UPDATED
SAFETY ANALYSIS REPORT
PULSTAR REACTOR

NORTH CAROLINA STATE UNIVERSITY
RALEIGH, NORTH CAROLINA 27695

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1 INTRODUCTION AND SUMMARY

1.1 Introduction

This Safety Analysis Report (SAR), originally submitted as the Preliminary Safety Analysis Report and on which Construction Permit CPRR-106 was issued, has been subsequently amended during the first licensing period. It is now amended to become the Safety Analysis Report (SAR) in support of North Carolina State University's (NCSU) application and renewal for a Facility License to operate a one megawatt reactor on the NCSU Campus at Raleigh, North Carolina.

This report provides the necessary information required by 10 CFR 50, and is sufficient to demonstrate the safety of the design and continued operation of the PULSTAR Facility. The Technical Specifications based on this SAR continue to identify those operational safety features of the facility which cannot be exceeded or changed without prior approval of the Nuclear Regulatory Commission (NRC).

The University's mission is to carry out the traditional teaching, research, community, and national service functions. The reactor and experimental facilities are used by students, faculty, and staff in carrying out this mission.

The design of the reactor and associated laboratories was completed and construction started in June, 1969. The general arrangement of the Burlington Engineering Laboratories (BEL) are shown in Figure 1-2 through Figure 1-5. The floor plan, elevations, and sections of the Reactor Building are shown in Figures 1-6 and 1-7. The Reactor Building is a separate confinement area within BEL. With the exception of the Primary Piping Vault (PPV), only minor changes have been made within the Burlington Engineering Laboratories since the original SAR was issued. The south wing of Burlington Engineering Laboratories is depicted in Figures 1-1 and 1-8.

The PULSTAR Reactor was manufactured by the American Machine and Foundry Company (AMF) and its design, fabrication, and installation are based on the proven prototype located at the Buffalo Materials Research Center (BMRC) at the State University of New York at Buffalo. It is a light water moderated and cooled, low enriched ^{235}U (4%), heterogeneous, pin-type fueled, thermal reactor with initial criticality occurring in September, 1972. It is operated by the Department of Nuclear Engineering within the College of Engineering. The PULSTAR Reactor operates at steady state power levels up to one megawatt. The PULSTAR Reactor was originally designed to be pulsed routinely to 2200 MW peak power and 38 MW·sec total energy release.

The PULSTAR Reactor core is located in a fixed position at the bottom of a 26.5 foot (8.08 m) deep pool of water. Its design and operational mode were originally based on the results of many years of testing of its prototype at BMRC. The reliable operation of the NCSU PULSTAR during the first license interval supports the fact that no additional research or developmental testing is necessary to confirm proven safety features, and therefore is operated in regions of proven technology and predictable responses. As part

of the relicensing of the PULSTAR Reactor, no significant deviations from the proven design and operation are proposed; the only exceptions are that pulsing has been discontinued, and a subterranean Primary Piping Vault (PPV) has been added to contain the previously buried N-16 delay tank and primary piping.

The University, specifically the Department of Nuclear Engineering within the College of Engineering, is fully responsible for the safe and efficient operation of the PULSTAR Reactor and its support facilities. The plan for carrying out this responsibility is based on the experience at both NCSU and BMRC and the assimilation of proven operational and functional principles at other research and training reactors.

Organizationally, the PULSTAR Reactor is operated under the direction of a Nuclear Reactor Program Director, with a staff organization that has proven quite successful during the first license interval on the PULSTAR, and is similar to that employed at other successful university reactor facilities. In addition, operational activities at the reactor are conducted with audit and approval of external safety committees.

The University established the Reactor Safety and Audit Committee (RSAC), the former subcommittee to the Radiation Protection Committee (RPC), to assist in ensuring that the reactor is operated in compliance with the facility license and all applicable regulations. The RSAC has initial review responsibilities to determine that any new or proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question in accordance with 10 CRF 50.59. Recommendations arising from committee review are forwarded to RPC for concurrence before being implemented.[1-1]

1.2 Design Highlights

1.2.1 Site Characteristics

The site is characterized by typical university surroundings, relatively light to moderate population areas, and a natural environment offering no undue or unusual characteristics. The area is free of any abnormal or hazardous conditions, e.g., explosive or chemical plants, etc. which would jeopardize the reactor or Reactor Building. Finally, in the unlikely event of release of radioactive materials to the Reactor Building, there is no evidence that any undue risks to the public would result from site characteristics.

1.2.2 Reactor System

The reactor was selected and designed to take advantage of proven characteristics of the power reactor type fuel, specifically the Doppler broadening, high retention rate of fission products by the matrix, and low thermal transmissivity. Finally, the design and operating limits fall well within the already approved and/or proven safety limits of its prototype at BMRC. Neither the reactor nor its support systems require any research or development to demonstrate continued inherent safety of design and operation.

1.2.3 Reactor Building Structure

The intrinsic shutdown mechanism of the PULSTAR is well established by experience at both the Buffalo prototype and at the NCSU PULSTAR, and the use of confinement areas continues to be an acceptable and successful replacement of containment. Therefore, the system (structure) required to confine the Maximum Credible Accident (MCA) as detailed in Section 13 has been modeled after the latest trends in reactor building envelopes, specifically confinement. The Reactor Building has proven to be a relatively air-tight structure design during its original operating period. The Reactor Building is a reinforced, monolithic concrete structure faced with a brick veneer. The PPV is an underground reinforced concrete structure with minimal penetrations which is coupled to the reactor building. The Reactor Building and PPV are designed to confine radioactive releases and at the same time provide for controlled release to the surrounding environment at levels below the 10 CFR 20[1-2] limits. The remainder of BEL administrative offices and laboratories are conventional structures for their particular functions. Several engineering safeguards are similar to those at other facilities and present neither uncommon solutions nor extrapolations in technology.

1.3 CHARACTERISTICS

1.3.1 Summary

Table 1-1 summarizes the existing characteristics of the reactor and its support systems. Variations or differences from the prototype are in a conservative direction and were made adjunct to the functional needs of the University's program and AMF's product improvements rather than any major technological or safety limitation of the prototype. The reactor physics information presented in the initial SAR issuance was for the 5 x 5 Standard Core (the original core arrangement for startup in 1972). Two other core arrangements used in the first licensing interval include the 5 x 5 Reflected Core No. 1 and the 5 x 5 Reflected Core No. 3, both of which will be discussed in detail in Section 3. The PULSTAR has always employed a 5 x 5 array of fuel assemblies (total of 25) and it will continue to be the core arrangement for safety related analysis. Data other than reactor physics information has remained the same during reactor operation to date.

1.3.2 Engineered Safety Features

Engineered safety features are employed to reduce the potential dose for operating personnel and the general public to less than the regulatory limits of 10 CFR 20[1-2]. The major step in accomplishment of this objective was the selection of a proven reactor and support systems. The next step was to limit the consequences of a credible but most improbable event, even if it should occur. This is accomplished by automatically isolating the Reactor Building and therefore controlling the release of activity.

