JUNE 1979 324 063

SAFETY ANALYSIS REPORT

FOR THE

NUCLEAR SCIENCE CENTER REACTOR
TEXAS A&M UNIVERSITY

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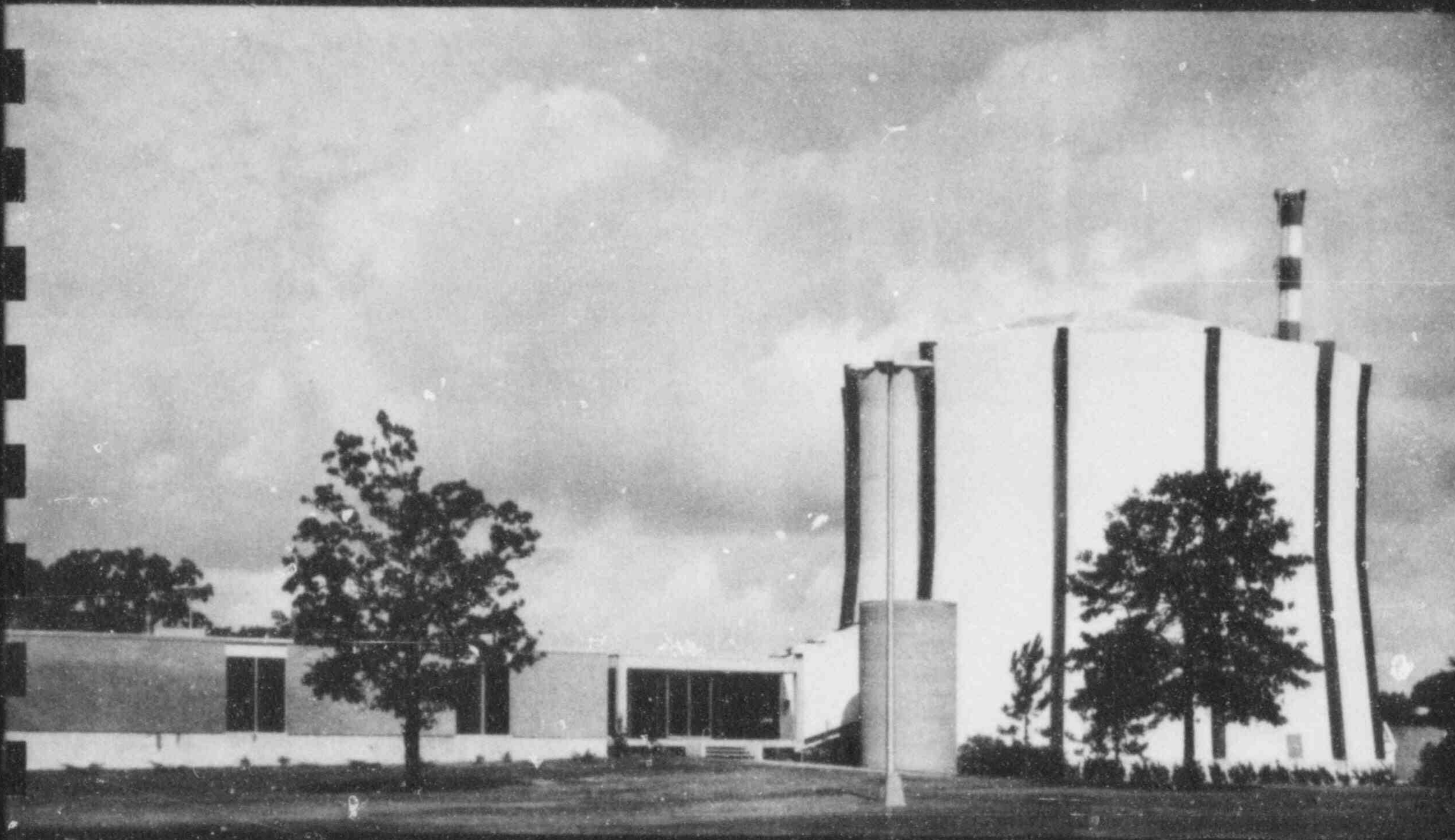

NUCLEAR SCIENCE CENTER
TEXAS ENGINEERING EXPERIMENT STATION
COLLEGE OF ENGINEERING
TEXAS A&M UNIVERSITY
COLLEGE STATION, TEXAS

S A F E T Y A N A L Y S I S R E P O R T

for the

Nuclear Science Center Reactor
Texas A&M University

June 1979

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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 standard TRIGA and/or FLIP TRIGA fuel and contains previously reviewed material from the August 1967 SAR and its supplements dated November 1972 and January 1975.

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.

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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 to where it presently supports an active nuclear program. The facility serves many campus departments, other universities and colleges, several city and state agencies, and other industrial and research organizations. By January, 1965, the use of the facility had increased to where 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 increased to 1,000 Kw. Only three years had elasped from initial reactor operations before 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.

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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 a number of existing research programs and 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 pulsing up to $3.00 reactivity insertion. The inherent safety of the TRIGA fuel allowed increased flexibility and utilization of the reactor. Pulsing was possible 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 plan covered a period of 3 1/2 years to completion in mid 1969. The plan not only changed the initial facility physical plant but also established a new reactor program.

Operating experience with the standard TRIGA fuel revealed a high fuel burnup rate resulting in fuel additions to maintain sufficient 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 for irradiation. 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. Present NSCR operation utilizes a mixed core loading.

*General Atomic - Fuel Life Improvement Program

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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 south-easterly winds occur a large majority of the time on an annual basis

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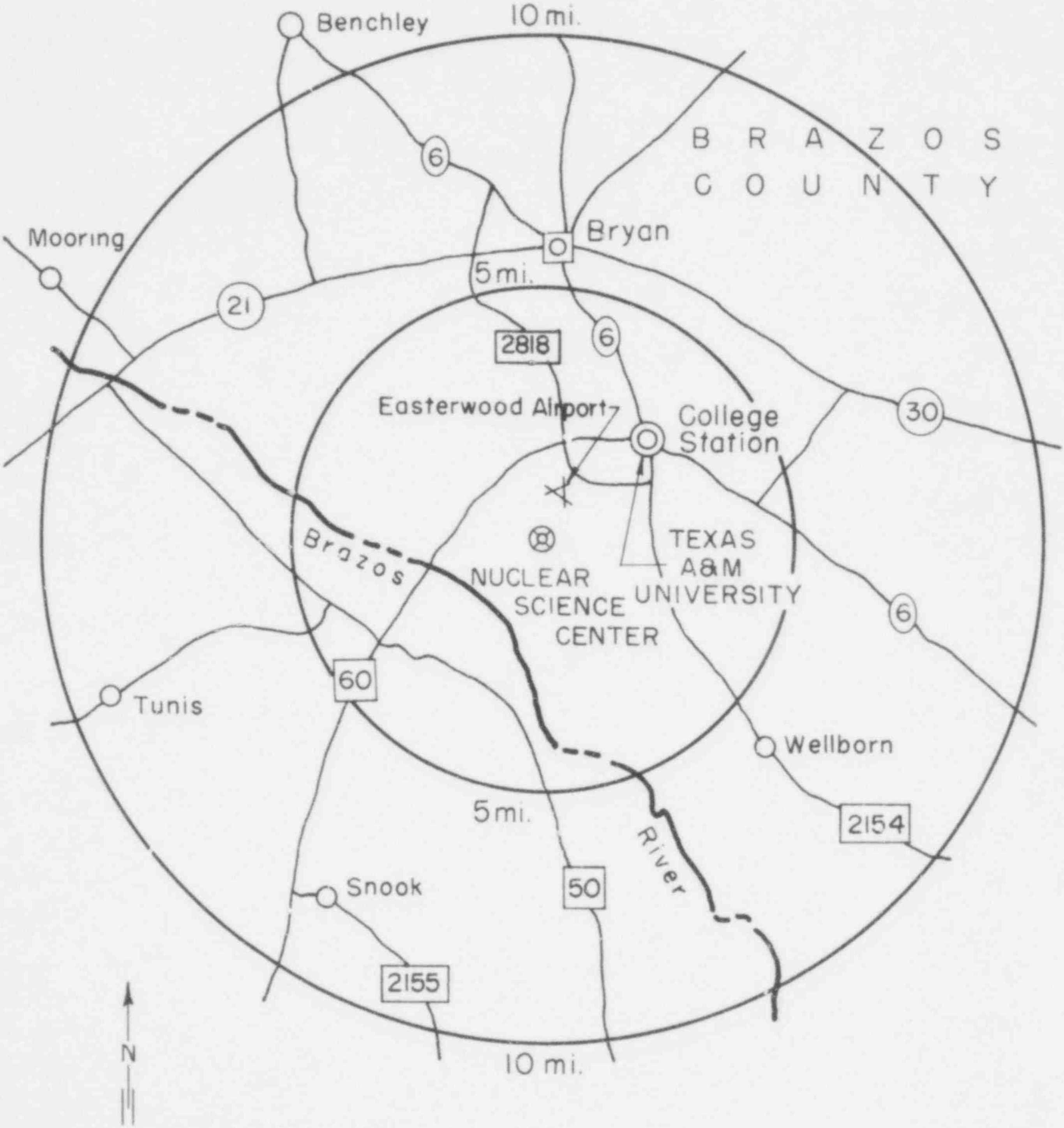

FIGURE 2-1 NUCLEAR SCIENCE CENTER REGIONAL MAP

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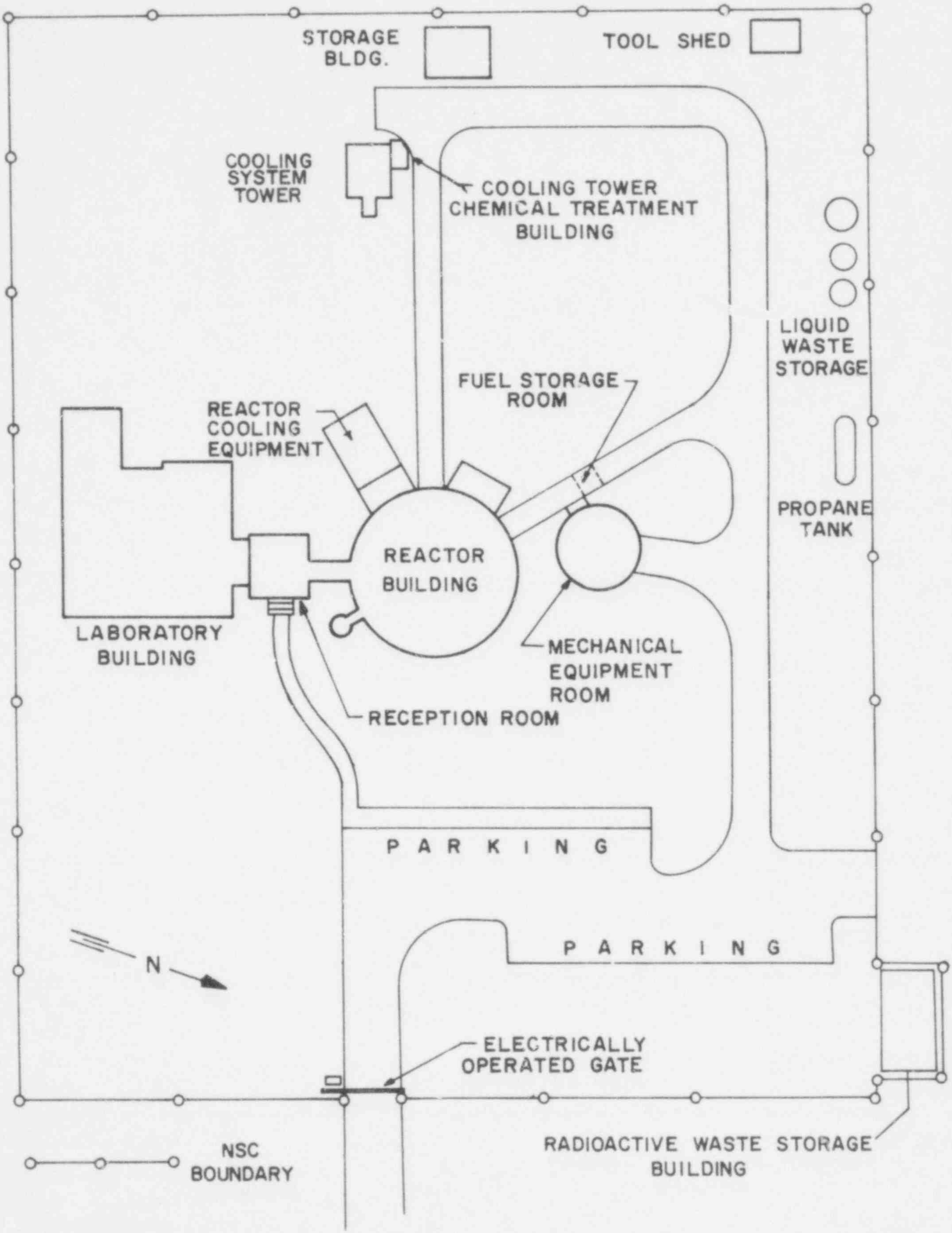

FIGURE 2-2 NUCLEAR SCIENCE CENTER SITE PLAN

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(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.

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 overpressure 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.

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

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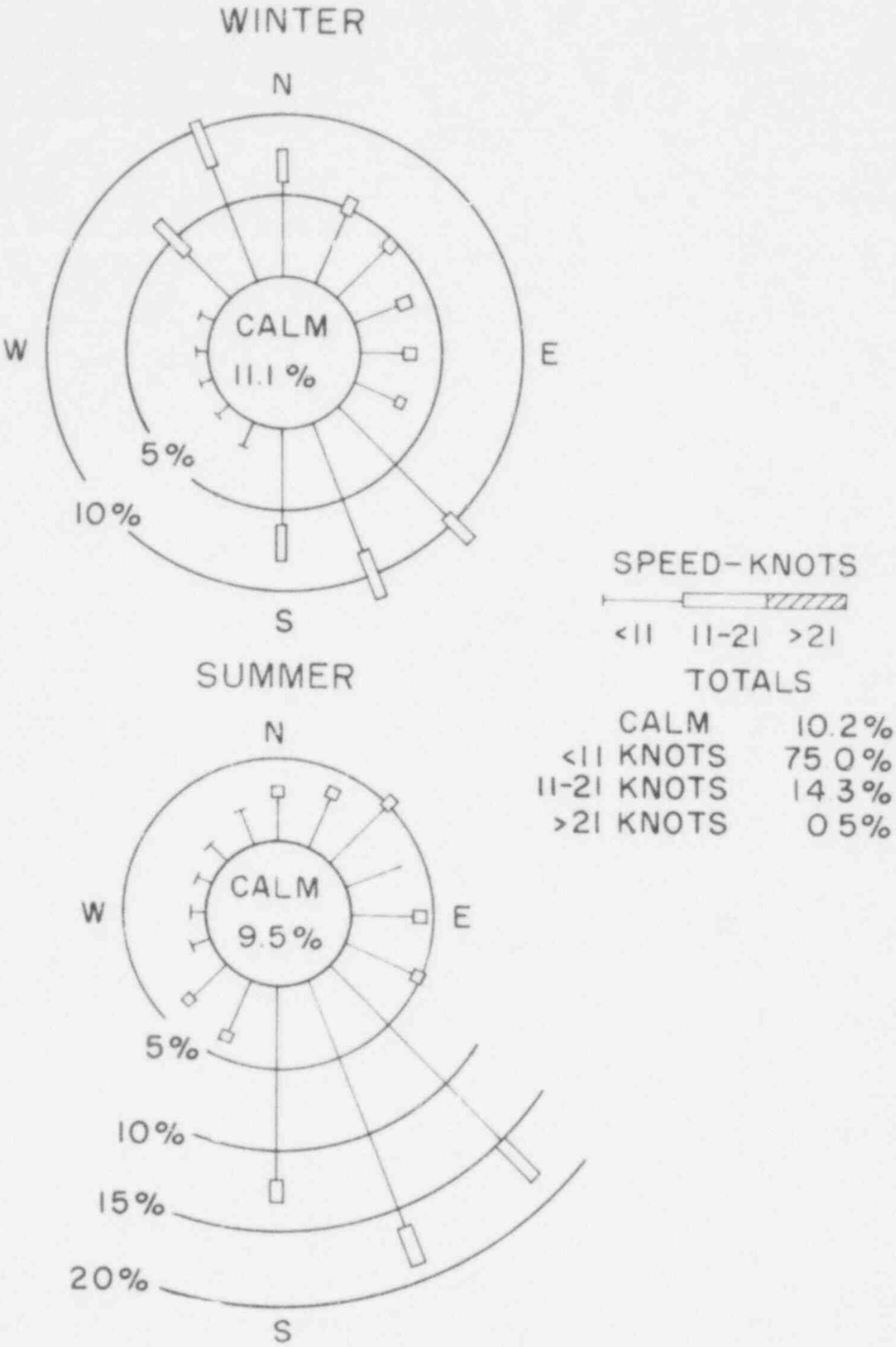

FIGURE 2-3 AVERAGE WIND FREQUENCY DISTRIBUTION AT THE NUCLEAR SCIENCE CENTER

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 acquifer is the Bryan Sandstone which underlies the Easterwood Shale. It is well below the depths required for building excavation.

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.

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

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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 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 all core faces were reflected by graphite. 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 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

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

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

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

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

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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 1/7 of a Mw-year) without a fuel addition.

Inasmuch as funds for a complete FLIP core were not available, it was necessary to consider operation with a core comprised of a mixture of FLIP and standard TRIGA fuel. A precedent for this had been established by General Atomic when they operated a standard core loaded with eighteen centrally located FLIP elements in a fuel test program.[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 satisfaction of producing substantially greater burnup in the standard fuel which is used in 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.

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

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

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

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

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

B. Mechanical Design

1. Reactor Bridge

The reactor core, the control rod drives, the ion chamber cannisters 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 stablizer frame is bolted to the bottom of the grid plate for vertical support. Stainless steel guides on the bottom of the stablizer fit between tracks on the pool floor. This allows accurate repositioning of the reactor core which is essential for numerous experiments. The stablizer 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

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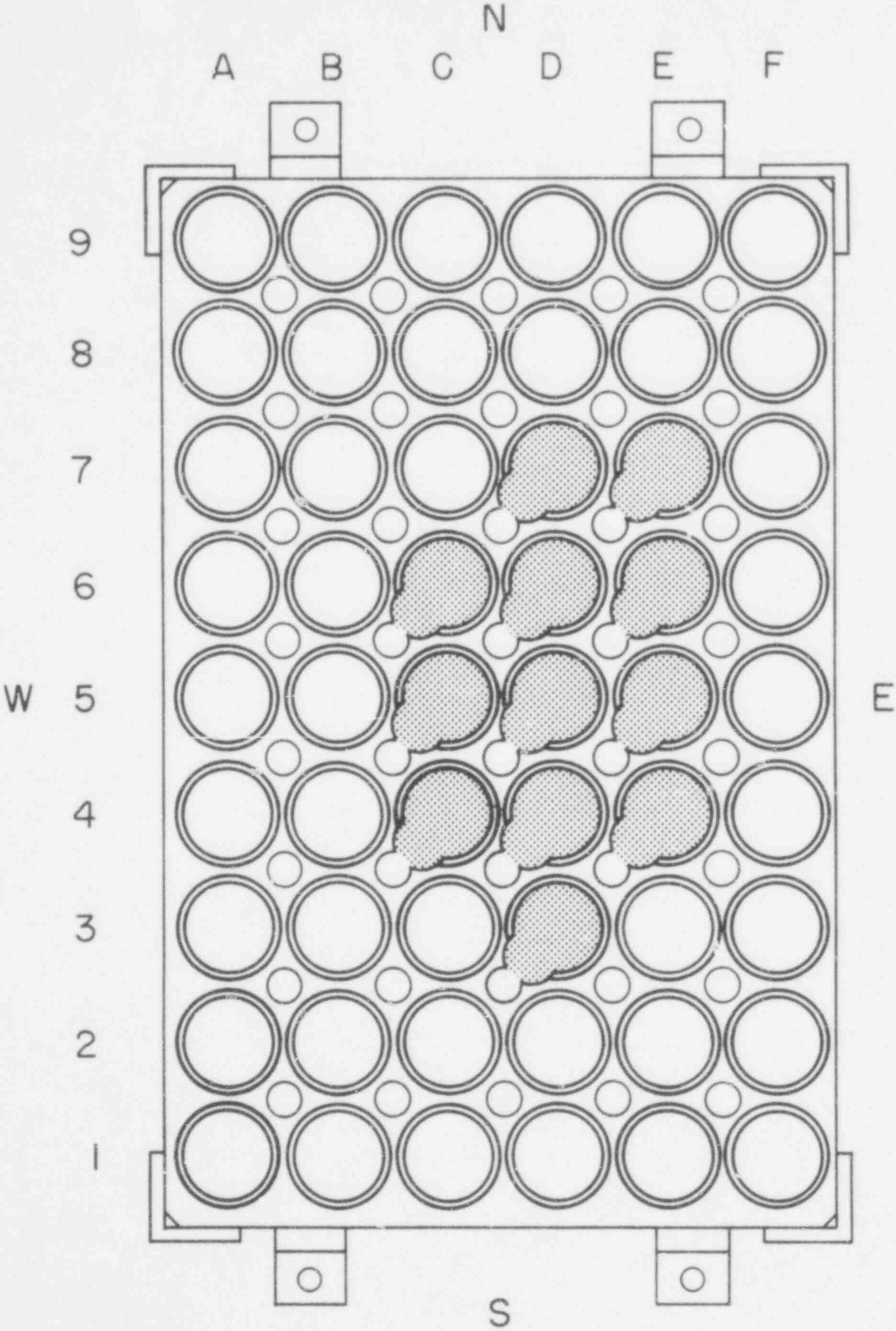

FIGURE 3-1 MODIFIED NSCR GRID PLATE

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elements are considerably heavier than the aluminum plate type fuel elements.

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 98 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 utilizes both standard and FLIP type fuel moderator elements in which a zirconium hydride moderator is homogenously combined with partially enriched uranium fuel. TRIGA fuel elements have a 15 inch long active fuel section. Standard TRIGA fuel contains approximately 8.5 weight % uranium enriched to 20% in ^{235}U. The fuel is homogenously mixed with $ZrH_{1.7}$. A FLIP type element is 8.5 weight-% uranium homogeneously mixed with $ZrH_{1.6}$. The uranium is 70% enriched in ^{235}U and is homogenously loaded with approximately 1.5 weight-% erbium as a burnable poison. Principal design parameters of both standard and FLIP elements are shown in Table 1.

To facilitate hydriding in both the FLIP and Standard elements, a 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 approximately 7.5 pounds with an average ^{235}U content of about 35 grams in the standard element and about 123 grams in the FLIP element. Serial numbers on the bottom end fittings are used to identify individual fuel rods. FLIP fuel elements are also identified by a machined flat at the tip of the top fitting.

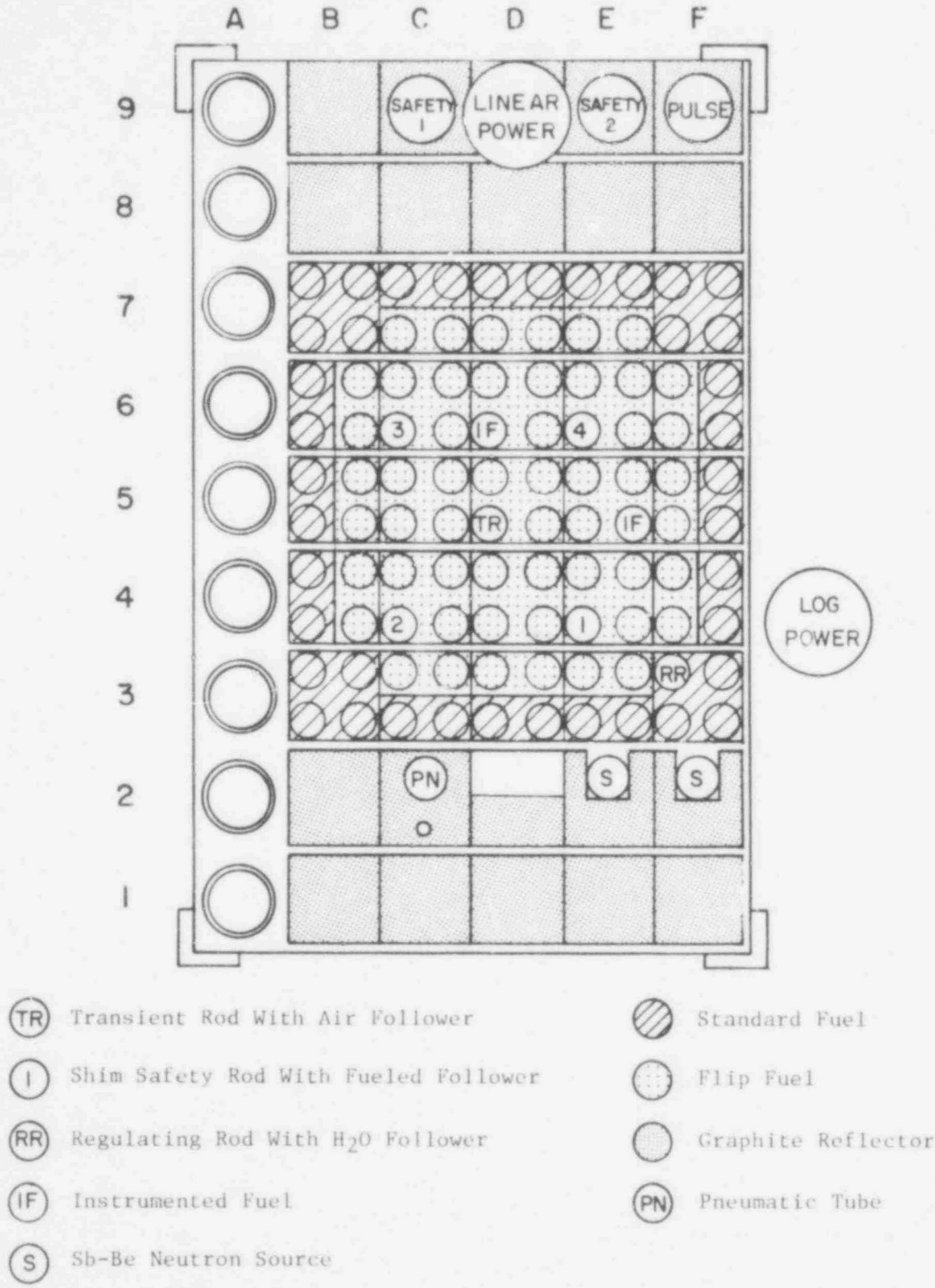

FIGURE 3-2 TYPICAL CORE CONFIGURATION

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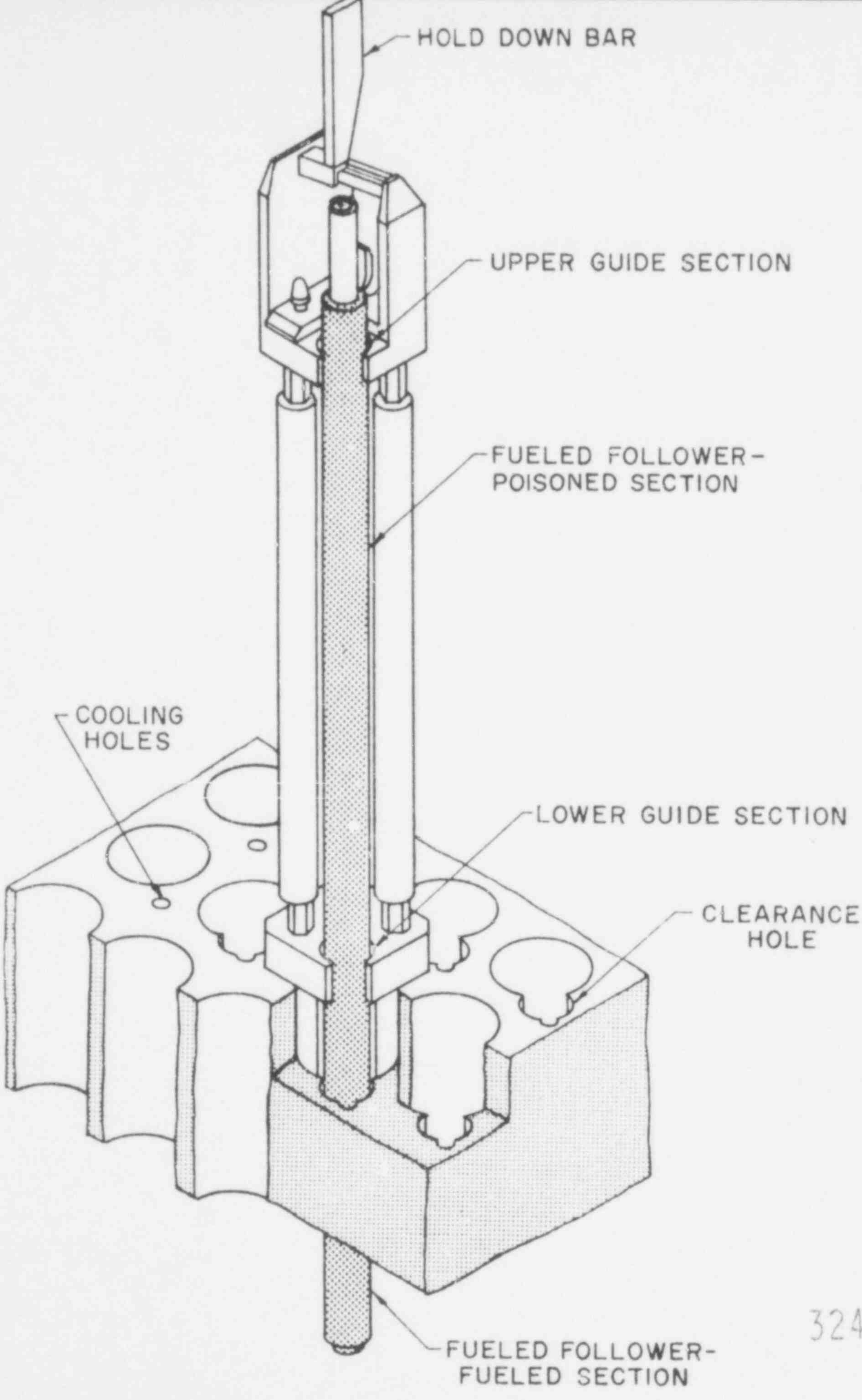

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FIGURE 3-3 FUELED FOLLOWER INSTALLATION

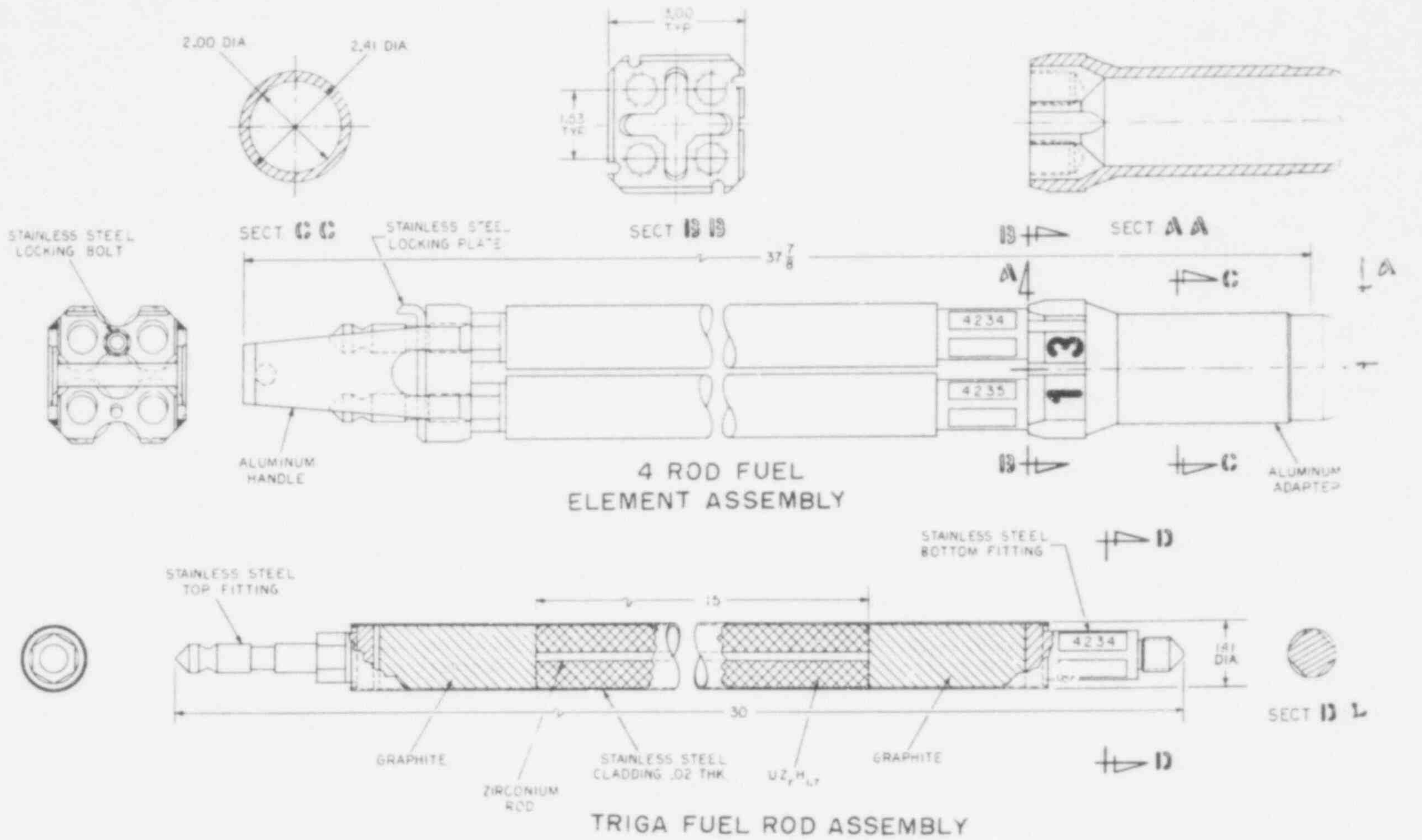

FIGURE 3-4 FOUR ROD FUEL ELEMENT ASSEMBLY FOR THE NSCR

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TABLE I

PRINCIPAL FUEL ELEMENT DESIGN PARAMETERS

Fuel Element Type	FLIP	STANDARD
Fuel-moderator material	U-ZrH	U-ZrH
Uranium content	8.5 Wt-%	8.5 Wt-%
U-235 enrichment	70%	20%
U-235 content (avg) per element	123 g	35 g
Burnable poison	natural erbium	none
Erbium content	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.

Special 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 center. 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 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 canti-lever fashion. The details of the four rod assembly are shown in Figure 3-4.

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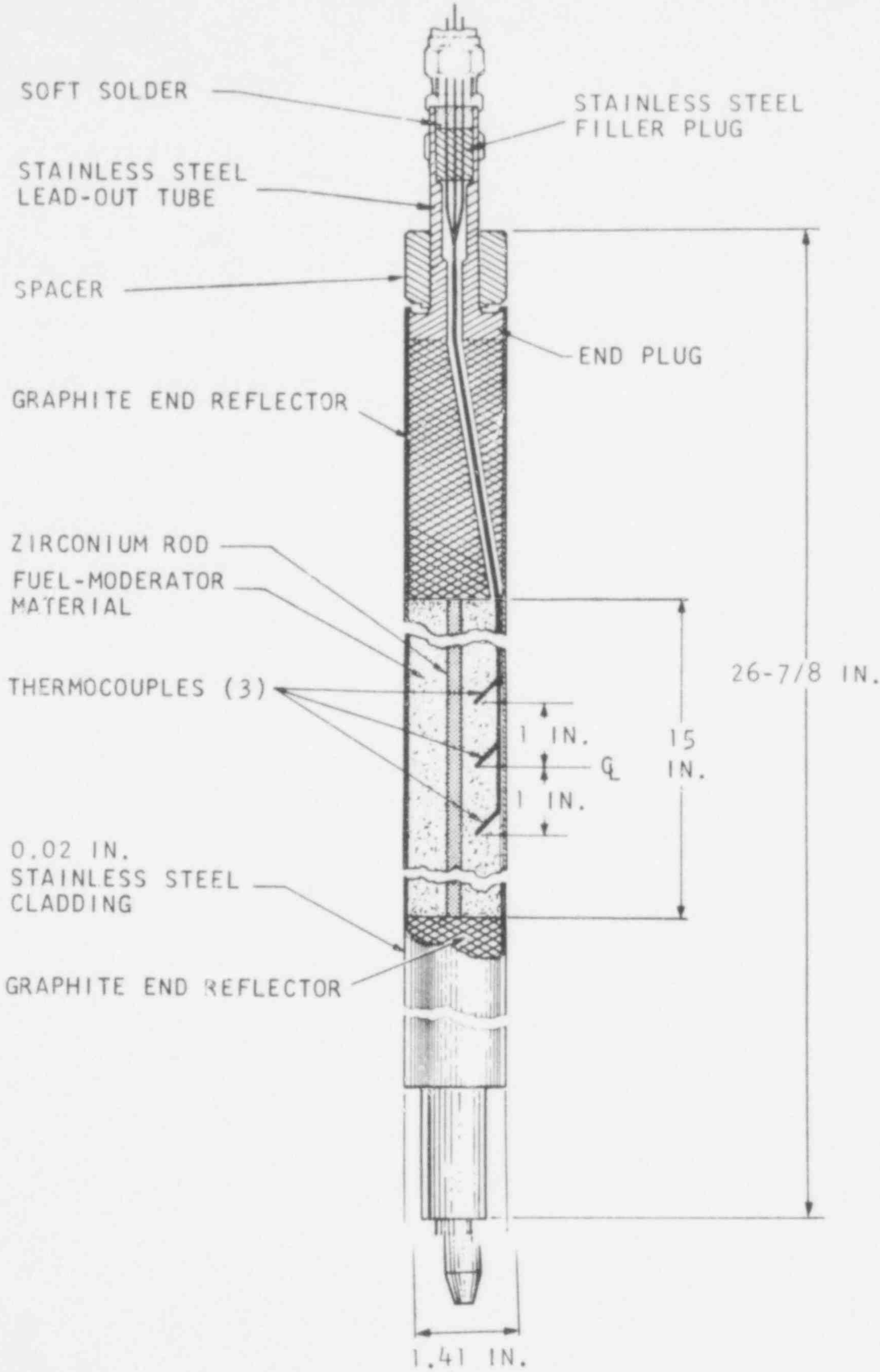

FIGURE 3-5 THE INSTRUMENTED FUEL ROD

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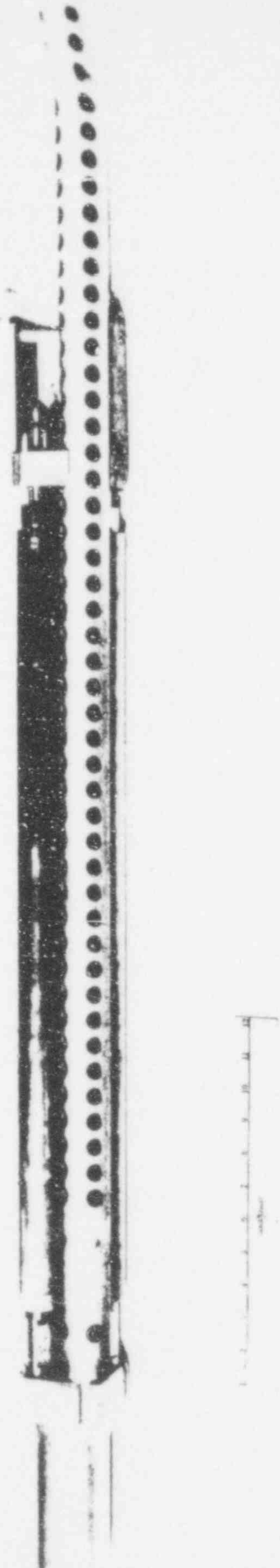

FIGURE 3-6 A CONTROL-BUNDLE WITH INSTALLED CONTROL ROD GUIDE TUBE

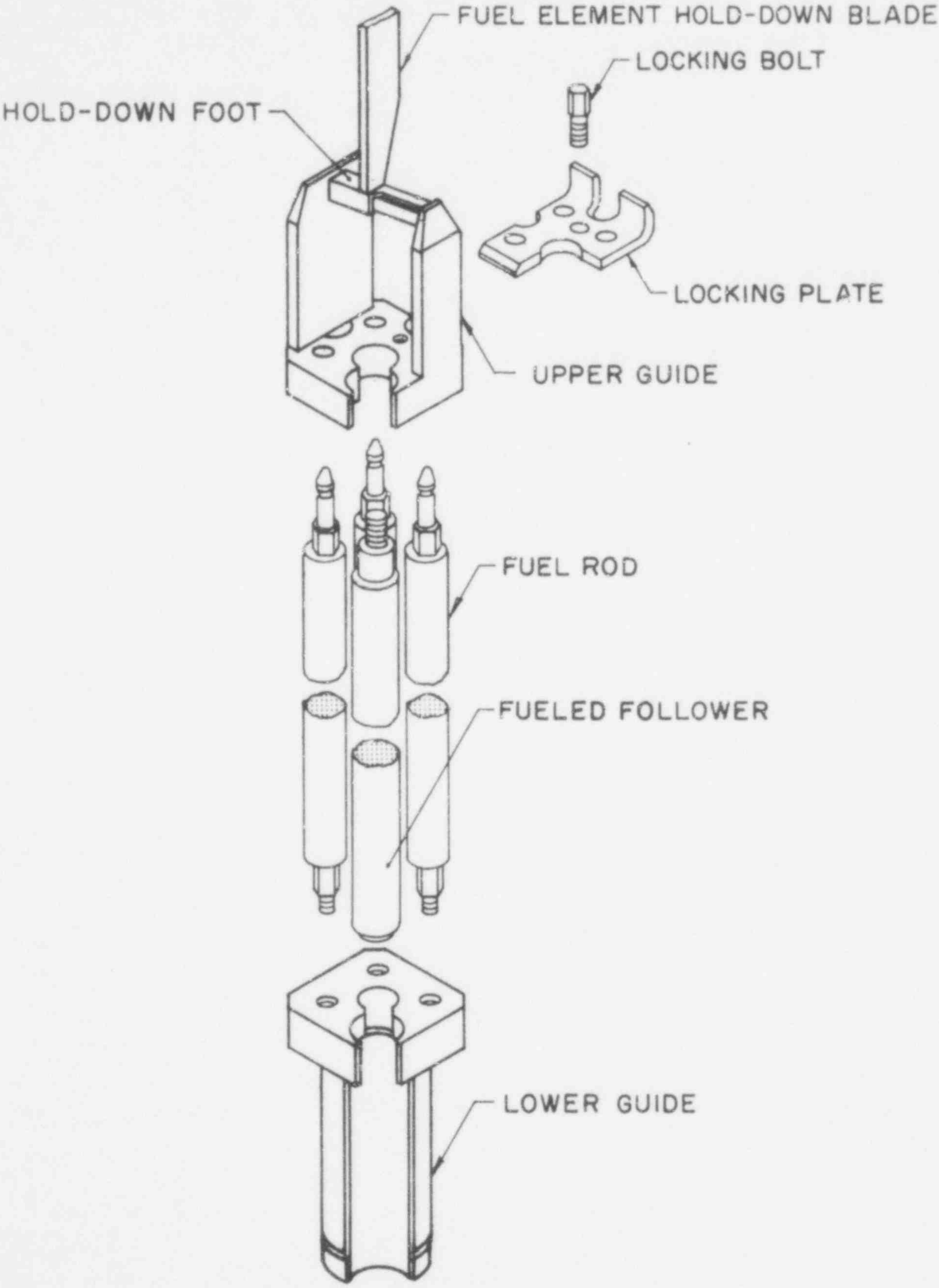

FIGURE 3-7 FUELED FOLLOWER - FUEL BUNDLE ASSEMBLY

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, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding. The regulating rod need not have scram capability and shall be a stainless rod or contain materials as specified for shim-safety control rods. The transient control rod shall have scram capability and contain borated graphite or boron and its compounds in a solid form in an aluminum or stainless steel clad. The shim-safety and regulating control rods may incorporate fueled followers (Figure 3-10), and the transient rod may incorporate an aluminum or air follower.

