	921	1006	1144	1182
$\overline{T}_{core}$ (°C)	177	286	347	155	264	330	365	422	438
$\hat{T}_{0.3}$ (°C) (5E4)	327	515	618	200	333	413	456	526	545

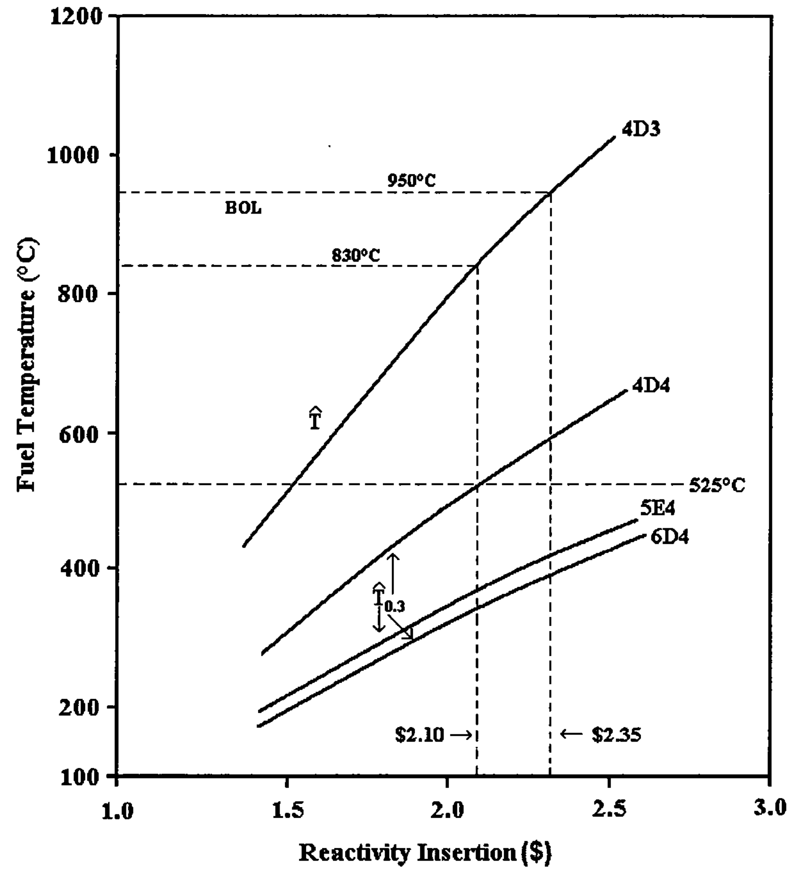


Figure 4-33: Peak Fuel Temperature and Typical IFE Measured Fuel Temperatures as a Function of Reactivity Insertion at Beginning of Core Life – NSC

4.5.13 Pulse Operation – NSC – BOL, Measured

The NSC conducted a series of pulses (all less than \$1.56) to determine various parameters of the new core and compare them to the calculated values. Pulses were conducted per the NSC Standard Operating Procedures with reactor power and temperature indication being collected by the NSC Reactor Controls Console Computer. The NSC PULSE program installed on the computer collected and plotted the pulse data.

Total pulse energy was calculated integrating pulse power from the time the pulse was initiated until the reactor was scrammed (approximately two seconds following pulse initiation). Peak core temperature was calculated using the peak measure core temperature and the peak temperature equation of Section 4.5.12.

Finally, total pulse energy release and peak power were plotted versus Δk_p and $\Delta k_p^{\,2}$ respectively to verify linearity. The measurements obtained from a series of pulses from \$1.15 to \$1.56 yielded the results of Table 4-21.

Table 4-21: NSC – BOL Pulse Measurements

Δk_p	FWHM	$\hat{P}$	E	T_{amb}	$\hat{T}_{0.3}$	$\hat{T}_{0.3} - T_{amb}$	$\hat{T}$	$\hat{T} - T_{amb}$
\$	ms	MW	MW-sec	°C	°C	°C	°C	°C
0.15	33.2	33.2	7.67	33.33	137.78	104.44	297.17	263.83
0.22	63.6	54.7	10.5	32.33	175.11	142.78	388.11	355.78
0.24	64.2	64.2	10.4	31.44	176.78	145.33	393.39	361.94
0.42	30.2	222.7	14.73	30.56	211.94	181.39	476.83	446.28
0.52	24.1	343.8	16.45	30.78	257.83	227.06	580.94	550.17
0.33	40.5	129.9	13.06	36.89	212.39	175.50	469.44	432.56
0.47	25.9	294.9	15.95	34.78	249.67	214.89	557.67	522.89
0.56	21.6	423.8	17.56	33.89	275.39	241.50	616.39	582.50

A plot of energy released during a pulse versus prompt reactivity insertion is a straight line as expected. A plot of maximum power in the core or in the instrumented element versus prompt reactivity insertion squared also give straight lines as expected as demonstrated on Figure 4-34 and Figure 4-35.

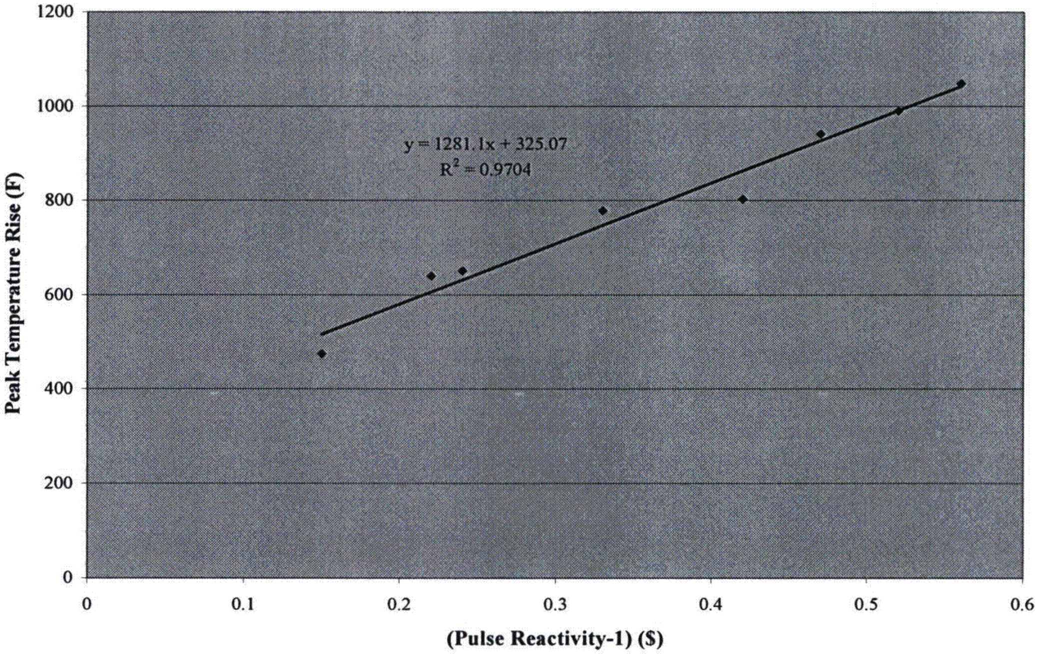


Figure 4-34: ($\hat{T}$ - T_{amb}) vs. Δk_p

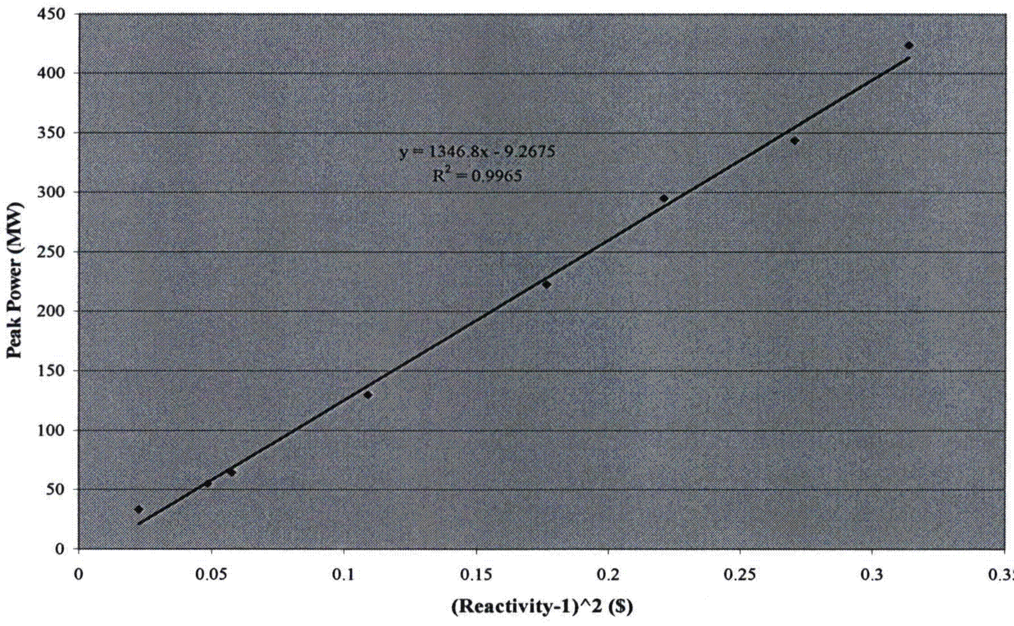


Figure 4-35: Peak Power vs. $\Delta k_p{}^2$

From this data, it is calculated that the maximum prompt insertion should be $1.91 so as to keep the maximum temperature in the core below 830°C. The BLOOST predicted $2.10.

The BLOOST predicts the energy released from a \$1.45 pulse is 11.3 MW-sec. Interpolation from measured data gives 15.3 MW-sec, but this value contains part of the residual tail in the pulse. The BLOOST predicts a 177°C rise as measured by the instrumented element during a \$1.45 pulse; interpolated from measured data, yields a rise of 188°C.

The measured and predicted data are in excellent agreement. For \$1.45 pulse the measured data agree with predicted data within experimental error. Error can arise both from the measured values themselves (i.e., thermocouple indications, and pulse power channel indications) and from error in the assigned value of the prompt insertion. The error in transient rod calibration can easily be a few cents and this error alone is sufficient to account for all the difference between the measured and predicted data.

Note: Using the data on Table 4-20, the NSC calculates peak core temperature with the following equation:

$$\hat{T} = 2.9445 \times 10^{-7} \times \left(\hat{T}_{0.3} - T_{ambient}\right)^2 - 9.2520 \times 10^{-4} \times \left(\hat{T}_{0.3} - T_{ambient}\right) + 2.6183 + T_{ambient}$$

The temperature in this calculation is in °C.

4.5.14 Pulse Operation – NSC – EOL

The BLOOST code was used to calculate the pulsing performance of the LEU 30/20 core at 2000 MWD burnup. The procedure is the same as used above for BOL conditions. Results are shown in Table 4-22 for reactivity insertions from low power (300W) of \$1.45, \$1.95, \$2.30, \$2.50, \$2.95, and \$3.21. Results are presented for peak pulsed power, integrated energy, peak fuel temperature in hottest fuel rod, average reactor core temperature, and thermocouple measured temperature.

Figure 4-36 illustrates the dependence of fuel temperatures on reactivity insertions at 2000 MWD burnup. A particular object is to explore whether the reactivity insertion of \$2.10 will give higher or lower peak fuel temperature compared to the results at start of core life. (See Figure 4-33 for beginning-of-life results.) A reactivity insertion of \$2.10 at 2000 MWD burnup will give a slightly lower peak fuel temperature (i.e., 810°C in contrast to the beginning value of 830°C).

In view of the fact that the peak fuel temperatures ($\hat{T}$) at 2000 MWD are lower rather than higher than the initial peak fuel temperature, the limiting reactivity insertion for pulse operation can safely remain at \$2.10, as determined earlier.

Similarly, the maximum allowed accidental pulse of \$2.95, which, was shown previously in Table 4-20 to accommodate the safety limit of 1150°C, is also shown in Table 4-22 to meet the safety limit. Therefore, the electro-mechanical interlock discussed in the Section 4.5.12 will prevent an accidental pulse from exceeding the safety limit both at BOL and EOL.

Table 4-22: Calculated Pulse Performance fo[redacted] for BOL and EOL – NSC

Parameter	LEU 30/20 Fuel [redacted]				LEU 30/20 Fuel [redacted]					
	$1.45	$1.95	$2.30	$2.50	$1.45	$1.95	$2.30	$2.50	$2.95	$3.21
$\hat{P}$ (MW)	227	1008	1873	2468	230	1028	1915	2536	3716	4046
E (MW-sec)	11.3	19.8	25.5	28.2	11.8	21.2	23.0	30.9	33.0	35.0
$\hat{T}$ (°C) (4D3)	459	755	921	1006	442	733	901	988	1142	1188
$\overline{T}_{core}$ (°C)	155	264	330	365	162	278	350	388	458	478
$\hat{T}_{0.3}$ (°C) (5E4)	200	333	413	456	208	354	444	492	577	602

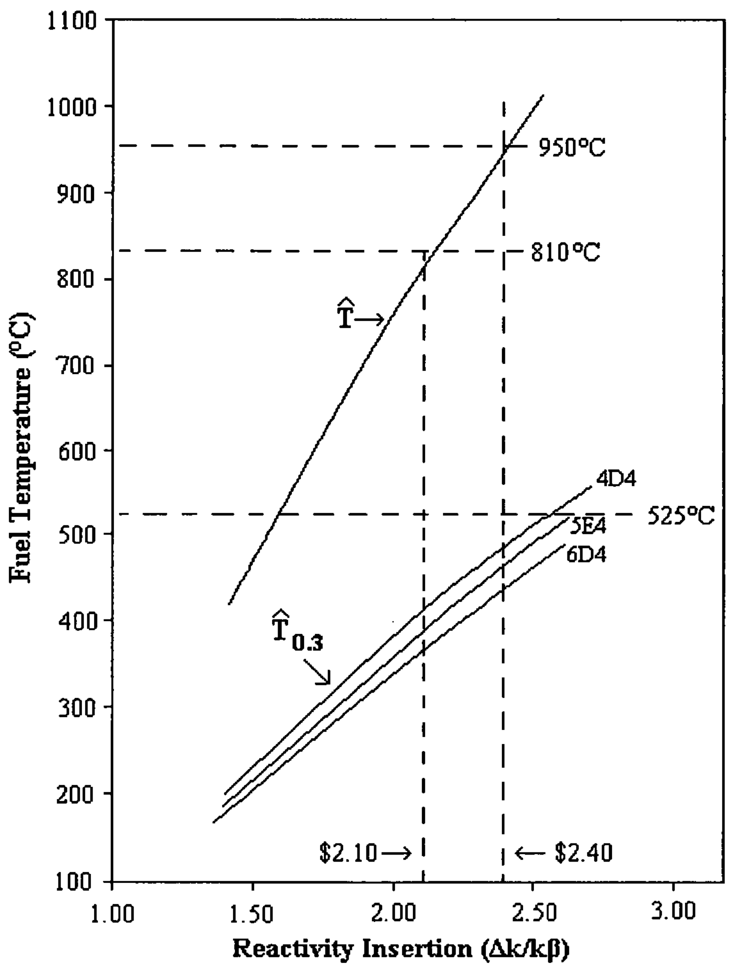


Figure 4-36: Peak Fuel Temperature ($\hat{T}$) and Typical IFE Measured Fuel Temperatures ($\hat{T}_{0.3}$) as a Function of Reactivity Insertion at 2000 MWD Burn-Up

4.6 Thermal-Hydraulic Analysis – PRNC

4.6.1 Analysis of Study State Operation

The following evaluation has been made for a TRIGA system operating with cooling from natural convection water flow around the fuel elements. In this study, the predicted steady state thermal-hydraulic performance of the PRNC TRIGA core was determined for the reactor operating at 1.4 MW and a water inlet temperature of 30°C. Although the PRNC TRIGA reactor was designed and operated at 2.0 MW, extensive measurements were made at 1.4 MW during the commissioning tests. This experimental data was used for the benchmark comparisons. An average powered fuel rod and a maximum powered fuel rod were analyzed. The STAT computer code[9] was used to determine the natural convection flow rate, the coolant and clad axial temperature profiles, and the clad wall heat flux axial profile. The STAT code was also used to determine the clad wall maximum heat flux versus coolant inlet temperature for departure from nucleate boiling. The TAC2D thermal analysis code[10] was used to determine the fuel average and fuel maximum temperatures.

4.6.2 STAT Code Analysis and Results

STAT is a computer program for calculating the natural convection heat transfer and fluid flow in an array of heated cylinders. It calculates the natural convection flow through a vertical water coolant channel bounded by cylindrical heat sources. The STAT model examines a single fuel element and a representative flow area associated with that single fuel element. This approach is conservative since the flow area is not shared by higher and lower powered fuel elements. Output from the STAT code includes: channel flow rate, outlet velocity, temperature rise of the fluid along the channel, maximum heat flux, and maximum clad temperature. The assumption is made that there is no cross flow between adjacent channels. Input to the program includes the following:

1. Size and spacing of the heat sources;
2. Axial heat source distribution;
3. Pressure head above the source;
4. Constants to be used in generic expressions for convective and sub-cooled boiling heat transfer coefficients and for the convective friction factor;
5. Inlet and exit pressure loss coefficients;
6. Inlet water temperature.

Analysis is performed on a single flow channel divided into axial segments. The natural convection system for the PRNC TRIGA was based on the four-rod cluster of fuel elements. The representation used herein establishes one flow channel bounded by four fuel elements. The reactor geometry, power factors and heat transfer and friction data for the STAT input are given in Table 4-23: STAT Input for Reactor and Core Geometry and Heat Transfer, PRNC.

A STAT thermal hydraulic analysis was done for an average flow channel and a maximum powered channel. The analysis was conducted by considering the hydraulic characteristics of a typical flow channel represented by the geometric data given in Table 4-24.

The heat generation in the fuel element is distributed axially in a cosine distribution chopped at the end such that the peak-to-average ratio is 1.27 (Table 4-15). It is further given that there are [redacted] in the initial core. The hot-rod power ratio is assumed to be 1.55 (Table 4-15) for initial core conditions.

The axial power profile for STAT is given by the analytical equation:

$$Q(z) = APF * \cos\left(\pi \frac{(z - ZL/2)}{Z_{ext}}\right) * \left(1 + G\,E * \exp\left[\frac{-AK * z}{L}\right]\right)$$

where Z_{ext}, GE, and AK are parameters to fit the shape of the axial profile determined by DIF3D having the following numerical values:

1. $Z_{ext} = 49.784$ cm
2. GE = 0.77
3. AK = 45.0

Table 4-23: STAT Input for Reactor and Core Geometry and Heat Transfer, PRNC

Parameter	Value
Core and Reactor Geometry	
Unheated core length at inlet, mm	100
Unheated core length at outlet, mm	103
Length from top of unheated length to top of locking plate, mm	21
Distance from top of pool surface to top of locking plate, mm	7170
Heat Transfer and Friction Data	
Void detachment fraction	0.0
Sub-cooled boiling correlation coefficient	0.2494
Sub-cooled boiling correlation exponent	3.797
Convection correlation coefficient	0.023
Convection correlation exponent for the Reynolds number	0.8
Convection correlation exponent for the Prandtl number	0.4
Friction factor correlation coefficient	0.079
Friction factor correlation exponent	0.25
Inlet pressure loss coefficient	1.672
Exit pressure loss coefficient	0.6
Ambient pressure at elevation 750 m, MPa	0.092
Peaking Factor (Axial and Radial)	
Initial Core Axial Power Factor	1.27
Initial Core Hot Rod Factor	1.55

Table 4-24: Hydraulic Flow Parameters for a Typical Flow Channel

Parameter	Value
Flow area (mm^2/element)	552.26
Wetted perimeter (mm/element)	112.6
Hydraulic diameter (mm)	19.62
Fuel element heated length (mm)	[redacted]
Fuel element diameter (mm)	[redacted]
Fuel element surface area (mm^2)	[redacted]

The driving force is supplied by the buoyancy of the heated water in the core. Countering this force are the contraction and expansion losses at the entrance and exits to the channel, the acceleration and potential energy losses, and fiction losses in the cooling channel itself. The pressure drops through the flow channel are dependent on the flow rate while the available static driving pressure is fixed for a known core height and ambient pressure.

Cross flow from one flow channel to another is ignored by restricting the flow modeling to a single fuel element. Cross flow would allow flow mixing from cooler, lower powered flow channels into the maximum powered flow channel. The total pressure drop in the average and maximum fuel rod channel is balanced by the pressure gain due to buoyant forces so that the pressures at the exit of the average and hot channels are equal[18]. The total pressure drop reported by STAT does not include the pressure change due to elevation and density differences between the hot fluid in the core and the cold fluid outside of the core, which determines the pressure of the water at the core inlet. The hydrostatic head is not counted as part of the pressure loss but as part of the buoyancy pressure gain. STAT integrates the density changes as the coolant passes up through the core.