To physically accomplish the above, the reactor and/or its support systems have the following:

a. Strong negative reactivity responses to reactor power surges, evidenced by the ability to routinely accept large pulsed power increases.

b. Heat capacity to accommodate fission product heating, even if the core is completely uncovered.

c. A building confinement system to control the release of any fission products.

d. High stack for dilution of fission products and their subsequent release to the environment.

e. Auxiliary power supply and instrumentation to accomplish the safeguards operations automatically.

1.4 Principal Design Criteria

The NCSU PULSTAR Reactor is designed to meet basic operational and safety practices. The criteria which represent our framework of reference and on which more detailed design could proceed are the following:

1.4.1 Criterion 1

Those features of reactor design essential to the prevention of accidents must be designed, fabricated, erected, and tested to establish quality and performance standards. This was accomplished by the selection of a tested and proven system from a reliable contractor and the use of nuclear consultants to assist in the design of appurtenances.

1.4.2 Criterion 2

The maximum reactivity worth of control rods and the rate at which reactivity can be inserted are held to values such that no single credible malfunction could create a transient capable of causing significant fuel failure.

1.4.3 Criterion 3

Reactivity shutdown is provided for any credible operating condition with the highest worth control rod fully withdrawn.

1.4.4 Criterion 4

Capability for control rod insertion under abnormal conditions is provided.

1.4.5 Criterion 5

The reactor facility has a single control room from which all actions can be controlled or monitored to insure a safe operation at all times. This room is provided with adequate means to insure the protection of its occupants.

1.4.6 Criterion 6

A reliable Reactor Protection System is provided to automatically initiate appropriate action and prevent exceeding safety limits. Testing of these systems for continued operability is provided. Redundancy and independence of vital channels are provided.

1.4.7 Criterion 7

The confinement structure must accommodate the environment of the largest credible energy release and site characteristics.

1.4.8 Criterion 8

Sufficient normal and auxiliary sources of electrical power have been provided to assure a capability for prompt shutdown of the reactor facility to a safe condition.

1.4.9 Criterion 9

Suitability of the facility and sub-systems will depend on demonstrated performance, reliability, and the extent of test and inspection during the life of the plant. This has been confirmed during the construction and checkout of the PULSTAR.

1.4.10 Criterion 10

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel elements.

1.4.11 Criterion 11

The facility is provided with systems that are capable of monitoring the release of radioactivity under normal and accident conditions.

1.5 Identification of Contractors

1.5.1 Reactor Designer

The American Machine and Foundry Company (AMF) who designed, fabricated, and installed the PULSTAR prototype at BMRC, was also the designer, fabricator, and installation and checkout supervisor of the NCSU PULSTAR Reactor. AMF also furnished the fuel.

1.5.2 Architect-Engineer

Charles W. Wheatley and Associates were the architects for the BEL complex and were supported by the nuclear consulting firm of Parsons-Jurden Company of New York City. Parsons-Jurden was responsible for the design and initial review of the reactor support systems. Detailed design of the waste handling, radiation monitoring and process systems outside the reactor was handled by the architect. The Title III services (field supervision, inspection, and acceptance test of all but the AMF design) was handled by the architect and his subcontracted engineers. The architect appointed a qualified individual to serve as the full time on-site Inspection Coordinator[1-3] responsible for the scheduling and monitoring of inspection services by the design engineers, giving reasonable assurance to the University and NRC that compliance by the contractors with the contract documents was effected, and ensuring that adequate records were available to substantiate the effectiveness of the quality control and assurance programs.

1.5.3 NC State Consultants and Advisors

The University established an Ad Hoc Building Planning Committee to manage the overall planning for delivery of the PULSTAR Reactor and BEL complex. The technical management of the project was handled by the Department of Nuclear Engineering. The faculty of the College of Engineering actively served as advisors to the Department's Project Engineer. The Department Committee was responsible for evaluating and passing on the Safety Analysis Report. Also within the Department were experts in instrumentation and controls, health physics, and experiment planning. The Chairman of the former Radiation Protection Council (now the Radiation Protection Committee) appointed a subcommittee to review the Safety Analysis Report (SAR).

The University defined and implemented a Quality Assurance Program (QAP)[1-3] with a Quality Assurance Coordinator (QAC) who directed/coordinated all quality assurance measures for the project, both on-site and off-site, as they related to the nuclear safety and facility operational aspects. The three levels of QA were: control by contractor, surveillance by design engineers, and audit by the QAC.

1.6 <u>References</u>

1-1 "ENVIRONMENTAL HEALTH AND SAFETY POLICIES AND PROGRAMS", North Carolina State University, Issue 4, August 1, 1995.

1-2 United States Nuclear Regulatory Commission RULES and REGULATIONS Title 10, Chapter 1, Code of Federal Regulations, Part 20, "STANDARDS FOR PROTECTION AGAINST RADIATION", September 30, 1994

1-3 J. R. Bohannon, "NUCLEAR SCIENCE AND ENGINEERING RESEARCH CENTER QUALITY ASSURANCE INSTRUCTION MANUAL", North Carolina State University, School of Engineering, July 1, 1969

TABLE 1-1
COMPARISON OF PULSTAR REACTORS

	NCSU		BMRC
Fuel			
Material	UO_2		
Form		Sintered Pellets	
Enrichment (weight % ^{235}U)	4%		6%
Design Inventory Core (kg UO_2)	359		285
Density (gm/cm^3)	10.5-10.76		10.3
^{235}U per Fuel Pin (gm)	20.2		30.7
Fuel Pin			
Pellet (diameter nominal in (cm))		0.423 (1.074)	
Diametrical Gap (nominal in (cm))		0.0085 (0.0216)	
Zircaloy-2 clad thickness (in (cm))		0.0205 (0.0521)	
Outside Diameter Pin (in (cm))		0.4725 (1.2002)	
Rectangular Spacing (center-to-center in (cm))		0.606 x 0.524 (1.54 x 1.33)	
Clearance (pin-to-pin in (cm))		0.051 x 0.133 (0.130 x 0.338)	
Clearance (pin-to-box in (cm))		0.025 x 0.066 (0.064 x 0.168)	
Height of Pellet Stack (in (cm))		24 (61)	
Pins per Core	625		500
Height of Pellet (in (cm))		0.60 (1.52)	
Fuel Box			
Material		Zr-2	
Inside Dimensions (in (cm))		2.620 x 3.030 (6.655 x 7.700)	
Wall Thickness (in (cm))		0.060 (0.152)	
Clearance between Assemblies (in (cm))		0.040 (0.102)	
Clearance between Control Rod Guide and Assemblies (in (cm))		0.060 (0.152)	
Fuel Pins per Assembly		25	
Weight (pounds (kg))		44(20)	
Moderator - Reflector - Coolant			
Material		Light Water	
Nominal Inlet Temp. (°F (°C))	105 (40.6)		100 (37.8)
Nominal Outlet Temp. (°F (°C))	118.8 (48.2)		111.8 (44.3)
Primary Flow Rate (gpm (ℓs^{-1})	500 (31.5)		1150 (72.5)
Secondary Flow Rate (gpm (ℓs^{-1})	700 (44.2)		800 (50.5)