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 cannot 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

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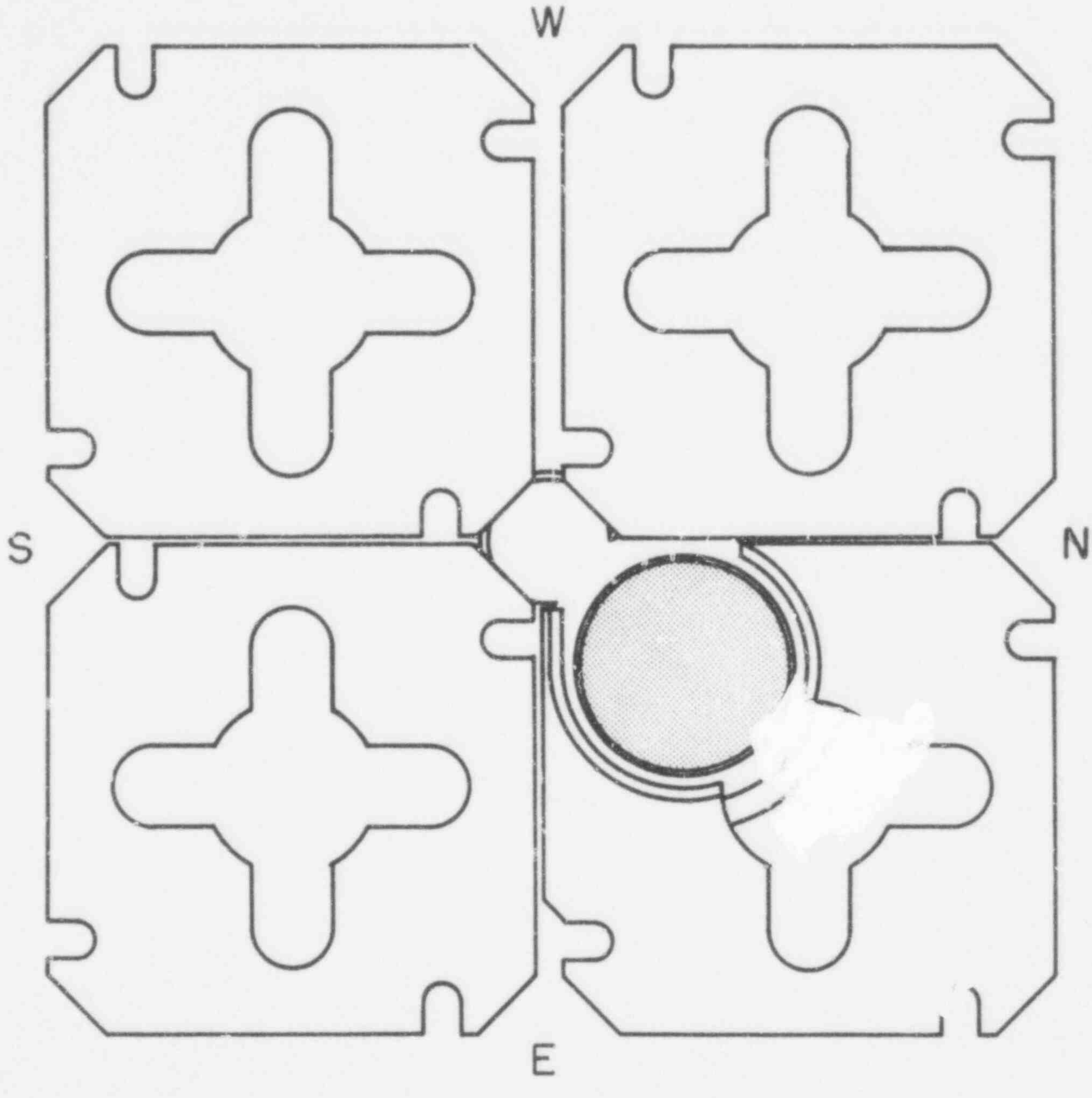

FIGURE 3-8 FUELED FOLLOWER IN FUEL ELEMENT BASE AND ITS RELATION TO ADJACENT BUNDLES

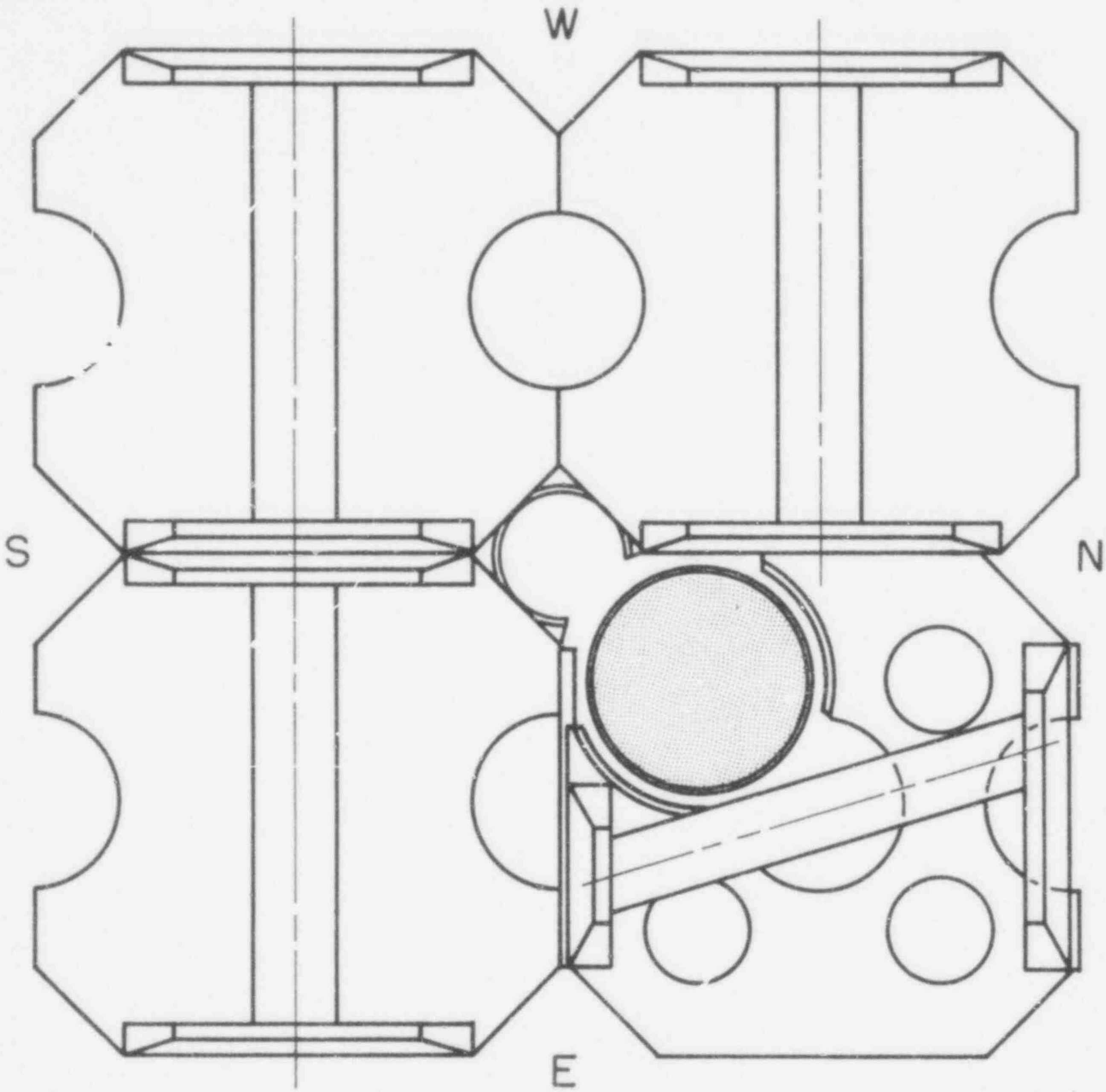

FIGURE 3-9 FUELED FOLLOWER IN FUEL ELEMENT TOP HANDLE AND ITS RELATION TO ADJACENT BUNDLES

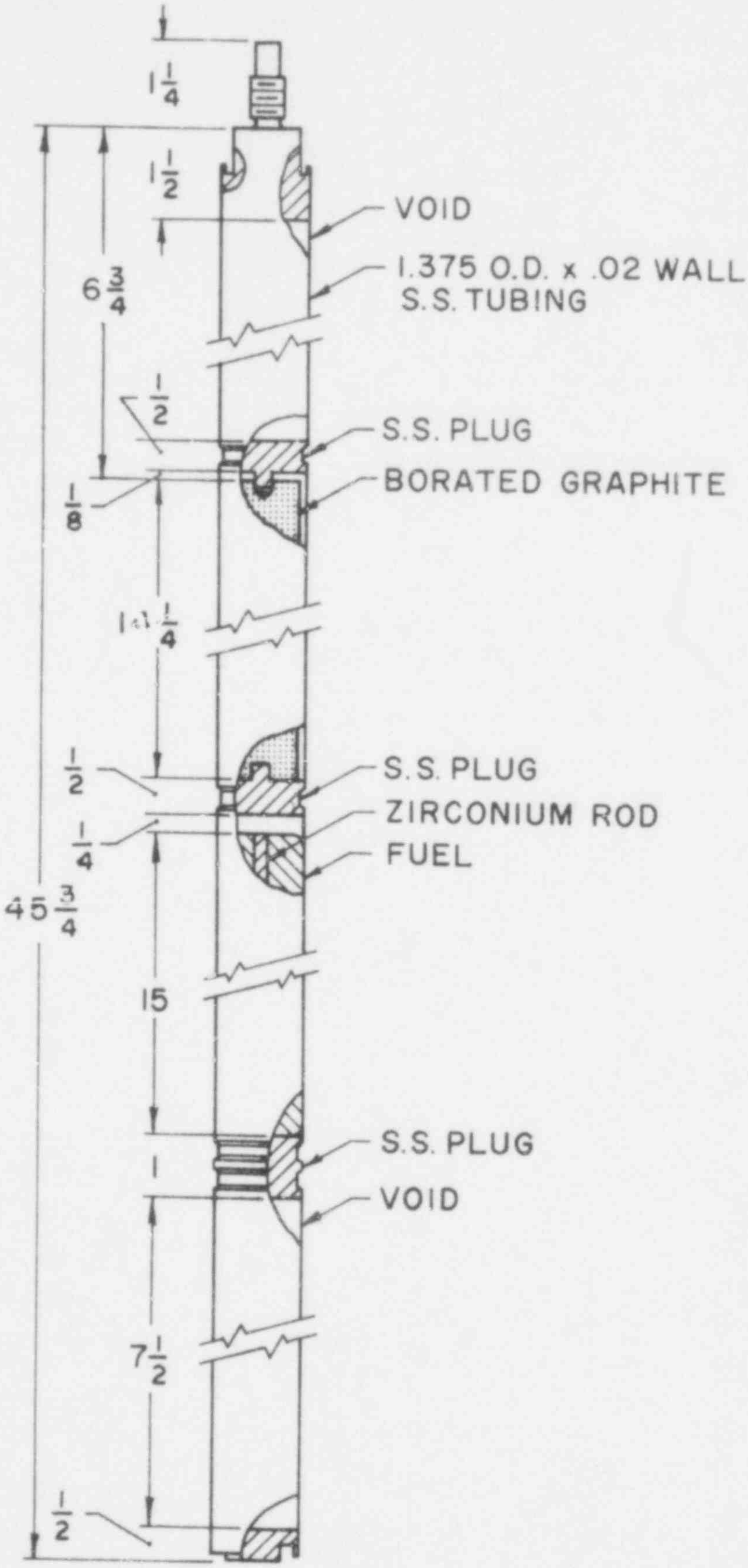

FIGURE 3-10 NSCR FUELED FOLLOWER CONTROL ROD

324 099

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.

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 Power 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.

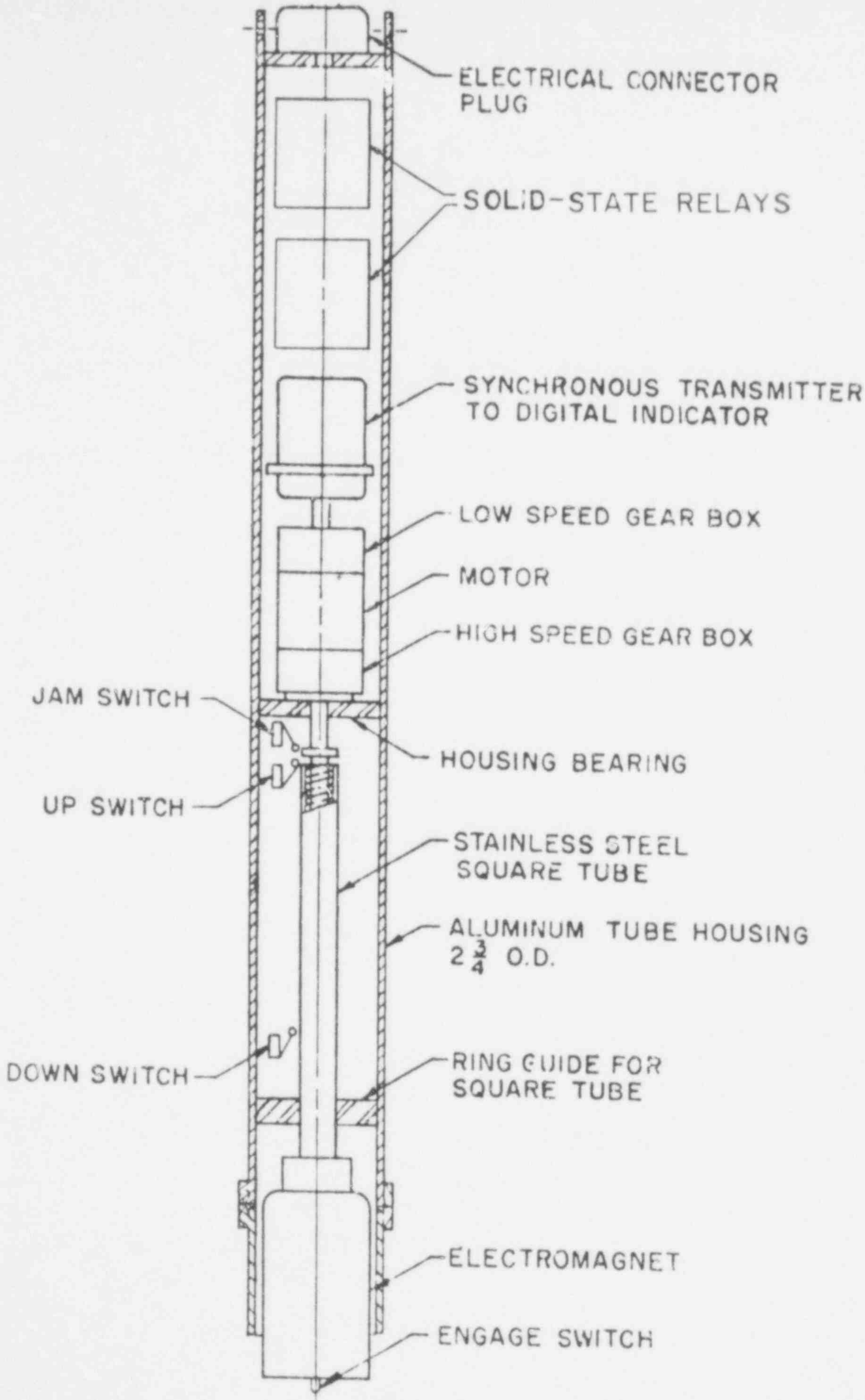

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FIGURE 3-11 CONTROL ROD DRIVE

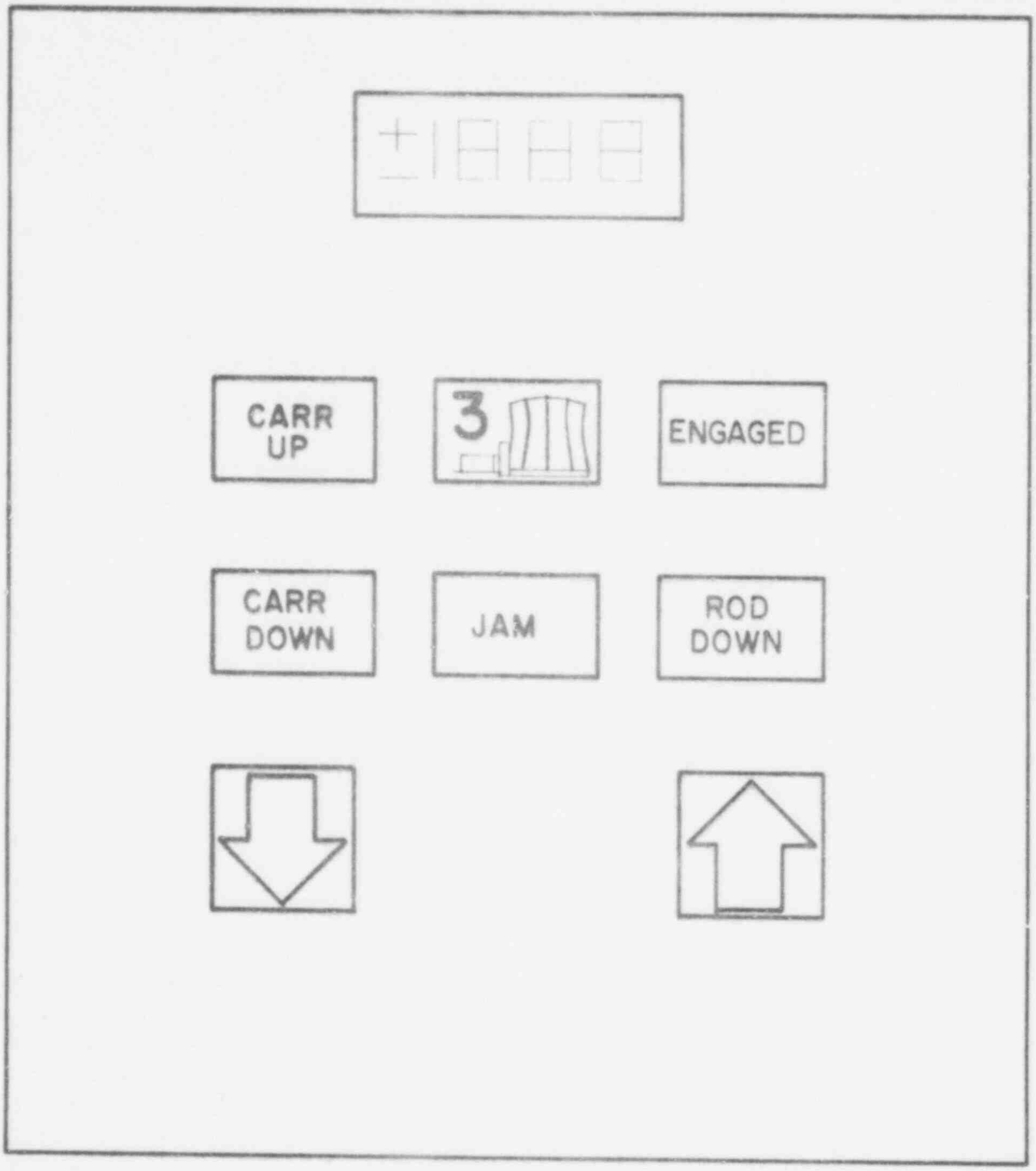


FIGURE 3-12 SHIM-SAFETY CONTROL UNIT

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 45^{o} 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 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 removing 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

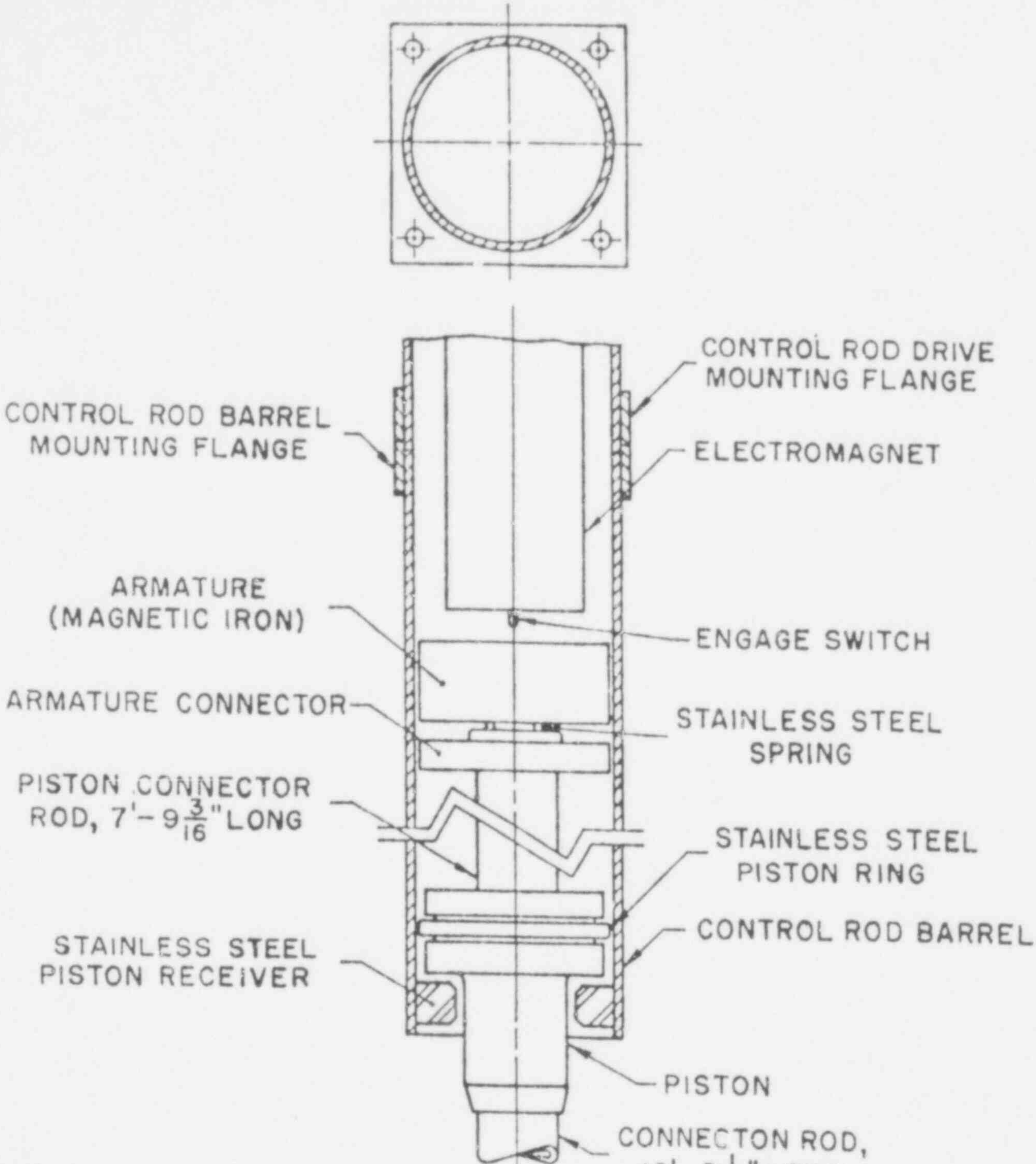


FIGURE 3-13 SHIM-SAFETY ARMATURE AND DAMPENING DEVICE ASSEMBLY

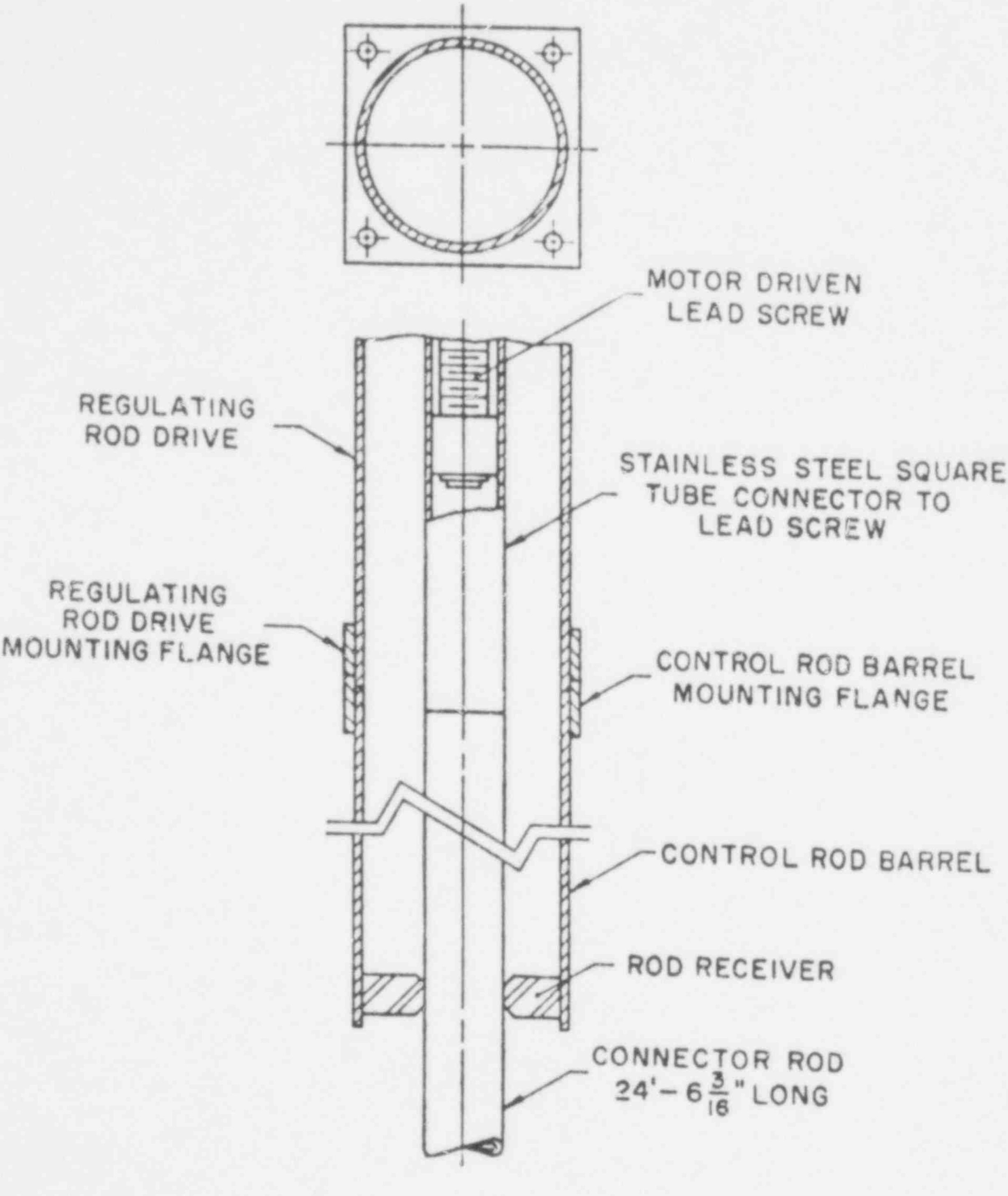
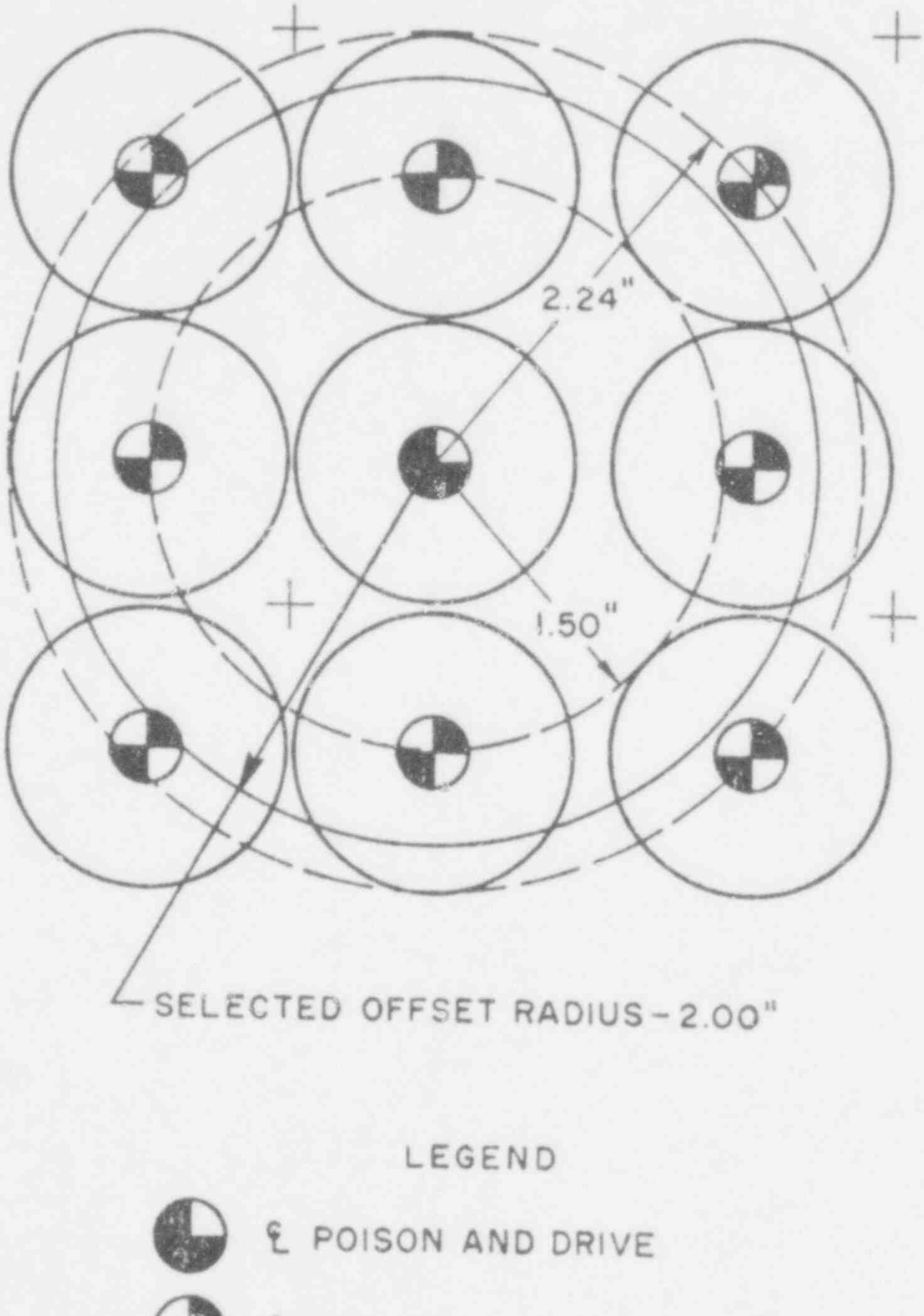


FIGURE 3-14 REGULATING CONTROL ROD ASSEMBLY



LEGEND

℄ POISON AND DRIVE

℄ POSSIBLE POISON LOCATIONS

FIGURE 3-15 CONTROL ROD OFFSET ORIENTATION

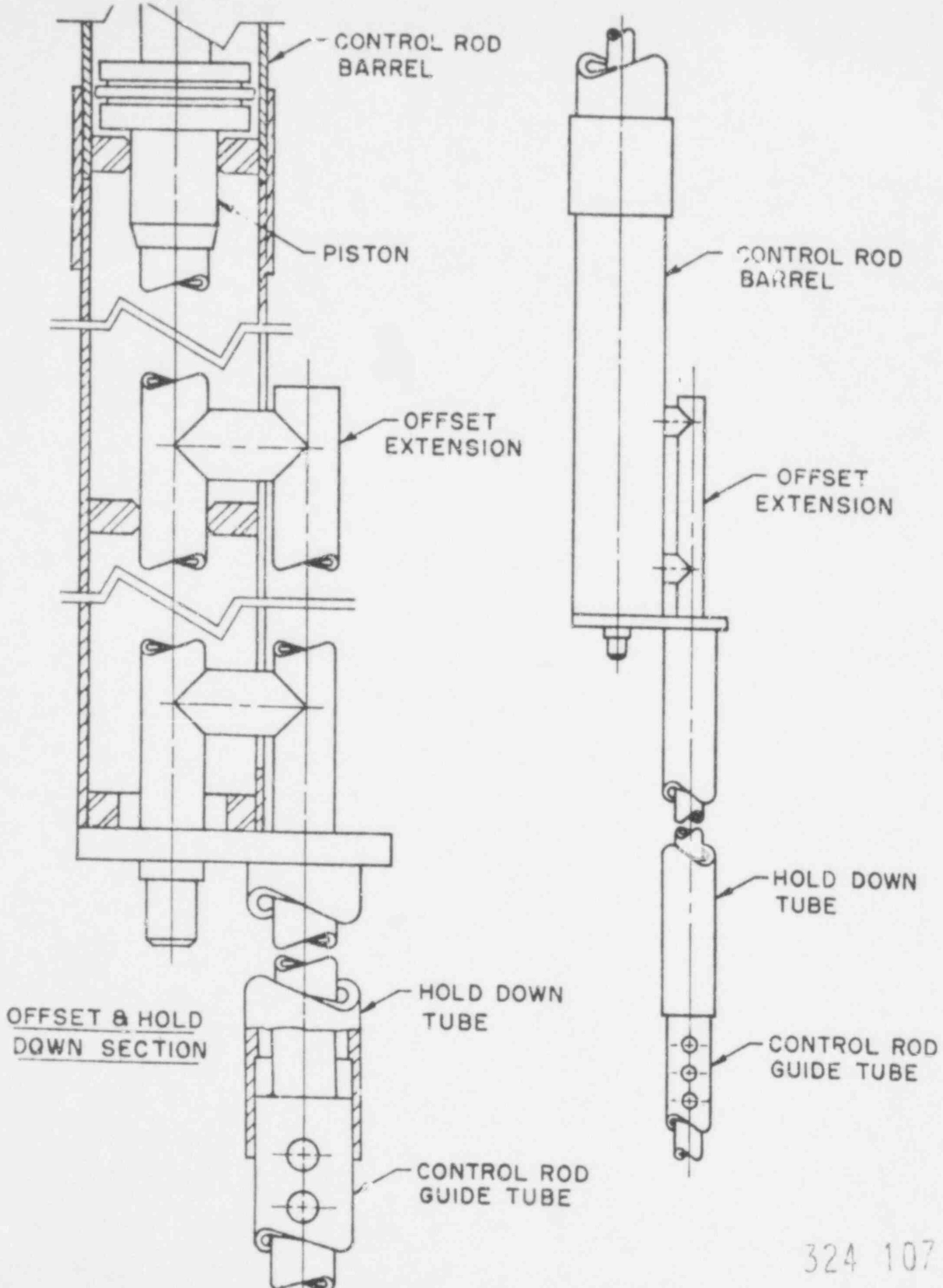


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FIGURE 3-16 CONTROL ROD OFFSET AND HOLD DOWN ASSEMBLY

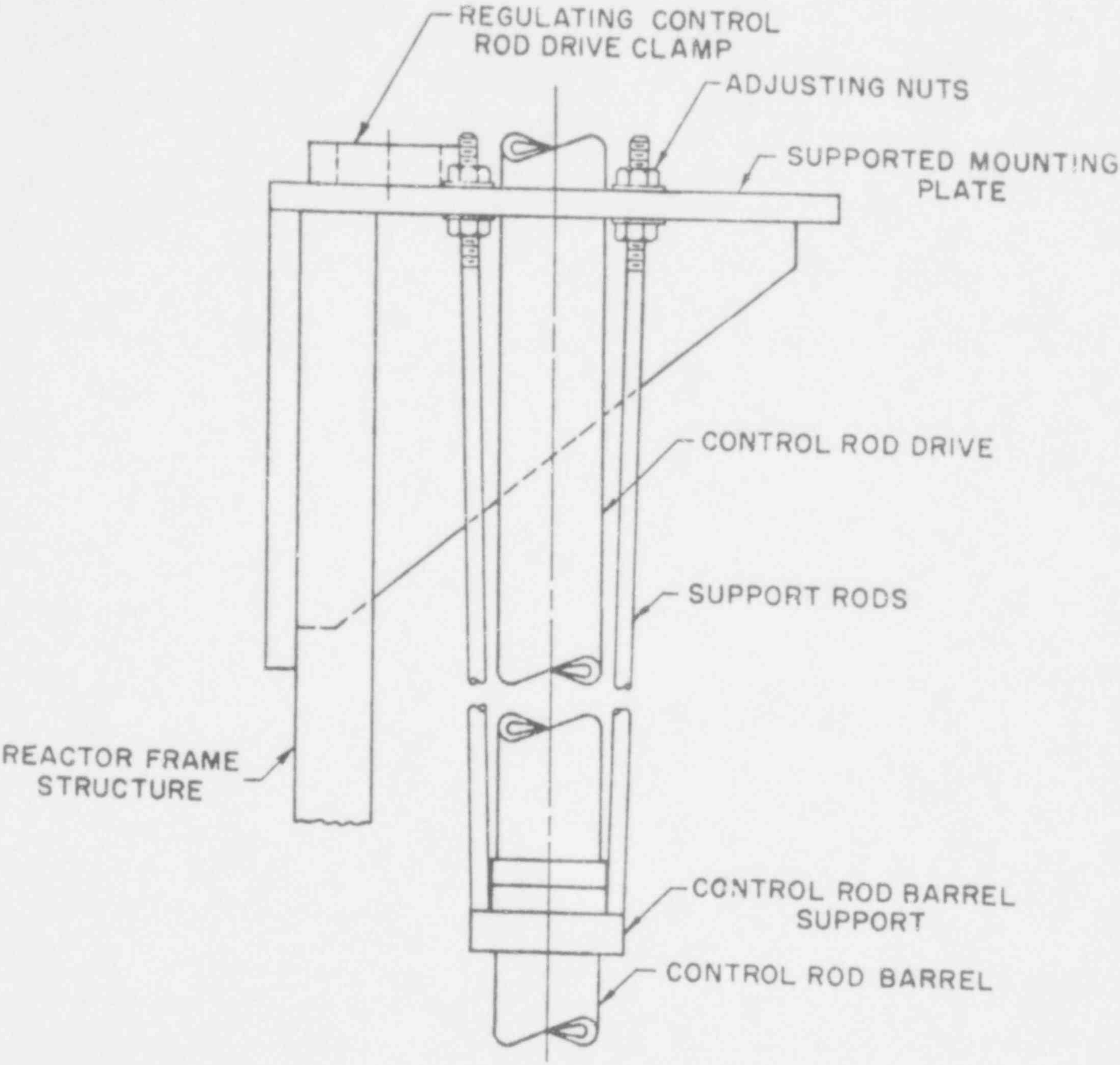

FIGURE 3-17 CONTROL ROD ASSEMBLY SUPPORT MECHANISM

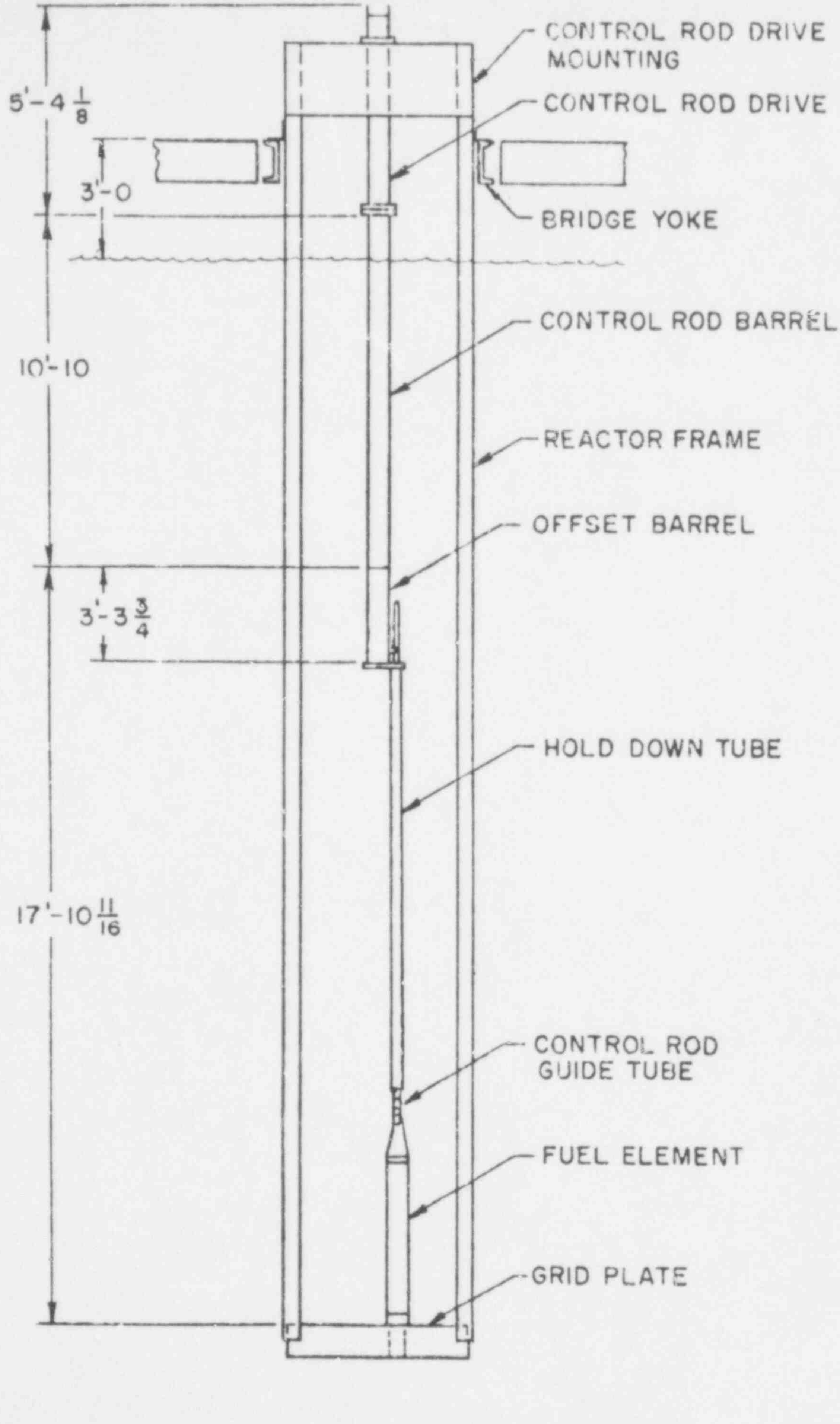

FIGURE 3-18 CONTROL ROD INSTALLATION

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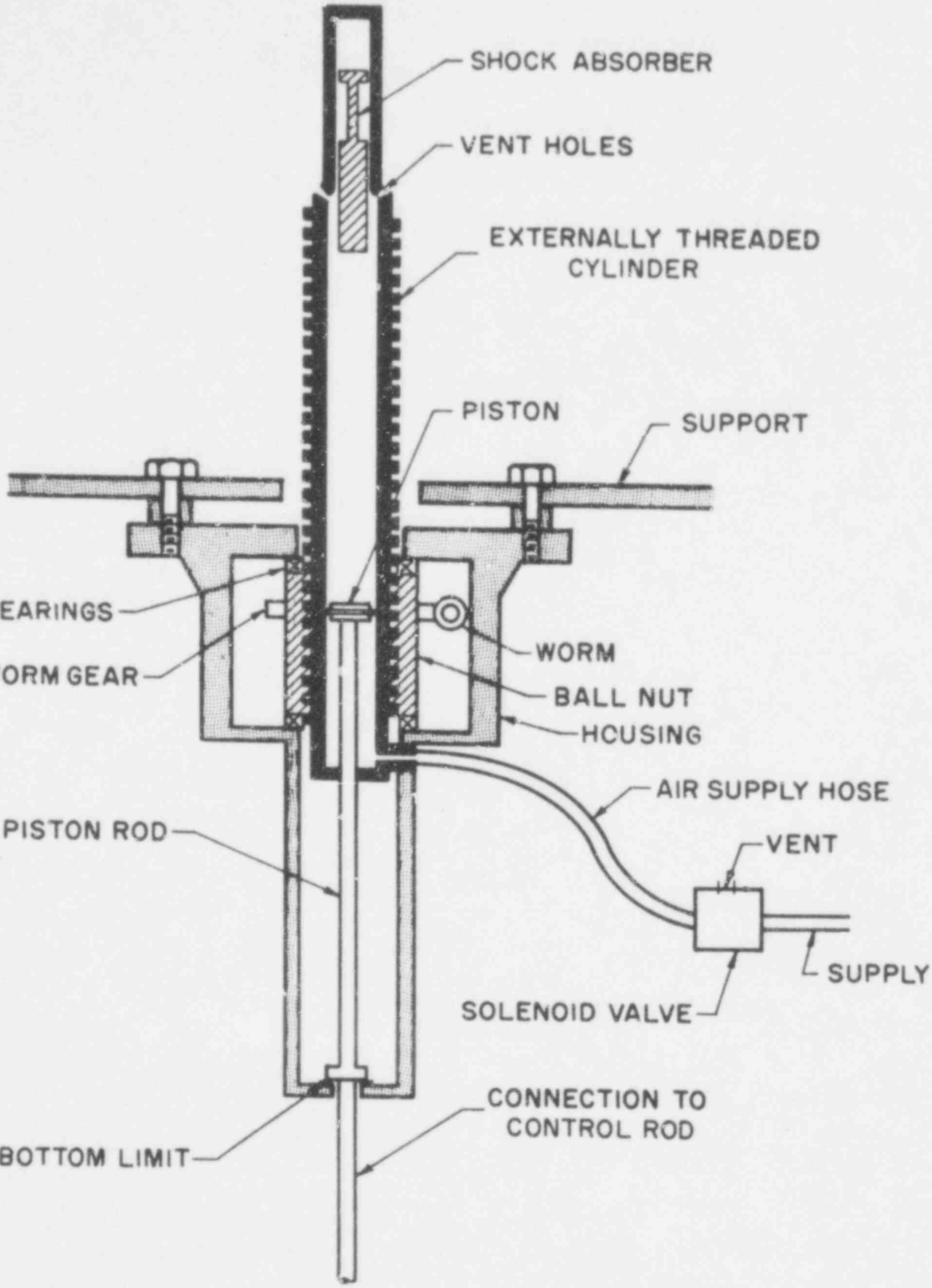

FIGURE 3-19 SCHEMATIC DRAWING OF TRANSIENT-ROD DRIVE

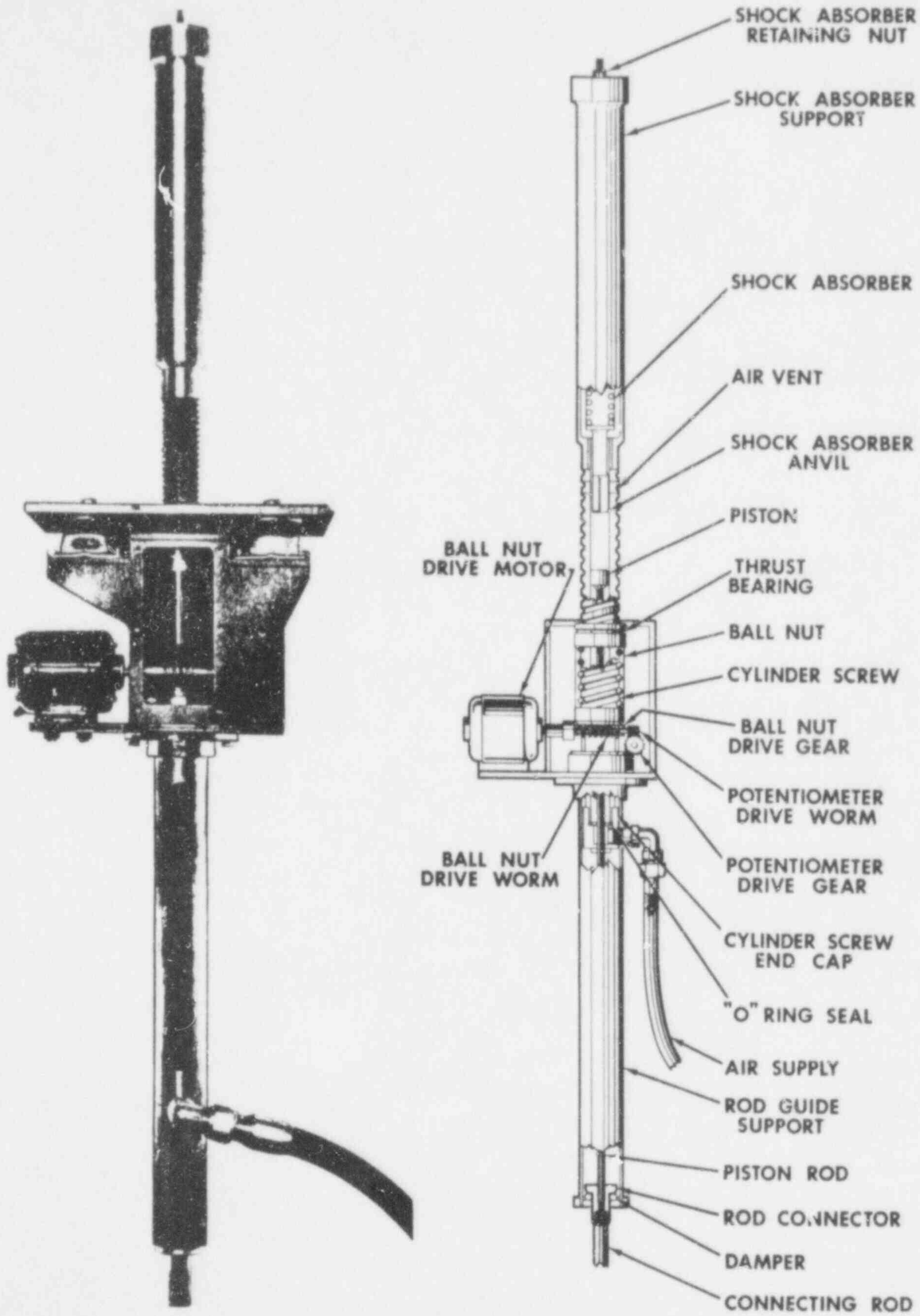

FIGURE 3-20 PNEUMATIC-ELECTROMECHANICAL TRANSIENT-ROD DRIVE

324 111

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.

C. Nuclear Design

1. Standard TRIGA Cores

The design and operating characteristics of standard TRIGA cores are well known as is the inherent safety characteristics of this class of TRIGA reactors.[11] The first NSCR Standard TRIGA core loading reached criticality in August 1968. The following operating characteristics were observed from the initial operational standard TRIGA core for the NSCR:

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

324 112

TABLE II

Steady State Power Level: 1 Mw
Critical Mass: 2,830 grams ^{235}U
Core Mass: 3,325 grams ^{235}U
Maximum Excess Reactivity: \$3.77
Prompt Negative Coefficient of Reactivity: -1.2×10^{-4} $\Delta\kappa/\kappa^{\circ}C$
Power Coefficient (1 Mw): \$3.60
Maximum Pulse Energy (\$2.00 insertion): 14.7 Mw-sec
Total Control Rod Worth: \$11.23
Maximum Pulse Reactivity Insertion: \$3.00

The pulsing of standard TRIGA cores at the present pulsing limit of \$2.35 results in very safe conditions because operational NSCR cores were regularly pulsed at \$3.00 insertions. The pulsing characteristics for a NSCR operational standard TRIGA core are shown in Figure 3-21. The pulsing analysis for standard TRIGA fuel is different from that of FLIP fuel. This is due to a rather constant negative temperature coefficient ($\sim 1.2 \times 10^{-4}$ $\Delta\kappa/\kappa^{\circ}C$) for standard TRIGA as compared to a variable temperature coefficient for FLIP fuel. More than 50% of the temperature coefficient for standard TRIGA cores comes from the "cell effect" or dependent disadvantage factor, and $\sim$ 20% each from doppler broadening of the ^{235}U resonances and temperature dependent leakage from the core. A discussion of the temperature coefficient for FLIP fuel is presented on page 47.

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% $\delta\kappa/\kappa$ (\$3.00).

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 3-22 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 $\sim$ 3 msec. Also shown in Figure 3-22 is a plot of the reciprocal of the measured width at half maximum power versus prompt reactivity insertion.