Pressures at different elevations of the average and hot channels are also essentially equal due to the balance of pressure losses and buoyancy pressure gains. Various experiments have revealed that this cross flow is negligible for tightly packed geometries such as PRNC and NSC[11, 12, 13, 14]. The ultimate effect of cross flow mixing is to increase Critical Heat Rate (CHR), so that its neglect through the use of subchannel approach is expected to result in conservative estimate of Critical Heat Flux (CHF).

The STAT code calculates both a convection and a boiling heat transfer coefficient and uses the larger of these two to determine the clad-to-fluid temperature difference. The boiling heat transfer correlation, the convection heat transfer correlation, and the friction factor equation as used by STAT are, respectively:

1. $Q_{boiling} = 0.2494(T_w - T_{sat})^{3.797}$
2. $Nu = 0.023\,\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.4}$
3. $f = \dfrac{0.079}{\mathrm{Re}^{0.25}}$

Where $Q_{boiling}$ is in BTU/hr-ft2 and T_w and T_{sat} are in °F[15]. The correlation is for sub-cooled boiling in a narrow vertical annulus at pressures of 2-6 atm and flow velocity of 1 to 12 ft/sec. The natural circulation flow loop established due to buoyancy difference between the coolant heated in the core and the cold pool water behaves like a low-head pump generating a low velocity flow through the core. It does not behave like free convection from either a horizontal or vertical surface. Under normal operating conditions of 1.0 MW, the flow velocity in the maximum powered channel is 0.5 ft/s but increases to 1 ft/s for departure from nucleate boiling (DNB) conditions as determined in the thermal hydraulic calculations using STAT.

The zero-void-detachment fraction given in Table 4-23 implies that none of the vapor bubbles that are generated due to subcooled boiling detach from the wall surface enter the main coolant and subsequently increase the buoyancy of the main coolant stream.

A summary of the STAT results for PRNC is given in Table 4-25.

Table 4-25: Steady State Results for PRNC, 1.4 MW (Results are for One Flow Channel)

Initial Core (95 Elements)	Average Rod	Maximum Rod
Channel natural convection mass flow rate, kg/sec	0.084	0.97
Channel total pressure drop, Pa	55.59	81.11
Exit coolant flow temperature, °C	71.86	86.50
Maximum wall heat flux, W/cm^2	43.49	67.41
Maximum flow velocity, cm/sec	15.57	18.14
Maximum clad temperature, °C	135.1	137.3
Exit clad temperature, °C	132.3	134.2

The STAT code also calculates the maximum nucleate boiling heat flux, that is, the heat flux at which there is a DNB and the transition to film boiling begins. This is also termed the critical heat flux. Two correlations are used to calculate this heat flux. The first, given by McAdams[16], indicates that the critical heat flux is a function of the fluid velocity and the fluid only. The second correlation is due to Bernath[17]. It encompasses a wider range of variables over which the correlation was made and it takes into account the effect of different flow geometries. It generally gives a lower value for the critical heat flux. The lower critical heat flux from the correlations is used here for determining the minimum DNB ratio, that is, the minimum ratio of the local allowable heat flux to the actual heat flux. The minimum DNB ratio is given in

Table 4-26.

The STAT code analysis has been run for the critical heat flux for the PRNC reactor operating at 1.4 MW. The data was obtained by first selecting an inlet temperature (30°C) and then systematically increasing the reactor power until STAT indicated a DNB ratio equal to one. The maximum power per fuel element for which the DNB ratio is 1 versus the core inlet water temperature is 44 kW/element. For a core with a rod peaking factor of 1.55, this maximum fuel element power corresponds to a maximum reactor power of 2.70 MW. Hence, the DNB ratio for the PRNC TRIGA at the stated conditions is 1.93. These values are for BOL core conditions. The minimum DNB ratio is listed in

Table 4-26.

Table 4-26: TRIGA Thermal and Hydraulic Parameters for PRNC, 1.4 MW

Parameter	Initial Core Value
Number of fuel elements	[redacted]
Diameter, mm (in.)	[redacted])
Length (heated), mm (in.)	[redacted]
Core flow area, mm^2 (ft^2)	52465 (0.565)
Core wetted perimeter, mm (ft.)	10,697 (35.10)
Flow channel hydraulic diameter, mm (ft.)	19.62 (0.0644)
Core heat transfer surface, m^2 (ft^2)	4.08 (43.87)
Hot rod factor	1.55
Axial peaking factor	1.27
Hot spot peaking factor*	1.97
Inlet coolant temperature, °C (°F)	30 (86)
Coolant saturation temperature, °C (°F)	117.0 (242.6)
Exit coolant temperature (average), °C (°F)	71.86 (161.3)
Exit coolant temperature (maximum), °C (°F)	86.50 (187.7)
Coolant mass flow, kg/sec (lb/hr)	8.02 (63,626)
Average flow velocity, mm/sec (ft/sec)	154 (0.506)
Peak fuel temperature in average fuel element, °C (°F)	333.6 (632.4)
Maximum wall temperature in hottest element, °C (°F)	137.3 (279.2)
Peak fuel temperature in hottest fuel element, °C (°F)	444.2 (831.5)
Core average fuel temperature, °C (°F)	232.2 (449.9)
Average heat flux, W/cm^2 (BTU/hr-ft^2)	34.35 (108,898)
Maximum heat flux in hottest element, W/cm^2 (BTU/hr-ft^2)	67.41 (213,734)
Minimum DNB ratio	1.93
*The "hot spot peaking factor" is defined as the product of the rod power factor and the axial peak to average power ratio.	

4.6.3 TAC2D Fuel Temperature Analysis and Results

STAT does not calculate fuel temperatures. The TAC2D general purpose heat conduction code was used to calculate steady state maximum and average fuel temperatures for the average powered rod and the maximum powered rod[7]. A radial-axial (R,Z) two-dimensional model of the center zirconium rod (6.35 mm diameter), the fuel annulus, the fuel-to-clad gap, and the 0.5 mm thick stainless steel cladding of a single fuel pin was constructed. The model included only the active length of the fuel pin.

TAC2D is a code for calculating steady state and transient temperatures in two-dimensional problems by the finite difference method. The configuration of the body to be analyzed is described in the rectangular, cylindrical, or circular coordinate system by orthogonal lines of constant coordinate called grid lines. The grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite difference equation is formulated for each nodal point in terms of its heat capacity, heat generation, and heat flow paths to neighboring nodal points.

The TAC2D code requires an input of a geometric description of the problem and properties of the materials considered. The radial and axial power distributions in the fuel are also provided as input. The problem is defined in cylindrical R-Z geometry. The axial distribution of the clad wall temperature from the STAT code was imposed on the outside surface of the clad for the TAC2D model outer radial boundary. Alternatively, the axial distribution of the surface heat transfer coefficient and coolant temperature from the STAT code could be used to model the outer radial boundary with consistent results. The fuel-to-clad interface conductance assumes the fuel pin is sealed with air and has an initial 0.003 mm cold gap. Reported fuel temperatures do not account for fuel swelling over long periods of operation and are therefore conservative. Some gap closure occurs due to the relative expansion of the fuel and cladding at normal operating temperatures. As the reactor is operated over time and irradiation effects in the fuel induce swelling, the gap closes even further, tending to increase the gap conductance and in turn decrease the peak fuel temperature.

A summary of the TAC2D results for PRNC has been given in Table 4-13 for 0.5, 1.0, and 1.4 MW operation with [redacted].

4.7 Thermal Hydraulic Analysis – NSC LEU

4.7.1 Analysis of Steady State Operation – NSC

The following evaluation has been made for the TRIGA LEU 30/20 fuel system with four-rod configuration operating with cooling from natural convection water flow through four-rod clusters of fuel. The steady state thermal-hydraulic performance of the NSC LEU TRIGA core was determined for operation at 1.0 MW with a water inlet temperature of 30°C.

An average powered fuel rod and a maximum powered fuel rod were analyzed. The STAT computer code[6,18] was used to determine the natural convection flow rate, the coolant and clad axial temperature profiles, and the clad wall heat flux axial profile. The STAT code was also used to determine the clad wall maximum heat flux versus coolant inlet temperature for departure from nucleate boiling. The TAC2D thermal analysis code[7] was used to determine the fuel average and fuel maximum temperatures.

4.7.2 STAT Code Analysis and Results – NSC

The STAT analysis was performed using the method outlined in Section 4.6. The reactor geometry, power factors, heat transfer and friction data for the STAT input are given in Table 4-27. The natural convection system for the NSC TRIGA was based on the four-rod cluster of fuel elements. The representation used herein establishes one flow channel bounded by four fuel elements.

Table 4-27: STAT Input for Reactor and Core Geometry and Heat Transfer, NSC

Parameter	Value
Core and Reactor Geometry	
Unheated core length at inlet, mm	100
Unheated core length at outlet, mm	103
Length from top of unheated length to top of locking plate, mm	21
Distance from top of pool surface to top of locking plate, mm	7170
Heat Transfer and Friction Data	
Void detachment fraction	0.0
Sub-cooled boiling correlation coefficient	0.2494
Sub-cooled boiling correlation exponent	3.797
Convection correlation coefficient	0.023
Convection correlation exponent for the Reynolds number	0.8
Convection correlation exponent for the Prandtl number	0.4
Friction factor correlation coefficient	0.079
Friction factor correlation exponent	0.25
Inlet pressure loss coefficient	1.672
Exit pressure loss coefficient	0.6
Ambient pressure at elevation 750 m, MPa	0.092
Peaking Factor (Axial and Radial)	
Initial Core Axial Power Factor	1.26
Initial Core Hot Rod Factor	1.565

A STAT thermal hydraulic analysis was done for an average flow channel and a maximum powered channel. The analysis was conducted by considering the hydraulic characteristics and flow parameters of a typical flow channel represented by the geometric data given in Table 4-28.

The heat generation in the fuel element is distributed axially in a cosine distribution chopped at the end such that the peak-to-average ratio is 1.26 (Table 4-16, Table 4-17). It is further given that there are [redacted] in the initial core. The hot-rod power ratio is assumed to be 1.565 for initial core conditions.

Table 4-28: Hydraulic Flow Parameters, NSC

Parameter	Value
Flow area (mm^2/element)	552.26
Wetted perimeter (mm/element)	112.6
Hydraulic diameter (mm)	19.62
Fuel element heated length (mm)	[redacted]
Fuel element diameter (mm)	[redacted]
Fuel element surface area (mm^2)	[redacted]

The parameters Zext, GE, and AK for the axial power profile equation Q(z) are:

1. Zext = 50.212 cm
2. GE = 0.70
3. AK = 45.0

A summary of the STAT results for the 1.0 MW NSC 30/20 is given in Table 4-29.

Table 4-29: Steady State Results for NSC, 1.0 MW

Initial Core (90 Elements)	Average Rod	Maximum Rod
Channel natural convection mass flow rate, kg/sec	0.077	0.089
Channel total pressure drop, Pa	44.15	63.87
Exit coolant flow temperature, °C	64.79	76.82
Maximum wall heat flux, W/cm^2	32.54	50.92
Maximum flow velocity, cm/sec	14.09	16.50
Maximum clad temperature, °C	133.8	135.9
Exit clad temperature, °C	131.4	133.2

4.7.3 TAC2D Fuel Temperature Analysis and Results – NSC

Using the methods given in Section 0, a TAC2D analysis was performed for the 1.0 MW NSC LEU 30/20 core for operation with [redacted]. See Table 4-18.

4.7.4 Steady State Analysis Results Summary – NSC

Table 4-30 lists the pertinent heat transfer and hydraulic parameters for the 1.0 MW NSC TRIGA reactor. Results are presented therein for an average channel and a maximum powered channel (hot channel) at initial core conditions. Also shown are the peak fuel temperatures in the hottest and average fuel element calculated with the TAC2D code.

Table 4-30: TRIGA Thermal and Hydraulic Parameters for NSC, 1.0 MW

No Parameter	Initial Core Value
Number of fuel elements	[redacted]
Diameter, mm (in.)	[redacted]
Length (heated), mm (in.)	[redacted]
Core flow area, mm^2 (ft^2)	49,703 (0.535)
Core wetted perimeter, mm (ft.)	10,134 (33.25)
Flow channel hydraulic diameter, mm (ft.)	19.62 (0.0644)
Core heat transfer surface, m^2 (ft^2)	3.86 (41.56)
Hot rod factor	1.565
Axial peaking factor	1.26
Hot spot peaking factor*	1.97
Inlet coolant temperature, °C (°F)	30 (86)
Coolant saturation temperature, °C (°F)	117.0 (242.6)
Exit coolant temperature (average), °C (°F)	64.79 (148.6)
Exit coolant temperature (maximum), °C (°F)	76.82 (170.3)
Coolant mass flow, kg/sec (lb/hr)	6.90 (54731)
Average flow velocity, mm/sec (ft/sec)	140 (0.459)
Peak fuel temperature in average fuel element, °C (°F)	282.4 (540.4)
Maximum wall temperature in hottest element, °C (°F)	135.9 (276.6)
Peak fuel temperature in hottest fuel element, °C (°F)	368.1 (694.6)
Core average fuel temperature, °C (°F)	206.8 (404.2)
Average heat flux, W/cm^2 ($BTU/hr\text{-}ft^2$)	23.24 (73,690)
Maximum heat flux in hottest element, W/cm^2 ($BTU/hr\text{-}ft^2$)	50.92 (161,438)
Minimum DNB ratio	2.42
*The "hot spot peaking factor" is defined as the product of the rod power factor and the axial peak to average power ratio.	

The STAT code analysis has been run for the critical heat flux for the NSC reactor operating at 1.0 MW. The data was obtained by first selecting an inlet temperature (30°C) and then systematically increasing the reactor power until STAT indicated a DNB ratio equal to one. The maximum power per fuel element for which the DNB ratio is 1 versus the core inlet water temperature is 42 kW/element. For a core with a rod peaking factor of 1.565, this maximum fuel element power corresponds to a maximum reactor power of 2.42 MW. Hence, the DNB ratio for the NSC TRIGA at the stated conditions is 2.42. These values are for BOL core conditions. The minimum DNB ratio is listed in Table 4-30.

4.8 Thermal Neutron Flux Values - NSC LEU Core

4.8.1 Thermal Neutron Flux Values in LEU Core, Calculated

The DIF3D code provides neutron flux values. Figure 4-37 presents a 3-D representation of the thermal neutron distribution throughout the core and into the surrounding water and graphite (including the coupling box).

Figure 4-38 shows the flux plot through the transient rod in a direction perpendicular to the face of the coupler box. In the region near the 24 cm position, the thermal neutrons peak over a 3-4 cm distance in the water adjacent to the core. Near the 70 cm position, the thermal neutrons peak over a significantly greater distance in the graphite of the coupler box.

Figure 4-39 presents a graphical representation of the neutron flux across the core through the transient rod in a direction parallel to the face of the coupler box. This plot starts in the water reflector/shield, crosses the reactor, and ends in the water reflector on the other side of the reactor core. Note that the thermal neutron flux is relatively weak inside of the fuel rods due to the erbium and large uranium loading, but rises substantially in regions of water or graphite.

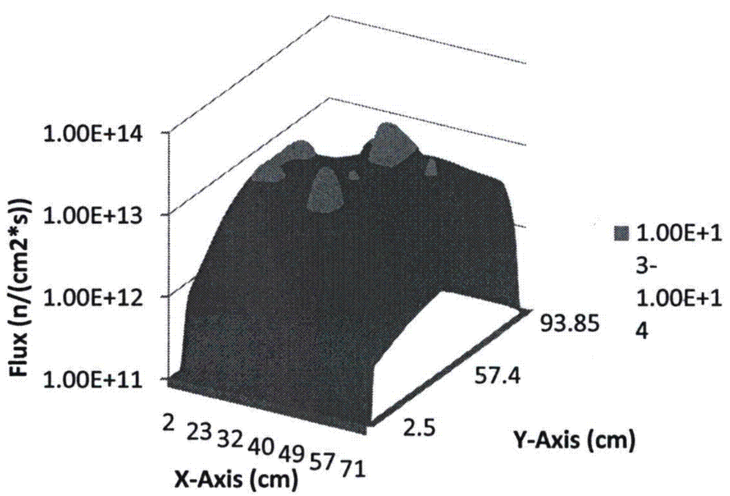


Figure 4-37: NSC Initial Cycle: Thermal Neutron Flux (E<0.42 eV) at Maximum Axial Power Peaking Position (76.7cm) – BOL

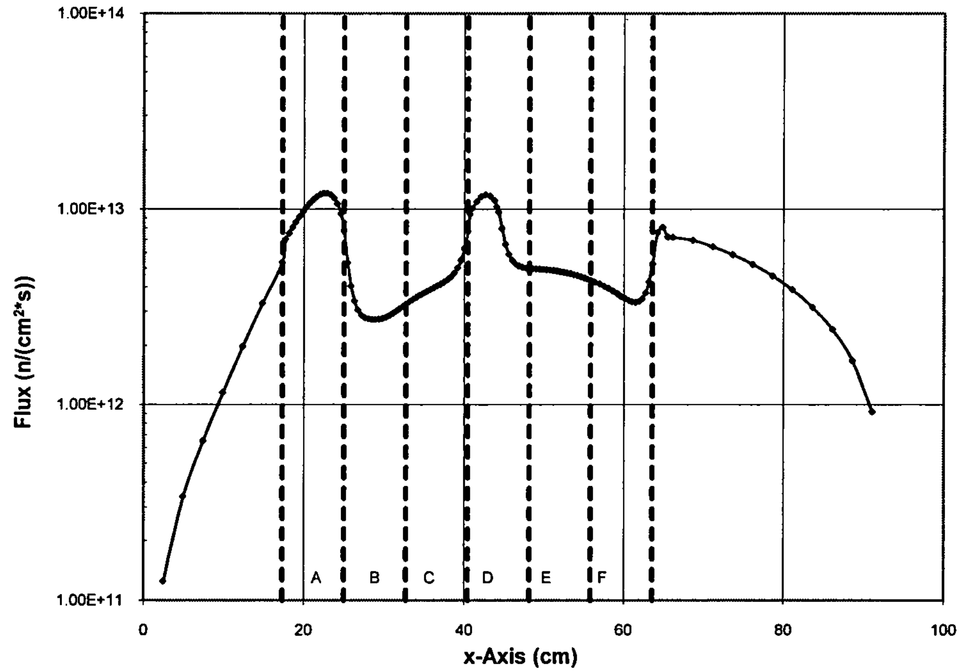


Figure 4-38: NSC Initial Cycle: Thermal Neutron Flux (E<0.42 eV) through the Transient Rod at Maximum Axial Power Peaking Position (76.7 cm) - BOL

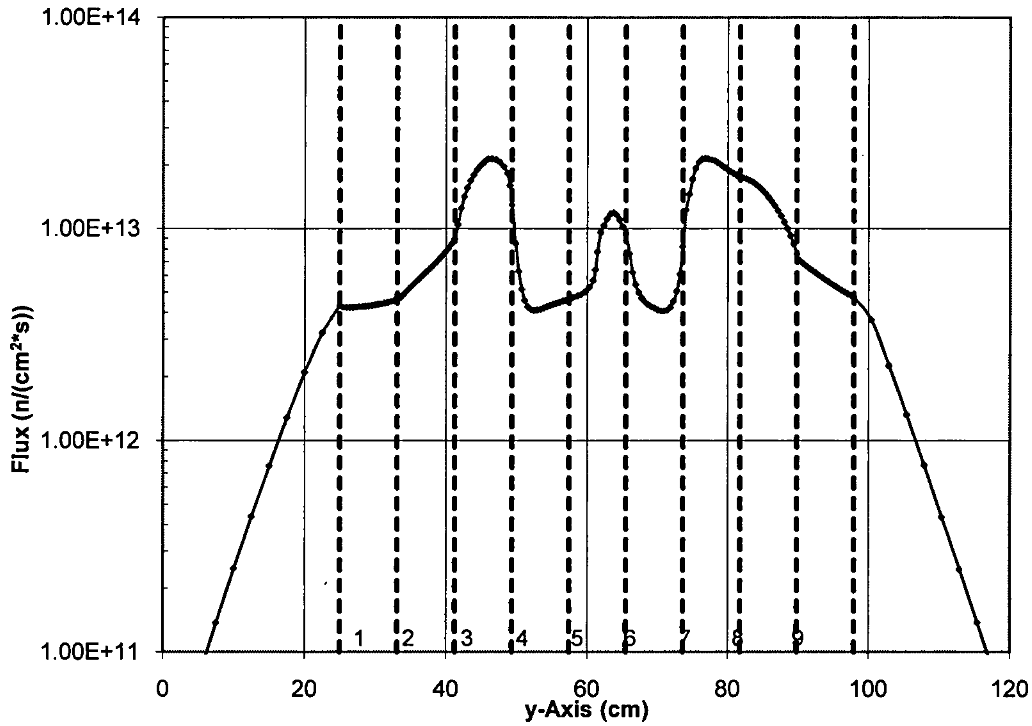


Figure 4-39: NSC Initial Cycle: Thermal Neutron Flux (E<04.42 eV) through the Transient Rod at Maximum Axial Power Peaking Position (76.7 cm) - BOL

The DIF3D code described above was used to calculate expected cross-sectional thermal neutron flux at the maximum axial power peaking position. The cross-sectional fluxes were calculated at 0.428 cm intervals within the core boundary and 2.470 cm intervals external to the core.