**Tabulated values for the NCSU PULSTAR are for the 5 x 5 Standard Core while values for the BMRC PULSTAR are for a 5 x 4 core loading, both water reflected.*

TABLE 1-1 (continued)

	NCSU		BMRC
Neutron Source	5 curie Pu-Be		Sb-Be
Control Rods			
Absorber Material		Ag-In-Cd (80-15-5)	
Guide Material		Aluminum	
Shape		Rectangular	
Transverse Dimensions (inches (cm))			
Guide		6.30 x 0.43 (16.0 x 1.09)	
Absorber		4.85 x 0.18 (12.32 x .46)	
Clearance Absorber to Guide (inches (cm))		0.00625 (0.01588)	
Clad Material	Sn/Ni		Ni
Number of Control Rods	3		5
Number of non-scrammable rods		1	
Core Dimensions			
Overall	15-7/8 x 15 x 24 (in) 40.3 x 38 x 61 (cm)		15-7/8 x 12-1/8 x 24 (in) 40.3 x 30.8 x 61 (cm)
Volume Fractions			
UO_2	0.3823		0.3789
Gap (Helium)	0.0155		0.0154
Cladding (includes warts)	0.0803		0.0795
Lattice Water	0.3858		0.3824
Assemblies (Zr-2)	0.0753		0.0747
Water Between Assemblies	0.0255		0.0252
Control Rod Guides (Al)	0.0128		0.0158
Water Inside Guides (Rods out)	0.0226		0.0280
Total Volume Fractions			
UO_2	0.3823		0.3789
Helium Gap	0.0155		0.0154
H_2O	0.4339		0.4356
Zr-2	0.1555		0.1542
Aluminum	0.0128		0.0158
Core Volume Ratios			
H_2O/UO_2	1.135		1.150
H_2O/U	2.06		2.11
Physics Parameters			
Effective Neutron Temperature(ev)	0.0509		0.0587
Disadvantage Factors			
Moderator to Fuel	1.243		1.309
Clad to Fuel	1.125		1.163
Neutrons per Thermal Fission (ν)		2.43	

TABLE 1-1 (continued)

	NCSU		BMRC
Capture to Fission Rate (α)		0.18	
Thermal Utilization	0.951		0.964
Fast Fission Number	1.397		1.050
Age (cm^2)	43.8		30.7
Resonance Escape (p)	0.558		0.767
Thermal Diffusion Area (L^2)	2.215		1.178
Reflector Savings (cm)	7.81		7.77
Bucklings (cm^2)			
Width	0.00342		0.00320
Length	0.00316		0.00260
Height	0.00168		0.00168
k_{eff}(cold clean with H_2O Reflector)	1.0178		1.045
Minimum Critical No. of Assemblies	20		16
Doppler Coefficient (pcm/°F)	-1.40		-2.15
Void Coefficient (pcm/cm^3)	-1.03		-1.1
Moderator Temperature Coefficient	-3.2		-8.0
Beam Tube Worths (air to water filled pcm)			
6" diameter	193		100
8" diameter	45		-
12" x 12"	20		-
Neutron Lifetime (sec)	3.6×10^{-5}		2.9×10^{-5}
β_{eff}	0.0073		0.0076
Steady State Power Level (MW)	1		2
Design Pulse Peak Power (MW)	2200		2000
Design Pulse Total Energy Release (MW·sec)	38		40
Design Step Input (pcm)	1720		1740
Maximum Rate of Reactivity Insertion by Control Rods (pcm/sec)	100		100
Reactor Bay			
Dimensions (ft (m))			
Height	55 (17)		52 (16)
Width	37 (11)		70 (21)
diameter			
Length (ft (m))	94 (29)		N/A
Free Air Volume (ft^3 (m^3))	86,500 (2,448)		186,000 (5,264)
Ventilation (cfm (m^3/sec))			
Normal	10,050 (4.7)		12,000 (5.66)
Emergency	600 (0.3)		3,200 (1.5)
Type Building	Confinement		Containment
Exhaust Stack Height (ft (m))	100 (30)		167 (50)
Pool Volume (gal (ℓ))	15,600 (59,052)		14,592 (55,231)

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9509200199-02

FIGURE 1.3
FIRST FLOOR PLAN, BEL

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FIGURE I-4
SECOND FLOOR PLAN, BEL

FIGURE I-5
THIRD FLOOR PLAN, BEL

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9509200199-06

2 SITE AND ENVIRONMENT

2.1 Site and Adjacent Areas

Pertinent information related to the NCSU PULSTAR Reactor site, including demographics and land use is presented in this section. Natural site characteristics have been studied and include considerations of meteorological, hydrological, geological, and seismological factors.

The site is usually free of external hazards associated with regular commercial or military air traffic, explosions or fire in industrial areas, strong winds, earthquakes, or floods. Raleigh-Durham International Airport is approximately 10 miles (16 km) from the PULSTAR Nuclear Reactor site.

Test borings indicate that a soil pressure value of 3,000 lbs/ft^2 (140 kPa) at 5 feet (1.5 m) below the existing grade can be used for foundation of the secondary part of the building including retaining walls. Maximum soil pressure values of 4,000 lb/ft^2 (190 kPa) from elevation 400 feet (122 m) to 390 feet (119 m) or 5,000 lb/ft^2 (240 kPa) from elevation 390 feet (119 m) to 385 feet (117 m) were recommended for spread footing or mat foundation schemes for the main building structure.

Ventilation and filtering systems for the Reactor Building are designed such that a maximum design release from the stack would not result in doses from radioactive noble gases and halogens exceeding safe limits. See Section 13 for detailed calculations concerning radiological implications. The aerodynamic effect due to the proximity of the stack to nearby buildings, including the Reactor Building, may be neglected since their heights are less than 40 percent of the stack height. Further justification for neglecting effects of local structures on stack effluent concentrations resides in the relative openness of the building area. The Reactor Building is located in a quadrangle surrounded by streets on all sides.

2.1.1 Location

The North Carolina State University PULSTAR Reactor is located in a separate bay (Reactor Building) of the Burlington Engineering Laboratories near the center of the campus. The campus is located in the western part of the City of Raleigh near the center of Wake County, North Carolina. Relative positions of the nearby campus buildings and the railroad line are found on the N.C. State University map, Figure 2-1 and the "Detailed Campus Layout Within Close Proximity of Burlington Engineering Laboratories Map", Figure 2-3.