Figures 3-23, 3-24, and 3-25 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 ($\delta\kappa/\kappa$ - \$1.00). Pulse width and integral powers,

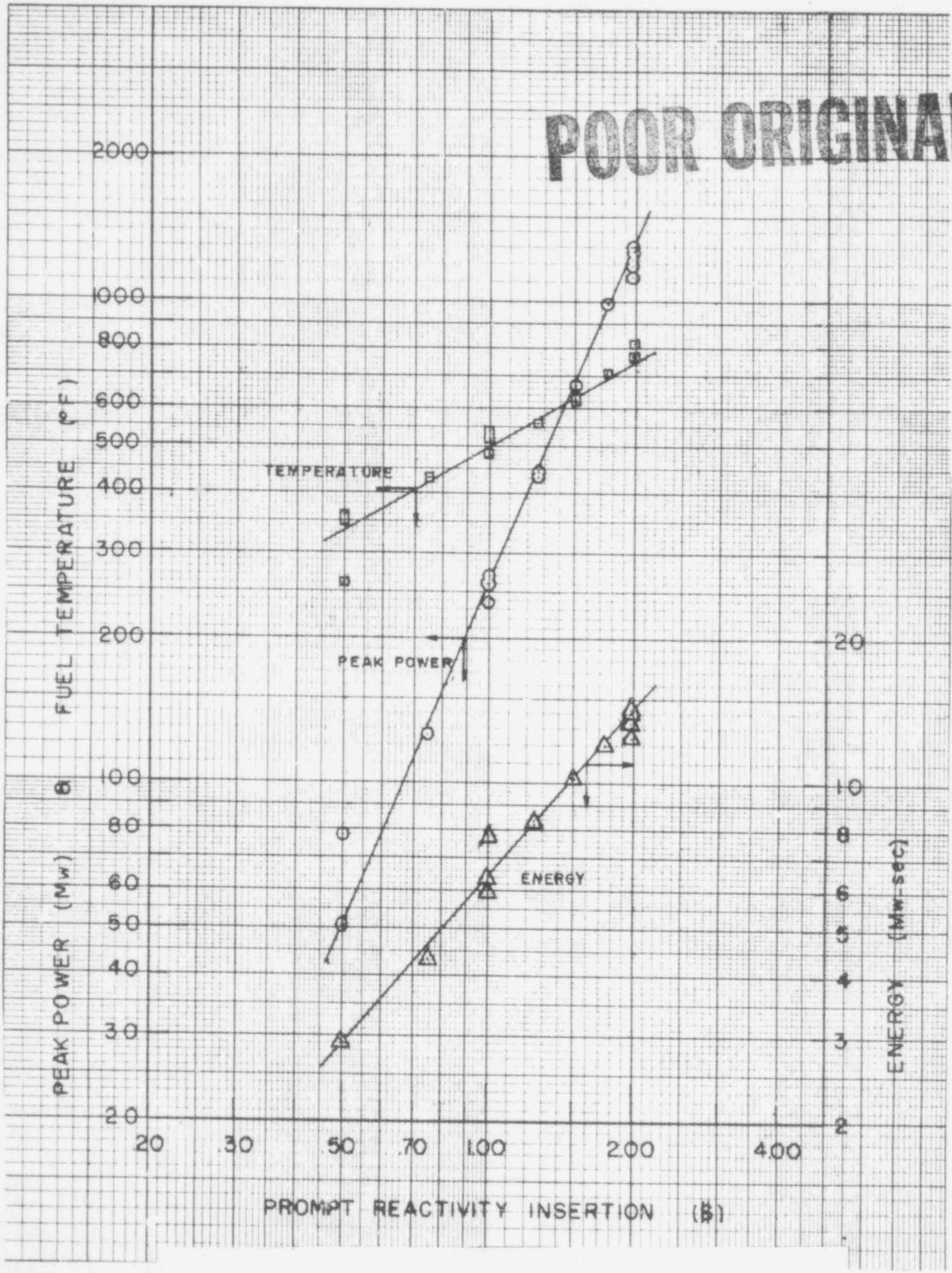

FIGURE 3-21 PULSING CHARACTERISTICS

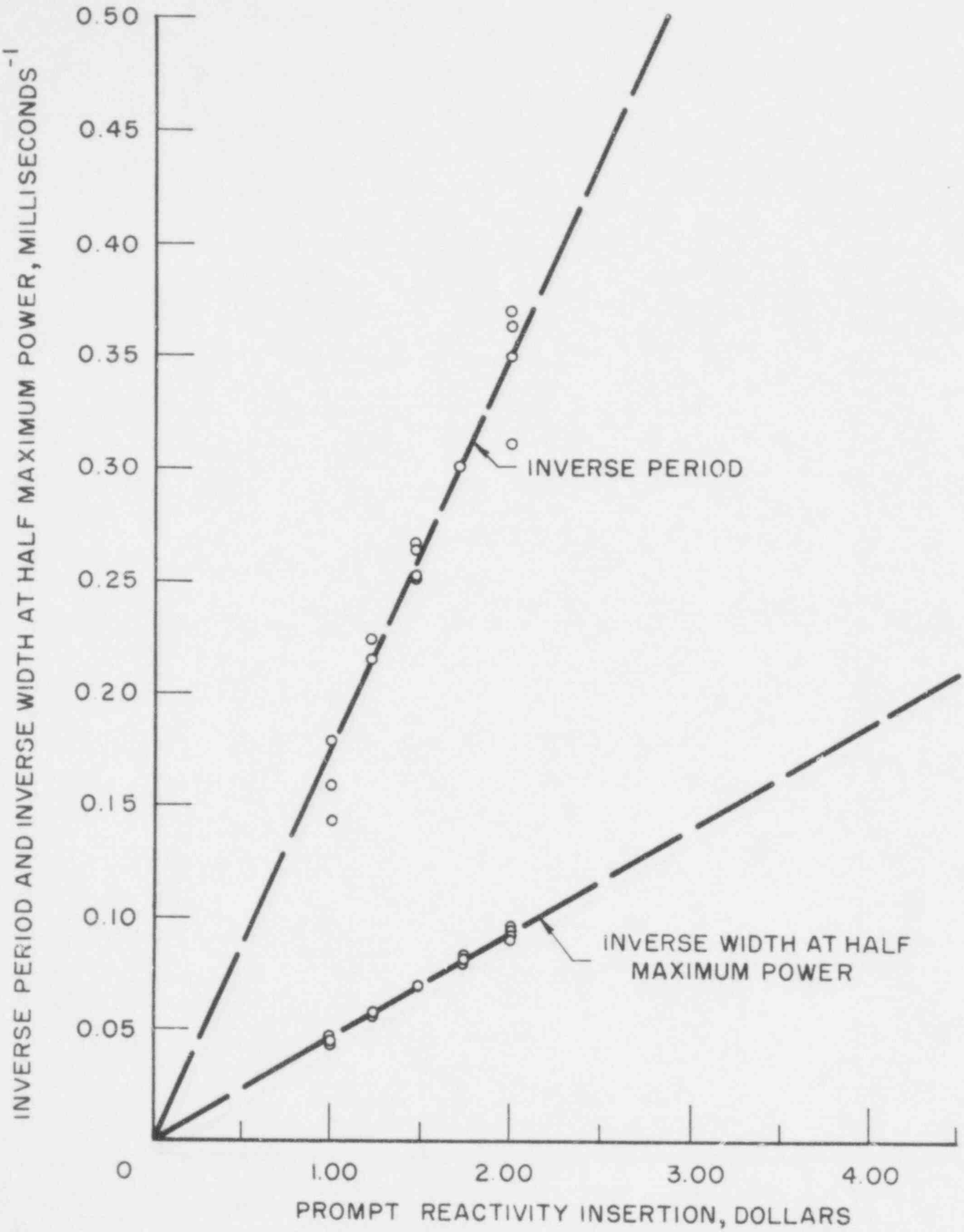

FIGURE 3-22 INVERSE PERIOD AND INVERSE WIDTH AT HALF MAXIMUM POWER VERSUS PROMPT REACTIVITY INSERTION—PROTOTYPE REACTOR

324 115

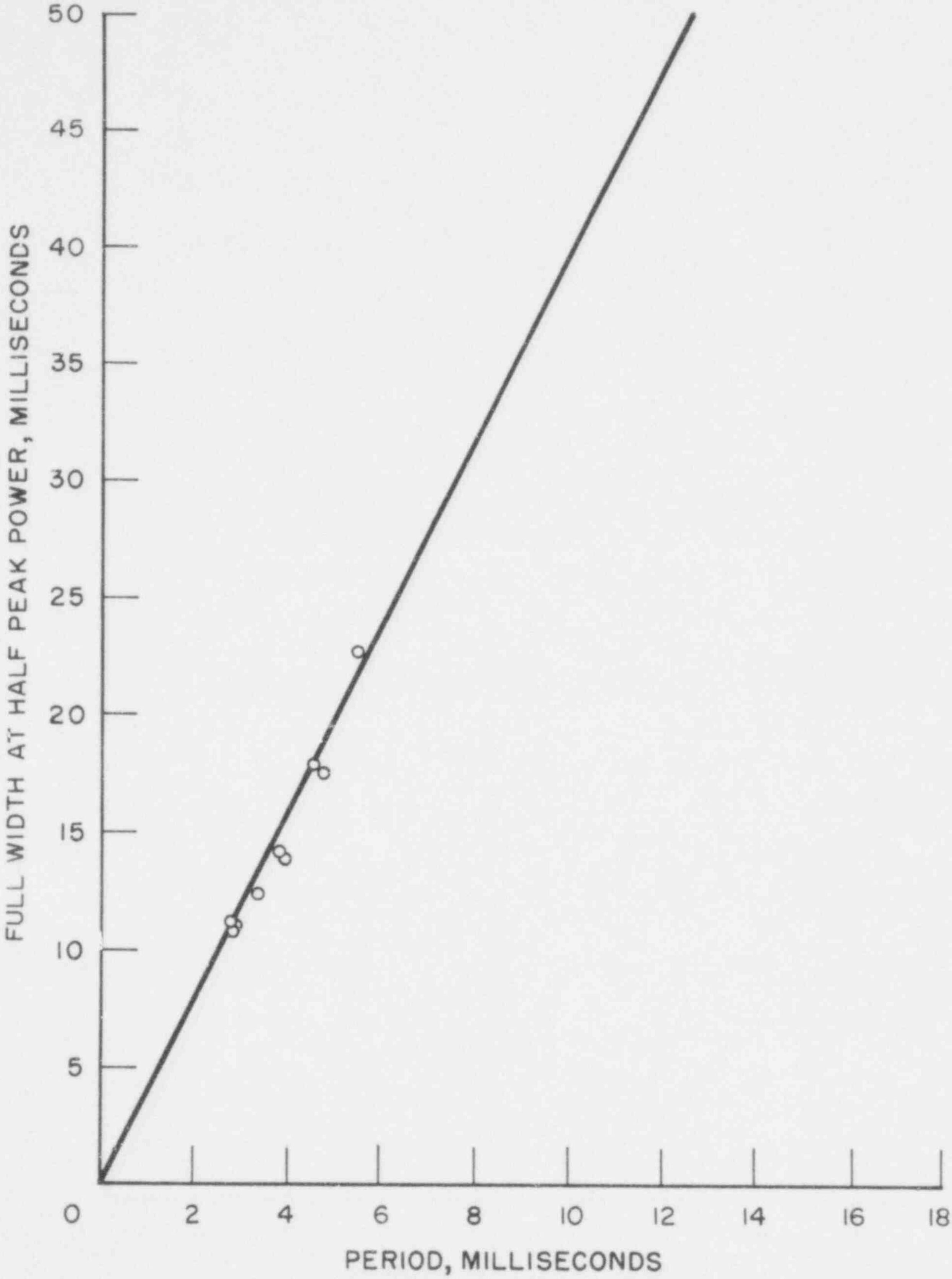

FIGURE 3-23 FULL WIDTH AT HALF PEAK POWER VERSUS PERIOD - PROTOTYPE REACTOR

324 116

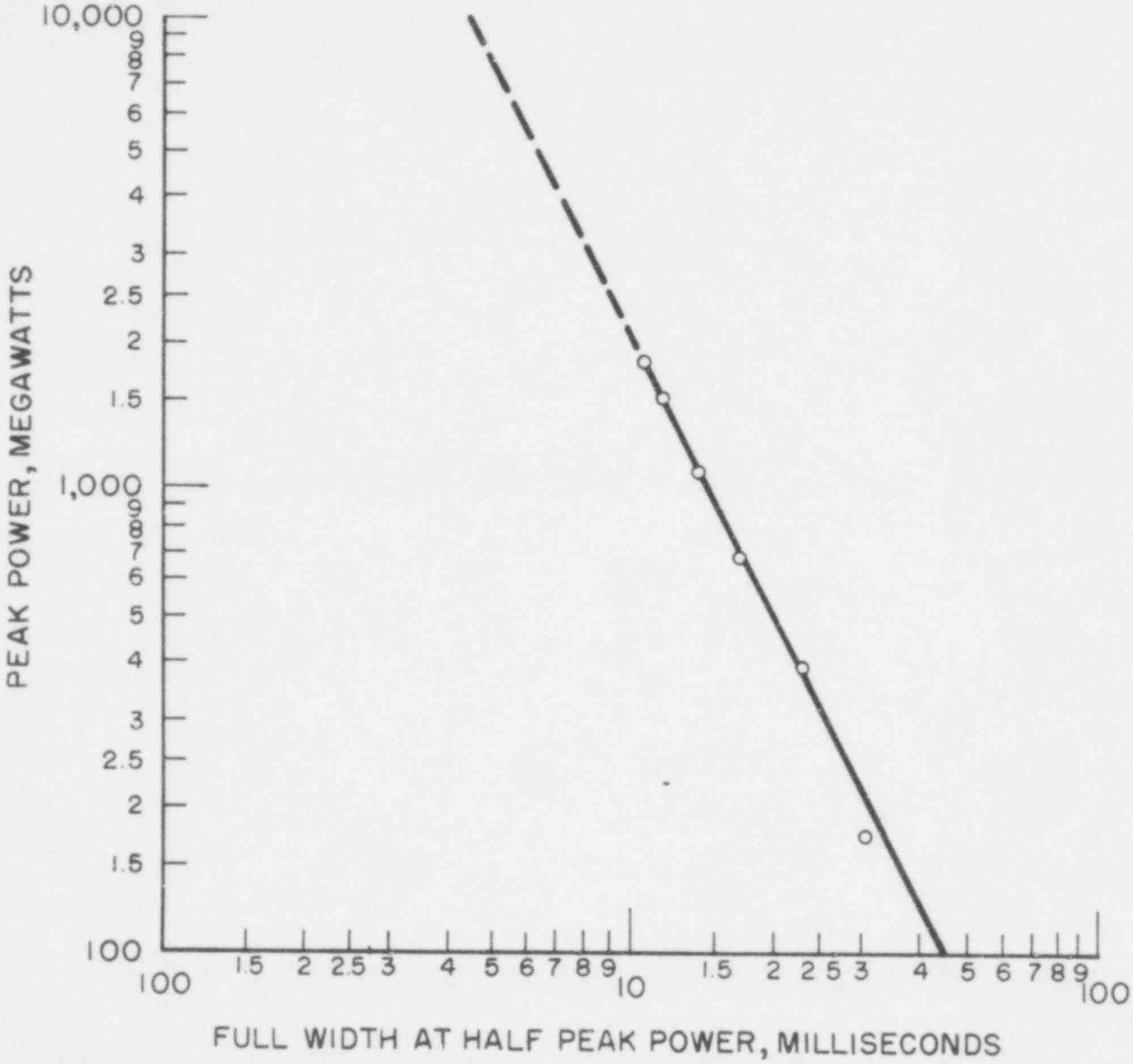

FIGURE 3-24 PEAK POWER VERSUS FULL WIDTH AT HALF PEAK POWER – PROTOTYPE REACTOR

324 117

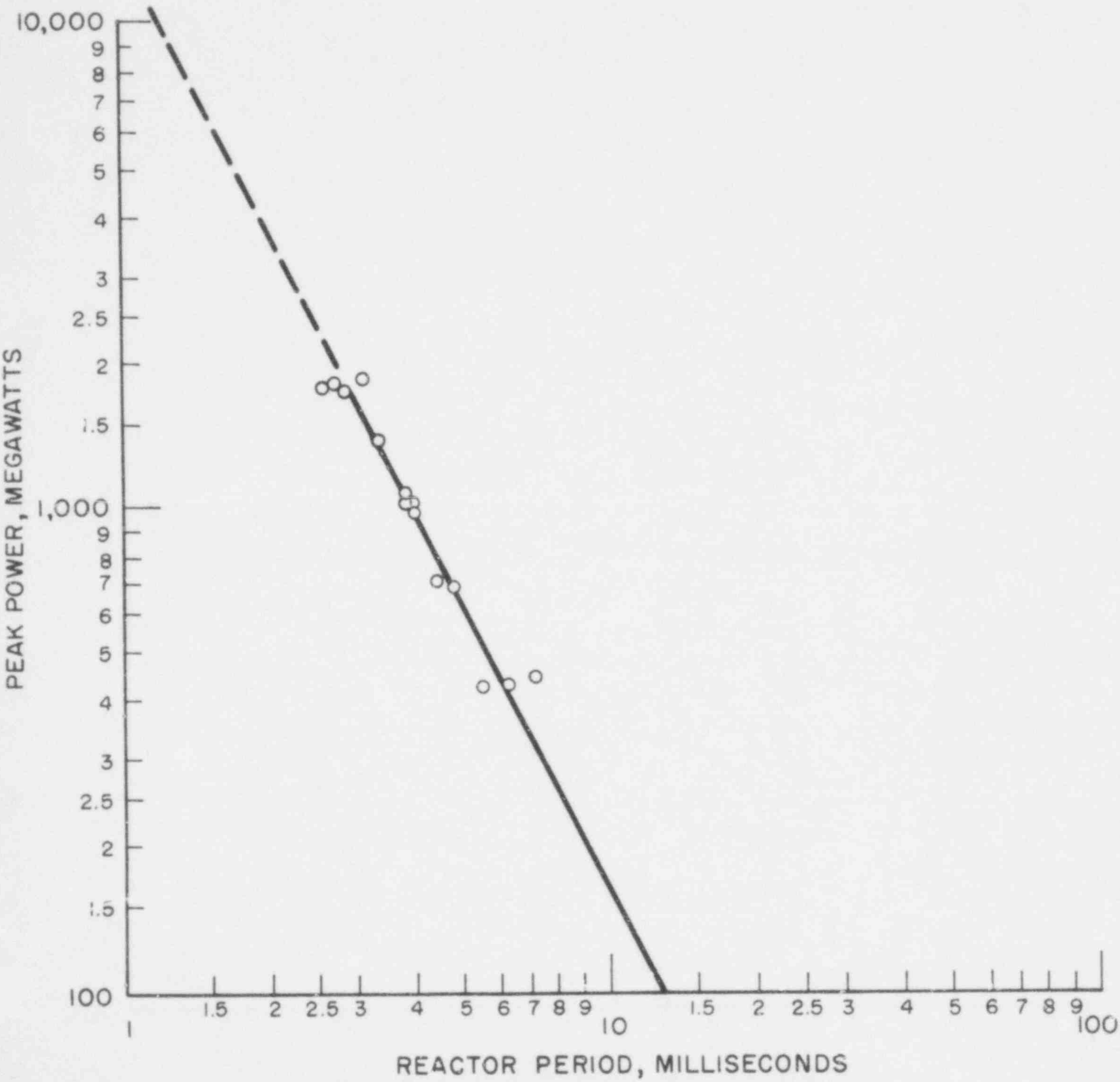

FIGURE 3-25 PEAK POWER VERSUS REACTOR PERIOD–PROTOTYPE REACTOR

on the other hand, are approximately linear functions of reactivity insertions above prompt critical so that their range is more limited.

2. Mixed and FLIP Cores

The NSCR presently operates a mixed core reactor composed of both TRIGA standard and FLIP elements (Figure 3-2). The decision to convert to a mixed core was reinforced by the considerable amount of information available from General Atomic on the TRIGA and FLIP systems. Studies made by the NSC for a variety of cores from all standard fuel to all FLIP fuel indicated that a core with a mixed loading would safely satisfy all operational requirements. FLIP fuel elements are located in a contiguous central region of the core, and future additions of FLIP fuel will be made such that the FLIP region grows outward as the outside core dimensions remain essentially the same. This procedure will be repeated until at an appropriate time a full FLIP core is obtained.

To obtain a code suited for this investigation an extensive survey of available computer codes was undertaken. The criteria used to determine the code best suited for the core studies were accuracy, programming language, machine storage requirements, and run times.

Exterminator-2 which was the code chosen is a two-dimensional, multigroup, diffusion theory code. The only limitations on problem size is actual machine storage space available since the code employs the technique of variable dimensioning. To facilitate the core studies, an optional total thermal flux print out and/or computer plot was added to Exterminator-2. An additional program was written to calculate the average power generation in each fuel cell for the cores under investigation. Basic input data such as cross sections, number densities, and bucklings were obtained from General Atomic since they have been proven by actual usage.

For core calculations, each fuel element and its associated moderator were homogenized over a cell, and likewise each control rod and its associated moderator were homogenized. Homogenized calculations were necessary because the spacing between fuel elements is small and diffusion theory is not valid for the heterogeneous problem. Fuel cell dimensions for NSCR cores are 3.854 cm. x 4.050 cm.

To test the accuracy of the input data and to insure that the results obtained using Exterminator-2 were meaningful, comparisons were made between calculated and experimental results. Three cases in particular can be used to show the validity of the code. The first test of the Exterminator-2 program was to repeat some calculations on a FLIP core that have been performed by General Atomic. Both programs gave essentially identical results. The second test was a comparison of the calculated and measured values of a relative thermal flux distribution in an operational NSCR TRIGA core. The results are shown in Figure 3-26. Good agreement is noted when it is recalled that the diffusion theory calculation only gives flux values that are averaged over a cell. The

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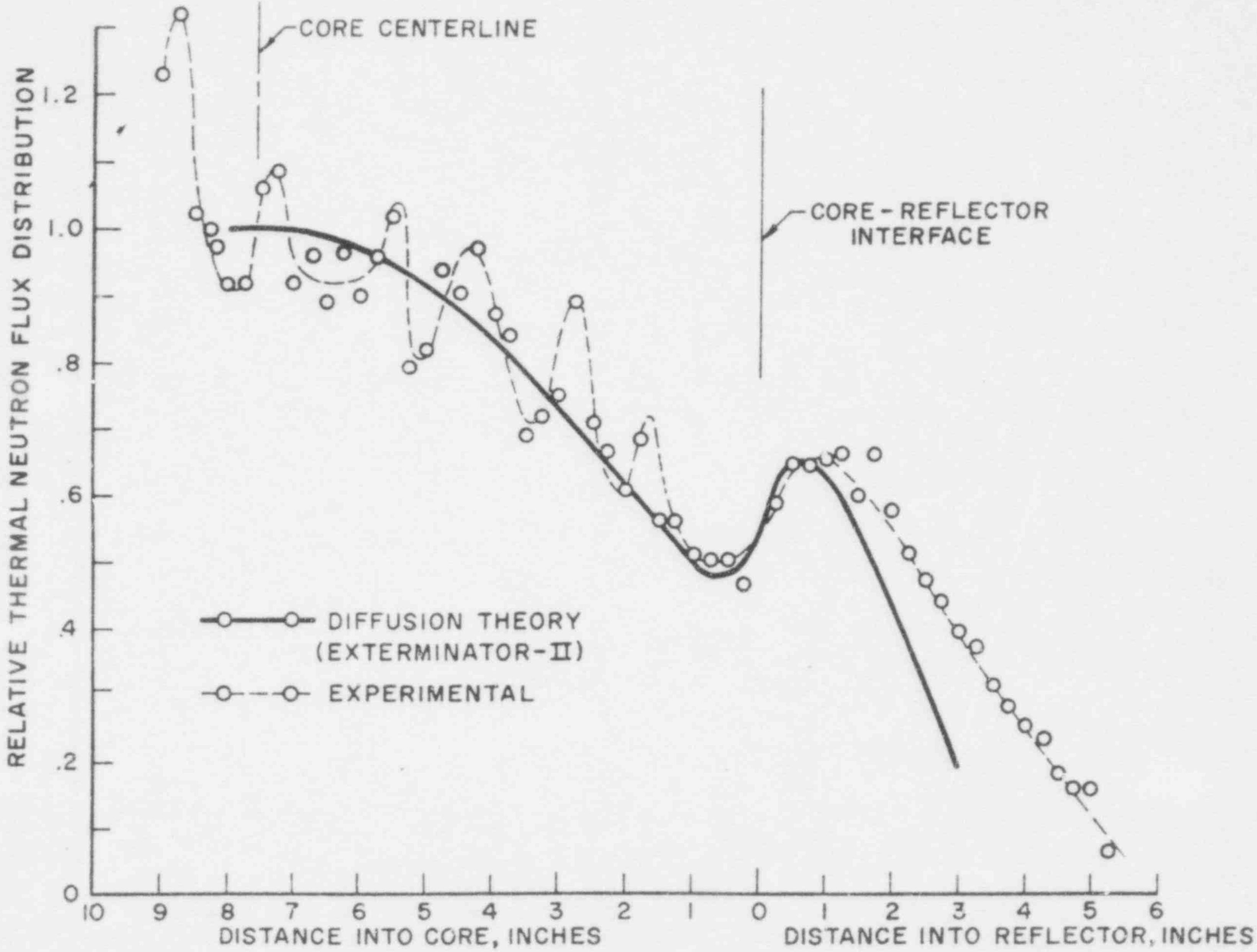

FIGURE 3-26 EXPERIMENT VS CALCULATED FLUX USING EXTERMINATOR-2

324 120

third test was a comparison of measured and calculated values of k_{eff} for several TRIGA cores. These results are shown in Table III.

TABLE III

COMPARISON OF MEASURED AND CALCULATED VALUES OF k_{eff}

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

The cores used for the comparisons were both 5 x 5 array (100 cell) cores. The NSCR IA(1) cores had 95 standard TRIGA fuel elements, 3 shim-safety control rods, 1 transient rod and 1 regulating rod. The Puerto Rico all-FLIP core considered had 94 FLIP fuel elements, 3 shim safety rods, 1 transient rod, 1 regulating rod, and 1 air void experiment hole. In both cases, the calculated values were slightly higher than the measured values but agreement was good and for safety purposes the results were conservative.

The results of the above three tests established confidence in the Exterminator-2 code and the G.A. cross sections. It is believed that any safety calculations obtained from this program would yield realistic and reliable results.

As mentioned earlier, the power generation in a given core was computed by Exterminator-2 for all fuel cells. A typical flux plot and generated cell power obtained using Exterminator-2 code is shown in Figure 3-27. However, the peak adiabatic fuel temperature during a pulse was computed on a selective basis following examination of calculated cell power generation values. For each core of interest, this calculation was carried out for those cells which exhibited the highest average power generation. The distribution of power density produced in the cell was obtained by weighting the average power with the microscopic flux distribution. This flux distribution was obtained as the product of the average flux computed by Exterminator-2 and the intra-cell flux computed by General Atomic. This method was shown to be in good agreement with two dimensional cell calculations of higher order which account separately for cell effects. A knowledge of the temperature as a function of the volumetric heat content allowed calculation of the maximum fuel temperature for a pulse of given energy.

Additional comparisons between calculated and experimental results were made for mixed TRIGA cores in operation by the NSCR. The first comparison is for the experimental and calculated values of pulse temperatures at a single thermocouple location. Pulse temperatures were observed for a \$2.00 pulse insertion following rotation of the instrumented element in 15^{o} increments. These results are compared to calculated values in Figure 3-28. The relative values for the pulse

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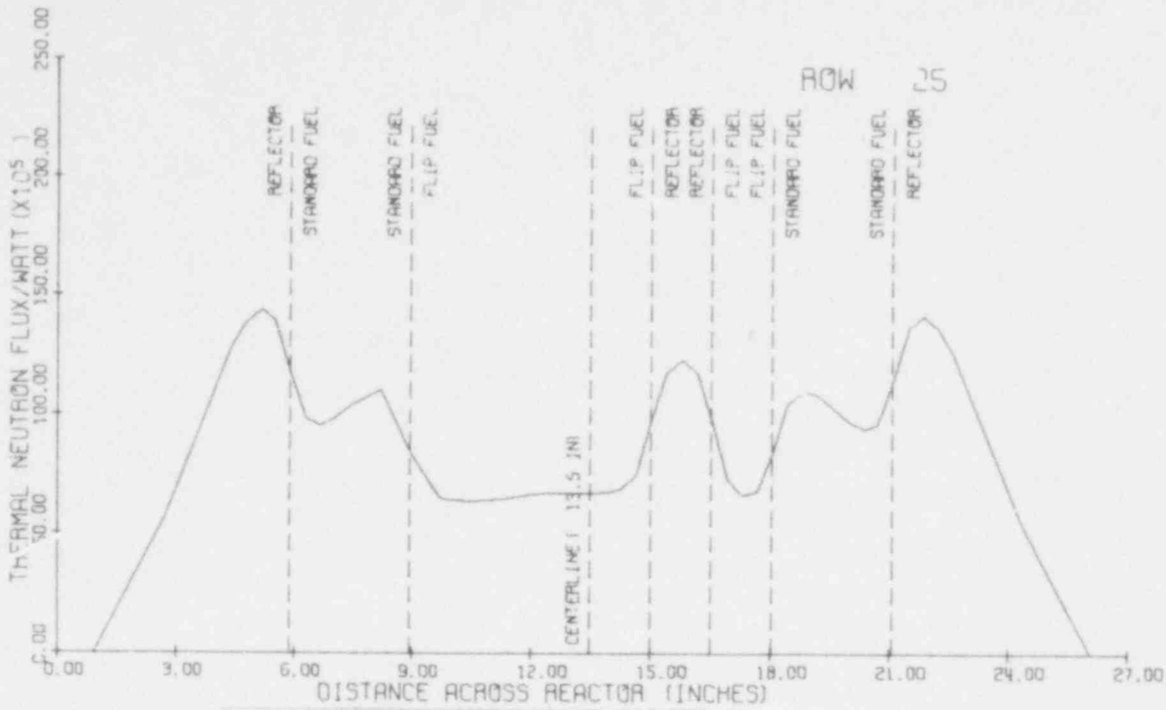
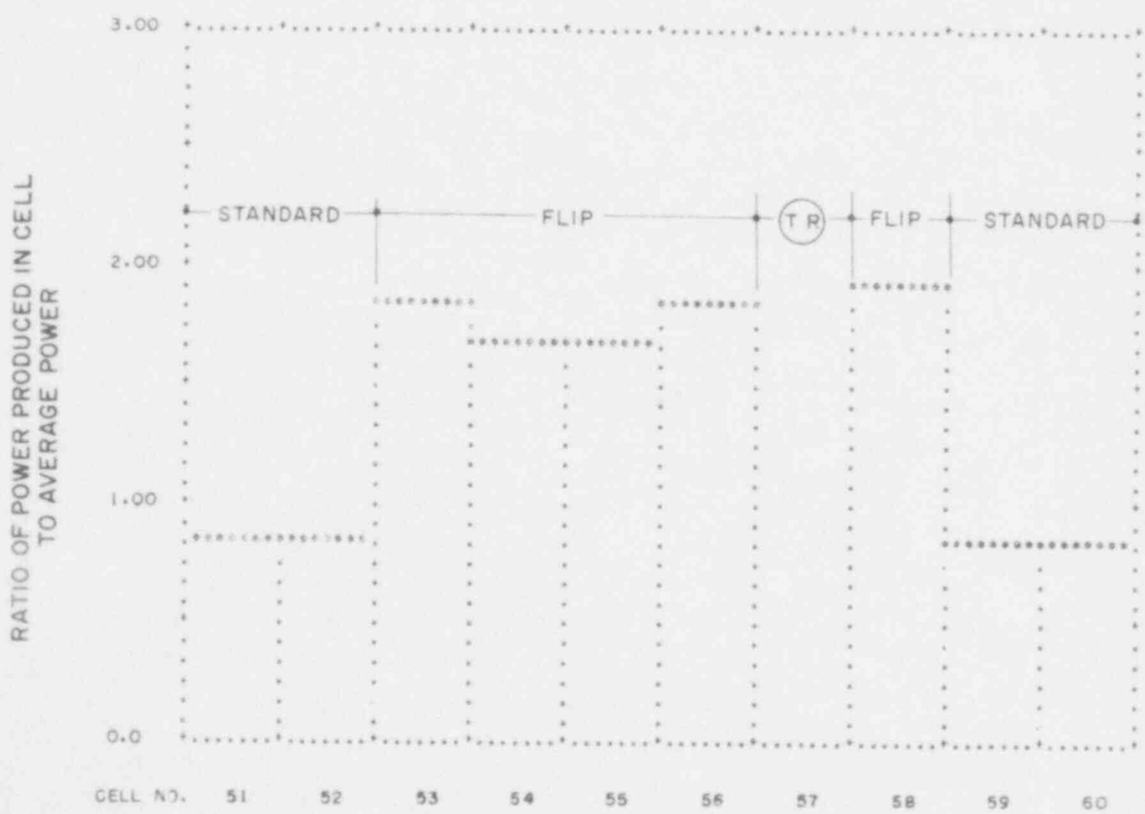


FLUX PLOT



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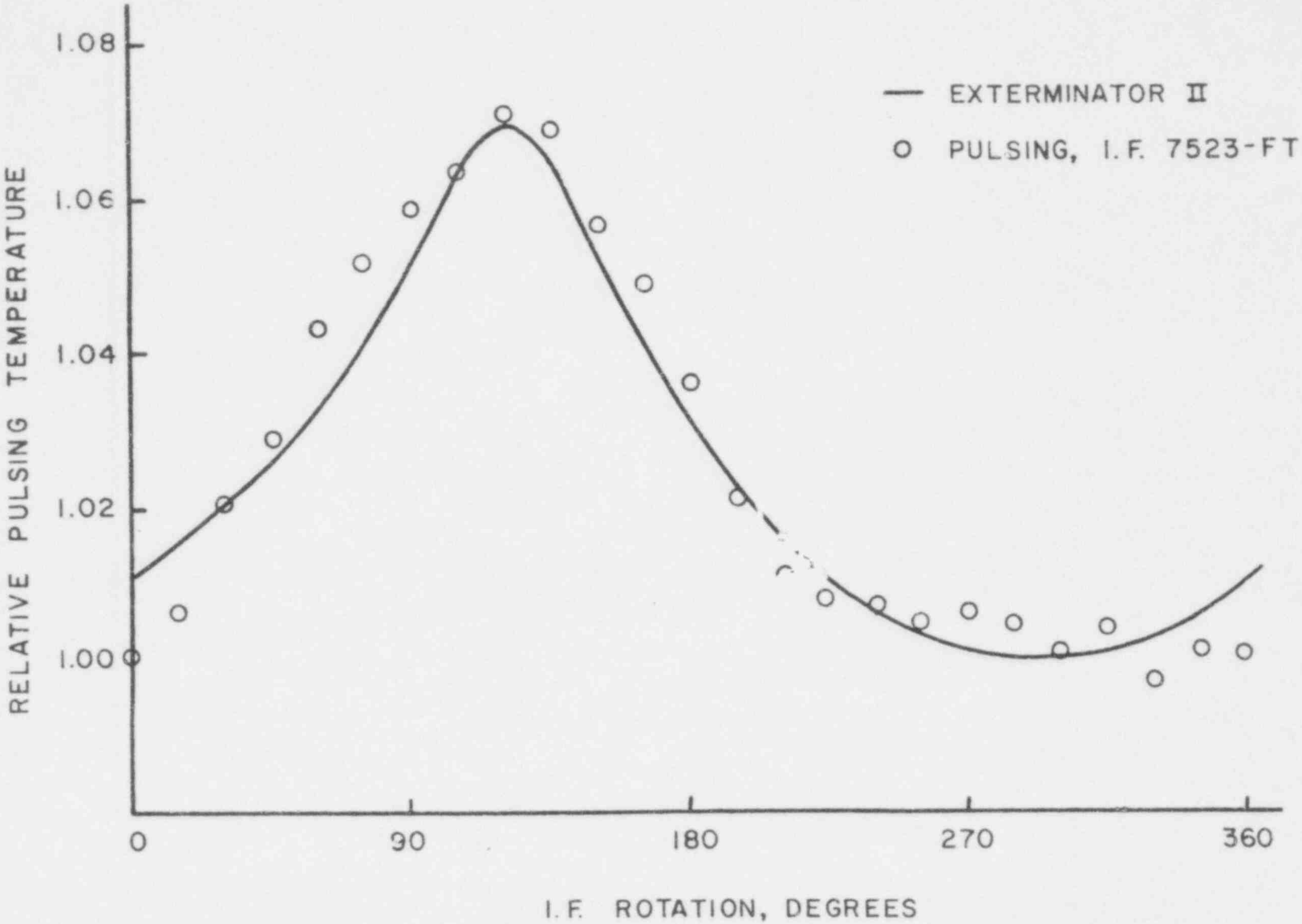
GENERATED POWER

FIGURE 3-27 23 ELEMENT FLIP THERMAL NEUTRON FLUX AND GENERATED POWER PLOTS FOR FUEL ROW CONTAINING MAXIMUM GENERATED POWER



324 123

FIGURE 3-28 EXPERIMENTAL vs. CALCULATED PULSE TEMPERATURES FOR ROTATED INSTRUMENTED ELEMENT

temperatures determined by the code and the experimental values agree very well.

The second comparison is the experimental and calculated values for the ratio of pulsing temperatures for two different instrumented element locations. Experimental values were obtained for observed pulse temperatures for \$2.00 insertions for an instrumented element at two different core locations. Calculated values of the thermocouple temperature rise at both locations were obtained for comparison. The experiment was repeated four times using two instrumented FLIP fuel elements and the same thermocouple for each set of values obtained. For a 35 FLIP element mixed core, the observed and calculated ratios agree to within 4%. The design core for the first NSCR loading was 35 FLIP and 63 standard TRIGA elements. Criticality of this core was achieved on July 1, 1973 and the core was initially pulsed July 4, 1973. The core loading designated as III-A is shown in Figure 3-29. The following operating characteristics were observed for core III-A.

TABLE IV

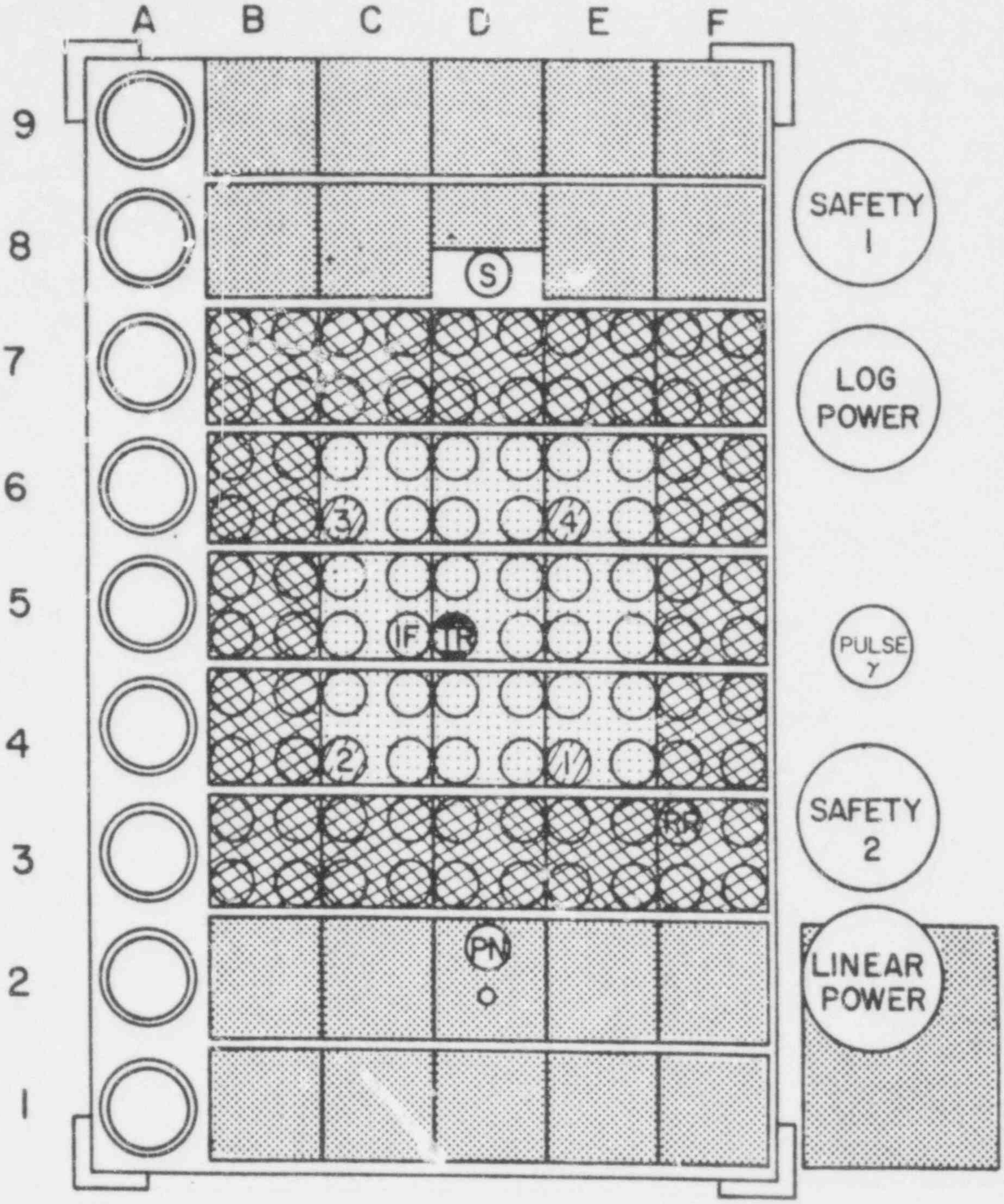
Steady State Power Level: 1 Mw
Critical Mass: 5,590 grams ^{235}U
Core Mass: 6,226 grams ^{235}U
Maximum Excess Reactivity: \$6.09
Power Coefficient (1 Mw): \$2.50
Maximum Pulse Energy (\$2.00 insertion): 15.7 Mw-sec
Total Control Rod Worth: \$15.57
Maximum Pulse Reactivity Insertion: \$2.00

Reactor core studies using Exterminator-2 predicted a core excess of \$6.20 for Core III-A as compared to the measured value of \$6.09. Pulsing characteristics observed for III-A are shown in Figure 3-30.

As mentioned earlier the design considerations for pulsing FLIP fuel as compared to standard TRIGA fuel are different due to a temperature dependent negative temperature coefficient for FLIP fuel.

For a TRIGA-FLIP fuel element, the uranium loading is about 3.5 times that of a standard TRIGA element and this causes the neutron mean free path in the FLIP element to be much shorter. For this reason, the escape probability for neutrons in the fuel is not greatly enhanced as the fuel-moderator material is heated. In the TRIGA-FLIP fuel the temperature hardened spectrum is used to decrease reactivity through its interaction with a low energy resonance material. Thus erbium, with its double resonance at $\sim$ 0.5 eV, is used in the TRIGA-FLIP 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 Er-167 resonance as the fuel temperature increases. In a standard TRIGA core, the temperature coefficient is prompt because the fuel is intimately mixed with a large portion of the

324 124



TR Transient Rod With Air Follower

1 Shim Safety Rod With Fueled Follower

RR Regulating Rod With H_2O Follower

IF Instrumented Fuel

S Sb-Be Neutron Source

Standard Fuel

Flip Fuel

Graphite Reflector

PN Pneumatic Tube

k_{ex} = \$6.09 (Meas.)

k_{ex} = \$6.20 (Calc.)

FIGURE 3-29 NSCR CORE III-A

324 125

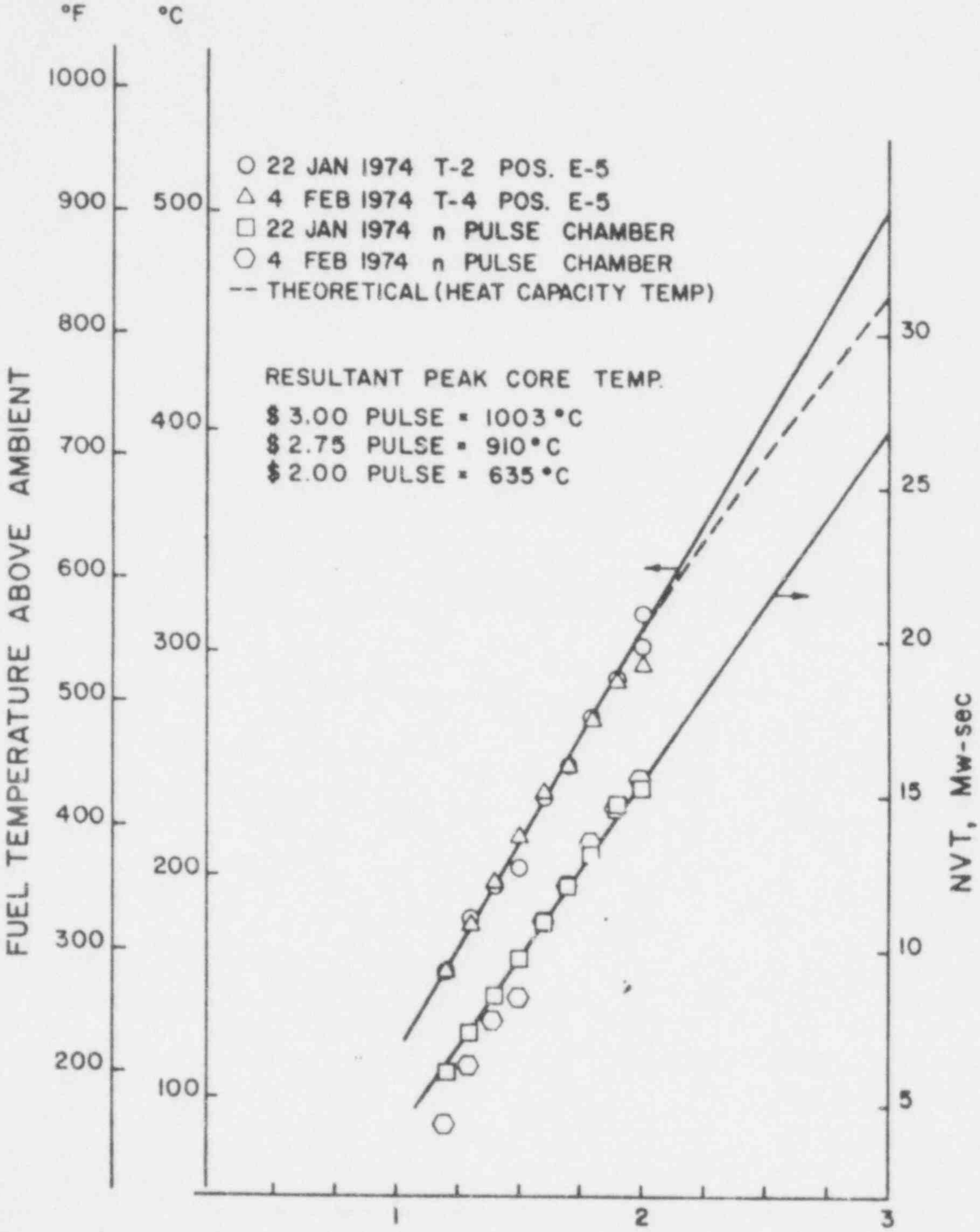


FIG. 3-30 CORE III-A PULSE DATA
IF POSITION E5(SE)

moderator, and thus, fuel and solid moderator temperatures rise simultaneously producing the temperature dependent spectrum shift.

In a TRIGA-FLIP core the results of cell structure on the temperature coefficient are small. Almost the entire coefficient comes from temperature dependent changes in ηf within the core and $\sim$ 80% of this effect is independent of the cell structure. The calculated temperature coefficients are shown in Figure (3-31) for standard, mixed and FLIP cores. The temperature dependent character of the temperature coefficient of a TRIGA-FLIP core is advantageous in that a minimum reactivity loss is incurred in reaching normal operating temperatures, but any sizeable increases in the average core temperature result in a sizeably increased prompt negative temperature coefficient to act as a shutdown mechanism. The burnup calculations indicate that after 3000 MW days of operation, the U-235 concentration averaged over the core is $\sim$ 67% and the Er-167 concentration is $\sim$ 33% of the beginning-of-life values. The end-of-life coefficient for a FLIP core is less temperature dependent than the beginning-of-life coefficient because of the sizeable loss of Er-167 and the resulting increased transparency of the $\sim$ 0.5 eV resonance region to thermal neutrons.

The effects of the temperature coefficient for standard and FLIP cores upon the pulse shape is demonstrated in Figure 3-32. Note that the FLIP pulse peak power is considerably higher than in the standard fuel, thus, higher fuel temperatures for the same reactivity insertions. The time span for full width at half maximum (FWHM) for a FLIP pulse is considerably less than that for the standard core pulse yet the total pulse energy is approximately the same.

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.