Table 4-31 provides the minimum and maximum cross-sectional thermal fluxes at four core positions utilized for NSC samples. These values are the maximum and minimum calculated fluxes within a 1.8 cm square centered at the sample center.

Table 4-31: Maximum and Minimum Calculated Flux in 1.8 cm Square Centered at Sample Positions

Flux $neutrons/(cm^2 \cdot sec)$	Core Position			
	A2	A4	A6	A8
Minimum Flux	4.03×10^{12}	8.18×10^{12}	8.45×10^{12}	3.20×10^{12}
Maximum Flux	5.72×10^{12}	1.10×10^{13}	1.12×10^{13}	4.50×10^{12}

4.8.2 Thermal Neutron Flux Values in LEU Core, Measured

Thermal and epithermal neutron flux was measured at the A4 and A6 core position at six core heights corresponding to typical NSC sample positions using bare gold and cadmium covered gold flux foils. The measurements were performed first with the four shim safety control rods banked (See Table 4-32). A second set of measurements was performed with the shim safety control rod skewed to tilt the core flux to the core A-row (See Table 4-33).

After irradiating the foils for two hours, the foil activity was measured using a HPGe (High Purity Germanium) detector. Using foil irradiation and decay time, and comparing the bare foils to the cadmium-covered foils, the NSC was able to determine the thermal neutron flux, epithermal flux, and the epithermal ratio. Measured thermal flux was then compared to the calculated thermal flux calculated by General Atomics for each of the four core positions at six sample positions.

With the shim safety control rods banked, measured thermal flux differed from calculated flux by 21 – 38 percent. With the shim safety control rods skewed, measured thermal flux differed from calculated flux by 25 - 48 percent.

A primary contributor to this difference can be attributed to flux foil sample placement. As demonstrated by Table 10, measured thermal flux at a sample position can differ from 20 to 50 percent by positioning the flux foil as little as 0.9 cm from the centerline sample position.

A second source of measurement error arises from the model used to calculate flux. The calculated thermal flux assumes no air void surrounding the foil samples, where in reality, the foil samples are surrounded by a 2.74 inch diameter air void. This can cause a significant difference in actual thermal flux due to reduced neutron attenuation and thermalization.

Since BOL, flux has also continuously trended upward and thermal neutron flux is well within the range of usability for the purposes of the NSC.

Table 4-32: A4 and A6 Axial Thermal Neutron Flux with Shim Safety Control Rods Banked

Height from Bottom of Fueled Portion (cm)	A4 Core Position			A6 Core Position		
	Calculated Thermal Flux (1)	Measured Thermal Flux (2)	% Difference	Calculated Thermal Flux (1)	Measured Thermal Flux (2)	% Difference
	$cm^{-2}s^{-1}$	$cm^{-2}s^{-1}$	%	$cm^{-2}s^{-1}$	$cm^{-2}s^{-1}$	%
9.38	9.52E+12	5.93E+12	-37.7%	9.63E+12	7.54E+12	-21.7%
13.13	9.70E+12	6.38E+12	-34.2%	9.90E+12	7.76E+12	-21.6%
16.88	9.49E+12	6.18E+12	-34.9%	9.74E+12	6.64E+12	-31.8%
20.63	8.93E+12	6.26E+12	-29.9%	9.13E+12	5.84E+12	-36.0%
24.38	8.04E+12	5.25E+12	-34.7%	8.32E+12	5.91E+12	-28.9%
28.13	7.04E+12	5.09E+12	-27.7%	7.30E+12	5.63E+12	-22.9%

Table 4-33: A4 and A6 Axial Thermal Neutron Flux with Shim Safety Control Rods Skewed

Height from Bottom of Fueled Portion (cm)	A4 Core Position			A6 Core Position		
	Calculated Thermal Flux (1)	Measured Thermal Flux (2)	% Difference	Calculated Thermal Flux (1)	Measured Thermal Flux (2)	% Difference
	$cm^{-2}s^{-1}$	$cm^{-2}s^{-1}$	%	$cm^{-2}s^{-1}$	$cm^{-2}s^{-1}$	%
9.38	1.05E+13	5.53E+12	-47.2%	1.08E+13	8.02E+12	-25.5%
13.13	1.09E+13	6.35E+12	-41.7%	1.12E+13	7.44E+12	-33.7%
16.88	1.09E+13	7.51E+12	-31.2%	1.12E+13	6.55E+12	-41.8%
20.63	1.05E+13	6.89E+12	-34.5%	1.09E+13	6.12E+12	-43.9%
24.38	9.80E+12	6.51E+12	-33.6%	1.01E+13	6.72E+12	-33.8%
28.13	8.66E+12	5.80E+12	-33.1%	9.11E+12	5.86E+12	-35.7%

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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 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 33 feet. The maximum heat exchanger tube pressure of approximately 22 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 5-2 shows the elevations of the water systems.

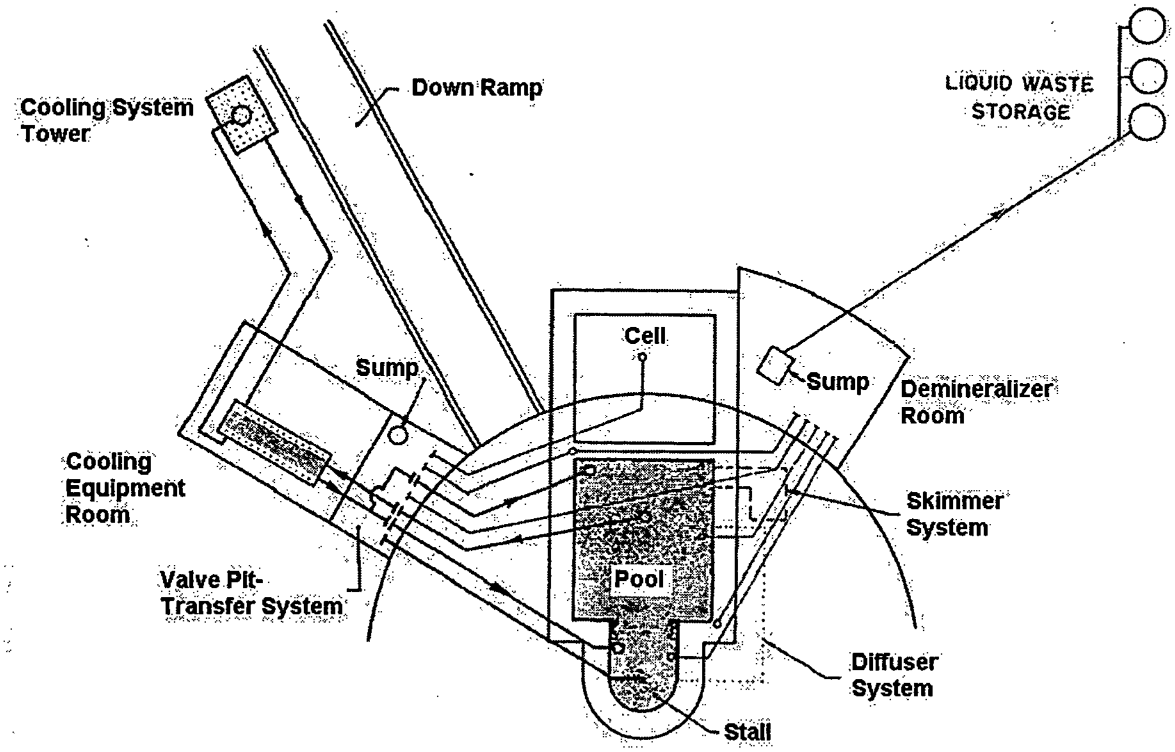

Figure 5-1: Pool Water Systems

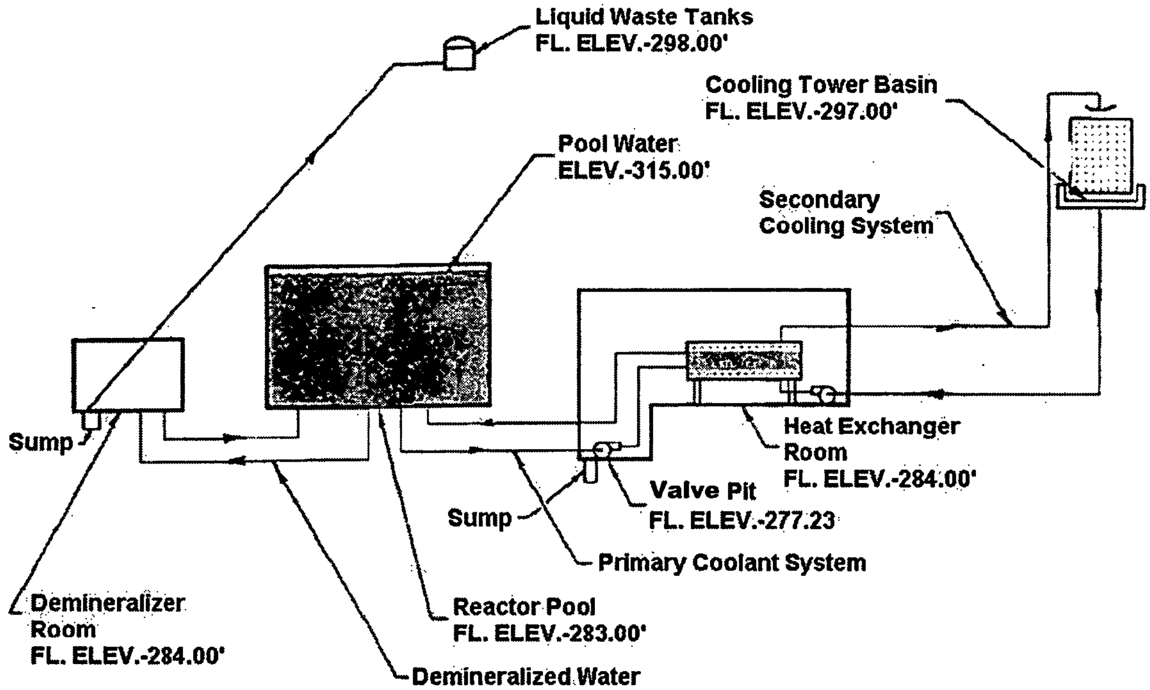

Figure 5-2: NSCR Pool Water Elevation

Two two-inch drain lines, one on the floor of each pool section, 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 ten-inch line beneath the liner at the center of the stall section routes pool liner leakage to the valve pit. The irradiation cell floor has a three-inch drain line. Figure 5-3 shows the piping penetrations in the reactor pool.

[redacted]

Figure 5-3: Reactor Pool Sections and Penetrations

5.2 Primary Coolant System

Three ten-inch water-cooling lines are located on the floor 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 2.0 MW with a 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.

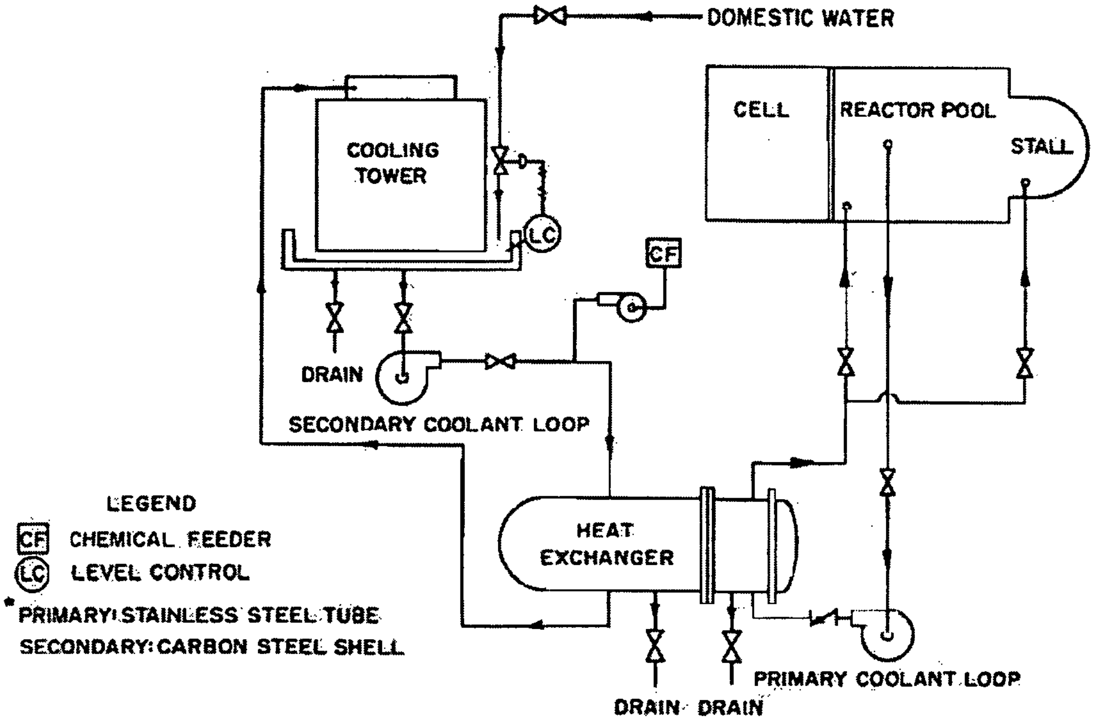

Figure 5-4: Reactor Pool Cooling System

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 a 78°F wet bulb air temperature.

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; 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. An auto/off/hand switch allow local operation and is 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 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:

- Maintain pool water purity,
- Provide a filtering mechanism for makeup water; and
- Provide a path for makeup water for filling the pool during both normal and emergency pool fills.

The demineralizer/recirculation system uses a regenerative mixed bed of ion exchanged resin in the demineralizer 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 microseimens per cm^3. Water quality is maintained by regeneration of the ion exchange resin.

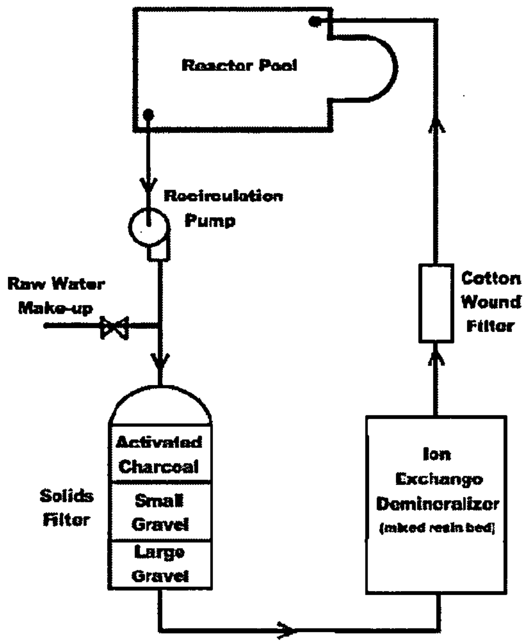

Figure 5-5: Recirculation/Demineralization System

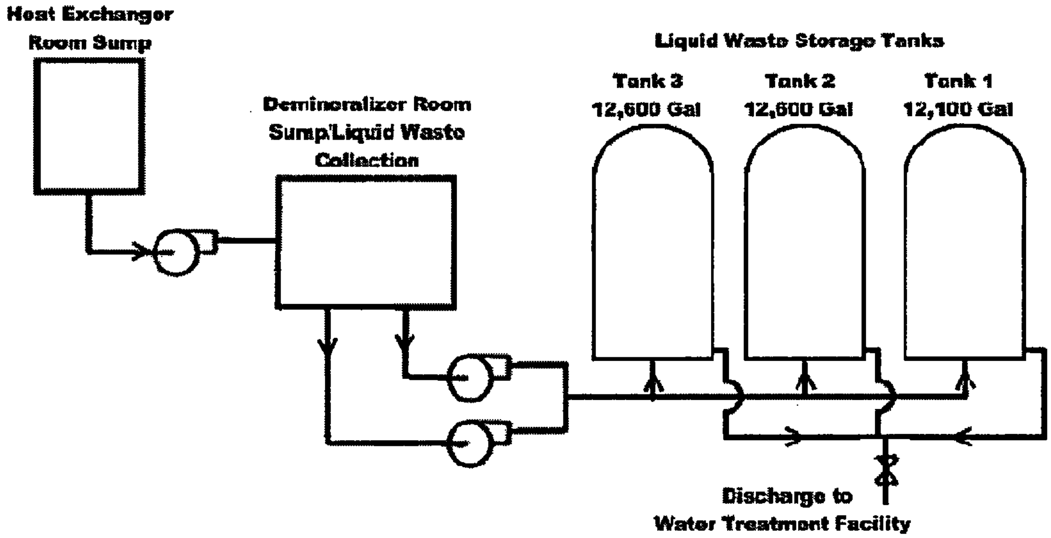

Figure 5-6: Liquid Waste Disposal System

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.

5.4.2 Skimmer System

The purpose of the skimmer system (Figure 5-7) 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.

5.4.3 Transfer System

The transfer system is located in the valve pit of the cooling equipment room. It consists of a stainless steel piping system and a 250 gpm pump, which interconnects the two pool sections, the irradiation cell, and the demineralizer room. The system is used to transfer water between pool sections for storage, transfer to the waste sump for disposal, or transfer to the demineralizer room for purification. An on/off pump switch is located at the pump and in the reactor control room for operation of the system. The demineralizer system and the transfer system are interconnected by a single three-inch crossover line for flexibility of operation. The transfer system is shown in Figure 5-8.

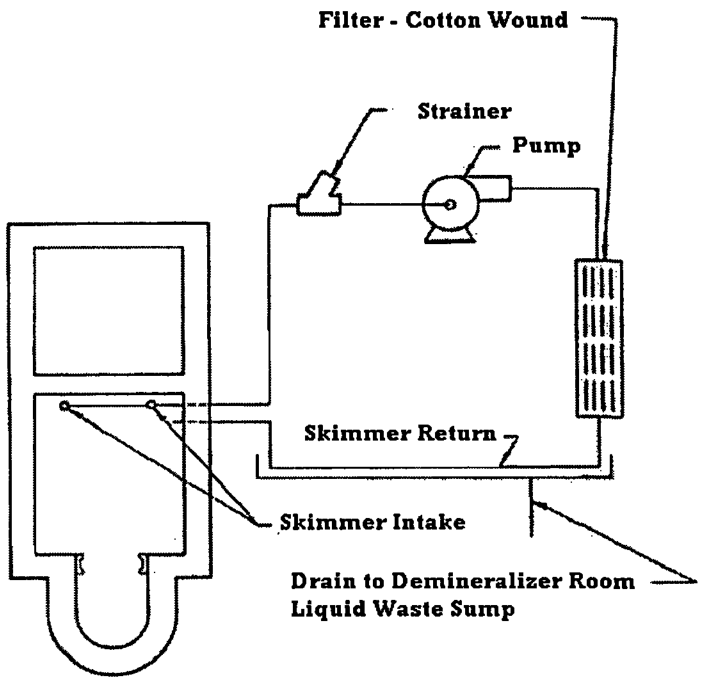

Figure 5-7: Skimmer System

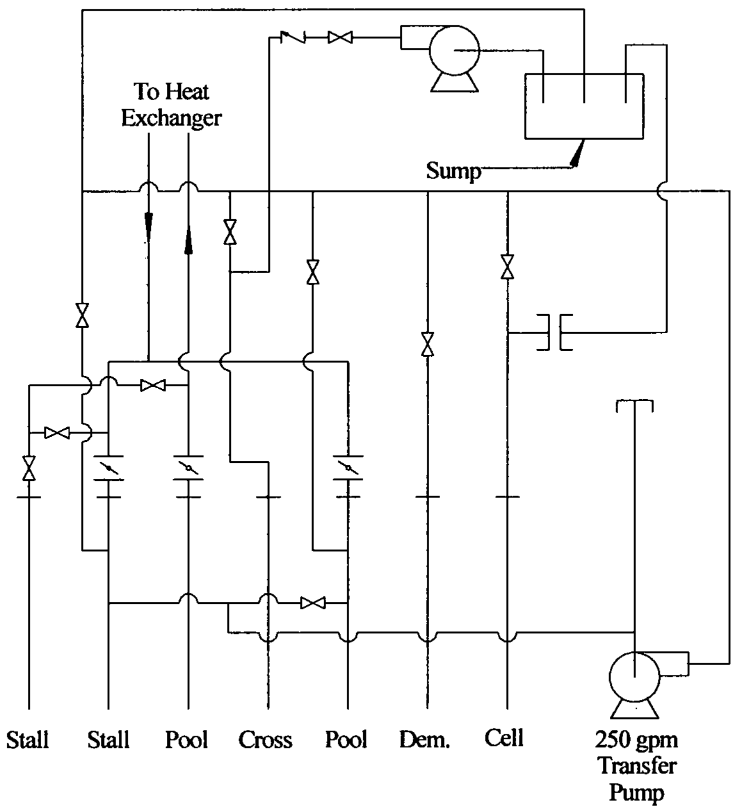

Figure 5-8: Transfer System

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 makeup.

