FG 03-93ER75884

FINAL REPORT

**Nuclear Safety Analyses and Core Design Calculations to Convert the
Texas A&M University Nuclear Science Center Reactor to Low Enrichment
Uranium Fuel**

Theodore A. Parish

Principal Investigator

Nuclear Engineering Department

Texas A & M University

College Station, TEXAS 77843-3133

March 2, 1995

BD final

DOE/ER/75884--T1

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Executive Summary

This project involved performing the nuclear design and safety analyses needed to modify the license issued by the Nuclear Regulatory Commission to allow operation of the Texas A&M University Nuclear Science Center Reactor (NSCR) with a core containing low enrichment uranium (LEU) fuel. The specific type of LEU fuel to be considered was the TRIGA 20-20 fuel produced by General Atomic.

Computer codes for the neutronic analyses were provided by Argonne National Laboratory (ANL) and the assistance of William Woodruff of ANL in helping the NSCR staff to learn the proper use of the codes is gratefully acknowledged. The codes applied in the LEU analyses were WIMSd4/m, DIF3D, NCTRIGA and PARET. These codes allowed full three dimensional, temperature and burnup dependent calculations modelling the NSCR core to be performed for the first time. In addition, temperature coefficients of reactivity and pulsing calculations were carried out in-house, whereas in the past this modelling had been performed at General Atomic. In order to benchmark the newly acquired codes, modelling of the current NSCR core with highly enriched uranium fuel was also carried out. Calculated results were compared to both earlier licensing calculations and experimental data and the new methods were found to achieve excellent agreement with both. Therefore, even if an LEU core is never loaded at the NSCR, this project has resulted in a significant improvement in the nuclear safety analysis capabilities established and maintained at the NSCR.

The steady state nuclear design calculations were largely concerned with identifying the "best" erbium neutron absorber concentration to be used in the LEU fuel. The erbium concentration needed to be set so as to maintain an adequate minimum control rod shutdown margin while achieving a useful core lifetime. The recommended erbium concentration in the TRIGA LEU 20-20 fuel for the NSCR core configuration was found to be .59 weight percent as compared to .47 weight percent for the erbium concentration in "standard" TRIGA LEU 20-20 fuel. By converting to LEU fuel, the NSCR core life is decreased significantly as compared to that for the current TRIGA FLIP fuelled core. In addition, since the LEU core's excess reactivity decreases with burnup, the control rod margin at the beginning of core life is significantly smaller than for the FLIP fuelled core. Nevertheless, the erbium concentration of .59 weight percent achieves adequate reactivity shutdown margin while allowing a core lifetime of approximately 4 full power years. Since the individual rod powers are not substantially affected by the change from FLIP to LEU fuel, the calculated limiting safety system setting (LSSS), i.e., the temperature rise as measured by a thermocouple inside an instrumented fuel rod, was not found to be much different than the LSSS calculated for an all FLIP fuelled core.

The pulsing and transient analyses were performed with PARET and relied on temperature coefficients of reactivity calculated using WIMSd4/m and DIF3D. Calculations for the current FLIP fuelled core were compared to experimental information as well as the results from earlier licensing analyses. The transient performance of the new LEU core was found to be generally similar to that of the FLIP fuelled core although there are some differences primarily due to the burnup dependence of the temperature coefficient of reactivity of the LEU 20-20 fuel.

The results of the nuclear safety analyses performed during this project are presented in this final report in the form of a "draft" Safety Analysis Report (SAR) for the NSCR that could eventually be submitted to the NRC in order to modify the current license to allow for operation with an all LEU fuelled core. All pertinent sections of the current SAR, originally submitted in June 1979, have been rewritten to reflect the changes needed to support loading of an all LEU 20-20 fuelled core. Many chapters of the SAR required only superficial changes while others needed to be totally rewritten. The SAR Sections/Chapters or Appendices that are either new or required substantial modification were 1) Chapter III (Reactor Design) Section C (Nuclear Design), Chapter XI (Safety Evaluation), Appendix I (WIMSd4/m and DIF3D), Appendix II (NCTRIGA) and Appendix III (PARET).

The work of several graduate assistants was essential in performing the analyses required for this project. Brad Rearden contributed significantly to both the transient modelling using PARET and the organization and writing of this final report. Without Brad Rearden's efforts, much coming after the initial DOE funding was terminated, this final report may never have been produced. The strong modelling work of Chien-Hsiang Chen is also acknowledged. Chien-Hsiang Chen was responsible for performing the WIMSd4/m and DIF3D calculations reported here and for carrying out the temperature coefficient of reactivity, shutdown margin, loading to initial criticality and erbium poison worth calculations. A third graduate assistant, Mark Bigler, made important contributions by determining the material compositions and dimensions of all of the core components and by constructing the initial three dimensional DIF3D model of the NSCR core. Finally, Russell Johns contributed to this project by performing the initial NCTRIGA and PARET calculations and by implementing a plotting capability for the DIF3D power and flux distributions using TECPLOT.

SAFETY ANALYSIS REPORT

for the

**Nuclear Science Center Reactor
Texas A&M University**

DRAFT

March 1995

Preface

This document is submitted in support of the renewal of License R-83 with the intention that it supersede all previous submittals in Docket 50-128 that pertain to reactor safety. This Safety Analysis Report (SAR) is a consolidated and updated safety analysis for the continued operation of the NSCR using FLIP and LEU TRIGA fuel and contains previously reviewed material from the August 1967 SAR and its supplements dated November 1972, January 1975 and June 1979.

The purpose of this SAR is to provide a description and safety analysis of structures, systems and components in terms of their ability to provide proper operational performance and functions for the 20 year term of the license renewal. The continual upgrading programs implemented since initial operation of the NSCR have improved reactor safety and prevented the need for restrictions on reactor operations due to age of structures or equipment.

Acknowledgments

This document was generated through the efforts of several members of the Nuclear Science Center staff. Dr. Ted Parish coordinated and advised the graduate students performing the analysis. These students are Mark Bigler, Chien-Hsiang Chen, Russell Johns, and Brad Rearden. Technical and financial support was provided by the Nuclear Science Center.

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I. INTRODUCTION

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

The application for a construction permit and operating license was submitted in March, 1958, along with the hazards Summary Report. Supplement I to the Hazard Summary Report was submitted in 1959. The construction permit, No. CPRR-38 was issued in August, 1959. This permit was converted to operating license, R-83, which authorized operation of a MTR swimming pool type reactor at 100 Kw.

The reactor was first taken critical on December 18, 1961. Since that time, the use of the facility has increased steadily and presently the NSCR supports active nuclear education and research programs. The facility serves many campus departments, other universities and colleges, several city and state agencies, and other industrial and research organizations. By January, 1965, the use of the facility had increased so that it was necessary to operate on a two shift basis three days a week and one shift operation for two days. Since July, 1966, the reactor has routinely operated two shifts for five days a week. In 1968 the reactor was converted to TRIGA fuel and the power level was increased to 1,000 kW. After only three years from initial reactor operation, a comprehensive upgrading program was implemented. In December of 1965 proposals were submitted to the National Science Foundation and the Atomic Energy Commission for funds to support a long range expansion program.

The expansion of the facility included four separate phases, all of which have now been completed. They are described briefly below:

Phase I. Pool Modification and Liner

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

Phase II. Cooling System

To allow steady state operation at power levels up to 1 MW, a cooling system has been provided for the reactor. The 1 MW reactor power was needed to improve the neutron flux for a number of existing research programs and to encourage initiation of new projects.

Phase III. Conversion of the Reactor Core

The reactor core was converted to employ Standard TRIGA fuel elements, and on July 31, 1968, an amended facility license allowed the NSCR to be operated at a maximum steady state power level of 1000 kilowatts and with pulsing up to a \$3.00 reactivity insertion. The inherent safety of the TRIGA fuel allowed increased operational flexibility and utilization of the reactor. Pulsing was

possible because of the prompt negative temperature coefficient of reactivity and the integrity of TRIGA fuel at the peak temperature attained.

Phase IV. Laboratory Building

The original research space within the Nuclear Science Center was quite limited. A laboratory building was constructed which adequately accommodates the present research load and allows for anticipated expansion of programs.

From initiation, the expansion plan covered a period of 3 1/2 years until completion in mid 1969. The plan not only changed the initial facility physical plant but also established a new reactor program.

Operating experience with standard TRIGA fuel revealed a high fuel burnup rate resulting in fuel additions to maintain sufficient excess reactivity. Core life was extended by modification of the reactor grid plate in late 1970 to provide for the installation of fuel followed control rods. This increased the core life by approximately 1 1/2 years. Subsequent operation, however, eventually required the addition to the core of all the standard TRIGA fuel on hand. This seriously reduced the fluxes that were available at the irradiation facilities. The solution to this problem was the initiation of a program to provide a core loading utilizing TRIGA FLIP* fuel. In June, 1973, the NSCR was licensed to operate Standard, Mixed or FLIP TRIGA cores. The mixed cores were licensed to operate at a maximum steady state power of 1,000 kW with maximum pulse reactivity insertion of \$2.00. In July, 1973, the first NSCR mixed TRIGA core containing 35 FLIP and 63 Standard elements was placed into service. In July 1975 the maximum pulse reactivity insertion was increased to \$2.70. In 1979 a core of all FLIP fuel was loaded at the NSCR. Since this time, the core has operated at a steady state power of 1,000 kW with all FLIP fuel elements.

*General Atomic - Fuel Life Improvement Program

II. SITE

A. The Site and Adjacent Areas

1. Description and Location

The Texas A&M University Nuclear Science Center (NSC) is situated on a rectangular six-acre site which is located 1,500 feet from the north-south runway of Easterwood Airport, six miles south of the city of Bryan, (est. pop. 46,600), three miles southwest of the main campus of Texas A&M University, two and one-half miles west-southwest of the city of College Station (est. pop. 42,400), and eight miles northwest of Wellborn (est. pop. 1,200), in Brazos County, Texas (See Figure 2-1).

2. Control

The land adjacent to all sides of the site is owned and controlled by the University. The indemnity confines of the site are defined by a chain-link steel fence which provides reasonable restriction of access to the site. The only entrance into the site is through a chain-link steel gate at the east end of the site (See Figure 2-2). The entire area inside the perimeter fence of the NSC is designated as a "Restricted Area. A sign at the main gate will direct incoming personnel to report directly to the Reception Room. It is the responsibility of the receptionist to observe incoming vehicles and personnel. Improper access will be brought to the attention of the Reactor Supervisor immediately. Entrance through the gate will actuate an audible signal in the Reception Room. Initial entry for the day will be through the Reception Room for all personnel. Personnel monitoring devices will be issued in the Reception Room. The radiation exposure of all individuals admitted to the Nuclear Science Center will comply with the limits set forth in 10CFR20. 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.

B. Meteorology

1. Survey of Yearly Weather Cycles

The Bryan-College Station area is located approximately 100 miles inland from the Texas Gulf Coast. The local weather is determined to a great extent by the high pressure areas which are predominant over the Gulf of Mexico. As a result of this condition, warm southeasterly winds occur a large majority of the time on an annual basis (Figure 2-3). Average annual rainfall is 30-35 inches. Snow occurs only rarely and sub-freezing temperatures are encountered infrequently for brief periods during the winter.

2. Seasonal Wind Characteristics

The passage of frontal systems is normally accompanied by northwest winds as shown in the winter wind rose diagram which is shown in Figure 2-3. Calms occur an average of 10% of the time, and wind speeds above 21 knots are seldom encountered.

FIGURE 2-1 NUCLEAR SCIENCE CENTER REGIONAL MAP

FIGURE 2-2 NUCLEAR SCIENCE CENTER SITE PLAN

Tornadoes are fairly common in Texas. Data on tornado frequency between 1950 and 1976 indicates that 17 tornadoes were reported within a 25 nautical mile radius of College Station.¹ The mean path length is 2.24 miles, and the mean path area is .23 square miles. The months of April and May have had the most occurrences of tornadoes over this period with the greatest probability of appearance being in the afternoon hours from about 2:00 to 7:00 p.m. The season usually starts in March and reaches a peak in May. A study of the movement of tornadoes indicates that a tornado will have 6% probability of having a westerly component in its direction of movement and a large percentage of these will move to the northwest.

The reactor building is designed to withstand 30 psi over pressure with the exception of the domed roof. It will withstand only 50 psf or .34 psi. In case a tornado passed nearby, the roof would probably act as a pressure relief mechanism. The basic steel structure in the roof would probably remain intact unless direct contact was made on the building by the tornado. The building is designed to withstand a straight wind of 90 mph velocity. The reinforced concrete construction and round shape of the building provide a considerable strength to withstand high winds.

3. Tornado Warning System

In the event a tornado is sited or detected within a 5 mile radius of TAMU, the NSC or the first available person on the NSC emergency notification roster will be notified by the radio operator at the TAMU Communications Center. 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.

C. Hydrology and Geology

1. Surface Geology

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

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

2. Subsurface Geology

Ground water is not expected to present any problems. The Nuclear Science Center is constructed on a formation known as the Easterwood Shale. The thickness of the formation is from 10-300 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.

¹National Severe Storms Forecast Center, Kansas City, Missouri, 1976.

FIGURE 2-3 AVERAGE WIND FREQUENCY DISTRIBUTION AT THE NUCLEAR SCIENCE CENTER

3. Geological Location

The reactor site is located in a geological region known as the Gulf Coastal Plain. The nearest fault zone (now dormant) is 100 miles to the west. This fault zone, known locally as the Balcones Escarpment, is the western boundary of the Gulf Coastal Plain.

The information on Hydrology and Geology was obtained from the staff of the Department of Geology, Texas A&M University.

D. Seismology

1. Earthquake Frequency

Texas lies in a region of minor seismic activity. Extreme West Texas, over 600 miles west of College Station is nearest 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. Only one earthquake of any significance has ever been recorded in Texas. This shock was at 30.6 N and 104.2 W on August 16, 1931. This occurred near El Paso in extreme West Texas and was a Class C (6.4) shock.²

2. Building Seismic Design

It is well known that a reinforced concrete structure provides good protection against earthquakes. The Nuclear Science Center wall structure and reactor pool walls are heavily reinforced, so it is anticipated that the building would withstand any minor shock that might occur.

²Seismicity of the Earth, B. Gutenberg and C. F. Richter.

III. REACTOR DESIGN

A. General Summary

The Nuclear Science Center Reactor (NSCR) operated from 1962 until 1967 with MTR-type curved aluminum plate elements. During this time the reactor was operated extensively at a maximum power level at 100 kW. In 1968 the reactor was converted to TRIGA fuel and the power level was increased to 1,000 kW.^{3,4} The initial core loading was quite satisfactory, but the experimental capability was soon affected by fuel burnup and samarium buildup.⁵ To restore excess reactivity, additional fuel was periodically added to the core and a graphite reflector was added to all faces. This eventually led to a 126-element core with a resultant decrease in the flux of almost 40% and the elimination of most of the irradiation facilities.

In August, 1970, fuel followed control rods were installed to gain excess reactivity and help 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.⁶ An average \$1.10 increase per fueled follower was achieved which extended the core life nearly 2 years. The high fuel burnup rate of standard TRIGA cores continued to be an operational problem for the NSCR. The NSCR has operated at a burnup of approximately 100 MW-days per year since 1969.

It was obvious that a solution was needed that could fit within the constraints of a university budget and limited federal support. A complete replacement of the core with new fuel was not seriously considered because of the considerable expense involved and the very short effective life of a standard core. Cycling new fuel into the core was no more attractive, since only small reactivity increases could be obtained with a reasonable amount of fuel since the average core burnup was only 10%. The solution to the problem was found in a new fuel developed and marketed by General Atomic.⁷ It is almost identical to the standard TRIGA fuel except that the enrichment was increased from 20% to 70%. The hydrogen to zirconium ratio was decreased from approximately 1.7 to 1.6, and 1.5 weight percent natural erbium was added as a burnable poison. The fuel designated as FLIP (Fuel Life Improvement Program) has a calculated lifetime of approximately 9 MW-years. This contrasts with experience for a standard core, where it was possible to operate only 6 months (approximately 1/2 of a MW-year) without a fuel addition.

³J.D. Randall, "Power Upgrading Experience Following Conversion of a Pool Reactor From Plate-Type to TRIGA Fuel Elements", Nuclear Safety, Vol. 10, No. 6, December 1969.

⁴W.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.

⁵D.R. Schad & J.D. Randall, "Operational Reactivity Considerations of the Texas A&M TRIGA", TRIGA Seminar, Denver, Colorado, 1970.

⁶D.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.

⁷F. 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.

Inasmuch as funds for a complete FLIP core were not available, it was necessary to initially 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.⁸ Calculations were performed at Texas A&M which led to the conclusion that satisfactory core arrangements were possible with a mixed core.⁹ As funds became available, the amount of FLIP fuel could be increased until a complete FLIP loading was achieved. This concept provides the additional advantage of producing substantially greater burnup in the standard fuel which is used during operation of the mixed core.

Sufficient funds for a partial loading of FLIP fuel were obtained and a 98-element core with a 35-element FLIP region was loaded. Criticality was achieved in July, 1973.¹⁰ The burnup data indicated that the burnup rate was initially 0.5¢ per MW-day and after samarium buildup the rate dropped to 0.2¢ per MW-day. Thus the incorporation of FLIP fuel had increased the lifetime of the core by a factor of three.

The NSCR has operated with two mixed core loadings containing 35 FLIP and 59 FLIP elements each since initial approval was granted in June 1973.

Beginning in 1979, all of the standard fuel had been replaced and Cores V-VIII have operated with FLIP fuel only. Although operation with FLIP fuel has been satisfactory, the U-235 enrichment of this fuel is 70%. To discourage nuclear weapons proliferation, the U.S. Department of Energy has embarked on a program to replace the highly enriched fuel in use at certain research reactors with fuel having an enrichment of 20%. As part of this program, General Atomic has produced TRIGA reactor type fuel rods using a lower enrichment. This fuel consists of the U-Zr-H (with erbium poison) type fuel similar to that used in FLIP elements but contains 20 weight percent uranium with a U-235 enrichment of 20%. It is designated as LEU 20-20 fuel. Outside fuel rod dimensions for FLIP and LEU fuel are identical. It is expected that the NSCR core will be converted to a full loading of fresh LEU fuel, including control rod followers, in the near future.

This safety analysis report presents the results and calculations performed in order to support the license modifications necessary to accomplish conversion of the NSCR core to GA LEU 20-20 fuel. Analyses have been performed to specify the particular erbium loading needed by the NSCR to achieve satisfactory excess reactivity control and cycle length. Operations with an LEU core during steady state, pulsing and accident conditions have also been analyzed to determine safety margins. Operation with mixed core loadings of FLIP and LEU fuel is not planned.

⁸G.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.

⁹D.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.

¹⁰D.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.

B. Mechanical Design

1. Reactor Bridge

The reactor core, the control rod drives, the ion chamber canisters, and the diffuser system are supported by a bridge that spans the reactor pool. Mounted on four wheels, the bridge travels on rails provided at the sides of the pool; thus, the reactor can be moved from one operating position to another. The bridge is hand operated and its speed of travel is limited due to the large gear ratios involved. Electric power, control-circuit wiring, and compressed air are supplied to the bridge. Slack for the bridge movement is provided by cable that lies in a covered trough which is parallel to the reactor pool.

Quick disconnect valves are mounted just below the grating on the upper research level at each end of the pool to facilitate easy movement of the reactor from the stall position to the large pool section or irradiation cell operating position. Flexible quick disconnect hoses have been installed for the diffuser system, air to the transient rod, pool water sampling line, pneumatic system lines, and the fission product monitor system.

An adjustable frame on the west side of the bridge called the bridge yoke serves as the mounting for the reactor suspension system. The yoke is raised or lowered mechanically using a large crank wheel and jack mechanism. The bridge yoke was modified prior to the installation of the transient rod drive assembly by adding a 7" I beam to the yoke frame. The I beam was installed to insure that the reactor frame could support the additional weight of the transient rod mechanism.

2. Reactor Support System

The reactor grid plate, shown in Figure 3-1 is welded to an aluminum suspension frame. The suspension frame is a welded structure of 3/8" x 2" x 2" aluminum angle. The west side of the frame is open toward the large section of the pool. The angle construction allows unrestricted flow of the cooling water. An aluminum stabilizer frame is bolted to the bottom of the grid plate 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 the core to be lowered until it bottoms to prevent sway, which could introduce annoying reactivity variations. This is accomplished using the jack mechanism which links the suspension frame to the reactor bridge and permits approximately a 6 inch vertical adjustment of the core position.

During the conversion to TRIGA fuel, 1/4" stainless steel pins were inserted through the aluminum frame into the grid plate on all four corners. This additional support was provided since the TRIGA fuel elements are considerably heavier than the aluminum plate type fuel elements. Since LEU rods have approximately the same weight as FLIP fuel, no additional support will be required for the core consisting of LEU fuel.

3. Reactor Grid Plate

Fuel elements and control rods are contained in bundle assemblies which are positioned and supported by a grid plate containing a 9 x 6 array of 54 holes (Figure 3-1). A reactor core loading could have several options for location in the grid plate. A typical core loading containing 90 elements and graphite reflectors is shown in Figure 3-2. In this loading the A row of the grid plate is available for positioning experiments. To accommodate a fuel followed control rod a 1 3/4"

diameter clearance hole through the grid plate is used to allow passage of the fueled section of the rod (Figure 3-3). A set of twelve clearance holes were drilled with each one being offset to be 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 has been installed beneath the reactor grid plate. Its purpose is to stop a control rod follower 2 inches below its normal down position should it become detached from its mounting.

4. Fuel Elements

The NSCR currently utilizes General Atomics TRIGA FLIP type fuel moderator elements. With the LEU conversion, TRIGA LEU type fuel moderator elements will be loaded. In both types of rods a zirconium hydride moderator is homogeneously combined with partially enriched uranium fuel. The major difference in the fuels is the atomic percentages of the constituent materials. This results in a difference in the expected burnups of the two fuel types. The expected fuel life of FLIP fuel has been stated to be 3000 MWD, although calculations support burnup as high as 4600 MWD. The expected life of a core composed of LEU fuel is 1300 MWD, approximately one third that of a core using FLIP fuel.

TRIGA LEU fuel elements have a 15 inch long active fuel section. These elements contain approximately 20 weight % uranium enriched to 20% in ^{235}U homogeneously mixed with ZrH, with approximately 0.59 weight % erbium as a burnable poison. This erbium content has been specified according to the shutdown reactivity requirements of the NSCR core configuration. Principal design parameters are shown in Table I.

To facilitate hydriding, an 0.18 inch diameter hole is drilled through the center of the active section; a zirconium rod is inserted in this hole after hydriding is complete. As shown in Figure 3-4, graphite slugs, 3 1/2 inches in length, act as top and bottom reflectors. The active fuel section and top and bottom graphite slugs are contained in a 0.020-inch thick stainless steel can. The stainless steel can is welded to the top and bottom end fittings. The approximate over-all weight of the rod is 7 pounds with an average ^{235}U content of about 99 grams. Serial numbers on the bottom end fittings are used to identify individual fuel rods.

Table I
PRINCIPAL FUEL ELEMENT DESIGN PARAMETERS

	LEU	FLIP
Fuel-moderator material	U-ZrH	U-ZrH
Uranium content	20 Wt-%	8.5 Wt-%
U-235 enrichment	20%	70%
U-235 content (avg.) per element	.99 g	123 g
Burnable poison	natural erbium	natural erbium
Erbium content	0.59 wt-%	1.5 wt-%
Shape	cylindrical	cylindrical
Length of fuel meat	15 in.	15 in
Diameter of fuel meat	1.371 in.	1.371 in
Cladding material	Type 304 SS	Type 304 SS
Cladding thickness	0.020 in.	0.020 in

Specially fabricated instrumented fuel elements containing three thermocouples embedded in the fuel, are used to measure fuel temperature during reactor operation. As shown in Figure 3-5, the sensing tips of the fuel rod thermocouples are located half-way from the vertical center line at the center of the fuel section and 1 inch above and below the vertical centerline. The thermocouple leadout wires pass through a seal contained in a stainless steel tube welded to the upper end fixture. This tube projects about 3 inches above the upper end of the element and is extended by tubing connected by swagelok unions to provide a watertight conduit carrying the leadout wires above the water surface in the reactor pool. The instrumented fuel rod is handled using the watertight conduit.

5. Fuel Assemblies (Fuel Bundles)

The NSCR fuel elements are assembled in two, three or four rod assemblies referred to as fuel bundles. This provides a simple means for converting MTR-type reactors to the use of TRIGA fuel. The four rod TRIGA fuel assembly is shown in Figure 3-4. Three rod bundles are shown in Figure 3-6 and 3-7.

The four rod assembly consists of an aluminum bottom adapter, four stainless steel clad TRIGA fuel rods, and an aluminum top fitting which serves as a handle. The bottom adapter fits into the NSCR grid plate, and the top of the adapter contains four tapped holes into which the fuel rods are threaded. The bottom fitting on the fuel rod is provided with a flange at the base of the threads so that the fuel rod seats firmly on the adapter and is rigidly supported in cantilever fashion. The details of the four rod assembly are shown in Figure 3-4.

Two rod bundles are identical to four rod bundles except that they contain only two fuel rods. Two rod bundles are used to produce thermal neutron flux traps (regions in which the thermal neutron flux is increased) by replacing fuel with water. Two rod bundles are positioned along the core's periphery in order to enhance the neutron flux available at certain specimen irradiation locations. The nonfuel section of a two rod bundle may also be used to contain an experiment.

A three rod fuel assembly may accommodate a control rod or instrumented fuel element or an experiment. The NSCR utilizes two separate types of three rod fuel assemblies for housing control rods. The first, shown in Figure 3-6, permits one fuel rod in an assembly to be replaced by a control rod guide tube which has an outside diameter of 1 1/2 inches. The handles on control rod elements are modified to accommodate the guide tube. A regulating rod and a transient rod without a follower will utilize this type of assembly. The second type, shown in Figure 3-7, is designed for use with shim safety control rods which are fuel followed. The transient rod which has a follower uses a specially designed control rod guide tube and must also have a base assembly as shown in Figure 3-7. The instrumented fuel rod fits into the bottom adapter. Not being an integral part of the bundle, the instrumented fuel rod is positioned into the bundle after it is in the grid plate.

Since the fuel follower must pass through the fuel element base, it was necessary to design a new base. This base serves as a guide for the control rod portion extending through the grid plate. The adjacent three fuel rods of the assembly are screwed into the base. Figure 3-8 is a plan view of 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. Figure 3-9 illustrates a similar plan view of the positioning of the top handle at a fueled follower assembly.