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

324 127

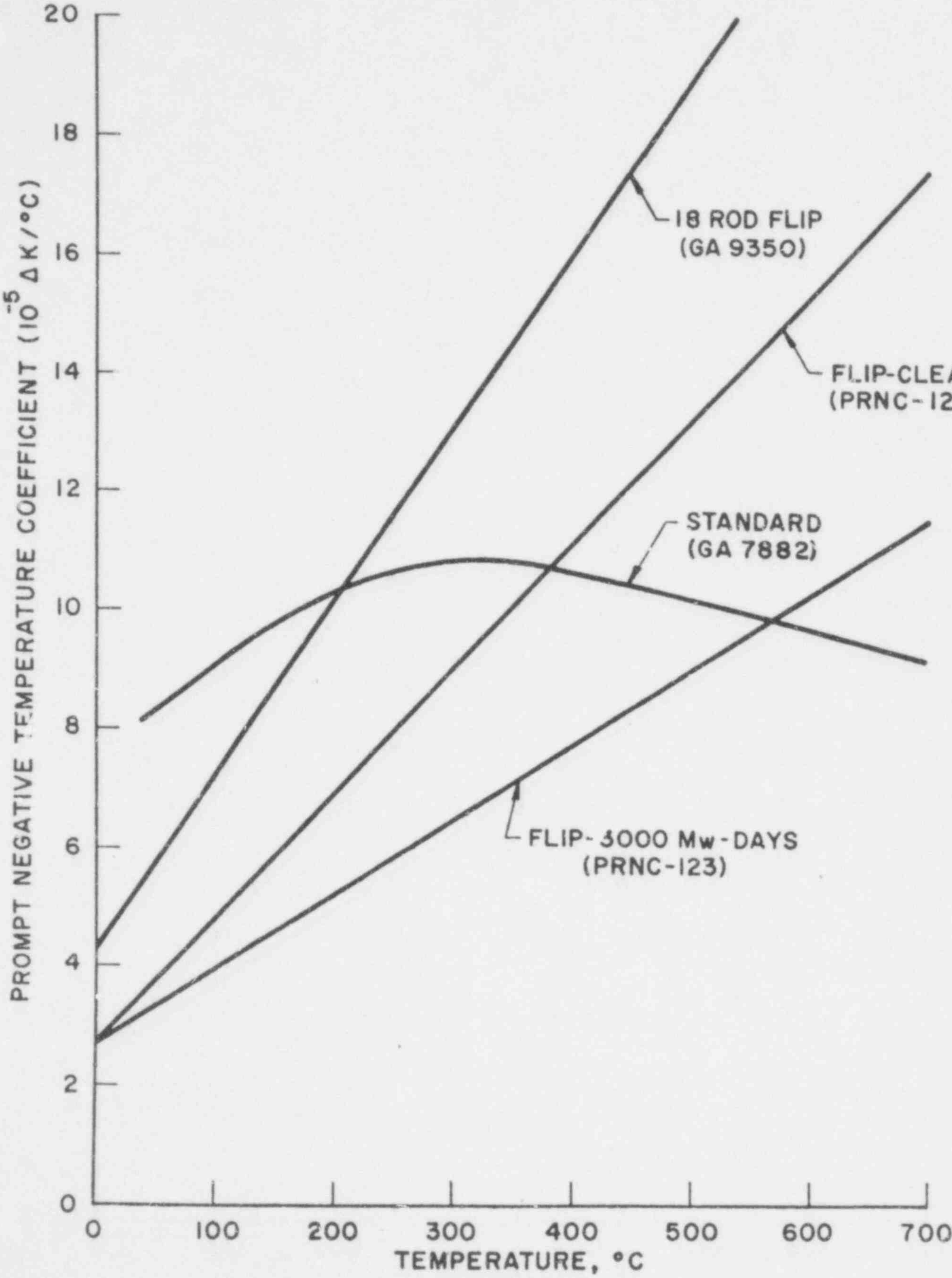


FIG. 3-31 TEMPERATURE COEFFICIENTS OF TRIGA FUELS

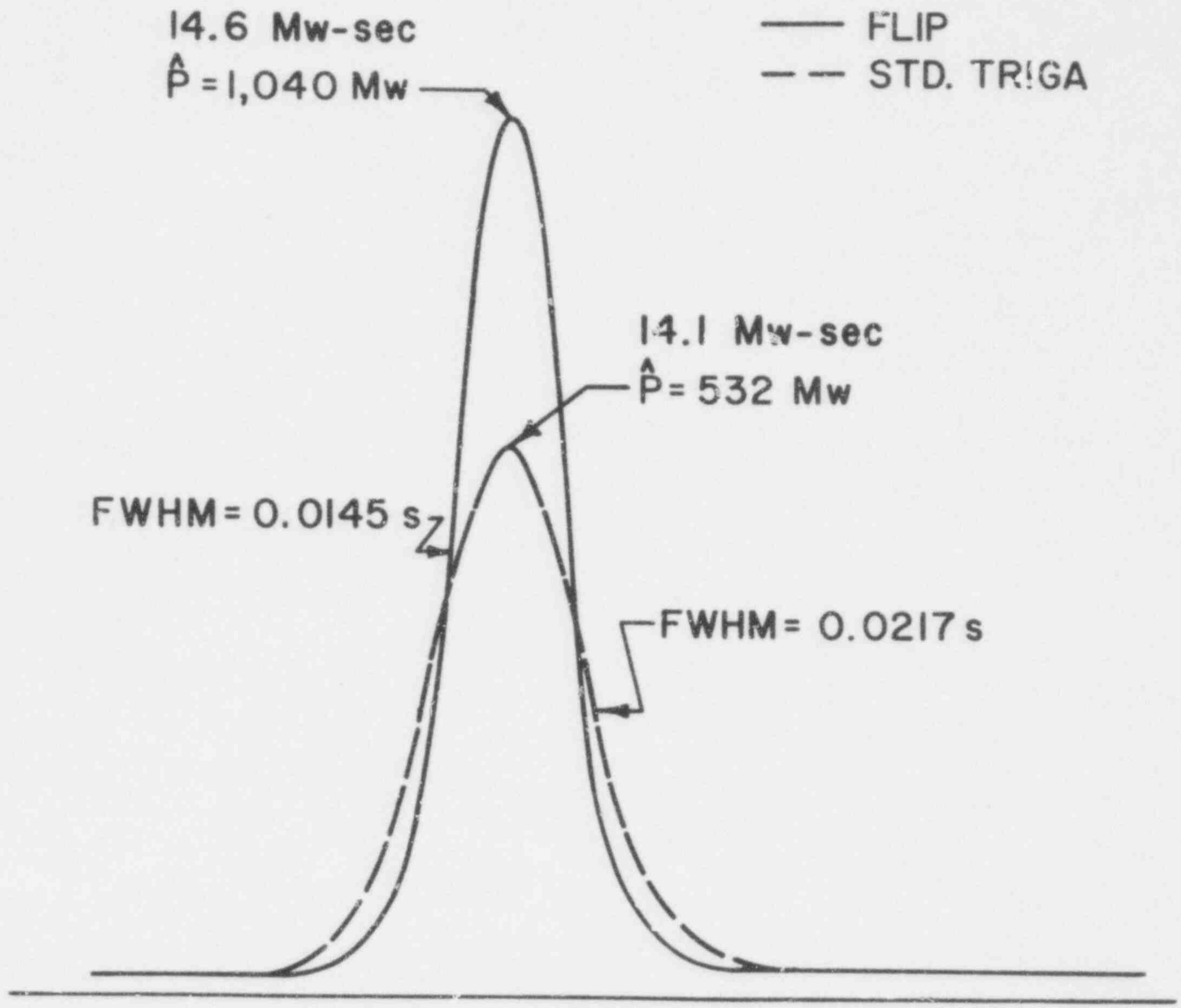


FIG. 3-32 COMPARISON OF FLIP TO STD. TRIGA PULSE FOR SIMILAR REACTIVITY INSERTIONS

324 129

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 3-33. 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.

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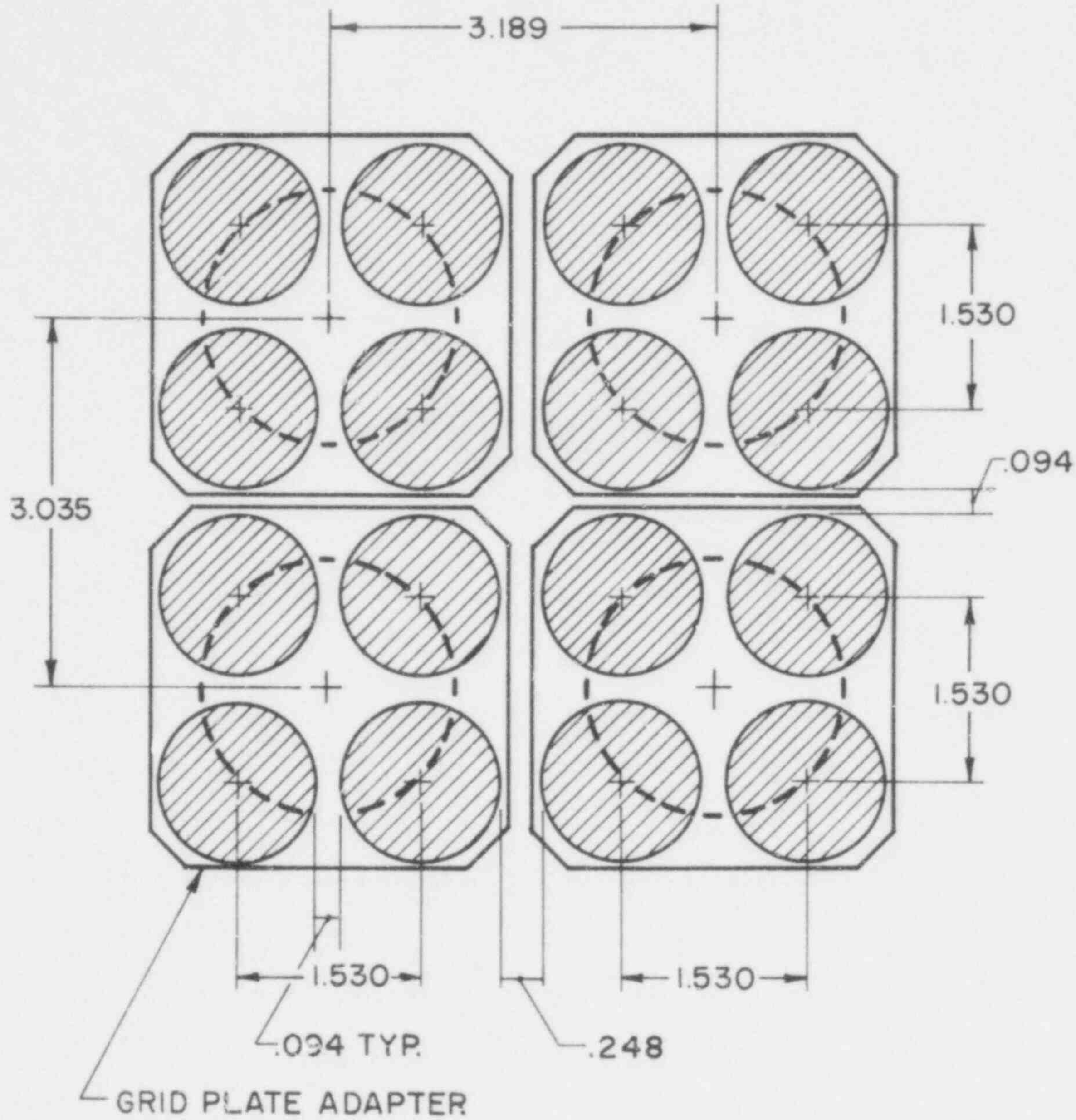

FIGURE 3-33 NOMINAL FUEL ROD SPACING IN THE NSCR CORE

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.

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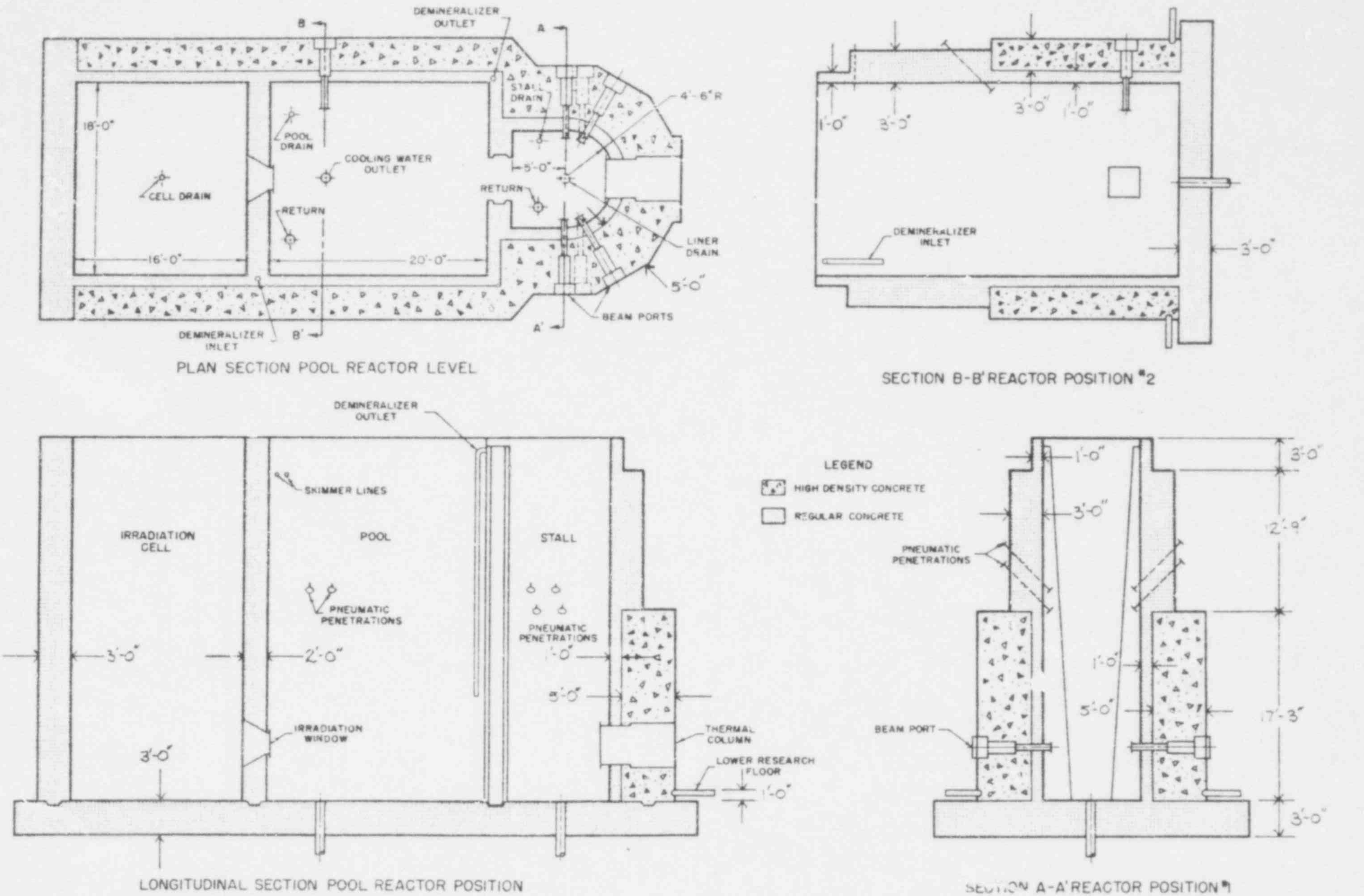

FIGURE 4-1 NSC REACTOR POOL SECTIONS AND PENETRATIONS

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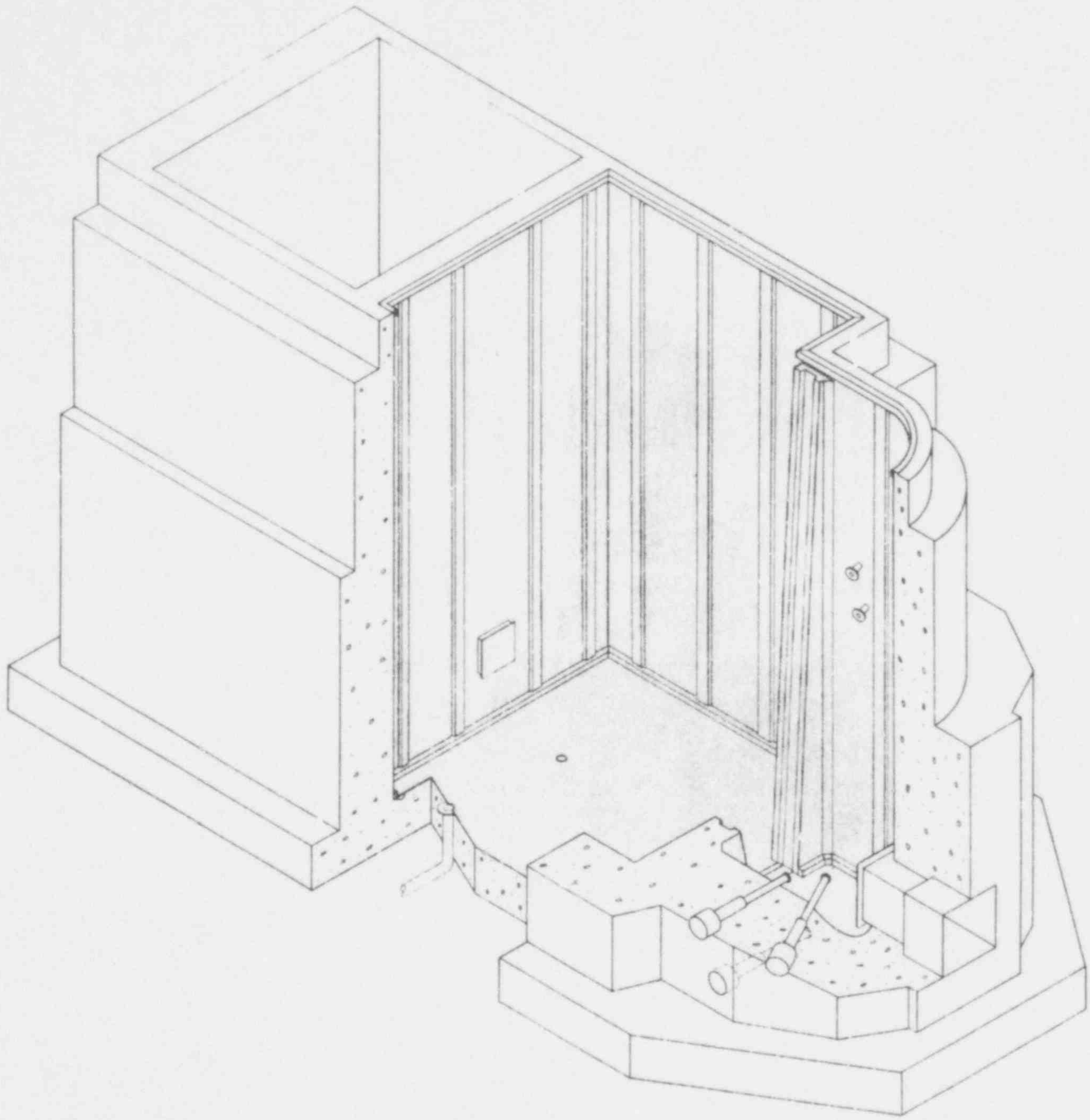

FIGURE 4-2 REACTOR POOL LINER

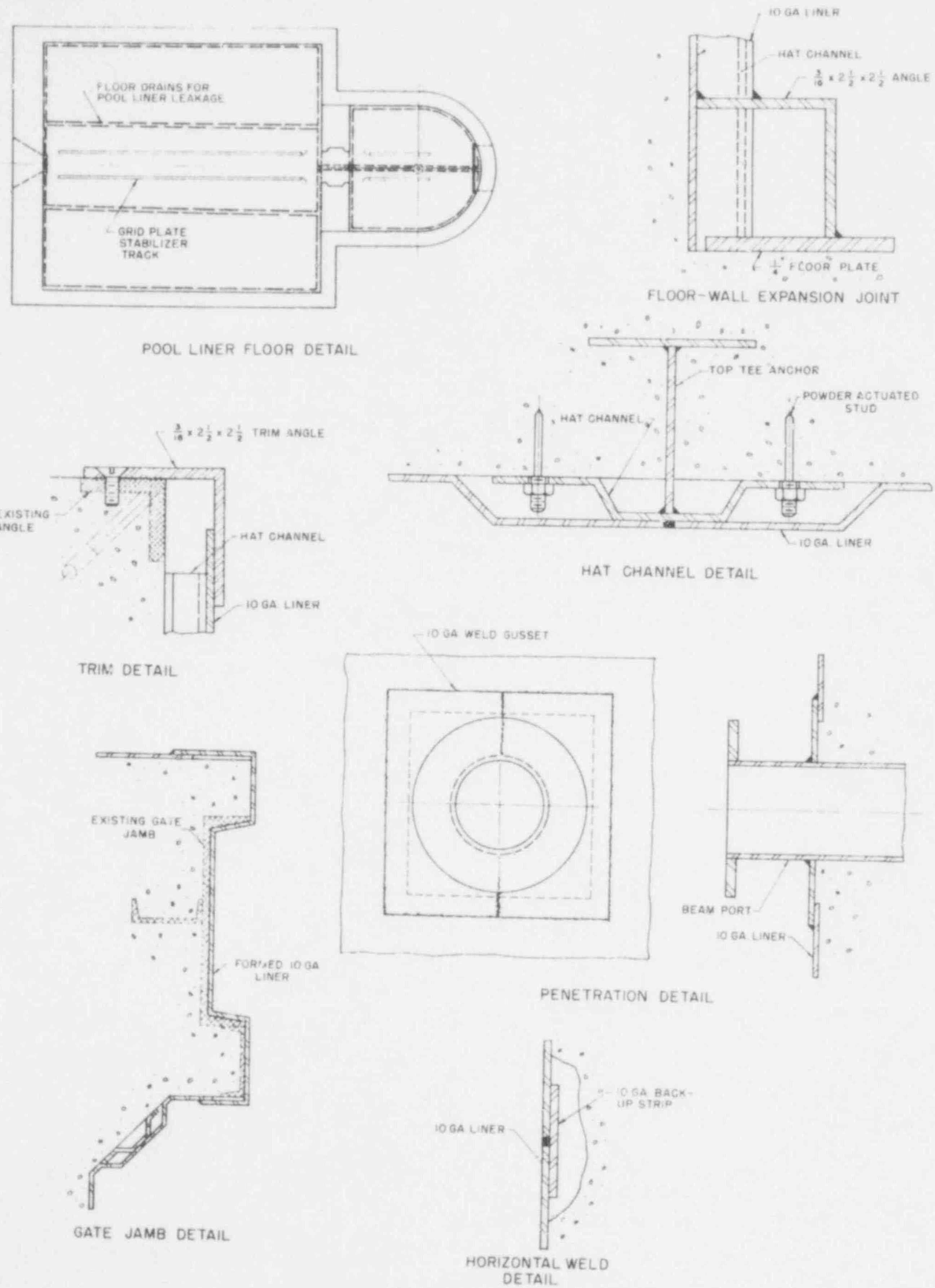

FIGURE 4-3 REACTOR POOL LINER DETAILS

324 135

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 in Figure 4-1.

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/in^2 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.

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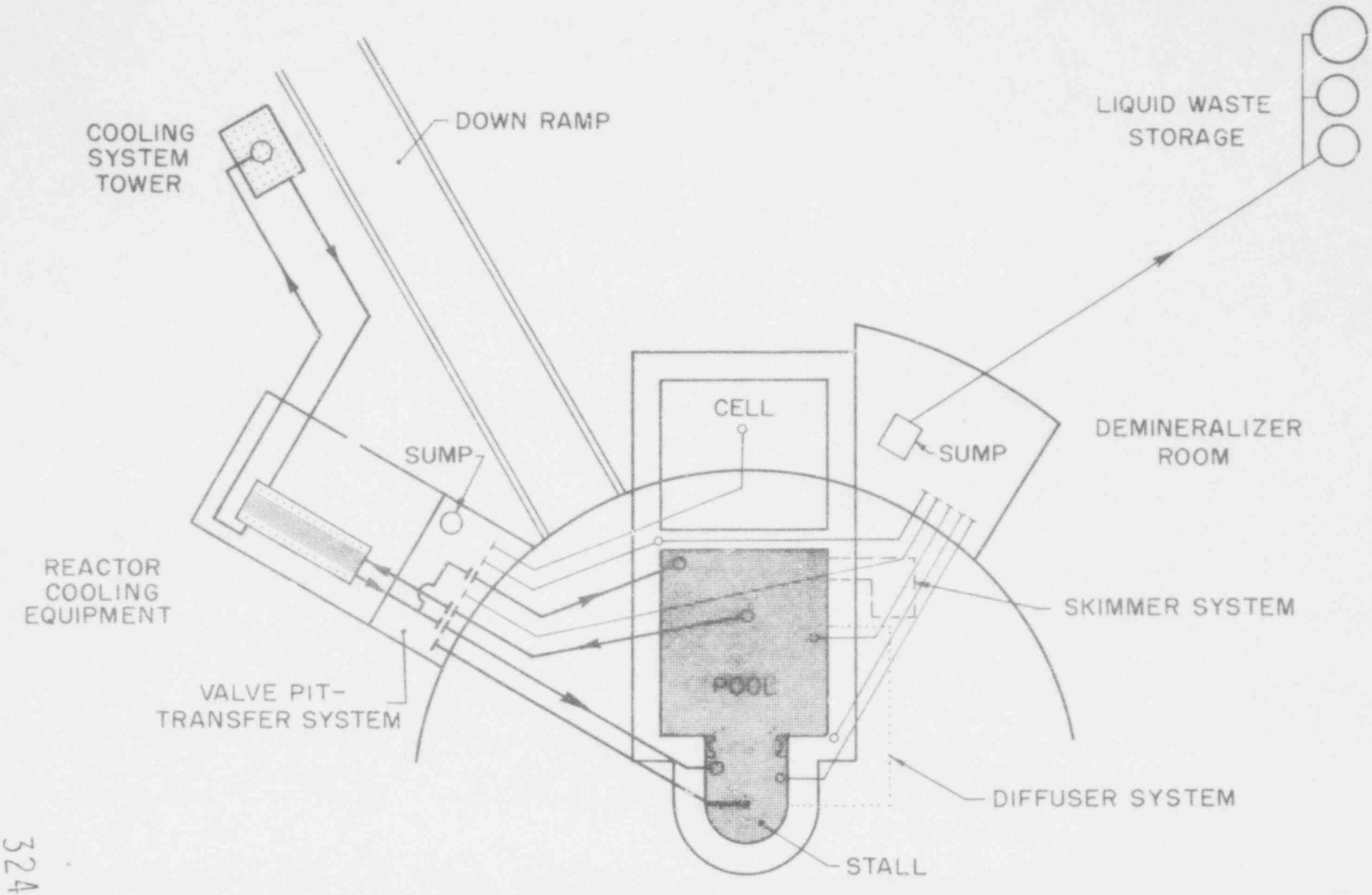

FIGURE 4-4 NSCR POOL WATER SYSTEMS

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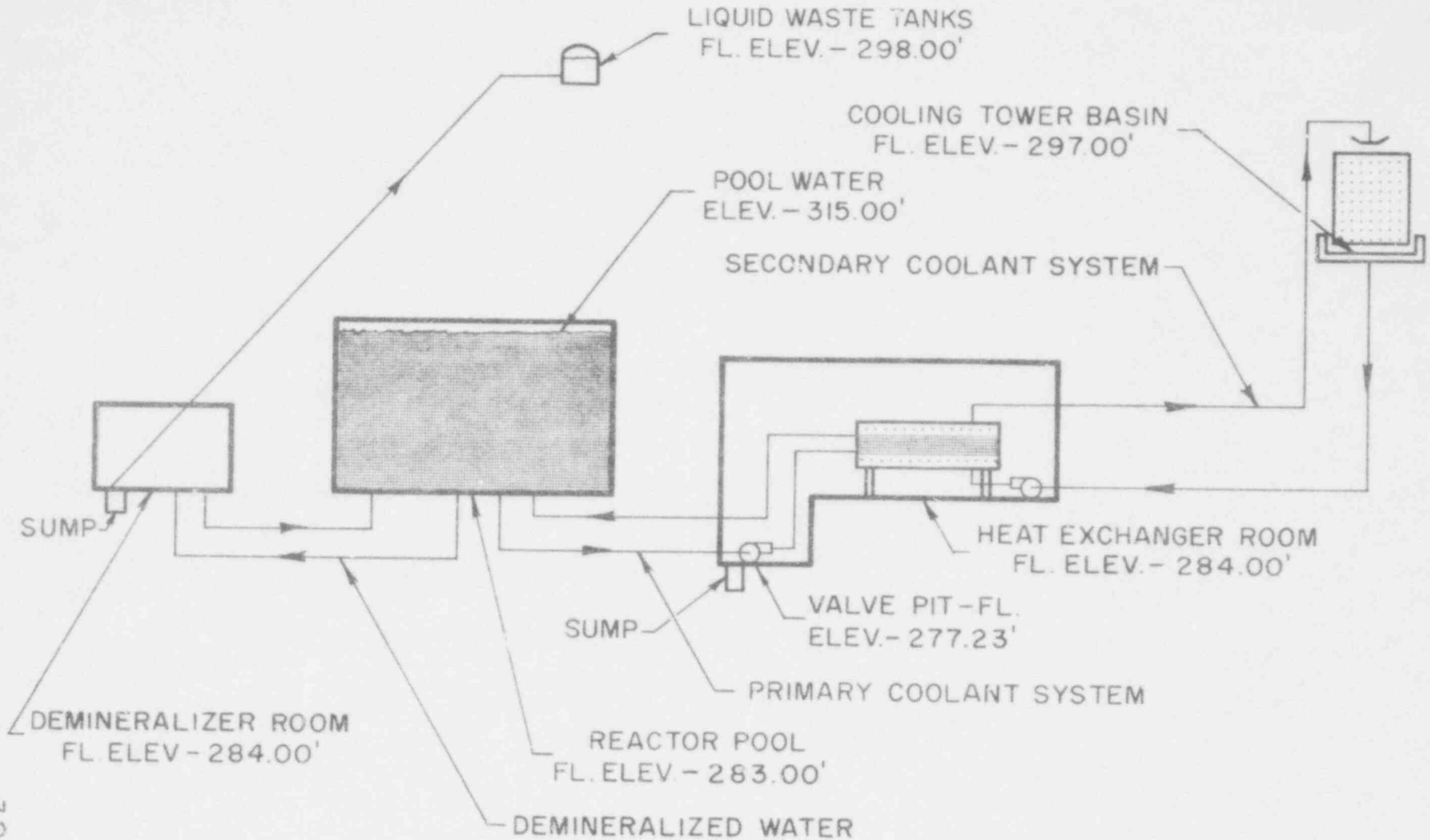

FIGURE 4-5 NSCR POOL WATER ELEVATION

324 138

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 $70^{\circ}F$ to $80^{\circ}F$. 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 $83^{\circ}F$ 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/in^2 for the primary side and 17.5 lb/in^2 in the secondary. Components for the primary cooling loop are located in the cooling equipment room on the lower research level.

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

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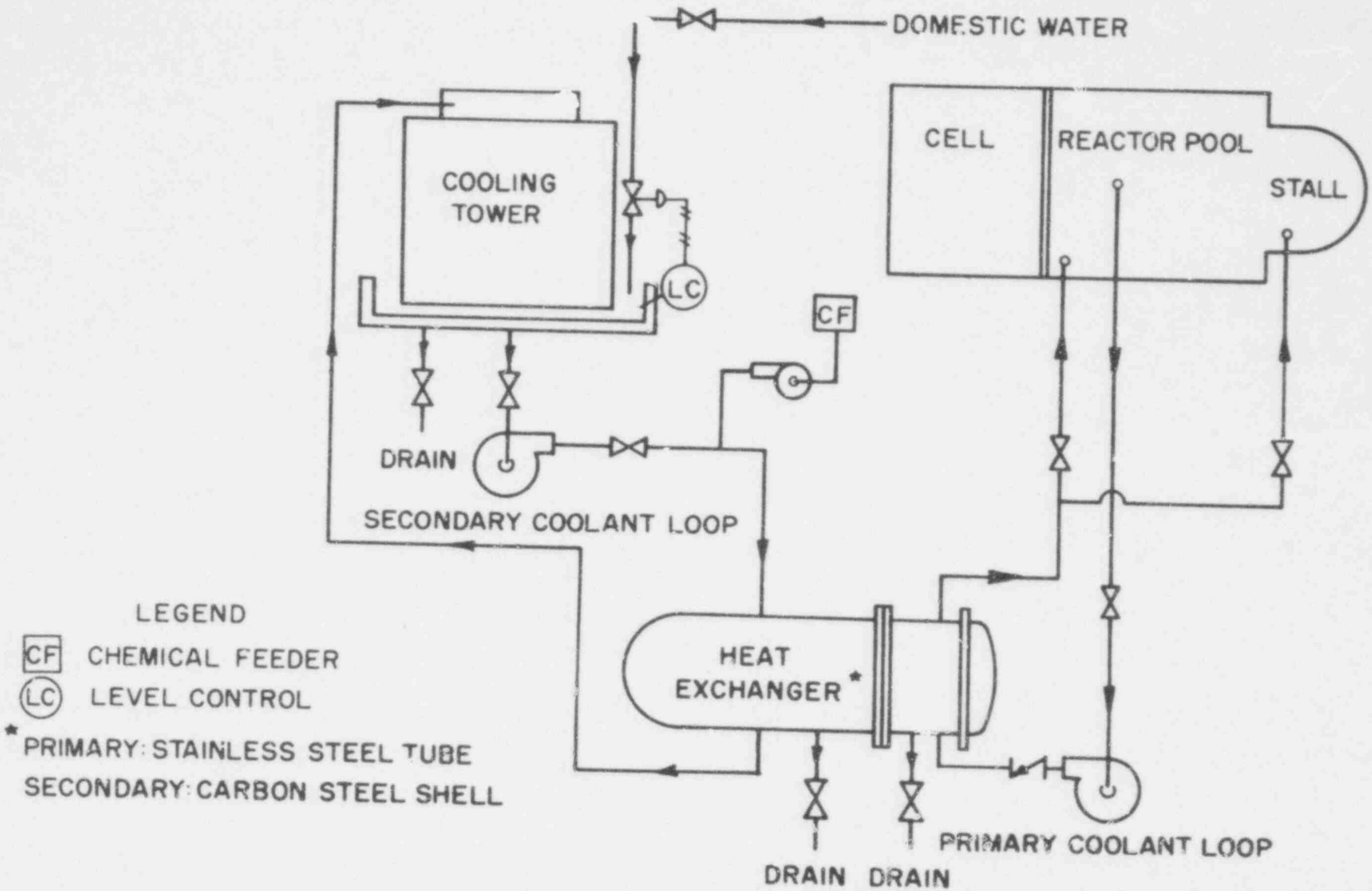

FIGURE 4-6 REACTOR POOL COOLING SYSTEM

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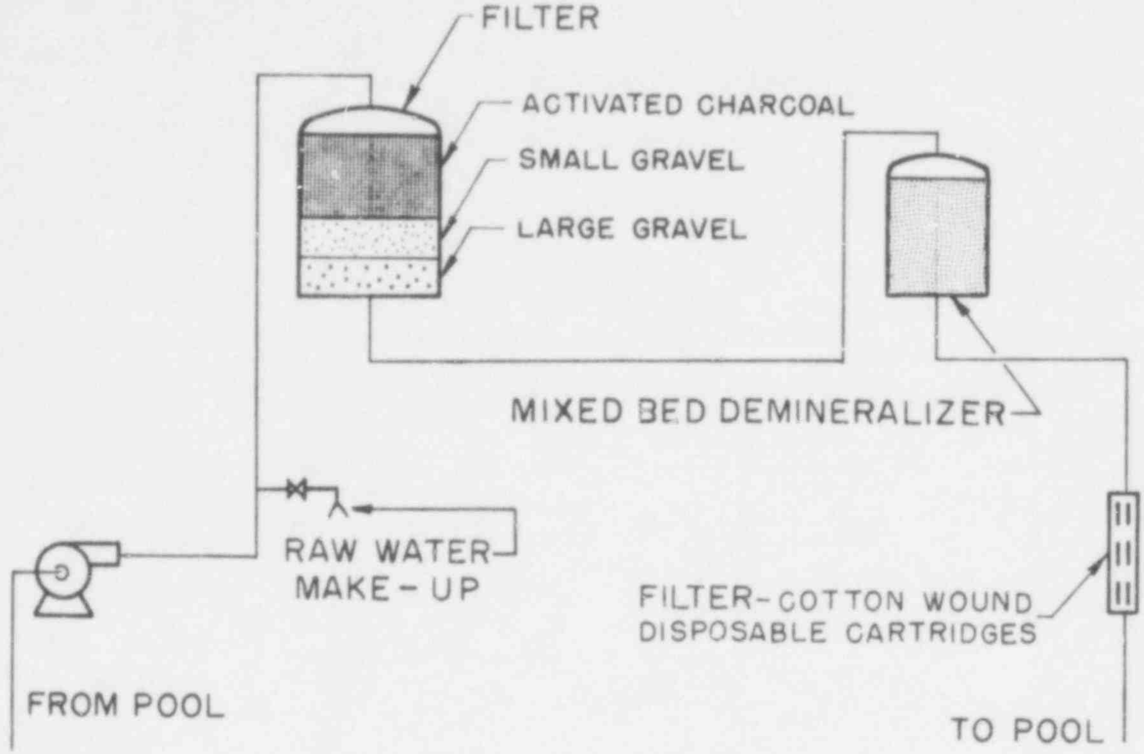
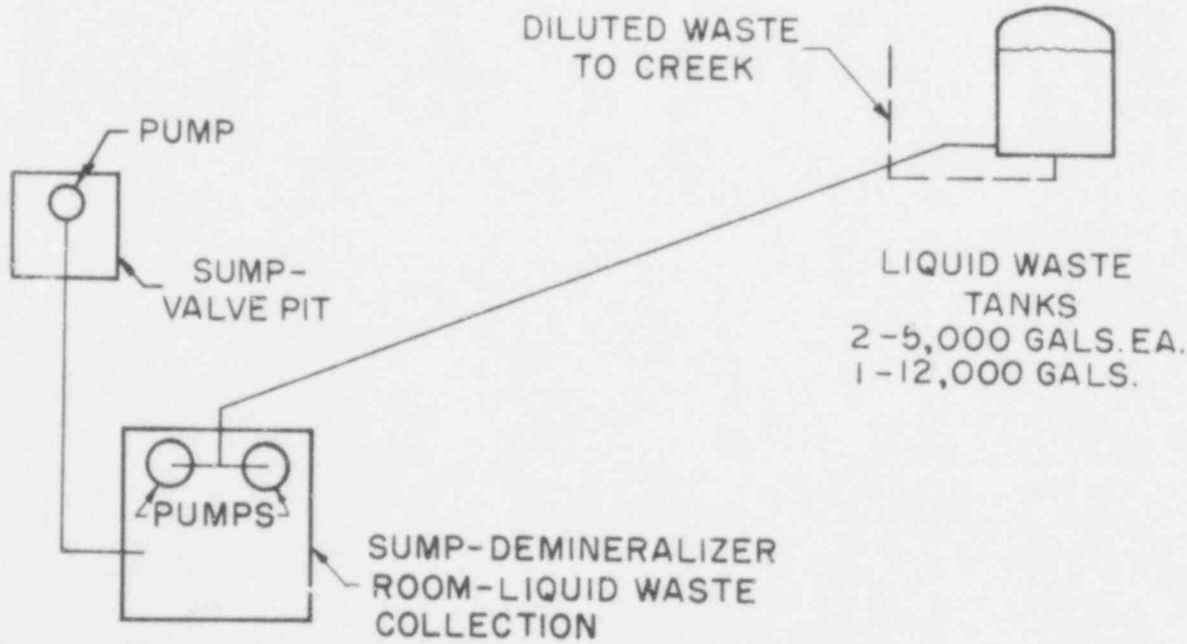


WATER PURIFICATION SYSTEM



LIQUID WASTE DISPOSAL SYSTEM

FIGURE 4-7 WATER PURIFICATION AND WASTE DISPOSAL SYSTEMS

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 make-up 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 cross-over 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.

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.

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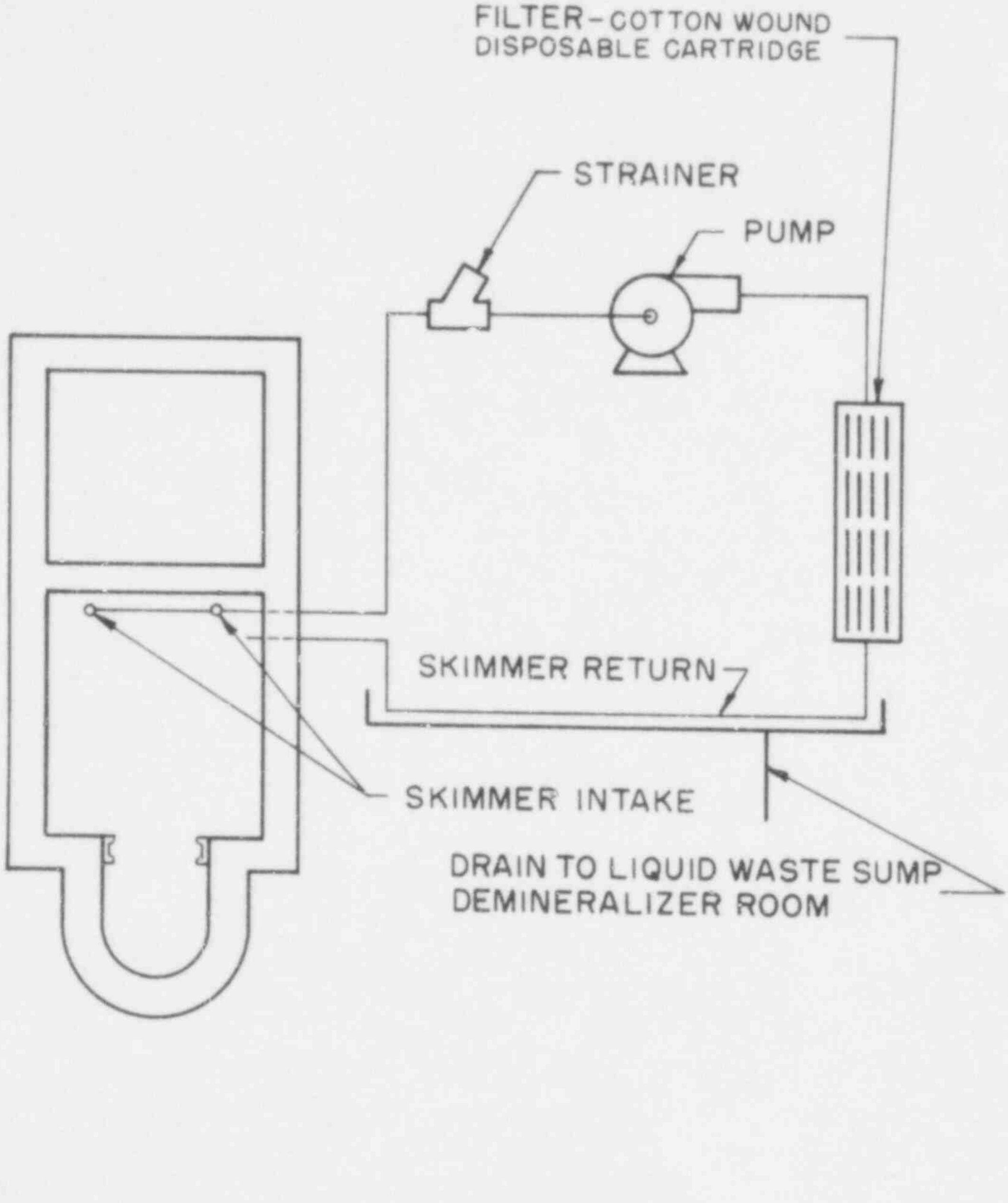

FIGURE 4-8 POOL SKIMMER SYSTEM

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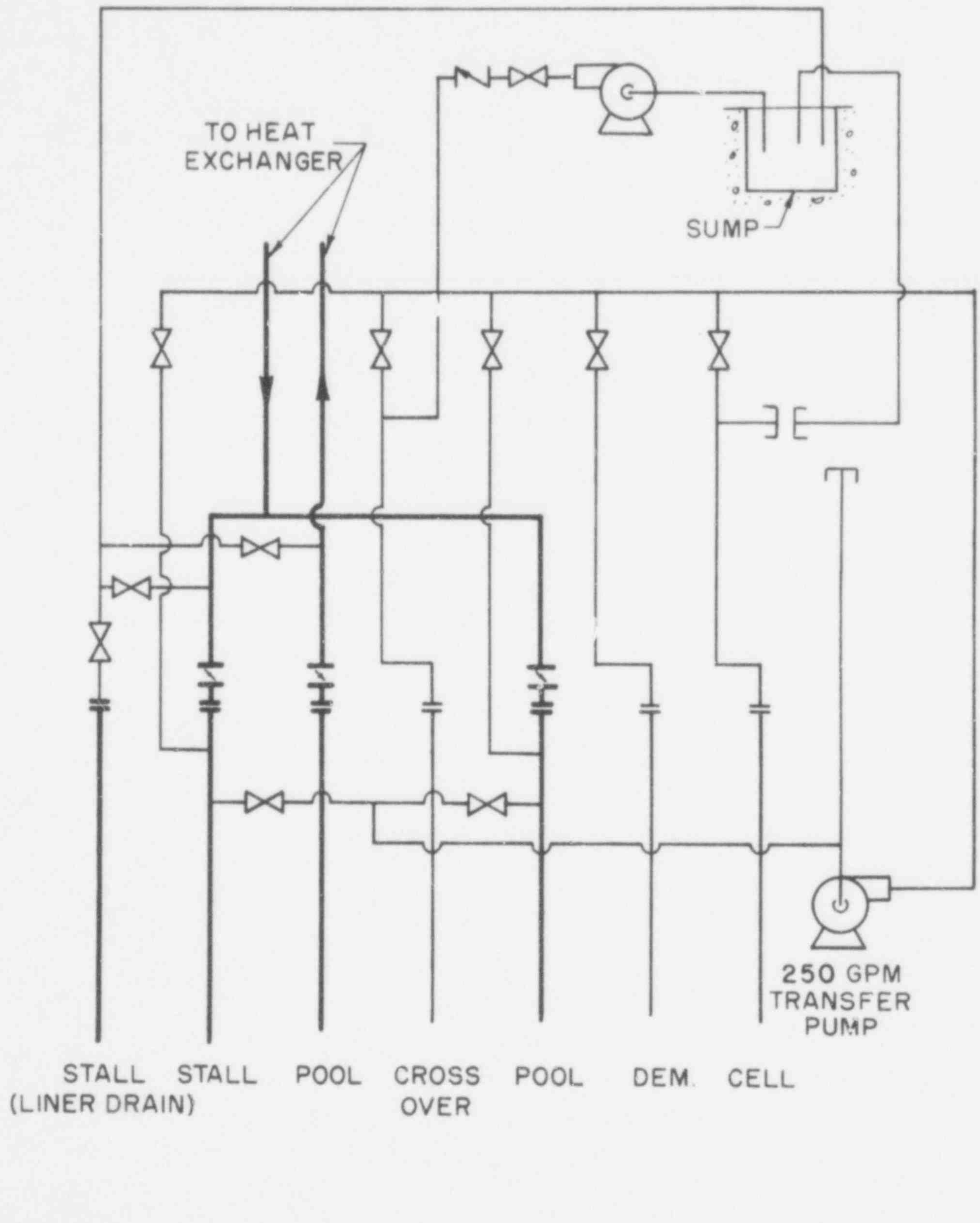

FIGURE 4-9 POOL WATER TRANSFER SCHEMATIC

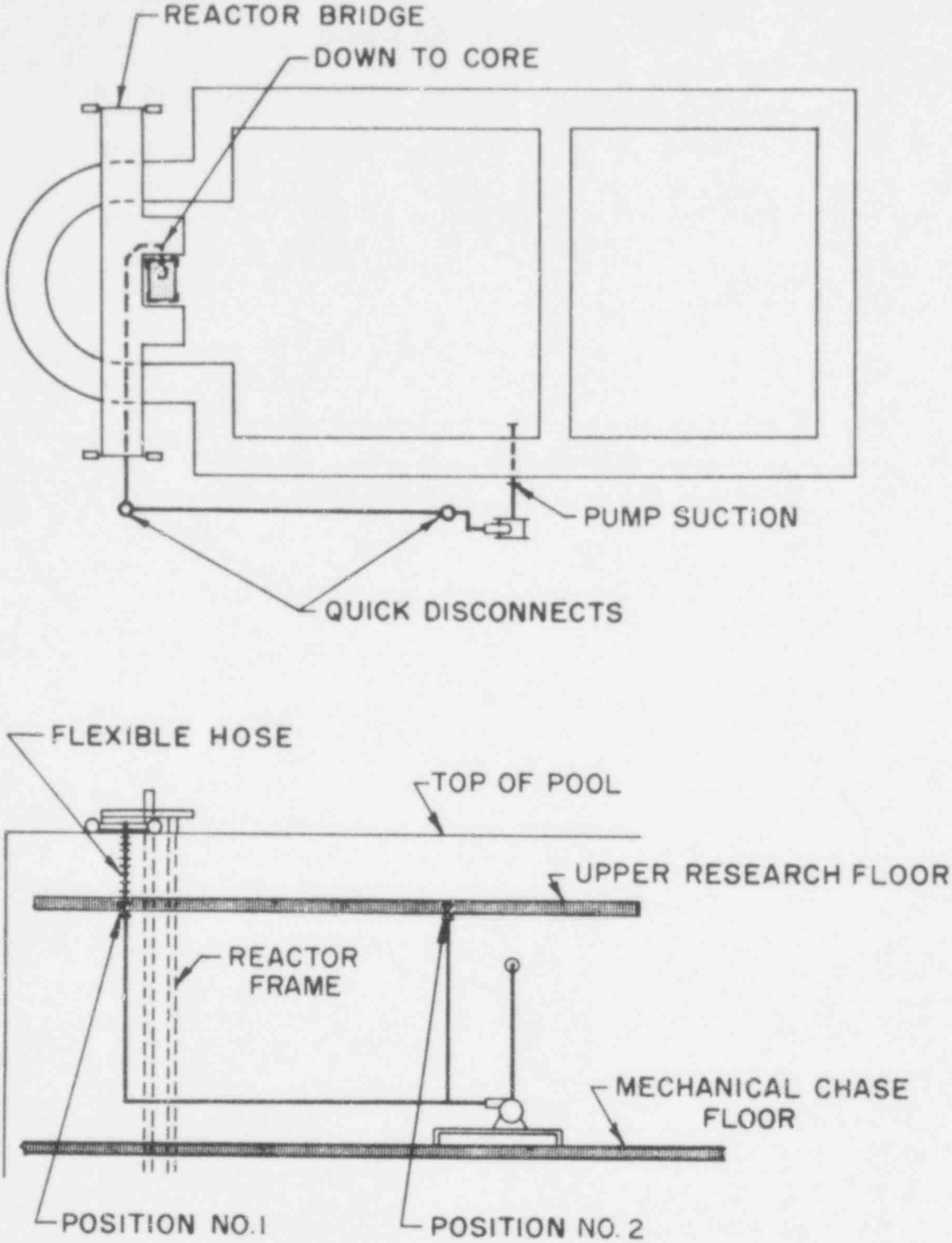

FIGURE 4-10 CORE DIFFUSER SYSTEM

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.