2.1.2 Topography

The reactor floor is situated at an elevation of 396 feet (121 m) above mean sea level. The campus grounds slope steeply from this site to the east, south, and west and affords good natural drainage and freedom from flooding as shown in Figure 2-2. The terrain of the area inside a radius of more than 15 miles (24 km) about the reactor site is made up of gently rolling land ranging in altitude from about 350 feet (107 m) to 450 feet (137 m). Except for

a few small lakes, the land is well drained; creeks carry the surface water to rivers that flow toward the southeastern part of the state. The nearest mountains lie more than 75 miles (121 km) to the northwest.

2.1.3 Population and Land Use

2.1.3.1 Part-Time Population

Except for occasional changes in campus student population, no marked variations in the local population exist. Tourism and other seasonal industries are not important factors in population size. On the campus, an important change in population takes place at night. Buildings near the PULSTAR Reactor are nearly empty and the campus population, including dormitory residents, changes as shown in Table 2-1 below:[2-1]

TABLE 2-1

Population for Campus of North Carolina State University
Fall of 1994-1995 Academic Year

GROUP	DAYTIME	NIGHTTIME
Students	27,577	2,426
Faculty	1,914	200 to 300
Administrative Staff	952	<50
Other Staff	3,497	<300
TOTALS	33,940	<3,076

During summer school the typical number of students on campus during the day is estimated to be approximately 8,000 at present.

2.1.3.2 Use of Land Areas and Permanent Population

Raleigh is the capital city of North Carolina and, as such, contains much of the State's government facilities and several State institutions. Much of the commercial activity in the area typically arises from the operation of State government and State institutions. There is limited industry within a 5 mile (8 km) radius of the reactor site, most of which lies to the east. The commercial activity within the radius is limited to mostly retail business activity.

Population details, housing, industry and commerce surrounding the PULSTAR Reactor site are presented in Figure 2-4, Figure 2-5 and Table 2-2.[2-2,2-3] Figure 2-4 depicts tax zones or parcels of land by which Wake County documents geographical and demographical information. For each column of tax zones depicted in Figure 2-4, detailed population and

land use data is presented in Table 2-2. A breakdown of what each type and "use unit" includes is detailed in Figure 2-5. The data depicted in Figure 2-4 and Table 2-2 is based on January 1990 census records provided by Wake County showing a population of about 185,000 within the 5 mile (8 km) radius and an approximate total population of 267,000, including the adjoining towns of Cary and Garner. The color map at the end of this section, Figure 2-4A, provides a pictorial representation of land use around the reactor site.

The population of grazing animals and dairy cattle within 1 and 5 mile (1.6 and 8 km) radii of the reactor site and for eight directional sectors is depicted in Figure 2-7.[2-4,2-5] Figure 2-8 shows the nearby land areas owned and controlled by North Carolina State University.[2-6]

2.2 Meteorology of Site and Vicinity

The purpose of this section is to present data which are directly applicable to the computation of radiation hazards which might be present during or after operating the N.C. State University PULSTAR Reactor.

Several detailed meteorological surveys of the site have been conducted. The first was conducted prior to the actual design and construction of the first Raleigh Research Reactor (NCSUR-1). Later in January 1952, additional data were presented.[2-7]

Downey's thesis includes photographs of smoke releases from the 100 foot (30.5 m) stack under a considerable variety of weather conditions.[2-8] Observations of the distance from the stack at which maximum ground concentrations developed during each smoke release were recorded. Also temperature gradient data over the stack height, and other pertinent data, are included in this thesis.

In addition to data collected by the National Climatic Data Center at the Raleigh-Durham International Airport, some meteorological data have been collected at the weather bureau on the NCSU campus. Campus wind observations were made on the roof of Ricks Hall, approximately 50 feet (15.2 m) above ground level. The campus and Airport data are comparable, climatic factors are similar and the Airport data is a satisfactory approximation to campus conditions.[2-9]

Recorded weather information, updated hourly by the National Weather Service, is available by campus telephone on a 24 hour basis.

In general, the climate is of a modified continental type, i.e., warm, temperate and moderately wet with no particularly dry season. In the summer months a maritime tropical air mass predominates. The high humidity, combined with surface heating, tends to make this air mass unstable, with the result that frequent thunderstorms occur in summer and early fall. Wintertime sees the principal air mass changed to a modified polar continental type with occasional replacement of modified maritime polar air from the Pacific area. Extreme cold waves are rare in this section due to the mountain barrier to the west and the modifying influence of the Atlantic Ocean to the east. Inversions usually accompany the incursion of maritime tropical air in the wintertime, due to the interaction of warm air from

the Gulf of Mexico with cold polar air, which usually results in widespread overcast. Winter snowstorms sometime accompany low pressure centers moving up the coast. North or northeast winds predominate during these periods, bringing in moist cold maritime air from the Atlantic Ocean.

2.2.1 Wind and Precipitation

2.2.1.1 Upper Air Movement

Although the immediate problem of diffusion in this area does not extend to the air above 5,000 feet (1 500 m), this data is included since upper air movement and surface patterns are not entirely separable. Table 2-3 depicts the monthly tabulation of the mean geostrophic wind at the 700 mb (70 kPa) constant pressure surface (approximately 10,000 feet (3 048 m)).

2.2.1.2 Surface Winds

Average climatological data measured at the Raleigh-Durham International Airport for the period of January 1990 to December 1994 is detailed in Table 2-4.[2-10] Table 2-5 details time-of-day average readings taken for the same period at the same location for wind and frequency of precipitation events.[2-11]

Two wind roses constructed from data provided by the National Climatic Data Center for the Raleigh-Durham International Airport for the five year period from 1990 through 1994 are shown in Figures 2-9 and 2-9A.[2-12,2-13] Figure 2-9 displays the percent direction of all winds along with the average speed in meters per second. Figure 2-9A displays the same data in a slightly different format. The percent of all winds 10 knots or less from all directions is shown in the center of the figure. Each range of wind speed (11 - 16, 17 - 21, 22 - 27, and 28 knots or greater) is displayed in succeeding bands along with the percent of winds from that direction. Table 2-6 gives the percent of calm frequency for the data collected during the same five year period. Figure 2-10 shows a wind rose for early data taken at NCSU which compares favorably with the most recent data shown in Figure 2-9.

2.2.1.3 High and Violent Winds

Within a five mile (8 km) radius of the reactor, there are approximately four occurrences of winds with tornado characteristics per 60 years. On November 28, 1988, Raleigh was struck by one or more tornadoes resulting in $77 million in property damage, 4 deaths and 157 injuries. The track of the Raleigh tornado was 84 miles (135 km) in length and was estimated at its onset as F4 severity on the Fujita Scale with winds estimated at 210 mph (338 kph). This particular tornado had an estimated ground speed of 50 mph (80 kph) and passed through Wake, Franklin, Nash, Halifax, and Northampton Counties. At its closest point, it passed within approximately 5.5 miles (8.8 km) of the PULSTAR Reactor facility. Prior to this date, only twelve other tornadoes had been reported within the state of North Carolina during November for the period of 1916 to 1987. For that same period only eight tornadoes were reported in December. During the period of 1884 through 1994, only six

F4 severity class tornadoes have been recorded in North Carolina. On the average, North Carolina experiences approximately 12 tornadoes per year.