6. Control Rod Elements

Six motor-driven control rods (4 shim-safety rods, a regulating rod, and a transient rod) control the reactor during steady-state operation. Typical control rod positions are shown in Figure 3-2. The shim-safety control rods have scram capability and shall contain either borated graphite, B₄C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding. The shim safety control rods consist of a 35.56 cm (14 in.) long poison region with a diameter of 1.69545 cm on top of a 38.10 cm (15 in.) fueled region with a composition identical to the fuel rods, but with a fuel diameter of 1.69545 cm. These are capped on both ends by void regions. The regulating rod consists of a 38.10 cm (15 in.) poison region with a diameter of 1.51638 cm. The transient rod has a 38.10 cm (15 in.) poison region on top of a 50.80 cm (20 in.) void section. Both of these sections have diameters of 1.51638 cm. The shim safety rods and the transient rod have scram capability. The regulating rod does not.

TRIGA control rods without fuel followers will operate within a guide tube that surrounds the rod and has holes for proper cooling (Figure 3-6). A fueled follower control rod can not operate in a guide tube because of restrictive cooling and will be assembled in a three rod fuel assembly as shown in Figure 3-7. Hold-down devices are used for control rods with fueled followers. In the absence of a guide tube, a hold-down foot (Figure 3-7) is designed to fit over the top handle cross bar and the blade of the foot extends high enough for clearance of the rod when it is in the full up position. The blade is attached to the side of a tube that houses the control rod extension. When the rod drive unit is secured to the reactor support structure, a 1/8" clearance is provided between the foot and fuel element top handle cross bar. This clearance permits small thermal expansion of the fuel without vertical restriction.

FIGURE 3-1 MODIFIED NSCR GRID PLATE

Figure 3-2 Flip Core Configuration

FIGURE 3-3 FUELED FOLLOWER INSTALLATION

FIGURE 3-4 FOUR ROD FUEL ELEMENT ASSEMBLY FOR THE NSCR

FIGURE 3-5 THE INSTRUMENTED FUEL ROD

FIGURE 3-6 A CONTROL- BUNDLE WITH INSTALLED
CONTROL ROD GUIDE TUBE

FIGURE 3-7 FUELED FOLLOWER - FUEL BUNDLE ASSEMBLY

FIGURE 3-8 FUELED FOLLOWER IN FUEL ELEMENT BASE
AND ITS RELATION TO ADJACENT BUNDLES

FIGURE 3-9 FUELED FOLLOWER IN FUEL ELEMENT TOP HANDLE
AND ITS RELATION TO ADJACENT BUNDLES

FIGURE 3-10 NSCR FUELED FOLLOWER CONTROL ROD

7. Control Rod Drive Systems

The shim-safety and regulating control rod systems consist of an electromechanical rod drive, a control unit, an offset and hold-down assembly and the control rod. The transient (pulse) rod system consists of an electromechanical-pneumatic rod drive assembly, a control unit, a hold-down assembly, an air supply system, and the transient rod. The control rod drives are mounted to the reactor frame structure above water level and are coupled to the control rod assembly. The shim safety control rods are connected to the rod drive by an electromagnet and armature, whereas the regulating rod is mechanically connected to the rod drive. High pressure air acting on a piston holds the transient rod against the rod drive assembly. In pulsing applications the transient rod is rapidly withdrawn using high pressure air. An offset and hold-down assembly permits installation of shim safety rods or the regulating rod in nine optional core positions. The pulse rod can be installed in only two core positions and is not offset.

The shim safety rod drives and regulating rod drive used in the NSCR are manufactured by Diamond Poker Speciality and are extremely compact. See Figure 3-11. The electromechanical portion of each drive unit is housed within a single 3" aluminum tube. An electric motor drives a lead screw through a gear reducer for movement of the control rod. Withdrawal speed for the shim-safety rod drive is 11.4 cm/min.

A gear driven synchronous transmitter is also connected to the motor on the other end of the shaft. Synchronous receivers located at the reactor console indicate rod position. The rod drive mechanism is coupled to the control rod by an electromagnet. A reactor scram is achieved when the current to the electromagnet is interrupted and the control rods fall into the core. The regulating rod drive unit is directly coupled to the control rod extension and has no scram capability.

The control rod control unit (Figure 3-12) which is located in the reactor console is electrically connected to the rod drive for operation and indication of the system. This unit has push buttons for up or down rod movement, and indicator lights are provided for magnet engagement, carriage up, carriage down, rod down, and jam conditions. A digital readout is provided for percent rod movement and can be read to 0.1%. A gang switch is located on the reactor console for simultaneous operation of the shim safety control rods.

The shim-safety control rod magnet and armature and rod assembly dampening device are located within the control rod barrel attached to the control rod drive unit as shown in Figure 3-13. 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 the clearance in the bottom of the control rod barrel. When the piston enters the piston receiver the clearance is reduced and maximum dampening of the rod occurs. Stainless steel and aluminum components are used to provide smooth movement and reduced wear of sliding parts.

The regulating rod control assembly is shown in Figure 3-14. The barrel contains a lower guide piece with no piston action since the control extension is bolted to the rod drive lead screw and does not scram. Withdrawal speed of the regulating rod is 24.4 cm/min.

The shim-safety control rods can be mounted collinearly with the control rod drive central axis or offset to align with eight optional surrounding control rod positions as shown in Figure 3-15. The offset assembly functions similarly to the bolt action of a rifle. Spacers which separate the two piston rods are retained in slots which allow vertical movement but restrict lateral movement. The offset barrel can be positioned in 450 increments by rotation about the end of the control rod barrel to which it is attached (Figure 3-16). The hold-down assembly provides a means to enclose the

FIGURE 3-II CONTROL ROD DRIVE

FIGURE 3-12 SHIM-SAFETY CONTROL UNIT

FIGURE 3-13 SHIM-SAFETY ARMATURE AND DAMPENING DEVICE ASSEMBLY

FIGURE 3-14 REGULATING CONTROL ROD ASSEMBLY

LEGEND

⊗ POISON AND DRIVE

⊗ POSSIBLE POISON LOCATIONS

FIGURE 3-15 CONTROL ROD OFFSET ORIENTATION

FIGURE 3-16 CONTROL ROD OFFSET AND HOLD DOWN ASSEMBLY

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 or the cross bar of the fuel bundle.

The control rods are mounted to the upper portion of the reactor frame structure by a horizontal plate with machined slots and clamps to hold the rod drives in position (Figure 3-17). A support ring which holds the shim-safety control rod assembly is attached to the control rod mounting plate by threaded aluminum rods. This assembly permits removal of the associated control drives for maintenance without moving the associated control rod from the core. The regulating rod assembly is fastened to the mounting plate by a clamp ring with a set screw which penetrates the wall of the drive unit. When maintenance is required for the regulating rod, it is removed from the reactor core and disassembled as required. The installation of a shim-safety control rod is shown in Figure 3-18.

The transient rod drive unit is a pneumatic-electromechanical rod drive, a standard system supplied by General Atomic (Figures 3-19, 3-20). The pneumatic portion of the transient rod drive is basically a single acting pneumatic cylinder. A piston within the cylinder is attached 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 is admitted at the lower end of the cylinder to drive the piston upward. As the piston rises, the air being compressed above the piston is forced out through vents at the upper end of the cylinder. During the final inch of travel, the rod is smoothly decelerated by a shock absorber which minimizes rod vibration 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 transient rod drive. A three-way solenoid valve controls the air supplied to the pneumatic cylinder. 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. With this operating feature, the transient rod remains in the full down position except when air is supplied to the cylinder.

The cylinder drive portion of the transient rod system consists of an electric motor, a ball-nut drive assembly, and the externally threaded air cylinder. Withdrawal speed of the transient rod is 63.3 cm/min. During electromechanical operation of the transient rod, the threaded section of the air cylinder acts as a screw in the ball nut drive assembly. The ball-nut assembly is in turn connected through a worm-gear drive to an electric motor. The cylinder may be raised or lowered independently of the piston and control rod by means of the electric drive. Adjustment of the position of the cylinder controls the upper limit of piston travel, and hence controls the amount of reactivity inserted for a pulse.

8. Reflector Elements

Reflectors, excluding experiments and experimental facilities, shall be water or a combination of graphite and water. At the time of installation of the fueled followers, new graphite elements were also added. In the past, 3" x 3" graphite blocks were attached to grid plugs and positioned in the core for use as reflectors. The assemblies did not always stand straight when in a grid plate position and on occasion, two elements would bind and were difficult to remove. A new design was devised where the ends of the graphite elements were machined to fit flush against a machined spacer. This permitted some degree of error in the alignment of threads in the graphite yet produced an aligned vertical assembly.

FIGURE 3-17 CONTROL ROD ASSEMBLY SUPPORT MECHANISM

FIGURE 3-18 CONTROL ROD INSTALLATION

FIGURE 3-19 SCHEMATIC DRAWING OF TRANSIENT-ROD DRIVE

FIGURE 3-20 PNEUMATIC-ELECTROMECHANICAL TRANSIENT-ROD DRIVE

C. Nuclear Design

1. TRIGA - FLIP and LEU Cores

The design and operating characteristics of standard TRIGA cores are well known as is the inherent safety characteristics of this class of reactors.¹¹

The pulsing of TRIGA-FLIP cores at the present pulsing limit of \$2.35 results in very safe conditions because earlier operational NSCR cores were regularly pulsed at \$3.00 insertions. The pulsing characteristics for a NSCR operational TRIGA-FLIP core are shown in Figure 3-21. The pulsing analysis of LEU cores indicates performance similar to that of FLIP cores.

In TRIGA FLIP and LEU fuel the temperature hardened thermal neutron spectrum is used to decrease the reactivity through its interaction with erbium, a low energy resonance material. Thus erbium, with its double resonance at ~0.5 eV, is used in the fuel both as a burnable poison and as a material to enhance the prompt negative temperature coefficient. The neutron spectrum shift pushes more of the thermal neutrons into the ¹⁶⁷Er resonance as the fuel temperature increases. For TRIGA LEU fuel, the ²³⁵U loading is less than that of TRIGA FLIP fuel, resulting in an increased concentration of ²³⁸U. This allows the generation of a similar prompt negative coefficient of reactivity with a lower erbium concentration since the negative effects of the higher ²³⁸U loading will be felt. Another important design feature of TRIGA fuel is the ZrH moderator present in the fuel structure. Neutrons are moderated in the fuel rod in which they are generated. As the fuel temperature increases, the density of the material decreases, resulting in a negative feedback effect.

The calculated temperature coefficients are shown in Figure 3-31 for both FLIP and LEU cores. The temperature dependent character of the temperature coefficient of a TRIGA FLIP or LEU core is advantageous in that an acceptable reactivity loss is incurred in reaching normal operating temperatures, but any further increases in the average core temperature result in a sizably increased prompt negative temperature coefficient to act as a shutdown mechanism. The burnup calculations indicate that after 3000 MWD of operation for FLIP fuel, the ²³⁵U concentration averaged over the core is ~67% and the ¹⁶⁷Er concentration is ~33% of the beginning-of-life values. For LEU fuel at 1300 MWD, the ²³⁵U concentration is ~47% and the ¹⁶⁷Er concentration is ~11% of the beginning-of-life values. The end-of-life coefficient for both fuel types is less temperature dependent than the beginning-of-life coefficient because of the sizable loss of ¹⁶⁷Er and the resulting increased transparency of the ~0.5 eV resonance region to thermal neutrons.

The effects of the temperature coefficient for FLIP and LEU cores upon the pulse shape at beginning of life is demonstrated in Figure 3-22. Note that the LEU pulse peaks at a power higher than that of the FLIP fuel; thus, higher fuel temperatures will result during the LEU pulse as shown in Figure 3-23. This relationship between the two fuel types will change as the core life progresses. By the end of core life the FLIP pulse will generate higher temperatures than the LEU pulse. This behavior is shown in Figures 3-24 and 3-25.

Extensive measurements were made of the various parameters relating to the pulsing operation of the General Atomic Prototype reactor. The most important of these are given below for step insertions of reactivity up to 2.1% $\delta k/k$ (\$3.00).

¹¹ GA - 3886 (Rev A) TRIGA Mark III Reactor Hazards Analysis, Feb. 1965.

Figure 3-21 FLIP Pulsing Characteristics

Prompt Reactivity Insertion (\$)

Figure 3-22 Comparison of FLIP and LEU Pulses

For \$1.80 Insertion, BOL

Figure 3-23 Pulsing Temperatures

For \$1.80 Insertion, BOL

Figure 3-24 Comparison of FLIP and LEU Pulses

For \$1.80 Insertion, EOL

Figure 3-25 Pulsing Temperatures

For \$1.80 Insertion, EOL

During pulsing operation, the reactor is placed in a super-prompt-critical condition. The asymptotic period is inversely related to the prompt reactivity insertion. Figure 3-26 shows the results of plotting the reciprocal of the measured period versus the prompt reactivity insertion. As can be seen, the minimum period obtained for reactivity insertions of \$3.00 (\$2.00 prompt) is ~ 3 msec. Also shown in Figure 3-26 is a plot of the reciprocal of the measured width at half maximum power versus prompt reactivity insertion.

Figures 3-27, 3-28, and 3-29 show the interrelationship between maximum transient power, pulse widths, and period. When considered together, these plots serve to describe the general features of the TRIGA Mark III core performance in the pulsing mode. For a given core configuration, the peak power, integral power in the prompt burst, and width of the pulse are determined by the amount of reactivity inserted. It can be seen from the plots that the peak power is controllable over a rather wide range since this parameter is very nearly proportional to $(\delta k/k - \$1.00)$. Pulse width and integral powers, on the other hand, are approximately linear functions for reactivity insertions above prompt critical so that their range is more limited.

The NSCR will operate a core composed of TRIGA LEU fuel elements (Figure 3-2). The LEU fuel was designed to have essentially the same properties as TRIGA FLIP fuel which is currently used in the NSCR.

The investigation of the design of a core composed of this type of fuel required the use of a computer code capable of producing all the necessary design parameters during both steady state operation and during accident scenarios. Accuracy of computations was of the utmost importance. Accuracy was assured by performing benchmark calculations and by comparing results for the LEU core to those for the FLIP core. Machine run time and storage capabilities were only of secondary importance since a powerful machine was available for exclusive use in performing the licensing calculations.

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

Once a temperature and burnup dependent library of homogenized cross sections had been generated, it was used in a three dimensional neutron diffusion calculation that modeled the core, the reflector and the irradiation facilities. The code selected for these calculations was DIF3D. This code is capable of computing the k_{eff} of the core for different fuel and control rod configurations, as well as the spatially dependent power and neutron fluxes for each energy group. Different control rod positions and fuel loading arrangements can be investigated, and the prompt temperature coefficients of reactivity necessary for the transient calculations can be generated on a full core calculational basis.

To prove the accuracy of these codes for modeling the NSCR, a test case run by General Atomics on a two megawatt TRIGA core was recreated.¹² This test case was chosen for a

¹²General Atomics, "Generic Enrichment Reduction Calculations for Rod-Type Reactors", Research Reactor Core Conversion From the Use of Highly Enriched Uranium to the Use of

Figure 3-26 Inverse Period and Inverse Width at Half Maximum Power Versus Prompt Reactivity Insertion - Prototype Reactor

Figure 3-27 Full Width at Half Peak Power Versus Period - Prototype Reactor

Figure 3-28 Peak Power Versus Full Width at Half Peak Power - Prototype Reactor

Figure 3-29 Peak Power Versus Reactor Period - Prototype Reactor

benchmark calculation because it models a core containing fuel similar to the TRIGA-LEU fuel proposed for use at the NSCR. Both the fuel used in the NSCR and the fuel used in the test case have essentially the same properties for each element. The major difference is the addition of a shroud around each four rod cluster in the GA test case. Also, the GA test case core was modeled at a power level of 2 MW while the NSCR core will operate at 1 MW, but this does not adversely affect the validity of the test case. A summary of the core parameters for this test case is shown in Table II.

Table II
Core Parameters of GA Test Case

Fuel cluster: TRIGA-LEU 20 wt% U in UZrH (76 x 80 x 508 mm)
 Fuel rods per cluster:
 Standard Cluster: 4
 Control Cluster: 3
 Nominal fuel rod dimensions:
 Fuel O.D.: 32.4 mm
 Clad O.D.: 33.5 mm (incoloy)
 Fuel Height: 508 mm
 Fuel loading: 548 mm U (20% enriched)/rod
 2.2 Kg U (20% enriched)/std cluster
 440 gm U-235/std cluster
 ~0.59 wt% Erbium as burnable absorber
 Number of fuel clusters in the core: 26±1
 Standard clusters: 21
 Control clusters: 5±1
 Reflector: Water
 Core size (liters): 78±2
 U-235 content/core (Kg): 10.6
 Core geometry: 4 x 6 arrangement
 Grid plate: 6 x 9 positions (normal conversion)
 Burnup Status of the core: equilibrium core
 Average core burnup (%): ~20
 Thermal-hydraulic data:
 Average power density (kW/liter): 26
 Coolant flow rate: 1000 gpm (227 m³/hr)
 Core inlet temperature: 38°C

One set of parameters that were generated and reported by GA for the test case was the peaking factors. These are listed below in Table III along with the corresponding values calculated at the NSC using WIMSd4/m and DIF3D.

Table III
Peaking Factors From GA Test Case

Type of Peaking	$\hat{P}/\bar{P}$ General Atomics	$\hat{P}/\bar{P}$ TAMU 0 MWD	$\hat{P}/\bar{P}$ TAMU 900 MWD
Core Radial	1.57	1.61	1.62
Core Axial	1.36	1.30	1.31
1D Cell (23°C)	1.48	1.41	1.29
1D Cell (310°C)	1.52	1.44	1.31
1D Cell (700°C)	1.61	1.49	1.34

The core radial and core axial values agree quite well, but there is a small discrepancy in the 1D cell values. The GA test case is supposedly run at a burnup of approximately 20% of the core life which corresponded to approximately 900 MWD. It is possible, however, that the 1D cell peaking factors presented in the document were calculated with no burnup on the core. This would put them in better agreement with the TAMU values.

Below is a comparison of the peak thermal flux values in the core and the water reflector. To find the thermal flux, the values of groups 5, 6, and 7 in the 7 group model were summed. These flux values are in close agreement.

Table IV

Estimated Peak Thermal Flux at 2 MW 4-Rod Cluster TRIGA-LEU Reactor

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

The final parameter that GA generated in their report was the prompt negative temperature coefficient. Figure 3-30 shows the Texas A&M values along with the GA values for this parameter.

Figure 3-30

These are the only parameters which could be verified against General Atomics data for the benchmark case, but they show that the TAMU models are capable of producing accurate results as

compared to the GA approved models. DIF3D is also capable of producing output in much more detail than was shown here. With the aid of an external plotting program (TECPLOT), three dimensional flux maps have been generated which will prove useful in future operations at the NSCR.

A steady state thermal hydraulic analysis was also performed based on the power distribution obtained from DIF3D in order to determine the maximum fuel and clad temperature during operation. To perform this steady state analysis, the code NCTRIGA was used. NCTRIGA is a one dimensional thermal hydraulic code that calculates temperatures at several nodes along a single fuel rod channel and determines the natural convection induced coolant flow rate. This code uses the power distribution generated from the neutronic analysis, along with information on the geometry and material composition of the flow channel, to produce temperatures in the fuel and to predict the coolant flow rate.

To perform calculations to determine peak fuel and clad temperatures during normal pulsing and accident transients for the both the FLIP and LEU cores, the code PARET from Argonne National Laboratory was used. This code is capable of performing 1-D radial heat transfer calculations under non steady state conditions at several axial nodes along a fuel rod. The initial temperatures and flow rates needed for input to PARET were generated using NCTRIGA. The temperature coefficients of reactivity needed by PARET were generated using DIF3D with temperature dependent cross sections taken from WIMSd4/m. These are shown in Figure 3-31 for FLIP and LEU fuels at various values of burnup. To acquire power distribution data for the (pulsing) PARET model, a DIF3D job was run at 300 watts steady state power with the transient rod fully inserted to obtain the initial power distribution. This data was compiled into a PARET input deck to produce a model for predicting performance of the peak fuel element and core during pulsing.

Figure 3-31

To benchmark this model, data was obtained from pulsing experiments on the Texas A&M Nuclear Science Center Reactor using FLIP fuel. When the peak pulsing powers were compared between the model and the experimental data, the following results were produced.

Figure 3-32

The comparison of the full width at half maximum of the pulsing peaks is shown below.

Figure 3-33

Both of the comparison of the peak energy and FWHM agree quite well. The next figure displays a comparison of the total energy generated during the pulse.

Figure 3-34

The mismatch at \$1.80 shown above is most likely because the experimental value is too small caused by the detector overloading due to the excessive amount of energy generated in this pulse.

The final comparison of PARET to experiment matched the peak temperature generated in the fuel during a pulse. These results are shown below.

Figure 3-35

This peak fuel temperature is difficult to quantify exactly due to the nature of the experimental data acquired. The experimental temperature is taken from thermocouple which is located mid-way between the fuel surface and the centerline of the instrumented element. Since the measured temperature falls between the predicted centerline and fuel surface temperatures, the results are accepted as reasonable. In addition, the calculated peak fuel temperature is considerably higher than the experimentally measured temperature.

To relate the thermocouple temperature to the predicted peak temperature in the element, a curve fit was used to develop a functional relationship for the ratio of the peak fuel temperature to peak thermocouple temperature as a function of peak centerline temperature. A plot of this for the data above is shown in figure 3-36.

Figure 3-36

When a second order polynomial in temperature is fit to this data the result is

$$R(T_{CL}) = 0.5066 + 4.046 \times 10^{-3} T_{CL} - 3.32667 \times 10^{-6} T_{CL}^2$$

where: $R(T_{CL})$ is the ratio of the peak centerline temperature to the peak thermocouple temperature

T_{CL} is the peak centerline temperature of the instrumented element as calculated by PARET.

Since the heat transfer properties of LEU fuel is similar to that of FLIP fuel, this ratio should hold for both fuel types.

2. LEU Cores

Using the methods described above, a LEU core was designed for the NSCR. This core has the following properties.

Table V
NSCR LEU Core Properties

Steady State Power Level:	1 MW
No. of Fuel Elements:	90
Critical Mass:	6,836 grams ²³⁵ U
Core Mass:	8,230 grams ²³⁵ U
Maximum Excess Reactivity:	\$3.43
Total Control Rod Worth:	\$16.91
SS1:	\$2.78
SS2:	\$1.85
SS3:	\$2.99
SS4:	\$5.10
RR:	\$1.39
TR:	\$2.80
Minimum Shutdown Margin:	\$0.91

A core map is shown below in Figure 3-37. All of the design parameters meet the requirements of the Technical Specifications for the NSCR. The shutdown margin is calculated by the procedure indicated in the Technical Specifications, and exceeds the required limit of \$0.25. A detailed description of the calculations used to determine the LEU core parameters in Table V can be found in Appendix I.

3. Neutron Startup Source

The reactor startup source will be either Sb-Be, Am-Be, or Po-Be. It is located in the core so that good multiplication data can be obtained on the startup channel. The source strength will be such that the startup channel will change by greater than 2 cps upon the insertion or removal of the source from the core at initial startup.

D. Thermal Design

The NSCR operates at 1 MW steady state and is cooled by natural convection and can be operated at any position along the pool center line except near the gateway between the stall and large pool section. The reactor core is constantly surrounded by pool water which is drawn in freely from the bottom and sides of the core during the convection cooling process.

Figure 3-37 LEU 20-20 Core Configuration

Referring to Figure 3-4, it can be noted that the four rod fuel element assembly has been designed to provide easy passage of cooling water through the element. Water flows by natural convection through the 2" diameter hole in the grid plate adapter. It passes through the large cruciform opening and then over the entire element until it leaves the core through the numerous openings in the aluminum handle. In addition to the coolant passages through the grid plate adapters, the NSCR grid plate has additional coolant holes 1/2" in diameter located at the corner of each four rod element.

Mark III standard fuel elements and FLIP elements have been successfully operated in TRIGA cores by General Atomic at steady state power levels of up to 1.5 Megawatts. The arrangement of fuel in the NSCR has been designed so that the minimum nominal spacing between the fuel rods provides adequate convection cooling of cores up to 2.0 MW operation. The nominal spacing of the fuel rods in the NSCR core is shown in Figure 3-38. Core cooling is considerably enhanced by this increased spacing and by the extra cooling holes at the corners of the bundle. Cooling of the NSCR is also improved due to the increased depth of pool. The NSCR core is normally covered by 26' of water which places it at a depth greater than that of most TRIGA installations. The resultant higher pressure will reduce nucleate bubble formation and improve convective heat transfer.

Figure 3-38 Nominal Fuel Rod Spacing in the NSCR Core

IV. REACTOR POOL AND WATER SYSTEMS

A. Reactor Pool

1. Pool Structure and Shield Design

The concrete pool structure and the pool water provide shielding for operation of the reactor. The shield was designed for 5 MW operation which is well above the proposed 1 MW TRIGA operation level. The movable reactor bridge permits operation of the reactor at any position on the pool center line running approximately east to west. The pool has two sections which are designated as the stall and the main pool (See Figure 4-1). These sections can be isolated using an aluminum gate then drained. The pool depth is 33 feet and it has a width of 18 feet in the main pool. The stall section is 9 feet across and has a 180° curved surface of 4.5 foot radius.

The upper 17 feet of the pool wall is constructed of standard concrete. The lower portion of the pool wall is constructed of barytes concrete and light concrete. Adjoining the main pool is a shielded structure designated as the "irradiation cell" (See Figure 4-1). This section may also be used for pool water storage. An irradiation window is located in the shield wall which separates the reactor pool and irradiation cell. The reactor can be positioned at desired distances from the window for irradiation of experiments in the cell.

2. Pool Liner

The reactor pool is lined with stainless steel for maximum water containment and water purity. The pool walls are lined with 10 gauge Type 304 stainless sheet and the floor is lined with 1/4 inch Type 304 stainless plate. All penetrations are stainless steel and are welded to the liner for water tightness. A drainage system is provided beneath the liner for collection of possible liner leakage. This leakage is drained through a 10" line into the sump of the valve pit and then handled as radioactive waste. Guide tracks for positioning the reactor on center line of the pool are located on the floor of the liner. The pool liner and the installation details are shown in Figure 4-2 and 4-3.

3. Pool Penetrations

- a. Experimental. Experimental penetrations consist of the thermal column, pneumatic tubes, beam ports, and the irradiation cell window. The removable ends of these penetrations have bolted flanges with mechanical seals for water tightness. Installation, modification, and maintenance of experiments using these penetrations can be performed by draining the appropriate pool section. Experimental pool penetrations are shown in Figure 4-1.