5.6 Nitrogen-16 Control Diffuser 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. Two diffuser pumps and associated piping are located in the mechanical chase as shown in Figure 5-9. Only one pump is operated during reactor operations, with the second serving as an installed spare to allow for reactor operation during pump maintenance. Two outlets permit operation of the system when the reactor is in the pool or the 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.

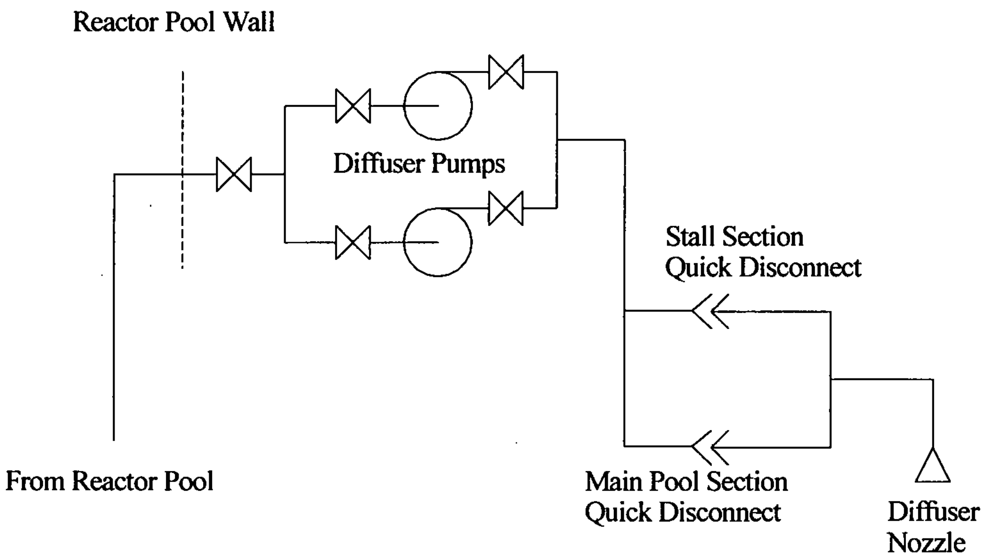

Figure 5-9: Diffuser System

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 13.5.1 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 70 feet in diameter and 70 feet high. Approximately 50 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 six degrees toward the center of the building and joined to a structural steel frame. The roof deck is a 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 level immediately below the upper research level 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. 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. The air within this tube flows through the existing exhaust system and facility air monitors (FAM) before release from the stack. This design allows for monitoring and controlled release of radioactive gases associated with operation of the pneumatic system.

Three 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 is less 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 zone of maximum negative pressure includes areas where contamination is most likely to occur, i.e., beam ports, thermal column, and through tubes.

Air-handling units provide and circulate fresh air in the building. All three units have controls on the control console in the reception room and the control room. These air handlers will shutdown simultaneously with the central exhaust fan if 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 inlets to all handling units, the fresh air bypass to the exhaust fan, and in the 85-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, as well as in the control 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 console is located in the reception room and in the control room for operation of the system. The air-handling units, exhaust-fan, and associated dampers can be operated from these consoles. The control room console is the primary operating station with the reception room console serving as an emergency station when building evacuation is required.

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 console in the reception room or control 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 1.0 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 in the bottom of the pool have manual closures to isolate the pool in the event of cooling system component failure.

7 INSTRUMENTATION AND CONTROL SYSTEM

7.1 Summary Description

The reactor operates in two modes:

1. Steady State Mode – steady power levels up to 1.0 MW, or
2. Pulse Mode – a rapid transient rod withdrawal (the technical specifications set the limit) causes a large power excursion.
3.

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, see Figure 7-1 for basic NSC Console Logic information.

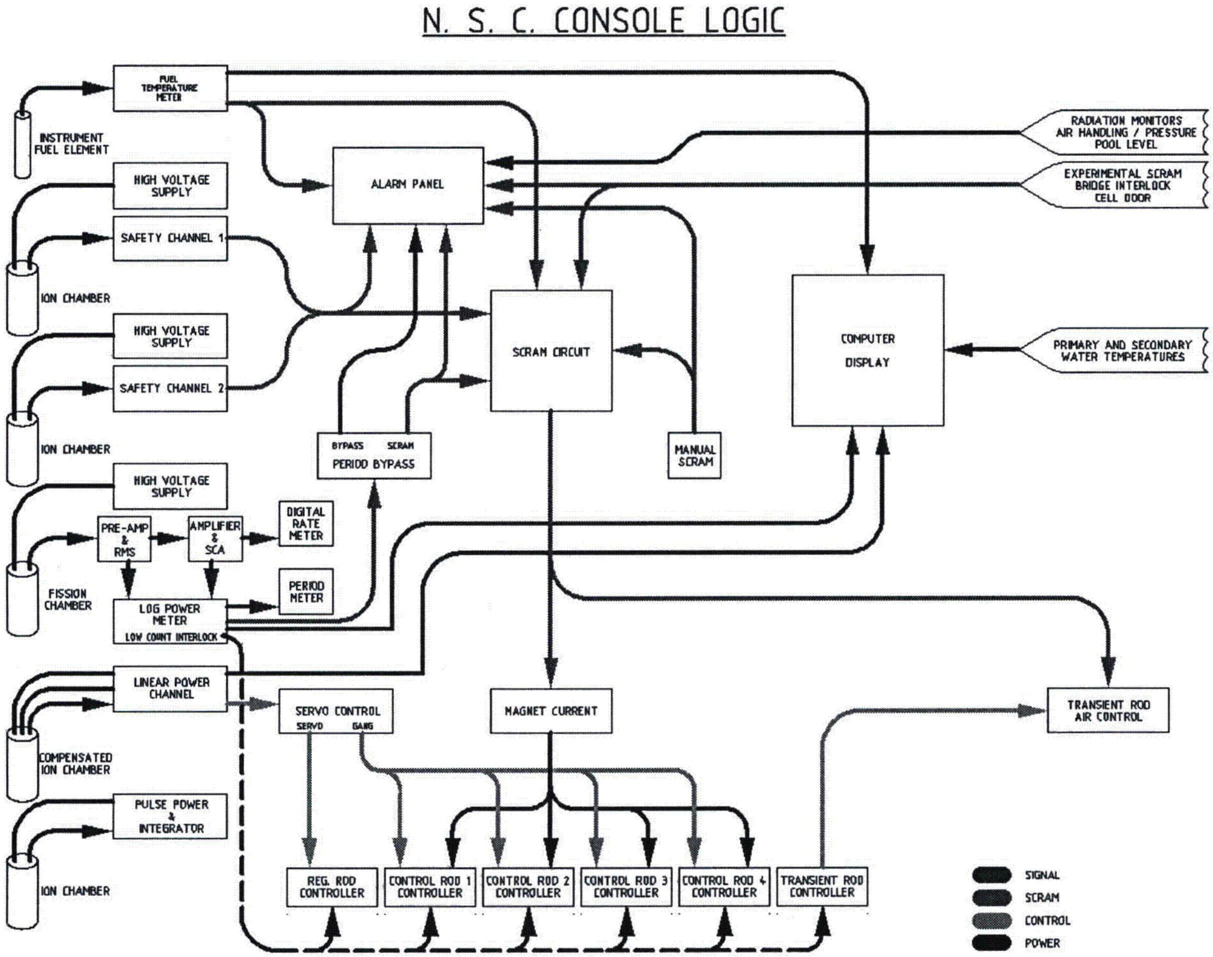

Figure 7-1: NSC Console Logic

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, the wide range linear drawer (with multiple scales) and the log drawer (with multiple instruments), provide indication over the entire operating range. Two other of these instruments, the safety drawers, provide indication of power beginning around 10 kW. Finally the fifth instrument, the pulse drawer, provides indication above 10 kW, and also 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 (in 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 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:

1. Information on the status of the reactor,
2. The means for insertion and withdrawal of control rods,
3. Automatic control of reactor power level,
4. 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 both the steady state and pulse modes, the reactor scrams when measured fuel temperature reaches the limiting safety system setting.

7.2.3 System Description

7.2.3.1 Log Power Channel

The log power channel consists of a fission chamber, preamp, amplifier, log meter, and digital neutron rate counter. The log power channel performs the following:

1. Provides reactor power indication over a of range of ten decades of reactor power,
2. Provides low count interlock which prevents withdrawing control rods without at least two counts per second, and

3. Provides period scram, which sends 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 RMS current to a logarithmic power indication. The instruments overlap to provide continuous indication. Although capable of providing indication over the entire range of steady state operation, the log drawer functions primarily as a start-up channel providing low power indication and providing necessary low power interlocks. At high power (>10 kW) it serves only as an operator aid, with the wide range linear channel serving as the primary power indication. Figure 7-2 shows a simplified diagram of the log power channel.

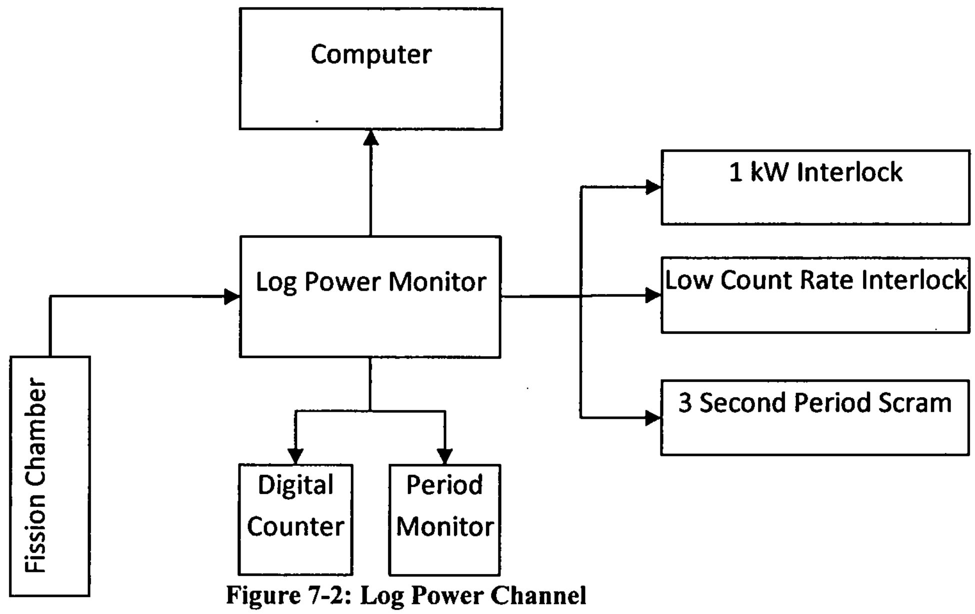

Figure 7-2: Log Power Channel

*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:

1. Steady State Mode
    a. Percent Power
2. Pulse Mode
    a. Percent Power
    b. Peak Power
    c. Energy (Mw-Sec)

*7.2.3.3 **Wide Range Linear Channel***

The linear power channel consists of a compensated ion chamber and a wide range linear drawer. The wide range linear drawer provides the following:

1. Power indication over the entire range of operating and shutdown levels

2. 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 radiation so that the system is sensitive and accurate[1].

7.2.3.4 Servo Control System

The servo controller automatically maintains power level by comparing the signal from the wide range linear 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. The 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 7-3 shows the linear power channel.

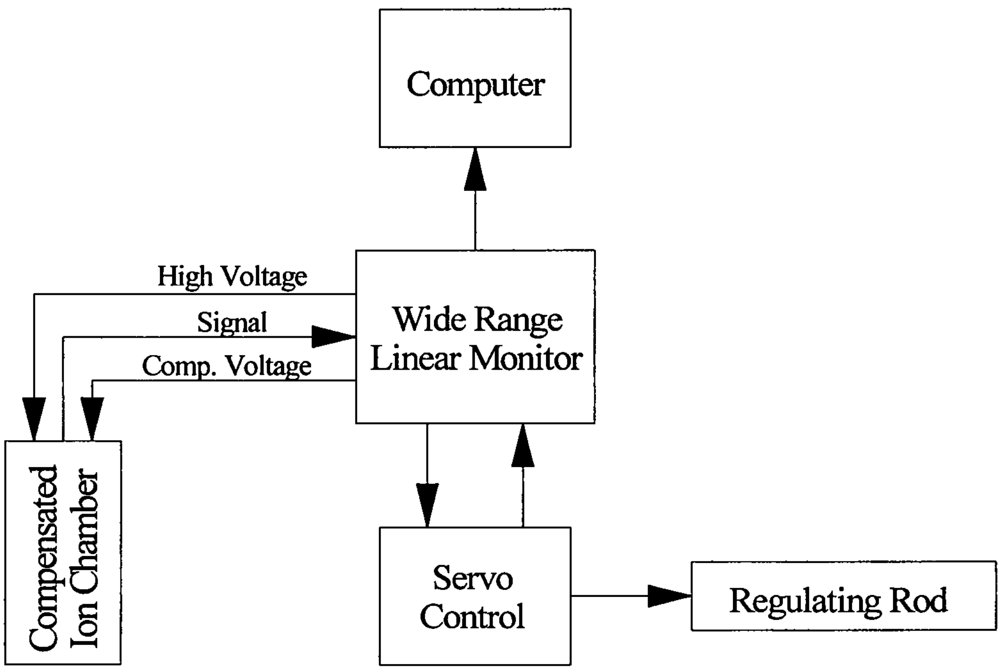

Figure 7-3: Wide Range Linear Channel

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.

There are two scrams associated with the safety drawer:

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

*7.2.3.6 **Scram System***

The scram system supplies current to the control rod magnets providing the mechanism for scramming the reactor. When the system receives a scram signal, it stops supplying current to the electromagnets, which hold the control rods in position. Without the magnets, the control rods fall to the fully inserted position with gravity providing the motive force.

The scram system also receives scrams that are not internal to the safety power drawer. Following are the scrams within the scram system:

1. High Power (125% Full Power)
2. Low Voltage (<150V)
3. Period (<3 sec)
4. Fuel Temperature (975°)
5. Manual (Console)
6. Bridge lock scram
7. Various experiment scrams, which allow experimenters to independently and locally scram the reactor. These are manual scram buttons located as follows:
   a. Beam Port Areas
   b. Irradiation Cell
   c. Reactor Bridge
   d. Pool Side
8. Interlocked scrams that ensure the reactor is shutdown when:
   a. BP- 4 cave door is open and the reactor is near the thermal column graphite coupler box
   b. Cell door is open and the reactor is within eight feet of the irradiation 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 indication is also available in the reception. Figure 7-4 shows a diagram of the fuel temperature channel.

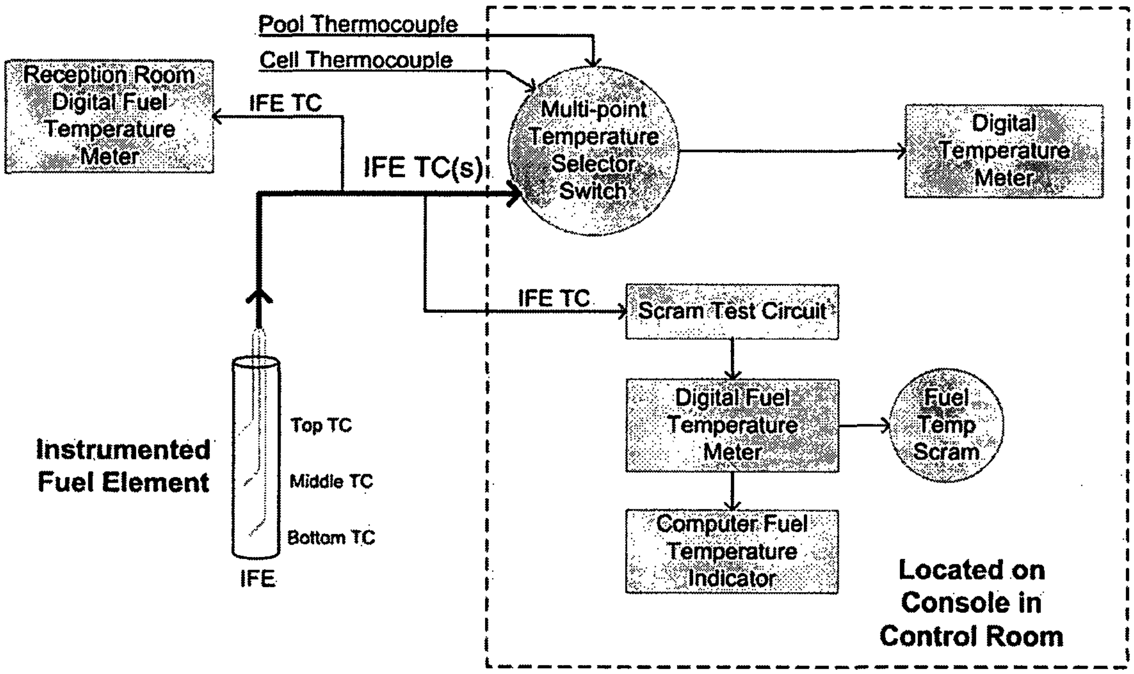

Figure 7-4: Fuel Temperature Channel

The fuel temperature channel is the limiting safety system with the limiting safety system setting (LSSS) at 975°F. The fuel temperature instrument performs the following functions:

1. Measures fuel temperature and scrams the reactor when the thermocouple temperature reaches 975°F, and
2. Captures peak measured 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 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 IFE can be in any of eight locations in the core. Three chromel-alumel thermocouples are embedded in the IFE. These thermocouples are located at the midpoint, one inch above, and below vertical center. They are positioned 0.3 inches from the radial center (Figure 4-9).

*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 power from oscillating following a pulse.

7.2.4 System Performance Analysis

The instrumentation and control system designs have been in routine operation for over 40 years. Replacing nearly all original measuring instruments with solid-state electronics 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 fuel temperature channel prevents exceeding the limiting safety system setting for fuel temperature. Other scram conditions including high power scram, loss of AC power, loss of safety detector voltage, and manual scram capability ensure the safe operation of the NSCR.

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 and 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:

1. Provides method for controlled addition of reactivity
2. Provides scram capability
3. Holds control rods and control rod bundles in position
4. Provide indication of control rod position
5. Provides rod withdrawal interlocks

Rod control modules in the reactor console:

1. Provide signal for individual rod motion
2. Provide logic interlocks
3. Provide rod and carriage position indication.

Each control rod drive mechanism 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 that the 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 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, 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, as well as a power supply for rod motion.

Figure 7-5 shows the control rod magnet, 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.

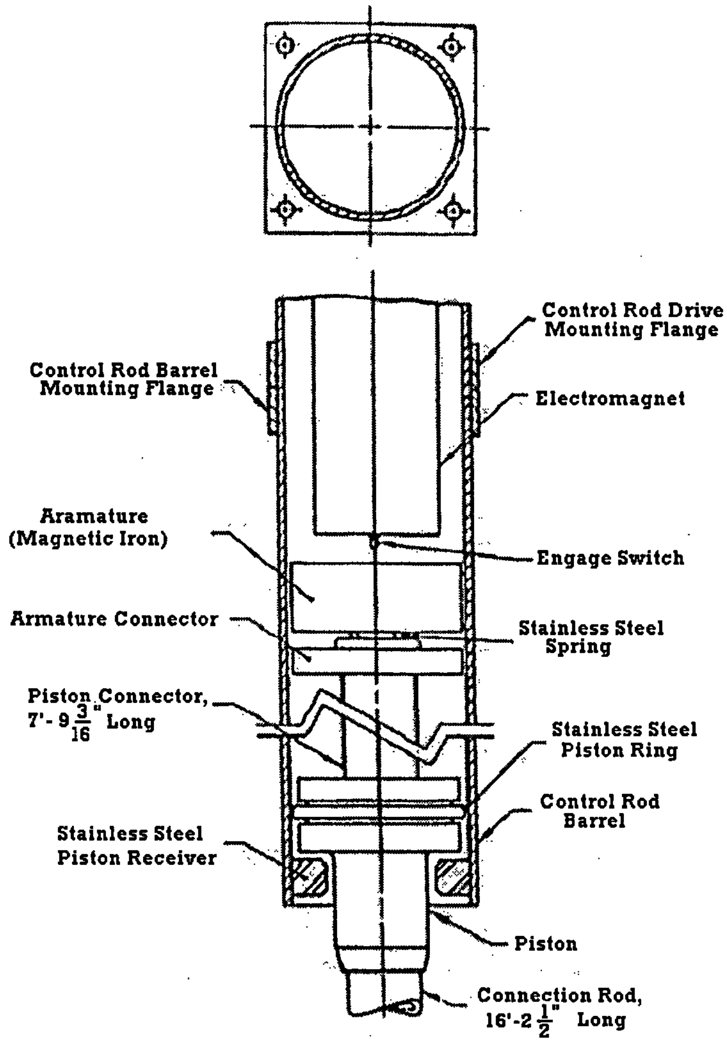

Figure 7-5: Shim Safety Armature and Dampening Device Assembly

The shim safety control rods have eight optional control rod positions. The offset assembly functions similarly to the bolt action of a rifle. Spacers that 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 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 7-6). 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 7-7 shows the installation of a shim safety control rod.