The only record of damaging hurricane winds in the Raleigh area, which occurred on October 15, 1954 (73 mph (117 kph) winds averaged over one minute), was due to Hurricane Hazel. In this case, all major damage was due to trees falling on houses. Winds during a hurricane are generally from the northwest; with the eye of hurricanes usually passing east of Raleigh, and likely off the coast. However, on 22 September of 1989, Hurricane Hugo came ashore at Charleston, South Carolina, and its eye later passed over Charlotte, North Carolina (and therefore, west of Raleigh). No damage was recorded on the NCSU campus or to the PULSTAR Reactor Facility as a result of Hurricane Hugo's inland movements.

2.2.1.4 Precipitation

The most recent data for precipitation taken at the Raleigh-Durham International Airport (January 1990 to December 1994) is given in Table 2-4 and 2-5. Table 2-4 gives five year averages each month for precipitation as well as temperature, relative humidity, and wind. Table 2-5 gives data at three hour intervals for resultant wind direction, speed and the number of precipitation events per year.

2.2.2 Temperature and Other Data

2.2.2.1 Surface Summaries

For the five year period of January 1990 to December 1994, Table 2-4 provides the most recent climatological data for temperature. It gives the average daily maximum and average daily minimum temperature for each month of the year.

Additionally, observations taken by the National Weather Service over a period of thirty years from 1961 to 1990 at the Raleigh-Durham International Airport are as follows:[2-14]

Yearly Average Daily Maximum	70.1°F	21.2°C
Yearly Average Daily Minimum	48.4°F	9.11°C
Highest Monthly Average (maximum)	88.0°F	31.1°C
Highest Monthly Average (minimum)	68.1°F	20.1°C
Lowest Monthly Average (maximum)	48.9°F	9.39°C
Lowest Monthly Average (minimum)	28.8°F	-1.78°C
Record High in July, 1988 (tie)	105°F	40.6°C
Record Low in January, 1985	-9°F	-23°C

Except for a record low in January 1985, there has been no significant change in the climatological data. The thirty year average for temperatures shown above compare very favorably with the five year averages given in Table 2-4.

2.2.2.2 Gradient and Diffusion Data

Measurements of temperature gradient along the PULSTAR air stack from 9 feet (2.7 m) to 104 feet (31.7 m) above ground level were made during the period of October 19, 1953 through March 7, 1954.[2-8]

The results are as follows:

	Time-of-Day Groups				
	I	II	III	IV	V
Inversions	2	2	7	7	21
Neutrals	2	7	8	17	8
Mild Lapses	5	13	32	9	5
Strong Lapses	-	6	64	-	-
Total Observations	9	28	111	33	34

The time-of-day groups and gradients classes are defined as follows:

Inversion	$+\ \Delta T_{stack} > 0°F$
Neutral	$0°F \leq -\ \Delta T_{stack} \leq 0.523°F$
Mild Lapse	$0.523°F \leq -\ \Delta T_{stack} \leq 1.05°F$
Strong Lapse	$-\ \Delta T_{stack} \geq 1.05°F$

Group I	Before sunrise to 30 minutes after sunrise
Group II	31 minutes to 2 hours after sunrise
Group III	All cases not covered in Groups I, II, IV, or V
Group IV	From 1 hour before sundown until sundown
Group V	Shortly after sundown

Typical values of Sutton's parameter, c^2, are given below as a function of height, h, and stability parameter, n.[2-15]

		h			
	n	25	50	75	100
Large Lapse Rate	0.20	0.043	0.030	0.024	0.015
Small Temperature Gradient	0.25	0.014	0.010	0.008	0.005
Moderate Inversion	0.33	0.006	0.004	0.003	0.002
Large Inversion	0.50	0.004	0.003	0.002	0.001

Assuming that once surface cooling begins, it takes about four hours to reach a maximum stability, remains at maximum for four hours, and then takes an additional four hours to dissipate, one can infer that the greatest stability exists at 12 percent of the time. Moderate inversions exist 24 percent of the time and the remaining 64 percent of the time, moderate to extreme instability persists.

The worst inversion conditions that occur within the course of a single day usually occur at night, especially in clear calm weather. Autumn is usually the worst season for air pollution due to this inversion condition.

2.2.2.3 Barometric Variation

There are no annual or regular seasonal barometric variations that can be called severe. During the course of a day there are usually two swings of small magnitude occurring at noon (Eastern Standard Time) and around midnight. The former is sun dependent while the latter is moon dependent. These variations are slightly higher and more erratic during the winter due to the occurrence of more low and high pressure movements over the area. The average barometric pressure is slightly higher for the colder months.

2.3 Hydrology of Reactor Site

2.3.1 Human Consumption

While about 55 percent of the population of the state of North Carolina depends on wells as a source of water, surface water provides the principal supply to Wake County in which the reactor site is located.[2-16] Figure 2-11 shows the major local streams and lakes.

2.3.2 Surface Water[2-17]

The natural drainage from the site is to the south into the Rocky Branch Creek located in a well defined valley about 1,700 feet (518 m) from the reactor. Rocky Branch Creek empties into Walnut Creek at a point about 2.25 miles (3.62 km) downstream from Lake Raleigh. Walnut Creek is a tributary of the Neuse River, which empties into the Atlantic Ocean. The Walnut Creek catchment area contains both Lake Raleigh (81 acres (328 000 m^2)) and Lake Johnson (174 acres (704 200 m^2)), which have a combined water storage of 840 million gallons (3.2×10^9 ℓ). The only other drainage area in the proximity of the reactor site is the Swift Creek catchment basin, which covers 67 square miles (1.7×10^8 m^2) lying south and southwest about 6 to 10 miles (9.7 to 16.1 km) from the NCSU campus. Swift Creek flows through Lake Wheeler and Lake Benson to the Neuse River. Lake Benson is located 9 miles (14 km) south of Raleigh and has a surface area of 490 acres (1 983 000 m^2) and a storage capacity of 1 billion gallons (4×10^9 ℓ).

Lake Benson, Lake Johnson, Lake Raleigh, and Lake Wheeler are no longer used for a primary drinking water supply for the City of Raleigh.[2-18] Instead, the City of Raleigh receives its drinking water supply from Falls Lake which is located approximately 12 miles (19 km) north of the reactor site. The water is supplied through underground pipes and is treated at the E. M. Johnson Water Treatment Plant approximately 8 miles (13 km) north of the reactor site. There is no direct natural drainage from the reactor site to Falls Lake. Lake Benson and Lake Wheeler are considered backup supplies while Lake Raleigh and Lake Johnson serve as recreational facilities.