- b. Pool Water Piping. Three 10" water cooling lines are located on the floor of the reactor pool. A single 10" line located on the center line of the main pool is the inlet line to the pool water cooling system. Cooled water is returned to the pool using the 10" lines in the stall and main pool. Diffusers are installed on each of the two return lines. Two 2" drain lines, one on the floor of each pool section, terminate in the demineralizer room. These lines are used for drainage and recirculation. Two 3" demineralizer recirculation and fill lines are located near the top of the pool. Two 1 1/2" lines are provided at the top of the pool for operation of the pool surface skimmer system. Pool liner leakage is routed to the valve pit by a 10" line beneath the liner at the center of the stall section. A 3" drain line is provided on the floor of the irradiation cell. Pool water piping penetrations are also shown

SECTION B-B' REACTOR POSITION #2

SECTION A-A' REACTOR POSITION 71

FIGURE 4-1 NSC REACTOR POOL SECTIONS AND PENETRATIONS

FIGURE 4-2 REACTOR POOL LINER

POOL LINER FLOOR DETAIL

TRIM DETAIL

HAT CHANNEL DETAIL

GATE JAMB DETAIL

PENETRATION DETAIL

HORIZONTAL WELD DETAIL

FIGURE 4-3 REACTOR POOL LINER DETAILS

B. Water Systems Description and Operation

1. General

The various pool water systems accomplish heat removal, purification, recirculation, make-up of pool water, pool surface skimmer, pool water transfer and storage, and liquid waste disposal. These systems are:

- (1.) Pool water cooling system
- (2.) Pool water purification system
- (3.) Pool skimmer system
- (4.) Pool water transfer system
- (5.) Liquid waste disposal system
- (6.) Core diffuser system

Systems handling pool water are constructed of stainless steel, aluminum, and plastic components to maintain maximum pool purity. Welded piping systems with mechanical seals insure minimum leakage for operation of the pool water systems. The maximum operating water pressure occurs in the heat exchanger tubes. The maximum pressure in the other systems corresponds to the reactor pool depth of 33 feet. The maximum heat exchanger tube pressure of approximately 22 lb/in² is well below the design pressure of 150 psi for all systems. The capacity of the pool cooling system is sufficient for continuous 1 MW operation of the reactor. The convection cooled TRIGA core does not present a problem of fuel melt down and resultant fission product release when there is a complete loss of coolant. 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 also follows that loss of electrical power to all coolant systems would not result in a hazardous condition. Remote operation of the pumping components of the pool water systems is provided in the reactor control room. Figure 4-4 is a schematic of the pool water systems. The elevations of the water systems are shown in figure 4-5.

2. Pool Water Systems

a. Pool Water Cooling System. The reactor pool water is cooled using a primary-secondary heat exchange principle. The pool cooling system, shown in Figure 4-6, has a design capacity of 2 MW with nominal pool operating temperature of between 70°F to 80°F. A closed primary loop of reactor pool water is pumped through the tube side of the heat exchanger for cooling and then returned to the reactor pool. The primary loop has a design flow rate of 1,000 gpm. Heat removed from the primary is transferred to the secondary cooling loop. The secondary cooling water is pumped from the basin of the cooling tower through the shell side of the heat exchanger and returned to the cooling tower. Heat removed from the secondary loop is released to the atmosphere at the cooling tower. The secondary loop has a measured flow rate of 1575 gpm. The cooling tower is designed to deliver 83°F water at 78° wet bulb.

The primary loop is constructed of stainless steel components to preserve pool water purity during the cooling process. The tubes, tube sheet, and header of the heat exchanger are stainless steel and the shell is carbon steel. Design operating pressures of the heat exchanger are 22 lb/in² for the primary side and 17.5 lb/in² in the secondary. Components for the primary cooling loop are located in the cooling equipment room on the lower research level.

FIGURE 4-4 NSCR POOL WATER SYSTEMS

FIGURE 4-5 NSCR POOL WATER ELEVATION

The secondary loop is chemically treated to increase the life of the components and reduce scale deposits in the heat exchanger. A control system is provided to continuously monitor and control recirculating cooling water. The system performs its control function by actuating chemical pumps and a bleed valve in response to measured chemical characteristics of the water. Two unique failsafes, "Corrosion Interlock" and "Alarm" are provided to prevent any damage to the plant or piping from an acid runaway or other corrosive upset. The system is normally operated with all function switches in the "Auto" position where the sensitive controllers will automatically compensate for load or water changes to maintain pH, inhibitor level, and total dissolved solids at their desired levels.

The cooling system is normally operated from the reactor control room. "On-Off" switches are located at the pumps and cooling tower and in the reactor control room for operation and maintenance of the system. A multipoint recorder indicates system temperatures and a flow rate meter indicates primary flow. Alarms are provided in the control room for primary and secondary pump power failures. Valve operating positions are manually set prior to operation of the cooling system.

b. Purification System. Reactor pool water purity is maintained using a commercial, regenerative mixed bed demineralizer unit in conjunction with micron filters and activated charcoal and gravel filters. This system, shown in Figure 4-7, is located in the demineralizer room on the lower research level. The pool water recirculation cycle for cleanup operates at a design flow rate of 75 gpm with an output conductivity of less than 1 micromho per centimeter. Pool water makeup is provided by processing raw water through the demineralizer system. Operation of the purification system is performed in the demineralizer room. A remote "on-off" switch is located in the reactor control room for operation of the demineralizer recirculation pump. Regeneration of the demineralizer is performed manually as required.

c. Pool Surface Skimmer System. The pool surface skimmers are located at the west end of the main pool section. Floating skimmer heads are attached to a piping system which serves as the suction line to a 30 gpm pump. The water is pumped through a 3 micron filter bank and then returned to the reactor pool. An "on-off" pump switch is located at the pump and in the control room for operation of the system. Periodic operation of the system is sufficient to maintain a clean pool surface. The surface skimmer system is shown in Figure 4-8.

d. Pool Water Transfer System. This 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 for transfer of 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 water transfer-storage system are interconnected by a single three inch crossover line for flexibility of operation. The pool water transfer-storage system is shown in Figure 4-9.

e. Liquid Waste Disposal System. Liquid waste from the reactor building is collected in the hot waste sump located in the demineralizer room. Two 100 gpm sump pumps lift the liquid waste for storage in one 12,000 gallon and two 5,000 gallon tanks located on the northwest corner of the reactor site. The sump pump is located below the base elevation of the reactor pool. Normally only one storage tank is connected to the sump. Liquid waste from the pool liner and cooling equipment room is collected in the valve pit sump and transferred to the demineralizer sump. A motor driven stirrer is used to mix the storage tanks prior to sampling and draining. A raw water mixing station is located at the tanks for dilution of liquid waste release. The liquid waste disposal system is shown in Figure 4-7.

FIGURE 4-6 REACTOR POOL COOLING SYSTEM

WATER PURIFICATION SYSTEM

LIQUID WASTE DISPOSAL SYSTEM

FIGURE 4-7 WATER PURIFICATION AND WASTE DISPOSAL SYSTEMS

f. Core Diffuser System. The NSCR 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 which is caused by N^{16} and Ar^{41} produced in the coolant water as it passes through the core. The diffuser pump and associated piping is located in the mechanical chase as shown in Figure 4-10. Two outlets permit operation of the system when the reactor is in the large pool or stall section. A flexible, quick disconnect water hose is used to connect the bridge piping to the diffuser outlets. The system is operated from the control room using the start-stop switch located on the water systems control panel.

C. Inspection and Maintenance of Water Systems

Maintenance and inspection of the pool water systems are performed by qualified personnel under supervision of the reactor supervisor on duty. Routine inspections are performed using check sheets which are signed by the person performing the inspection and the reactor supervisor. A record is kept of these inspections. Proper operation of the pool cooling system is verified daily by the reactor operator using a check sheet.

FIGURE 4-8 POOL SKIMMER SYSTEM

FIGURE 4-9 POOL WATER TRANSFER SCHEMATIC

FIGURE 4-10 CORE DIFFUSER SYSTEM

V. CONFINEMENT SYSTEM

A. Confinement Structure Design

The reactor confinement building is a cylindrical steel reinforced concrete structure which is approximately 70 feet in diameter and 70 feet high. Approximately 55 feet of the structure is above grade. Confinement is achieved within the structure by gasket seals on all outside doors. These door seals allow a negative pressure to be maintained within the building through the use of an exhaust blower and fresh air inlet louvers. Three major floor levels exist within the confinement building (See Figure 5-1). Access within the confinement structure from one level to another is provided by a cylindrical stairwell which is adjacent to the primary building and is connected to each level of the building.

1. Upper Research Level

The upper research level, shown in Figure 5-2, is the largest, by volume, of the three levels. A two-ton electric hoist is suspended from the ceiling and is capable of servicing all open areas of the upper research level, including the reactor pool. The exterior walls of this level and those of the central mechanical chase are constructed of reinforced concrete slabs poured in place between concrete-encased steel columns. The columns and slabs are slanted approximately 6 degrees toward the center of the building and joined to a structural steel frame. The roof is surfaced with built-up roofing on a covered roof deck of poured gypsum construction. Access to the upper research level is through the main stairwell, a personnel door from the reception room, and a large truck door at the west end of the reactor pool. Surrounding the reactor pool are the reactor control room, men's and women's restrooms, a cold change room, the electronics shop, and a materials handling area. The roof for these rooms provides a floor for a mezzanine area, a portion of which is enclosed for office space.

2. Central Mechanical Chase

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

3. Lower Research Level

The lowest level of the confinement building which is shown in Figure 5-3 is the lower research level. The floor and outer walls of this level are constructed of reinforced concrete. Access to this level is provided by the main stairwell and the lower level ramp truck doors. The north truck door contains a smaller personnel entry door. Facilities located at this level are the cooling system equipment room, research laboratories, the demineralizer room, and a chemistry laboratory. The lower portion of the reactor pool wall extends into this level, and several beam ports from the reactor core terminate at the outside wall of the pool shield. The floor area of this level is serviced by a three ton manual hoist. A laundry room is located on the west side of the main stairwell and is equipped with a washer and dryer. A restroom is located on the east side of the main stairwell for

FIGURE 5-1 NSCR BUILDING CROSS SECTION

FIGURE 5-2 UPPER RESEARCH LEVEL

the convenience of personnel working on the lower research level. Several steel tubes extend into the east wall of this level and provide storage facilities for beam port plugs.

4. Reception Room

The reception room is located on the south side of the confinement structure. All personnel initially entering the confinement structure enter through this building where a personnel log is maintained and where personnel dosimetry is issued. A master control panel for operation of exhaust and air conditioning systems in the confinement structure is located on the north wall inside this building.

5. Laboratory Building

To accommodate an increase in research load and to allow for expansion of programs a laboratory building was added to the south end of the reception room (Figure 5-4). The new laboratory building is located outside of the main reactor confinement and contains pneumatic receivers in laboratories 4, 5, and 6. All pneumatic systems connecting the NSCR to the new laboratory building are under USNRC regulation whereas within the new laboratory the licensing of the use of isotopes, the control of radiation exposures, and the release of radioactive material is within the jurisdiction of the State of Texas. Each pneumatic system to the new lab building is completely enclosed within a large airtight tube, and air within this tube is pulled through the existing exhaust system and monitored for radioactivity prior to release from the stack. This design allows for monitoring and controlled release of radioactive gases associated with operation of the pneumatic system.

B. Confinement Ventilation System

1. Air Handling Units

Four air handling units and an exhaust fan can be used to control pressure, temperature, and humidity within the reactor building. The facility can be divided into 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 building entry where contamination is least expected to occur. An intermediate zone of negative pressure includes the main research areas where infrequent contamination might occur. The third zone of maximum negative pressure includes areas where contamination of activation is likely to occur, i.e., beam ports, thermal column, and through tubes. Air is not recirculated in these areas and is monitored and exhausted directly to the stack.

Air handler unit A supplies air to the upper research level, unit B to the lower research level, unit C to the control room, and unit D to the restrooms and electronics shop. All four units may be operated from a control panel in the reception room and will shut down simultaneously with the exhaust fan when alarm levels are reached on the exhaust particulate monitor or the fission product monitor.

2. Dampers and Filters

Dampers are located at the air inlet to all handling units, the fresh air bypass to the exhaust fan, and in the exhaust stack. The height of the exhaust stack above ground level is 85 feet. In cases of emergency, these dampers can be simultaneously closed and the air handlers turned off to isolate the building by a switch which is located in the reactor control room. The inlet side of the air handling units is equipped with filters. An emergency exhaust air filter system is installed between

the exhaust fan and building stack. The emergency filter system consists of two particulate filter banks and one bank of activated carbon filters and may be operated manually during emergency air handling conditions.

FIGURE 5-3 LOWER RESEARCH LEVEL

FIGURE 5-4 LABORATORY BUILDING AND PNEUMATIC SYSTEM EXTENSION

the convenience of personnel working on the lower research level. Several steel tubes extend into the east wall of this level and provide storage facilities for beam port plugs.

3. Emergency Operation

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

VI. EXPERIMENTAL FACILITIES

A. Beam Ports

1. Description

Five permanent beam ports of Type 304 stainless steel are cast into the pool wall at the lower research level. One of the beam ports, Number 5, is located in the north wall of the main pool (See Figure 6-1). The other beam ports are located in the stall end of the pool and are most frequently employed in beam port experiments, since a number of experiments may be performed simultaneously with the core located in the stall position. The thermal column has been modified to provide additional beam ports, numbers 6, 7, and 8. The stall section with beam ports 2, 3, 6, and 8 in use is shown in Figure 6-2.

The beam ports are fabricated of stainless steel with sections 6, 10, and 19 inches in diameter divided longitudinally into 3, 2 1/2, and 1 foot segments, respectively. This design prevents neutron streaming when concrete shield plugs are in place. The 6 inch and the 10 inch sections are lined with 1/4" boral with exception of the six inch section of beam port No. 4. Each of the above ports ends flush with the external face of the pool wall and is sealed by a hinged 2-foot square, 4-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 microswitch actuates an annunciator light on the console when these doors are opened.

Beam port plugs are aluminum cylinders filled with barytes concrete, each about one foot long with a handle recessed in the exposed end for ease of handling. Three of these plugs are used in the six inch diameter beam port section and two are used in the ten inch diameter section when the port is not in use. A 19-inch diameter, one foot thick section is available to plug the final recessed section of the beam port.

Each beam port is connected by 2" diameter pipe 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.

2. Intended Use

These beam ports may be used in a variety of experiments such as the extraction of a well collimated beam of neutrons and/or gamma rays from the reactor. A variety of extensions are bolted to the beam ports as required by different experiments. The extensions are designed to prevent interference with the movement of the reactor frame and grid plate and to prevent pool water leakage. Each experiment which utilizes the beam port is carefully reviewed to prevent personnel exposure or possible damage to the reactor system. Since most of the utilization of the beam ports involves the extraction of neutron capture gamma radiation, the operating position of the reactor in the stall (Figure 6-2) is such that a trough can be suspended between the tips of beam ports 1 & 4 tangential to the east face of the reactor core. A bismuth shield in the trough provides isolation of the two ports. The design allows individual encapsulated target material to be removed and replaced without disturbing the target on the opposite side of the bismuth. Beam ports 2 & 3 are radial ports with extensions ending outside the reactor frame. Beam ports 6 & 8 have removable, weighted extension tubes that can be mounted tangential to the core on the north and south sides. Beam port 7 is located in the thermal column between beam ports 6 & 8. A removable extension tube that is normal to the east face of the reactor core can be used with this beam port. The extensions are

FIGURE 6-1 NSCR EXPERIMENTAL FACILITIES

FIGURE 6-2 - REACTOR STALL AND BEAM PORT INSTALLATIONS

designed to prevent interference with movement of the reactor frame and grid plate and to prevent pool water leakage.

A television monitoring system is installed for monitoring of beam port areas on the lower research level. The system consists of cameras positioned on the lower research level and a monitor in the reactor control room. A switch is provided for selection of cameras. The system is used to observe personnel entry and activities in the beam port experimental areas. This is in addition to a C-2 alarm device which also indicates personnel entry.

B. Through Tube

1. Description

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 are positioned along a colinear axis, a straight 6-inch diameter connecting tube can be bolted to the flanged pool ends of the tubes providing a continuous 6-inch diameter passage completely through the pool. Concrete plugs are described above provide the necessary shielding in this tube to prevent streaming of radiation. The through tube is also vented to the central exhaust system.

2. Intended Use

Transit experiments which pass through this tube or fixed experiments may be used in conjunction with this experimental facility. 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.

C. Thermal Column

1. Description

The thermal column, located in the east end of the stall portion of the pool, is constructed of stainless steel and aluminum. The pool wall is penetrated at core level by a 3 1/2 foot square section on the pool side and enlarges to a 4 foot square opening on the experimenter's side. The walls of the thermal column are welded to the stainless steel pool liner. A gasketed aluminum cover plate is bolted to the inside flange of the cavity providing the water seal.

A vent line from the thermal column cavity extends directly to the central exhaust system, where the air is monitored prior to release. A movable thermal column door, constructed of lead and concrete shielding material, is mounted on tracks embedded in the lower research level floor.

2. Intended Use

Besides use as a thermal column, this facility has been modified to provide additional beam ports as is shown in Figure 6-2.

D. Pneumatic Tubes

1. Description

The NSC pneumatic system consists of an Electronic Programmer, a Control Chassis for control of intercom and "initiate" locations, lab and core receivers that may be connected in any combination through quick connectors and a CO₂ valve panel which provides a regulated carbon dioxide gas supply. The programmer provides for variation of the irradiation and transit times, and is capable of starting an analyzer or other counting systems. A continuous flow of carbon dioxide may be flushed through the transfer tube to the core to prevent Argon-41 buildup between sample transits. The Control Chassis cannot be used in conjunction with more than one combination of lab and core receivers at a time.

The pneumatic tube itself consists of a core receiver, polyethylene tubing, protective metal sheathing at the reactor bridge and a receiver in any of several laboratories (Figures 6-3, 6-11.). The pool wall pneumatic penetrations shown in Figure 6-1 have not been used for some time due to 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.

2. Intended Use

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

E. Irradiation Cell

1. Description

The irradiation cell is located at the west end of the reactor pool. This cell is approximately 18 feet wide by 16 feet deep by 10 feet high. The frame for the concrete roof is fabricated from 8 x 8 inch steel "I" beam columns connected with 6 x 15 inch steel "I" beam joists. An overlay of 4 x 6 inch timbers provide decking for the concrete blocks which are 2 x 2 x 4 feet. The blocks are stacked 4 feet high with an opening of approximately 5 x 5 feet left directly over the cell window. A motor driven concrete shield 2 feet thick is installed over the opening (Figure 6-5).

Access is provided by an elevator which is raised and lowered from the upper research level by the overhead crane. The elevator can accommodate sample containers up to 51 inches wide by 49 inches deep by 73 inches high. With the exception of the elevator opening the upper level of the cell is decked with steel plate. A small section of the deck plate is hinged to provide access to the emergency ladder which runs from the upper level to the top of the concrete shield.

Concrete steps lead up to the top of the cell cover on each side of the pool. Handrails are installed around the elevator opening. The area provides an excellent vantage point for facility visitors. The irradiation cell window is cast into the 2 feet thick wall which separates the cell from the reactor pool. The window is 2 feet square on the pool side and flares out to 4 feet square on the cell side. A 1/2 inch aluminum plate is bolted 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. The cavity formed by an aluminum gasketed plate attached to the window on the irradiation cell side is used as a water shutter. A plenum at the bottom of the plate allows for filling and dumping of water collected within the shutter. A console on the upper research level floor area provides controls for filling, dumping, and determining level of water within the shutter. Annunciator lights on the shutter console and in the control room indicate water level in the shutter.

Electrical power for the motor driven shield is supplied through a breaker and a reversing switch to open or close the cell. The breaker can be locked to control opening of the shield. Mechanical stops are always attached to the NSCR bridge rail to prevent inadvertent movement of the NSCR closer than eight feet from the irradiation cell window. Also, a bridge interlock scram is

FIGURE 6-3

NSC PNEUMATIC SYSTEM
SOUTH SIDE - LABS 4 & 5

FIGURE 6-4

NSC PNEUMATIC SYSTEM

FIGURE 6-5 IRRADIATION CELL

provided in the event the irradiation cell door is opened whenever the reactor is positioned within eight feet of the cell.

To handle removal of Argon-41 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. Thus, all air from the cell will be monitored prior to its discharge to the outside environment. The controls for the cell air exhaust are located on the reactor console. Also, an experiment scram button and an intercom are located inside the cell.

F. Neutron Radiography - Beam Port 4

This facility is a concrete block structure on the lower research level located adjacent to the pool shield wall contain and shield a thermal neutron beam extracted from beam ports No. 4 (Figure 6-6). The cave structure is designed for remote positioning of samples with the beam port in operation. A hydraulic shutter at the beam port exit may be raised to shield the neutron beam between exposures. A dark room is available for loading and unloading cassettes and film processing.

An alarm in the reactor control room will alert the operator upon personnel entry into the sample preparation room or cave and a flashing red light signal is visible to the person entering. An entry device on the cave door is designed to cause a scram if the cave door is opened when the reactor is positioned against the radiography reflector. The bridge rail stop, which restricts moving the reactor any closer than 18 inches from the reflector must be in place for reactor operation when the cave entry door is open.

SCALE: 1" = 4'-0"

CONTAINMENT VOLUME

FIGURE 6-6 BEAM PORT 4
NEUTRON RADIOGRAPHY
FACILITY

VII. INSTRUMENTATION AND CONTROL

A. Control System Concept

The NSCR operates in two standard modes. Mode 1 is steady state operation at power levels up to 1000 Kw (thermal). Mode 2 is pulsed operation produced by the rapid withdrawal of the transient rod that introduces a step insertion of reactivity which results in peak powers of up to about 1,600,000 Kw. The reactor is operated from a console that displays all pertinent reactor operating conditions. The console also displays information about the cooling system, environmental monitoring, and experimental facilities. The control system consists of five power measuring channels utilizing four ion chambers and one fission counter. Test circuits and calibration signals are provided for the safety measuring channels. Shim safety rods, the regulating rod, and the pneumatic electromechanical transient rod drives are controlled from the console. A scram signal interrupts magnet current to the shim safety rod drives and air pressure to the transient rod drive for rapid insertion of control rods by gravity fall to shut the reactor down. The regulating rod is not connected to the scram circuits. A selector switch is provided for selecting the steady state or the pulsing mode of operation. For steady state operation, the control rods are withdrawn slowly by manual control until the desired power level is reached. The servo system may be used to automatically maintain the power level by controlling movement of the regulating rod.

B. Nuclear Instrumentation

1. Steady State Mode

In the steady state mode of operation reactor power is monitored by a log power channel, a linear power channel, and two safety channels which provides excellent overlap indications during reactor startups and shutdowns. The fuel temperature is monitored using an instrumented fuel element positioned adjacent to the central bundle of the core.

The log power channel consists of a fission chamber detector, preamp, amplifier, ratemeter, and log power recorder. The log power channel has a range of ten decades of reactor power, and interlocks are provided to prevent startups without at least 2 counts per second and to prevent pulsing at powers above 1 Kw. A reactor scram occurs in the event of a reactor period of 3 seconds or less. A simplified diagram of the log power channel is shown in Figure 7-1.

The linear power channel consists of a compensated ion chamber, a linear picoammeter, digital power readout, linear power recorder, and a servo controller. The detector is positioned above a tapered graphite reflector element which scatters the neutron flux from the core face to the detector. This configuration provides excellent linearity and significantly reduces the contribution due to gamma rays so that the system is sensitive and accurate at low power levels even after extended operation at 1 MW.¹³

The linear channel is connected to the servo controller which operates the regulating rod to maintain a constant power level during operation. A permit switch allows manual or automatic operation when the reactor power level reaches 15% of the set point on the linear recorder.

¹³T.A. Godsey and J.D. Randall, "A Solution to the Varying Response of the Linear Power Monitor Induced by Xenon Poisoning," presented at TRIGA Owner's Conference III, Albuquerque, N.M., February 1974.

FIGURE 7-1 LOG POWER CHANNEL

Regulating rod control is automatically shifted back to manual if the level drifts out of the 15% range. A signal from either the servo controller or from the manual control switch is connected to the regulating rod drive mechanism. The linear power channel is shown in Figure 7-2.

The Safety Power Channel Amplifier consists of two identical, isolated sections, each one consisting of an uncompensated ion chamber detector, a linear amplifier, two bi-stable trips, and power supplies. A diagram of the Safety Power Channel is shown in Figure 7-3.

The safety amplifier is the heart of the safety circuitry of the reactor control system. The instrument supplies current to the control rod magnets and provides the mechanism for scrambling the reactor. A signal from the ion chamber detector is fed to an amplifier which will disrupt the current to the control rod magnets when a preset level is exceeded. This level is set at 125% or less of full power. The current from each chamber is fed simultaneously to two independent amplifiers, and each circuit relies on the function of its own relays and other components, thereby providing excellent backup performance.

Additional scrams may also be coupled to the safety amplifier. As indicated in the above section, the period scram is fed to the safety amplifier. The additional scrams that are connected in the external scram chain are: fuel temperature scram (LSSS), the manual scram, the bridge lock scram, and various experiment scrams. Experiment scrams are provided in areas where an accident or other unusual circumstances could cause the individuals working with the experiment to receive high radiation exposures unless the reactor were rapidly shut down. Such a scram does not rely on communication with operations personnel but is initiated by the experimenter sensing the difficulty. Manual and experiment scram buttons are located near the beam port areas, in the irradiation cell, on the reactor bridge, and along pool side.

An instrumented fuel rod 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 provides flexibility in that 8 locations in the core are allowed, and another significant advantage is that thermocouple response in these locations is nearly independent of the amount of FLIP fuel in the core for both pulsing and steady state operations. Three chromel-alumel thermocouples are embedded in the instrumented fuel rod and are located at the vertical center and one inch above and below the vertical center and radially 0.3 inches from the center (Figure 3-5).

The fuel element temperature measuring channel consists of an instrumented fuel element and the fuel element temperature recorder. The recorder has a scram function which will scram the reactor if the temperature at the thermocouple position reaches 975⁰F. A digital thermocouple indicator and thermocouple selector switch is also available to read out the temperature of thermocouples in the fuel, the pool water, and the irradiation cell. The digital thermocouple indicator is switched to the peak retention mode when the reactor is pulsed. A diagram of the fuel element temperature channel is shown in Figure 7-4.

2. Pulsing Mode

After a moderate power level less than 1,000 watts in the steady state operating mode is reached, the mode switch is changed to the pulsing mode so that the reactor can be pulsed. When the switch is turned to this mode, the normal neutron channels are disconnected and the high level pulsing chamber becomes the monitoring channel. The pulsing chamber is a gamma or neutron ion chamber that is located adjacent to the reactor core and its output is fed to an integrator circuit which provides a digital display of the integrated pulse power (NVT). The integrator also provides

FIGURE 7-2 LINEAR POWER MEASURING CHANNEL

FIGURE 7-3 SAFETY POWER MEASURING CHANNEL

FIGURE 7-4 FUEL ELEMENT TEMPERATURE CHANNEL

outputs to record the reactor power level (NV) and the integrated power (NVT) as a function of time. Prior to pulsing, the transient rod cylinder is positioned for the desired pulse reactivity insertion and the transient rod remains in the full down position. Changing the mode switch to pulsing removes the inter- lock that prevents application of air to the transient rod. To pulse the reactor air is applied to the transient rod for rapid withdrawal. After a pulse occurs, the transient rod is reinserted into the reactor core after a preset time delay of less than 15 seconds. The pulse power measuring channel is shown in Figure 7-5. As indicated in this figure, the digital temperature meter (DORIC) is switched to fuel temperature peak retention during a reactor pulse.