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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

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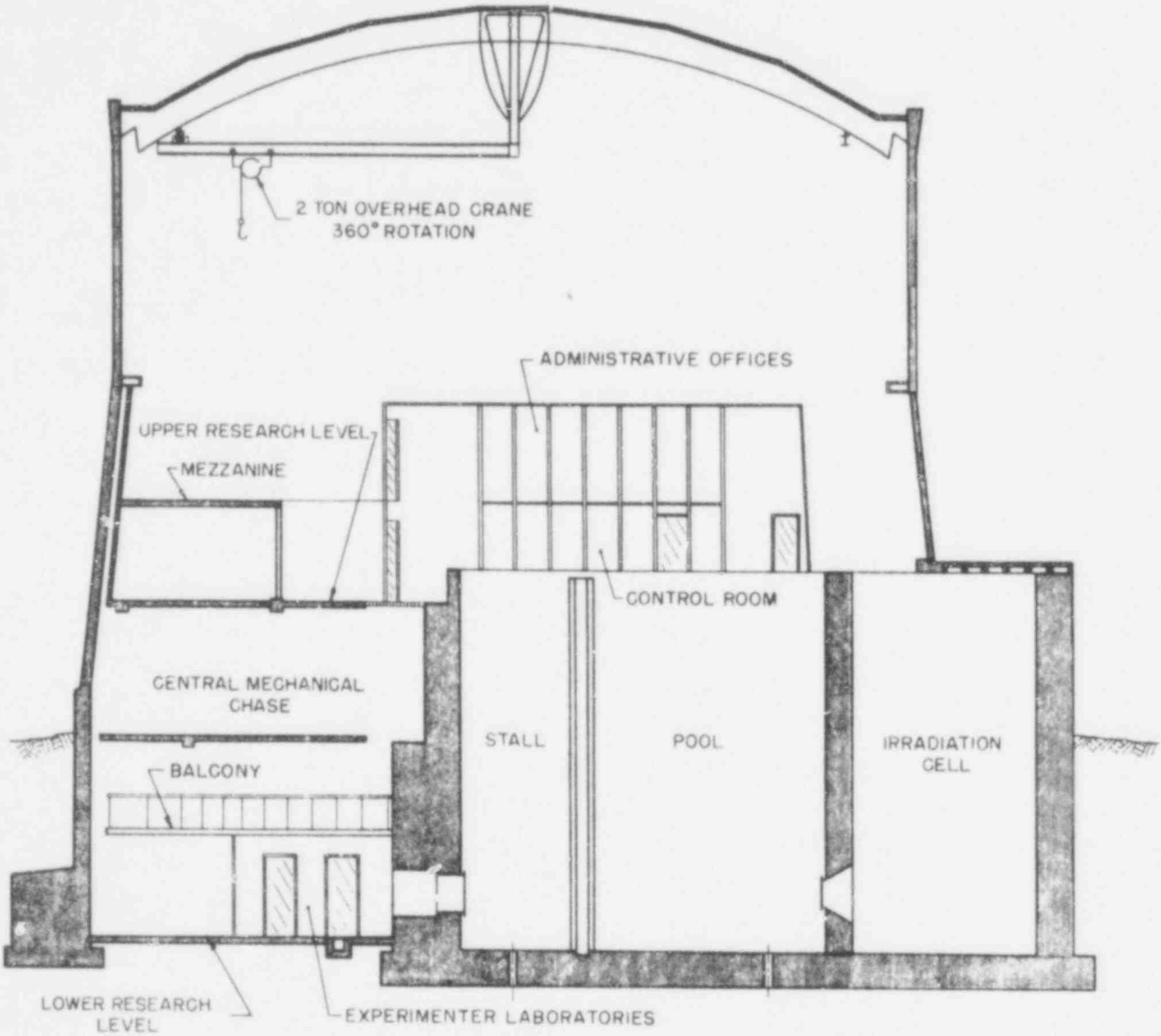

FIGURE 5-1 NSCR BUILDING CROSS SECTION

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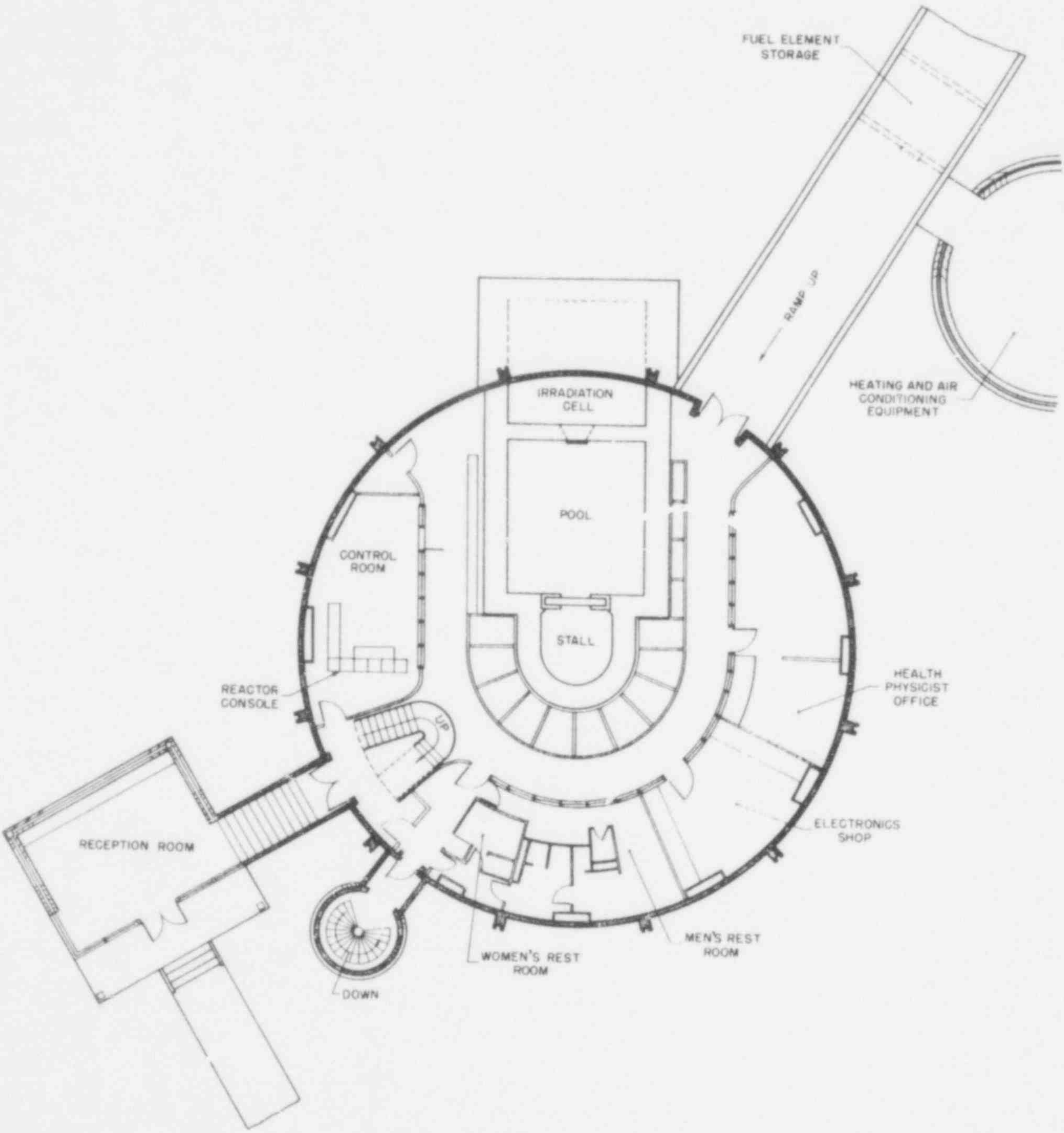

FIGURE 5-2 UPPER RESEARCH LEVEL

324 149

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 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.

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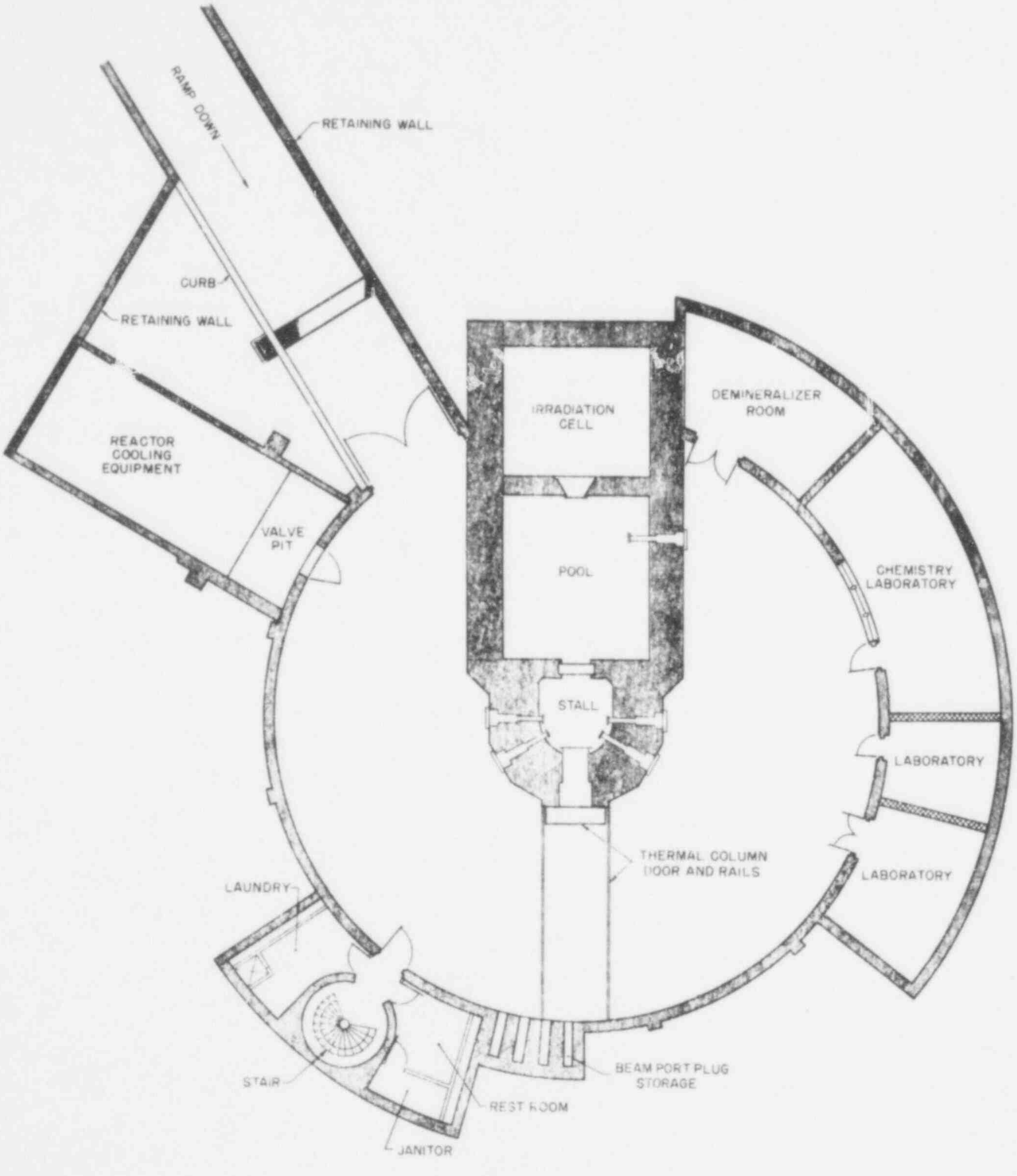

FIGURE 5-3 LOWER RESEARCH LEVEL

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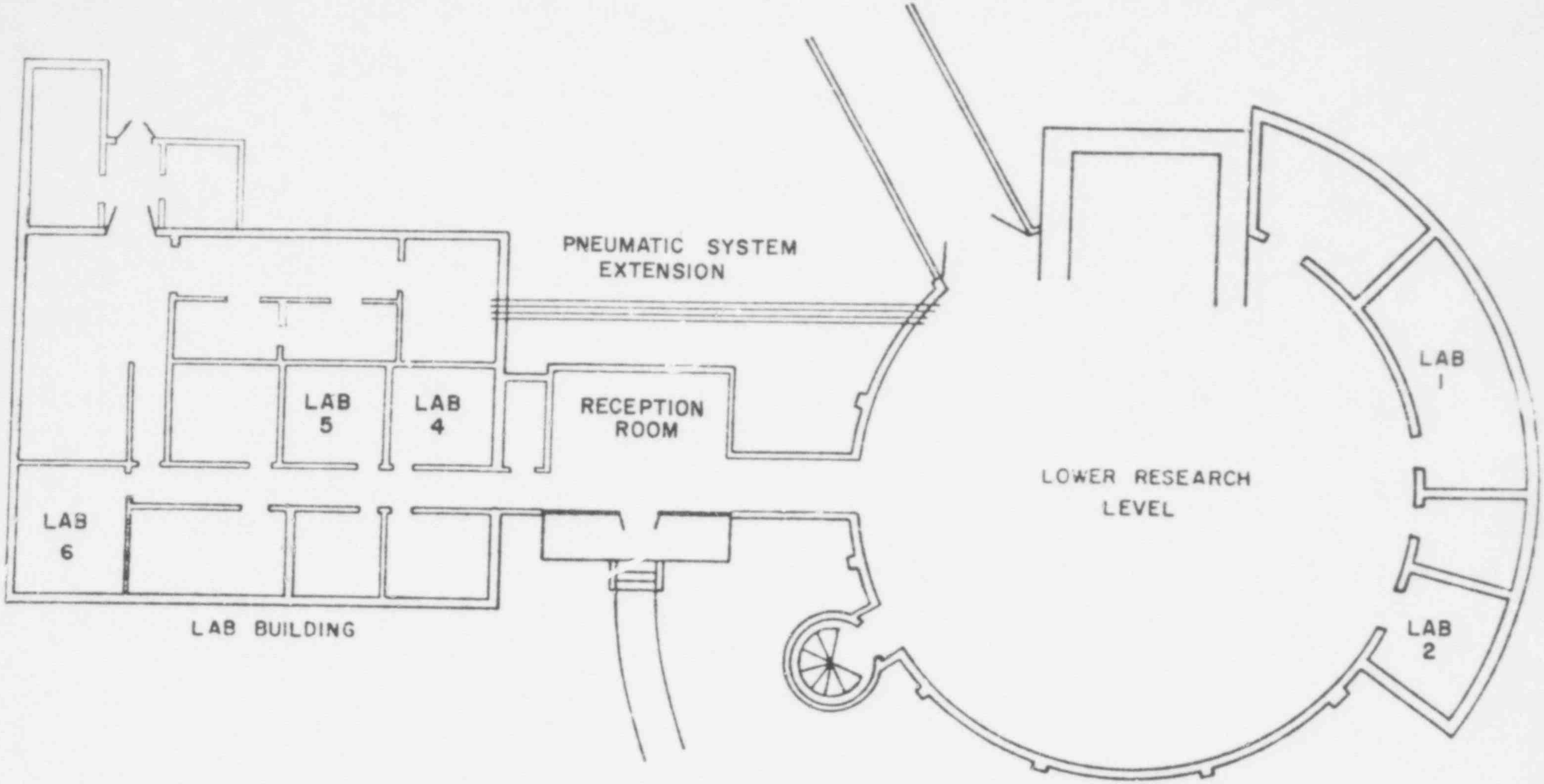

FIGURE 5-4 LABORATORY BUILDING AND PNEUMATIC SYSTEM EXTENSION

324 152

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 exhaus stack. The height of the exhaust stack above ground level is 85 f 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.

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.

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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.

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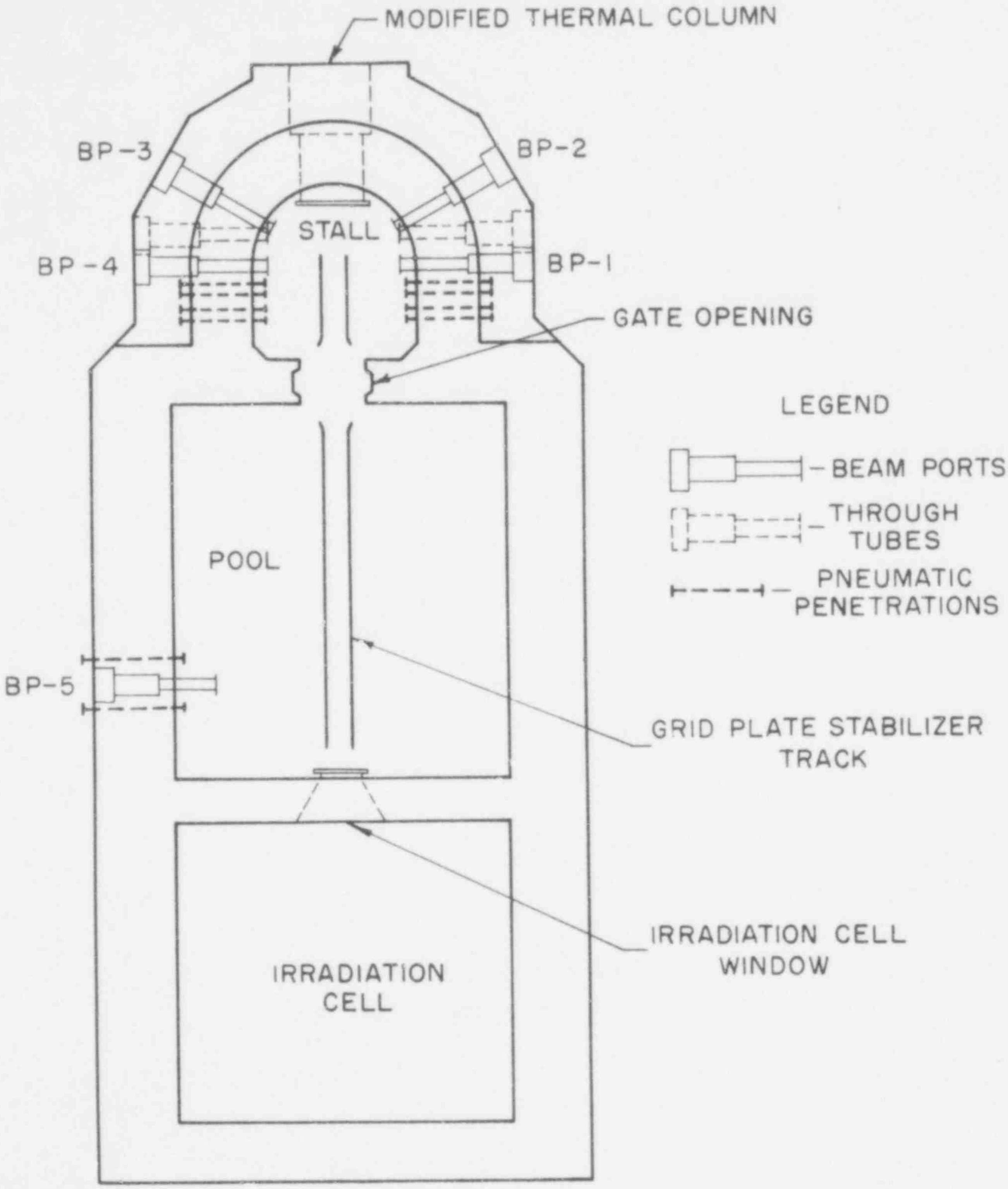

FIGURE 6-1 NSCR EXPERIMENTAL FACILITIES

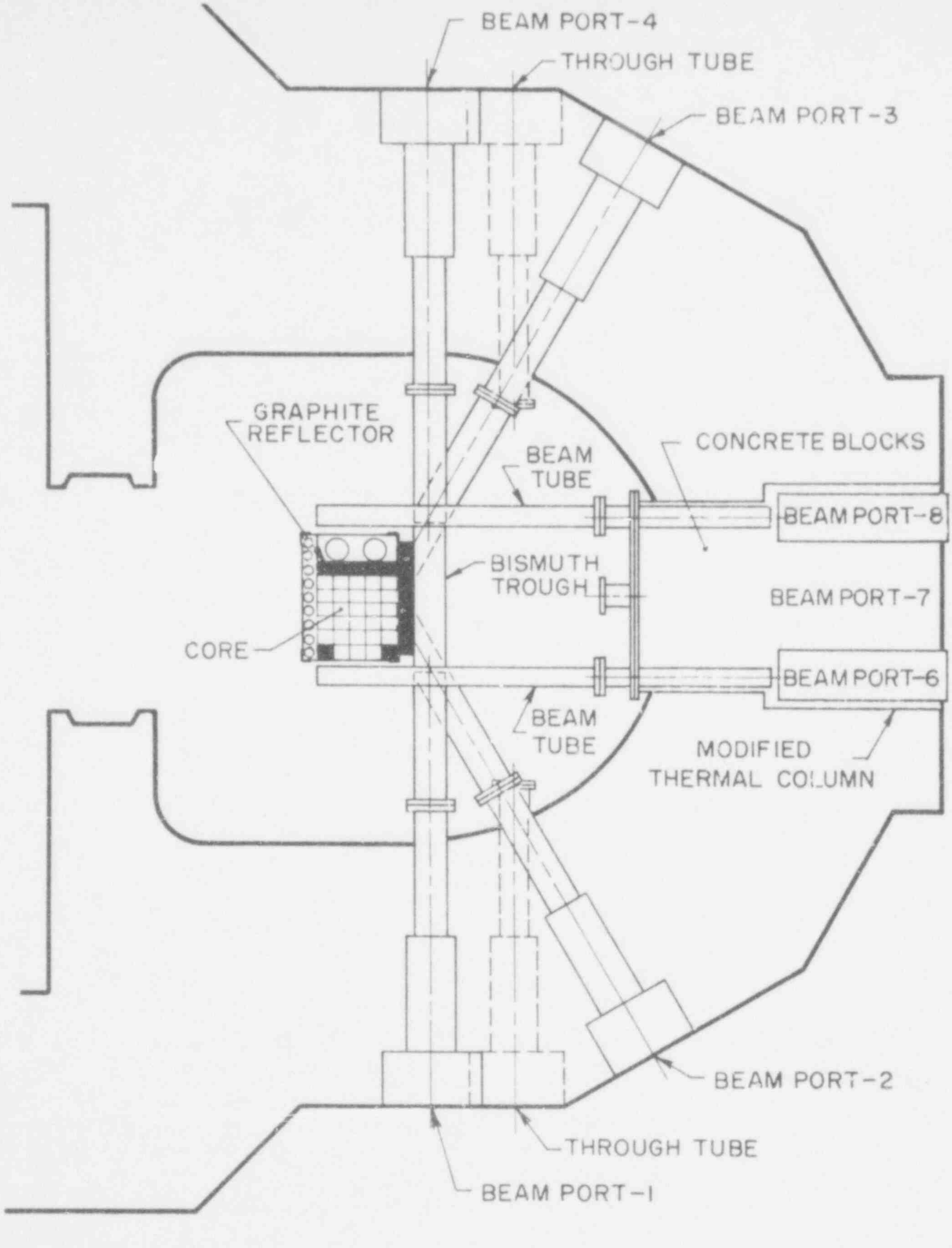

FIGURE 6-2 - REACTOR STALL AND BEAM PORT INSTALLATIONS

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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 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.

324 157

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 CO_2 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, 6-4). The pool wall pneumatic penetrations shown in Figure 6-1 have not been used for some

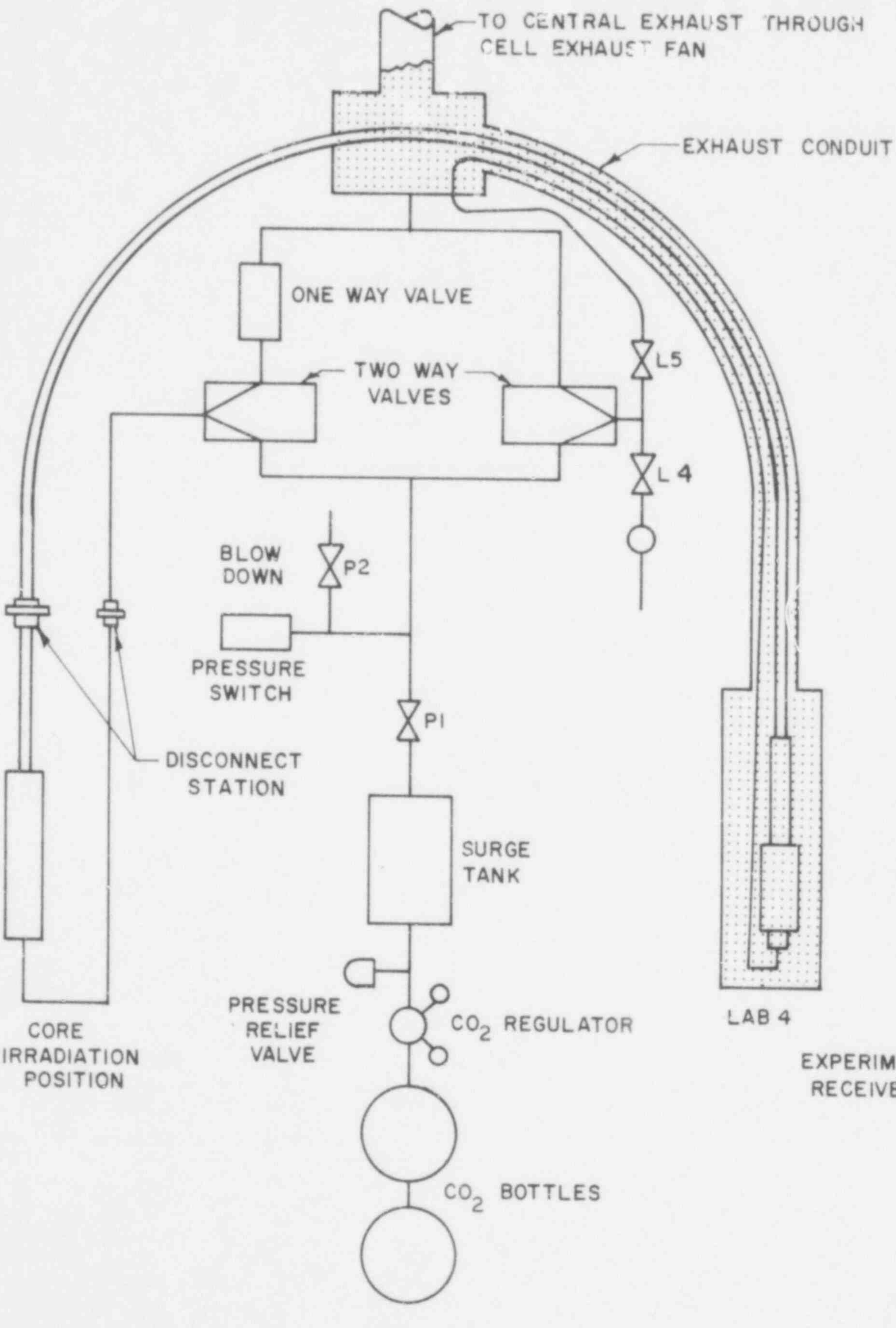

FIGURE 6-3 NSC PNEUMATIC SYSTEM
SOUTH SIDE - LABS 4 & 5

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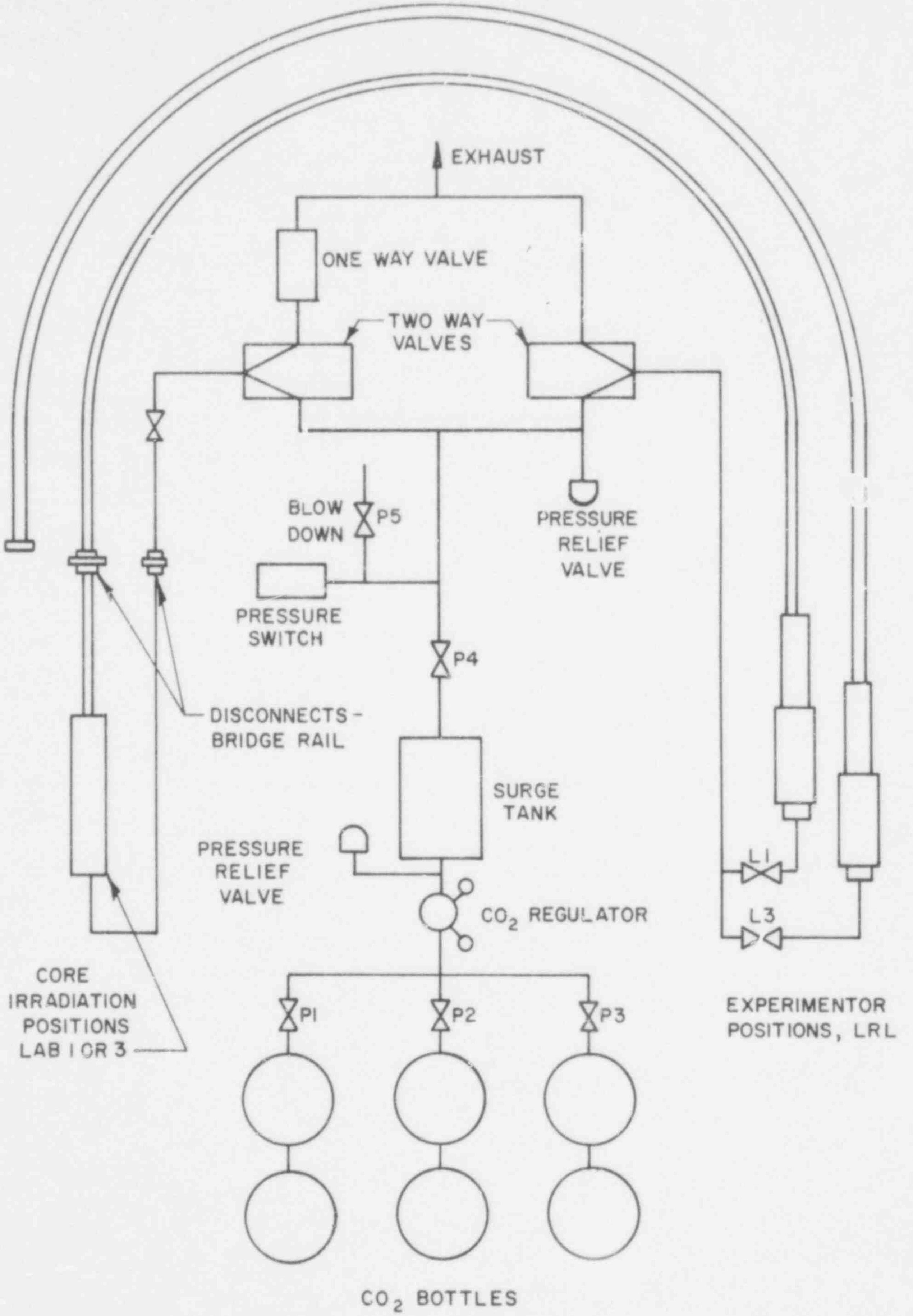

FIGURE 6-4 NSC PNEUMATIC SYSTEM NORTH SIDE - LABS 1 & 3

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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. Hand rails 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

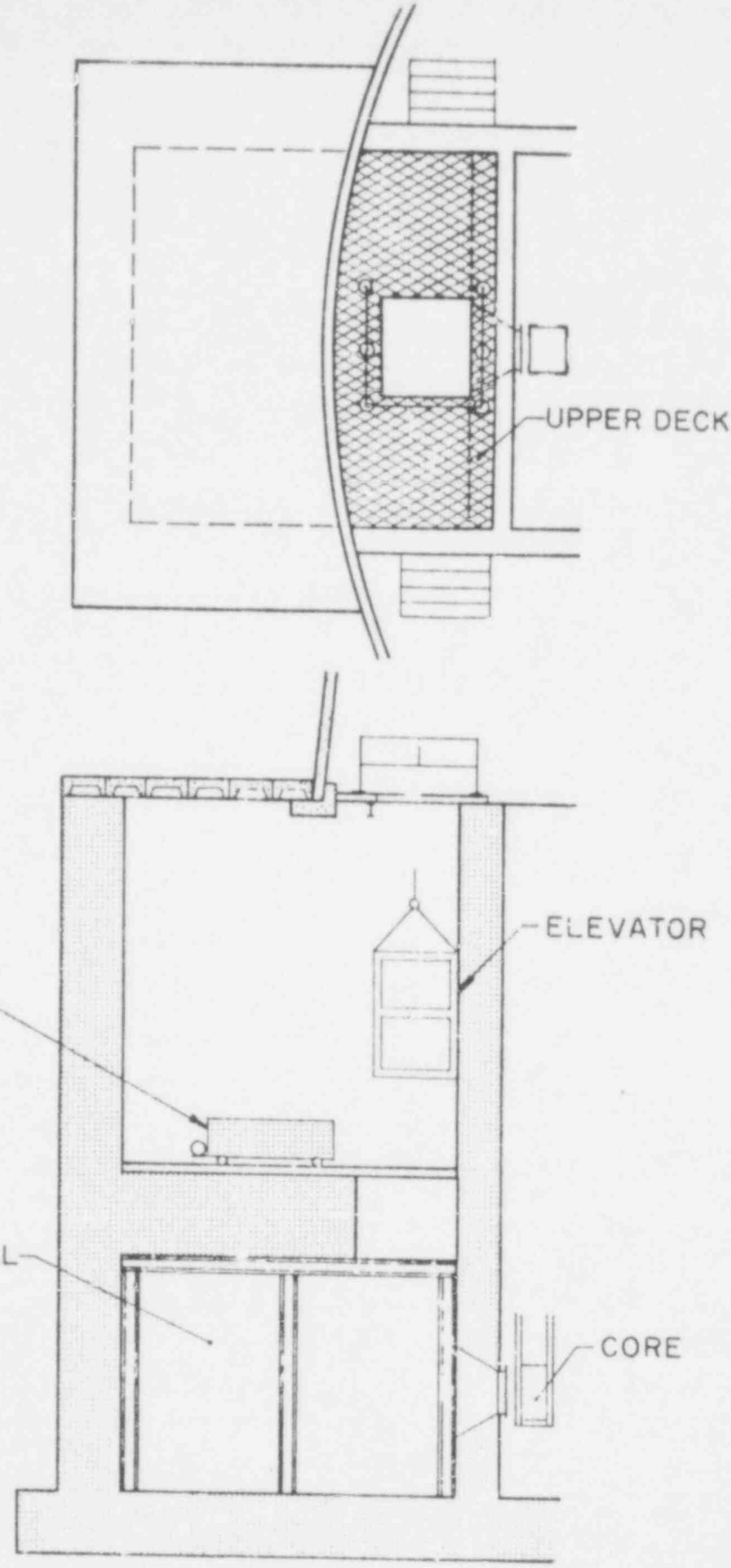

FIGURE 6-5 IRRADIATION CELL

always attached on the NSCR bridge rail to prevent inadvertant movement of the NSCR closer than eight feet from the irradiation cell window. Also, a bridge interlock scram is 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 to contain and shield a thermal neutron beam extracted from beam port 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.

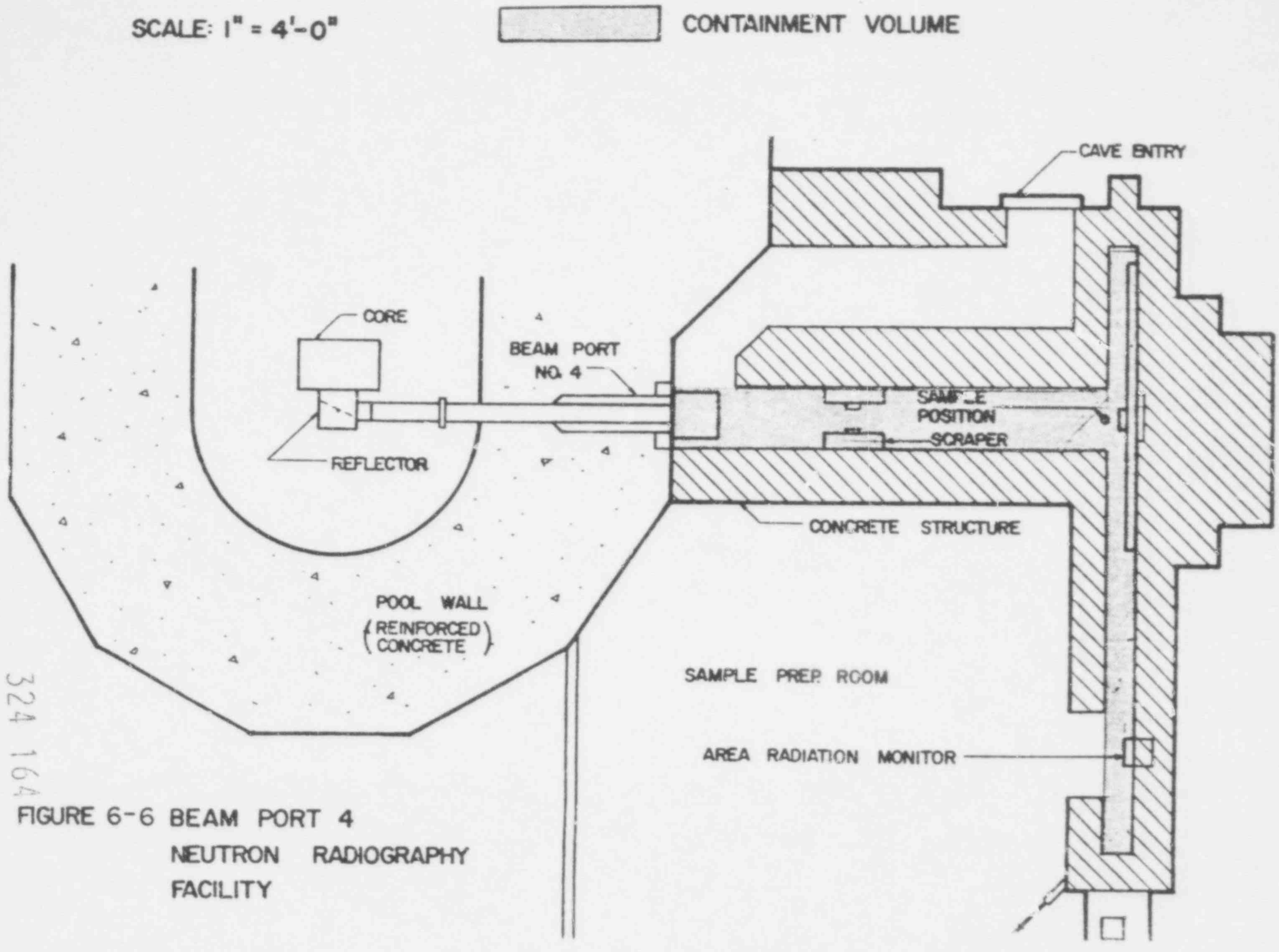

FIGURE 6-6 BEAM PORT 4
NEUTRON RADIOGRAPHY
FACILITY

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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.

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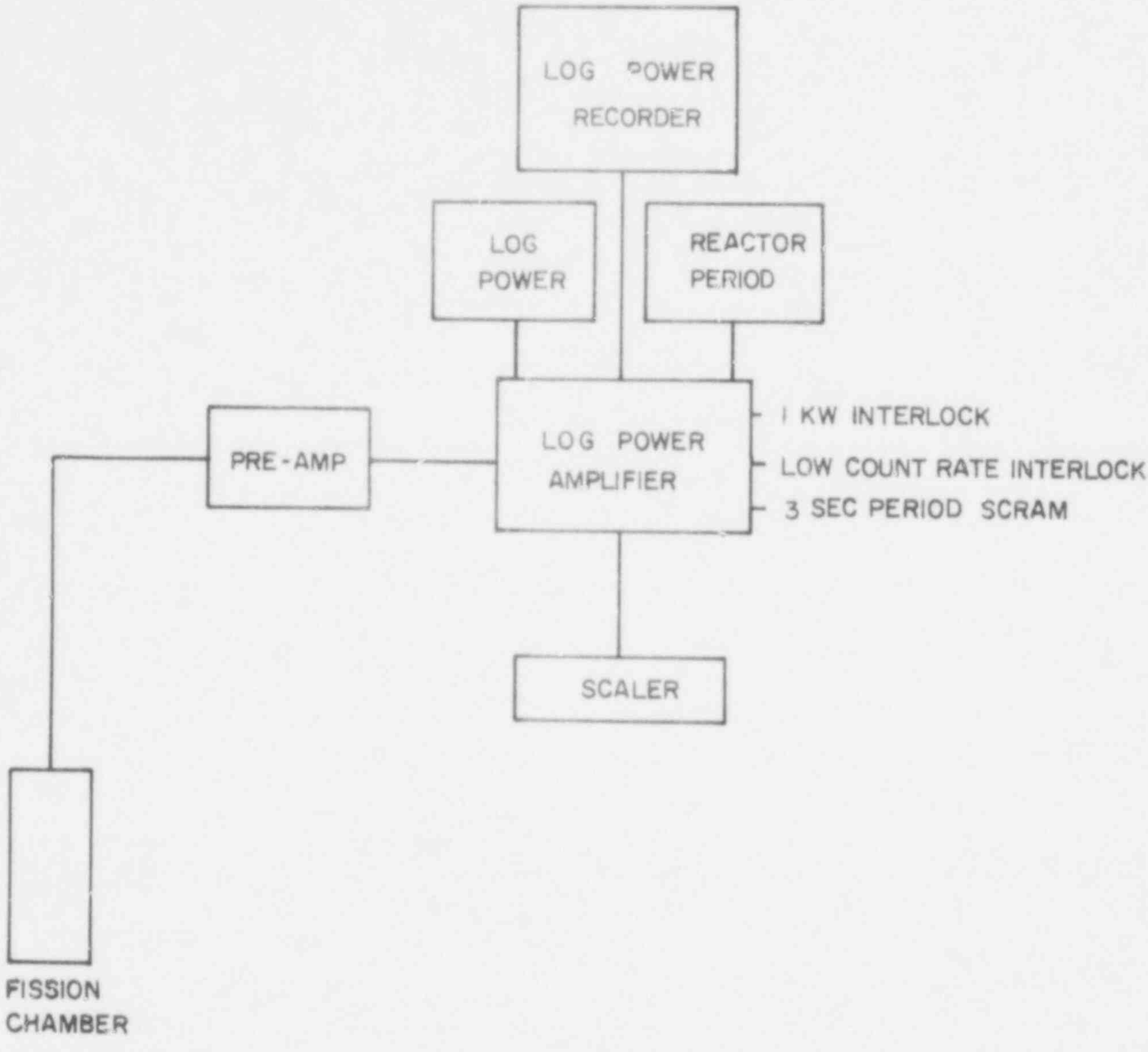

FIGURE 7-1 LOG POWER CHANNEL

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.[12]

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 ± 5% of the set point on the linear recorder. Regulating rod control is automatically shifted back to manual if the level drifts out of the ± 5% 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 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 scramming 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

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

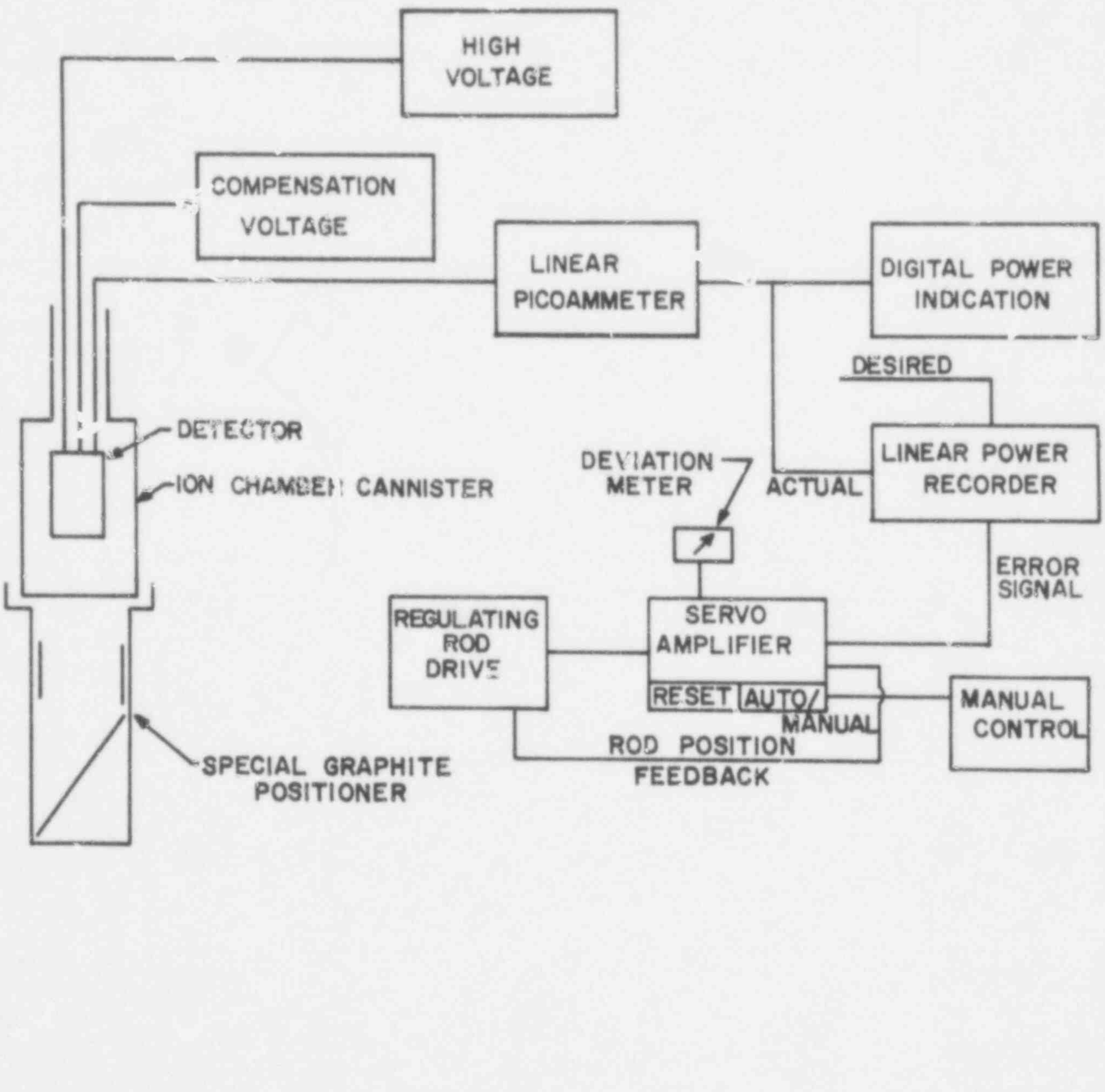

FIGURE 7-2 LINEAR POWER MEASURING CHANNEL

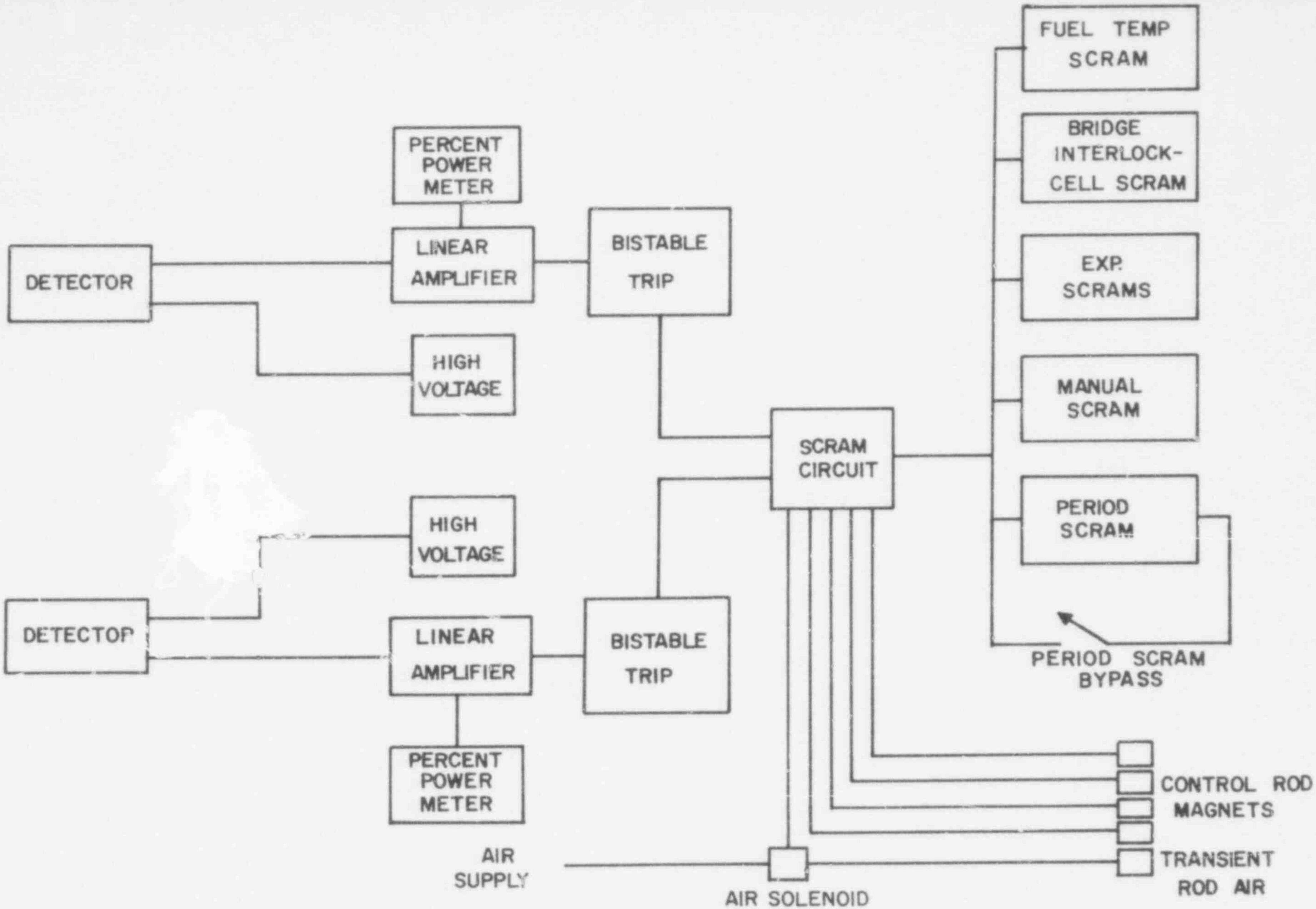

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FIGURE 7-3 SAFETY POWER MEASURING CHANNEL