The daily flow of the Neuse River is measured near Clayton, N.C. about 15 miles (24 km) southwest of Raleigh by U. S. Geological Survey personnel.[2-19,2-20] Since the Falls Lake Dam located northwest of Raleigh went into operation in December 1983, the flow of the river has been regulated. Data for the period of 1983 through 1994 as well as historical flow data for the Neuse River near Clayton, North Carolina are given below:

NEUSE RIVER FLOW RATES (1983 - 1994)

	Flow (gal/day)	Flow (liters/sec)	Date
Minimum	88×10^6	3.9×10^3	September 1985
Maximum	$3,170 \times 10^6$	1.39×10^5	March 1989
Mean	663×10^6	2.90×10^4	--

NEUSE RIVER HISTORICAL EXTREMES

	Flow (gal/day)	Flow (liters/sec)	Date
Minimum	28×10^6	1.2×10^3	15 Sep 1932
Maximum	$14,800 \times 10^6$	6.48×10^5	19 Sep 1945

The flow of streams in the Piedmont section can be described as average slow when compared with the rest of the country. The time constant, or surging flow, cannot be applied to stream flow due to widespread variability. A better description would be critical or noncritical; it can be said that flow reaches criticality (turbulent rather than laminar) at many times, and probably exists as critical more often than not.

2.4 Geology and Ground Water

2.4.1 Ground Water[2-21]

In the immediate North Carolina State University campus area, the soil is mostly moderately impervious Cecil Clay loam. The mineralogy of the clay fraction is approximately 50 percent kaolinite, with 40 percent subsidiary vermiculite and 10 percent free iron oxides.[2-22] The cation exchange capacity of the surface soil is about 5 milliequivalents per 100 grams of soil, and increases to only 8 to 10 milliequivalents per 100 grams of subsoil. Thus, the soil would generally have a low capacity for decontaminating water by cation exchange.[2-23] The coarse fraction which makes up about 25 percent to 30 percent of the soil is almost entirely quartz. The soil varies from permeable to virtually impermeable, i.e., from loose sand to tight alloy. The average depth of the soil on the NCSU campus is about 20 feet (6 m) with a range of from about 5 (1.5 m) to 50 feet (15.2 m) to bedrock which is predominately micagneiss and contains some micaschist.

Ground waters move slowly through the soil under the influence of gravity ranging in rate from a few feet per year to a few feet per day. After percolating downward through the pore space in the soil, the ground water is shunted almost laterally by the bedrock and discharges into Rocky Branch Creek, which has been previously described. Wells in the surrounding area provide generally small, sometimes moderate, supplies of water of only a few gallons per minute. According to the Wake County Health Department, there are no major wells that are still in service located between the reactor site and Rocky Branch Creek to the south.[2-24]

In view of the separation of surface drainage, and since prevailing winds are westerly or southwesterly, it does not seem likely that significant amounts of radioactive contamination originating from the PULSTAR Reactor could occur in the Raleigh Metropolitan major water supply.

2.4.2 Test Boring Data

Eight deep and three shallow test borings have been made at the reactor site.[2-25] In general, the data indicates a dark brown stratum of sandy silt beneath the topsoil overlaying variable color strata of coarse to fine sand with a trace of silt. Sandy strata of micaceous disintegrated rock weathered in place where encountered at the bottom of all the test borings before reaching the bedrock. Typical test borings data are shown in Figure 2-12.

2.5 Seismological Data for North Carolina

North Carolina is usually considered to be a nonseismic state. The United States Geological Survey has listed 3 earthquakes that are significant. They are the following:

- February 21, 1916, felt over 200,000 square miles (5×10^{11} m^2)
- October 20, 1924, felt over 56,000 square miles (1.5×10^{11} m^2) (not recorded on seismographs)
- November 20, 1928, felt over 40,000 square miles (1×10^{11} m^2)

These three earthquakes had a maximum intensity of at least VI by the modified Mercalli Scale. Table 2-7 details earthquakes that have been documented within North Carolina during the period from 1776 to present.[2-26,2-27] Figure 2-6 shows the map of North Carolina on which epicenters of earthquakes listed in Table 2-7 could be located. It is to be noted that no recent epicenters have been recorded in the immediate vicinity of the North Carolina State University PULSTAR Reactor site. No additional earthquakes of intensity comparable to the three listed above have been recorded to date.

This immediate area of the state had no recent displacements of faults in the earth.[2-28] The nearest faulted area of any notable size is the Deep River Coal Field located 45 miles (72 km) southwest of Raleigh, North Carolina.[2-29] The faults in this basin were formed in the latter Triassic times, estimated to be from 155 to 165 million years ago. There is a very minor fault, which may have been formed during Triassic times on the NCSU campus approximately 1,300 feet (400 m) from the reactor site.[2-30] There has been no additional information found which would lead one to believe that there has been any faults in this area since that time. The historical record indicates that the reactor site is safe from damage of major consequence from earthquakes.

2.6 Summary

In summary, the reactor site is free from natural hazards of wind and earthquake, and a ground release of radioactive material would not tend to contaminate local drinking water supplies. Airborne releases would be small due to retention in the filter system and would pass from the stack to the atmosphere with little aerodynamic effect from nearby buildings. Consequently, safe levels of airborne contamination would be maintained during an inadvertent release.

2.7 References

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TABLE 2-2
Population and Number of Type and Use Units Surrounding PULSTAR Reactor Site

Codes: POP - Population; RES - Living Units; DCC - Day Care Center; BNK - Bank Building; BWL - Bowling; GRG - Garage Building; GAS - Gas Station; HTL - Hotel; MTL - Motel; OFF - Office Building; RST - Restaurant; STB - Store Building; SHP - Shopping Center; STC - Store Combination; OTH - Other Building; IND - Industrial Bldg

MAP GRID	POP	RES	DCC	BNK	BWL	GRG	GAS	HTL	MTL	OFF	RST	STB	SHP	STC	OTH	IND
0772-5 Total	14,974	6,503	3	4	1	15	1		6	14	14	33	3	1	28	63
0781-6 Total	27,670	12,017	5	7	1	23	3		13	145	19	60		2	38	124
0791-6 Total	39,618	17,206	6	18	1	25	9	10	3	5	39	54	1	104	37	84
1701-6 Total	49,645	21,561	3	35		143	18	3	18	610	52	364	3	54	50	526
1711-6 Total	46,671	20,269	10	26	1	83	17	3	43	412	62	211		7	44	312
1723-4 Total	6,866	2,982	2	3		20	2		3	41	12	27			25	97

TABLE 2-3

Mean Geostrophic Winds (mph)
At 700 mb over Raleigh, North Carolina

Month	(direction/speed)	
January	WEST	38
February	WEST	20
March	WEST	29
April	WEST	29
May	WEST	18
June	WEST	17
July	NORTHWEST	21
August	SOUTHWEST	18
September	WEST	19
October	WEST	24
November	WEST	24
December	WEST	26

TABLE 2-4

Average Climatological Data
for Raleigh-Durham International Airport
January 1990 - June 1994