3. Control Rod Drives

Each shim-safety rod mechanism is connected to a control unit at the reactor console. The magnet carriage up and down position is indicated by lights and full range positioning by a digital readout calibrated in percent withdrawal. An engagement light indicates contact of the control rod and the magnet. A control rod down light is also available. Push buttons permit operation of each rod drive independently of the other control rods. All shim-safety rod drives may be inserted or withdrawn simultaneously by the operator by use of a gang switch. An interlock is provided which prevents withdrawal of the transient rod when any of the four shim-safety rod drives are being withdrawn. The regulating rod does not have a scram capability and is manually controlled by the operator or automatically controlled by the servo mechanism.

The transient control rod is provided with a pneumatic-electromechanical rod drive. This drive system which is controlled from the reactor console is shown in Figure 3-14 and 3-15. The transient rod drive is mounted on a support frame which bolts to the reactor bridge. A holddown tube extends from the control rod guide tube of the fuel bundle to the transient rod mounting frame and assures that the fuel bundle will remain in place when the transient rod is withdrawn.

An interlock is provided which prevents rapid pneumatic withdrawal of the transient rod for reactor power levels in excess of 1 Kw. For steady state reactor operations the transient rod position is controlled by the electromechanical portion of the transient rod drive. The pneumatic cylinder must be in the full down position to apply air to the piston for steady state operation of the rod. Once air is applied, the transient rod is controlled by movement of the pneumatic cylinder at a rate of about 25 inches per minute.

Limit switches provide an indication when the air cylinder has reached its upper or lower limit of travel. Similarly, a limit switch is actuated when the piston reaches its lower limit of travel. A continuous position indicator with digital readout is provided to indicate the position of the air cylinder.

4. Minimum Reactor Safety Circuits and Interlocks

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

C. Cooling System Instrumentation

The reactor cooling system control panel is located in the reactor control room. Prior to operation of the system, valve positions must be established and verified by the reactor supervisor. The system is controlled by the reactor operator using the start-stop switches to the cooling tower fan, primary pump, and secondary pump. A multipoint temperature recorder and primary flow rate indicator provide continuous performance data of the system, and a conductivity meter is used for readings of the bulk pool water purity. A graphic panel is also provided for display of the cooling system flow

schematic. Alarms are provided for primary pump failure, secondary pump failure, and loss of secondary flow.

D. Control Room

1. General Layout

The reactor control room is located on the upper research level next to the reactor pool as shown in Figure 5-2. All of the reactor controls are within the immediate reach and visibility of the reactor operator. The operator may view the reactor pool to his left, the reactor console directly in front of him, and auxiliary instrumentation to his right. The control room layout is shown in Figure 7-6. In addition to the reactor controls and alarms, the console contains instrumentation for the pool water systems, area radiation monitoring, building ventilation, facility air monitoring, building intercommunications system, and the telephone system. The arrangement of the instruments in the main and auxiliary panels of the console is shown in Figures 7-7 and 7-8.

2. Summary of Information Displayed and Recorded

Table VII lists the controls and the information that is displayed and recorded on the reactor console.

3. Occupancy Requirements

At all times when the console is turned on, a licensed reactor operator or licensed senior reactor operator will be in the control room. Reactor operators in training may operate the reactor in the presence of a licensed reactor operator or licensed senior reactor operator in the control room. All fuel additions to the reactor core and critical experiments require the presence of the Director, the Associate Director, the Manager of Reactor Operations, or the Reactor Coordinator.

Timer Functions:

- ① Digital NVT Display shown .5 sec after pulse
- ② TR Scram <15 sec after pulse
- ③ Doric thermocouple indicator switched to peak retention 2.5 sec after pulse

FIGURE 7-5 PULSE POWER MEASURING CHANNEL

TABLE VI

Minimum Reactor Safety Channels

<u>Safety Channel</u>	<u>Number Operable</u>	<u>Function</u>	<u>Effective Mode</u>	
			<u>S.S.</u>	<u>Pulse</u>
Fuel Element Temperature	1	SCRAM @ LSSS	X	X
Hi Power Level	2	SCRAM @ 125%	X	
Console Scram Button	1	SCRAM	X	X
Hi Power Level Detector Power Supply	1	SCRAM on loss of supply voltage	X	
Preset Timer	1	Transient rod scram 15 seconds or less after pulse		X
Log Power	1	Prevent withdrawal of shim-safeties at <4 x 10 ⁻³ watts	X	
Log Power	1	Prevent pulsing above 1 kW		X
Transient Rod	1	Prevent application of air unless fully inserted	X	
Shim-safeties & Regulating Rod Position	1	Prevent withdrawal		X
Pool Level	1	Alarm at 90% normal operating level	X	X

FIGURE 7-6 CONTROL ROOM LAYOUT

FIGURE 7-7 ARRANGEMENT OF MAIN PANEL OF THE REACTOR CONSOLE

FIGURE 7-8 ARRANGEMENT OF AUXILIARY PANEL OF THE REACTOR CONSOLE

TABLE VII

Summary of Information Displayed and Recorded on Reactor Console

	<u>Control</u>	<u>Indication</u>	<u>Record</u>
<u>Reactor Safety Systems</u>			
Log N-Power		x	x
Log N-Period		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	
<u>Water Systems</u>			
Pool Water Cooling System	x	x	x
Pool Recirculation System	x	x	
Pool Skimmer System	x	x	
Diffuser System	x	x	
Transfer System	x	x	
Secondary Treatment System	x	x	x
<u>Personnel Control and Radiation Protection</u>			
Area Radiation Monitors		x	
Facility Air Monitors		x	x
Air Handling System Shutdown	x	x	
Emergency Evacuation Horn	x	x	
Irradiation Cell Exhaust	x		
Television Monitors		x	
Facility "Door Open" Alarms		x	
<u>Experimental Facilities</u>			
Pneumatic System	x	x	
Motor Rotisserie	x	x	
"C-2" Experiment Personnel Control Alarm	x	x	

VIII. EMERGENCY SYSTEMS AND ENGINEERED SAFEGUARDS

A. Building Isolation System

To prevent or minimize the uncontrolled release of radioactivity to the environment of the Nuclear Science Center, the air discharged from the building is continuously monitored. If a spill or a release increases radioactivity levels above a preselected set point, an alarm sounds, the evacuation horn can be sounded, and the building can be isolated by the action of dampers in the air handling system. A control panel is located in the reception room from which the building air circulators and exhaust system can be operated, building air can be filtered, and air can be monitored and released from the building.

B. Exclusion Area

The site for the NSC was selected to achieve a more than adequate isolation distance between the Nuclear Science Center and the nearest non-University area. Access is controlled to the six acre fenced area upon which the NSC is located, and the surrounding land is fenced and owned by the University. The NSC implements a security plan with the aid of the University Police.

C. Storage of Special Nuclear Material

Technical specifications require that fuel elements be stored in a geometrical array where the k -effective is less than 0.8 for all conditions of moderation. Irradiated fuel elements are to be stored in an array which permits 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 NSCR. These areas are the fuel storage room and the reactor pool handling and storage areas. Unirradiated fuel can be temporarily stored in approved shipping containers used by the fuel manufacturers for shipment in the reactor building which is protected by keyed locks and intrusion security. The normal storage location for unirradiated fuel is the fuel storage room located under the up-ramp to the NSCR. Elements stored in this room are in an environment of dry, cool air and are under lock and key and intrusion security. Fuel bundles (4 elements maximum per bundle) and individual elements are positioned in cadmium lined aluminum tubes which are secured to an aluminum frame mounted to the concrete walls of the fuel storage room. The fuel storage room and storage rack assembly are shown in Figure 8-1. Irradiated fuel elements are stored under water in racks mounted on the reactor pool walls or in a storage rack located on the floor of the reactor pool. Wall storage racks have aluminum tube positions for 6 fuel bundles and 12 individual elements. The fuel element rack positioned on the floor of the pool is constructed of aluminum angle and tubing and has facilities for 24 fuel bundles. This facility has a large aluminum lid which covers the tubes and protects stored fuel. A wall storage rack is shown in Figure 8-2, and the floor storage rack is shown in Figure 8-3. A typical fuel storage arrangement in the reactor pool is shown in Figure 8-4. A fuel element maintenance jig is used for storage of a fuel bundle during periods of fuel element inspection or fuel bundle maintenance. Location of the maintenance jig is shown in Figure 8-4. Special storage facilities are provided for instrumented fuel and control rod fuel followers.

D. Emergency Personnel Control

In the unlikely event of a nuclear accident provisions have been made for the care and treatment of accident cases involving bodily injury with radioactive contamination or acute radiation exposures. These personnel can be transported by either university vehicles or the city of College Station ambulance service

to St. Joseph's Hospital, East 29th Street, Bryan, Texas. Arrangements have been made with the hospital and a segregated area is available for treatment of these personnel.

E. Emergency Equipment

The Nuclear Science Center maintains an adequate inventory of equipment located strategically throughout the facility. First aid supplies and a stretcher are located in the laboratory building adjacent to the reception room, plus an emergency shower is available in the laboratory shop area. A portable supply cart which contains supplies for contamination control, first aid, personnel monitoring, and protective clothing and respiratory equipment is available in the lab building. CO₂, water, and dry chemical fire extinguishers are distributed throughout the NSC site.

In case of an emergency, in which the equipment and supplies of the Nuclear Science Center are expended, additional equipment as outlined below is available from the Radiological Safety Office and other campus facilities.

- (1.) Self Contained Breathing Apparatus
- (2.) Protective Clothing including coveralls, gloves, hoods, shoe covers, plastic suits, tape, and assault masks.
- (3.) Portable survey instruments including neutron survey meters, ion chambers with ranges up to 10,000 R/hr and G.M. survey meters.
- (4.) Counting equipment including G.M. counting system, proportional counting system and pulse height analyzers.
- (5.) Portable air sampling equipment
- (6.) Communications equipment - 2 way radios

F. Reactor Heat Removal System

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

G. Reactor Coolant Leakage Control System

1. Pool Level Float Switch

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. 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 could be corrected before the core would be uncovered.

2. Pool Isolation Valves

The two coolant return lines and the coolant extraction line in the bottom of the pool can be manually closed to isolate the pool in the event of cooling system component failure.

FUEL STORAGE ROOM

FUEL STORAGE CONTAINER

FIG. 8-1 FUEL STORAGE ROOM AND RACK

FIG. 8-2 FUEL STORAGE RACK FOR REACTOR POOL WALL

FIG. 8-3 FUEL STORAGE RACK FOR REACTOR POOL FLOOR

FIG. 8-4 REACTOR POOL FUEL STORAGE ARRANGEMENT

H. Emergency Pool Fill System

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

I. Emergency Lighting System

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

J. Facility Service Systems

1. Fire Protection

Fire protection is provided at the Nuclear Science Center by numerous fire extinguishers which are located throughout the NSC site. Additionally, the College Station Fire Department provides the NSC with fire protection services and is on call twenty-four hours a day. Fire department personnel receive training in radiological hazards and NSC site familiarization on an annual basis.

2. Console Instrument Cooling

Cooling for the NSCR control console comes from air handler Unit C, thus providing a controlled environment for primary reactor control instrumentation.

IX. RADIATION PROTECTION AND RADIOACTIVE EFFLUENTS

A. Introduction

All activities will be conducted in such a manner as to comply with 10CFR20 "Standards for Protection Against Radiation". Exposure of individuals and release of radioactivity to the environment will be controlled to maintain compliance with all applicable sections of the regulations. Methods and instrumentation which are used to establish compliance are outlined in the following sections.

B. Liquid Waste

1. Generation of Liquid Waste

Low level liquid waste originates from four primary sources at the Nuclear Science Center. These sources are: (1) floor drains, shower and radio chemistry laboratory on the lower research level; (2) demineralizer room filter and ion bed; (3) condensate from air handling units on mechanical chase; and (4) valve pit sump in cooling equipment room.

2. Liquid Waste Handling System

Liquid waste flows through common headers to a liquid waste sump located below the grade of the lower research level (See Figures 4-4 and 9-1). 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 22,000 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 dumped to the unrestricted environment.

When a tank is filled, it is valved off, stirred to insure uniformity, sampled, and the activity determined. Care must be taken in the collection and preparation of the liquid waste sample to insure that the concentrations determined are representative of what is being released. Thus the tank must be isolated from incoming water prior to sampling, the liquid should be thoroughly mixed when sampled, and care must be taken in sample preparation to prevent foreign contamination. Based on the activity determination, the waste will either be drained, stored for decay, or diluted with fresh water to release levels in compliance with 10CFR20 limits and drained.

C. Solid Waste

1. Generation of Solid Waste

Solid radioactive waste in the form of rags, paper towels, used laboratory equipment, sample containers, aluminum, etc., is generated in normal operation of the Nuclear Science Center Reactor. Activated materials such as used experimental hardware also are generated.

FIGURE 2.1 LIQUID WASTE DISPOSAL SYSTEM

2. Solid Waste Handling and Disposal

Low level solid 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 (See Figure 2-2). This waste is transferred to the Texas A&M University by-product material license for disposal.

Activated equipment is normally stored in the high level waste storage area adjacent to the outside wall of the irradiation cell. If equipment cannot be reused, it may be transferred in the same manner as low level waste if possible under state regulations.

D. Gaseous and Particulate Waste

1. Generation of Radioactive Waste

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

Nuclear Science Center Technical Report No. 32, "Determination of Argon-41 Production at the Texas A&M Nuclear Science Center Reactor" documents that approximately 4.7 Ci of ^{41}Ar are released on an annual basis. Applying a dilution factor of 5×10^{-3} , the releases produce approximately .8% of the permissible concentration specified in 10CFR20. The results of 4.7 Ci was based on 100 MW-day operation of the NSCR.

Several important findings were made concerning ^{41}Ar production at the NSC. On a long term basis, the pool accounts for more than 95% of the facility's production. The average cell concentrations were measured versus time for the cell exhaust on and off for 4 hours at 1 MW. As expected the exhaust-on values were lower than the exhaust off, with peak values of 6.7×10^{-5} pCi/cc and 1.2×10^{-4} pCi/cc, respectively. The pneumatic system was examined for absolute production on each firing as a function of time at 1 MW before the first firing, and results showed that the release increased from 6.8 pCi present before reactor startup to a plateau value of about 208 pCi after 6 hours at 1 MW. In all cases the system was purged of argon after 5 firings. As expected, the dry tube showed no contribution to release, but the beam port measurements showed a level of 2.15×10^{-3} pCi/cc at 1 MW in Beam Port #1, closest to the core. Thus, although the pool is the major production source in the long run, the other sources can rival the pool release rate on occasion.

2. Gaseous Waste Handling

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.

3. Gaseous Waste Disposal

Gaseous waste is disposed to the environment through the building stack which is 85 feet in height.

E. Dilution Factor Calculations

The equations used in developing the dilution factors calculated below are those presented by F. A. Gifford, Jr.^{14,15} 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 m².

The equation for the atmospheric dilution factor is:

$$X = \frac{Q}{\pi \sigma_y \sigma_z u} \exp \left\{ -\frac{1}{2} \left(\frac{y^2}{\sigma_y^2} + \frac{h^2}{\sigma_z^2} \right) \right\}$$

where

X = Concentration in grams or curies per cubic meter

Q = Original source strength in grams or curies per second

u = Mean wind speed in meters per second

y = Crosswind in meters from the plume axis

h = Source height in meters

σ_y, σ_z = Dispersion coefficients in m²

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

$$X = \frac{Q}{(\pi \sigma_y \sigma_z + cA)u}$$

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, σ_z for stable conditions and a wind speed of 1 m/sec (2 MPH) to determine the dilution factor available under pessimistic conditions. ($Q = 1$) at a distance of 100 meters from the point of release.

$$X = \frac{1}{(\pi \sigma_y \sigma_z + cA)u}$$

$$X = \frac{1}{(3.14 \times 4 \times 2 \times 0.5 \times 357)}$$

$$X = \frac{1}{202}$$

This calculation indicates that the minimum dilution at 100 meters is 200/1 under the most adverse conditions. From the wind rose diagram shown in Figure 2-3 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 (10 MPH) is substituted into this equation the dilution factor becomes:

$$X = \frac{1}{903}$$

¹⁴F. A. Gifford, Jr., Nuclear Safety, December, 1960.

¹⁵F. A. Gifford, Jr., Nuclear Safety, July, 1961.

Again this is a pessimistic approach since the dilution was calculated at only 100 meters (approximate boundary of 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 it still is 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.

F. Facility Air Monitoring System

Argon-41 activity is monitored with a gas detector which utilizes a 3" NaI (TI) scintillation crystal and a gamma spectrometer. 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 on the console and a warning light on the console and in the reception room. The system is shown schematically in Figure 9-2.

Stack particulate activity is monitored with a moving tape type, continuous air monitor. 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 indicating the channel alarming and also causes an automatic shutdown of the air handling system to isolate the facility.

Building gas activity is monitored by a gas detector which 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 tape type, 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 with a low sensitivity for the detection of gases is used to essentially eliminate high detector backgrounds due to ^{41}Ar gas. The air sampling region is located approximately one foot above the pool surface at the reactor bridge. Air is drawn through the line and through the monitor filter paper using an air suction pump. A G.M. detector monitors the filter paper 180° from the point of collection. The monitor primarily detects particulates that are produced by decay of fission product gases collected in the sampling line. This monitor is equipped with an alarm circuit which activates an audible alarm and a warning light. An alarm on this system will automatically shut down the air handler units and shut air dampers to isolate the facility.

G. Area Radiation Monitors

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 annunciator panel. A red light on the indicating meter and on the detector identify the particular area. A block diagram of a typical system is shown in Figure 9-3.

FIGURE 9-2 FACILITY AIR MONITORING SYSTEM

FIGURE 9-3 AREA RADIATION MONITOR SYSTEM

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.

H. Health Physics

I. Personnel Monitoring

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.

2. Protective Clothing and Equipment

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.

3. Change Room Facility

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

4. Radioactive Materials Handling Area

A radioactive materials 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 10CFR20 requirements.

5. Laboratory Facility

A standard radio chemistry laboratory on the lower research level is available for research experiments and health physics use. Equipment for routine radiochemical procedures is maintained. Laboratory procedures as required for fulfillment of the radiation protection regulations will be developed as needed.

6. Environmental Monitoring Program

An environmental monitoring program has been established between Texas A&M University and the Texas Department of Health. Through this program, vegetation and water samples are collected from NSC creek, White creek, the upper and lower Brazos River, and the sanitary outflow. These samples are analyzed for gross gamma and beta radioactivities and radioisotope identification. Data from these samples have remained basically unchanged since 1974 and no results which would have a significant impact on the environment have been found.

7. Portable Radiation Survey Instruments

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 10CFR20 limits and established ALARA* limits.

8. Health Physics Counting Equipment

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.

* As Low As Reasonably Achievable

X. CONDUCT OF OPERATIONS

A. Organization and Responsibility

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

The internal administration of the NSC is comprised of the Reactor Administration and the Facility Administration. The Facility Administration is not directly related to reactor safety and is established internally by the Director.

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

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

B. Training

A training program for reactor operations personnel exists to prepare personnel for the USNRC Operator or Senior Operator examination. This training program normally contains twenty hours of lecture, outside study, and requires approximately twenty reactor startups. At the conclusion of the program, the Director or Associate Director of the Nuclear Science Center conducts an examination of the trainee to ascertain whether or not he is qualified to take the USNRC examination.

C. Written 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 best protect the health and safety of the public. This philosophy is augmented in practice by detailed, written procedures. The procedures are followed by all personnel using the facilities of the Nuclear Science Center. 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.

D. 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.

A supervisor log is maintained which contains information on facility or reactor changes, and items of a more detailed nature concerned with operational aspects of the facility. A review is performed of each unscheduled shutdown along with corrective action taken prior to the next startup.

Nuclear Science Center Reactor Administrative
Organizational Chart

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

E. Review and Audit of Records

A Reactor Safety Board (RSB) acts as a review panel for new reactor experiments, procedural changes, and facility modifications. The RSB thus provides an independent audit of the operations of the Nuclear Science Center. Problems of a nuclear safety nature 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 (See Figure 10-1).

F. Reactor Operating Safety Philosophy

All operations involving the reactor will be conducted in compliance with the regulations specified in 10CFR50 and 10CFR55. The reactor will be operated within the limits of the license and technical specifications.

XI. SAFETY EVALUATION

A. General Summary

The values given in this SAR were generated using the computer codes described in chapter III. The current values are given for both FLIP and LEU fuel types. To continually prove the validity of our method, the values accepted in the previous version of the SAR for an all FLIP core are also given. The current values for the FLIP core show little difference from the previously accepted values, and therefore, serve to demonstrate the validity of the new methods.

B. Fuel Description and Safety Limits

The fuel currently used in the NSCR is GA FLIP fuel, and the type to be loaded is GA LEU 20-20 fuel. The two types of elements are identical in geometry and differ physically only in the fuel "meat" section, with a larger uranium fraction, lower U-235 enrichment, lower burnable poison content, and lower hydrogen-to-zirconium ratio. Table I lists the principal design parameters of both FLIP and LEU elements.

The safety limitations on the fuel are those imposed by the loss of fuel element integrity. During a reactivity excursion the limiting condition is the fuel temperature and the corresponding hydrogen overpressures at which clad rupture may occur. Studies show that in FLIP fuel the hydrogen pressure which would result from a transient for which the peak fuel temperature is 2100° F (1150° C) would not produce a stress in the clad in excess of its ultimate strength. TRIGA fuel with a hydrogen-to-zirconium ratio of at least 1.65 has been pulsed to temperatures of about 2100° F (1150° C) without any damage to the clad. As a safety limit, the peak fuel temperature to be allowed during transient conditions is set at 2100° F (1150° C) for FLIP fuel. For LEU fuel, the temperature limit is also based on dissociation of hydrogen. The temperature limit is also 2100° F (1150° C) for LEU fuel. For steady-state operation (non-adiabatic case) fuel temperatures are dependent upon the heat transfer characteristics of the element and coolant, thus, an experimental limit on power density is selected to insure fuel integrity. This limit is well below the maximum allowable power density which corresponds to a heat flux value at which there is a departure from nucleate boiling. The maximum steady-state power density generated in the Torrey Pines TRIGA Mark III is 32 kW per element and the maximum generated in the NSCR is approximately 18 kW per element, for either LEU or FLIP cores. Since the Texas A&M TRIGA pool depth is approximately six feet greater, improved cooling characteristics are expected.

C. Evaluation of the Limiting Safety System Setting

The Limiting Safety System Setting (LSSS) is established to insure that the safety limits will not be reached. A peak core temperature of 950°C in FLIP and LEU fuel is established for all modes of operation to provide a minimum safety margin of 200°C. Since the LSSS responds to a temperature measured in an instrumented fuel element, the location of this element must be considered. The LSSS can be established once the ratio of the temperature in the maximum power density element to the temperature in the instrumented fuel element is determined. This ratio is not the same for the steady state and pulsing cases so they will be considered separately.

1. Thermocouple Location

There is a trade-off to be considered in determining the location of the instrumented element. If complete freedom is to be allowed in the positioning of the thermocouple then the LSSS must be

set low to allow for positions with a low power density. Experience has shown that this approach unnecessarily restricts operations. Maximum latitude in operations can be attained by specifying the exact thermocouple location but this would generate rigid specifications for the core configurations. It is believed that a satisfactory compromise has been found. By specifying that the thermocouple will be located adjacent to the central bundle excluding the corner positions, eight locations are allowed. For the steady state analysis it is the ratio of the total power produced in the highest power element to that produced in the instrumented element that determines the variation in thermocouple responses. These ratios have been computed for both FLIP and LEU cores. The maximum value obtained from the eight possible locations was calculated for both FLIP and LEU cores at a power level of 1 MW. These values are shown below in Table VIII.

TABLE VIII
POWER RATIOS FOR THE THERMOCOUPLE LOCATIONS
ALLOWED BY TECHNICAL SPECIFICATIONS

<u>Core</u>	<u>Pwr produced in max Pwr element/ Pwr produced in thermocouple element</u>
FLIP	1.17
LEU	1.18

The value generated in the previous SAR for an all FLIP core was 1.18 which agrees well with the new computation. The value for the LEU core is comparable to that for the FLIP core so similar limits are expected. Contour maps of the power distributions used in this analysis are shown in Figures 11-1 and 11-2 for FLIP and LEU fuel.

For pulsing mode operation, the variation in the thermocouple response is determined by the ratio of the peak temperature rise in the core to the temperature rise at the thermocouple location. The temperature must be calculated for the hottest fuel element for a particular insertion, then the lowest temperature in a possible thermocouple location must be calculated for the same insertion. This calculation is described in a later section.

2. The LSSS for Steady State Operation

To determine the limiting safety system setting, NCTRIGA was run for varying power levels (using power distributions from DIF3D), until the peak fuel element attained a centerline temperature of 950°C. The code was then run to determine the peak temperature at the same power level for the element with the lowest peaking factor in an allowed instrumented fuel rod position.

For FLIP fuel, the power level necessary to produce a peak core temperature of 950°C was 2800 kW. This gave a lowest possible instrumented element peak temperature of 816°C. Since the highest observed ambient temperature for the pool is 37°C, the peak temperature rise of this instrumented element is:

$$\Delta \hat{T}_{TC} = 816^{\circ}C - 37^{\circ}C = 779^{\circ}C$$

In steady state mode the radial temperature distribution through a fuel element is not highly sensitive to the power density distribution. Thus, the radial temperature distribution which was

Figure 11-1 Critical Power Map - FLIP Fuel

Figure 11-2 Critical Power Map - LEU Fuel

calculated for the PRNC reactor is applicable to the NSCR. For this case the ratio of the maximum temperature, which occurs at the center of the fuel element, to that at the thermocouple location is 1.16. Thus, for the element discussed above, the thermocouple should read

$$\frac{779^{\circ}\text{C}}{1.16} = 672^{\circ}\text{C above ambient}$$

Thus, for the FLIP core using the new methods, the LSSS is

$$\text{LSSS} = 672^{\circ}\text{C} + 37^{\circ}\text{C} = 709^{\circ}\text{C}$$

This is more conservative than the value of 728°C presented in the previous SAR.

LSSS value for a LEU core is obtained in a similar manner and is listed in Table IX.

TABLE IX

STEADY STATE VALUES OF THE LSSS FOR DIFFERENT CORE CONFIGURATIONS

<u>Core</u>	<u>Steady State LSSS</u>
FLIP	709°C
LEU	730°C

For steady state operation at 1MW, a LSSS of 730°C should be set for the LEU core. This will insure that the peak core temperature will not exceed 950°C when the instrumented element is placed in any of the eight approved positions.