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 975°F. 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 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 interlock 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

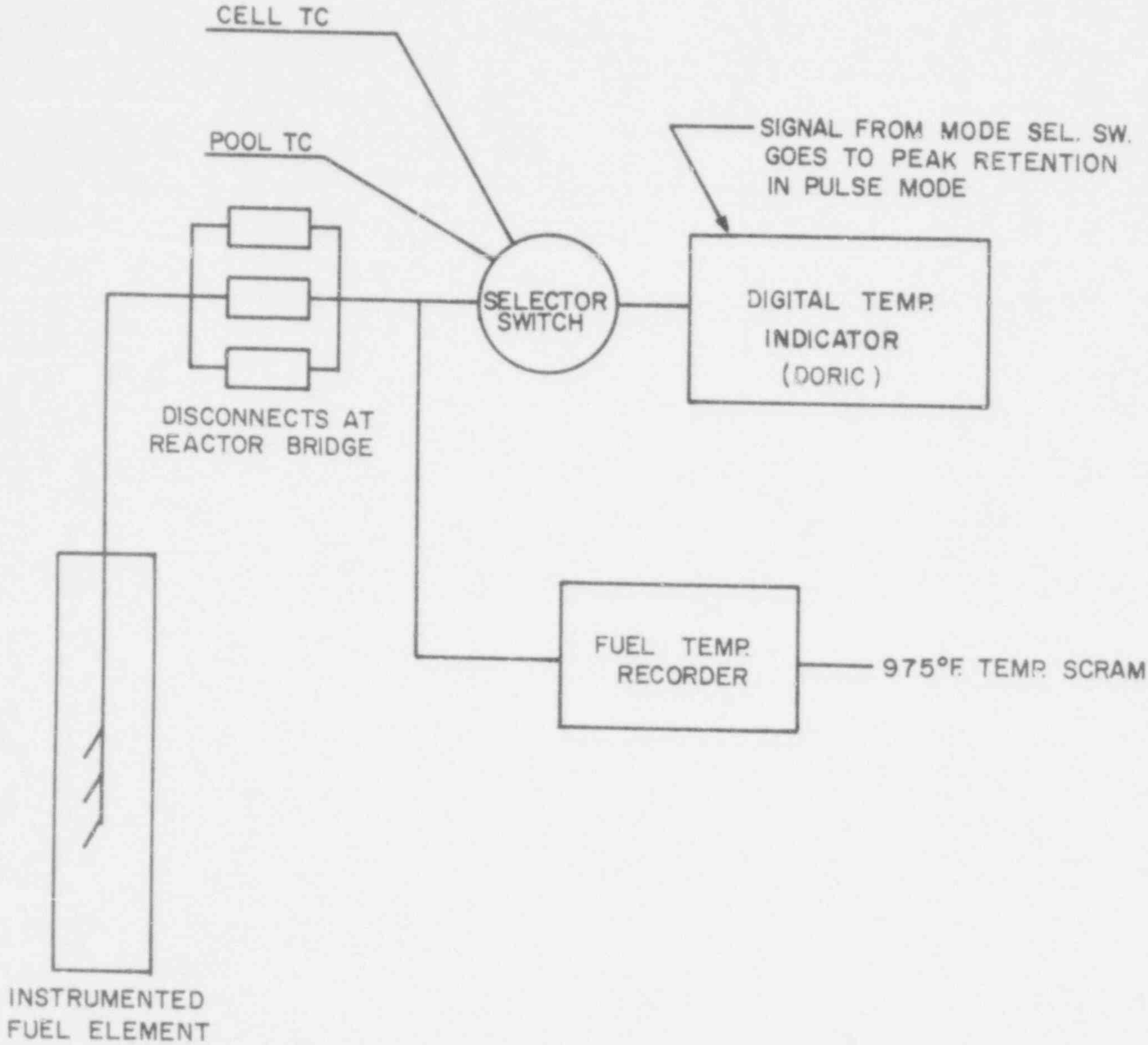

FIGURE 7-4 FUEL ELEMENT TEMPERATURE CHANNEL

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

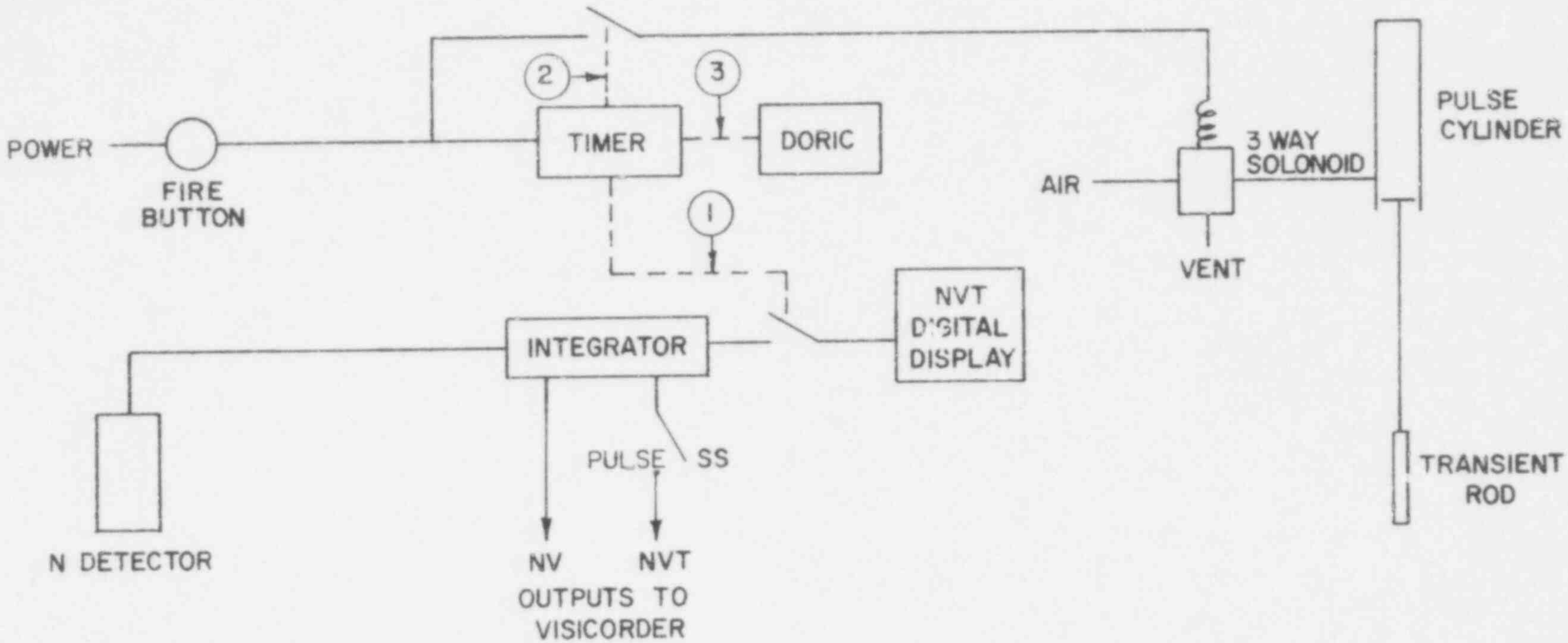

FIGURE 7-5 PULSE POWER MEASURING CHANNEL

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 V 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

TABLE V

Minimum Reactor Safety Channels

Safety Channel	Number Operable	Function	Effective Mode S.S.	Effective Mode Pulse
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 $<4 \times 10^{-3}$ 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

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 VI 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.

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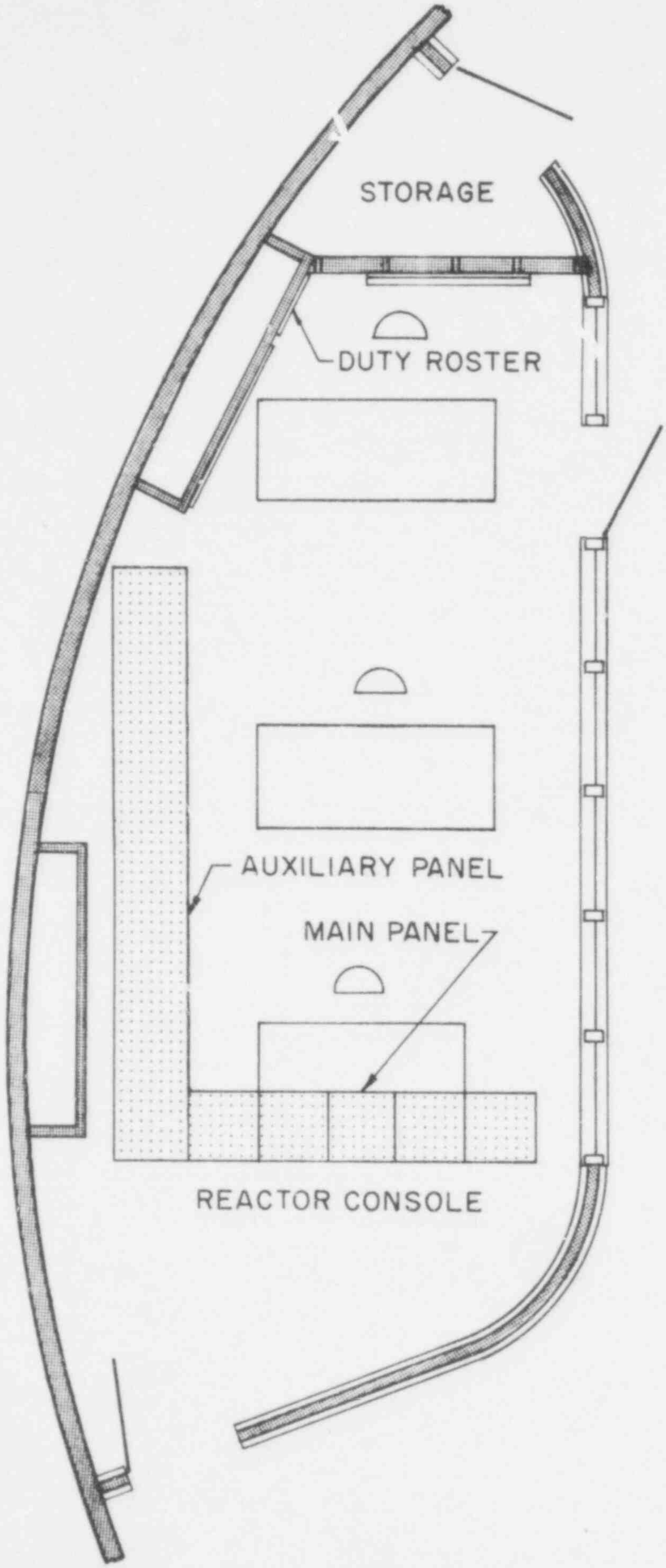

FIGURE 7-6 CONTROL ROOM LAYOUT

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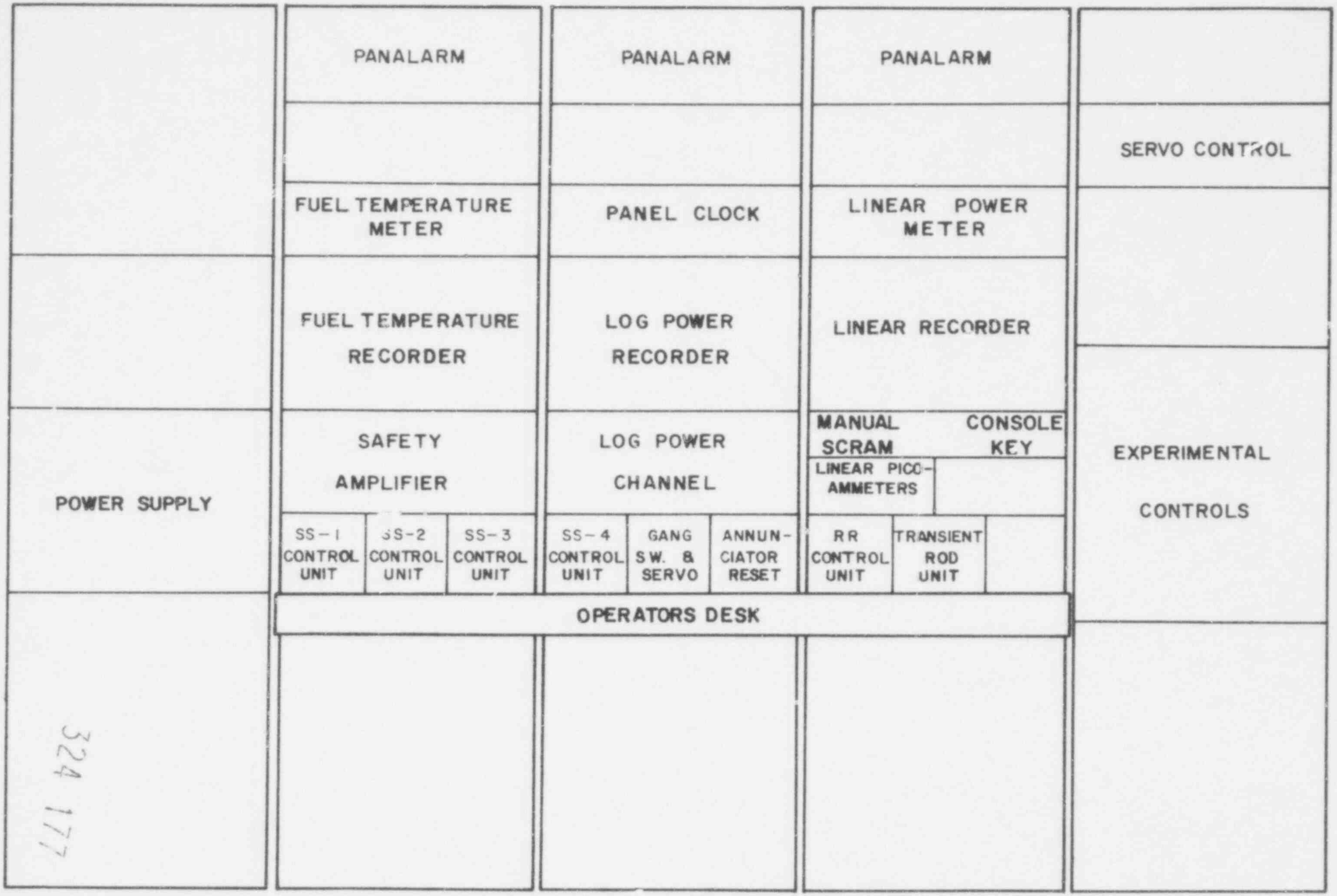

FIGURE 7-7 ARRANGEMENT OF MAIN PANEL OF THE REACTOR CONSOLE

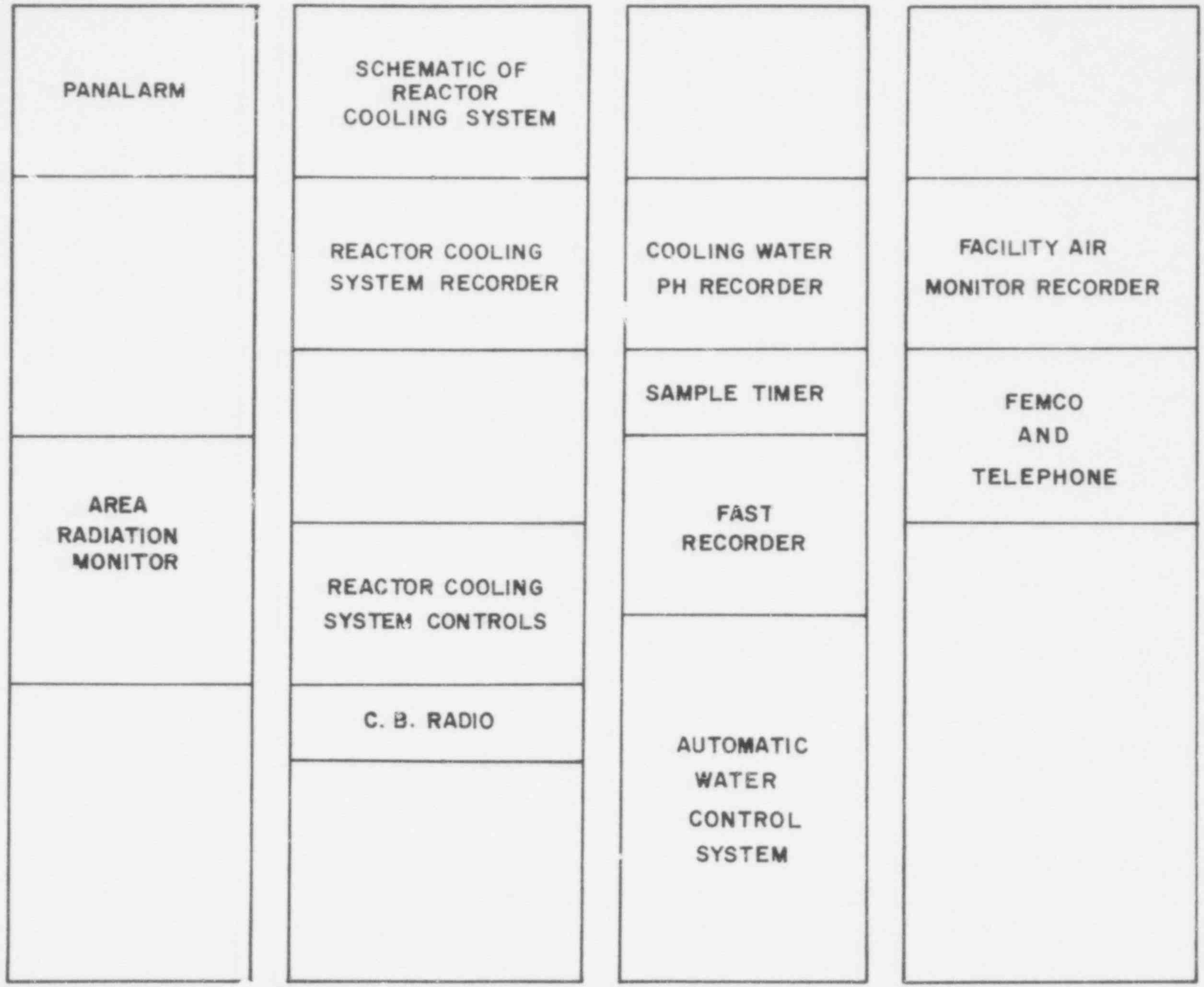

FIGURE 7-8 ARRANGEMENT OF AUXILIARY PANEL OF THE REACTOR CONSOLE

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TABLE VI

Summary of Information Displayed and Recorded on Reactor Console

	Control	Indication	Record
<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	

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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

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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. CO_2, 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.

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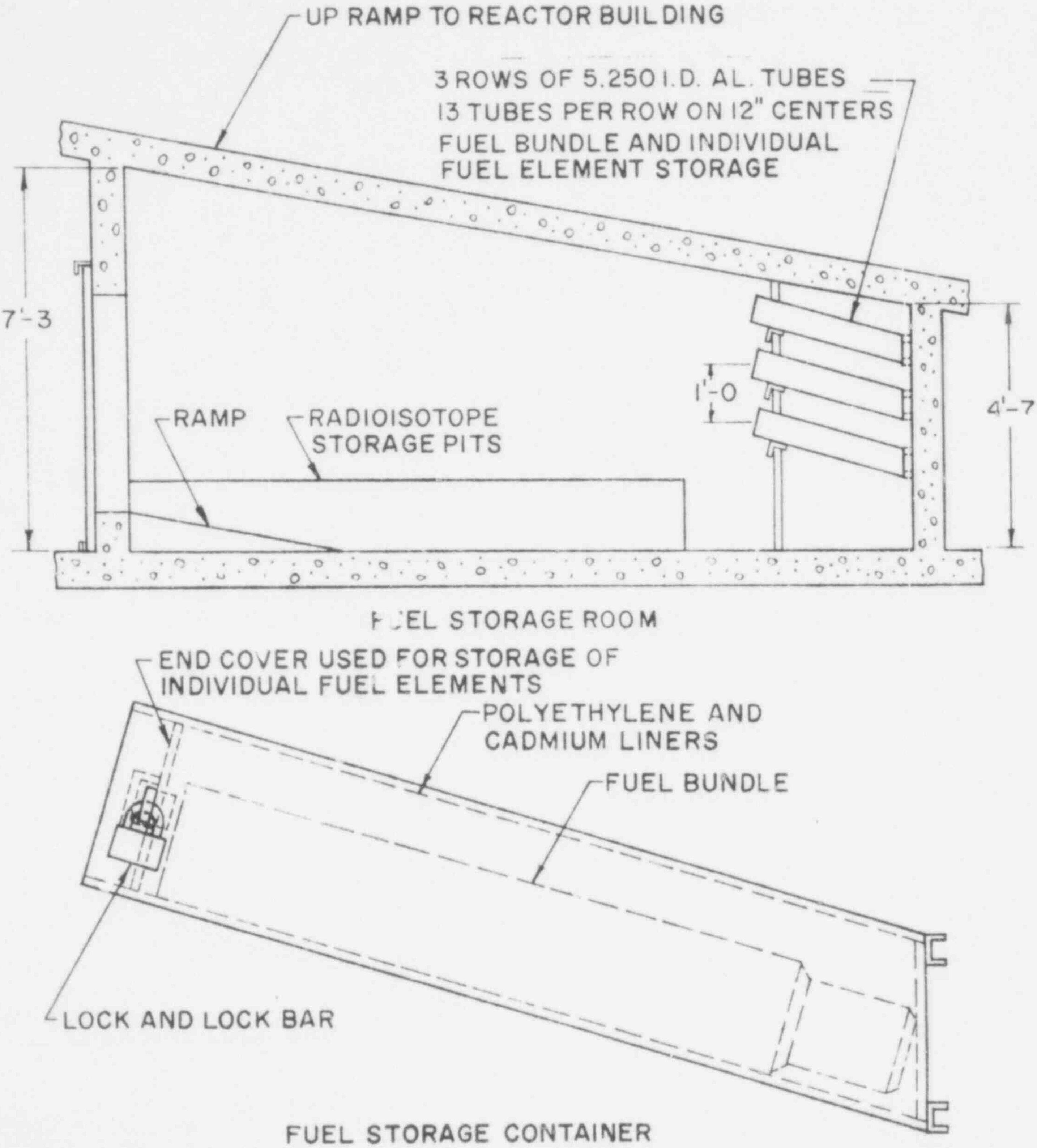

FIG. 8-1 FUEL STORAGE ROOM AND RACK

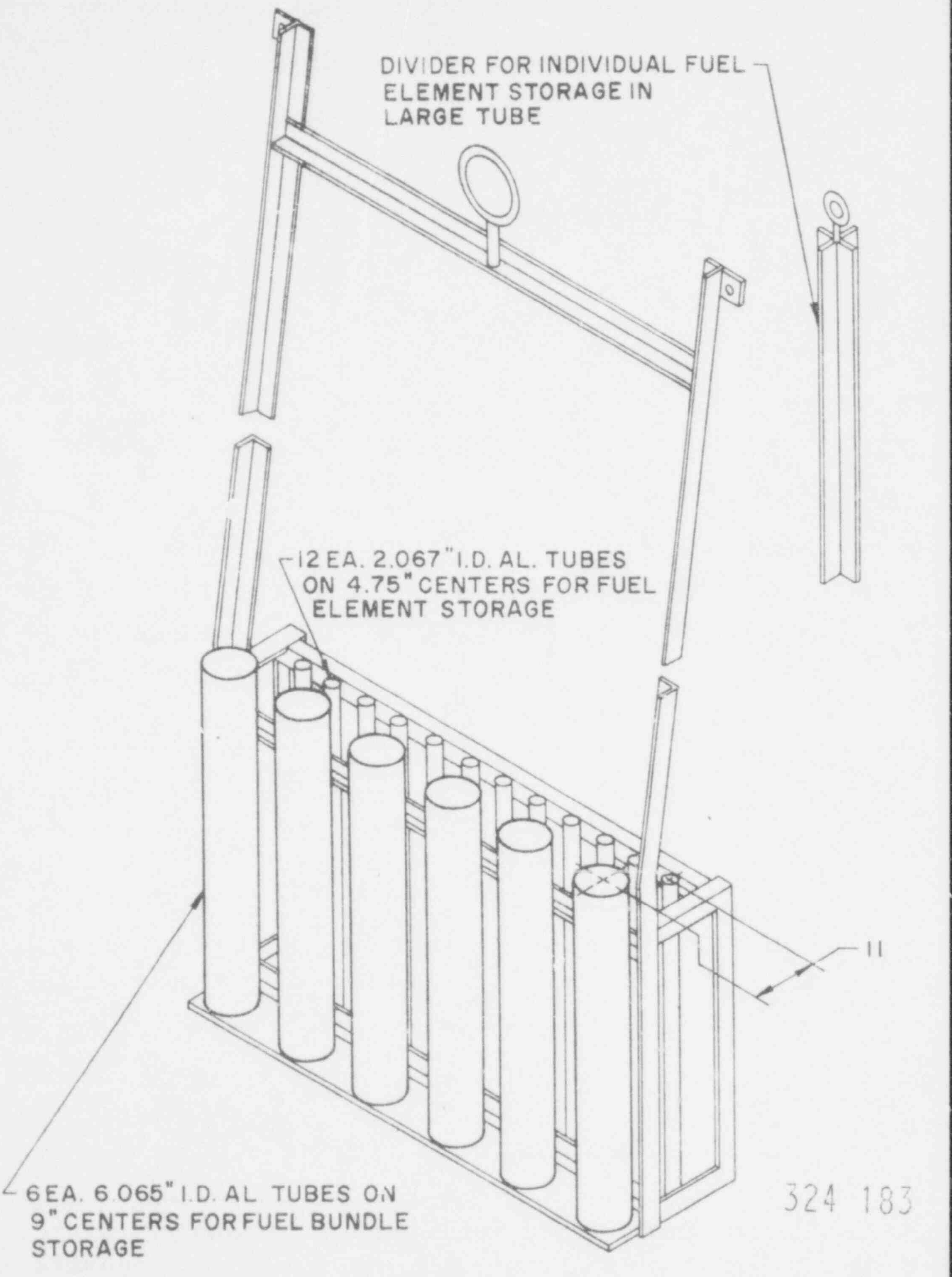

FIG. 8-2 FUEL STORAGE RACK FOR REACTOR POOL WALL

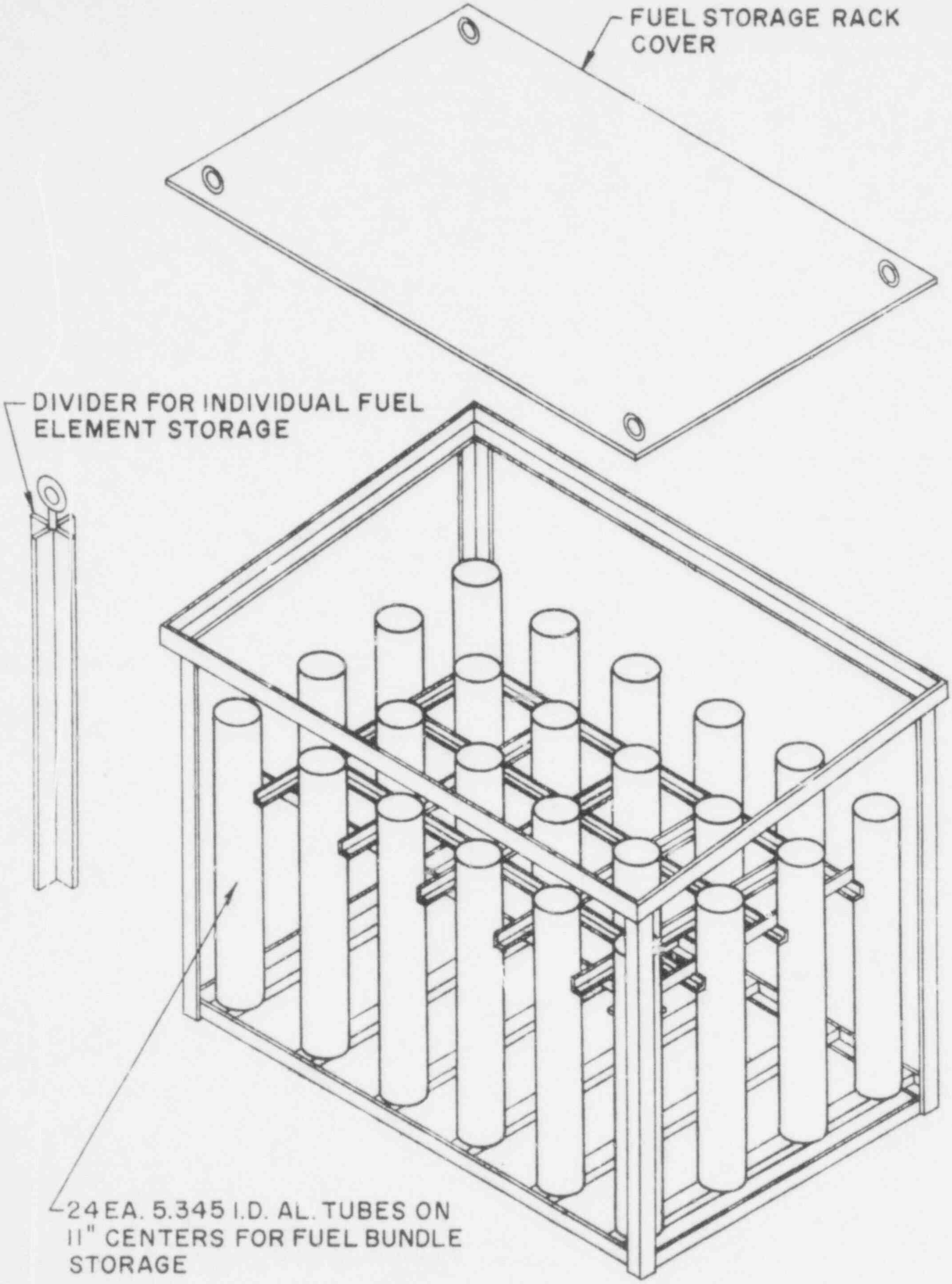

FIG. 8-3 FUEL STORAGE RACK FOR REACTOR POOL FLOOR

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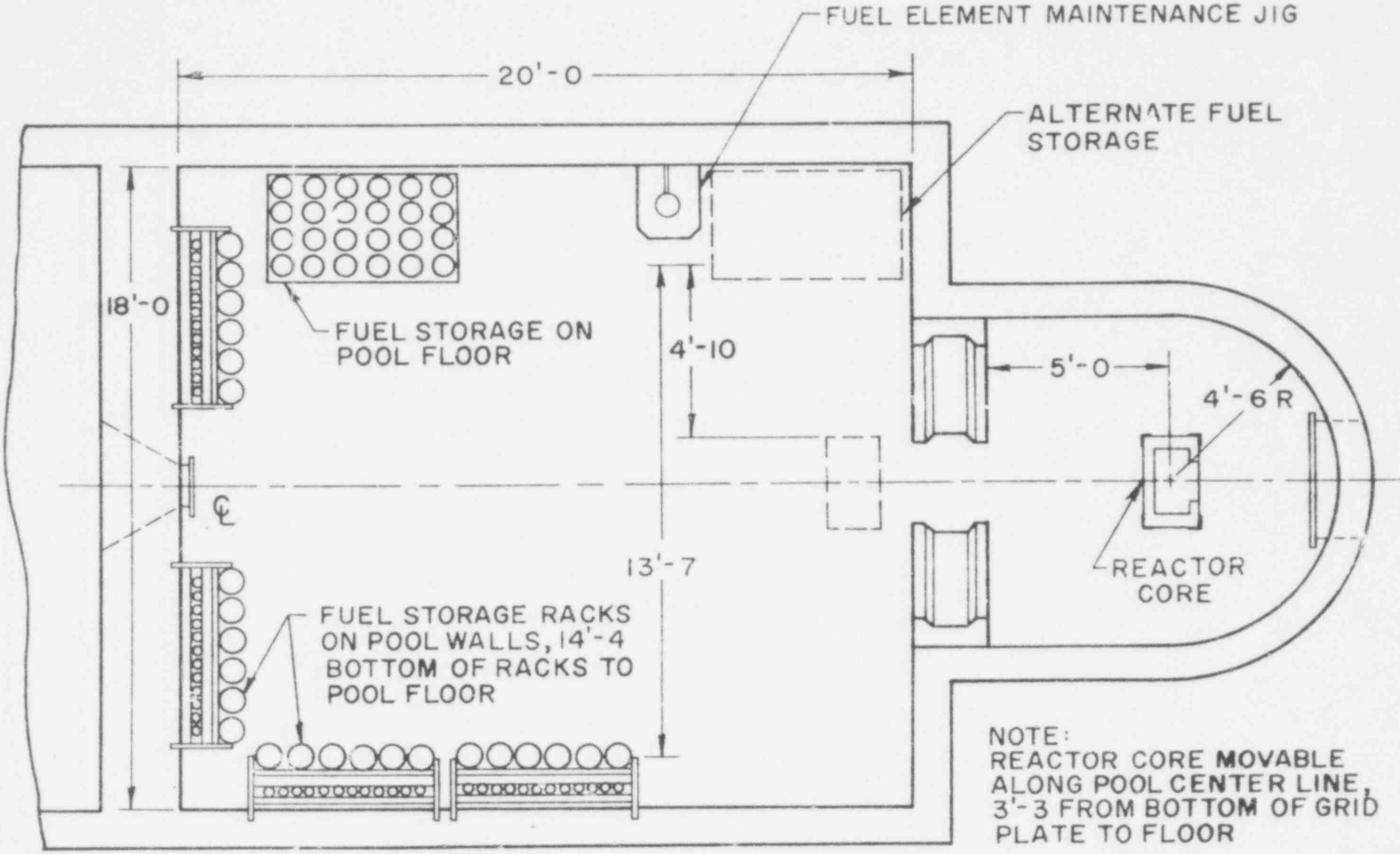

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FIG. 8-4 REACTOR POOL FUEL STORAGE ARRANGEMENT

(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.

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 familarization 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

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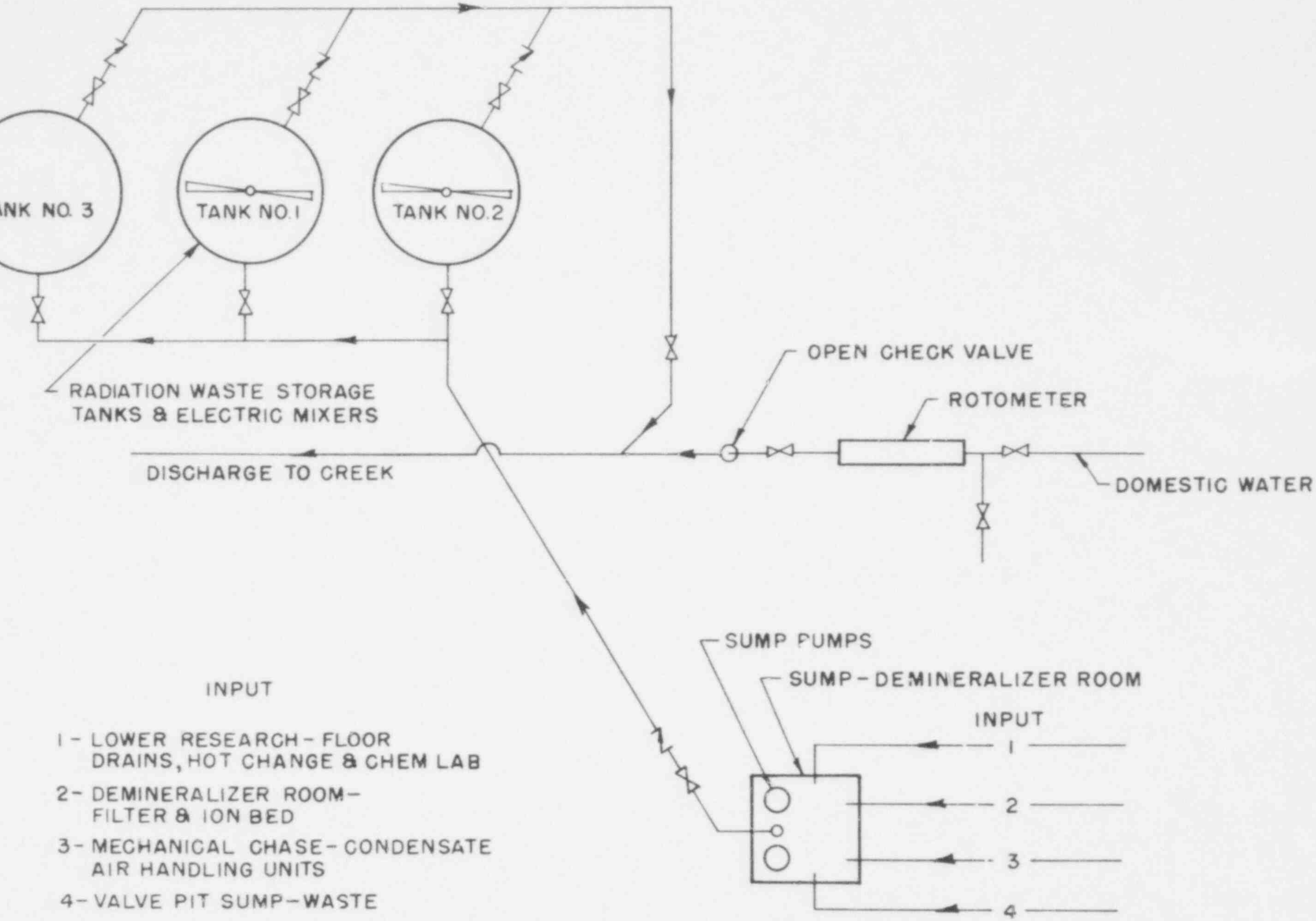

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FIGURE 9-1 LIQUID WASTE DISPOSAL SYSTEM

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.

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 x 10^{-5} μCi/cc and 1.2 x 10^{-4} μCi/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 μCi present before reactor startup to a plateau value of about 208 μCi 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 x 10^{-3} μCi/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.[13,14] These calculations are based on release at ground level and utilize the building dilution factor D_B = cAu, where A is the cross sectional area of the building normal to the wind and u is wind speed in meters/second. From reference 13, C is estimated to be 0.5. The cross sectional area of the Nuclear Science Center is 357 m^2.

The equation for the atmospheric dilution factor is:

$$X = \frac{Q}{\pi \sigma_y \sigma_z u} \exp \{-1/2 (\frac{y^2_2}{\sigma_y} + \frac{h^2_2}{\sigma_z})\}$$

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

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

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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 m^2

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

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

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.

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

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

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

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:

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

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.

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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 (Tl) 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

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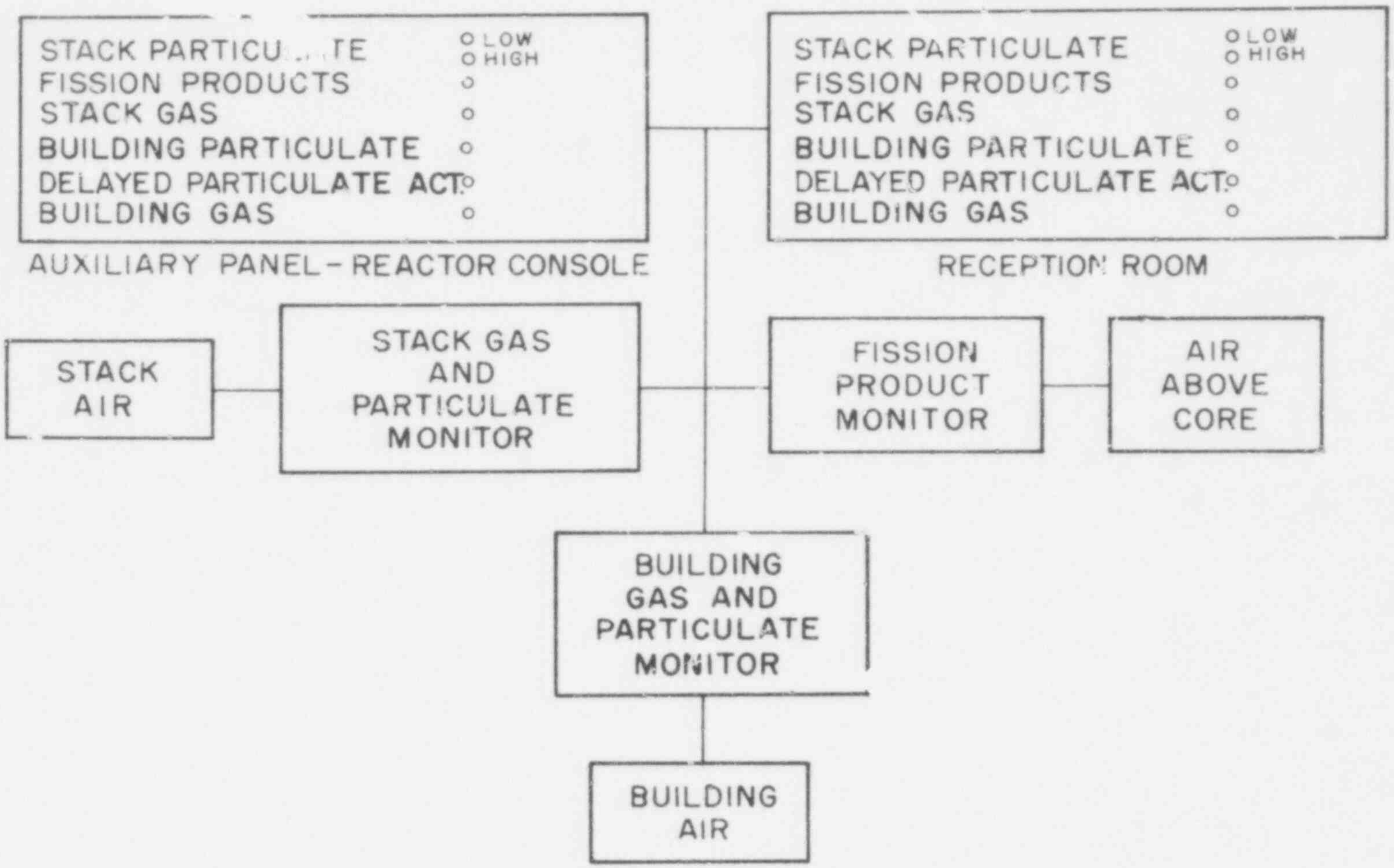

FIGURE 9-2 FACILITY AIR MONITORING SYSTEM

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the particular area. A block diagram of a typical system is shown in Figure 9-3.

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

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

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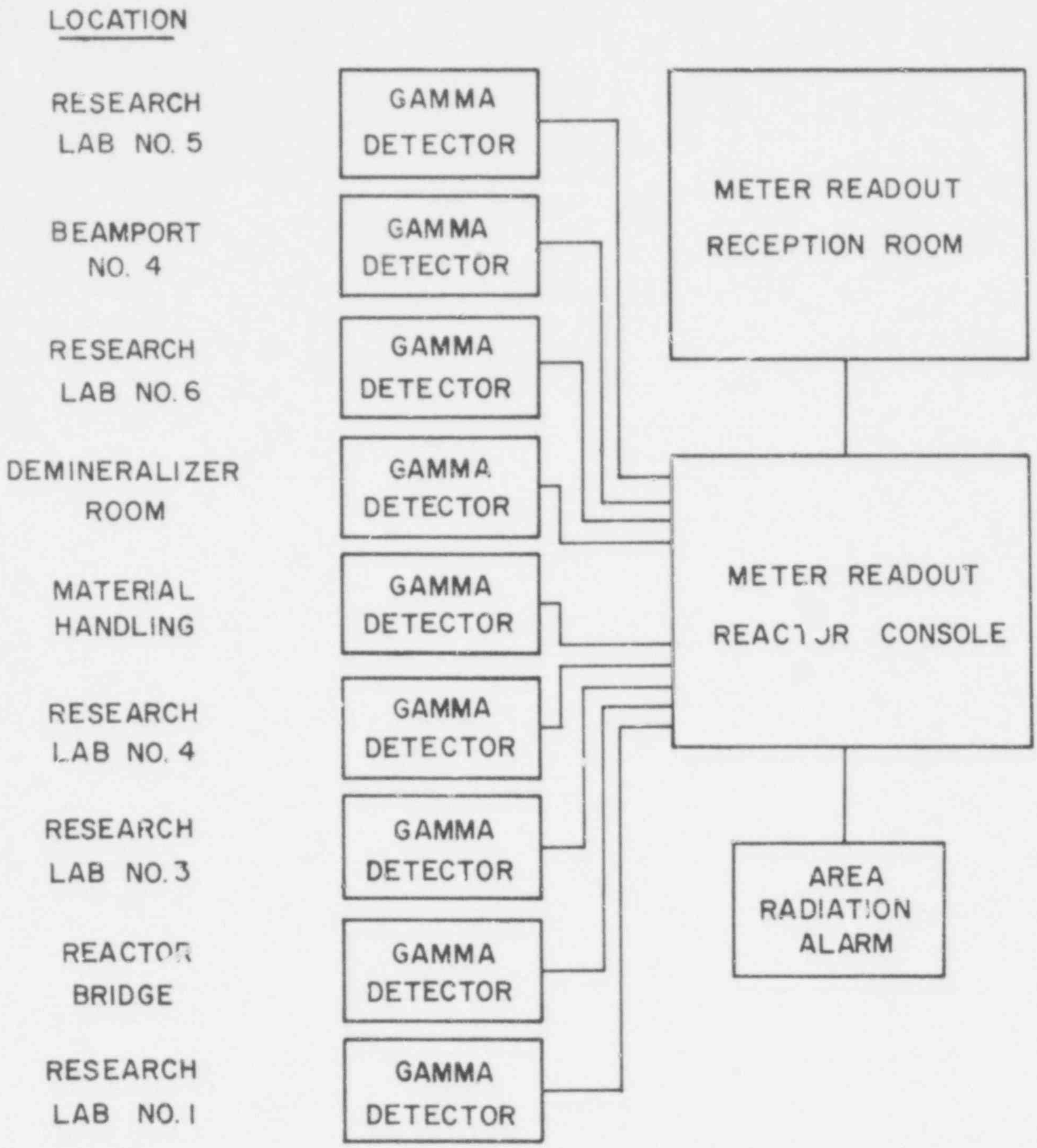

FIGURE 9-3 AREA RADIATION MONITOR SYSTEM

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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

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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 through the Associate Director of the Texas Engineering Experiment Station for the administration and the proper and safe operation of the facility. The Director of the Texas Engineering Experiment Station reports to the President of the University. 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 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.

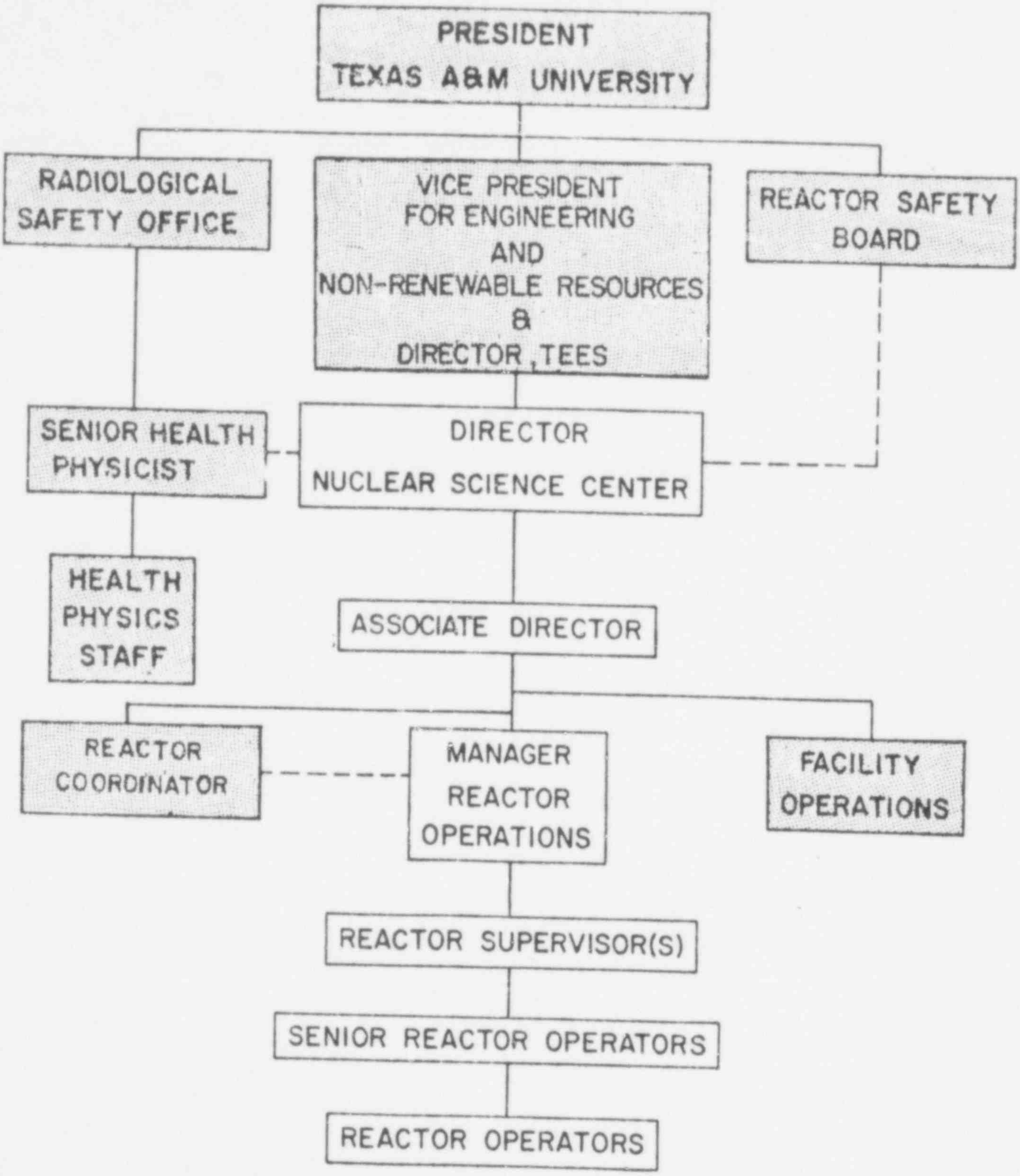

Figure 10-1
Nuclear Science Center Administrative Organizational Chart

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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 start-up 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.