MONTH	JAN	FEB	MAR	APR	MAY	JUN
TEMPERATURE						
Average Daily Maximum (°F)	52.9	56.9	63.5	73.0	78.1	85.8
(°C)	15.1	13.8	17.5	22.8	25.6	29.9
Average Daily Minimum (°F)	32.5	34.7	40.3	47.9	56.3	64.6
(°C)	0.28	1.50	4.61	8.83	13.5	18.1
RELATIVE HUMIDITY (%)						
Hour 0100 (Local Time)	76	70	70	75	83	86
Hour 0700	82	77	82	82	85	87
Hour 1300	56	51	49	45	54	55
Hour 1900	65	57	55	54	65	66
PRECIPITATION (Inches)						
Water Equivalent Total	3.81	2.39	4.92	2.17	3.73	2.15
WIND						
Resultant Direction (degrees)	271	287	257	227	206	191
Resultant Speed (mph)	2.2	1.6	9.2	2.2	0.7	1.4
Average Speed (mph)	8.1	8.6	8.6	8.8	8.0	6.7

TABLE 2-4 (continued)

Average Climatological Data
for Raleigh-Durham International Airport
July 1990 - December 1994

MONTH	JUL	AUG	SEP	OCT	NOV	DEC
TEMPERATURE						
Average Daily Maximum (°F) (°C)	90.6 32.6	84.6 29.2	82.0 27.8	72.3 22.4	63.9 17.7	55.1 12.8
Average Daily Minimum (°F) (°C)	70.6 21.4	67.2 19.6	61.2 16.2	48.5 9.17	40.8 4.89	35.2 1.78
RELATIVE HUMIDITY (%)						
Hour 0100 (Local Time)	88	90	87	87	79	77
Hour 0700	89	92	91	91	84	80
Hour 1300	57	60	55	53	49	55
Hour 1900	69	73	74	77	67	65
PRECIPITATION (Inches)						
Water Equivalent Total	4.54	3.62	2.23	3.69	2.25	2.63
WIND						
Resultant Direction (degrees)	214	144	072	032	302	305
Resultant Speed (mph)	2.7	0.7	0.4	1.2	0.8	2.1
Average Speed (mph)	7.0	6.0	6.3	6.6	7.2	8.0

TABLE 2-5

Wind and Precipitation Data for Raleigh-Durham
January 1990 - December 1994

	Resultant Wind Direction (degrees)	Resultant Wind Speed (mph)	Average Speed (mph)	Average Number of Precipitation Events/Year
January				
0100	275	1.1	7.0	6.8
0400	305	1.8	6.9	6.6
0700	324	1.4	6.8	6.8
1000	302	2.5	9.2	6.2
1300	292	3.9	10.3	5.2
1600	278	3.6	10.1	4.4
1900	276	1.0	7.7	5.6
2200	271	1.0	7.6	6.2
February				
0100	299	1.0	8.0	4.8
0400	312	0.6	6.9	3.2
0700	335	1.2	7.1	5.4
1000	306	2.5	10.7	5.8
1300	280	3.5	10.7	4.8
1600	277	4.0	10.7	3.8
1900	223	1.1	7.4	4.0
2200	183	0.7	7.8	3.8
March				
0100	199	1.0	7.7	6.2
0400	235	0.1	7.8	5.8
0700	232	0.5	7.4	5.8
1000	288	2.4	11.0	5.2
1300	265	4.1	11.0	4.6
1600	256	4.7	11.1	4.2
1900	239	1.2	7.8	4.8
2200	208	0.8	8.1	5.0
April				
0100	194	2.1	7.9	4.0
0400	188	1.7	6.5	4.2
0700	213	0.9	6.9	3.0
1000	273	2.9	10.8	2.6
1300	247	4.3	11.2	2.2
1600	240	4.7	10.9	3.6
1900	205	1.5	7.8	2.6
2200	173	2.1	7.9	2.8

TABLE 2-5 (continued)

Wind and Precipitation Data for Raleigh-Durham
January 1990 - December 1994

	Resultant Wind Direction (degrees)	Resultant Wind Speed (mph)	Average Speed (mph)	Average Number of Precipitation Events/Year
May				
0100	193	1.9	6.9	5.0
0400	152	0.6	6.1	5.0
0700	017	0.1	7.1	4.8
1000	292	1.4	9.4	2.8
1300	248	2.1	10.0	3.0
1600	237	2.4	10.0	3.6
1900	147	0.9	7.1	3.8
2200	141	2.0	6.7	3.2
June				
0100	171	2.5	5.6	2.8
0400	196	1.8	4.9	2.0
0700	223	1.4	6.0	2.6
1000	262	1.3	8.2	2.4
1300	220	1.7	8.7	2.8
1600	225	1.8	9.0	2.6
1900	153	1.8	6.4	4.2
2200	140	2.3	5.6	2.6
July				
0100	197	2.9	5.8	4.6
0400	206	2.7	5.0	2.6
0700	228	2.5	5.8	2.2
1000	258	2.8	8.1	1.6
1300	238	3.6	8.9	1.6
1600	212	3.4	9.7	3.8
1900	188	2.2	6.7	5.0
2200	175	3.1	5.7	3.8
August				
0100	159	1.3	4.7	3.8
0400	210	0.4	4.2	3.0
0700	001	0.3	5.1	3.2
1000	359	0.6	7.4	3.0
1300	194	0.9	7.9	4.0
1600	156	1.5	7.9	5.2
1900	125	1.3	5.7	4.2
2200	131	2.1	4.6	3.6

TABLE 2-5 (continued)

Wind and Precipitation Data for Raleigh-Durham
January 1990 - December 1994

	Resultant Wind Direction (degrees)	Resultant Wind Speed (mph)	Average Speed (mph)	Average Number of Precipitation Events/Year
September				
0100	112	0.9	4.9	2.6
0400	017	0.4	4.6	2.8
0700	009	1.1	4.8	2.4
1000	002	1.4	8.2	2.2
1300	276	0.4	8.5	2.6
1600	181	0.3	8.2	2.4
1900	120	1.9	5.3	2.4
2200	128	2.3	5.1	3.2
October				
0100	076	1.1	5.3	3.6
0400	023	1.6	4.9	4.2
0700	016	2.6	5.3	4.0
1000	011	1.6	8.8	2.8
1300	234	0.7	9.1	3.4
1600	002	1.9	8.5	3.2
1900	097	1.5	5.2	3.4
2200	105	1.8	5.7	3.6
November				
0100	311	0.4	6.0	3.4
0400	322	1.0	5.6	4.0
0700	337	1.1	5.8	4.0
1000	330	2.0	9.4	3.0
1300	284	2.4	9.7	4.0
1600	134	1.8	9.0	4.2
1900	156	1.3	6.5	3.8
2200	192	0.8	6.6	3.8
December				
0100	302	1.2	7.0	6.2
0400	305	2.1	6.9	6.8
0700	317	1.9	7.1	6.8
1000	324	3.0	9.7	6.0
1300	303	3.7	10.3	3.8
1600	295	2.7	9.4	4.0
1900	289	0.7	6.8	4.8
2200	297	1.1	7.3	6.2