3. The LSSS and Maximum Allowable Reactivity Insertion for Pulsing

The temperature scram that will occur when the LSSS is reached cannot prevent a pulse from causing the safety limit to be exceeded. This control is achieved by limiting the allowed reactivity insertion. However, fuel element failure could possibly be prevented in the event of a pulsing accident if a scram occurred when the peak fuel temperature reached 950°C for the FLIP or LEU cores. This would reduce the total energy produced in the peak fuel element by clipping the pulse "tail." Since establishing the LSSS on this basis does not restrict steady state operation at 1 MW, this conservative approach is applied.

The minimum LSSS for pulsing was determined for the thermocouple location adjacent to the central fuel bundle that establishes the largest value for the power ratio ($\hat{P}/P_{TC}$) for each core examined. This power ratio is the ratio of the maximum core power, $\hat{P}$, to the power at the thermocouple location, P_{TC} .

To calculate this power ratio, PARET was run using the power distributions for the hot channel generated by DIF3D for prepulsing conditions. Since it is well known that the end of fuel life pulses produce the highest temperatures for a given reactivity insertion, only cases at end of core life were examined.

For the FLIP core, several pulses were modeled until the peak core temperature of 950°C was obtained. The reactivity insertion required to produce a peak core temperature of 950°C was found to be \$2.49. This compares well with the previously accepted value of \$2.48. A plot of the pulse power and energy is shown in Figure 11-3, and a plot of the temperatures of the hot channel is shown in Figure 11-4. PARET was then run to determine the temperature response at the location of the lowest power instrumented location. This gave a peak temperature of 811°C. The allowed temperature of the thermocouple must then be determined from this. Using the numerical fit from Chapter III

$$R(811^{\circ}\text{C}) = 0.5066 + 4.046 \times 10^{-3} \times (811) - 3.32667 \times 10^{-6} \times (811)^2$$

$$R(811^{\circ}\text{C}) = 1.60$$

so the temperature difference at the thermocouple is

$$\frac{811^{\circ}\text{C} - 37^{\circ}\text{C}}{1.60} + 37^{\circ}\text{C} = 521^{\circ}\text{C}$$

This is more conservative than the value calculated (and accepted) in the previous SAR of 544°C for an all FLIP core.

For the LEU core, the maximum reactivity insertion to produce a peak temperature of 950°C is \$3.42, which is higher than that allowed for a FLIP core. This difference is due to the different temperature coefficients of reactivity for the two fuel types. A plot of the pulse power and energy is shown in Figure 11-5, and a plot of the temperatures of the hot channel is shown in Figure 11-6. When the same reactivity insertion (\$3.42) was examined for the allowed thermocouple location with the lowest temperature, the thermocouple was evaluated from PARET to reach a peak temperature of 753°C. When the temperature ratio for pulsing is applied to this, the LSSS becomes

$$\frac{753^{\circ}\text{C} - 37^{\circ}\text{C}}{1.67} + 37^{\circ}\text{C} = 466^{\circ}\text{C}$$

For further details on this calculation, see Appendix IV.

The LSSS and maximum allowed reactivity insertion for FLIP and LEU fuels are presented in Table X.

TABLE X

PULSING VALUES OF THE LSSS FOR DIFFERENT CORE CONFIGURATIONS

<u>Core</u>	<u>Pulsing LSSS</u>	<u>Maximum Allowed Reactivity Insertion</u>
FLIP	521° C	\$2.49
LEU	466° C	\$3.42

Figure 11-3 Limiting FLIP Pulse

Figure 11-4 Limiting FLIP Pulse Temperatures

Figure 11-5 Limiting LEU Pulse

Figure 11-6 Limiting LEU Pulse Temperatures

The LSSS values obtained in this manner are considerably lower than those derived for the steady state mode. The current LSSS of 525°C for FLIP cores remains valid, and the LSSS to be applied for LEU cores is 466°C.

E. Accidental Pulse at Full Power

It is necessary to examine this situation in spite of the interlocks which are intended to prevent this from happening. The main objective is to consider the lowest pulse that could produce a peak temperature in the core of 950°C for both FLIP and LEU fuel. For the FLIP fuel, the insertion was considered at the end of fuel life, 3000 MWD. The reactor was modeled in a critical configuration with the transient rod fully inserted. The power distribution data from DIF3D was then used to create a PARET input deck, and several pulses were simulated until a peak fuel temperature of 950°C was reached. The insertion to cause this temperature was \$2.78. This is slightly higher than the accepted value of \$2.42 predicted by GA in the previous SAR. In either case, the insertion limit imposed by the 300 W pulsing case is more restrictive.

For an LEU core, since the fuel life is considerably shortened, the reactor cannot be placed in a critical configuration with the transient rod inserted, except at the beginning of life. After a short burnup period of only a few hundred MWD, enough negative reactivity has been added to the core to prevent criticality unless the transient rod is withdrawn. The pulse from 1 MW was therefore examined at the beginning of fuel life. Due to the large negative temperature coefficient at the beginning of life shown in figure 3-29, the reactor cannot reach a peak temperature of 950°C, even if the transient rod is set for a maximum insertion. Therefore, this is not a limiting scenario.

F. Control Rod Run-Out

The magnitude of the result of the withdrawal of all control rods at maximum speed was considered for the PRNC reactor. The magnitude of the effect of this accident is dependent primarily on the speed of rod withdrawal and the value of the temperature coefficient. The PRNC reactor consisted of FLIP fuel, but as seen in figure 3-31, the temperature coefficients for FLIP and LEU fuel are comparable. Since the control rod worths are also comparable, any analysis performed on the PRNC reactor is applicable to the NSCR.

For the calculation of the PRNC control rod run-out accident the withdrawal time used was 16.2 seconds. The withdrawal time of the shim-safety rods of the NSCR reactor is .347 minutes (20.82 seconds). The NSCR is set to scram at 1.25 MW (or less) as opposed to 2.2 MW for the PRNC reactor. Since the temperature coefficient will be the same or larger and the control rod removal rate is also slower, the increase in fuel temperature with reactor power level will compensate for the rod insertion so that the excess reactivity will be maintained near zero. Thus, when the reactor trip occurs the core temperature will nearly correspond to the case of a reactor operating at steady state. The maximum power generated in any cell will be approximately 25 kW which is well below the maximum permitted.

G. Loss of Coolant Accident

If the reactor pool is accidentally drained of water, the fission product decay heat will be removed primarily by natural convection of air. An analysis was performed by GA for FLIP fuel for the NSCR. Since the temperature limit on LEU fuel is higher than that of FLIP fuel, and the expected lifetime of the fuel is less, any calculations performed on the FLIP fuel are applicable to LEU fuel. If the decay-heat production is sufficiently low or if there is a long enough interval between reactor shutdown and coolant loss, the convective cooling by air will be enough to maintain

the fuel at a temperature which will not damage the fuel elements. The analysis of this accident for the NSCR yielded the following results:

- a. The maximum temperature that FLIP fuel can tolerate in air without damage to the clad and subsequent release of fission products is 940°C. For LEU fuel, the limit is 950°C.
- b. This temperature will not be exceeded under the conditions of coolant loss if the maximum power density in an element is equal 23 kW even if the reactor is operated for an infinite time prior to the accident.
- c. If reactor operations are limited to 70 MW-hrs per week, power densities up to 28 kW/element will not cause element damage in the event of loss of coolant. The calculations that produced the above results are presented in Appendix II.

H. Design Basis Accident

The Design Basis Accident 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.

This analysis was performed in detail for the FLIP core in the previous SAR. Since the power density limits and thermal conductivities of the two fuels are similar, the centerline and fuel surface temperatures for the limiting cases are similar. In addition, the maximum power density in the LEU 20-20 fuel for the NSCR LEU core is considerably less than the power density limit of 28 Kw used in the Design Basis Accident analysis in the earlier SAR. Finally, the FLIP fuel has a much longer fuel life than the LEU fuel, so a smaller fission product inventory will accumulate in the LEU fuel. For these reasons, the FLIP analysis is more conservative than an LEU analysis. Since this analysis was accepted in the previous SAR, it is presented here as an extreme case for the LEU fuel.

The fission product release fraction as determined experimentally by General Atomic is a property of the fuel and its operating temperature and is not dependent on facilities. To determine the operating temperature of the failed element it was assumed that it was operating at the highest power density permitted by the proposed Technical Specifications. For limited operation the maximum power density is 28 Kw for FLIP fuel. Since all significant power is generated in the steady state mode it is the temperatures in this mode that are required. An experiment was performed to obtain the correlation between steady state fuel element temperatures and power density since the heat transfer calculations are subject to large errors. Fuel temperatures were obtained at several locations in NSCR Core II - G for which the power density had been calculated by Exterminator-2. The results shown in Figure 11-7 show the maximum temperature in the fuel element rather than the temperatures measured by the thermocouple. These results are superimposed on calculations which were made for the Puerto Rico Nuclear Center Reactor as well as a single experimental datum point which was obtained from that facility. It can be seen that the linear extrapolation used to extend the NSCR data is conservative when compared to those numbers. A linear extrapolation yields a value of 535° C for the maximum temperature at 28 Kw generated in an element. With a maximum temperature of 535° C and a power density of 28 Kw/element the "minimum" or surface temperature would be 150° C. The fission product release fraction averaged over the fuel volume is 2.6×10^{-5} . The saturated activities of the significant fission products at 1 MW in a single fuel element are:

Total iodine fission products - 6,432 curies
Total halogen fission products - 7,611 curies
Total gaseous fission products - 10,760 curies

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

Total gaseous activity - 280 mc
Total iodine activity - 167 mc
Total halogen activity - 198 mc

If the release accident occurred with the pool water in place the halogens will remain in the water. The resulting concentration would be $3.65 \times 10^{-4} \mu\text{c}/\text{cm}^3$. Within 24 hours this value would decay to $8.34 \times 10^{-5} \mu\text{c}/\text{cm}^2$. These soluble fission products would be removed by the demineralizer and disposed of as liquid wastes.

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

Figure 11-7 Steady State Fuel Temperature as a Function of Power Generation

TABLE XII

Summary of Radiation Exposure Following Failure of
CLAD of the Highest Power Density FLIP Fuel Element Cladding

A. Building Ventilation Operating:

1. Maximum Exposure to Population Outside Building

	<u>WBGD*</u>	<u>Thyroid Dose</u>
Pool Water Remaining	3.5×10^{-3}	--
Pool Water Drained	1.4×10^{-2}	3.7 mr

2. Exposure to Operating Personnel in One Hour After Release

	<u>WBGD</u>	<u>Thyroid Dose</u>
Pool Water Remaining	0.84 mr	--
Pool Water Drained	1.75 mr	10.5 r

B. Building Ventilation Shut Down

1. Maximum Exposure to Population Outside

Building (12-hours)

	<u>WBGD</u>	<u>Thyroid Dose</u>
Pool Water Remaining	3.6×10^{-3} mr	--
Pool Water Drained	2.1×10^{-2} mr	18 mr

2. Exposure to Operating Personnel in One Hour After Release:

	<u>WBGD</u>	<u>Thyroid Dose</u>
Pool Water Remaining	1.75 mr	--
Pool Water Drained	4.2 mr	49 r

*WBGD - Whole Body Gamma Dose

I. Effects of Experimental Facilities on Reactivity

Before any new experiment involving the reactor is performed approval must be obtained from the Reactor Safety Board. This committee reviews all experiments for safety and for compliance with the operating license and NRC regulations. Experiments affecting the reactivity of the core are not to be loaded, unloaded or moved without the permission of the Reactor Supervisor.

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 is believed to exist. The reactivity worth of any non-secured experiment shall have reactivity worths less than \$1.00. This specification is intended to provide assurance that the worth of a single unfastened experiment will be limited to a value such that the safety limit will not be exceeded if the positive worth of the experiment were to be suddenly inserted. 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 from being attained.

J. Irradiation of Explosives in Beam Port 4 Neutron Radiography Facility

In February, 1979 a proposal was made to the USNRC to modify the Technical Specifications to Facility License No. R-83 to permit the irradiation of up to five pounds of explosive material in the radiography facility utilizing Beam Port No. 4 of the NSCR. The safety evaluation is presented in Appendix III.

K. Conclusion

The previous analyses and discussions show that the TRIGA core is inherently safe and reliable. This has been clearly demonstrated by the numerous TRIGA's that have logged many hours without a significant incident. Since the NSCR control system has also proven to be safe and reliable through years of operation, it is believed that we can clearly operate in a manner that will not endanger operating personnel or the public. The analyses for the LEU type fuel shows that no significant operational changes are necessary in order to convert from FLIP fuel.

Appendix I - WIMSD4/m and DIF3D Modeling

A. WIMSD4/m Modeling

The code selected for the generation of few group neutron cross sections representative of various subregions in the core and its surroundings is WIMSD4/m¹, a one-dimensional neutron transport code capable of solving for flux distributions using many fine energy groups. WIMSD4/m is an advanced version of WIMSD4. The associated cross section data library has three new isotopes: hydrogen in ZrH, Er¹⁶⁶, and Er¹⁶⁷. WIMSD4/m reads 69 group cross section data for each material present, and creates a spatially and energetically averaged set of cross sections in the seven standard energy groups used in TRIGA analyses for each subregion of the core. These cross sections are stored in the ISOTXS format for use by DIF3D, a three dimensional finite difference multigroup computer code². The steps followed in the calculational strategy is presented in the flow chart shown in Figure I-1.

Since neutron cross sections are highly temperature dependent, WIMSD4/m must be run for many cases covering the entire operational temperature range for each type of subregion in the core for both normal and accident conditions. The composition of the core components and techniques used for the WIMS modeling are described below.

1. Fuel Rod Model

Both the FLIP and LEU fuel rods have identical geometry and differ only in the composition of the fuel material. This geometry can be seen in Figure I-2. Axially, the fuel element is modeled as 38.10 cm (15 in.) of fuel material, capped on both ends by 8.89 cm each of graphite reflector. The bottom fitting is 8.89 cm of stainless steel; the top fitting is 11.43 cm of stainless steel and aluminum.

1.1 Fuel Cell

At the center of the fuel meat is a 0.4572 cm zirconium rod. This rod is surrounded by fuel-moderator material with an outer diameter of 3.48234 cm. The meat is surrounded by a stainless steel clad with a thickness of 0.0508 cm. The rest of the 3.9520 cm x 3.9520 cm cell is filled with water. The atom density for each isotope in each material is listed in Table I-1. Information is given in the table for both FLIP and LEU fuel rods. A sample WIMS input deck is shown in Table I-2. A brief description of the input cards in Table I-2 follows.

- * TRIGA Flip Fuel Cell --- 500 MWD Burnup Test

This card is a short description of this job. The "*" indicates that the following words in the same line are a comment.

- CELL i

This defines the cell type.

- i = 4 : homogeneous
- i = 5 : pin-cell without energy condensation
- i = 6 : pin-cell with energy condensation
- i = 7 : cluster

- SEQUENCE i

This defines the main transport routine to be used.

- i = 1 : DSN

¹ C. I. Costescu, W. L. Woodruff and D. G. Cacuci, WIMSD4m --- WIMS with Isotopes Capability Basic Information for Users, University of Illinois at Urbana-Champaign, Nuclear Engineering Department, 1992.

² K. L. Derstine, DIF3D : A Code to Solve One-, Two- and Three-Dimensional Finite-Difference Diffusion Theory Problems, ANL-82-64, Argonne National Laboratory, Argonne, Illinois. 1984.

--- i = 2 : PERSEUS
 --- i = 4 : PIJ-PERSEUS
 --- i = 5 : PRIZE-PERSEUS

- **NGROUP i j**

i : the number of main transport routines groups
 j : the number of reaction edit groups
 Here i is set to 23 and j is set to 7 in the first deck of Table I-2.

- **NMESH i j**

i : the number of main transport routines mesh points
 j : the number of edit mesh points
 Here both i and j are set to 15 in the first deck of Table I-2.

- **NREGION i j k**

i : the number of slabs or annuli
 j : the number of annuli containing rods (default = 0)
 k : the number of edit regions (default = i + 3j)
 Here i, j and k are set 4, 0 and 4, respectively in the first deck of Table I-2.

- **NMATERIAL i j**

i : the total number of problem materials
 j : the number of materials which undergo burnup
 Here i is set to 4 and j is set to 1 in the first deck of Table I-2..

- **PREOUT**

This must be the last prelude data card.

- **INITIATE**

The first card of a complete set of main data.

- **ANNULUS j r m**

j : the annulus number counting from the centre
 r : the outer radius in cm
 m : the material number

For example, the card " ANNULUS 1 0.22860 1", means annulus number 1 has a radius of 0.22860 cm and is composed of material number 1. The other two ANNULUS cards in Table I-2 are similar to the above.

- **SQUARE j r m**

This card is similar to the ANNULUS card. The definitions of j and m are the same as on the ANNULUS card. The definition of r is the radius (in cm) of the inscribed circle of the square. Here j, r and m are set to 4, 1.9760 and 4. respectively.

- **MATERIAL m -l t n list**

m : the material number.
 -l : the type of input variables
 t : the material temperature in degrees Kelvin
 n : the spectrum type (1 : fuel, 2 : can, 3 : coolant, 4 : moderator)
 list : consists of pairs of nuclide identifier and atom density (in atoms/barn-cm)
 For example, the first deck of Table I-2:

```

MATERIAL 2 -1 650 1 235.2 8.8382E-4 2238.4 3.7878E-4 2191 0.054363 $
91 0.033977 166.1 1.0560E-4 1167.1 7.2132E-5 3239.1 1.0E-18
  
```

The above card shows that material number 2 is 650 K fuel composed of uranium 235 (identifier is 235.2, atom density is 8.8382E-4 atoms/barn-cm), uranium 238 (2238.4, 3.7878E-4), hydrogen in ZrH (2191, 0.054363), zirconium 91 (91, 0.033977), erbium 166 (166.1, 1.0560E-4), erbium 167 (1167.1, 7.2132E-5), and a small amount of plutonium 239 (3239.1, 1.0E-18).

There are several interesting things about this card. First of all, there is a dollar sign (\$) at the end of the first line, which means the next line is a continuation of the first line. Secondly, the burnable nuclides, like uranium 235, 238, erbium 166, 167, and plutonium 239, have different versions of cross section data. Erbium 166 and 167, currently, only have .1 versions. The 2238.4 version of uranium 238

and the 3239.1 version of plutonium 239 are recommended. Uranium 235 has .2, .3 and .4 versions. 235.2 is the original version created by the British. 235.3 and 235.4 are alternative versions of 235.2, which were created in the United States. The different versions of uranium 235 can cause some variance in multiplication factor. The last thing worth noting about this card is that a small amount ($1.0\text{E-}18$ atoms/barn-cm) of plutonium 239 has been added to the fresh fuel in order to obtain cross sections for plutonium that builds up as the fuel burns.

- FEWGROUPS list

This list specifies how the library groups are divided during the calculations of the main transport routine. This card in Table I-2 shows how the 69 groups are condensed to 23 groups before the main transport routine performs its calculations. Table I-3 presents a list of the energy boundaries for the 69 "fine" groups.

- MESH list

List is the number of mesh intervals of equal volume to be used in each of the annuli or slabs. In deck 1 of Table I-2, this card reads "MESH 2 9 1 3" which means that there are 2, 9, 1 and 3 equal-volume mesh intervals in regions 1, 2, 3 and 4, respectively.

- POWERC INDQ RQ RTAU INDNB

This card is used to control the power level for burnup calculations.

INDQ : units for RQ

--- INDQ = 1 : MW/Te heavy elements

--- INDQ = 2 : fissions/sec-cm³ averaged over the cell

--- INDQ = 3 : fissions/sec-cm³ averaged over burnable materials

--- INDQ = 4 : total flux

RQ : value of power level based on the unit of INDQ

RATU : the time step between successive criticality calculations (in days)

INDNB : the number of timesteps

In the first deck of Table I-2, INDQ, RQ, RATU and INDNB have values of 2, 5.835E11, 100 and 1, respectively.

The power level of 5.835E11 fissions per second per cm³ corresponds to the fission rate in the homogenized fuel cell based on the following information :

--- power level : 1 MW

--- volume of cells : 3.952 cm x 3.952 cm x 38.10 cm (15 in) x 90 (fuel rods)

--- energy released in one fission : 200 MeV

- BEGNC

The final main data card.

- PARTITION list

List specifies the combination of fine library groups required in the edit of the "broad" group cross sections.

IN the first deck of Table I-2, this card is used to condense the 69 fine groups to 7 broad groups.

- REGION n m

The cell edit will be taken over regions from n to m. In the first deck of Table I-2, we have "REGION 1 4" which indicates an edit should be produced over regions 1 to 4.

- OPTION i

The WED edit contains (i) flux per unit volume and per mesh interval, (ii) cross sections for each material x flux per unit volume and (iii) cross sections for each material x flux per mesh interval.

--- i = 0 : gives the full edit

--- i = 1 : gives items (i) and (iii)

--- i = 2 : gives item (i)

--- i = 3 : gives no output (default i = 3)

- ISOXS imacro imicro nburn

This card is used to specify the type of the output cross section data file.

imacro = 0 : macro ISOTXS cross sections will not be produced

1 : macro ISOTXS cross sections will be produced and stored in a binary file named fort.19

imicro = 0 : micro ISOTXS cross sections will not be produced

1 : micro ISOTXS non-transport corrected cross sections will be produced and stored in fort.20

2 : micro ISOTXS transport corrected cross sections will be produced and stored in fort.20
nburn : number of POWERC burnup cards

In the first deck of Table I-2, imacro, imicro and nburn are 1, 0 and 5, respectively. Since DIF3D only needs macroscopic cross section data, only the macro ISOTXS file is used in later calculations.

- VECTOR list

List is a partition vector referring to the main transport group structure. Notice that the condensed group structure is limited to 10 groups. This card in our case shows how to convert 23 groups to 7 groups. It must be consistent with the FEWGROUPTS and PARTITION cards.

- BEGINC

The last card of the edit data.

- POWERC 2 5.835E11 100 1

BEGINC

BEGINC

These three cards shown in the first deck of Table I-2 indicate one burnup step of 100 MWD is to be computed. Since this test is for the case of 500 MWD burnup, there are another 4 sets like these added at the end of the file. One interesting point is that there is another way to do the 5-100-MWD-step-burnup calculation, that is, using only one POWERC card : POWERC 2 5.835E11 100 5. The difference is that the flux spectrum of the latter case will not be updated as burnup steps are calculated, while the former case recomputes the spectrum after each 100 MWD step. The two methods do not create a significant difference for short burnup. create a significant change if the burnup is short time. For long burnup, the former method is more accurate.

1.2 Graphite End Reflector of Fuel Rod

The graphite end reflector cell has the same outside diameter and thickness for the clad as the fuel cell (see Figure I-2).

1.3 Top and Bottom Structural Materials

The top structure of the fuel rod includes a volume ratio of 24.7% for the stainless steel plug, 17.9% for the aluminum handle and 57.4% for the surrounding water. The estimation of these values is not exact, and the process for their calculation is not presented here. Similarly, in the bottom structure, approximately 32.5% by volume is stainless steel and 67.5% by volume is water.

To obtain the neutronic data for the top graphite end reflector and the top structure of a fuel rod, a four-region slab geometry WIMSD4/m case is constructed as shown in Figure I-3a. The WIMSD4/m input deck for this case is listed in Table I-4. In the similar way, Figure I-3b shows a five-region slab geometry WIMSD4/m case that was used to obtain the homogenized neutronic data for the bottom graphite end reflector and the bottom structure of a fuel rod. The WIMSD4/m input file for this case is listed in Table I-5. Notice that the homogenized neutronic data for the aluminum grid plate is also obtained from this WIMSD4/m case.

2. Control Rod Model

There are three types of control rods in NSCR: shim safety rods, a transient rod and a regulating rod. Their geometries are depicted in Figures I-4, I-5 and I-6, respectively. Details are described in the following three sections.

2.1 Shim Safety Rod

Each shim safety rod is composed of three portions; a 35.56-cm(14-in)-long poison region, a 38.10-cm(15-in)-long fuel follower under the poison, and two void regions on the top and bottom with 12.70 cm (5 in) and 17.78 cm (7 in) in length, respectively (see Figure I-3). The poison in the shim safety rod is

borated graphite (B₄C 25%, graphite 75% by weight) and its diameter is 3.3909 cm. The outside clad is 0.0508-cm-thick stainless steel. The fuel follower is similar to a fuel rod except that the fuel meat is 3.3909 cm in diameter. The void region has the same diameter as the poison region, but is filled with void instead of poison.

An example of the WIMSD4/m input file for the poison region of the shim safety rod cell is listed in Table I-6. It is worth noting that the control rod cell is surrounded by eight equal-sized fuel rod cells in the WIMSD4/m calculation and the cross sections for the shim safety rods are obtained by editing over only the center rod.

2.2 Transient Rod

The transient rod only has two portions: a 38.10-cm(15-in)-long poison region and a 50.80-cm(20-in)-long void region (see Figure I-5). The poison used in the transient rod is also borated graphite. The diameter of the region filled with poison is 3.03276 cm. The thickness of the stainless steel clad is 0.0712 cm. The void cell is very similar to the poison cell except that it is filled by void instead of poison.

2.3 Regulating Rod

The regulating rod is composed of a single 38.10-cm(15-in)-long poison region (see Figure I-6). The rod size is the same as that of the transient rod. The only difference is that the poison in the regulating rod is pure B₄C powder (density: 2.5 g/cm³).

3. Graphite Reflector Element

The size of the parallelepiped graphite reflector elements used on the north and south sides of the NSCR is 7.62 cm (3 in.) x 7.62 cm (3 in.) x 71.17 cm (28 in.) (in length). When placed next to the core, the graphite elements are surrounded by water. Therefore, in the area of the homogenized north and south graphite reflectors 92.94% by volume is graphite and the rest is water. To obtain the neutronic data for the graphite reflector from WIMSD4/m, a multicell configuration is set up as shown in Figure I-7. The WIMSD4/m input file for this case is listed in Table I-7. Notice that three of four sides of the graphite reflector unit are neighbored other graphite and only one neighbor is a fuel cell, since generally the graphite elements are located on the either the north or south of the core.

4. Water

The needed neutronic data for water is obtained by setting up a WIMSD4/m multicell case which is similar to that used to model the control rods. In this case, the central cell is filled by pure water (density: 1.0g/cm³, temperature: 300 K) and the other eight cells are fuel.

Figure I-1. Running WIMS and DIF3D Flow Chart

Figure I-2. Fuel Rod Model

Figure I-3.

a. Slab Model for Top Structure and Graphite End Reflector of Fuel Rod

b. Slab Model for Bottom Structure (Includes Grid Plate) and Graphite End Reflector of Fuel Rod

Figure I-4. Control Rod Cell Model : Shim Safety Rod

Figure I-5. Control Rod Cell Model : Transient Rod

Figure I-6. Control Rod Cell Model : Regulating Rod

Figure I-7. Graphite Reflector Element Model

B. DIF3D Modeling

1. Geometry of Core

To set up a DIF3D job for NSCR reactor, a full three dimensional model of the core and its surroundings was created. Based on Figure 3-2, a simplified core map profile in the x-y plane is shown in Figure I-8. In the core, there are eighty-six normal fuel rods, four shim safety rods with fueled followers, one transient rod and one regulating rod. The details of fuel rods and control rods were described previously in section A. Graphite reflectors are located on the north and south sides of the core. In the DIF3D model, the entire core is surrounded by a 35 cm thick region of water.