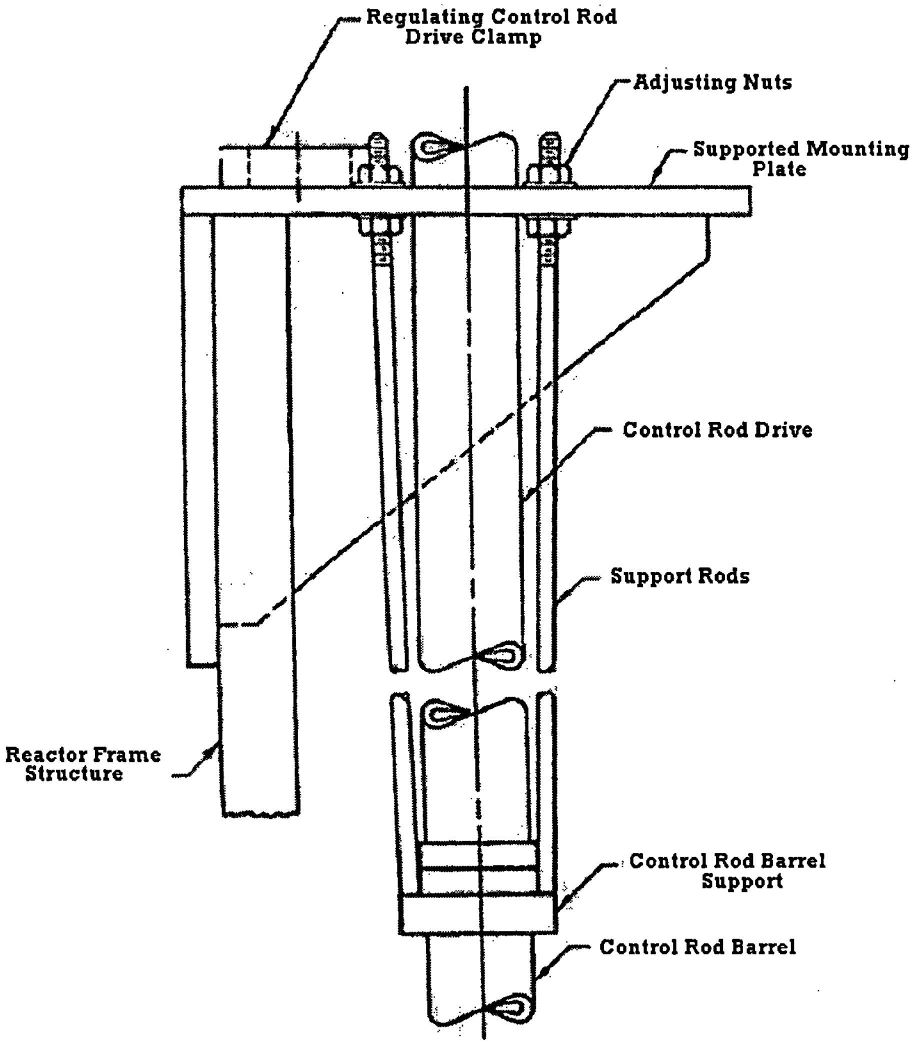

Figure 7-6: Control Rod Assembly Support Mechanism

[redacted]

Figure 7-7: Control Rod (Shim Safety) 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:

1. Provides digital indication of carriage height
2. Provides Rod In/Rod Out signal to CRDM
3. Provides indication of the following:
    a. Rod Engaged
    b. Rod Down
    c. Rod Jammed
    d. Carriage Down
    e. Carriage Up
4. Resets the rod position indication to 1.0% when:
    a. The engaged switch changes from disengaged to engaged
       -while-
    b. Carriage is driving in
       -and-
    c. The rod down switch is pressed
5. Provides the following logic interlocks:
    a. Prevents rod insertion for jammed rod
    b. Prevents rod insertion if:
        i. Carriage height is 0.0%
           -and-
        ii. Rod is engaged and down
    c. Prevents rod withdrawal if rod height is 100.0%
    d. Prevents rod withdrawal if the gang switch is in the gang down position
6. Provides individual rod withdrawal and insertion capability

Interlocks associated with the shim safety control rods are as follows:

1. Rod Jammed: Prevents driving carriage down when lead screw presses the jam switch
2. Rod Down: Prevent driving carriage down when:
    a. Carriage height indication at 0.0%
       -and-
    b. Rod engaged
3. Rod Out Interlock: Prevents rod withdrawal if rod height is 100.0%
4. 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.
5. Shim Safety Pulse Interlock: Prevents withdrawing control rods in the pulse mode
6. Low Count Interlock: Prevents withdrawing control rods with < 4mW on the log power channel

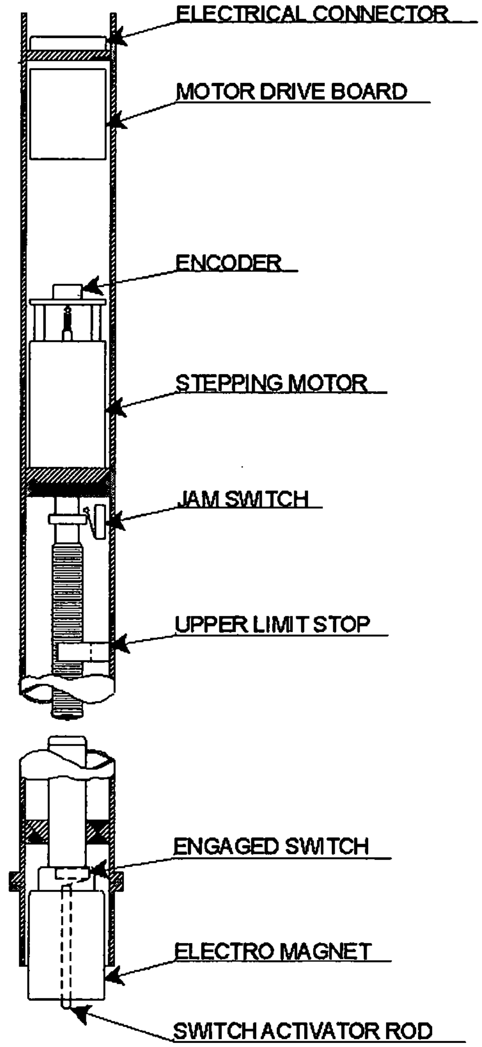

Figure 7-8: Control Rod Drive Mechanism for Shim Safety Control Rods

The CRDMs (Figure 7-8) are electromechanical assemblies, which move the 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:

1. Engaged Switch
    a. Push button on the bottom face of the electromagnet
    b. Lights engaged light on rod control module
    c. Provides input to rod position reset circuit to reset indication at 1.0% when the control rod extension presses the engaged switch.

    d. Provides input to rod down Interlock (prevents further insertion if rod at 0.0% and rod engaged)
2. Rod Down Switch
    a. Magnetic reed switch external to the CRDM and adjacent to the armature on top of the control rod extension assembly
    b. Lights rod down light when control rod is <5% withdrawn
    c. Provides input to rod position reset circuit
3. Rod Jammed
    a. A micro-switch pressed when the lead screw drives in without the carriage lowering
    b. Lights rod jammed light
    c. Provides input to rod jammed interlock, which prevents rod insertion if rod is jammed

7.3.1.1 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, and a 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.

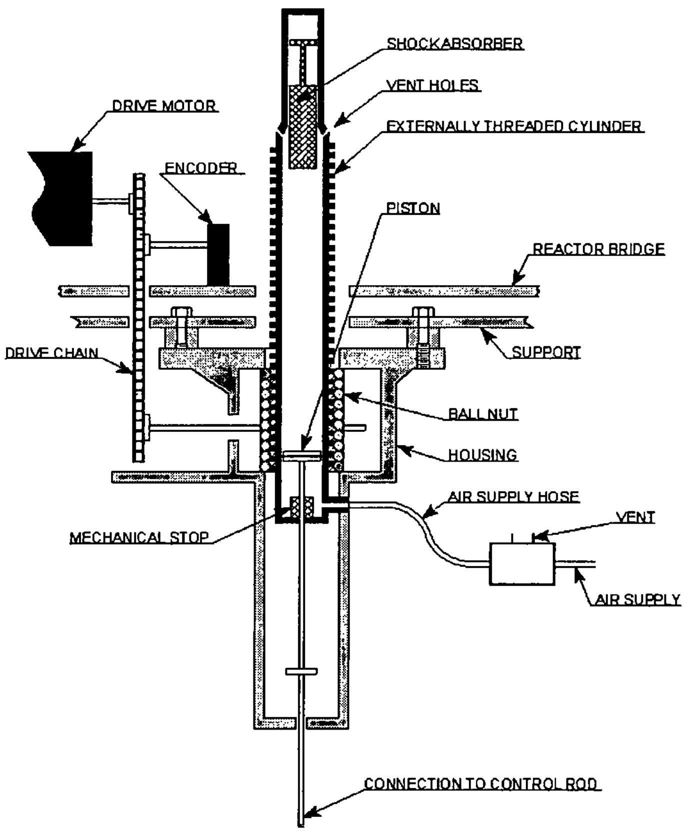

Figure 7-9: Transient Rod Drive

Figure 7-9 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.

The following describes the operation of the transient rod for both modes of operation:

1. Steady State:
    a. High-pressure air acting on a piston holds the transient rod against its rod drive carriage (specifically the shock absorber)
2. Pulse:
    a. High-pressure air pushes the transient rod rapidly to the carriage position.
    b. 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:

1. Provides digital indication of carriage height
2. Provides Rod In/Rod Out signal to CRDM
3. Provides indication of the following:
    a. Air Applied
    b. Rod Down

   c. Carriage Down
   e. Carriage Up
   f. TR Fire Ready
4. Resets the rod position indication to 1.0% when:
   a. The Carriage down switch makes
      -while-
   b. Carriage is driving in
      -and-
   c. The rod down switch is made
5. Provides the following logic interlocks
   a. Prevents rod insertion if:
      i. Carriage height is 0.0
         -and-
      ii. Carriage down switch is made
   b. Prevents rod withdrawal if rod height is 100.0%
6. 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:

1. Rod Down: Prevent driving carriage down when:
   a. Carriage height indication at 0.0%
      -and-
   b. Rod carriage is down
2. Rod out interlock: Prevents rod withdrawal if rod height is 100.0%
3. Air applied interlock: Allows applying air when:
   a. Mode selector switch in pulse
      -and-
      i. Power is less than 1kW
      -or-
   b. Mode selector switch is in steady state
      -and-
      i. Carriage is down
      -or-
      ii. 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.)
4. TR Withdrawal: Prevents withdrawing control rods in the pulse mode
5. Low count interlock: Prevents withdrawing control rods with < 2cps on the log power channel

6. Pulse Stop Interlock: Prevents applying air to the transient rod when:
    a. Mode selector switch is in pulse
       -and-
    b. Mechanical pulse-stop is not installed

The CRDM (Figure 7-9) is an electromechanical assembly that moves the transient rod and holds 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:

1. Carriage Down Switch:
    a. Micro switch activated by carriage
    b. Provides input to rod position reset circuit to reset indication at 1.0% when
        i. The carriage presses the carriage down switch
           -while-
        ii. Rod is driving in
           -and-
        iii. Rod is down
    c. Provides input to rod down interlock (prevents further insertion if rod at 0.0% and carriage is down)
2. Rod Down Switch
    a. Micro switch activated by the piston rod
    b. Lights rod down light when control rod is <5% withdrawn
    c. Provides input to rod position reset circuit

7.3.1.2 ***Regulating Rod Control***

The rod control system for the regulating control rod allows the operators to control the rod manually or automatically via the servo controller. The regulating rod has a rod control module, control rod drive mechanism, control rod barrel and hold-down tube. In addition, the regulating 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 7-8 except 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 regulating rod physically attaches to the connector rod, the CRDM must be disassembled.

The regulating rod CRDM connects to a rod control module at the reactor console. Push buttons permit operation of the rod drive. Travel speed for the regulating 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 (0.0%). Rod height and carriage height are the same.

The rod control module for the regulating rod performs the following functions:

1. Provides digital indication of carriage height
2. Provides Rod In/Rod Out signal to CRDM
3. Provides indication of the following:
    a. Carriage <20%
    b. Carriage >80%
    c. Rod Jammed
    d. Carriage Down
    e. Carriage Up
4. Resets the rod position indication to 1.0% when:
    a. Carriage is driving in
       -and-
    b. The rod down switch is made
5. Provides the following logic interlocks
    a. Prevents rod insertion for jammed rod
    b. Prevents rod insertion if:
        i. Carriage height is 0.0%
           -and-
        ii. Rod down switch is made
    c. Prevents rod withdrawal if rod height is 100.0%
6. Provides signal for shimming required alarm: Alarms when reg rod is <20% for >80% fully withdrawn

Interlocks associated with the regulating rod are as follows:

7. Rod Jammed: Prevents driving carriage down when lead screw presses the jam switch
8. Rod Down: Prevent driving carriage down when:
    a. Carriage height indication at 0.0%
       -and-
    b. Rod down switch is made
9. Rod Out Interlock: Prevents rod withdrawal if rod height is 100.0%
10. Shim Safety Pulse Interlock: Prevents withdrawing control rods in the pulse mode
11. Low Count Interlock: Prevents withdrawing control rods with < 4mW on the log power channel

The CRDMs for the regulating rod operates exactly like the shim safety control rod drive mechanisms except that there is no electromagnet.

The regulating rod CRDM has a jam and rod down switch. These physical switches perform the following and have the following physical descriptions:

1. Rod Down Switch
    a. Micro switch activated by carriage
    b. Provides input to rod position reset circuit
2. Rod Jammed
    a. A micro-switch pressed when the lead screw drives in without the carriage lowering
    b. Lights rod jammed light
    c. Provides input to rod jammed interlock, which prevents rod insertion if rod is jammed

7.3.1.3 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:

1. Steady State:
    a. Prevents applying transient rod air unless the following conditions are met:
        i. Transient rod carriage down

           -or-
        ii. 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.)
2. Pulse:
    a. Allows transient rod air applied with pulse stop installed
    b. Activates Preset Timer: Preset timer scrams the transient rod less than 15 seconds after applying air
    c. Prevents control rod withdrawal
    d. Disables safety drawer amplifiers
    e. 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.

Table 7-1: Minimum Reactor Safety Channels

Safety Channel	Number Operable	Function	Effective Mode	
			SS	Pulse
Fuel Element Temperature	1	Scram at LSSS	X	X
Safety Power	2	Scram at 125%	X	
	2	Scram on low detector power supply voltage (< 150V)	X	
Console Scram Button	1	Scram	X	X
Preset Timer	1	Transient Rod Scram < 15 seconds after pulse		X
Log Power	1	Prevent rod withdrawal with reactor power < $4x10^{-3}$W	X	
Log Power		Prevent Pulsing above 1 kW		X
Transient Rod Air Apply	1	Prevent application of air unless rod fully inserted	X	
Shim Safety & Regulating Rod Pulse Interlock	1	Prevent withdrawal of rods in Pulse Mode		X
Pulse Stop Interlock	1	Prevent application of air unless pulse stop is installed		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 is for operation at power levels up to 1000 kW (thermal). Pulse mode is for the condition resulting from the rapid withdrawal of the transient rod, introducing 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 chamber. 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

System	Control	Indication	Record
Reactor Safety Systems			
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 and reactor Conditional Alarms		X	
Water Systems			
Pool Water Cooling System	X	X	X
Pool Recirculation System	X	X	X
Pool Skimmer System	X	X	X
Diffuser System	X	X	X
Transfer System	X	X	X
Secondary Treatment System	X	X	X
Personnel Control and Radiation Protection			
Area Radiation Monitors		X	
Facility Air Monitors		X	X
Air Handling System Shutdown	X	X	
System	**Control**	**Indication**	**Record**
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

Alarm
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)

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 ARMs are located near the beam port areas, demineralizer room and other potentially radioactive areas.

The ARMs 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 FAM 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 monitors by FAM channel including their functions and sample points.

1. FAM Channel 1 (FAM-1) – Stack Particulate
    a. Monitors for radioactive particles in the air entering the exhaust stack
    b. Automatically shuts down the air handling system
2. FAM Channel 2 (FAM-2) – Fission Product
    a. Monitors for radioactive particulate above the reactor core
    b. Automatically shuts down air handling system
3. FAM Channel 3(FAM-3) – Stack Gas
    a. Monitors for 41Ar entering the exhaust stack
4. FAM Channel 4 (FAM-4) – Building Particulate
    a. Monitors for radioactive particles in the confinement building
5. FAM Channel 5 (FAM-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. FAM Channel 6 (FAM-6) – Building Gas
    a. Monitors for 41Ar in the confinement building
    b.

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.

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[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 in front of the Gamma Shack on the NSC site.

8.1.1 480 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.

8.1.2 120/208 VAC Electrical Power

Various distribution panels receive 450 VAC for loads in the facility.

8.1.2.1 Reactor Building Panels, Labeled RB, are located the confinement building

- RB 'A' and RB 'B' are located in the electrical shop.
- RB 'C' and RB 'E' are on the chase level.
- RB 'B' supplies power to RB 'D', which is in the control, room on the wall behind the main control panel.

8.1.2.2 Laboratory Building Panels, Labeled LB, are located in the laboratory building

- LB 'A' and LB 'B', located in lab seven of the laboratory building, supply loads for the laboratory area.

8.2 Emergency Electrical Power System

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.

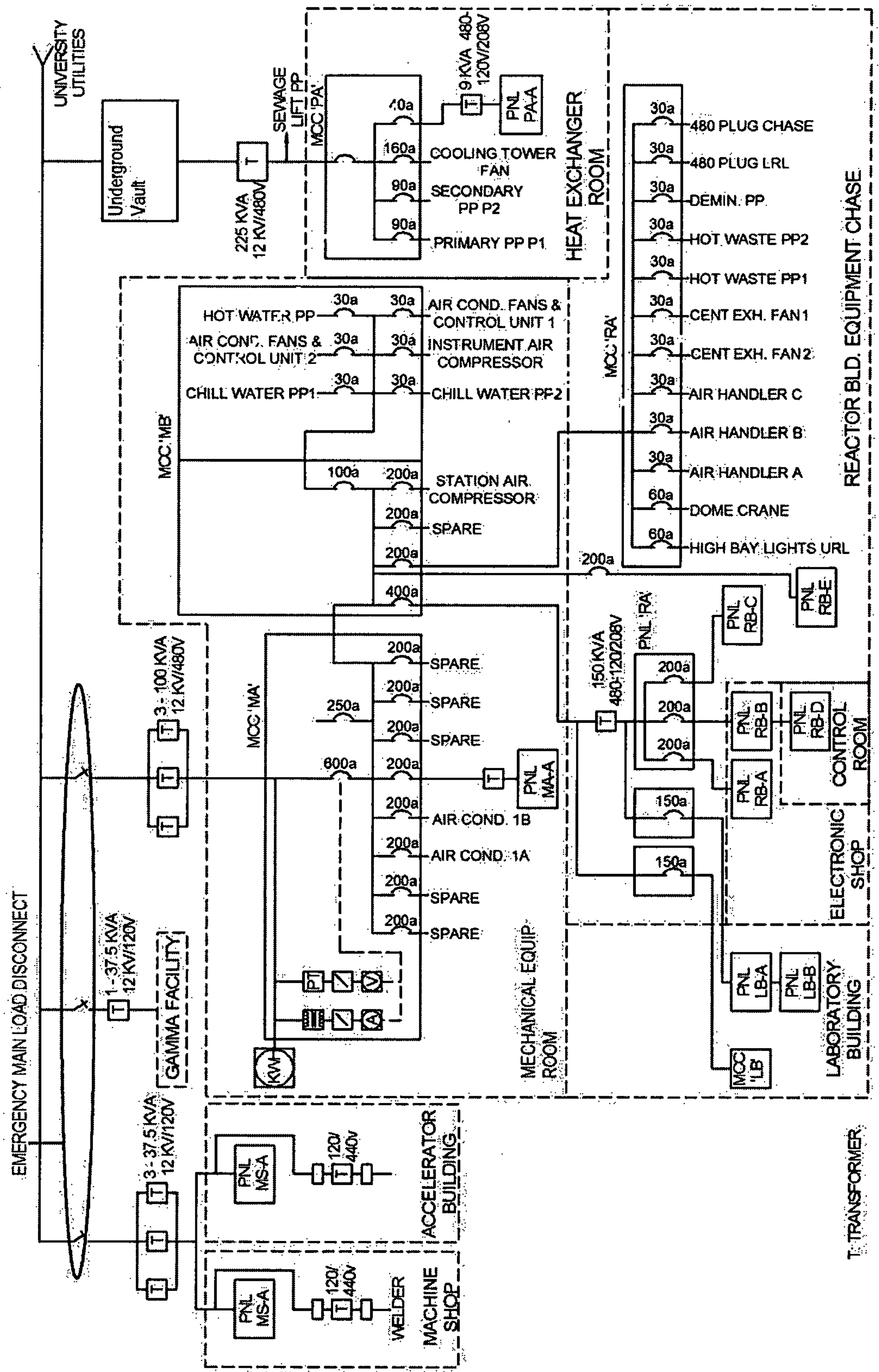

Figure 8-1: NSCR Electrical Distribution

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning (HVAC) 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 Unit

Three 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. The 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, which 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
    b. Restrooms
    c. Electronics shop
2. Air Handler B
    a. Lower research level
3. Air Handler C
    a. Control room

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.