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 decision to operate mixed cores at the NSCR was reinforced by the considerable amount of information available on the TRIGA and FLIP systems. Many standard TRIGA cores have been in use for years and their characteristics are well documented. The Puerto Rico Nuclear Center recently operated an all FLIP core at 2,000 kw and General Atomic has operated the Mark III at 1,500 kw with FLIP fuel for some time. Thus, considerable information on the characteristics and performance of FLIP cores is available. During the development and testing of FLIP fuel, General Atomic performed a series of experiments using a mixed standard-FLIP core which provided considerable information on mixed cores as well. The Texas A&M study which was performed for a variety of cores from all standard fuel to all FLIP fuel indicates that a core with a mixed loading would safely satisfy all operational requirements.

Operation of mixed TRIGA cores has shown that these studies were satisfactory and only one significant incident has occurred during operation of the NSCR that questions design or fuel characteristics. The discovery of damaged FLIP fuel elements adjacent to the transient rod occurred in September of 1976. There was no failure of the fuel clad and no release of fission products. The fuel was removed from service and a report of the incident and status of the investigation was issued to the NRC on November 1, 1976. Pulsing activities were terminated and will not resume until a final analysis and report of the damaged fuel incident is completed. The fuel damage occurred during pulsing of a 35 FLIP mixed core using $2.70 reactivity insertions. Studies indicate that a reduction of the pulse reactivity insertion to a maximum $2.35 will not result in fuel damage for the useful life of the core.

The NSCR mixed cores are similar to other operational reactor systems such as those operating at General Atomic and Washington State University, and many safety analyses have been performed and much operating experience has been obtained using identical fuels. With this information and experience available, it was considered unnecessary to repeat any safety analyses that apply in general to these reactors but to elaborate only on those items which were unique to NSCR cores.

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B. Fuel Description and Safety Limits

The fuel in a mixed core is both standard TRIGA and FLIP fuel elements. The two types of elements are identical in geometry and differ physically only in U-235 enrichment, burnable poison content, and hydrogen-to-zirconium ratio. It is possible to visually distinguish the element types, however, by the modification on the upper tip of the FLIP fuel. Table I lists the principal design parameters of both FLIP and standard TRIGA 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 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 the 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 adiabatic fuel temperature to be allowed during transient conditions is set at 2100°F (1150°C) for FLIP fuel. Since standard nominally contains more hydrogen than FLIP fuel, its corresponding safety limit is reduced to 1830°F (1000°C). For steady-state operation (non-adiabatic case) fuel temperatures are dependent upon the 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. Since the Texas A&M TRIGA pool depth is approximately six feet greater, improved cooling characteristics are expected.

To evaluate power densities in NSCR cores safety analyses were performed for a number of water reflected cores with a 5 x 5 array of fuel bundles. The shim-safety rods all had fueled followers, the transient rod had a voided follower, and the regulating rod which was located away from the central region had no followers. Thus, the cores studied were comprised of a total of 98 fuel elements. As indicated previously both all standard and FLIP cores were studied as well as variations between these extremes (See Figure 11-1).

The power generated in each fuel element for the various cores was calculated for steady-state operation at 1 Mw. The maximum power generated in a single element for 23-element FLIP, 35-element FLIP, and 37-FLIP was 22.24 kw, 19.67 kw, and 19.48 kw respectively. Thus, as the FLIP region becomes larger, the ratio of the peak to average power decreases. In most cores, the maximum power generation was in the element located next to the withdrawn transient rod. This was

caused by the flux peaking in the water around the void which followed the transient rod. Notice that all of the maximum values were well below the value of 32 kw based on fuel cooling. Thus, for steady-state operation no cooling problems are expected. However, the loss of coolant accident imposes a more restrictive limitation on the power generations in a single element.

For pulsing calculations, a conservative pulse of 25 Mw-sec was considered. The results of these calculations showed that such a pulse in all cores shown in Figure 11-1 would produce maximum fuel temperatures less than the limiting safety system setting for FLIP and standard fuel.

C. Potential Hazards Considered

1. Fuel Bundle Rotation

To achieve symmetry with the central FLIP region it is often necessary to load both FLIP and standard elements in a single bundle. Since the fuel bundles can be physically rotated 180° and still fit the grid plate, the inadvertent rotation of a four-element fuel bundle containing one FLIP and three standard elements was considered. In the event of such a rotation, the FLIP element would be completely surrounded by standard fuel as shown in Figure 11-2. The analysis showed no appreciable increase in average power generated in either the FLIP element, or in the standard elements. The maximum fuel temperature that would be obtained during a 30 Mw-sec pulse with the rotated bundle is 1450°F (788°C) which is well below the safety limit. Table VII shows the results of the calculations of peak average power per cell during 1 Mw steady-state operation and the maximum fuel temperature due to a 30 Mw-sec pulse with and without rotation of one bundle. Obviously, no safety problem will be created due to inadvertent rotation of a fuel bundle.

TABLE VII

Fuel Bundle Rotation Study For
Maximum Power and Maximum Temperature

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

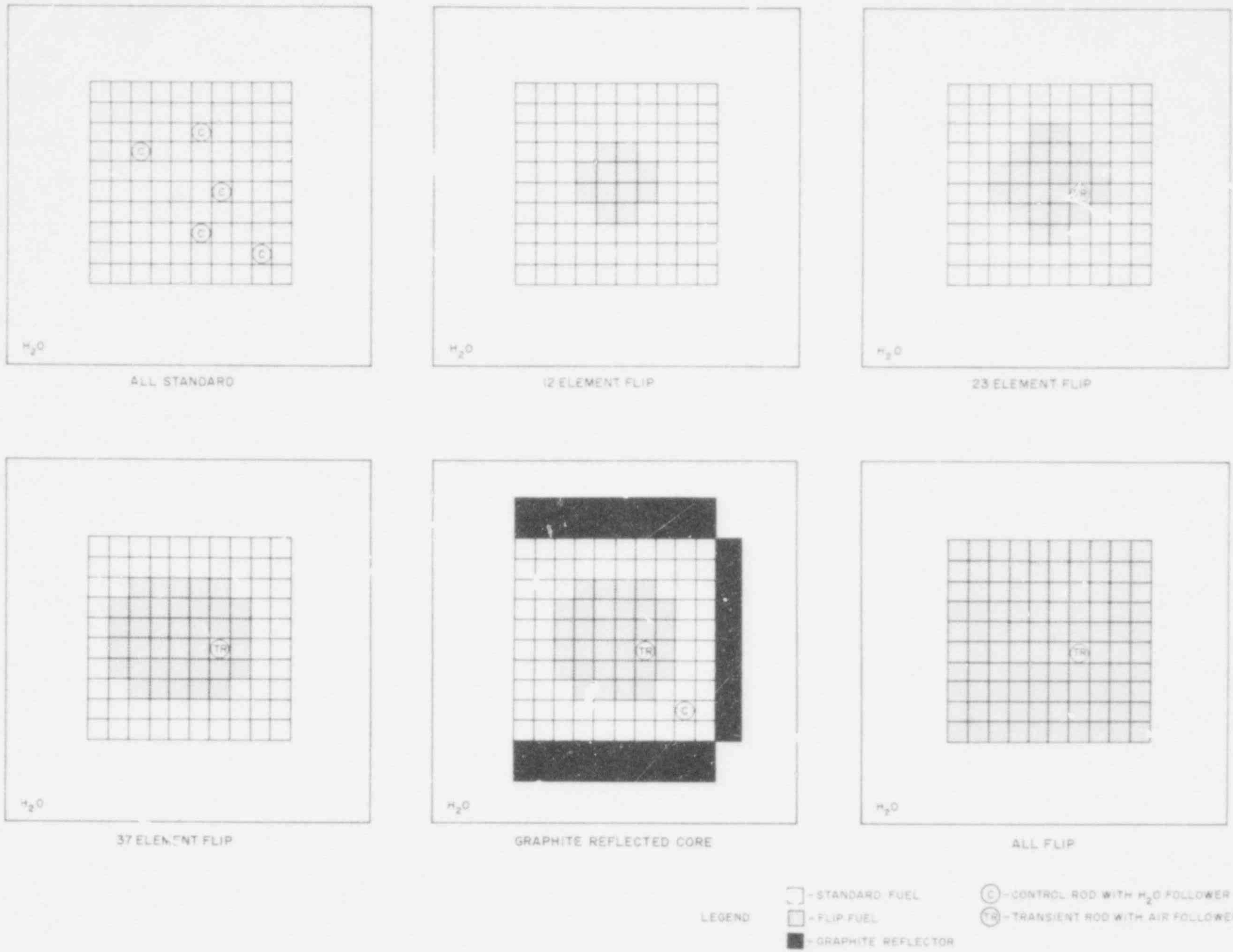

FIG. I-1 CORE CONFIGURATIONS STUDIED FOR THE TEXAS A&M TRIGA REACTOR

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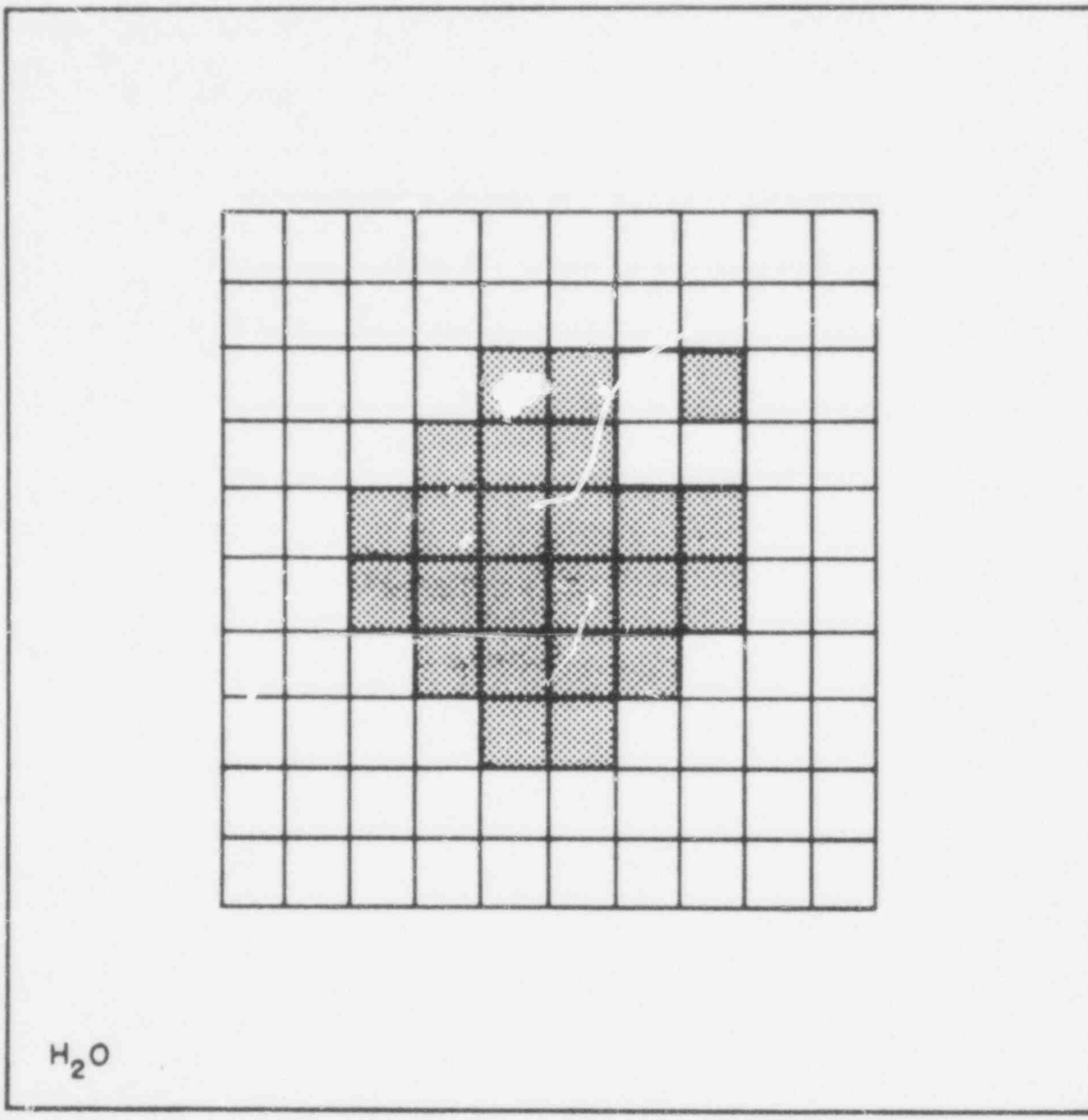

LEGEND

☐ - STANDARD FUEL ▩ - FLIP FUEL

FIG. 11-2 MIXED CORE WITH FUEL BUNDLE CONTAINING A SINGLE FLIP ROD ROTATED 180°

2. Control Rod Run-Out

The magnitude of the result of the withdrawal of all control rods at maximum speed was considered for the PRNC reactor. The magnitude of the effect of this accident is dependent primarily on the speed of rod withdrawal and the value of the temperature coefficient. Since proposals for the Texas A&M reactor included core loadings which varied from all standard to all FLIP TRIGA fuel the temperature coefficient had to be examined for these variations. Calculations for a variety of cores were performed by General Atomic with the results shown in Figure 3-31. As can be seen, the standard fuel with no poison had a temperature coefficient which was relatively constant with temperature. The addition of FLIP fuel resulted in temperature coefficient which increased linearly with temperature. Most of the core loadings anticipated for the NSCR will lie between 18-element FLIP and all FLIP curves. It is doubtful that an entire core would ever achieve the 3000 Mw/day burnup indicated for the lowest curve due to the planned loading sequence. Even for this limiting case, however, the magnitude of the temperature coefficient was large enough to allow safe operation of the reactor.

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

D. 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 fuel and 800°C in standard 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 case and for pulsing 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. This not only provides adequate flexibility but has another significant advantage. The thermocouple response in these locations is nearly independent of the amount of FLIP fuel in the core for both pulsing and steady state operations. 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 an all standard core, 35 and 59 FLIP element cores, and an all FLIP core. The maximum value obtained from the eight possible locations for each core is listed in Table VIII.

TABLE VIII

POWER RATIOS FOR THE THERMOCOUPLE LOCATIONS
ALLOWED BY TECHNICAL SPECIFICATIONS

Core	Pwr produced in max Pwr element/ Pwr produced in thermocouple element
All standard	1.06
35 FLIP	1.21
59 FLIP	1.19
ALL FLIP	1.18

It is the maximum value of this ratio that determines the LSSS. Note that the variation from the 35 FLIP element core to full FLIP is less than 3%.

For pulsing mode the variation in thermocouple response is determined by the ratio of the peak adiabatic temperature rise in the core to the adiabatic temperature rise at the thermocouple location. The maximum value of this ratio for the 8 possible locations resulting from the maximum insertion is listed in Table IX.

TABLE IX

RATIOS OF ADIABATIC TEMPERATURE INCREASES
FOR THE THERMOCOUPLE LOCATIONS ALLOWED
BY TECHNICAL SPECIFICATIONS

Core	Peak $\Delta T/\Delta T$ at thermocouple
All standard	1.61
35 FLIP	2.15
59 FLIP	2.14
All FLIP	2.13

Again note that the 35 FLIP elements core is limiting and that the variation is neglible in cores containing additional FLIP elements. Therefore, if the LSSS is established for a thermocouple located in one of the 8 specified locations for a 35 FLIP element core it will be safe for all other mixed cores.

2. The LSSS for Steady State Operation

Extensive fuel temperature measurements have been made using both standard and FLIP fuel. These results for a number of different IF's are shown in Figure 11-3. The wide differences that are observed are due to variations in the heat transfer coefficient of the element. The results that are used in the following calculations represent the highest temperatures that have been observed.

The details of the LSSS calculation are presented for the 35 FLIP element core. Normally, the maximum ambient fuel temperature that is observed is $37^{o}C$. Thus, the maximum temperature rise permitted in the core is

$$\hat{\Delta T}_{core} = 950^{o}C - 37^{o}C = 913^{o}C$$

The "safety limit less safety margin" used for mixed cores is that for FLIP fuel since the maximum power will occur in FLIP fuel. A value of $800^{o}C - 37^{o}C = 763^{o}C$ was used for the all standard core.

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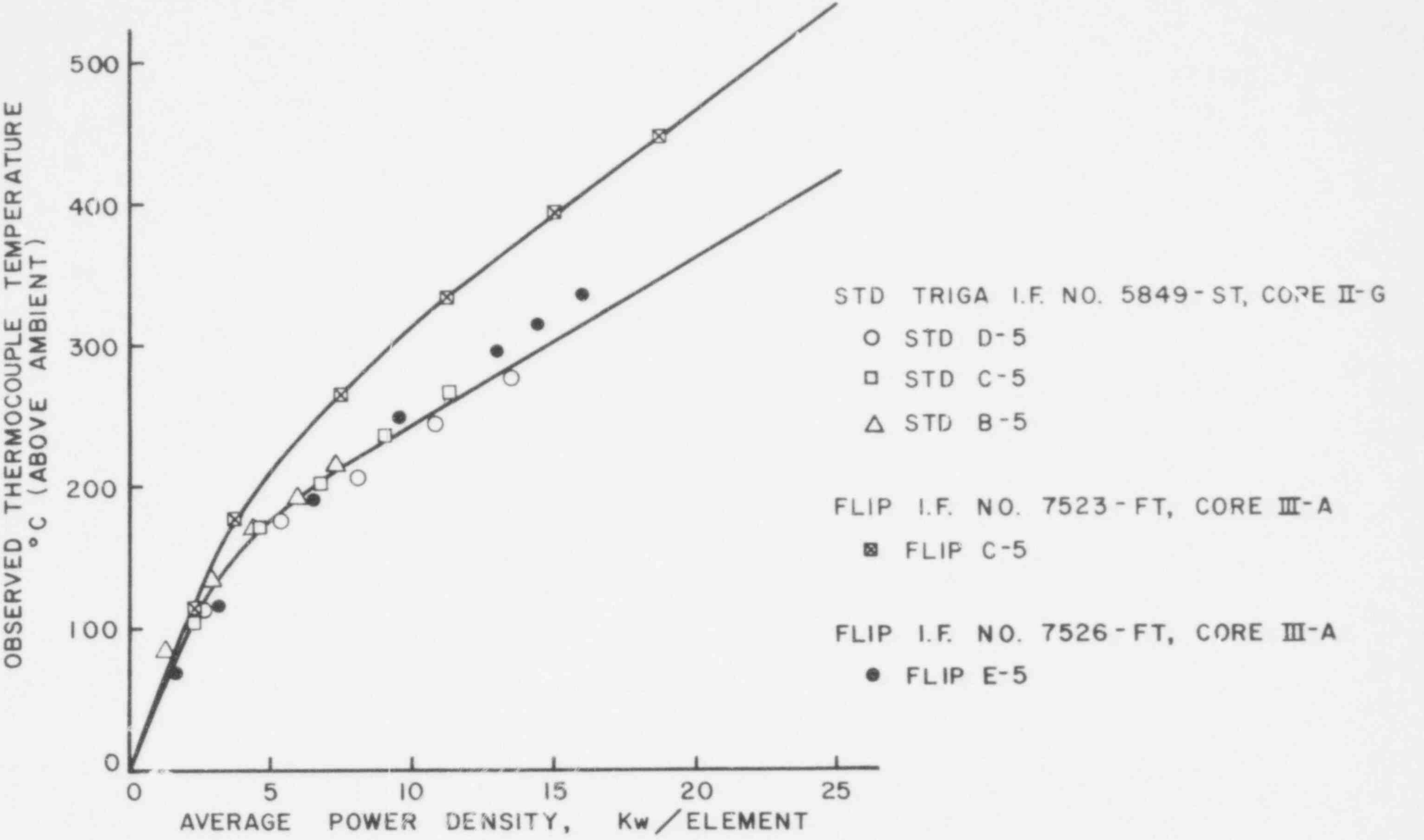

FIGURE II-3 CORE III-A STEADY STATE
FUEL ELEMENT TEMP. vs. POWER DENSITY

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In the 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, if an IF were located at the position of the maximum power element the thermocouple would read

$$\frac{913^{\circ}C}{1.16} = 787^{\circ}C \text{ above ambient.}$$

Using the conservative extrapolation of Figure 11-3, the power that fuel element would generate at $787^{\circ}C$ is 41.7 kw.

The power ratio between the element with the maximum power and the instrumented element was determined by Exterminator-2 to be 1.21 (Table III). Applying this ratio to the maximum power corresponding to $\Delta \hat{T}_{core}$ one obtains the power in the instrumented element. Thus,

$$P_{TC} = \frac{41.7 \text{ kw}}{1.21} = 34.4 \text{ kw}$$

From the extrapolation (Figure 11-3) the temperature in the thermocouple element at 34.4 kw is $678^{\circ}C$. This is the temperature above ambient observed by the thermocouple for a peak core temperature of $950^{\circ}C$. The LSSS is simply this value with the ambient temperature restored.

$$LSSS = 678 + 37 = 715^{\circ}C$$

LSSS values for other cores were obtained in a similar manner and are listed in Table X.

TABLE X

STEADY STATE VALUES OF THE LSSS FOR DIFFERENT CORE CONFIGURATIONS

Core	Steady State LSSS
All standard	$667^{\circ}C$
35 FLIP	$715^{\circ}C$
59 FLIP	$724^{\circ}C$
All FLIP	$728^{\circ}C$

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For steady state operation the all standard core has the lowest LSSS which is due to the lower safety limit. Thus, if a value of 667°C were chosen for the LSSS, it would prevent peak core temperatures from exceeding 950°C for standard cores, full FLIP cores, and for all mixed cores that comply with the Technical Specifications.

3. The LSSS and 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, a 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 mixed cores or 800°C for a standard core. This would reduce the total energy produced in the element by clipping the pulse "tail." Since establishing the LSSS on this basis does not limit steady state operations this conservative approach is considered to be the prudent thing to do.

The minimum LSSS for pulsing was determined for the thermocouple location adjacent to the central 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}. It establishes the relation between the peak core adiabatic temperature rise and the adiabatic thermocouple temperature rise. A sample calculation of the pulsing LSSS is presented for the 35 FLIP element core. Values of the LSSS for additional cores were obtained in the same manner.

The limiting peak core temperature rise, $\Delta\hat{T}_{core}$, for a peak core temperature of 950°C and an ambient temperature of 37°C is

$$\Delta\hat{T}_{core} = \hat{T}_{core} - 37^{o}C = 913^{o}C$$

The energy density required to raise the FLIP fuel to 913°C is 3580 watt-sec/cc. Applying the power ratio, the energy density at the thermocouple becomes

$$E_{TC} = \frac{3580 \text{ w.sec/cc}}{2.85} = 1256 \text{ w.sec/cc}$$

The corresponding adiabatic temperature rise ΔT_{TC} at the thermocouple is 438°C. The LSSS is obtained by determining the observed thermocouple reading. The observed reading differs from the adiabatic value due to heat flow during the first few seconds after the pulse.

The adiabatic temperature distribution corresponds to the power density distribution where the surface value is considerably higher than that at the element center. After the pulse the heat not only flows out of the element but initially some will flow towards the lower, central temperatures. This will cause an increase in the indicated thermocouple reading. The maximum thermocouple reading must therefore be used. The accepted value for the ratio of the peak temperature reading to the

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adiabatic value is 1.13. The 1.13 heat flow factor and ambient temperature applied to the thermocouple rise established an LSSS of

$$\text{LSSS} = T_{TC} = (\Delta T_{TC} \times 1.13) + 37$$

$$= (438 \times 1.13) + 37 = 532^{\circ}\text{C}$$

LSSS values for standard, mixed, and all FLIP cores are presented in Table XI.

TABLE XI

PULSING VALUES OF THE LSSS FOR DIFFERENT CORE CONFIGURATIONS

Core	Pulsing LSSS
All Standard	591°C
35 FLIP	532°C
59 FLIP	537°C
All FLIP	544°C

The LSSS values obtained in this manner are considerably lower than those derived for the steady state mode. Using the limiting value which is for the 35 FLIP element core and reducing it to easily remembered numbers results in an LSSS of 525°C or 975°F. This will result in a minimum safety margin for steady state operation of 368°C for an all standard core and 460°C for a 35 FLIP element core. Thus, the recommended LSSS remains highly conservative.

E. Evaluation of the Maximum Allowable Reactivity Insertion for Pulsing

Considerable pulsing experience has been obtained with several standard cores and a mixed core containing 35 FLIP elements. It is well known that pulsing limits for standard cores are much higher than for those that contain FLIP fuel. This section will therefore be directed toward mixed and FLIP cores and any values obtained will be conservative for all standard cores.

It has been observed that the pulsing temperatures observed at a given core location and thermocouple orientation vary little between different thermocouples. The observed values of the temperature rise due to pulsing the 35 FLIP element core is shown in Figure 11-4. A straight line, determined by a least squares fit of these data is represented by

$$\Delta T(^{\circ}\text{F}) = 322 \frac{\Delta k}{k} (\$) - 99.5^{\circ}\text{F}$$

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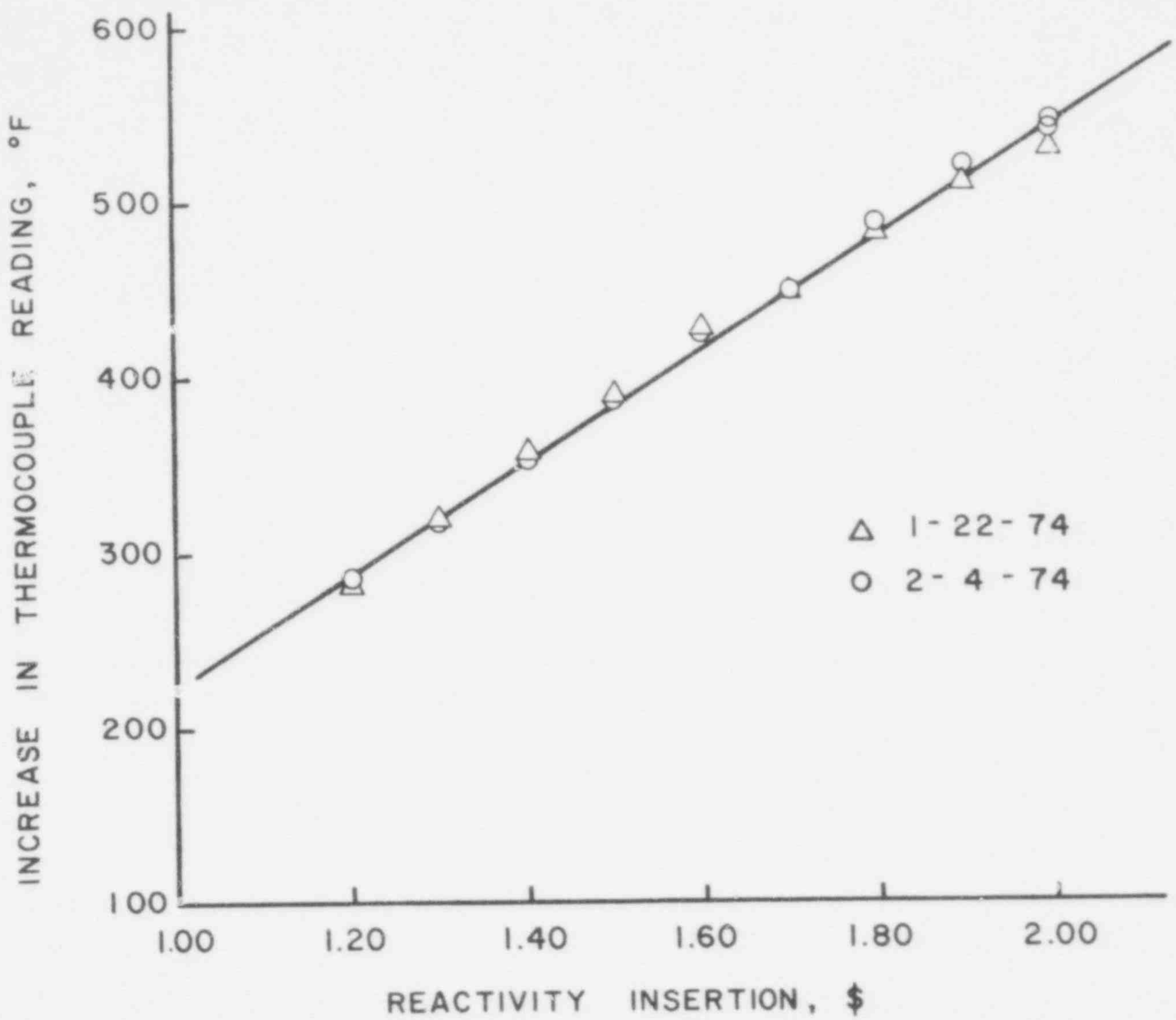

FIGURE 11-4 PULSING DATA FOR A 35 FLIP ELEMENT CORE

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A linear extrapolation of these measured data will yield conservative results since the heat capacity of the fuel increases with fuel temperature and there will be some cooling occurring during the pulse. Thus, the extrapolated temperatures will not actually be attained. Converting to Centegrade temperature,

$$\Delta T(^{o}C) = 179 \frac{\Delta k}{k} (\$) - 55.3^{o}C$$

Due to the heat flow towards the center of the element after a pulse, this value over-estimates the adiabatic temperature by 13%. Thus, the adiabatic temperature is

$$\Delta T_a (^{o}C) = \frac{\Delta T(^{o}C)}{1.13} = 158 \frac{\Delta k}{k} (\$) - 49.0^{o}C$$

The location and orientation of the thermocouple which obtained the pulsing data was such that the ratio between the maximum temperature rise in the core and temperature rise at the thermocouple location was 2.20. Thus the maximum core temperature for various pulse insertions is given by

$$\hat{\Delta T} (^{o}C) = \Delta T_a (^{o}C) \times 2.20 = 348 \frac{\Delta k}{k} (\$) - 108^{o}C$$

To obtain the maximum fuel temperature the initial fuel temperature must be added to the temperature rise. The maximum observed ambient core temperature is $37^{o}C$. Thus,

$$\hat{T}(^{o}C) = 348 \frac{\Delta k}{k} (\$) - 71^{o}C$$

The pulse insertion that yields a maximum core temperature of $950^{o}C$ is therefore,

$$950^{o}C = 348 \frac{\Delta k}{k} (\$) - 71^{o}C$$

$$\frac{\Delta k}{k} = \$2.93$$

1. Effect of Amount of FLIP Fuel

Since it is convenient to have a single insertion limit for all allowed cores, the effect of increasing the amount of FLIP fuel was examined. The reactivity that can be inserted to yield a peak core temperature of $950^{o}C$ as a function of number of FLIP elements in a 98 element core has been calculated (See Appendix I and Figure 11-5). It is found that the insertion for a full FLIP core is 96.3% as large as that for a 35 element FLIP core to achieve the same peak core temperature. If the insertion limit is reduced by this amount it will then be acceptable for all mixed cores from 35 elements to full FLIP.

Therefore, the limit becomes

$$\$2.93 \times .963 = \$2.82$$

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As mentioned earlier, this value is well below pulsing limits for standard cores which are as high as \$3 and \$4.

2. Effect of Burnup

Correcting for changes in neutron lifetime and the temperature coefficient due to burnup results in an allowable insertion which decreases with core utilization. Calculations were performed to determine the effect of these changing parameters resulting from burnup. Calculations were performed for full FLIP only since the effect would be greatest for this case.

The insertions calculated were low compared to the extrapolation of experimental results, but the ratio is considered valid. The values obtained for the insertions which yielded 950°C peak core temperatures are \$2.36 for the beginning-of-life and \$2.08 for the end-of-life (8.2 Mw-yrs). The ratio is $\frac{2.08}{2.36} = .881$. Thus, after 8.2 Mw-yrs of operation the maximum allowable insertion will be

$$\$2.82 \times .881 = \$2.48$$

A maximum allowable insertion of \$2.35 is allowed by technical specifications and is well below the above value for the lifetime of the core. The safety margin exceeds 200°C in FLIP fuel for a \$2.35 pulse insertion and is considerably greater than this for standard fuel. These margins allow amply for uncertainties due to the accuracy of the measurement of extrapolation of the measured data.

F. Accidental Pulse at Full Power for Standard TRIGA Cores

In order for this operation to take place, the reactor operator would have to deliberately violate approved procedures, the technical specifications, and a series of reactor safety interlocks and scrams. The reactivity insertion would take place as is outlined below. The reactor is loaded to \$7.00 excess and the transient rod is set for a maximum \$3.00 transient. The operator slowly withdraws all the control rods (except the transient rod) until all the rods are completely out and the reactor is operating at a high steady-state power. From data on steady-state power and fuel temperatures as a function of compensated reactivity (measured at the prototype TRIGA Mark III reactor), the steady-state power and fuel temperatures for a reactivity of \$4.00 (that is \$7.00 - \$3.00 = \$4.00) is 1.4 Mw, 405°C peak fuel temperature, and 237°C average fuel temperature.

The next step in this incident takes place as the operator inserts the full worth of the transient rod by ejecting it from the hot reactor operating at 1.4 Mw. This prompt insertion of \$3.00 results in an average temperature rise in the core of 233°C and a peak temperature rise

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of 399^{o}C. Thus, the average and peak temperatures at the conclusion of the pulse at 1.4 Mw are 470^o and 804^{o}C, respectively, well below 1000^{o}C, the rupture temperature of the cladding. The equilibrium hydrogen pressure over the $ZrH_{1.7}$ fuel resulting from this type of reactivity insertion would be about 60 psi, well below the rupture pressure of the fuel element clad (i.e., 1800 psi).

G. Accidental Pulse at Full Power for Mixed and FLIP TRIGA Cores

It is necessary to examine this situation in spite of the interlocks which will prevent this from happening. The calculations were performed by General Atomic using the BLOOST 2 code assuming adiabatic processes. The details of the calculations are presented in Appendix I. The experimental parameters and core power distributions were supplied by Texas A&M. The results desired were the reactivity insertion from power that would produce a peak core temperature of 950^{o}C. For valid comparisons the values for pulsing from 300 watts were also calculated. Figure 11-5 shows the results obtained as a function of number of FLIP elements in the core at the beginning-of-life. The end-of-life case was calculated only for a full FLIP core. As can be seen for every case, if the reactor is pulsed from 1 Mw considerably more reactivity is required to obtain 950^{o}C. Therefore, it is pulsing from low power that limits the amount of insertion and not the "accident" situation.

H. 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. 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 standard fuel can tolerate in air without damage to the clad and subsequent release of fission products is 1650^{o}F (900^{o}C). For FLIP fuel, this value is 1720^{o}F (940^{o}C).

b. This temperature will not be exceeded under the conditions of coolant loss if the maximum power density in an element is equal to or less than 21 kw for standard fuel and 23 kw for FLIP fuel 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 25 kw/element for standard fuel and 28 kw/element for FLIP fuel will not cause element damage in the event of loss of coolant. The calculations that produced the above results are presented in Appendix II.

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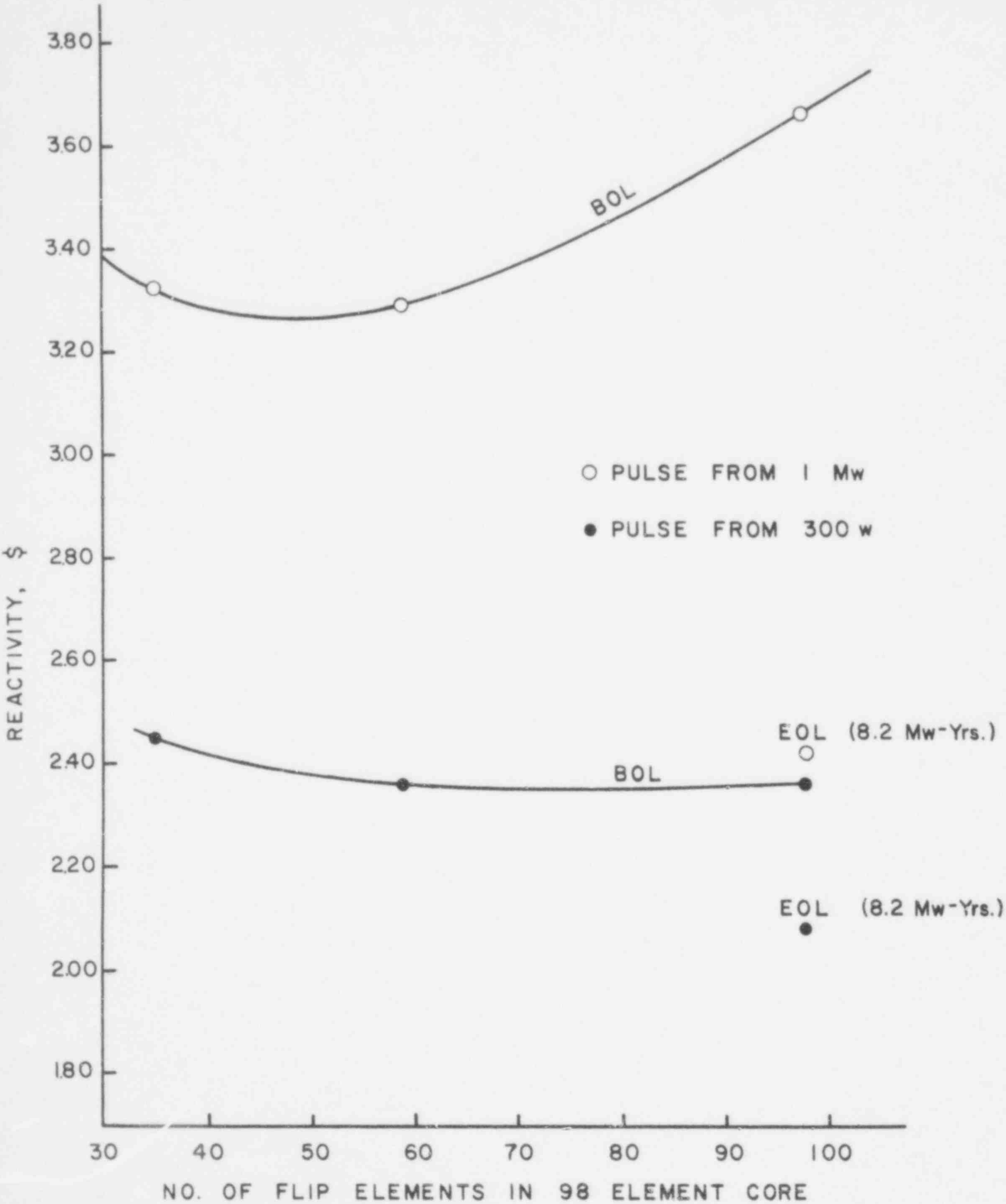

FIGURE 11-5 PULSE TO PRODUCE
950°C PEAK TEMPERATURE

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I. 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.

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 11-6 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 3.65×10^{-4} μc/cm³. Within 24 hours this value would decay to 8.34×10^{-5} μc/cm³. These soluble fission products would be removed by the demineralizer and disposed of as liquid wastes.

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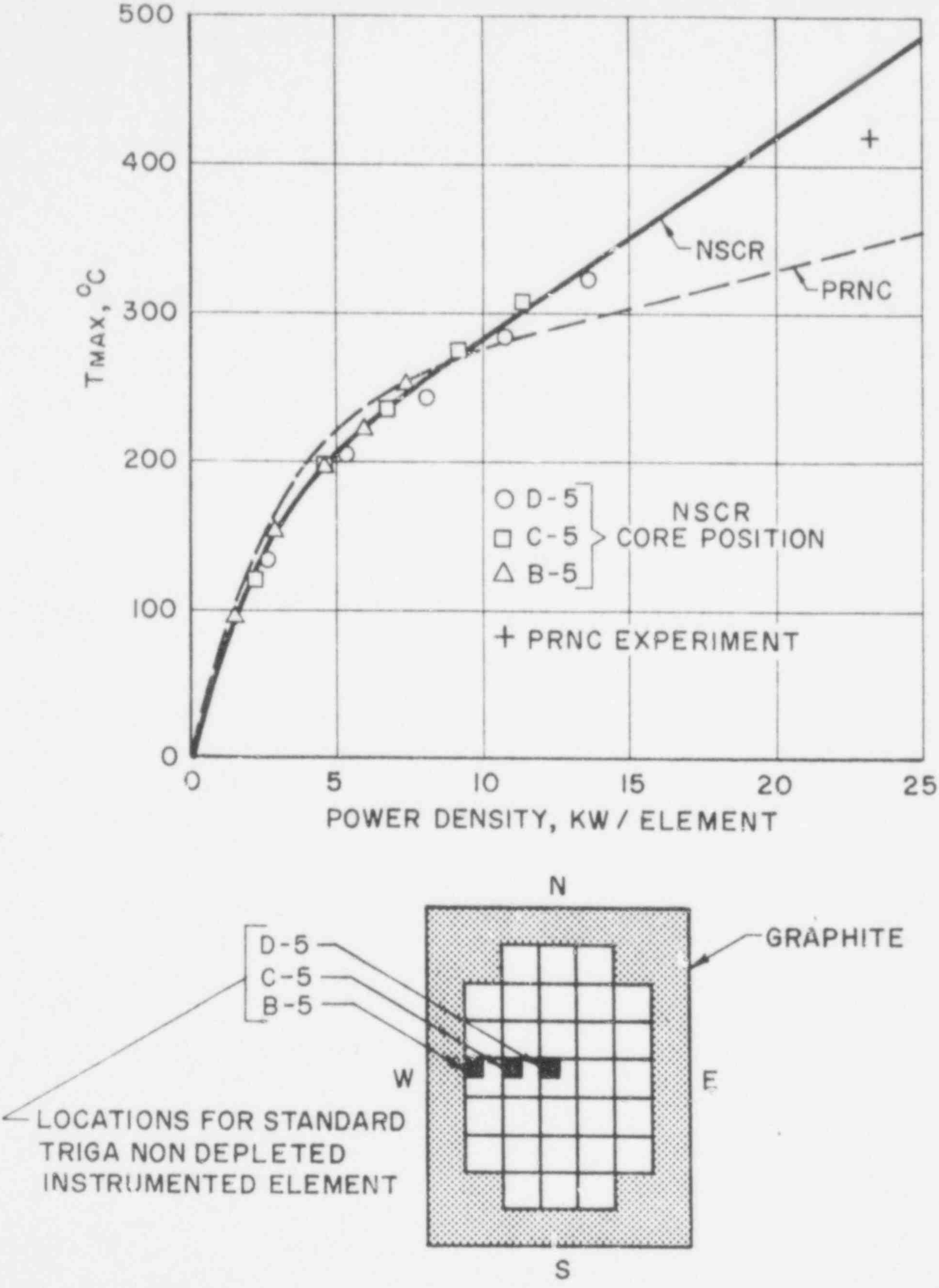

STANDARD TRIGA CORE II-G
(10% DEPLETED)

FIG. 11-6 STEADY STATE FUEL TEMPERATURE AS A FUNCTION OF POWER GENERATION

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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.

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

	WBGD*	Thyroid Dose
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

	WBGD	Thyroid Dose
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)

	WBGD	Thyroid Dose
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:

	WBGD	Thyroid Dose
Pool Water Remaining	1.75 mr	--
Pool Water Drained	4.2 mr	49 r

*WBGD - Whole Body Gamma Dose

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J. 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.

K. 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.

L. 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.

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Appendix I

THE PULSING ACCIDENT IN MIXED AND FLIP CORES

The scope of this study performed by General Atomic was defined as:

1) Determining the size of the pulse producing 950°C peak temperature as a function of the number of FLIP elements in the core and the burnup by assuming adiabatic processes, and
2) If the results of 1) do not allow pulses of an acceptable magnitude, to refine the calculations by including the effects of heat transfer which would give a more realistic assessment of the effects of the reactivity addition.

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

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

The calculations were made using BLOOST 2. The input parameters that were common to all problems were:

No. of elements = 98

Delayed neutron fraction = 0.007

Fuel specific heat = 720.0 + 1.48 T (w-sec/°C-element)

Water specific heat = 860 (w-sec/°C-element)

Fuel thermal resistance = 10000 (°C/Mw/element)

Coolant thermal resistance = 1175 (°C/Mw/element)

Initial average fuel temperature at 1Mw = 238°C

Initial average coolant temperature at 1Mw = 45°C

Pulse insertion time = 100 msec

Scram delay time = 15 msec

Rod drop time = 0.985 sec

For the pulses from 300w it was assumed that the system had an initial temperature of 25°C and that there was no scram.

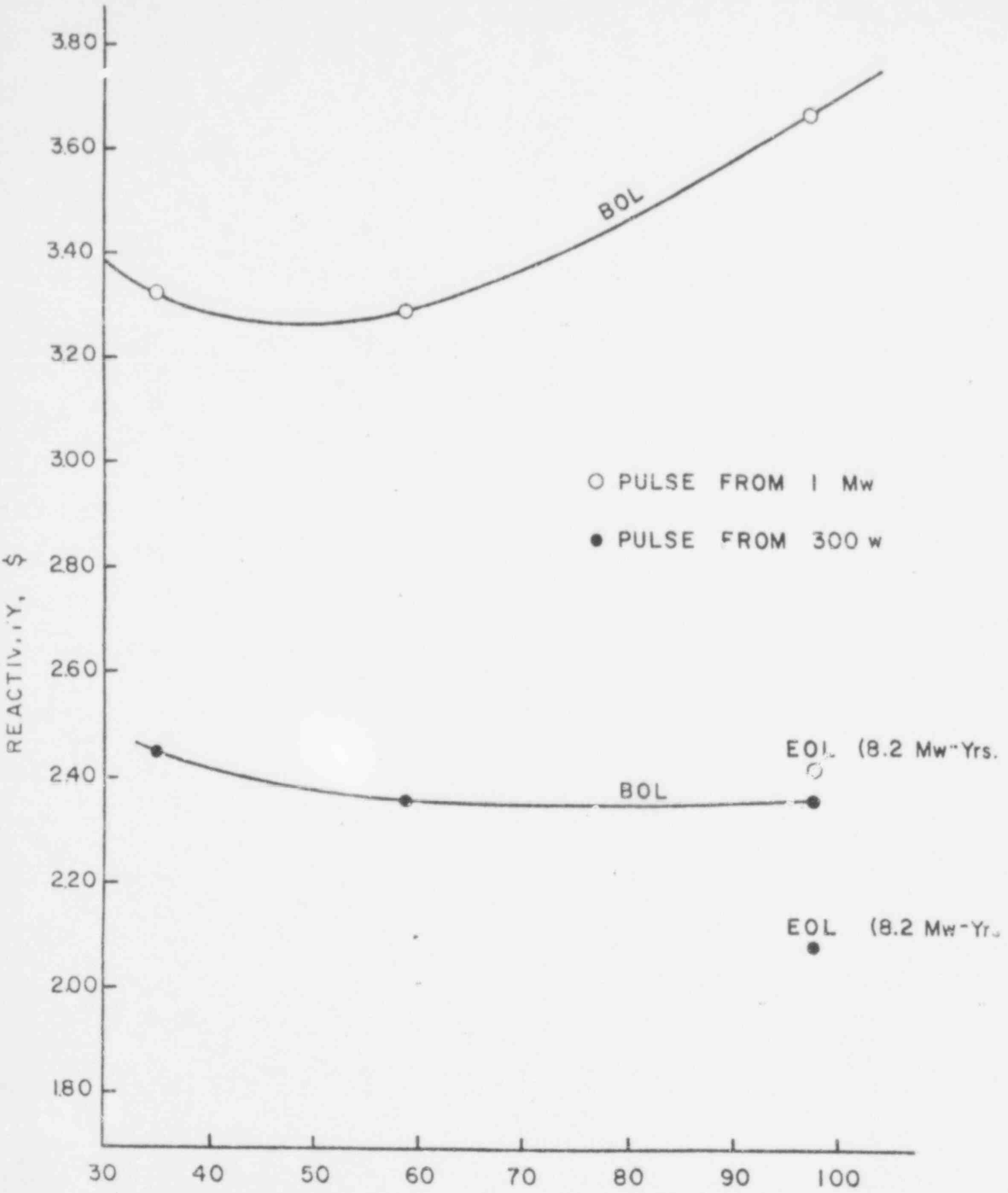

FIGURE 1 PULSE TO PRODUCE 950°C PEAK TEMPERATURE

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In Figure 2 there is shown the transient rod integral worth as measured. The "ramp" table was constructed by assuming this rod was uniformly accelerated from its initial position to 605 units (i.e., \$3.25). The initial position was chosen to provide the desired worth of the pulse. The \$3.25 position for the upper end of the insertion was chosen simply to sharpen the pulse as the integral worth curve flattens out drastically above that point. Using the full out position as the final point would tend to clip the pulse although that is a trivial consideration here. The ramp as a function of time is shown in Figure 3 for \$2.00, \$2.25, and \$2.75 pulse. These curves are plots of the ramp table.