The NSCR core profile in the z-direction is shown in Figure I-9. The fuel region is 38.10 cm (15 in) long. On the top and bottom of the fuel region are the regions of the graphite end reflectors. At both ends of the graphite reflectors are the regions of structural material. At the bottom of the fuel rod structure is the region of the grid plate. Finally, the top and bottom of the entire core is surrounded by water. Two cases of control rod position -- "all rods in" and "all rods out" are also indicated in Figure I-9. Notice that, in the axial region containing the fuel, there are fifteen equal-length fine meshes in the z-direction, that is, the control rod positions relative to the fuel can be adjusted individually inch by inch in z-direction.

2. Reactor Critical Test

After the DIF3D core model for the core was set up, a critical situation from the operation with a FLIP core was computed. A set of experimental critical data from December, for the FLIP core was used. The estimated core burnup map for the FLIP fuel based on the December 1993 critical data just mentioned is shown in Figure I-10. The average core burnup is approximately 1400 MWDs. The control rod positions are that all shim safety rods are 66% out, the transient rod is 100% out and the regulating rod is 45% out.

According to this information, an input deck for DIF3D was set up and it is listed in Table I-8. The results were that the multiplication factor, $K_{\text{effective}}$ is found to be 1.00253616 which is \$ 0.3623 (\$ 1.00 = 0.007 $\Delta k/k$) higher than the expected critical situation. This value of $K_{\text{effective}}$ is defined as our "critical" multiplication factor for computing core life and control rod worths.

3. Calculation of Burnup

Figure I-11 shows $K_{\text{effective}}$ versus full power time for the FLIP core and the LEU 20-20 core with an initial erbium content of 0.590% by weight in the fuel meat. The life time of the LEU core is approximately 3.7 MWYs.

4. Calculation of Temperature Coefficient of Fuel

Since WIMS cross section data library has thermal scattering kernels for ZrH only at temperature 25, 200, 500 and 750°C, the temperature coefficients of the fuel at these four were calculated and used to sketch the change of the temperature coefficient of reactivity with the change of the fuel temperature. Table I-9 shows the negative fuel temperature coefficient at these four temperatures and at several burnup values for both FLIP and LEU fuels. Based on these data, the relationship for the prompt negative temperature coefficient versus the fuel temperature were obtained. These data are plotted in Figure 3-31.

5. Estimation of Control Rod Worth

To estimate the control rod worth, first of all, an all-rods-in situation is calculated. Then the change of $K_{\text{effective}}$ for one totally withdrawn control rod is obtained by running DIF3D. Table I-10 and I-11 show the estimated control rod worth for the FLIP core at 1400 MWDs burnup and the fresh LEU core, respectively. Table I-10 also lists the values obtained by experiment at the NSCR.

6. Estimation of Shutdown Margin Calculation

The shutdown margin is defined by the NSCR technical specifications as the remaining control rod worth at a power of 300 watts after deductions are made for the worths of the most reactive control rod, a non-secured experiment that is assigned a worth of \$1.00 and the worth of the non-scrammable rods. In the case of the NSCR, the non-scrammable rod is the regulating rod and the control rod with the highest worth is shim safety rod number 4.

The estimation of the control rod worth has already been described in the previous section. The next step is to find core excess at 300 watts. First, the critical control rod positions at 300 watts are found. Then based on how much each control rod is inserted for the critical case at 300 watts, the shutdown margin can be easily calculated from its definition.

Table I-12 and I-13 show the results for the calculated shutdown margins for the FLIP core at 1400 MWDs burnup and for the fresh LEU fuel, respectively. Table I-12 also lists the values obtained by experiment at NSCR.

Figure I-8. Core Map Profile at X-Y Plane

Figure I-9. Control Rods In-core Profile in Z-direction

Figure I-10.

Flip Core Burnup Map Profile

Based on the Experimental Data of December 1993

Figure I-11. K-effective versus Burnup

Table I-1.

Materials	Components	Atom Densities (atoms/barn-cm)
Zirconium Rod	Zirconium	4.2909E-2
Flip Fuel Meat	Uranium-235	8.8382E-4
	Uranium-238	3.7878E-4
	Hydrogen in ZrH	5.4364E-2
	Zirconium	3.3977E-2
	Erbium-166	1.0560E-4
	Erbium-167	7.2132E-5
LEU 20-20 (Er: 0.59%) Fuel Meat	Uranium-235	6.5482E-4
	Uranium-238	2.6140E-3
	Uranium-234	4.9946E-6
	Uranium-236	8.2393E-6
	Hydrogen in ZrH	5.4540E-2
	Zirconium	3.5260E-2
	Erbium-166	4.4216E-5
	Erbium-167	3.0202E-5
	Carbon	1.3029E-3
Graphite	Carbon	8.1299E-2
Clad of Fuel Rod	Iron	6.0414E-2
	Chromium	1.7384E-2
	Nickel	8.1033E-3
	Carbon	3.1716E-4
SS304	Iron	5.9206E-2
	Chromium	1.7385E-2
	Nickel	7.6995E-3
	Manganese	1.7320E-3
Aluminum	Aluminum	6.0240E-2
Water	Hydrogen in H ₂ O	6.6691E-2
	Oxygen	3.3346E-2

Table I-2

* TRIGA Flip Fuel Cell --- 500 MWD Burnup Test

CELL 6

SEQUENCE 2

NGROUP 23 7 7

NMESH 15 15

NREGION 4 0 4

NMATERIAL 4 1

PREOUT

INITIATE

ANNULUS 1 0.22860 1

ANNULUS 2 1.74117 2

ANNULUS 3 1.79197 3

SQUARE 4 1.97600 4

MATERIAL 1 -1 650 2 91 .042909

MATERIAL 2 -1 650 1 235.2 8.8382E-4 2238.4 3.7878E-4 2191 0.054363 \$

91 0.033977 166.1 1.0560E-4 1167.1 7.2132E-5 3239.1 1.0E-18

MATERIAL 3 -1 300 2 56 0.060414 52 0.017384 58 0.0081033 \$

12 0.00031716

MATERIAL 4 -1 300 3 2001 0.0666913 16 0.0333456

FEWGROUPS 5 6 14 15 21 25 26 27 30 33 34 35 38 45 47 52 55 58 59 \$

60 63 66 69

MESH 2 9 1 3

POWERC 2 5.835E11 100 1

BEGINC

PARTITION 6 14 34 47 55 60 69

REGION 1 4

OPTION 3

ISOXS 1 0 5

VECTOR 2 3 11 15 17 20 23

BEGINC

POWERC 2 5.835E11 100 1

BEGINC

BEGINC

POWERC 2 5.835E11 100 1

BEGINC

BEGINC

POWERC 2 5.835E11 100 1

BEGINC

BEGINC

POWERC 2 5.835E11 100 1

BEGINC

BEGINC

TABLE 3 I-3

69-GROUP ENERGY BOUNDARIES FOR WIMS

GROUP	ENERGY		ENERGY WIDTH	LETHARGY WIDTH	GROUP	ENERGY eV		ENERGY WIDTH	LETHARGY WIDTH
	<u>MeV</u>								
1	10.0	- 6.0655	3.9345	0.49997	28	4.00	- 3.30	0.700	0.19237
2	6.0655	- 3.679	2.3865	0.49998	29	3.30	- 2.60	0.700	0.23841
3	3.679	- 2.231	1.448	0.50019	30	2.60	- 2.10	0.500	0.21357
4	2.231	- 1.353	0.878	0.50013	31	2.10	- 1.50	0.600	0.33647
5	1.353	- 0.821	0.532	0.49956	32	1.50	- 1.30	0.200	0.14310
6	0.821	- 0.500	0.321	0.49592	33	1.30	- 1.15	0.150	0.12260
7	0.500	- 0.3025	0.1975	0.50253	34	1.15	- 1.123	0.027	0.02376
8	0.3025	- 0.183	0.1195	0.50260	35	1.123	- 1.097	0.026	0.02342
9	0.183	- 0.1110	0.072	0.49996	36	1.097	- 1.071	0.026	0.02399
10	0.111	- 0.06734	0.04366	0.49978	37	1.071	- 1.045	0.026	0.02458
11	0.06734	- 0.04085	0.02649	0.49985	38	1.045	- 1.020	0.025	0.02421
12	0.04085	- 0.02478	0.01607	0.49987	39	1.020	- 0.996	0.024	0.02381
13	0.02478	- 0.01503	0.00975	0.49999	40	0.996	- 0.972	0.024	0.02439
14	0.01503	- 0.009118	0.005912	0.49980	41	0.972	- 0.950	0.022	0.02289
					42	0.950	- 0.910	0.040	0.04302
					43	0.910	- 0.850	0.060	0.06821
					44	0.850	- 0.780	0.070	0.08594
					45	0.780	- 0.625	0.155	0.22154
					46	0.625	- 0.500	0.125	0.22314
15	911.8.0	- 5530.0	3588.0	0.50006	47	0.500	- 0.400	0.100	0.22314
16	5530.0	- 3519.1	2010.9	0.45198	48	0.400	- 0.350	0.050	0.13353
17	3519.1	- 2239.45	1279.65	0.45198	49	0.350	- 0.320	0.030	0.08961
18	2239.45	- 1425.1	814.35	0.45199	50	0.320	- 0.300	0.020	0.06454
19	1425.1	- 906.898	518.202	0.45197	51	0.300	- 0.280	0.020	0.06899
20	906.898	- 367.262	539.636	0.90395	52	0.280	- 0.250	0.030	0.11333
21	367.262	- 148.728	218.534	0.90396	53	0.250	- 0.220	0.030	0.12783
22	148.728	- 75.5014	73.2266	0.67797	54	0.220	- 0.180	0.040	0.20067
23	75.5014	- 48.052	27.4494	0.45187	55	0.180	- 0.140	0.040	0.25131
24	48.052	- 27.700	20.352	0.55085	56	0.140	- 0.100	0.040	0.33647
25	27.700	- 15.968	11.732	0.55085	57	0.100	- 0.080	0.020	0.22314
26	15.968	- 9.877	6.091	0.48038	58	0.080	- 0.067	0.013	0.17733
27	9.877	- 4.00	5.877	0.90391	59	0.067	- 0.058	0.009	0.14425
					60	0.058	- 0.050	0.008	0.14842
					61	0.050	- 0.042	0.008	0.17435
					62	0.042	- 0.035	0.007	0.18232
					63	0.035	- 0.030	0.005	0.15415
					64	0.030	- 0.025	0.005	0.18232
					65	0.025	- 0.020	0.005	0.22314
					66	0.020	- 0.015	0.005	0.26768
					67	0.015	- 0.010	0.005	0.40547
					68	0.010	- 0.005	0.005	0.69315
					69	0.005	- 0.	0.005	-

Table I-4

* Graphite End Reflector and Structural Material Above Core --- Slab Geometry

CELL 6
 NPLATE 1
 SEQUENCE 2
 NGROUP 23 7 7
 NMESH 32 32
 NREGION 4 0 4
 NMATERIAL 4 0
 PREOUT
 INITIATE
 SLAB 1 20.00 1
 SLAB 2 31.43 2
 SLAB 3 40.32 3
 SLAB 4 59.37 4
 MESH 8 8 8 8
 FEWGROUPS 5 6 14 15 21 25 26 27 30 33 34 35 38 45 47 52 55 58 59 \$
 60 63 66 69
 MATERIAL 5 -1 300 1 235.2 5.2967E-4 2238.4 2.2700E-4 2191 3.2580E-02 \$
 91 2.0813E-02 2001 2.3615E-02 16 1.1808E-02 56 2.1809E-03 \$
 58 2.9253E-04 52 6.2756E-04 12 1.1449E-05 166.1 6.3286E-5 \$
 1167.1 4.3229E-5
 MATERIAL 3 -1 300 3 12 4.9577E-2 56 2.1375E-3 52 6.2763E-4 \$
 58 2.7797E-4 55 6.2528E-05 2001 2.3652E-2 16 1.1826E-2
 MATERIAL 2 -1 300 3 2001 3.8281E-2 16 1.9140E-2 52 4.2940E-3 \$
 56 1.4624E-2 58 1.9018E-3 55 4.2779E-4 27 1.0783E-2
 MATERIAL 1 -1 300 3 2001 0.0666913 16 0.0333456
 BEGINC
 OPTION 3
 PARTITION 6 14 34 47 55 60 69
 THERMAL 15
 REGION 1 1 2 2 3 3 4 4
 ISOXS 1 0 1
 HSETID
 TOP GRAPHITE END REFLECTOR AND STRUCTURAL MATERIAL
 VECTOR 2 3 11 15 17 20 23
 BEGINC

Table I-5

* Grid Plate, Graphite End Reflector and Structural Material Below Core --- Slab Geometry

```

CELL 6
NPLATE 1
SEQUENCE 2
NGROUP 23 7 7
NMESH 40 40
NREGION 5 0 5
NMATERIAL 5 0
PREOUT
INITIATE
SLAB 1 20.00 1
SLAB 2 32.70 2
SLAB 3 41.59 3
SLAB 4 50.48 4
SLAB 5 69.53 5
MESH 8 8 8 8 8
FEWGROUPS 5 6 14 15 21 25 26 27 30 33 34 35 38 45 47 52 55 58 59 $
60 63 66 69
MATERIAL 5 -1 300 1 235.2 5.2967E-4 2238.4 2.2700E-4 2191 3.2580E-02 $
91 2.0813E-02 2001 2.3615E-02 16 1.1808E-02 56 2.1809E-03 $
58 2.9253E-04 52 6.2756E-04 12 1.1449E-05 166.1 6.3286E-5 $
1167.1 4.3229E-5
MATERIAL 4 -1 300 3 12 4.9577E-2 56 2.1375E-3 52 6.2763E-4 $
58 2.7797E-4 55 6.2528E-05 2001 2.3652E-2 16 1.1826E-2
MATERIAL 3 -1 300 3 2001 4.4910E-2 16 2.2455E-2 52 5.6500E-3 $
56 1.9242E-2 58 2.5023E-3 55 5.6288E-4
MATERIAL 2 -1 300 3 2001 3.33456E-2 16 1.66728E-2 27 3.0120E-2
MATERIAL 1 -1 300 3 2001 0.0666913 16 0.0333456
BEGINC
OPTION 3
PARTITION 6 14 34 47 55 60 69
THERMAL 15
REGION 1 1 2 2 3 3 4 4 5 5
ISOXS 1 0 1
HSETID
GRID PLATE, BOTTOM GRAPHITE END REFLECTOR AND STRUCTURAL MATERIAL
VECTOR 2 3 11 15 17 20 23
BEGINC

```

Table I-6

* Shim Safety Rod Poison (Borated Graphite) --- Multicell Model

```

CELL 6
SEQUENCE 2
NGROUP 23 7 7
NMESH 28 28
NREGION 7 0 7
NMATERIAL 7 0
NCELLS 2
PREOUT
INITIATE
CELL 1 8
ANNULUS 1 0.22860 1
ANNULUS 2 1.74117 2
ANNULUS 3 1.79197 3
SQUARE 4 1.97600 4
MESH 2 9 1 3
PCELL 1 (0.875 0.125)
CELL 2 1
CSPECTRUM 1
PCELL 2 (1.0 0.0)
ANNULUS 1 1.69545 5
ANNULUS 2 1.74625 6
SQUARE 3 1.97600 7
MESH 9 1 3
MATERIAL 1 -1 650 2 91 .042909
MATERIAL 2 -1 650 1 235.2 8.8382E-4 2238.4 3.7878E-4 2191 0.054363 $
91 0.033977 166.1 1.0560E-4 1167.1 7.2132E-5
MATERIAL 3 -1 300 2 56 0.060414 52 0.017384 58 0.0081033 $
12 0.00031716
MATERIAL 4 -1 300 3 2001 0.0666913 16 0.0333456
MATERIAL 5 -1 300 2 11 0.0264239 12 0.0714147
MATERIAL 6 -1 300 2 56 0.060414 52 0.017384 58 0.0081033 $
12 0.00031716
MATERIAL 7 -1 300 3 2001 0.0666913 16 0.0333456
FEWGROUUPS 5 6 14 15 21 25 26 27 30 33 34 35 38 45 47 52 55 58 59 $
60 63 66 69
BEGINC
PARTITION 6 14 34 47 55 60 69
REGION 1 4 5 7
OPTION 3
ISOXS 1 0 1
HSETID
SHIM SAFETY ROD POISON
VECTOR 2 3 11 15 17 20 23
BEGINC

```

Table I-7

* Graphite Reflector Element --- Multicell Model

```

CELL 6
SEQUENCE 2
NGROUP 23 7 7
NMESH 25 25
NREGION 5 0 5
NMATERIAL 5 0
NCELLS 2
PREOUT
INITIATE
CELL 1 1
ANNULUS 1 0.22860 1
ANNULUS 2 1.74117 2
ANNULUS 3 1.79197 3
SQUARE 4 1.97600 4
MESH 2 9 1 3
PCELL 1 (0.75 0.25)
CELL 2 1
CSPECTRUM 1
PCELL 2 (0.25 0.75)
SQUARE 1 1.9760 5
MESH 10
MATERIAL 1 -1 650 2 91 0.042909
MATERIAL 2 -1 650 1 235.2 8.8382E-4 2238.4 3.7878E-4 2191 0.054363 $
91 0.033977 166.1 1.0560E-4 1167.1 7.2132E-5
MATERIAL 3 -1 300 2 56 0.060414 52 0.017384 58 0.0081033 $
12 0.00031716
MATERIAL 4 -1 300 3 2001 0.0666913 16 0.0333456
MATERIAL 5 -1 300 2 2001 5.1986E-3 16 3.0993E-3 12 7.5559E-1
FEWGROUPS 5 6 14 15 21 25 26 27 30 33 34 35 38 45 47 52 55 58 59 $
60 63 66 69
BEGINC
REGION 1 4 5 5
OPTION 3
PARTITION 6 14 34 47 55 60 69
ISOXS 1 0 1
HSETID
GRAPHITE REFLECTOR
VECTOR 2 3 11 15 17 20 23
BEGINC

```

Table I-8. Example of DIF3D Input Deck

```

BLOCK=OLD
DATASET=ISOTXS
BLOCK=STP021
UNFORM=A.DIF3D
01 CORE VIII-A - 1MW CRITICAL WITH STRUCTURAL MATERIAL (1993 BURNUP)
02 200000 2500000
03 0 0 0 4500 50 0 10000000000 5 0 0
04 0 0 0 00 000 11 111 1 1 0 0
05 1.0E-5 1.0E-5 1.0E-5
06 1.0 .001 .04 1.0E6
UNFORM=A.HMG4C
01
02 20000 0 0 0 0 0 0 0
UNFORM=A.NIP3
01
02 0 0 10000 0 20000 0 0 0 0 0
03 44
04 2 2 2 2 2 2
06 CORE      0.0      108.72      0 51      0.0      143.03
06 GRID      27.368    73.64      5 10      35.08     107.95
06 GRAFN     35.08     73.64     10 42      35.08     51.278
06 GRAFS     35.08     73.64     10 42      91.758    107.95
06 B7NW      35.08     38.936    18 33      51.278     55.326
06 B7NE      38.936    42.792    18 33      51.278     55.326
06 B7SW      35.08     38.936    18 33      55.326     59.374
06 B7SE      38.936    42.792    18 33      55.326     59.374
06 C7NW      42.792    46.648    18 33      51.278     55.326
06 C7NE      46.648    50.504    18 33      51.278     55.326
06 C7SW      42.792    46.648    18 33      55.326     59.374
06 C7SE      46.648    50.504    18 33      55.326     59.374
06 E7NW      58.216    62.072    18 33      51.278     55.326
06 E7NE      62.072    65.928    18 33      51.278     55.326
06 E7SW      58.216    62.072    18 33      55.326     59.374
06 E7SE      62.072    65.928    18 33      55.326     59.374
06 F7NW      65.928    69.784    18 33      51.278     55.326
06 F7NE      69.784    73.64     18 33      51.278     55.326
06 F7SW      65.928    69.784    18 33      55.326     59.374
06 F7SE      69.784    73.64     18 33      55.326     59.374
06 B6NE      38.936    42.792    18 33      59.374     63.422
06 B6SE      38.936    42.792    18 33      63.422     67.47
06 C6NW      42.792    46.648    18 33      59.374     63.422
06 C6NE      46.648    50.504    18 33      59.374     63.422
06 C6SE      46.648    50.504    18 33      63.422     67.47
06 D6NW      50.504    54.36     18 33      59.374     63.422
06 D6NE      54.36     58.216    18 33      59.374     63.422
06 D6SW      50.504    54.36     18 33      63.422     67.47
06 D6SE      54.36     58.216    18 33      63.422     67.47
06 E6NW      58.216    62.072    18 33      59.374     63.422
06 E6NE      62.072    65.928    18 33      59.374     63.422
06 E6SE      62.072    65.928    18 33      63.422     67.47
06 F6NW      65.928    69.784    18 33      59.374     63.422
06 F6NE      69.784    73.64     18 33      59.374     63.422
06 F6SW      65.928    69.784    18 33      63.422     67.47
06 F6SE      69.784    73.64     18 33      63.422     67.47
06 B5NW      35.08     38.936    18 33      67.47      71.518
06 B5NE      38.936    42.792    18 33      67.47      71.518
06 B5SW      35.08     38.936    18 33      71.518     75.566
06 B5SE      38.936    42.792    18 33      71.518     75.566
06 C5NW      42.792    46.648    18 33      67.47      71.518
06 C5NE      46.648    50.504    18 33      67.47      71.518
06 C5SW      42.792    46.648    18 33      71.518     75.566
06 C5SE      46.648    50.504    18 33      71.518     75.566

```


06 D5NW	50.504	54.36	18 33	67.47	71.518
06 D5NE	54.36	58.216	18 33	67.47	71.518
06 D5SE	54.36	58.216	18 33	71.518	75.566
06 E5NW	58.216	62.072	18 33	67.47	71.518
06 E5NE	62.072	65.928	18 33	67.47	71.518
06 E5SW	58.216	62.072	18 33	71.518	75.566
06 E5SE	62.072	65.928	18 33	71.518	75.566
06 F5NW	65.928	69.784	18 33	67.47	71.518
06 F5NE	69.784	73.64	18 33	67.47	71.518
06 F5SW	65.928	69.784	18 33	71.518	75.566
06 F5SE	69.784	73.64	18 33	71.518	75.566
06 B4NE	38.936	42.792	18 33	75.566	79.614
06 B4SE	38.936	42.792	18 33	79.614	83.662
06 C4NW	42.792	46.648	18 33	75.566	79.614
06 C4NE	46.648	50.504	18 33	75.566	79.614
06 C4SE	46.648	50.504	18 33	79.614	83.662
06 D4NW	50.504	54.36	18 33	75.566	79.614
06 D4NE	54.36	58.216	18 33	75.566	79.614
06 D4SW	50.504	54.36	18 33	79.614	83.662
06 D4SE	54.36	58.216	18 33	79.614	83.662
06 E4NW	58.216	62.072	18 33	75.566	79.614
06 E4NE	62.072	65.928	18 33	75.566	79.614
06 E4SE	62.072	65.928	18 33	79.614	83.662
06 F4NW	65.928	69.784	18 33	75.566	79.614
06 F4NE	69.784	73.64	18 33	75.566	79.614
06 F4SW	65.928	69.784	18 33	79.614	83.662
06 F4SE	69.784	73.64	18 33	79.614	83.662
06 B3NW	35.08	38.936	18 33	83.662	87.71
06 B3NE	38.936	42.792	18 33	83.662	87.71
06 B3SW	35.08	38.936	18 33	87.71	91.758
06 B3SE	38.936	42.792	18 33	87.71	91.758
06 C3NW	42.792	46.648	18 33	83.662	87.71
06 C3NE	46.648	50.504	18 33	83.662	87.71
06 C3SW	42.792	46.648	18 33	87.71	91.758
06 C3SE	46.648	50.504	18 33	87.71	91.758
06 E3NW	58.216	62.072	18 33	83.662	87.71
06 E3NE	62.072	65.928	18 33	83.662	87.71
06 E3SW	58.216	62.072	18 33	87.71	91.758
06 E3SE	62.072	65.928	18 33	87.71	91.758
06 F3NE	69.784	73.64	18 33	83.662	87.71
06 F3SW	65.928	69.784	18 33	87.71	91.758
06 F3SE	69.784	73.64	18 33	87.71	91.758
06 GRAFT1	35.08	73.64	33 37	51.278	91.758
06 GRAFB1	35.08	73.64	14 18	51.278	91.758
06 F2NW	65.928	69.784	18 33	91.758	95.806
06 F2NE	69.784	73.64	18 33	91.758	95.806
06 F2SW	65.928	69.784	18 33	95.806	99.854
06 F2SE	69.784	73.64	18 33	95.806	99.854
06 GRAFT2	65.928	73.64	33 37	91.758	99.854
06 GRAFB2	65.928	73.64	14 18	91.758	99.854
06 TOPF1	35.08	73.64	37 42	51.278	91.758
06 BOTF1	35.08	73.64	10 14	51.278	91.758
06 TOPF2	65.928	73.64	37 42	91.758	99.854
06 BOTF2	65.928	73.64	10 14	91.758	99.854
06 SS1W2	58.216	62.072	0 5	79.614	83.662
06 SS1V1	58.216	62.072	5 12	79.614	83.662
06 E4SW	58.216	62.072	12 28	79.614	83.662
06 SS1P	58.216	62.072	28 43	79.614	83.662
06 SS1V2	58.216	62.072	43 48	79.614	83.662
06 SS1W	58.216	62.072	48 51	79.614	83.662
06 SS4W2	58.216	62.072	0 5	63.422	67.47
06 SS4V1	58.216	62.072	5 12	63.422	67.47
06 E6SW	58.216	62.072	12 28	63.422	67.47
06 SS4P	58.216	62.072	28 43	63.422	67.47
06 SS4V2	58.216	62.072	43 48	63.422	67.47
06 SS4W	58.216	62.072	48 51	63.422	67.47