Control stations for the HVAC system are located in the control room as well as the reception room. An interlock prevents running units A, B, or C 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:

1. The FAMs generate an emergency shutdown signal
2. The air handling shutdown button is pressed on the reactor console
3. The high temperature sensor above the emergency filter bank exceeds its set point
4. 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. An emergency 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. Placing the filter bank on service can be performed from the control room or the reception room HVAC computers.

9.1.4 Emergency Operation

The HVAC computers in the control room or the reception room provide 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 two general areas at the NSC. These areas are the fuel storage vault and the reactor pool handling and storage areas. Un-irradiated fuel can be temporarily stored in approved shipping containers, used by the fuel manufacturers for shipment to 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 at 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 Vault

The fuel vault, [redacted], provides a storage location for un-irradiated fuel. Elements stored in this room are under lock and key and intrusion security. Fuel bundles (four elements maximum per bundle) and individual elements can be stored in cadmium-lined aluminum tubes secured to an aluminum frame mounted to the concrete walls of the fuel storage room. Figure 9-1 shows the dry storage for fresh, un-irradiated fuel. Each aluminum tube is lined with polyethylene and cadmium. The cadmium sleeve provides additional insurance should some erbium be missing from one or more fuel rods – a highly unlikely circumstance. Thirty-nine storage tubes are available for four-rod clusters and individual un-mounted fuel rods.

[redacted]

Figure 9-1: Fuel Storage Room Rack

The model approach to fuel storage is very conservative, with the aim to establish upper bounds for criticality. For the dry, fresh fuel storage, the computer model included the aluminum tubes with a four-rod cluster inside. For conservatism, the cadmium sleeve and polyethylene sleeve were not modeled. A concrete wall was added on five sides to simulate the storage room. The computer program was run twice, once with air in the storage room, and once with the storage room flooded.

1. Storage of LEU 30/20 in air: $k_{eff} = 0.12994 \pm 0.00024$
2. Storage of LEU 30/20 flooded: $k_{eff} = 0.45526 \pm 0.00043$

When unirradiated fuel is stored in the fuel vault, a monitoring system capable of detecting a criticality that generates radiation levels of 300 Rem/hour one foot from the fuel is monitored locally and in the control room. The monitor alarm set point is adjustable and is maintained between 5 and 20 mrem per hour in accordance with 10 CFR 70.24 (a)(2). Should the alarm set point be reached, visual and audible alarms are activated, both locally and in the control room.

In conclusion, the storage of fresh LEU 30/20 fuel in the fuel storage vault maintains the fuel in a safe condition with regard to criticality, even in the event of fuel vault flooding. In addition, the storage vault's anticipated use is only during the infrequent changes of reactor core.

9.2.2.2 Reactor Pool Storage Areas

Figure 9-2 illustrates the in-pool storage facility. This consists of one large four by six storage rack free standing on the pool floor and four wall mounted racks, in each of which twelve (12) individual fuel rods and six (6) four-rod clusters can be stored.

[redacted]

The model approach to in-pool fuel storage has been equally conservative, with the aim to establish upper bounds for criticality. For this four by six stand-alone floor storage facility, only four-rod clusters were modeled. All aluminum structure of the floor rack was omitted. In two computer runs for this structure, one used full water reflector on all sides; in a second run, concrete was placed on one long side and on the bottom. As expected, the results for LEU 30/20 fuel gave very small values of k_{eff}, far below the limit of 0.8.

1. Storage of LEU in 4x6, array, water reflected: $k_{eff} = 0.46428 \pm 0.00045$
2. Storage of LEU in 4x6, water reflected plus 2 sides of concrete: $k_{eff} = 0.46420 \pm 0.00045$

The wall mounted storage racks were also modeled simply and conservatively. All material in the racks except the fuel elements was ignored. Using the well-known rule that one (1) foot (305 mm) of water adequately isolates one set of fissile material from another, the fourth wall rack is considered by itself. All four wall mounted rack storage are completely separate from the four by six storage assembly on the reactor pool floor. The k_{eff} is calculated for the group of three wall racks. The fourth, single rack is not evaluated because its k_{eff} is obviously smaller than that for the set of three.

The computed keff values are the following:

1. Storage of LEU in pool wall storage, full water reflected $k_{eff} = 0.47105 \pm 0.00044$
2. Storage of LEU in pool wall storage, two concrete sides plus water: $k_{eff} = 0.47110 \pm 0.00043$

In conclusion, the storage of fresh or irradiated LEU 30/20 fuel in reactor pool storage facilities maintains the fuel in a safe condition with regard to criticality.

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 for 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 non-fuel element. The difference between these two changes over the life of the element and provides for elongation determination.

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 24 hours a day. Fire department personnel receive training in radiological hazards and NSC site familiarization.

Smoke detectors placed at strategic locations throughout the confinement building and laboratory building sound a local audible alarm as well a visual an audible alarm in the control room. When an alarm is actuated, an alarm at the University Communication Center is automatically triggered prompting the Communication Center to contact the College Station Fire Department.

Should fire-fighting efforts be required, the building is designed such that water from all levels drains to the lower research level. From the lower research level, water eventually drains to the heat exchanger room sump or the demineralizer room sump. From there, it is automatically pumped to the liquid waste collection tanks for filtration, sampling and discharge.

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 Communication Center at Texas A&M University. The radios are battery-operated and spare charged batteries are kept on site, so that in the event of a loss of power to the facility, communications are maintained using these radios.

Finally, the Communications Center maintains an emergency recall roster listing telephone contact information (typically home, cellular, or beeper) 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 Experimental Facilities

10.1.1 Beam Ports

Five permanent beam ports of Type 304 stainless steel are cast into the pool wall at the lower research level. Beam Port (BP)-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 BP-6, 7 and 8 in Figure 10-2. Figure 10-2 shows the stall section of the pool with an unmodified thermal column. Normally, BP-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.

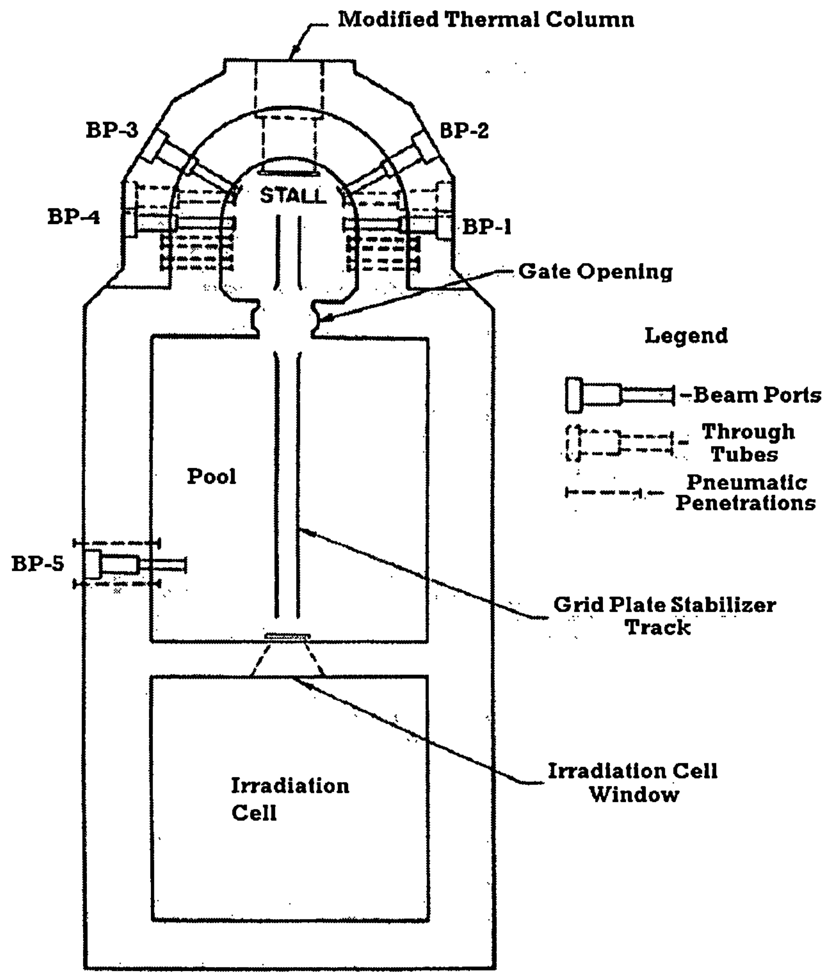

Figure 10-1: Pool Experiment Facilities

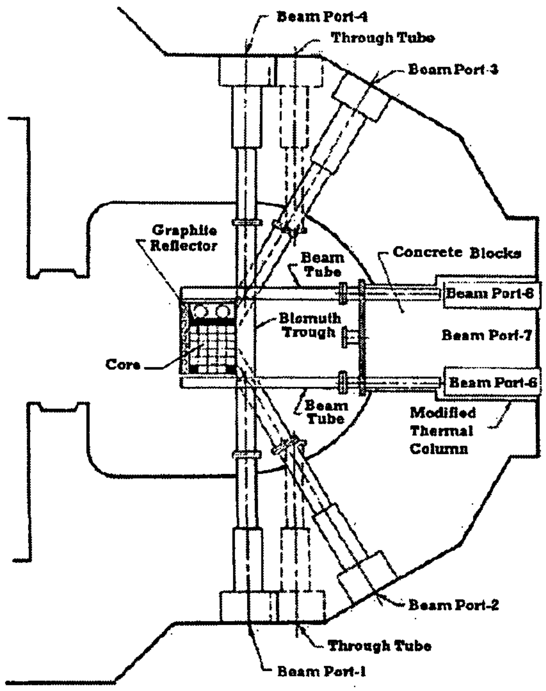

Figure 10-2: Stall Beam Port Installations with Dismuth Trough

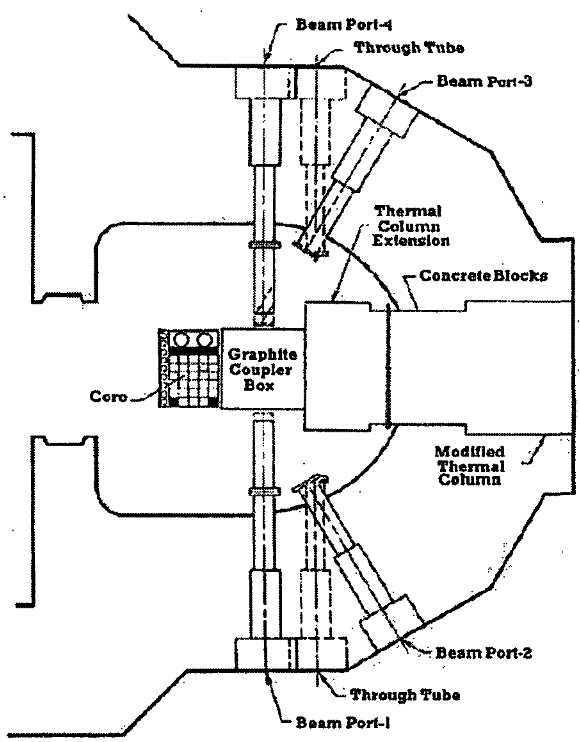

Figure 10-3: Stall Beam Port Installations with Graphite Coupler Box and Thermal Column Extension

The beam ports are type 304 stainless steel in sections of 6, 10, and 19 inches in diameter. These sections are 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 BP-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 BP-1 and 4 and the graphite coupler box removes water from between the coupler box and the beam port. BP-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.

10.1.2 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 FAM 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.

10.1.3 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 the ability to manipulate the length of irradiation, but the final pneumatic permit is granted by the control room operator. The experiment receivers provide a means for the pneumatic systems operator 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 inject the sample into the core, and then 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 an experiment receiver in multiple 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.

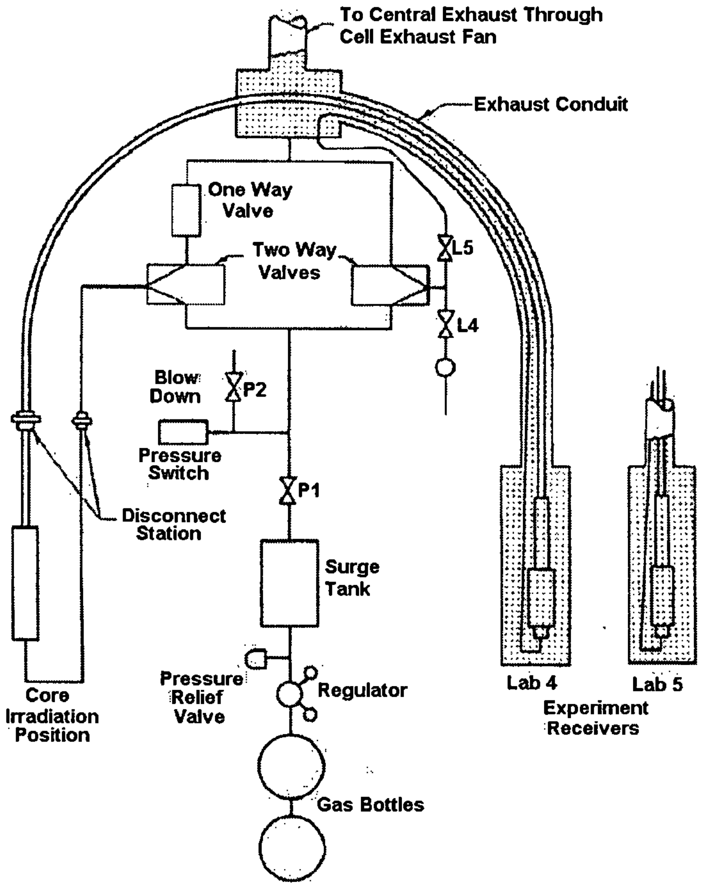

Figure 10-4: Pneumatic System

10.1.4 Irradiation Cell

The irradiation cell is located at the west end of the reactor pool. This cell is approximately 18 feet wide by 16 feet long by 10 feet high. The frame for lower irradiation cell's 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, provides decking for the concrete blocks, which are two by two by four feet. These blocks are stacked four feet high with a five by five foot opening, which is positioned directly over the cell window. A motor driven concrete shield covers the opening to the lower irradiation cell (Figure 10-5). The lower cell's concrete roof provides the floor for the upper irradiation cell.

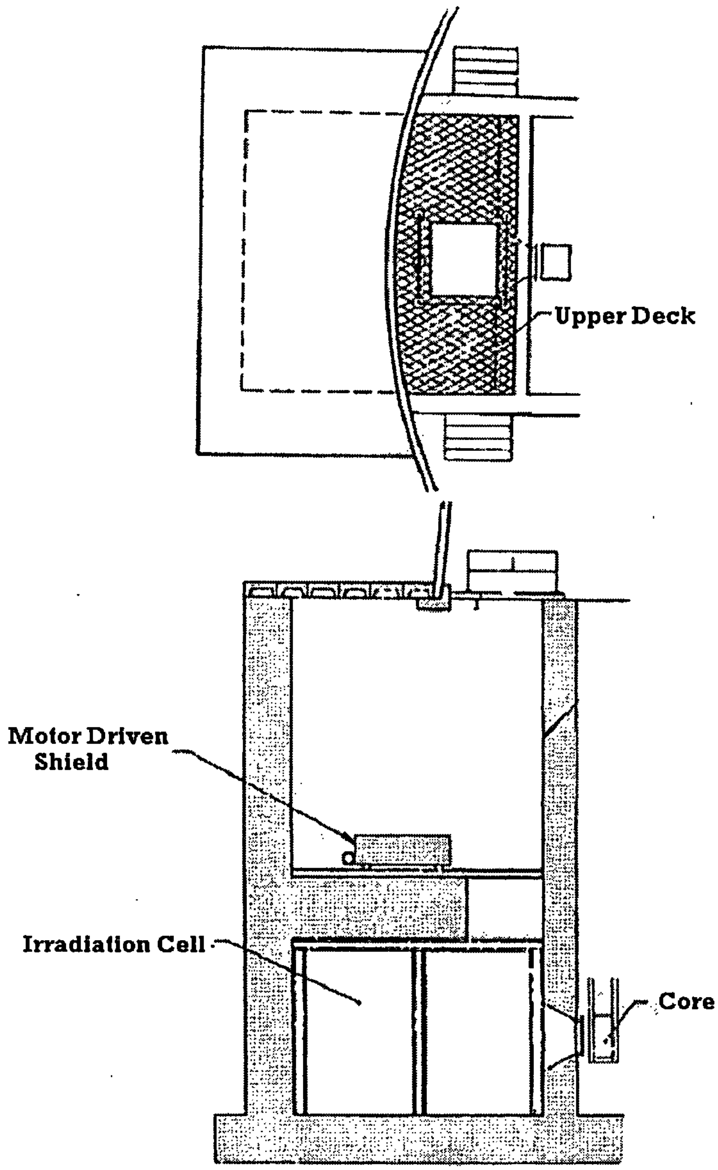

Figure 10-5: Irradiation Cell

A steel ladder extends from the observation deck to the upper irradiation cell and provides access to the upper irradiation cell. A separate removable ladder provides access to the lower irradiation cell. The upper deck over the cell is steel plate with an opening to allow movement of equipment, samples, and people to the lower irradiation cell. 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 upper deck on the south side of the pool. The upper deck is used as an observation area which provides an excellent vantage point for facility visitors. The irradiation cell window is in the two foot thick wall which 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 void 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. Locked mechanical stops on the rail prevent accidental positioning of the reactor closer than eight feet 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 is located inside the cell and radios are used to communicate between the cell and the control room.

10.1.5 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 BP-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.

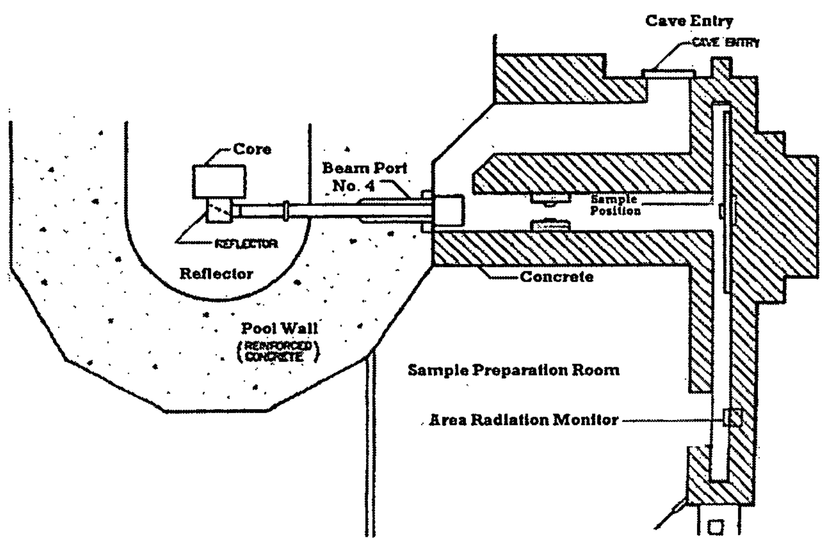

Figure 10-6: BP-4/Radiography Cave

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.

10.2 Experiment Review

The NSC Standard Operating Procedures (SOP) 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 NSCR's Technical Specifications.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

All activities will be conducted in such a manner as to comply with 10 CFR Part 20 "Standards for Protection Against Radiation." Exposure of individuals and release of radioactivity to the environment will be adequately assessed and controlled to maintain compliance with all applicable sections of the regulations. Waste materials containing radioactive isotopes are governed by the operating license and shall be controlled. Methods and instrumentation that are used to establish compliance are outlined in the following sections.

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). The design of the experimental facilities, reactor pool, and the reactor shield includes protective measures and devices, which limit radiation exposures and release of radioactive material to the environment. Information on these aspects of the radiation control program is included in the sections of this report that describe the equipment. General requirements such as dosimeter use and records, certification of training, survey frequency, leak testing of sources, and overall ALARA program are discussed in the Standard Operating Procedures.

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 if the fuel clad is breached. This possibility is one of the accidents considered in Chapter 13 of this report in the analysis of the design basis accident.

- Releases from 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.

The Nuclear Science Center Annual Report, documents the total release of ^{41}Ar from the normal operation of the reactor. Based on the 75 MW-days of operation, the release of ^{41}Ar is approximately 5.24 Ci. Applying a dilution factor of 5×10^{-3}, the releases produce approximately 0.8% of the effluent concentration of 41Ar as specified in 10 CFR Part 20, "Standards for Protection Against Radiation". The EPA COMPLY[1] code indicates that the maximally exposed receptor would receive an annual dose of one mrem from airborne emissions.