The scram table was constructed by assuming that the total excess available was \$7.00, that \$3.50 was held down by the 1 Mw temperature, that the total worth of the rods available for the scram was \$12.00, that the shape of the rod worth is represented by the General Atomic "standard" shape, and that the rods fall with uniform acceleration. A plot of the scram table is given in Figure 4.

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

Peaking factors for the power distribution were:

35 FLIP elements: axial = 1.36; radial = 2.00; cell = 1.95

59 FLIP elements: axial = 1.36; radial = 1.91; cell = 1.95

98 FLIP elements: axial = 1.36; radial = 1.56; cell = 1.95

Prompt neutron lifetimes were estima[illegible] from the following calculated data:

BOL - 18 FLIP/54 standard	[illegible].0 µsec
BOL - All FLIP	ℓ 17.5 µsec
BOL - All FLIP	ℓ = 21.0 µsec

Linear interpolation based on the fraction of FLIP elements in the core gives for the cores studied:

35 FLIP (35.7% FLIP) BOL ℓ = 26.5 µsec

59 FLIP (60.2% FLIP) BOL ℓ = 23.1 µsec

[illegible]LIP (100% FLIP) BOL ℓ = 17.5 µsec

[illegible]LIP (100% FLIP) EOL ℓ = 21.0 µsec

Figure 5 shows the calculated prompt negative temperature coefficients for the 18 rod FLIP at BOL and the full FLIP at BOL and EOL. The BOL coefficient for the mixed cores between 18 rod (which corresponds to 25% FLIP) and full FLIP was estimated by making a linear interpolation between the two curves with the ratio of FLIP to total as the porportionality constant. In Figure 6 the integral temperature coefficients, as used in BLOOST 2, are plotted for the four core configurations considered.

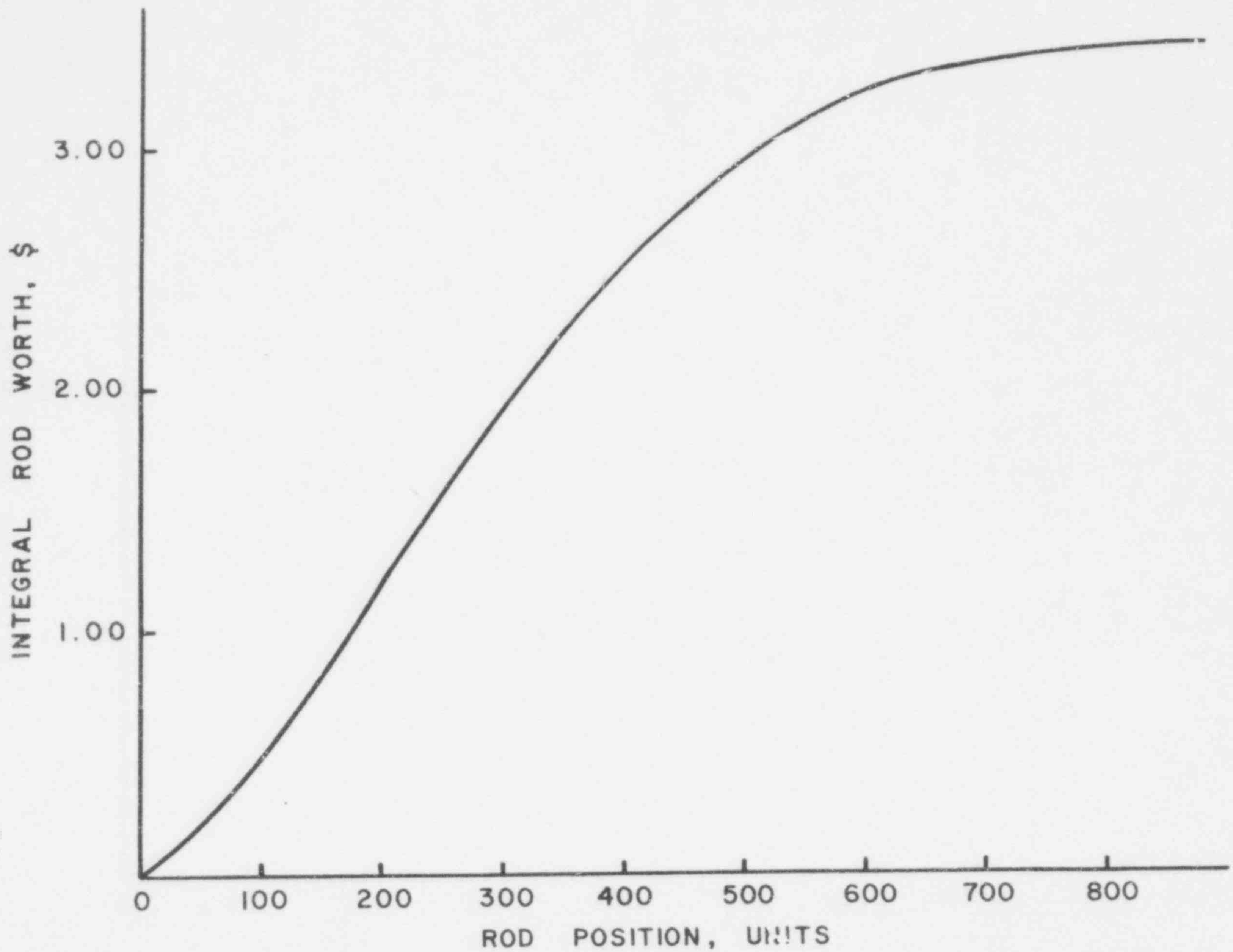

FIGURE 2 TRANSIENT ROD INTEGRAL WORTH

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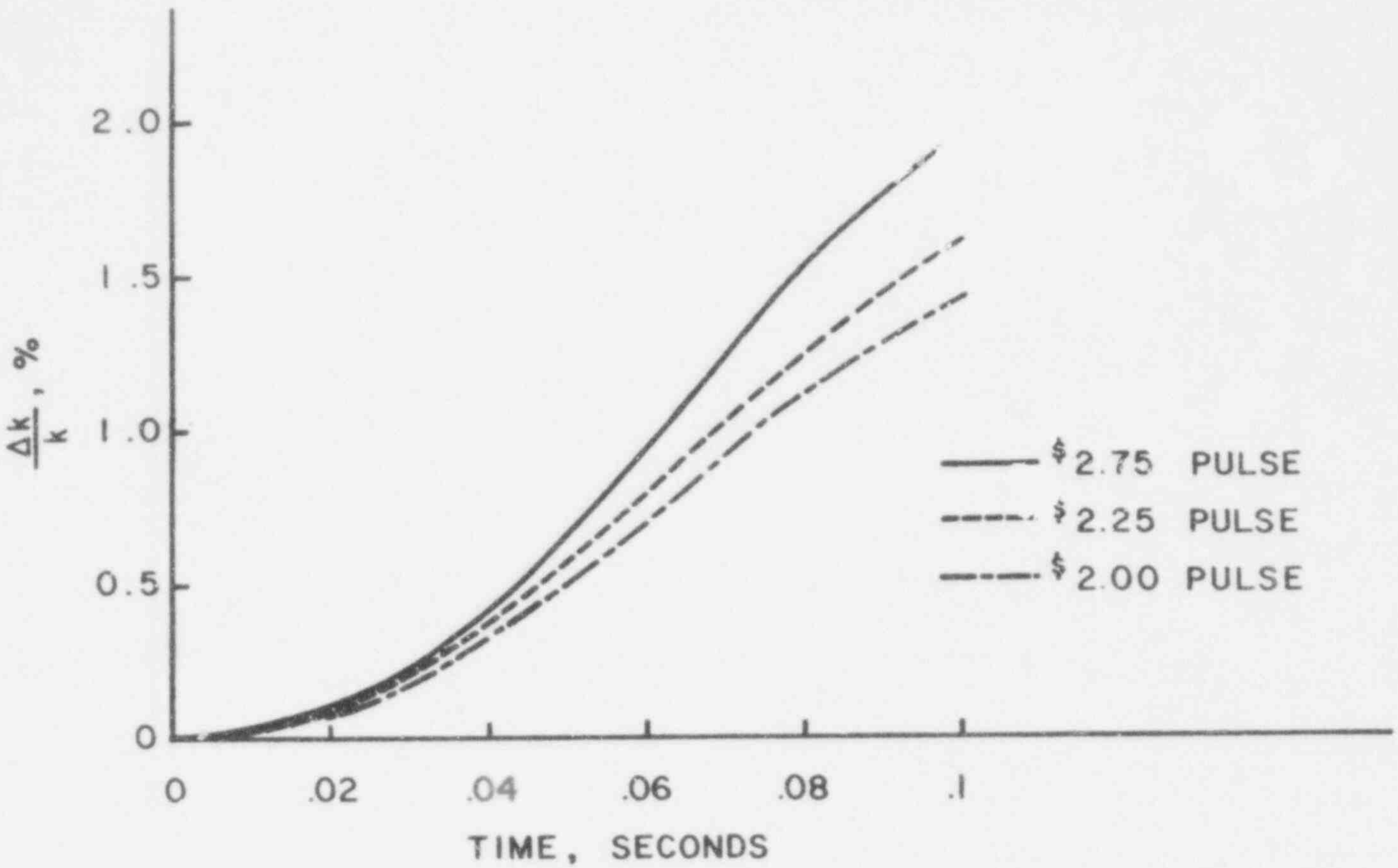

FIGURE 3 TIME DEPENDENT PULSE REACTIVITY INSERTION USED TO OBTAIN RAMP TABLE

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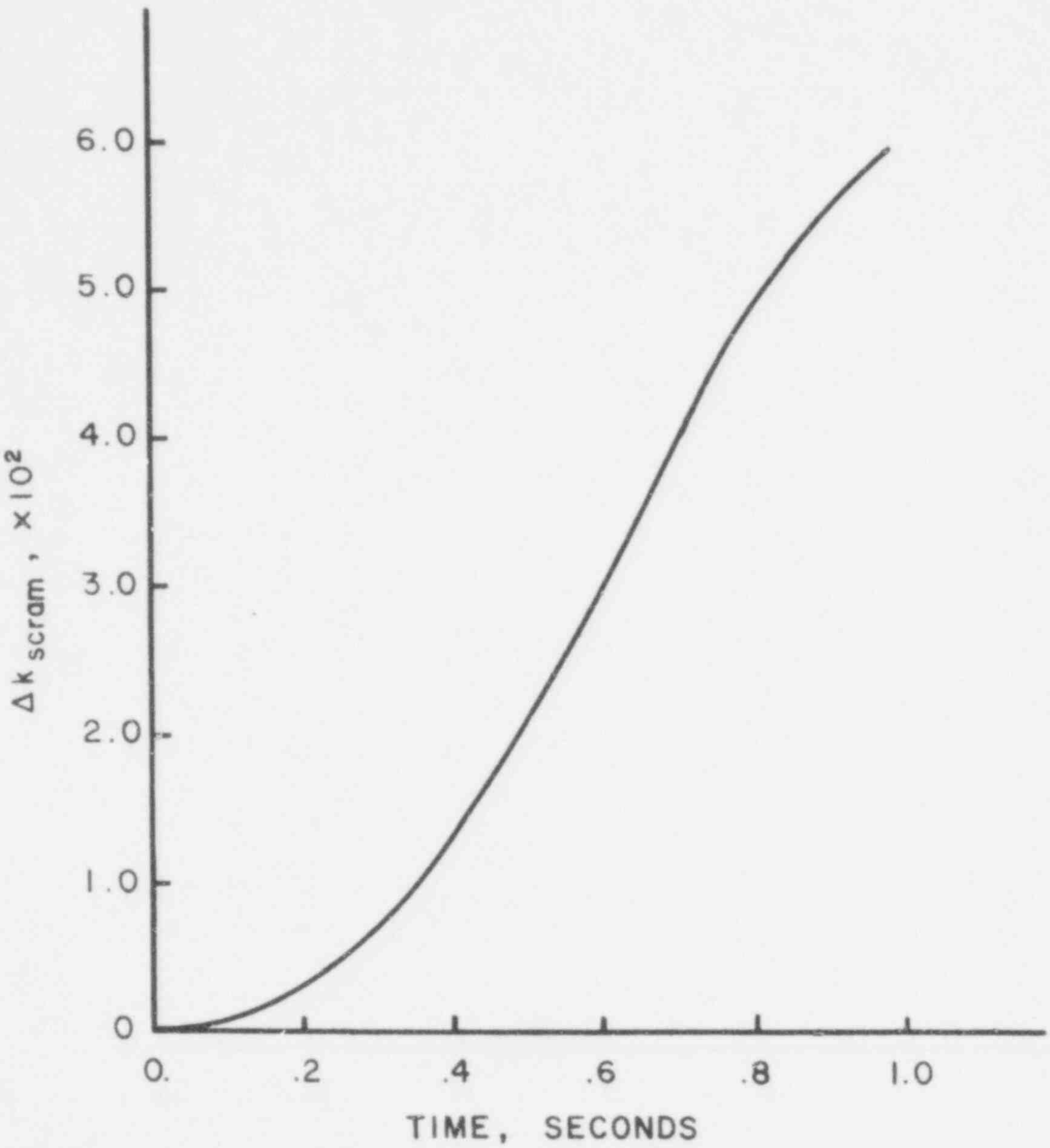

FIGURE 4 TIME DEPENDENT REACTIVITY INSERTION USED TO GENERATE SCRAM TABLE

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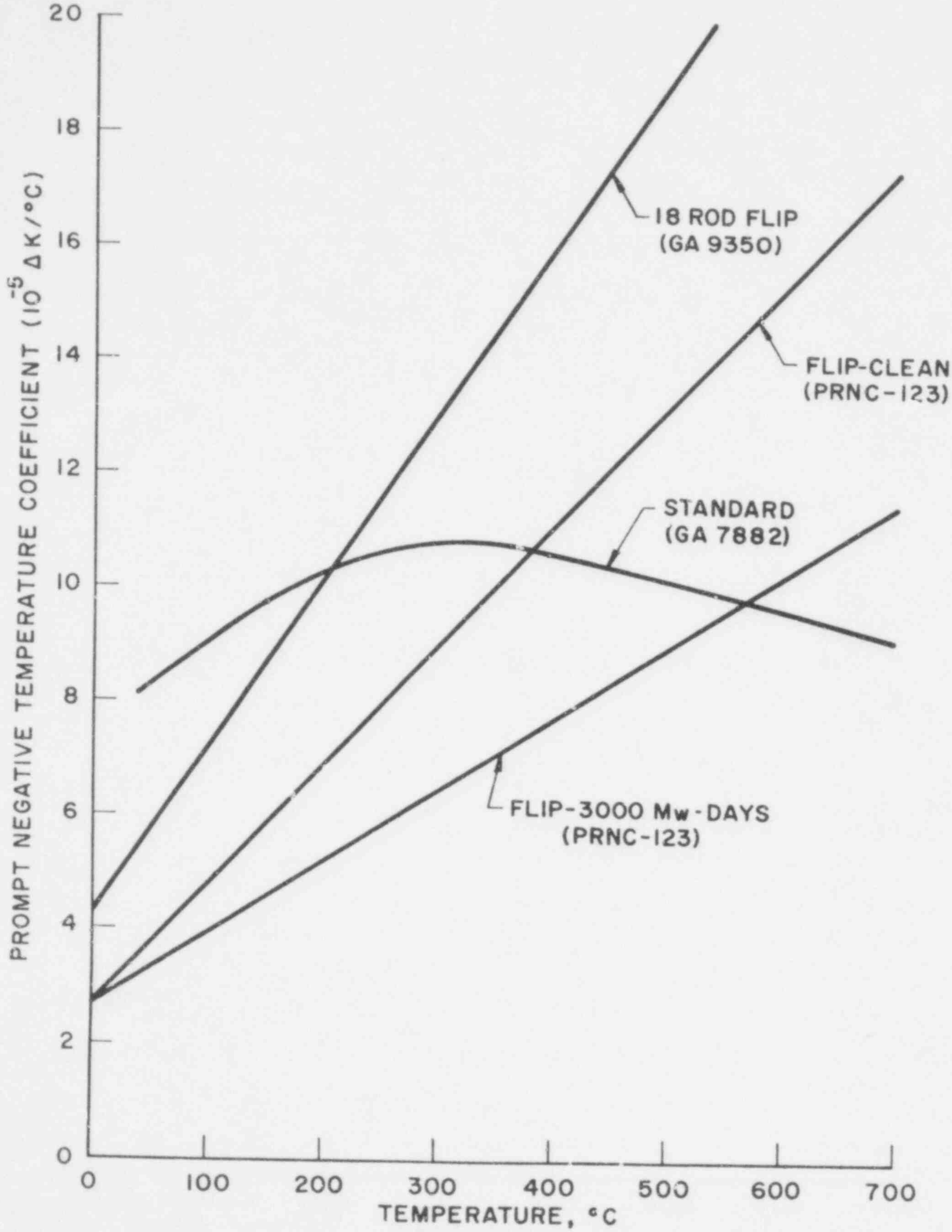

FIGURE 5 TEMPERATURE COEFFICIENTS OF TRIGA FUELS

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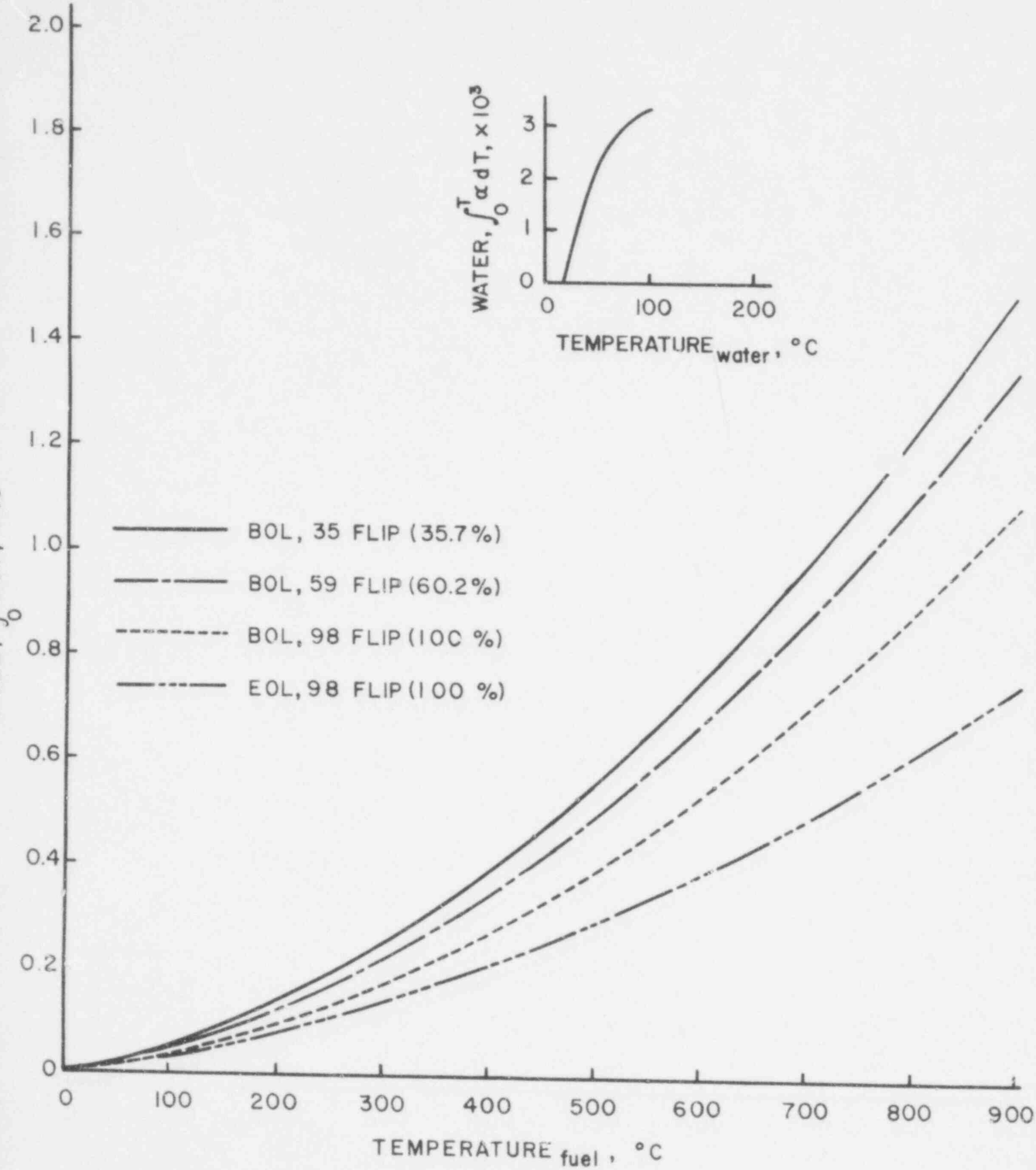

FIGURE 6 INTEGRAL TEMPERATURE COEFFICIENT

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The integral water temperature coefficient is the standard FLIP coefficient and is shown in the inset to Figure 6. Since the BLOOST 2 does not calculate the peak temperature properly when the pulse is from power, the peak temperature for these cases was calculated by hand. This was done by the following process:

1) Determine the average power density in the "hottest" element at 1Mw.
2) From this power density determine the steady state temperature at the periphery of the fuel at the axial centerline.
3) Calculate the energy content at the point at 1Mw.
4) Calculate the energy added at that point in the pulse.
5) Determine the adiabatic temperature that would result from that energy content.

The peak edge temperature in the fuel before the pulse is:

35 FLIP Max SS edge temp = $165^{o}C$ Energy content = $0.139 \frac{Mw}{elem}$

59 FLIP Max SS edge temp = $170^{o}C$ Energy content = $0.144 \frac{Mw}{elem}$

98 FLIP Max SS edge temp = $180^{o}C$ Energy content = $0.154 \frac{Mw}{elem}$

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

PULSE FROM 1 Mw

Prob #	$	No. FLIP	Burnup	P_{max}	Prompt Energy	$\bar{T}$	$\hat{T}$
11	2.00	35	BOL	677Mw	12.4Mw	345	667.3
12	2.75	35	BOL	1383Mw	18.4Mw	394	844.1
13	2.00	59	BOL	817Mw	13.5Mw	354	685.9
14	2.75	59	BOL	1579Mw	19.6Mw	403	855.2
15	2.00	98	BOL	1110Mw	15.8Mw	373	673.7
16	2.25	98	BOL	1420Mw	18.1Mw	392	728.9
17	2.00	98	EOL	1502Mw	22.4Mw	425	826.2
18	2.25	98	EOL	1951Mw	26.4Mw	455	910.0

PULSE FROM 300W

Prob #	\$	No. FLIP	Burnup	P_{max}	Prompt Energy	Energy to 2s	$\bar{T}$	$\hat{T}$
21	2.00	35	BOL	1263Mw	17.7Mw	24.0	225	767
22	2.75	35	BOL	3625Mw	25.5Mw	34.4	324	1057
23	2.00	59	BOL	1619Mw	19.6Mw	26.5	244	802
24	2.75	59	BOL	4631Mw	31.6Mw	38.0	349	1094
25	2.00	98	BOL	2645Mw	24.1Mw	32.5	286	802
26	2.25	98	BOL	4016Mw	28.9Mw	36.8	328	907
27	2.00	98	EOL	2561Mw	29.0Mw	39.4	329	911
28	2.25	98	EOL	3905Mw	35.0Mw	45.7	379	1031

To find the values of the reactivity insertion that would result in peak fuel temperatures of 950°C, an interpolated value was found from the data in the tables above. These values are given in the next table.

REACTIVITY TO GIVE 950°C PEAK TEMPERATURE

No. FLIP	Burnup	Pulse from 300w	Pulse from 1Mw
35	BOL	\$2.45	\$3.32
59	BOL	2.35	3.29
98	BOL	2.36	3.65
98	EOL	2.08	2.42

It was felt that these results were sufficient to allow effective operation so it was not necessary to do the second part of the program which would consider heat flow from the fuel after the pulse.

Appendix II

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 1.7 for standard fuel and 1.6 for FLIP 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 $600^{\circ}C$.

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

$$p_h = \exp \{1.76 + 10.3014x - 19740.37/(T_k)\}$$

where x = ratio of hydrogen atoms to zirconium atoms

and T_k = fuel temperature ($^{\circ}K$).

The pressure exerted by the fission product gases is given by

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

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 $^{\circ}K$),

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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

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

where T_o = maximum fuel temperature in the element during normal operation (^oK).

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

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

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.

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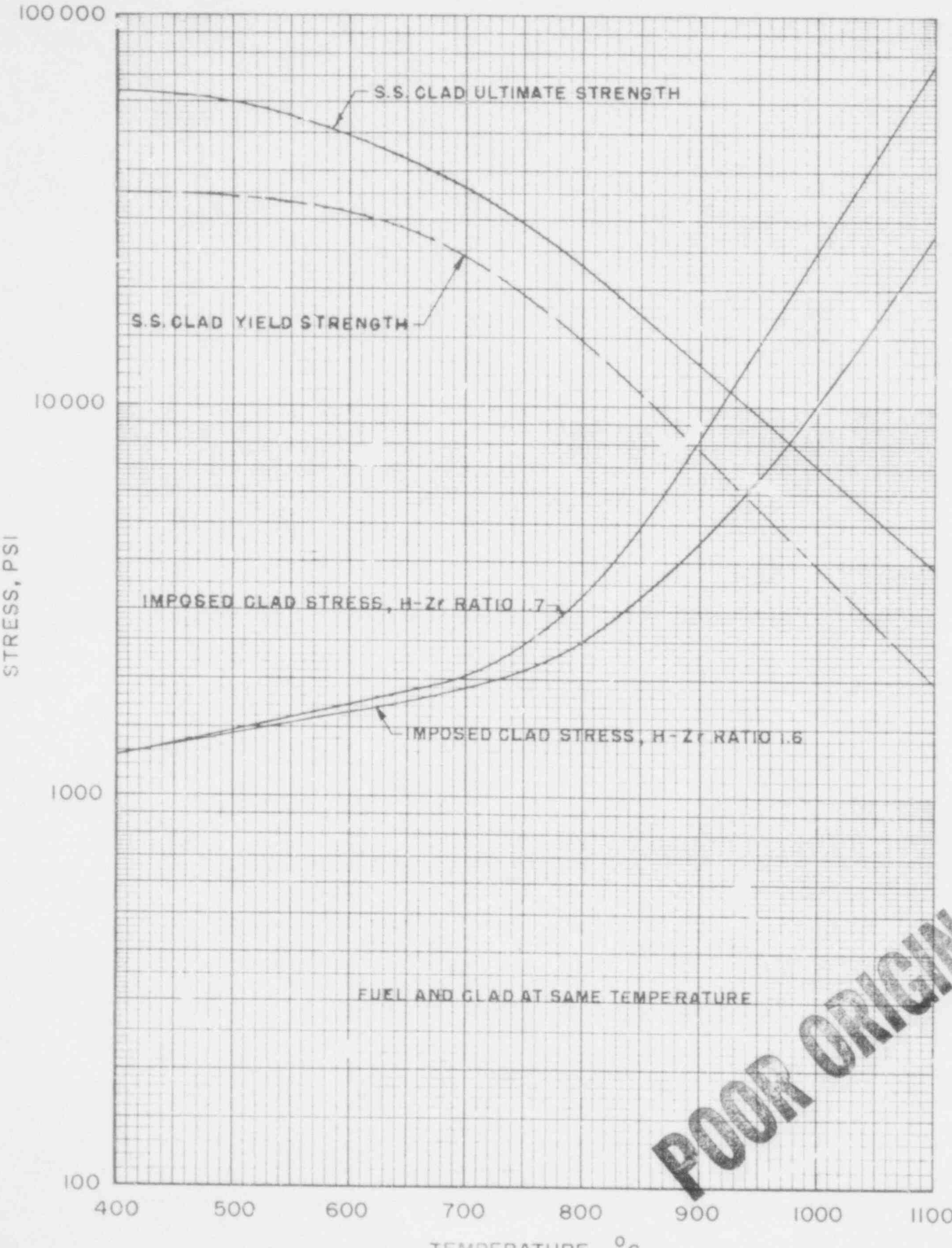

FIGURE 1 STRENGTH AND APPLIED STRESS AS A FUNCTION OF TEMPERATURE FOR 1.7 AND 1.6 H-Zr TRIGA FUEL

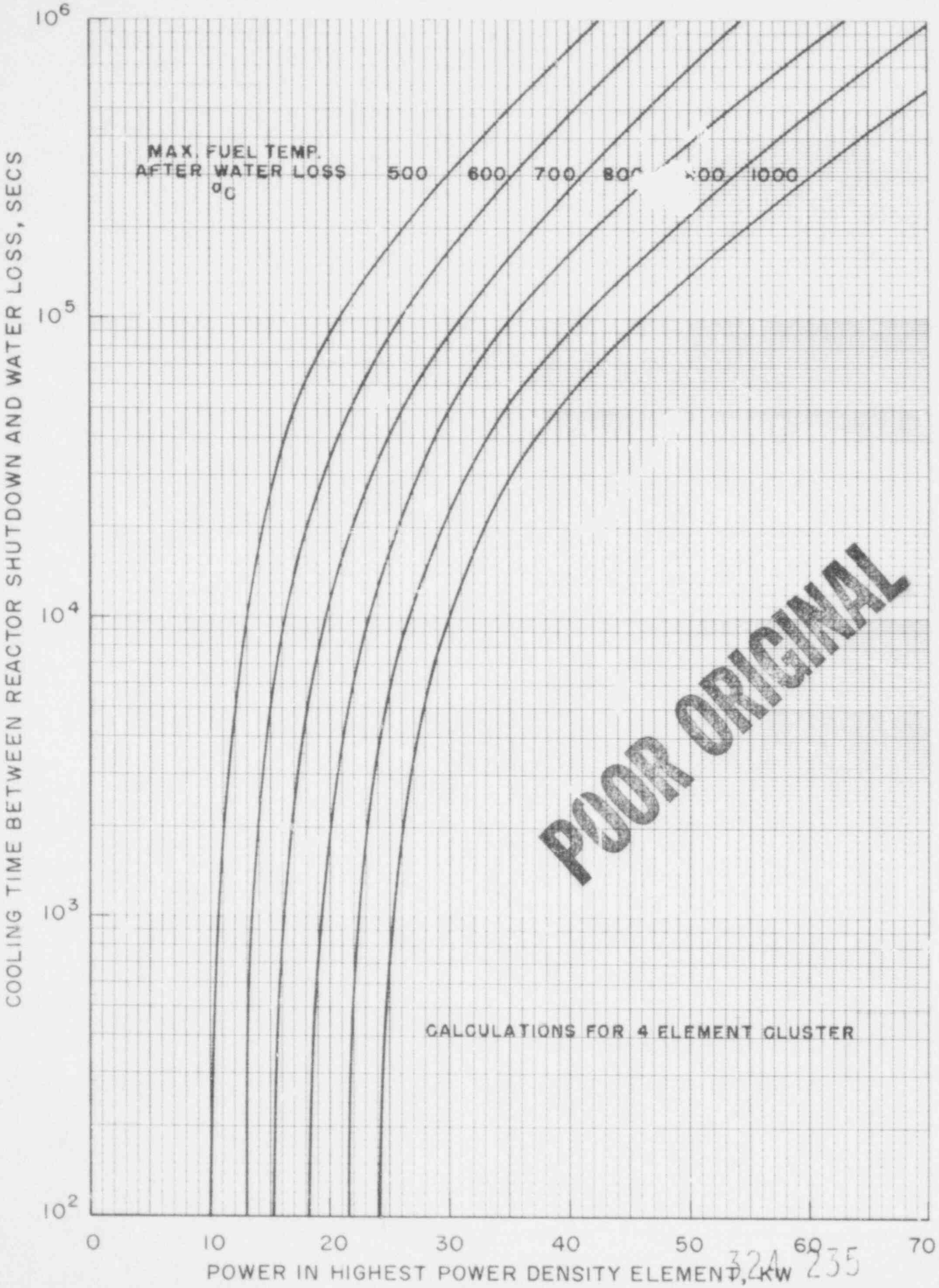

FIGURE 2 MAXIMUM TEMPERATURE IN FUEL ELEMENT
FOLLOWING LOSS OF COOLANT

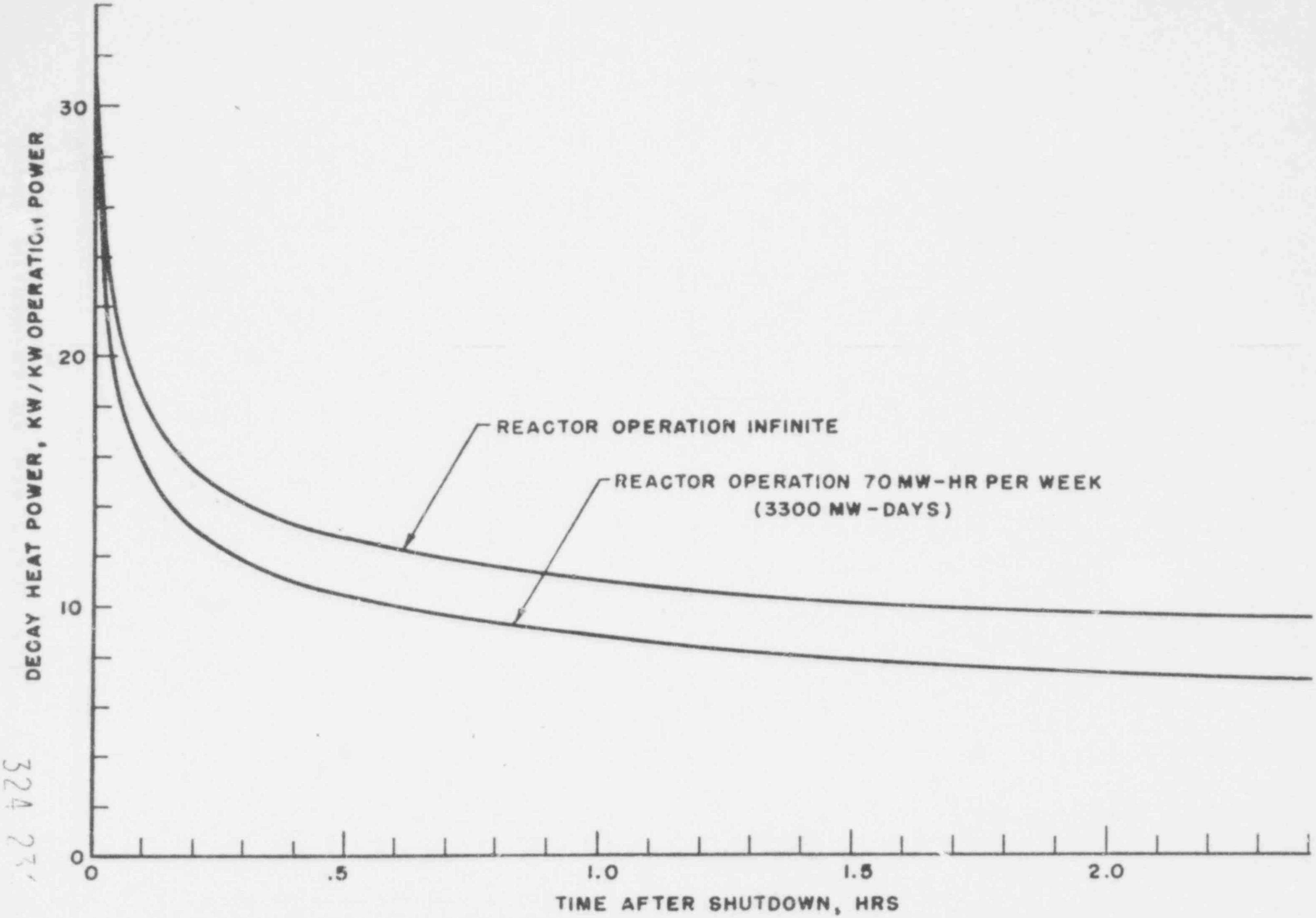

FIGURE 3 DECAY HEAT POWER GENERATION FOLLOWING LOSS OF COOLANT FOR INFINITE REACTOR OPERATION AND PERIODIC REACTOR OPERATION

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Appendix III

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.

Emperical 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 ft^3) 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 ft^2. The quantity of explosive will be considered to be the maximum of 5 pounds allowed in the facility. For a containment volume of 270 ft^3 (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,

$$100 \text{ psi} \times A_{BP} = 100\ (\pi)\ (3)^2$$
$$= 2800 \text{ lb}_f.$$

(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,

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

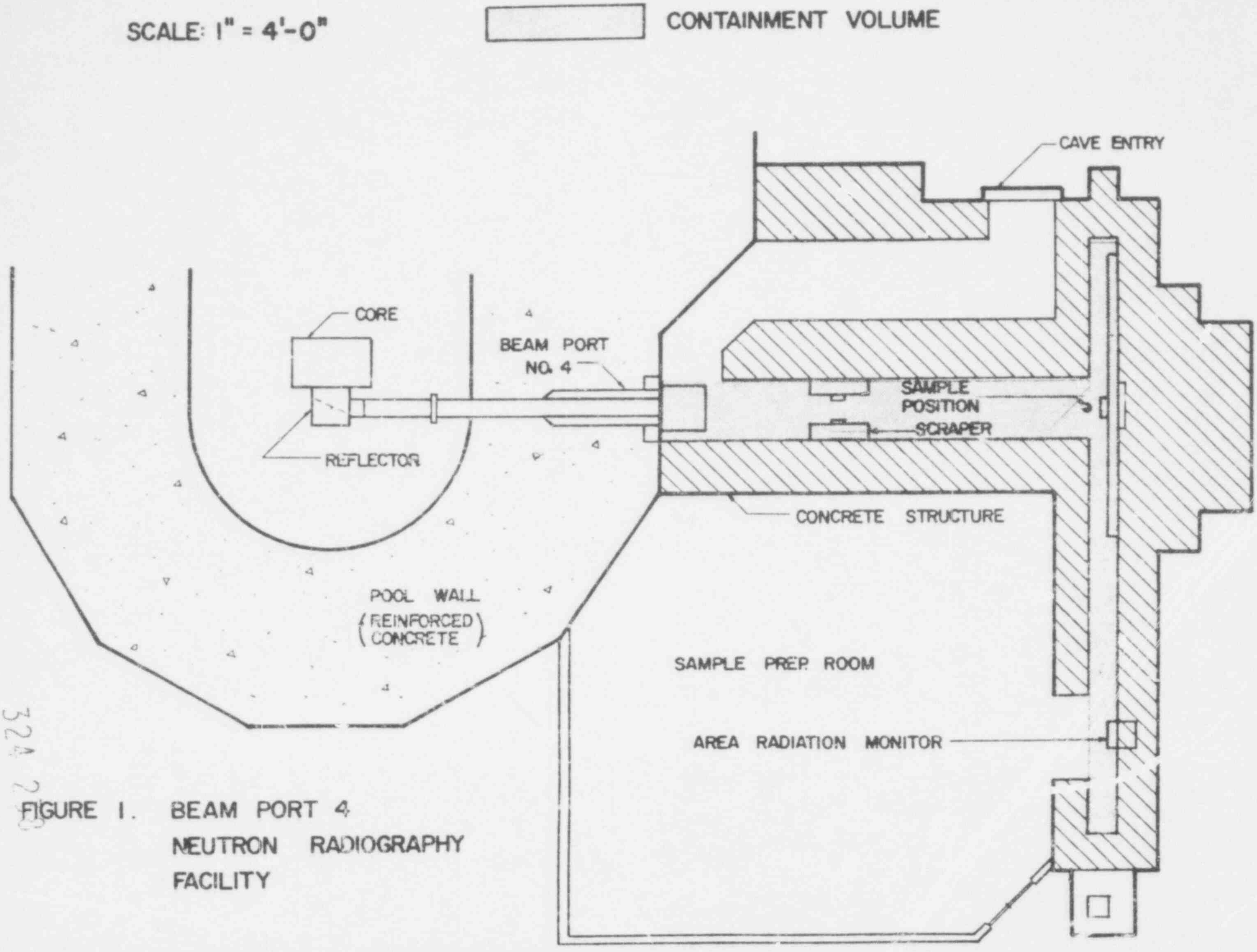

FIGURE 1. BEAM PORT 4
NEUTRON RADIOGRAPHY
FACILITY

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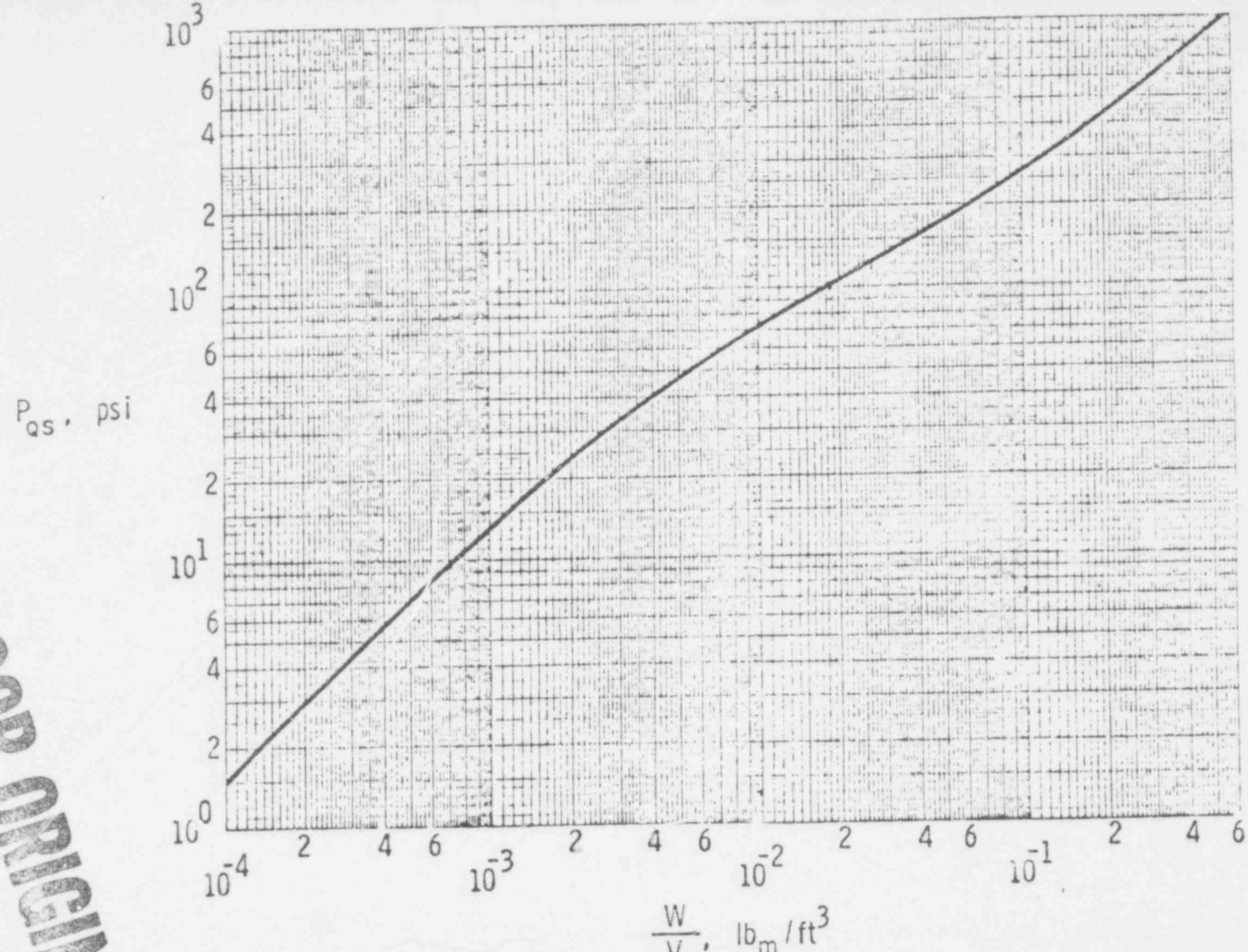

FIGURE 2 PEAK QUASI-STATIC PRESSURE FOR TNT EXPLOSION IN CHAMBERS

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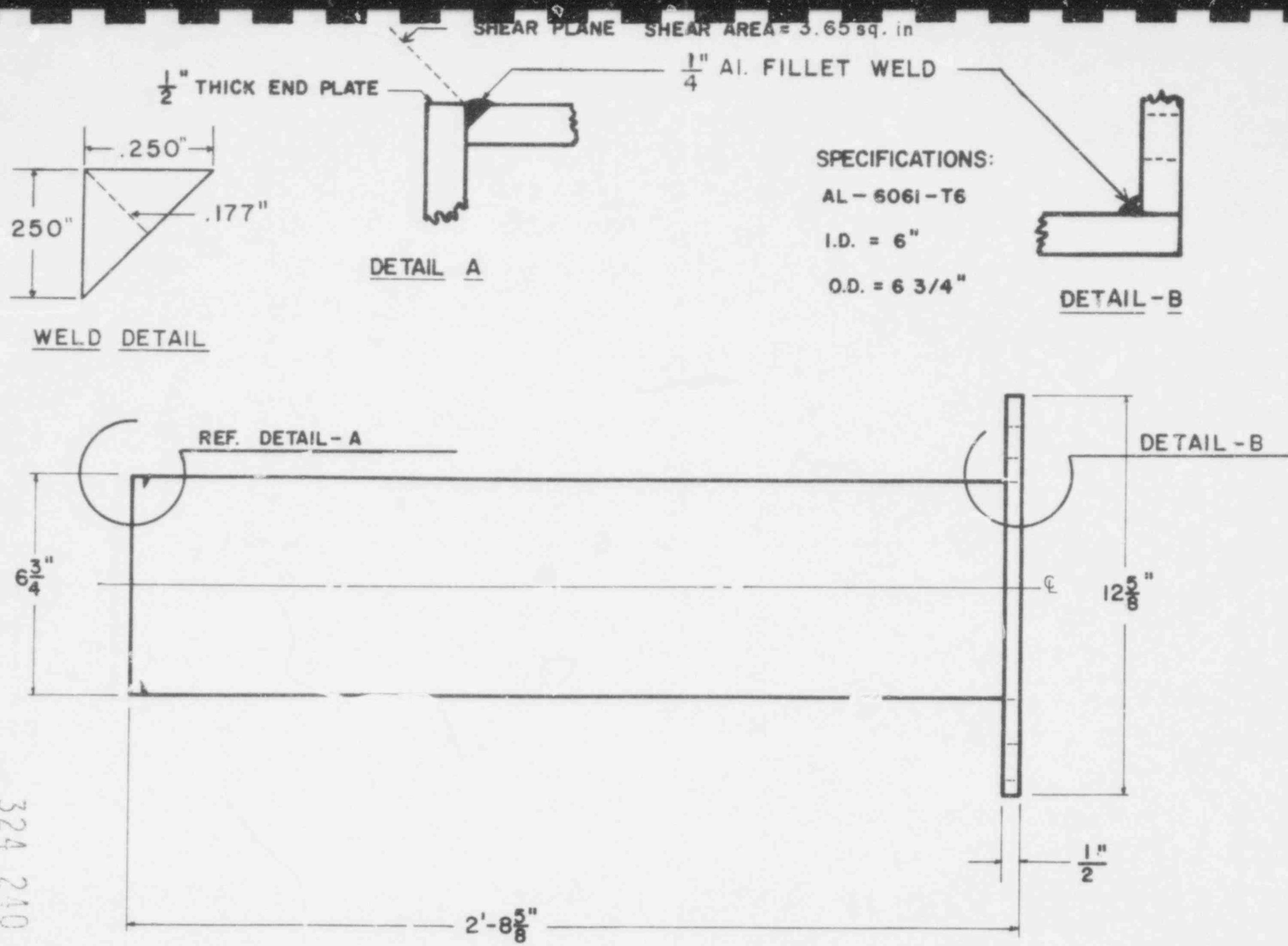

FIGURE 3 . (MODIFIED)

NSCR NEUTRON RADIOGRAPHY BEAM PORT EXTENSION

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

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

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

A comparison to the calculated peak blast pressure which results in a 2,800 lb_f 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

$$P_\ell = 5,000 \text{ psi}.$$

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 preceed 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.

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