06 TRW1	50.504	54.36	0 18	71.518	75.566
06 TRV	50.504	54.36	18 39	71.518	75.566
06 TRP	50.504	54.36	39 51	71.518	75.566
06 SS2W2	42.792	46.648	0 5	79.614	83.662
06 SS2V1	42.792	46.648	5 12	79.614	83.662
06 C4SW	42.792	46.648	12 28	79.614	83.662
06 SS2P	42.792	46.648	28 43	79.614	83.662
06 SS2V2	42.792	46.648	43 48	79.614	83.662
06 SS2W	42.792	46.648	48 51	79.614	83.662
06 SS3W2	42.792	46.648	0 5	63.422	67.47
06 SS3V1	42.792	46.648	5 12	63.422	67.47
06 C6SW	42.792	46.648	12 28	63.422	67.47
06 SS3P	42.792	46.648	28 43	63.422	67.47
06 SS3V2	42.792	46.648	43 48	63.422	67.47
06 SS3W	42.792	46.648	48 51	63.422	67.47
06 RRW1	65.928	69.784	0 25	83.662	87.71
06 RRP	65.928	69.784	25 41	83.662	87.71
06 RRW2	65.928	69.784	41 51	83.662	87.71
06 H2O1	50.504	58.216	10 42	51.278	59.374
06 H2O2	50.504	58.216	10 42	83.662	91.758
06 H2O3	35.08	38.936	10 42	59.374	67.47
06 H2O4	35.08	38.936	10 42	75.566	83.662
06 H2O5	35.08	42.792	10 42	99.854	107.95
09 X 2	27.368	4 35.080	4 42.792		
09 X 4	46.648	2 50.504	4 54.360		
09 X 2	58.216	4 62.072	2 65.928		
09 X 4	69.784	2 73.640	4 78.720		
09 X 2	108.720				
09 Y 2	30.000	2 35.080	3 47.230		
09 Y 8	63.422	4 67.470	2 71.518		
09 Y 4	75.566	2 79.614	8 87.710		
09 Y 10	107.950	2 113.030	2 143.030		
09 Z 13	33.020	2 35.560	21 88.900		
09 Z 2	91.440	13 124.460			
14 B700	A1A7	1.0			
14 B1000	A1A10	1.0			
14 B1100	A1A11	1.0			
14 B1200	A1A12	1.0			
14 B1300	A1A13	1.0			
14 B1400	A1A14	1.0			
14 B1500	A1A15	1.0			
14 B1600	A1A16	1.0			
14 B1700	A1A17	1.0			
14 B1800	A1A18	1.0			
14 B1900	A1A19	1.0			
14 H2O	B5B0	1.0			
14 FUEL	C3C0	1.0			
14 SSV	B5D0	1.0			
14 SSRP	B5E0	1.0			
14 TRAN	B5F0	1.0			
14 TV	B5G0	1.0			
14 REG	B5H0	1.0			
14 GRAF	B5I0	1.0			
14 SSRF	A1J12	1.0			
14 TOP	B2C0	1.0			
14 BOT	C3K0	1.0			
14 GRIDP	B2K0	1.0			
15 B700	E5SW				
15 B1000	E7SW E7NW D6SW				
15 B1100	C5NE C5SW C5SE E7NE E7SE C7SW C7NW D6SE F4NE F4NW				
15 B1100	F4SE D5NW D5SE D5NE				
15 B1200	B5NE B5SW F7SE F7SW F7NW F7NE B3SE B3NE B6SE F4SW				
15 B1200	E6SE				
15 B1300	F3SW F3NE F3SE F6SE F6NE F6SW F6NW B7NE B7SE C7SE				
15 B1300	C7NE B6NE B4SE B4NE				
15 B1400	D4SE D4NE E3NW E3SE B3NW B3SW F2SW F2NE F2NW				

15 B1500 B5NW D4NW D4SW E3SW F5NW F5NE C4NE
15 B1600 B5SE E3NE B7NW B7SW E5SE
15 B1700 F5SE F5SW D6NW
15 B1800 C3SE C3NW C3SW C3NE
15 B1900 C5NW E4NE E4NW E4SE F2SE D6NE C4NW C4SE E5NE E5NW
15 B1900 C6SE C6NW C6NE E6NW E6NE
15 SSRF E4SW C4SW C6SW E6SW
15 H2O CORE SS1W SS2W SS3W SS4W TRW1 H2O2 RRW1 RRW2 H2O1
15 H2O H2O3 H2O4 H2O5 SS1W2 SS2W2 SS3W2 SS4W2
15 FUELG GRAFT1 GRAFB1 GRAFT2 GRAFB2
15 SSRP SS1P SS2P SS3P SS4P
15 SSV SS1V1 SS1V2 SS2V1 SS2V2 SS3V1 SS3V2 SS4V1 SS4V2
15 TRAN TRP
15 TV TRV
15 REG RRP
15 GRAF GRAFN GRAFS
15 TOP TOPF1 TOPF2
15 BOT BOTF1 BOTF2
15 GRIDP GRID

Table I-9

Fuel Temp (C)	Temperature Coefficients of Flip Fuel at Burnup of					
	Beginning	100 MWD	1000 MWD	1400 MWD	2000 MWD	3000 MWD
27.0	0.48461E-04	0.45887E-04	0.37257E-04	0.34783E-04	0.32484E-04	0.32139E-04
177.0	0.10080E-03	0.94828E-04	0.71728E-04	0.64254E-04	0.56408E-04	0.51715E-04
477.0	0.15669E-03	0.14574E-03	0.94518E-04	0.76860E-04	0.56605E-04	0.39335E-04
727.0	0.18141E-03	0.16736E-03	0.93812E-04	0.68543E-04	0.40479E-04	0.18442E-04

Fuel Temp (C)	Temperature Coefficients of LEU 20-20 (Er=.590%) Fuel at Burnup of		
	Beginning	700 MWD	1300 MWD
27.0	0.64988E-04	0.57789E-04	0.56438E-04
177.0	0.10804E-03	0.88526E-04	0.82516E-04
477.0	0.14662E-03	0.99609E-04	0.79508E-04
727.0	0.16322E-03	0.93430E-04	0.64231E-04

Table I-10. Control Rod Worth for Flip Fuel at 1400 MWD Burnup

Power Level : 1 MW

Control Rod Position	K-eff	dK	Worth (\$)	Rod Worth (\$) Measured in NSCR	
				(01/12/94)	(01/07/93)
All Rods In	0.92087243607	-----	-----		
SS1 Out	0.93992390299	0.01905146692	2.72163813	2.72	2.70
SS2 Out	0.93377725493	0.01290481886	1.84354555	2.11	1.86
SS3 Out	0.94109756552	0.02022512945	2.88930421	3.14	3.10
SS4 Out	0.95461376596	0.03374132989	4.82018998	4.56	4.69
RR Out	0.93058517599	0.00971273992	1.38753427	0.81	0.84
TR Out	0.93951615085	0.01864371478	2.66338783	3.15	3.20
Sum			16.32559997	16.49	16.39

Table I-11. Control Rod Worth for Fresh LEU 20-20 (Er: 0.59%)

Power Level : 1 MW

Control Rod Position	K-eff	dK	Worth (\$)
All Rods In	0.92159891433	-----	-----
SS1 Out	0.94105510640	0.01945619207	2.779456010
SS2 Out	0.93456406937	0.01296515504	1.852165006
SS3 Out	0.94255210407	0.02095318974	2.993312820
SS4 Out	0.95727101772	0.03567210339	5.096014770
RR Out	0.93131554874	0.00971663441	1.388090630
TR Out	0.94121515946	0.01961624513	2.802320733
Sum		0.11837951978	16.911359969

Table I-12. Estimation of Shutdown Margin for Flip Fuel at 1400 MWD Burnup

Core Excess Test at 300 watts

Critical Rod Position : SS 50.0% Out, RR 46.7% Out, TR 100% Out

Control Rod Position	Remaining Worth (\$)	Measured in NSCR	
		(01/12/94)	(01/07/93)
SS1 Out	0.89394985	1.30	1.25
SS2 Out	0.65034359	0.98	0.90
SS3 Out	0.89557011	1.55	1.53
SS4 Out	1.42276192	2.49	2.42
RR Out	0.58966038	0.48	0.46
Total	4.45228585	6.80	6.56

Computation of Shutdown Margin :

Situation	Calculation	Measured in NSC	
		(01/12/94)	(01/07/93)
Total Control Rod Worth	16.32559997	16.49	16.39
Core Excess at 300 Watts	-4.45228585	-6.80	-6.56
Highest Worth Non-Secured Experiment	-1.00000000	-1.00	-1.00
Most Reactive Rod Worth	-4.82018998	-4.56	-4.69
Non-Scrambled Rod Worth	-1.38753427	-0.81	-0.84
Shutdown Margin	4.66558987	3.32	3.30

Table I-13. Estimation of Shutdown Margin for Fresh LEU 20-20 (Er: 0.59%)

Core Excess Test at 300 watts

Critical Rod Position : SS 28.6% Out, RR 66.7% Out, TR 100% Out

Control Rod Position	Remaining Worth (\$)
SS1 Out	2.091368254
SS2 Out	1.326950750
SS3 Out	1.824503637
SS4 Out	2.987349539
RR Out	0.290317990
Total	8.520490170

Computation of Shutdown Margin :

Situation	Cost (\$)
Total Control Rod Worth	16.91135997
Core Excess at 300 Watts	-8.52049017
Highest Worth Non-Secured Experiment	-1.00000000
Most Reactive Rod Worth	-5.09601477
Regulating Rod Worth	-1.38809063
Shutdown Margin	0.90676440

Appendix II - NCTRIGA

A. Code Description

NCTRIGA is a one dimensional heat transfer code that calculates temperatures at several axial nodes along a single fuel rod channel and determines the natural convection induced coolant flow rate. This code uses the power distribution generated from the neutronic analysis, along with information on the geometry and material composition of the flow channel, to produce temperatures in the fuel clad and coolant and to predict the coolant flow rate. This code is a modification of the Argonne code NATCON which was specifically written for reactors with TRIGA type fuel¹. Verification of this code was performed by Argonne with data collected by General Atomics. This is shown in Table II-1. It verifies that reasonable outlet temperatures and flow rates can be expected when applying NCTRIGA to TRIGA cores. The input for NCTRIGA is presented in Table II-2.

Table II-1- Comparison of NCTRIGA Results to Experimental Data

Power MW	No. of Elements	Source	Flowrate kg/sec	Error %	Outlet Temp °C	Error %
1.0	91	GA	8457	-7.0	70.2	-10.0
		NCTRIGA	7890		67.2	
1.5	91	GA	9555	-3.2	76.6	-2.6
		NCTRIGA	9259		75.6	
2.0	101	GA	11080	-0.3	86.1	-14.0
		NCTRIGA	11051		80.1	

Table II-2 - NCTRIGA Input Parameters

<u>Variable Name</u>	<u>Definition</u>
CHANHT	Coolant channel height, m.
CHIMMY	Unheated section of the reactor above top of fuel rods, m.
CLADTK	Thickness of the clad, m.
CPWR	Fixed power of whole core, kW.
DELTA	Criterion for convergence of frictional and buoyant forces to the same value.
DEPTH	Distance from pool surface to the bottom of fuel rods, m.
FKVT(I)	Fuel conductivity, W/mK, as a function of temperature, TEMPK(I).
FUELHT	Length of fuel meat, m.
FUELRD	Radius of fuel meat, m.

¹ Smith, R.S., Comparison of NCTRIGA Results to GA Data, NCTRIGA Input Format, and Revision of NATCON, Argonne National Laboratory, Intra-Laboratory Memo, Sept. 18, 1992.

Table II-2 - continued

FW, FQ, FH	Hot channel factors for flow (or enthalpy), heat flux, and heat transfer coefficient.
GAPCON	Conductance, W/m^2K , between the fuel rod and clad.
ICONFG	Code for orientation of pins. If = 3, a triangular pitch geometry is assumed (appropriate for circular geometry commonly used in TRIGA reactors); if = 4 a square pitch is assumed.
ICTC	If = zero, clad conductivity of Inconel 800 is assumed and computed internally in the code. If not = zero, user must input a single, temperature independent value of clad conductivity as UCLADK.
IPRT	If = zero, intermediate iteration and other output will not be printed. If not = zero, the output will be printed.
IQVZ	If = zero, a standard axial power distribution is calculated. If not = zero, user must input the axial distribution as arrays: QVZ(I) and ZR(I).
NK	No. of data values in fuel conductivity, FKVT(I), TEMPK(I). If = zero, fuel conductivity of Zirconium Hydride is assumed and computed internally in the code. If not = zero, user must input a temperature dependent array of fuel conductivity as FKVT(I) and TEMPK(I).
NN	No. of axial nodes in reactor coolant channel.
NTRGEL	No. of TRIGA elements in the whole core.
ONEP5	Additive parameter in code computed fuel conductivity relation that ranges from +1.5 to -1.5. For nominal value use 0.0.
POVERD	Pitch to diameter ratio (P/D).
QVZ(I)	Axial power distribution whose averages is one. ZR is the axial average position array.
RPEAK	Power factor by which this channel differs from an average channel.
TEMPK(I)	Temperature array corresponding to FKVT(I).
TPOOL	Average temperature of water in pool. °C.
UCLADK	User input value of clad conductivity, W/mK .
UHZB, UHZT	Unheated lengths at bottom and top of fuel pin, m.
VGUESS	Initial guess for inlet velocity of coolant, m/s.
ZR(I)	Axial position normalized to one (Z/FUELHT).

NCTRIGA models only one fuel element-flow channel at a time. To change the channel modeled in NCTRIGA for the NSCR core, since the geometry of all of the fuel elements is identical, only the axial power distribution, radial peaking factor, and total core power has to be changed in the NCTRIGA model. A sample input deck is shown in Table II-3.

Table II-3 - NCTRIGA Input Deck

TAMU NSC TRIGA HEU CORE prepulsing case axial dist from 300 watt-E5 SW
& INPUT

```

      NN=15
      , NK=5
      , NTRGEL=90
      , IPRT=0
      , IQVZ=1
      , IFK=1
      , ICTC=1
      , ICONFG=4
      , CPWR=1000.0
      , TPOOL=37.
      , UHZB=0.889
      , UHZA=0.889
      , FUELHT=.381
      , FUELRD=.017411
      , ONEP5=0
      , GAPCON=22111
      , POVERD=1.10268
      , CLADTK=.000508
      , UCLADK=16.8
      , CHANHT=.5588
      , CHIMNY=0.000
      , DELTA=.00001
      , VGUESS=0.2
      , DEPTH=8.84
      , RPEAK=1.65778649
      , FW=1.0
      , FQ=1.0
      , FH=1.0
      FKVT=18.0, 18.0, 18.0, 18.0, 18.0.
      TEMPK= 0.0, 200., 400., 600., 800.,
      ZR=0.000, 0.0667, .01333, 0.2000, 0.2667, 0.3333, 0.4000, 0.4667,
          0.5333, 0.6000, 0.6667, 0.7333, 0.8000, 0.8667, 0.9333, 1.0000,
      QVZ=0.4734, 0.5504, 0.6611, 0.7999, 0.9338, 1.0596, 1.1666, 1.2451,
          1.2895, 1.2966, 1.2652, 1.1961, 1.0922, 0.9609, 0.8426, 0.8074
      &END

```

In the above input deck, there are 90 TRIGA fuel elements in the core. The total core power is 1 MW, or 1000 kW. All of the geometry parameters are taken from the fuel description given in the SAR.

The fuel thermal conductivity is taken to be a constant with respect to fuel temperature. The value used is 18.0 watts/mK as recommended by GA².

B. SAR Parameters

A description of the methods used to generate the parameters contained in Chapters III and XI of the SAR is given below.

1. Steady State Parameters

To generate the LSSS for steady state operation, NCTRIGA jobs were run with the power distribution obtained from DIF3D for the beginning of life (1 MW critical case) for each fuel type. NCTRIGA was set up to model the hottest rod in the core, and the power level was increased until the peak temperature in the rod was 950°C. A job was then run for the core power that produced 950°C in the peak rod in order to determine the lowest temperature in an allowed instrumented element location. The within rod peaking factor was applied to the centerline temperature obtained from this job, and the result is the lowest possible thermocouple temperature reading when the peak core temperature is 950°C. The results of the jobs run in determining the LSSS are shown graphically below in Figure II-1.

Figure II-1

For FLIP fuel, the power level necessary to produce a peak core temperature of 950°C was 2800 kW. This gave a lowest possible instrumented element peak temperature of 816°C. Since the highest observed ambient temperature for the pool is 37°C, the peak temperature rise of this instrumented element would be

$$\Delta \hat{T}_{TC} = 816^{\circ}\text{C} - 37^{\circ}\text{C} = 779^{\circ}\text{C}$$

In steady state mode, the radial temperature distribution through a fuel element is not highly sensitive to the power density distribution. Thus, the radial temperature distribution which was calculated for the PRNC reactor is applicable to the NSCR. For this case the ratio of the maximum temperature, which occurs at the center of the fuel element, to that at the thermocouple location is 1.16. Thus, for the element discussed above, the thermocouple should read

² Schlicht, Roger, FAX correspondence, General Atomics, San Diego, CA, October 25, 1993.

$$\frac{779^{\circ}\text{C}}{1.16} = 672^{\circ}\text{C above ambient}$$

Thus, for the FLIP core using the new methods, the LSSS is

$$\text{LSSS} = 672^{\circ}\text{C} + 37^{\circ}\text{C} = 709^{\circ}\text{C}$$

For LEU fuel, the total core power required to produce a 950°C peak temperature is 2600 kW, and the corresponding peak temperature in the lowest power allowed instrumented element location is 841°C . Using the same methods as above

$$\frac{841^{\circ}\text{C} - 37^{\circ}\text{C}}{1.16} + 37^{\circ}\text{C} = 730^{\circ}\text{C}$$

Thus, for LEU fuel the LSSS for steady state operation is 730°C .

2. Pulsing Parameters

For pulsing analyses, the code PARET, presented in Appendix III, was used. One required input parameter for PARET is the flow rate in the channel of interest. NCTRIGA was used to generate the mass flow rate. To accomplish this, NCTRIGA was run for cases with different rod burnups and powers at prepulsing conditions which consist of the reactor operating at 300 watts with the transient rod fully inserted. It was determined that, for both the FLIP and LEU fuel, the flow rates were between 10 and 11 $\text{kg/m}^2\text{s}$ for all cases considered. For simplicity, an initial flow rate of 10.49 $\text{kg/m}^2\text{s}$ was chosen for all PARET cases from 300 watts. A discussion of the mass flow rates used as the transient progresses is presented in Appendix III.

Appendix III - PARET

A. Code Description

PARET is a computer program designed for use in predicting the course and consequences of nondestructive reactivity accidents in small reactor cores. It is basically a coupled neutronics-hydrodynamics-heat transfer code employing point kinetics, one-dimensional hydrodynamics, and one-dimensional heat transfer¹. The core can be represented by one to four regions. Each region may have different power generation, coolant mass flow rate, and hydraulic parameters as represented by a single fuel pin and its associated coolant channel. The time dependent temperatures within the fuel element is computed on the basis of the one-dimensional heat conduction equation solved in each of up to 21 axial sections. The hydrodynamics solution is also one-dimensional for each channel at each time node. The heat transfer may take place by natural or forced convection, nucleate, transition, or stable film boiling, and the coolant is allowed to range from subcooled liquid, through the two-phase regime, up to and including super-heated steam. PARET will also calculate flow reversal.

B. PARET Input

Below is a listing of the parameters needed in the generation of a PARET input deck. Each section contains the values used, along with a description of the methods used to produce these parameters. The headings given below are the same as those used in the PARET manual for ease of reference.

RESTART CARD

The first value is the identifier to indicate a restart problem 1, or initial problem 0, the second value is the frequency with which the restart file should be written in number of time steps.

0 200

I. Title Line

This is just the name of the job to be printed on the header. The first character must be an asterick.

*PARET: NSC TRIGA \$1.80 pulse at 37 deg C 300 watts

II. General Information

This information should be input to the 10xx series cards in the format described above.

<u>Item</u>	<u>Variable</u>	<u>Value</u>	<u>Description</u>
1	NCHN	-2	For the NSCR model, there are two channels. The first is the hot channel with the volume of one actual channel in the core, and the second is the average channel having the volume of the rest of the core. The minus signals that SI units will be used.

¹ C.F. Obenchain, "PARET--A Program for the Analysis of Reactor Transients." IDO-17282, January 1969.

2	NZ	15	In this model, there are 15 axial nodes in the fuel meat. Each is 1 in.
3	NR	7	There are seven radial nodes. Five in the fuel, one for the gap, and one for the clad.
4	IGEOM	1	Cylindrical geometry
5	IPROP	1	Reactivity specified problem. If 0 were chosen, the power level could be specified in table 9 unstead of the reactivity.
6	IRXSWT	1	This lets the code calculate the quality.
7	IPOP	0	Inlet pressure is specified.
8	KINTS	0	This allows the code to compress the time step, if necessary for kinetics calculations.
9	IDLYGP	6	Six delayed neutron groups.
10	KINPRT	0	No intermediate printout of kinetics parameters.
11	ISUPPR	0	No average temperature printout
12	MAXHCC	20	Limits heat transfer interations to 20 per node.
13	POWER	0.3-3	Initial Reactor Power in MW.
14	PF	0.032656	Total volume of fuel in the core (m ³)
15	PRESUR	1.90327+5	Operating pressure in Pa.
16	ENTHIN	-37.00	Inlet moderator temperature in -deg C. If a positive value is entered, it represents enthalpy
17	RS	0.01792	Fuel pin radius in m
18	RF	0.017411	Radius of fuel meat in m
19	RC	0.01742	Radius to inner surface of clad in m
20	PW	0.0	Plate width, doesn't apply here.
21	FW	0.0	Used only for plates
22	AL	0.381	Active length of fuel in m
23	ALDDIN	0.1016	Lenght of unfueled inlet, m.
24	ALDEEX	0.1016	Length of unfueled exit, m.
25	BBEFF	0.0071	Beta effective

26	EL	27.9-6	Prompt neutron generation time, s.
27	GRAV	9.80664	Acceleration due to gravity, m/s^2 .
28	QW	0.00679	Heat source for moderator. This is the fraction of heat generated in the moderator, multiplied by the ratio of fuel meat volume to moderator volume. This value gives about 2% of the energy generation occurring in the moderator.
29	TRANST	15.0	Total time of investigation, s.
30	RXXCON	0.80	A parameter used in void volume generation.
31	RXXEP	1.0	Exponent used in void volume generation.
32	RHOREF	988.67	Moderator reference density kg/m^3 . This is the density of the moderator at the initial cond.
33	GAMMA0	-0.47296	All GAMMAx are coefficients used in fitting the reactivity feedback as a function of temperature. These must be generated from the reactivity changes determined from DIF3D runs at different temperatures. A detailed description is given in the following section.
34	GAMMA1	-0.20201-2	
35	GAMMA2	1.15817-5	
36	GAMMA3	0.00	
37	GAMMA4	0.00	
38	DOPPN	1.00	Exponent used in reactivity fit.

The above parameters are numerical fits to the curves show below in Figure III-1. These were generated with data from DIF3D.

Figure III-1

Since the integral reactivity feedback is burnup sensitive, a different set of parameters must be entered according to the burnup desired for the problem of interest. A list of the curve fits to this data is given in Table III-1.

Table III-1

Integral Reactivity Feedback Curve Fit

FLIP Fuel:

BOL	$-0.99258 - 7.57122 \times 10^{-4} T + 1.26832 \times 10^{-5} T^2$
1000 MWD	$-1.63758 + 3.40290 \times 10^{-3} T + 5.25240 \times 10^{-6} T^2$
1400 MWD	$-1.92043 + 4.96090 \times 10^{-3} T + 2.90688 \times 10^{-6} T^2$
2000 MWD	$-2.27714 + 6.87820 \times 10^{-3} T + 1.30048 \times 10^{-7} T^2$
3000 MWD	$-2.66930 + 8.86260 \times 10^{-3} T - 2.43340 \times 10^{-6} T^2$

LEU Fuel:

BOL	$-1.38676 + 1.2853 \times 10^{-3} T + 1.09952 \times 10^{-5} T^2$
MOL	$-2.47546 + 6.6732 \times 10^{-3} T + 3.80030 \times 10^{-6} T^2$
EOL	$-3.21306 + 9.3854 \times 10^{-3} T + 5.21579 \times 10^{-7} T^2$

These fits are used with the constant as GAMMA0, the coefficient of the first order term as GAMMA1, and the coefficient of the second order term as GAMMA2. Since second order fits

give acceptable agreement with the NSCR data, the other terms GAMMA3, GAMMA4 and DOPPN were not needed.

39	EPS4	0.001	Upper limit for kinetics time test.
40	DNBQDP	0.00	This allows the code to calculate the DNB heat flux at each node.
41	TAUUNB	0.001	Nucleate boiling bubble collapse time. This value came with the sample deck.
42	TAUUTB	0.001	Transition boiling bubble collapse time. This value came with the sample deck.
43	ALAMNB	0.05	Fraction of clad surface heat flux which is utilized in producing vapor in sub-cooled region. This value came with the sample deck.
44	ALAMTB	0.05	Same as 43 for transition boiling. This value came with the sample deck.
45	ALAMFB	0.05	Same as 43 for film boiling.
46	HTTCON	1.4	Natural convection heat transfer constant for correlation. This value came with the sample deck.
47	HTTEXP	0.33	Natural Convection heat transfer exponent for original correlation. This value came with the sample deck.

III. Additional General Information

This data will be input to the 111x cards.

CARD 1111

1a	PSUBC	0.049768	This is the total area of flow for all the channels in the core in m ² .
2a	FACT2(I)	1.00	This appears to be some parameter that the code no longer uses. It is recommended to use 1.00 to keep the code from crashing. This value must be repeated for the number of channels used.

CARD 1112

1b	IONEP	0	The Dittus-Boelter correlation is used for single phase heat conduction
2b	ITWOP	1	The McAdams correlation is used for two phase heat transfer

3b	IMODE	1	The transition model is used.
4b	ICHF	2	The Mirshak DNB correlation is used.
5b	IHT	0	The original single phase heat transfer routine is used.
6b	QAVE	0	No value is needed for the average core heat flux.

CARD 1113

1c	RDRATE	3.81	Rate of control rod movement. m/s. This parameter has no effect on the rod movement for the case described by this input deck. The reactivity insertion rate is interpolated from the 9000 cards.
2c	TDLAY	0.2	Delay time before control rod movement begins after trip detection. (seconds).
3c	POWTP	10000.0	Overpower trip setpoint (MW).
4c	FLOTP	0.00	Low flow trip setpoint.

CARD 1114

1d	HNCTOP	0.0889	Height above the core considered for natural convection effects.
2d	HNCBOT	0.0889	Height below the core considered for natural convection effects.

2000 SERIES CARDS

These cards are used to describe the thermal conductivities and heat capacities of the materials used in the fuel. They are fit parameters that must be generated for each type of material. See equations 71 and 72 of the PARET manual for details.

The following numbers can be used for FLIP fuel.

CARD 2001

ALPHA1, ALPHA2, ALPHA3, ALPHA5 = 0.0

ALPHA4	18.00	This is the constant thermal conductivity of the fuel meat (W/mK). General Atomics has found that the thermal conductivity of the fuel is independent of temperature ² .
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CARD 2002

BETA1, BETA4 = 0.00

² Schlicht, Roger, FAX correspondence, General Atomics, San Diego, CA, October 25, 1993.