On a long-term basis, the pool accounts for more than 95% of the facility's ^{41}Ar production. The ^{41}Ar which is produced in the beam ports and in the irradiation cell is exhausted directly to the building central exhaust, and thus, through the stack. The ^{41}Ar that is produced by activation of the air, which is in the pool water, is transferred through the building ventilation system to the central exhaust. The airborne effluents are discharged to the environment through the building stack, which is 85 feet in height. ^{41}Ar release calculation includes the production from beam port, irradiation cell, and from pneumatics.

On average, the ^{41}Ar production during the beam port operation is about 288 mCi, during the irradiation cell operation is about 354 mCi based on the 75 MW days of operation.[2]

The equations used in developing the dilution factors calculated below are those presented by F.A. Gifford, Jr.[3,4] 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. From reference 13, c is estimated to be 0.5. The cross sectional area of the Nuclear Science Center is 357 m^2.

The equation for the atmospheric dilution factor is:

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

where:

1. X is the concentration in grams or curies per cubic meter;
2. Q is the original source strength in grams or curies per second;
3. u is the mean wind speed in meters per second;
4. y is the crosswind in meters from the plume axis;
5. h is the source height in meters; and
6. 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:

$$X = \frac{Q}{\left(\pi\sigma_y\sigma_z + cA\right)\cdot u} \qquad \text{Eq. 11-2}$$

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.

$$X = \frac{1}{\left(\pi\sigma_y\sigma_z + cA\right)\cdot u} \qquad \text{Eq. 11-3}$$

$$X = \frac{1}{\pi \cdot 4 \cdot 2 + 0.5 \cdot 357} \qquad \text{Eq. 11-4}$$

$$X = 1/203 \qquad \text{Eq. 11-5}$$

This calculation indicates that under the most adverse conditions, the minimum dilution at 100 meters is 200. 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. If the average wind velocity (ten mph) is substituted into this equation, the dilution factor becomes:

$$X = 1/903 \qquad \text{Eq. 11-6}$$

Again, this is a pessimistic approach since the dilution was calculated at only 100 meters (approximate boundary of the exclusion area). The calculation at 1500 meters under stable conditions and with a wind speed of 10 mph yields a dilution factor of 6,920. If the wind speed is reduced to only 2 mph, the dilution factor is still 1,570.

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

^{41}Ar activity is monitored with a gas detector, which utilizes a three-inch NaI(Tl) scintillation crystal and associated electronics. The detector, which is calibrated for ^{41}Ar activity, continuously samples air from the building exhaust plenum. The system is equipped with an adjustable contact, which provides an audible alarm and a warning light on the reactor console and in the reception room.

Stack particulate activity is monitored with a moving-filter type, continuous air monitor using a beta scintillator. This monitor samples air from the building exhaust plenum. This monitor is equipped with an alarm circuit, which activates an audible alarm and a warning light in the control room. An alarm in the system will automatically shut down the air handler units and air dampers to isolate the facility.

Building gas activity is monitored by a gas detector, which utilizes a NaI(Tl) scintillation crystal and associated electronics that is calibrated for ^{41}Ar activity. Air is sampled on the chase level by this monitor. An alarm circuit actuates an audible alarm when preset alarm levels are reached and a warning light is actuated.

Building particulate activity is monitored with a moving-filter type, beta scintillator continuous air monitor. Air is sampled on the chase level by this monitor. An alarm circuit actuates an audible alarm when preset alarm levels are reached and a warning light is actuated.

A fission product monitor that is essentially a beta scintillator is used to monitor primarily particulates that are produced by decay of fission product gases collected in the sampling line. The air-sampling region is located approximately one foot above the pool surface under the reactor bridge. Air is drawn through the line and through the monitor filter paper using an air suction pump. This monitor is equipped with an alarm circuit, which activates an audible alarm and a warning light. An alarm on this system will automatically shutdown the air handler units and the air dampers to isolate the facility.

The area radiation monitoring system provides a continuous indication at the reactor console and in the reception room of the radiation level in each of the monitored areas. An adjustable contact on each indicating meter provides an alarm on the console indicator panel. Red lights on the indicating meter and the detector identify the particular area. The area radiation monitors are located at strategic points throughout the building where the radiation levels might increase and reflect an abnormality or hazard in operations.

Pneumatic tube stations are located outside the NSC confinement building. Each station is installed within a fume hood. The air discharged from each hood is passed through the central exhaust system (fan has a capacity of about 8000 cfm) and then exhausted to the atmosphere.

For routine operations, efforts will be made to keep the concentration of contaminants in the atmosphere released from the NSC are well below the limits as stated in 10 CFR Part 20.

11.1.1.2 Liquid Radioactive Sources

The only activity produced in liquid form in amounts sufficient to be a personnel exposure hazard is Nitrogen-16, which is produced in the reactor coolant as it passes through the reactor core when operating at power levels above 100 kW. ^{16}N is controlled by the use of the diffuser system (Section 5), which reduces the dose rate at the pool surface to two to three mrem/h during full power operation. If the diffuser system fails during full power operation, the dose rate at the pool surface would be limited to less than 100 mrem/h.

Small quantities of liquid radioactive waste are produced from the regeneration of the demineralizer system and effluents from designated facility drains and sinks. Radiation levels from these liquid radioactive wastes are extremely low and do not produce radiation exposure hazards. Disposal of this liquid waste is addressed in section 11.2.2. Typical liquid releases range from 0-12,000 gallons. The NSC annual report notes the isotopic concentrations of liquid waste releases.

11.1.1.3 Solid Radioactive Sources

The major source of radiation and radioactivity is the fission product generation in the reactor fuel. Typical four-element fuel bundles will generate fields of 100 R/h to more than 1000 R/h in air at 3 feet if removed from the reactor pool. As long as the fuel is contained within the water-filled pool, this source of radiation dose presents no personnel hazard. Loss of pool water is considered in Chapter 13. The pool is designed to preclude loss of pool water, and operation would not take place if there were any difficulty in maintaining pool level.

Other possibilities of significant radiation exposure from solid radioactive material are the TRIGA core, samples irradiated for isotopic production, reactor components which have spent a long time near the core, and the reactor startup source. All of these are small sources compared to fuel fission product activity in the operating core. Sample handling equipment, procedures, and the use of aluminum for most structures near the core reduce exposure rates from activated materials to levels which generate no significant personnel hazard during operation and maintenance of the reactor. Activity produced during irradiations is calculated before the irradiations are performed and equipment and procedures are in place to deal with the activity after the sample irradiation is completed.

11.1.2 Radiation Protection Program

All personnel entering the facility will be provided with appropriate personnel monitoring devices. Personnel monitoring devices will include but not be restricted to 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. Respiratory protective equipment is also available for emergency use. However, no allowance for its use will be taken in determining exposure of individuals to airborne radioactive material without specific USNRC authorization.

A change room is provided on the upper research level for use by personnel. Lockers and showers are provided. Showers connected to the "hot" drain are provided in the upper and lower research level for decontamination of personnel. Washing of contaminated clothing can be accomplished on the lower research level where the drain from the washing machine is connected to the "hot" drain.

A radioactive material handling area is located adjacent to the reactor on the upper research level. This area is used for processing and packaging radioactive materials. Protective clothing and equipment are available for use in this area. Access to the area is controlled by internal procedures. The area is posted in accordance with 10 CFR Part 20 requirements.

A standard radiochemistry laboratory on the lower research level is available for research experiments and health physics use. Equipment for routine radiochemical procedures is maintained. Laboratory procedures are developed as required to ensure compliance with radiation protection regulations.

An environmental monitoring program has been established between Texas A&M University and the Texas Department of State Health Services. The program is modified as changes are made to the facility operation and/or equipment.

Portable survey meters are provided to survey operations in restricted areas and to survey all experimental activities to assure that personnel are not inadvertently exposed to excessive radiation levels, and to assure compliance with 10 CFR Part 20 limits and established ALARA limits.

Appropriate counting equipment will be provided 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, which reflect the management commitment to ALARA principles. An annual ALARA review is conducted by the NSC health physics staff and by the campus Environmental Health and Safety staff with a report of the results of the review being submitted to the Reactor Safety Board (RSB).

11.1.3.1 Radiation Monitoring and Surveying

NSC's Standard Operating Procedures reflect these requirements. Installed radiation and facility air monitors are described in Section 7.7 of this report. Area radiation surveys are conducted each month, including checks for contamination and particulate radioactivity. Sample irradiation procedures and forms require checks of radiation level each time a sample is removed from an irradiation facility. Experiment reviews and approvals required radiation surveys for new experiments and modifications of experiments.

11.1.4 Radiation Exposure Control and Dosimetry

NSCR standard operating procedures specify requirements on radiation control and dosimetry. NSC health physics staff administers the dosimetry program. Whole body and extremity dosimeters are used for operating personnel, and experimenters using the NSC on a regular basis and records are maintained at the NSC. Pocket ion chambers are used and doses recorded for tour groups and visitors. Experiment approval requires that no high radiation areas are created external to the experiment shielding. Some experiments have shield cavities larger for personnel entry, however, and high radiation levels can exist inside the shield. Should an experiment design be approved with a high radiation area or very high radiation area within the experiment shield, any entry points to the area will be provided with an alarm that sounds at the reactor console and at the entry point if a person attempts to open the entry point barrier.

Radiation doses received by visitors and tour groups are low; tour groups are not allowed in any area with dose rate exceeding two mrem/hr. No student dosimeter has ever received a measurable exposure from reactor operation. Occupational exposures of operations and maintenance personnel have been low, seldom exceeding one rem total effective dose equivalent (TEDE) in a year.

11.1.5 Contamination Control

NSCR SOPs specify requirements on contamination control. As noted in Section 11.1.4, monthly contamination surveys are conducted. Operating policy is that no detectable removal contamination is allowed; any contamination discovered is immediately decontaminated.

For routine cleaning of radiation areas, the facility has cleaning equipment that is dedicated to use in the radiation areas. Floor sweepings are surveyed for radioactivity before disposal.

11.1.6 Environmental Monitoring

Environmental thermoluminescent dosimetry (TLD) monitors are used and evaluated on a quarterly basis. The TLDs are placed on the fence surrounding the NSC. At present, more than 20 points around the facility are monitored.

11.2 Radioactive Waste Management

NSCR's Standard Operating Procedures specify requirements for dealing with radioactive waste.

11.2.1 Radioactive Solid Waste

Low-level, solid radioactive wastes in the form of gloves, paper towels, used laboratory equipment, sample containers, aluminum, etc., are generated in normal operations associated with the Nuclear Science Center Reactor. Activated materials, such as used experimental hardware, also are generated.

This waste is accumulated in plastic-lined, waste containers located at strategic points throughout the facility. When filled, these containers are monitored, the plastic liner sealed and removed, and the waste stored in the radioactive waste storage building (Figure 2-2). Short-lived radioactive waste is sorted, segregated and disposed as non-radioactive in a local landfill. Long-lived waste is compacted and packaged and sent to a final disposal facility.

Activated equipment is normally stored in the reactor pool or a high-level waste storage area adjacent to the confinement building for future reuse or disposal under applicable regulations.

11.2.2 Radioactive Liquid 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 (Figure 2-6 and Figure 5-6). Waste is transferred by a sump pump to one of three storage tanks located above grade 200 feet northwest of the building. These tanks have a total storage capacity of approximately 37,000 gallons. Each tank is equipped with an inlet valve, outlet valve, volume indicator, and sampling line. There is a valve on the master outflow line, which is secured with a keyed supervisor lock. Fresh water is available to the master outflow line for diluting and flushing the liquid waste being discharged.

The NSC discharges liquid waste from the hold-up tanks to the sanitary sewer system. When a tank is full, it is isolated, and re-circulated to ensure uniformity and sampled to determine activity. The liquid waste from the tank is filtered so that no un-dissolved particulate activities are discharged into the sewer. Sampling, analysis, and release from the tanks are governed by a procedure that assures releases are within 10 CFR part 20, Appendix B, Table 3 limits, and that the pH is within local limits for discharge to the sewer.

[1] U.S. Environmental Protection Agency, COMPLY Program Rev. 2, October 1989.
[2] Technical Report Number 32, "Determination of Argon-41 Production at the Texas A&M Nuclear Science Center Reactor"
[3] F.A. Gifford, Jr., Nuclear Safety, December 1960
[4] 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 deputy director of the TEES for the administration and the proper and safe operation of the facility. Figure 12-1 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 Radiation 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

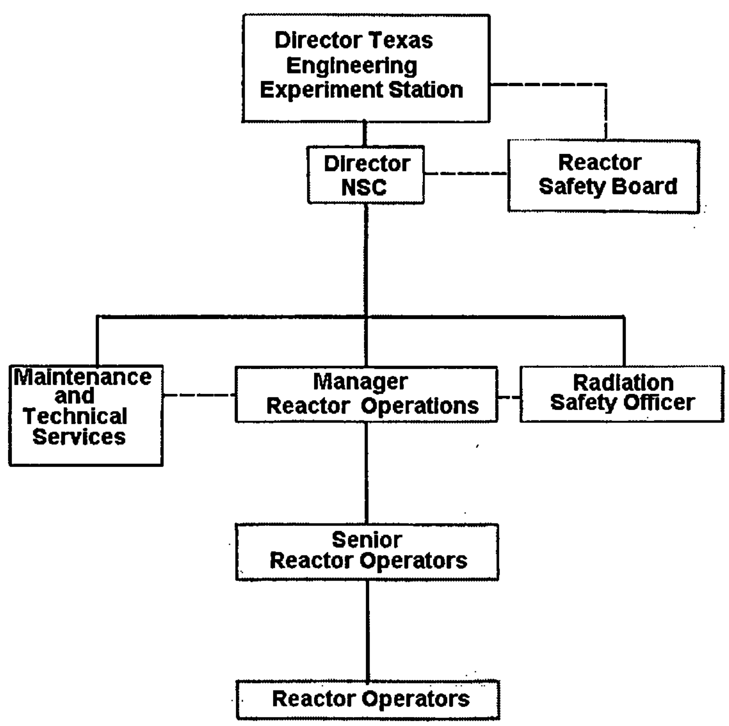

Figure 12-1: Organization Chart for Reactor Administration

A line management organizational structure provides administration and operation of the reactor facility. The deputy director of the TEES and the director of the NSC have line management responsibility for adhering to the terms and conditions of the NSCR license, technical specifications, and for safeguarding the public and facility personnel from undue radiation exposure. The facility shall be under the direct control of the NSC Director or a licensed senior reactor operator (SRO).

12.1.1.1 Management Levels

1. Level 1: Deputy Director TEES (Licensee): Responsible for the NSCR facility license.
2. Level 2: NSC Director: Responsible for reactor facility operation and shall report to Level 1.
3. 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.
4. Level 4: Reactor Operating Staff: Licensed reactor operators, 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

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

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

   Note: During periods of reactor maintenance as specified in 14.1.29 (b), the reactor operator or the operator trainee may be replaced by maintenance personnel. The licensed senior reactor operator may be permitted as the only operations person present at the facility to perform a pre-startup check of the reactor or perform general reactor maintenance not specified in 14.1.26 (b).

2. A licensed reactor operator or senior reactor operator will be in the control room.
3. The NSC Director 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 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), to provide advice and technical assistance in the area of radiation protection.

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

The following designated individuals shall direct the events listed:

1. The NSC Director or his designated alternate shall direct any loading of fuel or control rods within the reactor core region.
2. The NSC Director 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 NRC Operator or Senior Operator examination. This training program normally contains twenty hours of lecture, outside study, and requires several reactor startups. The selection and training of operations personnel shall be in accordance with the following:

1. Responsibility:
    a. The NSC Director 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.

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, as well as sampling water and air. Section 11 of this document details the radiation safety program for this license.

12.2 Reactor Safety Board Review and Audit Activities

The 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 12-1).

12.2.1 RSB Composition and Qualifications

The 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, and
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 un-reviewed safety question or a change in technical specifications;
4. Review of abnormal performance of plant equipment and operating anomalies having a 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, 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 NSC follow the Standard Operating Procedures (SOP). For example, 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 NSC Director, 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;

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

The NSC Director and the Reactor Safety Board shall make substantive changes to the above procedures effective only after documented review and approval. The NSC Director 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 of the Reactor Safety Board, and reports shall be made to the NRC in accordance with Section 14.6.6.2 of the Technical 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 NSC Director or his designated alternate;
2. The NSC Director or his designated alternate shall be notified and corrective action taken with respect to the operations involved;
3. The NSC Director or his designated alternate shall notify the Chairman of the RSB;
4. A report shall be made to the RSB which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence;
5. A report shall be made to the NRC in accordance with 14.6.6.2 of the Technical Specifications;
6. The 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:

1. 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;
2. Tabulation of the energy output (in megawatt days) of the reactor, hours the reactor was critical, and the cumulative total energy output since initial criticality;
3. The number of emergency shutdowns and inadvertent scrams, including reasons thereof;
4. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;
5. A brief description, including a summary of the safety evaluations of changes in the facility or in procedures and of tests and experiments carried out pursuant to Section 50.59 of 10 CFR Part 50;
6. A summary of the nature and amount of radioactive effluents released or discharged to the environment 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.
    a. Liquid Waste (summarized on a monthly basis)
        i. Radioactivity discharged during the reporting period.
            (1) Total radioactivity released (in Curies).
            (2) The Effluent Concentration used and the isotopic composition if greater than 1×10^{-7} mCi/cc for fission and activation products.
            (3) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.
            (4) Average concentration at point of release (in mCi/cc) during the reporting period.
        ii. Total volume (in gallons) of effluent water (including dilutent) during periods of release.
    b. Gaseous Waste (summarized on a monthly basis)
        i. Radioactivity discharged during the reporting period (in Curies) for:
            (1) ^{41}Ar
            (2) Particulates with half-lives greater than eight days.
    c. Solid Waste
            (3) The total amount of solid waste transferred (in cubic feet).
            (4) The total activity involved (in Curies).
            (5) The dates of shipment and disposition (if shipped off site).
7. A summary of radiation exposures received by facility personnel and visitors, including dates and time where such exposures are greater than 25% of that allowed or recommended.

8. 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 fax or similar conveyance. This is to be followed by a written report, which describes the circumstances of the event within 14 days of any of the following:
    a. Violation of safety limits (Section 12.4.1);
    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 technical 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 environment
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 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 reviewed 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 (QA) 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. Furthermore, experiment reviews use written requirements to ensure 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 ensures 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 of reactor manipulations to ensure proficiency, and operator evaluations to ensure an effective skill level.

12.11 Reactor Reload Consideration

As already discussed in Section 4.5.7, the NSC LEU 30/20 core is a "lifetime" core. It will have capability to operate at 1.0 MW on demand for 35 years following a weekly schedule of $\leq$70 MW Hr. However, it is conceivable that one or more fresh instrumented fuel elements may be acquired and installed in a heavily burned core. The concern for excessive power peaking is evaluated and discussed in the following paragraph.

To assess extreme power peaking that can occur when adding fresh fuel elements to a depleted core of fuel elements, the following case was analyzed. Three fresh fuel elements were added to the NSC core after 2000 MW-days of burnup. Because the IFEs, after such a burnup, may have failed and need to be replaced, the IFE locations were selected for placing fresh fuel elements. Specifically, fresh fuel was placed in locations 6D4, 5E4, and 4D4. The resulting rod power factors for these locations are 1.401, 1.502, and 1.546. All of these rod power factors are lower than the maximum rod power factor of 1.565, which occurs at BOL. Peak temperatures during steady state operation would likewise be lower than peak temperatures at BOL. For pulsing transients, the peak temperature is dependent on the peak power factor $(\hat{P}_{rod}/\overline{P}_{core})$. For the three fresh fuel elements in 6D4, 5E4, and 4D4, the peak power factor is 2.777, 3.064 and 4.585 respectively. The peak power factor of 4.585 is higher than the peak power factor at BOL which means either that the maximum allowable reactivity insertion during a pulse would be lower or that a new IFE should not be placed in 4D4. The power peaking associated with fresh fuel reload would be lessened if heavily depleted fuel in the core center is replaced with lightly depleted fuel from the core periphery and additional fresh fuel elements are placed in the core periphery.

12.12 Startup Plan

This Safety Analysis Report does not include a startup plan because the facility is currently conducting routine operations at power.

12.13 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 2.0 MW_{th} 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 SAFETY AND ACCIDENT ANALYSIS

13.1 Safety Analysis

The safety of TRIGA fuel is due entirely to the design features. The safety features of a standard TRIGA fueled core are well known. Each of the LEU fuel types is designed to replace a standard fuel (8.5 wt % U, 20% enriched) element, as regards reactivity; that is, [redacted] in a compact configuration is intended to have about the same core excess reactivity as [redacted] the more heavily loaded TRIGA fuel elements.