BETA2	0.417+4	This is the coefficient of the T term in the fit equation of the heat capacity.
BETA3	2.04+6	This is the constant in the fit equation of the heat capacity.
BETA5	-273.15	This term adjusts the T to units of °C instead of K which the code generates to match the units of the curve fit.

The method used to generate these fits for FLIP and LEU fuel as recommended by GA in Reference 2 is shown below.

The specific heat of $ZrH_{1.6}$ is given by

$$C_p(ZrH) = 0.75148 T + 363.0938 \text{ J/kg}^\circ\text{C}$$

The specific heat of U is given by

$$C_p(U) = 0.1305 T + 109.4 \text{ J/kg}^\circ\text{C}$$

So the specific heat of UZrH is given by

$$C_p(UZrH) = w(U)C_p(U) + w(ZrH)C_p(ZrH)$$

$$C_p(UZrH) = w(U)(0.1305 T + 109.4) + w(ZrH)(0.75148 T + 363.0938) \text{ J/kg}^\circ\text{C}$$

Applying the appropriate weight percents of U and ZrH for FLIP and LEU fuels and multiplying by the density of each fuel type, respectively,

FLIP:

$$C_p(UZrH) = ((0.085)(0.1305 T + 109.4) + (0.915)(0.75148 T + 363.0938) \text{ J/kg}^\circ\text{C}) \times 5970 \text{ kg/m}^3$$

LEU:

$$C_p(UZrH) = ((0.20)(0.1305 T + 109.4) + (0.80)(0.75148 T + 363.0938) \text{ J/kg}^\circ\text{C}) \times 6770 \text{ kg/m}^3$$

The following results are produced

$$C_p(UZrH) = 0.417 \times 10^4 T + 2.05 \times 10^6 \text{ J/m}^3^\circ\text{C} \text{ for FLIP fuel}$$

$$C_p(UZrH) = 0.424 \times 10^4 T + 3.16 \times 10^6 \text{ J/m}^3^\circ\text{C} \text{ for LEU fuel}$$

This information must be input on the above cards depending on the type of fuel to be investigated.

CARDS 2003, 2004

These are the values used in the fit of the gap conductivity and heat capacity. These are assumed to be constant with respect to temperature.

ALPHA3 0.19900

BETA3 6.66340+2

CARD 2005, 2006

These are the values used in the fit of the clad conductivity and heat capacity. These are assumed to be constant with temperature

ALPHA3 16.8

BETA3 3.975+6

3000 CARDS

These cards give the radial description of the fuel pin in the PARET model. For the TRIGA, seven radial regions are modeled. There are five nodes in the fuel meat, one in the gap, and one in the clad. The first value on each card describes the thickness of the region in meters, the second value denotes the last region with this thickness. The third value denotes the material composition of the region, as described in the 2000 series cards, i.e. 1 is fuel, 2 is gap, 3 is clad. The fourth value denotes the fraction of the total energy generated in this channel that is generated in the particular region of the channel.

Below are the cards. 3001 is for fuel, 3002 for gap, 3003 for clad.

3001,	4.35275-	3	5	1	0.980
3002,	0.9-5	6		2	0.00
3003,	5.00-4	7		3	0.000

The value of 0.980 on the 3001 card means that 98% of the energy is generated in the fuel. The other 2% is generated in the coolant. This coincides with the value input as item 28, QW, on the 1000 series cards.

4000 CARDS

These cards are used to describe the axial geometry in the active region of the fuel pin. In this model there are 15 regions of one inch (0.0254 m) each. The card is given below.

4001,	2.54-2	1	2.54-2	14	2.54-2	15
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5000 CARDS

These cards give information about each channel. For the first channel, the cards are numbered from 5100, for the second channel, the cards are numbered from 5200. The data is input as follows.

<u>ITEM</u>	<u>VARIABLE</u>	<u>CH1 VALUE</u>	<u>CH2 VALUE</u>	<u>DESCRIPTION</u>
2a	IFLOW	1	1	The flow for all channels will be specified on the 10000 series cards as a function of time.
3a	DELP	0	0	Not needed for this model

4a	RN	0.02794	0.02794	This is the distance in meters from the center of the pin to the node in the center of the water channel. For pin geometry, this is taken to be $\text{pitch}/2*\text{sqrt}(2)$
5a	BM	1.111-2	9.999-1	This is the reactivity weighting factor for the channel. These values make 1/90 of the core from the hot channel, and 89/90 of the core from the average channel.
6a	ALOSCN	0.5	0.5	Unrecoverable loss coefficient for the inlet.
7a	ALOSCX	0.55	0.55	Unrecoverable loss coefficient at the exit.
8a	SIGIN	1.00	1.00	Ratio of channel area to inlet plenum area.
9a	SIGEX	1.00	1.00	Ratio of channel area to outlet plenum area.
10a	DVOID	0.00	0.00	Void/density coefficient that is not needed here.
11a	DTMP	0.00	0.00	Coolant temperature coefficient that is not used in this model.

CARD 5101, 5201

These cards represent the plenea. These same values are used for both channels and are as follows.

1b	ALPPIN	0.0889	This is the length of the inlet plenum.
2b	ALPPEX	0.0889	This is the length of the outlet plenum.
3b	DEEIN	4.38511-2	This is the equivalent diameter of the inlet plenum.
4b	DEEEX	4.38511-2	This is the equivalent diameter of the outlet plenum.

The remaining 5000 series cards contain information describing the axial power description, followed by weighting factors which have all been set to 1.00 since they are not used in this model. The axial power description is the ratio of the local to average power. This is not normalized to one, except for the average channel. The hot channel should be normalized to the radial peaking factor. Fifteen values must be input. The first and the last are the peaking factors for the top and bottom of the fuel, not the nodes output from DIF3D. These values must be extrapolated from data generated by DIF3D. The position of the other nodes in PARET corresponds exactly to the nodes output from DIF3D, so no interpolation is necessary. The value input on these cards represent the state of the reactor prior to the transient..

The entire set of 5x00 cards must be repeated for each channel modeled. A complete set of 5100 cards and 5200 cards are shown below.

5100,	1	0	0.02794	1.1111-2	0.5	0.55
5100,	1.00	1.00	0.00	0.00		
5101,	0.0889	0.0889	4.38511-2	4.38511-2		
5102,	0.6563	1.00	1.00	1.00		
5103,	0.8587	1.00	1.00	1.00		
5104,	1.0636	1.00	1.00	1.00		
5105,	1.2791	1.00	1.00	1.00		
5106,	1.5037	1.00	1.00	1.00		
5107,	1.6854	1.00	1.00	1.00		
5108,	1.8191	1.00	1.00	1.00		
5109,	1.9022	1.00	1.00	1.00		
5110,	1.9326	1.00	1.00	1.00		
5111,	1.9091	1.00	1.00	1.00		
5112,	1.8314	1.00	1.00	1.00		
5113,	1.7007	1.00	1.00	1.00		
5114,	1.5208	1.00	1.00	1.00		
5115,	1.3093	1.00	1.00	1.00		
5116,	1.1314	1.00	1.00	1.00		
5200,	1	0	0.02794	9.9999-1	0.5	0.55
5200,	1.00	1.00	0.00	0.00		
5201,	0.0889	0.0889	4.38511-2	4.38511-2		
5202,	0.5087	1.00	1.00	1.00		
5203,	0.6323	1.00	1.00	1.00		
5204,	0.7771	1.00	1.00	1.00		
5205,	0.9084	1.00	1.00	1.00		
5206,	1.0343	1.00	1.00	1.00		
5207,	1.1393	1.00	1.00	1.00		
5208,	1.2172	1.00	1.00	1.00		
5209,	1.2641	1.00	1.00	1.00		
5210,	1.2776	1.00	1.00	1.00		
5211,	1.2567	1.00	1.00	1.00		
5212,	1.2009	1.00	1.00	1.00		
5213,	1.1112	1.00	1.00	1.00		
5214,	0.9902	1.00	1.00	1.00		
5215,	0.8505	1.00	1.00	1.00		
5216,	0.7438	1.00	1.00	1.00		

6000 CARDS

These cards provide the information concerning the delayed neutron emission. The parameter IDLYGP, the ninth item on the 1000 series cards specifies the number of delayed neutron groups that are used. This data is in ordered pairs, the first number is the yield fraction for the group and the second number is the decay constant.

This model contains six groups. The data is given below.

6001,	0.3824-1	0.12722-1	0.21194	0.31737-1	0.1878	0.1161
6002,	0.40684	0.31137	0.12879	1.4001	0.2639-1	3.8706

9000 CARDS

These cards represent the reactivity as a function of time in this model. They can also represent the power level as a function of time, depending on the type of problem desired. The 9000 card contains one integer value signaling the number of sets of data contained on the subsequent cards. The minimum number of data sets is two.

The remaining cards begin their numbering at 9001. The first value is the reactivity desired and the second value is the time at which this reactivity will be reached. PARET performs linear interpolation for times between the input values. The first value input must be for time 0.00 and the last time must be greater than the total time of the transient.

Sample cards for a \$1.80 pulse are shown below.

9000,	6					
9001,	0.000	0.000	0.000	0.15	1.80	0.20
9002,	1.800	2.000	0.00	2.20	0.00	1000.0

Note that the reactivity is inserted between 0.15 and 0.20 seconds and is removed between 2.000 and 2.20 seconds. This represents the transient rod ejection under pressure, which causes faster rod motion than the insertion under the influence of gravity.

10000 CARDS

These cards represent the inlet mass flow rate as a function of time. For the flow option 1 on the 5x00 card, these determine the mass flow rate as the transient progresses. As long as a sufficient flow rate is provided to allow the code to converge on a positive temperature, the flow rate has little effect on the peak temperature. This comparison is shown in Figure III-2. NCTRIGA was used to approximate initial mass flow rates. It was determined that for normal prepulsing conditions, for both fuel types, the mass flow rate was approximately 10.49 kg/m²s. This is used as the initial mass flow rate in all pulsing cases. The minimum flow rate that allows convergence is used for later times in the transient since 10.49 kg/m²s is not sufficient for convergence at higher powers.

These cards work much like the 9000 cards. Here the 10000 card consists of one integer value denoting the number of ordered pairs on the subsequent cards. The cards numbered from 10001 consist of ordered pairs of the flow rate, in kg/m²s, and the time in seconds at which the flow rate takes effect. Again, the code interpolates linearly between the values for intermediate times.

Sample cards are shown below.

10000,	6				
10001,	10.49	0.00	300.0	0.30	
10002,	300.0	0.50	100.0	5.00	
10003,	10.49	45.0	10.49	1000.0	

11000 CARDS

These cards represent the thermal expansion of the clad as a function of temperature. This is another series of cards with the first card telling how many sets of data are contained on the other cards, and the next cards containing data. The data is entered in pairs, the first parameter is the percent of expansion of the clad. The second is the temperature in °C at which this percent expansion exist.

Sample cards are shown below.

11000, 2

11001. 0.0 10.00 0.5 2000.0

Figure III-2 Centerline Temperature for Varying Flow Rates

\$1.80 FLIP Pulse 1400 MWD

12000 CARDS

These cards are not used in our model. The pressure drop as a function of time is calculated by the code. All values are set to zero.

14000 CARDS

These cards represent the calculational time step as a function of transient time. This is important for producing accurate results with optimum efficiency. A small time step is desired when the power or temperature is changing rapidly, and a large time step is desired when the parameters are changing slowly. Sample cards are shown below.

```
14000, 6
14001, 0.001 0.0 0.0001 0.20 0.00005 0.25
14002, 0.001 1.0 0.001 2.20 0.005 3.0
```

16000 CARDS

These cards determine the frequency which the major and minor edits are output. The values input here are determined by how much resolution is desired in the printed results. This does not effect the calculations, the code will continue to calculate in the steps given above, but it will not print the results at every time step. The output files from PARET are typically quite large. The size of the file produced using the following cards for the 15 second tranient is about 15 Mb. The cards work in sets of three numbers. The first is the time increment for major edits, the second is the frequency on intermediate edits, and the third is the time at which these go into effect.

```
16000, 5
16001, 0.10 50 0.0 0.0010 10 0.10
16002, 0.05 10 1.00 0.005 20 2.50
16003, 0.5 10 3.0
```

17000 CARDS

These cards represent relative pump mass velocity as a function of time. For the NSCR case, there is no forced convection, all values are set to 1.00.

```
17000, 2
17001, 1.0 0.0 1.00000 455.0
```

18000 CARDS

These cards represent the worth of the control elements as a function of position. These cards have no effect on the results when the reactivity is input as a function of time using the 9000 cards.

```
18000, 2
18001, 0.0 0.0 -4.00 0.381
```

C. Code Verification

To benchmark this model, data was obtained from pulsing experiments on the Texas A&M Nuclear Science Center Reactor using FLIP fuel. These experiments were performed on April 8, 1993. The average core burnup was approximately 1400 MWD. DIF3D was run to create a critical case with the same core configuration as the experiment. The power distribution of the instrumented element and average element were input to the PARET model. The curve fit for the temperature feedback at 1400 MWD was also used.

When the peak pulsing powers were compared between the model and the experimental data, the following results were produced.

Figure III-3

The comparison of the full width at half maximum of the pulsing peaks is shown below.

Figure III-4

Both of the comparison of the peak energy and FWHM agree quite well. The next figure displays a comparison of the total energy generated during the pulse.

Figure III-5

The mismatch at \$1.80 shown above is most likely because the experimental value is too small, caused by the detector overloading due to the excessive amount of energy generated in this pulse.

The final comparison of PARET to experiment matched the peak temperature generated in the fuel during a pulse. These results are shown below.

Figure III-6

This peak fuel temperature is difficult to quantify exactly due to the nature of the experimental data acquired. The experimental temperature is taken from a thermocouple which is located mid-way between the fuel surface and the centerline of the instrumented element. Since the measured temperature falls between the predicted centerline and fuel surface temperatures, the results are accepted as reasonable. In addition, the calculated peak fuel temperature is considerably higher than the experimentally measured temperature.

Plots of the data obtained from PARET for these pulses are included as Figures III-7 - III-17. These show the power, energy, centerline temperature, fuel surface temperature, and cladding temperature for the instrumented element for each pulse.

Figure III-7 \$1.20 FLIP Pulse

Figure III-8 \$1.20 FLIP Pulse

Figure III-9 \$1.35 FLIP Pulse

Figure III-10 \$1.35 FLIP Pulse

Figure III-11 \$1.50 FLIP Pulse

Figure III-12 \$1.50 FLIP Pulse

Figure III-13 \$1.65 FLIP Pulse

Figure III-14 \$1.65 FLIP Pulse

Figure III-15 \$1.80 FLIP Pulse

Figure III-16 \$1.80 FLIP Pulse

Appendix IV - Loss of Coolant Calculations

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

1. The fuel and clad are at the same temperature.
2. The hydrogen-to-zirconium ratio is 1.6 for FLIP and LEU fuel.
3. The free volume within the element is represented by a space 1/8 in. high within the clad.
4. The reactor contains fuel that has experienced burnup equivalent to 7700 MW-days.
5. Maximum operating temperature of the fuel is 600°C.

The fuel element internal pressure p is given by

$$P = P_h + P_{fp} + P_{air}$$

where P_h = hydrogen pressure,
 P_{fp} = pressure exerted by volatile fission products,
 and P_{air} = pressure exerted by trapped air.

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

where $P_h = \exp \{1.76 + 10.3014x - 19740.37/(T_k)\}$
 x = ratio of hydrogen atoms to zirconium atoms
 and T_k = fuel temperature (°K).

The pressure exerted by the fission product gases is given by

$$P_{fp} = f \frac{n}{E} \frac{RT_k}{V}$$

where f = fission product release fraction,
 n/E = number of moles of gas evolved per unit of energy produced (moles/MW day)
 R = gas constant (8.206×10^{-2} liters-atmospheres/mole °K),
 V = free volume occupied by the gasses (liters),
 and E = total energy produced in the element (MW-day).

The fission product release fraction is given by

$$f = 1.5 \times 10^{-5} + 3.6 \times 10^3 \exp\{-1.34 \times 10^4/(T_0)\}$$

where T_0 = maximum fuel temperature in the element during normal operation (°K).

The fission product gas production rate n/E is not independent of power density (neutron flux) but varies slightly with the power density. The value $n/E = 1.19 \times 10^{-3}$ moles/MW-day is accurate to within a few percent over the range from a few kilowatts per element to well over 40 kW/element. The free volume occupied by the gases is assumed to be a space 1/8 in. (0.3175 cm) high at the top of the fuel so that

$$V = 0.3175 \pi r_i^2$$

where r_i = inside radius of the clad (1.745 cm).

For standard TRIGA fuel the maximum burnup is about 4.5 MW-days/element, but the TRIGA-FLIP fuel is capable of burnup to about 77 MW-days/element. As the fission product gas pressure is proportional to the energy released, it will be assumed that the FLIP fuel in the reactor has experienced maximum burnup.

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

$$P_{\text{air}} = RT_k/22.4,$$

where it is assumed that the initial specific volume of the air is 22.4 liters/mole. Actually, the air forms oxides and nitrides with the zirconium so that after relatively short operation, the air is no longer present in the free volume inside the fuel element clad. The results of the stress imposed on the clad for standard and FLIP fuels are shown in Figure 1.

A two dimensional transient-heat transport computer code developed by General Atomic was used for calculating the system temperatures after the loss of pool water. Heat removal parameters were derived by General Atomic and programmed into the calculations. It was assumed that the reactor was shutdown 15 minutes before the core was uncovered. This is the time between the actuation of the pool level alarm and the uncovering of the fuel. It was calculated assuming the catastrophic failure of a 10 in. stainless steel line underneath the pool.

The maximum temperatures reached by the fuel are plotted as a function of operating power density in Figure 2 for several cooling or delay times between reactor shutdown and loss of coolant from the core. For reactor operation with maximum power density of less than 21 kW/element for standard and 23 kW/element for FLIP fuel, loss of coolant water immediately upon reactor shutdown would not result in fuel clad failure and subsequent release of fission products.

If the reactor is operated 70 MW-hrs or less per week, power generation per element values approximately 20% higher can be utilized. Thus, 25 kW/element for standard and 28 kW/element for FLIP fuel are established. A comparison of decay heat generation versus time following loss of coolant for infinite reactor operations and 70 MW-hrs per week cycle operation are shown in Figure 3.

FIGURE 1 STRENGTH AND APPLIED STRESS AS A FUNCTION OF TEMPERATURE FOR 1.7 AND 1.6 H-Zr TRIGA FUEL

FIGURE 2 MAXIMUM TEMPERATURE IN FUEL ELEMENT FOLLOWING LOSS OF COOLANT

FIGURE 3 DECAY HEAT POWER GENERATION FOLLOWING LOSS OF COOLANT FOR INFINITE REACTOR OPERATION AND PERIODIC REACTOR OPERATION

Appendix V - Safety Evaluation Of The Irradiation Of Explosives In The Beam Port No. 4 Neutron Radiography Facility

The proposed changes were intended to provide for the radiography of explosive materials within the beam port 4 facility. Located on the north side of the lower research level, the facility consists of a concrete block structure installed adjacent to the pool shield wall to contain and shield a thermal neutron beam extracted from beam port 4 (see Figure 1). The cave structure is designed for remote positioning of samples with the beam port in operation. A hydraulic operated shield door is raised to shield the neutron beam during loading of a sample, then lowered for the exposure. Since the specification restricts explosive materials from areas where the reactor safety systems would be endangered, the most serious accident involves the detonation of a charge while it is being radiographed. The beam port is exposed at this time and its rupture could result in loss of pool water. The analysis therefore centered on predicting the pressure developed on the weakest part of the beam port.

Empirical studies are available concerning the prediction of peak pressures within a vented or unvented structure.^{1,2} The quantities involved in this calculation are the ratio of the amount of explosive (W, in pounds equivalent TNT) to the volume of the containment structure (V, in ft³) and the vent area through which the pressure may be released. For a more conservative analysis the vent area will be assumed to be zero, i.e., the entire blast is contained within the radiography facility. This ignores two open doorways with a combined surface area of 27 ft². The quantity of explosive will be considered to be the maximum of 5 pounds allowed in the facility. For a containment volume of 270 ft³ (see Figure 1),

$$\frac{W}{V} = \frac{5 \text{ lbs}}{270 \text{ ft}^3} = 0.0185 \text{ lb / ft}^3$$

From Figure 2 the peak pressure (P_{qs}) is 100 psi.³ Using the conservative assumption that this maximum pressure will be transmitted through shields and collimators, the force exerted upon the end of the beam port is,

$$\begin{aligned} 100 \text{ psi} \times A_{BP} &= 100 (\pi) (3)^2 \\ &= 2800 \text{ lbf} \end{aligned}$$

(A_{BP} = the cross sectional area of the beam port.)

The rupture strength of the beam port is determined by the fillet weld between the beam tube and end plate (see Figure 3). The area of the shear plane is,

$$A = \pi t(D-t),$$

¹ Kingery, C. N., Schumacher, R. N., and Ewing, W. O., Jr. (1975), "Internal Pressures from Explosions in Suppressive Structures", BRL Interim Memorandum Report No. 403, Aberdeen proving Ground, Maryland, June 1975.

² Sewell, R. G. S. and Kinney, G. F. (1974), "Internal Explosions in Vented and Unvented Chambers", Minutes of 14th Explosives Safety Seminar. New Orleans, 8010 November 1973, Department of Defense Explosives Safety Board, pp. 87-98.

³ Baker, W. E., (1978), "Internal Blast Loading", A Short Course on Explosion Hazards Evaluation, Southwest Research Institute, pp. 3-16 to 3-34.

where t = weld thickness = 0.177 in. (see Figure 3)
 D = tube O.D. = 6.75 in.
 $A = \pi (0.177)(6.75 - 0.177)$
 $A = 3.65 \text{ in}^2$

For a shear yield stress of 12,000 psi and applying a safety factor of 4 the design weld shear force for failure is

$$RS = \frac{(3.65)(12,000)}{4} = 10,950 \text{ lbf}$$

A comparison to the calculated peak blast pressure which results in a 2,800 lbf reveals a wide safety margin for failure of the welds.

The limiting circumferential stress and bursting pressure for the beam port extension tubing is determined as follows:

$$S_c = P_c r / t$$

where P_c is the internal bursting pressure,
 r is the inside radius of the tube, and
 t is the tube thickness.

The bursting pressure is

$$P_c = \frac{(20,000 \text{ psi})(0.375 \text{ in.})}{3 \text{ in.}} = 2500 \text{ psi}$$

The longitudinal tearing pressure is twice the bursting pressure for failure and is

$$P_t = 5,000 \text{ psi.}$$

The failure strength of the beam port extension is determined by examination of the weld stress between the beam tube and end plate and the rupture (circumferential) and tearing (longitudinal) stresses. When these pressures for failure are compared to the 100 psi peak blast pressure, again a wide safety margin is indicated. Thus, failure of the weld at the end plate would be the probable failure mode of the beam port.

In predicting the peak blast pressure it was also assumed that the peak pressure curves would apply. This is valid if it is noted that the blast occurs at the cassette location, approximately twenty five feet from the end of the beam port extension where the pressure is calculated. This distance, the beam tube size, and the small aperture in the beam tube, provide adequate dampening to eliminate the transients which precede the "uniform state" condition.

The analysis indicates there will not be a rupture of the beam port for detonation of a 5 pound explosive charge during radiography. However, should a rupture occur there is no unreviewed safety question presented, since the analysis for the loss of pool water has already been performed.

Damage to the cave structure was not considered since it serves only as neutron shielding for personnel working in the area. Loss of all or part of this shielding would in no way endanger the reactor or any of its safety systems. Therefore, to simplify calculations and provide the maximum value for the peak blast pressure, the cave structure was assumed to remain intact.

Missiles generated by the blast would be in the form of shrapnel from the sample casing or fragments of structural components. The demineralizer room is nearby but all piping, etc., is behind a 12" reinforced concrete wall as well as the neutron shielding blocks. Thus, the piping is adequately protected. It is possible that a small projectile directed along the axis of the beam port could pass through the aperture and strike the Plate at the end of the beam port extension. However, if we assume isotropic distribution of the emitted fragments, the probability of passing through the aperture is 1.1×10^{-6} . Even though such an occurrence is unlikely, tests were performed to predict the effect such a missile would have on the extension. A plate of similar material (5/16" - 6061 - T6 Al) was fired into repeatedly with a .38 caliber revolver from a distance of 10 yards with negligible effect on the plate. It is doubtful that missiles with equivalent momentum could be generated by the blast. Thus, it is highly unlikely that a fragment would pass through the beam tube aperture, and if it did so, no damage is expected. Even if the end plate were breached, the result would be a loss of pool water which is much less severe than the catastrophic loss of pool water considered in this report. Therefore no new safety consideration is presented.

Materials entering the facility shall be transported to a storage room by the most direct path. Since the peak pressures developed within the lower research level by detonating a 5 lb explosive charge would be less than 2 psi, this action minimizes the possibility of damaging the systems within these rooms. It should be pointed out, however, that the only safety systems susceptible to damage are water systems which are located behind 1 foot thick reinforced concrete walls and which are contained in stainless steel piping. Any damage to the pipes would result in a loss of pool water less severe than the catastrophic loss discussed earlier. Therefore, the transport of explosive materials in quantities less than 5 lb equivalent TNT through the lower research level presents no unreviewed safety questions.

In evaluating the effect of a fire or an explosion in the neutron radiography facility, it is necessary to fully understand the location of that facility in the building relative to the reactor safety system. The level above the radiography facility contains mechanical and electrical equipment. The floor is six inch thick reinforced concrete. There is another similar floor which defines the upper research level which is constructed of four inch thick reinforced concrete. The control room and all critical safety cables are located on the opposite side of the reactor pool so that it is completely shielded from any explosion that could occur in the neutron radiography facility. Even during transport of the explosives through the lower level of the facility the control room is shielded by two reinforced concrete floors and the control cables by one floor. Since the radiography facility is constructed of some wood and paraffin, it is conceivable that an explosion or electrical short could cause these materials to burn. If this should happen these would be the only materials to burn since there are no other significant combustibles on the lower research level. All of the walls and floors and ceilings are concrete. The smoke and possibly some flames could pass through the air conditioning grates in the ceiling above the neutron radiography facility, but there is no reactor safety equipment located in that area.

All that would happen would be some smoke deposition in the mechanical level of the reactor building. There is an experiment scram which allows the reactor to be shut down from the lower research level which could be damaged by a blast or fire. However, this would probably shut the reactor down since a scram is initiated if the circuit is opened. It is therefore concluded that an explosion and/or fire resulting in the complete consumption of combustibles in the radiography facility would not endanger the reactor or any of its support systems.

SCALE: 1" = 4'-0"

CONTAINMENT VOLUME

FIGURE 1. BEAM PORT 4
NEUTRON RADIOGRAPHY
FACILITY

FIGURE 2 PEAK QUASI-STATIC PRESSURE FOR
TNT EXPLOSION IN CHAMBERS

FIGURE 3 . (MODIFIED)

NSCR NEUTRON RADIOGRAPHY BEAM PORT EXTENSION