As part of the Reduced Enrichment for Research and Test Reactors (RERTR) program, various tests were performed on high-uranium content, low-enriched TRIGA fuels and the test results were submitted to the NRC. The NRC concluded in their Safety Evaluation Report that both the 20-20 and 30-20 uranium-zirconium hydride fuels "are generally acceptable for use in other licensed TRIGA reactors, with the provision that case-by-case analyses discuss individual reactor operating conditions in applications for authorization to use them"[1].

In the present document, it is shown that one-for-one, fresh TRIGA FLIP fuel and fresh TRIGA LEU 30/20 fuel behave very similarly under cold, clean critical operations. However, both these types of TRIGA fuel react strongly to in core water filled regions. In PRNC, all control rods had water followers that were accounted for in the pulsing model. At the NSCR, all control rods near core center will have air or metal followers; however, two water- filled regions representing future experiments generated high pulsed temperatures in certain in-core fuel rods.

13.2 Safety Limits

The safety of the operating TRIGA reactor system with LEU 30/20 fuel is related directly to the maximum temperature of the fuel and the continued availability of coolant. As demonstrated for all TRIGA fuel elements, the safety limit for water-cooled fuel is taken conservatively as 1150°C. The safety limit for these fuel elements when air-cooled is 950°C.

As analyzed in this report, all proposed reactor operations will involve low fuel temperatures with large margins of safety. The peak fuel temperature in steady state operation at 1.0 MW (1.3 MW momentarily for the purpose of testing power level scrams) is 373°C (440°C). Both of these temperatures have large margins of safety for the 1150°C safety limit. In normal pulsing, the NSC facility has chosen 830°C as the limit for the peak fuel temperature and controls the reactivity insertion to maintain this limit. The limiting peak pulsing fuel temperature (830°C) has been chosen to address the problem of hydrogen migration resulting from long term high power operation.

The two power level scrams (125% of 1.0 MW) are used to assure a reactor power at a level that gives acceptable fuel temperatures as noted above.

The fuel temperature scram (1) (525°C) is maintained in all modes of operation. The high power level scrams (2) (125%) are effective in steady state mode of operation.

In addition to the protection provided by the several, redundant scrams, administrative procedures and written operating procedures contribute additional limits on operation to protect the reactor, the facility, and the public.

13.3 Evaluation of LSSS for NSC LEU 30/20 Fuel

13.3.1 Steady State Mode

The value of the limiting safety system setting (LSSS) is chosen to prevent the TRIGA safety limit (1150°C) from being reached in any mode of operation. The LSSS in an IFE has been selected as 525°C. The location of the fuel cluster containing the instrumented fuel element shall be chosen to be as close as possible to the hottest fuel element in the core. (The hottest element in the core is (5D3) in Figure 4-2 adjacent to the transient rod. In the present analysis, the instrumented fuel element is located at (5E4). Other locations can be chosen for the IFE.) The LSSS temperature setting is smaller than the safety limit by an amount to account for several factors, including:

1. Accuracy of temperature calibration
2. Precision of electronic readout/scram circuitry
3. Account taken of location of sensing tip of thermocouple 0.3 inch from axial center line of IFE
4. Difference in peak temperature in IFE compared to that in the hottest fuel element

The basis for selecting 525°C as the limiting safety system setting is the following. The limiting safety system setting is a temperature, which, if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from being exceeded. A part of this margin is used to account for the difference between the maximum core temperature and measured temperatures resulting from the actual location of the thermocouple, 0.3 inches from the axial center line of the fuel element. If the instrumented fuel element were located in the hottest position in the core, the difference between the true and measured temperatures would be small. However, this fuel position is not available for technical reasons due to the core configuration with the four-rod fuel clusters and to the location of the transient rod. The location of the instrumented elements is therefore restricted to the positions close to the central fuel element. Calculations indicate that, for this case, the true temperature at the hottest location in the core at 1.0 MW will differ from the measured temperature by about 13.4% (373/329 = 1.134) (Table 4-18). Thus, for the steady state mode of operation, if the temperature in the thermocouple element were to reach the trip setting of 525°C, the true temperature at the hottest location in the core would be less than 600°C, providing a safety margin of at least 550°C for LEU 30/20 type elements. At a steady state reactor power of 1.3 MW (for momentary testing and surveillance measurements) the peak fuel temperature $\hat{T}$ in the hottest fuel in the NSC LEU 30/20 core would only be at most 440°C. These resulting safety margins are ample to account for any remaining uncertainty in the accuracy of the fuel temperature measurements and any overshoot in reactor power resulting from a reactor transient during steady state mode operation.

13.3.2 Pulse Mode

In the pulse mode of operation, the same safety limit will apply. However, the temperature channel will have no effect on limiting peak powers generated because of the relatively long time constant (seconds) for the recorded temperature as compared with the width of the pulse (few milliseconds). In this mode, however, the temperature trip, if activated, will cause all scrammable rods to fall and will act to reduce the amount of energy generated in the entire pulse transient by cutting the "tail" of the energy transient even if the pulse rod remains stuck in the fully withdrawn position.

13.4 Maximum Allowable Pulsed Reactivity Insertion

In Sections 4.5.12 and 4.5.14, the pulsing performance of the LEU 30/20 core has been reviewed for both beginning of core life and at 2000 MWD burnup. At beginning-of-life, a reactivity insertion of $2.10 is calculated to produce the limiting peak pulse fuel temperature of 830°C. At a core burnup of 2000 MWD, a reactivity insertion of $2.10 was calculated to produce a fuel temperature of approximately 810°C.

In the initial commissioning and pulse calibration of the NSCR LEU core in 2006 the maximum reactivity insertion that would yield the limiting peak pulse fuel temperature of 830°C was measured to be $1.91. In view of the initial calibration data and the small change in calculated peak fuel temperatures with core burnup, a maximum reactivity insertion of $1.90 is a reasonable choice, as the $2.10 limit has no margin of error. The calculated $1.90 limit is to serve as a lifetime limit; however, the NSC shall periodically perform pulsing measurements to determine if a more restrictive reactivity insertion limit is required due to actual core performance.

13.5 Accident Analysis

13.5.1 Analysis Changes to DBA/MHA Event

The design basis accident (DBA) is defined as the loss of integrity of the fuel cladding for one fuel element and the simultaneous loss of pool water resulting in fission product release. NUREG/CR-2387[2] suggests, and NRC accepts, that for a 1.0 MW TRIGA reactor, the DBA is the release of fission products into the air from a single irradiated fuel element. The loss of pool water is typically treated separately as a loss of coolant accident (LOCA). For NSC SAR, the simultaneous loss of coolant and rupture of a single fuel element in air is considered a DBA-MHA (maximum hypothetical accident) event.

In this section, the radiological impact of the loss of fission products from a single TRIGA fuel element is reviewed. To compare the relative abundance of fission products, the pertinent operating parameters are compared for a FLIP core and a LEU 30/20 core. For the LEU 30/20 core, the core average fuel temperature at 1.0 MW is 237°C (Table 4-18) compared to 234°C (Table 4-13) for a FLIP core. The major difference lies in the fission product inventory due to burnup. A [redacted] FLIP core has a life of about 2350 MWD versus about 2000 MWD for the NSC LEU 30/20 core. The energy burnup capability based on >50% of the U-235 has been evaluated as 77 MWD per fuel element for FLIP fuel.[1] For NSC LEU 30/20 fuel, the energy burnup capability based on >50% of the U-235 gives 57 MWD per fuel element. The evaluated dose resulting from the fission product release from a single FLIP fuel element was found acceptable by the NRC.

The hottest fuel element in the LEU core at the end-of-life will contain less activity than assumed in the DBA analysis for FLIP fuel. Thus, the radiological impact from a DBA event on the members of the public and the facility workers will be less than that evaluated for FLIP fuel.

A summary report of these studies3 indicate that release from the $UZrH_{1.6}$ fuel meat at the steady state operating temperatures is principally through recoil into the fuel-clad gap. At high temperatures (above 400 to 500°C), the release mechanism is through a diffusion process and is temperature-dependent, unlike recoil.

In steady state operation, the peak fuel temperature establishes the fission products in the fuel-clad gap. Since the axial fuel temperature determines the fission product release, there is no further latitude in fission product release. For 1.0 MW operation, the peak integrated fuel temperature corresponds to a release fraction of 2.6×10^{-5}.

The fission product inventories in the Table 13-1 were calculated using ORIGEN. Both FLIP and LEU 30/20 inventories were calculated for a single fuel element. The element was assumed to have a conservative power density of 28 kW. The burnup calculation was performed for 200 days (5.6 MWD) in order to achieve saturation levels for all of the isotopes except ^{85}Kr. The burnup calculation for ^{85}Kr was extended to 77 MWD for FLIP fuel and 54 MWD for LEU 30/20 fuel.

[redacted]

Using the data in Table 13-1, the saturated activities of the significant fission products for a single LEU 30/20 fuel element at 1.0 MW are:

[redacted]

Applying the release fraction of 2.6×10^{-5} to the total inventory in a single element operating at 1.0 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 3.2×10^{-4} µCi/cm^3. Within 24 hours, this value would decay to 7.3×10^{-5} µCi/cm^3. The demineralizer system (Section 5.4.1) would remove these soluble fission products and they would eventually be transferred to the liquid waste system.

NSC calculated the results of the release of fission products from a single fuel element without water in the reactor pool, and with the ventilation system shut down[3]. Table 13-2 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 five minutes after release. The maximum exposure is 3.7 rem to the thyroid. The exposure to personnel in unrestricted area during the accident is calculated to be minimal. Thus, no realistic hazard of consequence will result from the DBA.

Table 13-2: Summary of Radiation Exposures Following the Release of Fission Products from a Single Fuel Element

Exposure Criteria	Whole Body Dose	Thyroid Dose
	mrem	mrem
Maximum exposure to operating personnel at 5 minutes after release	0.25	3720
Maximum exposure to personnel in unrestricted area	6.2	71
Exposure to personnel in nearest permanently occupied area (nearest resident)	0.14	1.8

The evaluated dose resulting from the fission product release from a single LEU 30/20 fuel element was found acceptable by the NRC.

13.5.2 Analysis of Changes to LOCA Event

A detailed analysis[4] has evaluated the safety of irradiated fuel elements in a loss of coolant accident (LOCA) for long term operation at 1.0 MW. With power levels no higher than 21 kW per element, air cooling is sufficient to prevent excessive fuel temperatures with essentially no delay between reactor scram and loss of coolant. For power densities up to 23 kW per element, a 15 minute delay is required between reactor scram and loss of coolant. TAMU has further refined the analysis[5] to account for the fact that the 1.0 MW operations is limited to ≤70 MW hr/week. As a result, fuel elements with power densities up to 28 kW per element can be safely cooled by air with a 15 minute delay between reactor scram and complete loss of coolant.

For the LEU 30/20 core, operating at 1.0 MW, the largest power per element is 17.6 kW per element. This is well below the 21 kW per element noted above as not requiring any delay between reactor scram and complete loss of coolant. It also should be noted that it is not credible to drain over 100,000 gallons of coolant water from the reactor tank in zero time. Table 13-3 provides actual drain time to the various reactor pool penetration heights from an initial pool volume of 106,000 gallons.

[redacted]

Several of the assumptions in the accident analysis[3] are tabulated in Table 13-4 for FLIP fuel and for the NSC LEU 30/20 fuel. Each of the listed parameters for the LEU fuel will reduce the estimated pressure inside the fuel clad and increase the safety margin for the LOCA event.

Table 13-4: Comparison of Assumptions for FLIP (HEU) and LEU 30/20 Fuel for LOCA Event

Parameter	FLIP (HEU)	LEU 30/20
Energy Burn-up Capability (50% of the U-235)	77 MWD per Element	57 MWD/Element
H-Zr Ratio	1.7	≤1.65
Peak Temperature (Steady State)	600°C	≤400°C
Lifetime Operation	Continuous	≤70 MW Hr/Week
Air Backfill	Present in Analysis at EOL	Not present at EOL*

* After a few hours of operation at full power, oxides and nitrides deplete the air backfill

In conclusion, the margin of safety during a LOCA event with the NSC LEU 30/20 fuel is greater than that previously evaluated for the NSC FLIP (HEU) fuel. Based on the previous analysis[1], the peak power/fuel element of 17.6 kW per element lies comfortably below the NSC limit established for a maximum weekly operation of 70 MW hr and a 15 minute delay time before complete loss of cooling water. In fact, for power/fuel element ratio up to 21 kW per element, all fuel can be cooled by air flow alone with no delay between reactor scram and complete loss of coolant water.

13.5.3 Accidental Pulsing from Full Power

13.5.3.1 BOL, Beginning of Core Life

The rapid insertion of a large amount of positive reactivity in the reactor operating at 1.0 MW is postulated. The method of inserting this reactivity is either by the ejection of an inserted transient rod or unplanned removal of a two-dollar experiment (an experiment that is required by technical specifications to be securely mounted in core). The technical specification limits the maximum insertion of a transient rod to that value which produces a maximum allowed fuel temperature no higher than the safety limit (1150°C). This reactivity is about $2.95 for the NSC LEU 30/20 core. The full worth of the transient rod is limited to reactivities that produce temperatures less than the safety limit (1150°C) by installing a mechanical stop which physically prevents the withdrawal of the transient rod at reactivities greater than $2.95. Furthermore, an electro-mechanical interlock prevents air from being applied to the transient rod in pulse mode when the mechanical stop is not installed.

Pulsing from full reactor power (1.0 MW) would be clearly an accident. In addition to the mechanical and electro-mechanical pulse stop safety interlocks there is another safety interlock that prevent application of air to the piston for any reactor power above 1 kW. In addition, administrative procedures prevent the reactor operator from switching the mode switch to PULSE during high power operation and pushing the TR FIRE READY button, which applies air to the transient rod. For the operator to do so would clearly violate three standard operating procedures.

The sequence of events leading to the postulated transient rod ejection accident at BOL is the following:

[redacted]

The consequences of the above sequence of events are the following:

1. The reactor power increases from 1.0 MW to a peak pulsed power of about 827 MW.
2. The maximum fuel temperature (864°C) is reached immediately after the peak power.
3. The energy release is about 15.8 MW-sec in about 1.0 second when the maximum measured fuel temperature ($\hat{T}_{0.3}$ = 614°C) is reached.
4. At peak fuel temperatures substantially below 1150°C, the strength of the clad maintains clad integrity so long as it remains water cooled.

This accident can be viewed in a number of ways. The transient rod reactivity is limited to $2.95 or less by the pulse stop, not the full stroke value of $3.21. The equilibrium pressure of hydrogen over the fuel is not achieved during the abbreviated pulse. It was incorrectly assumed in the earlier analyses that the backfill air left in the fuel during manufacturing is still present at EOL, whereas in fact it has disappeared, forming oxides and nitrides during the first days of operation

at full power. With the action of the dual power scrams at 125%, the scram of the control rod bank starts even before peak pulsed power is attained.
The pulsing calculations from power have involved hand calculations since BLOOST cannot handle a pulse from power. BLOOST is a zero-dimensional, combined reactor kinetics-heat transfer code. It cannot handle the "inverted U" temperature distribution in fuel operating in steady state coupled with the "U shaped" power distribution in a pulsed fuel element. BLOOST calculates an average core temperature as a function of time.

Calculations were made to establish the average fuel temperature at the steady state starting power of 1.0 MW. A value of 237°C was determined for $\overline{T}_{core}$.

The BLOOST calculations indicate that the highest average fuel temperature in the pulsed core immediately after the transient pulse is 356°C. From three-dimensional diffusion theory calculations, the peak-to-average power ratio was determined for steady state operation to be 4.2. Although the highest temperatures occur at the center of the hottest fuel element during 1.0 MW steady state operation (373°C) and before the pulse, the maximum fuel temperature after pulsing occurs at the edge because of the large power peak.

The peak-to-average ratio value at the edge of the hottest pulsed element is 4.2. Using these power ratios and considering the energy release during the transient superimposed on the energy density levels under steady state, coupled with the volumetric heat content of the fuel, a maximum fuel temperature of 864°C was obtained based on the average core temperatures computed by BLOOST.

An alternative method of producing the accidental pulse from full power is to remove an installed, two-dollar experiment. The technical specification requires such an experiment to be securely locked in position in the core. If somehow, the experiment is loosened and quickly removed from the core because of its mass, the removal time is typically assumed to be 0.3 second (greater than the 0.1 second for the engineered transient rod drive). The much slower withdrawal time will result in activating the 125% power scrams at a lower portion of the reactivity insertion curve. This will result in lower peak power and lower peak fuel temperatures than those for a $2.95 transient rod.

Neither of the aforementioned accidental pulses will endanger the reactor. To review:

The accidental pulsing of the transient rod at full power requires the following:

[redacted]

[redacted]

The accidental pulsing by removal of a $2.00 experiment at full power requires the following:

[redacted]

[redacted]

13.5.3.2 EOL Pulsing from Full Power at 2000 MWD Burn-up

The BLOOST code was run with input appropriate for the LEU core at 2000 MWD. Pulsing results were obtained for a series of reactivity insertions under the same conditions as set forth in the above section. Table 13-5 lists these results along with those for the beginning of core life.

Table 13-5: Comparison of Results from Pulsing from 1.0 MW at Beginning-of-Life (BOL) and End-of-Life (EOL)

Parameter	BOL			EOL(2000 MWD)		
$\Delta k/k\beta$ ($)	$2.10	$2.95	$3.21	$2.10	$2.95	$3.21
$\hat{T}$ (°C)	740	864	866	740	888	889
$\overline{T}_{core}$ (°C)	356	423	423	370	454	456
$\hat{T}_{0.3}$ (°C)	555	614	614	576	654	654
MW-sec	10.9	15.8	15.9	12.3	18.8	18.8
$\hat{P}$ (MW)	471	827	830	511	983	986

In Table 13-5, the effects are evident for shifts in the power distribution with burnup for the core. The measured temperature ($\hat{T}_{0.3}$) increases from 555°C to 576°C with burnup. The peak temperature in the hottest fuel element rises slightly, from 864°C to 888°C; but even 888°C is well below the 1150°C safety limit.

The conclusion is that accidental pulsing from full power is not a hazard for the reactor, either at BOL or at EOL (2000 MWD burnup).

13.5.4 Loss of Coolant Flow

The NSC reactor uses natural convection flow; therefore, this accident is not applicable.

13.5.5 Mishandling or Malfunction of Fuel

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

13.5.6 Experimental Malfunction

The Reactor Safety Board must approve any new class of experiments 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 TRIGA installations appear to be comparable and therefore no hazard exists. The reactivity worth of any non-secured experiment shall be less than $1.00, which will ensure that the removal 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.

Section 13.5.3 addresses the results of an experiment with a reactivity worth of $2.00 being removed from the core at power.

13.5.7 Loss of Normal Electrical Power

The NSCR will shut down upon loss of electrical power. Electricity is required to maintain magnet current in the control rod drive, which 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.

13.5.8 External Events

Floods, hurricanes, earthquakes and tornados are the four 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 Section 2 of this report states, earthquakes are extremely unlikely in this area. Finally, tornados may 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. Due to the confinement building structure and the below grade location of the reactor core, impact 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.

A group of explosive experts from the U.S. Army analyzed the University of Wisconsin reactor facility after a bomb severely damaged the building during the Vietnam era. The Wisconsin pool is similar to the NSC pool. It is eight feet thick at the bottom rather than five for the NSCR pool. However, it is of normal concrete rather than the high density concrete that makes up the NSC pool. In addition, the Wisconsin pool has aluminum lining rather than stainless steel. Both reactor buildings are accessible by automobiles. The following is an excerpt from the Wisconsin Reactor SAR that describes the conclusions of the explosive experts:

"...an explosive charge placed outside the building walls where vehicular access was available would not cause any damage within the reactor pool. Further, we were told that the amount of explosive material required to breach the pool walls, other than by use of shaped charges, was large enough that it would take a considerable amount of time for a person with access to the laboratory to assemble and tamp that our security system should give us more than enough warning. If a person has access to the top of the pool some damage to core internals could result from large explosive charges, but the security system we had in place would alert our security warning (system). We conclude that no accident with results more severe than those considered elsewhere in this chapter is credible."

The NSC reaches the same conclusion for the NSCR as the University of Wisconsin reached for its reactor.

13.6 Summary and Conclusion

It has been demonstrated that none of these credible accidents will result in consequences to the public health and safety.

[1] Lawrence, R.D., "The DIF3D Nodal Neutronics Option for Two-and-Three-Dimensional Diffusion Theory Calculations in Hexagonal Geometry," Doc. No. ANL-83-1, Argonne National Laboratory, March 1983.
[2] NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," USNRC, August 1987.
[3] Vasudevan, L., "Radiation Exposure Following the release of Fission Products During a Design Basis Accident Condition," Nuclear Science Center Internal Report, August 2006.
[4] F.C. Foushee, "TRIGA Four-Rod Cluster Loss of Coolant Accident Analysis," GA Report No. E-117-196 (October 1972).
[5] Details available in Appendix II to the TAMU License revised through Amendment No. 15, 1999.