TABLE 2-6

Percent of Calm Frequency[2-12]

Month	Percentage Calm
January	6.4 %
February	5.8 %
March	4.6 %
April	4.6 %
May	6.2 %
June	9.2 %
July	6.4 %
August	11.3 %
September	11.4 %
October	12.9 %
November	12.0 %
December	7.6 %

TABLE 2-7
North Carolina Earthquakes
1776 through 1994

Month / Day		Year	Magnitude	Latitude	Longitude
11	05	1776	3.3	35.2 N	83.0 W
11	09	1787	2.7	36.1 N	80.2 W
12	13	1808	2.7	35.8 N	78.6 W
11	27	1811	3.3	36.1 N	80.2 W
08	23	1823	2.7	36.1 N	80.2 W
11	11	1826	2.7	36.1 N	80.2 W
05	11	1827	3.3	36.1 N	81.2 W
11	29	1834	2.7	36.1 N	80.2 W
03	30	1850	3.5	35.4 N	78.0 W
08	11	1851	3.5	35.6 N	82.6 W
08	31	1861	5.1	36.1 N	81.1 W
04	16	1871	3.5	34.3 N	78.0 W
04	21	1871	2.7	36.4 N	78.6 W
02	10	1874	3.5	35.7 N	82.1 W
02	22	1874	3.3	35.7 N	82.1 W
03	17	1874	3.3	35.7 N	82.1 W
03	26	1874	3.3	35.7 N	82.1 W
04	14	1874	3.3	35.7 N	82.1 W
04	17	1874	3.3	35.7 N	82.1 W
04	26	1877	2.7	35.2 N	83.4 W
11	23	1878	2.7	35.1 N	84.0 W
12	13	1879	2.7	35.2 N	80.8 W
12	13	1879	3.3	35.2 N	80.8 W
01	28	1880	2.7	35.7 N	82.1 W
01	29	1880	2.7	35.7 N	82.1 W
02	10	1880	2.7	35.7 N	82.1 W
01	08	1882	3.3	34.6 N	76.5 W
10	15	1882	2.7	35.1 N	84.0 W
10	23	1882	3.3	35.1 N	77.0 W
09	21	1883	3.5	36.1 N	79.8 W
01	18	1884	3.5	34.3 N	78.0 W
01	18	1884	3.5	34.3 N	78.0 W
07	00	1884	2.7	35.7 N	82.5 W
08	06	1885	3.5	36.2 N	81.6 W
10	07	1895	3.5	35.9 N	77.5 W
02	11	1896	3.3	36.3 N	78.6 W
02	11	1898	2.7	35.8 N	78.6 W
10	29	1915	3.3	35.8 N	82.7 W
10	29	1915	3.4	35.8 N	82.7 W

TABLE 2-7 (continued)

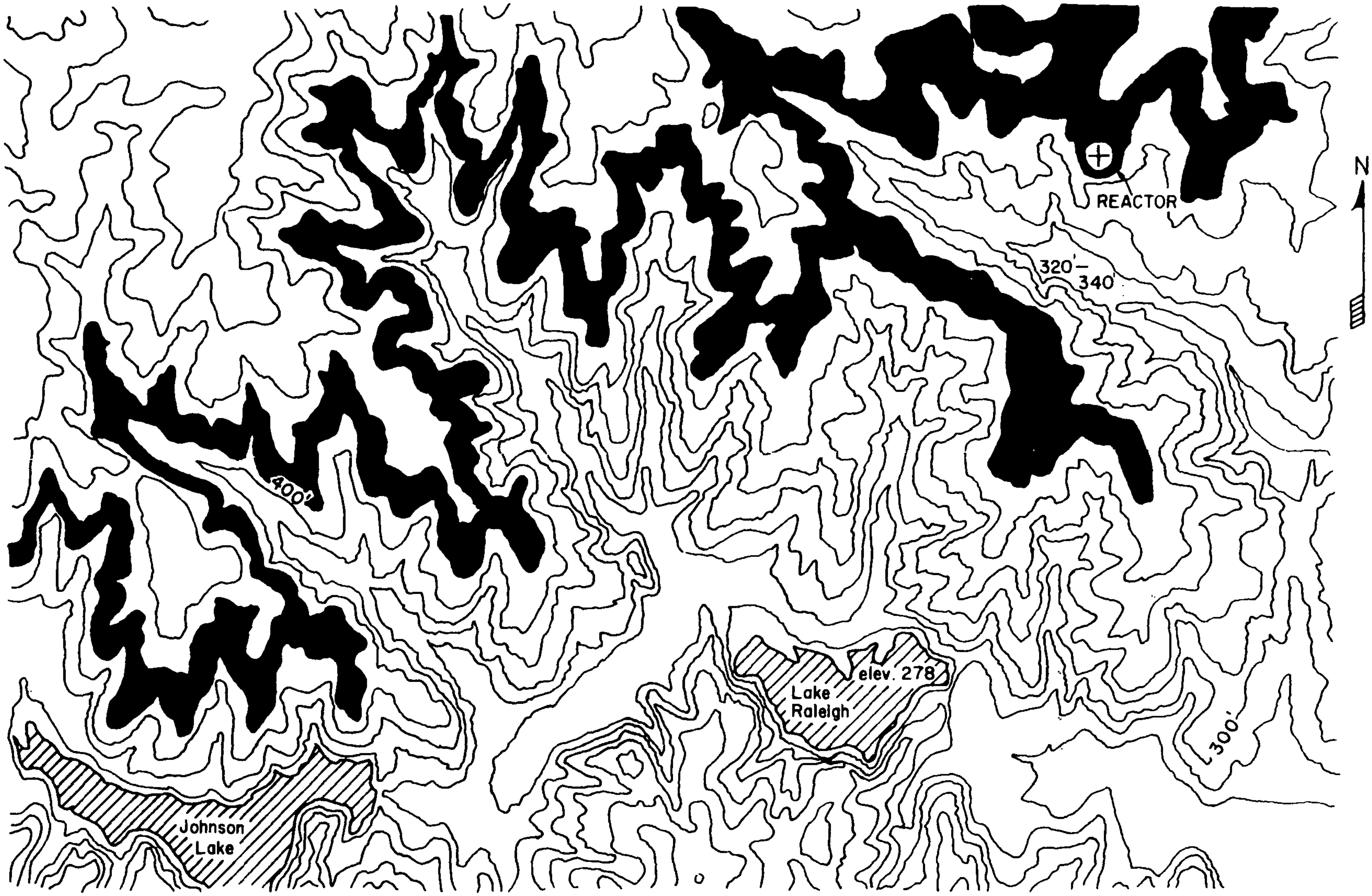
Month	Day	Year	Magnitude	Latitude	Longitude
02	21	1916	5.5	35.5 N	82.5 W
08	26	1916	3.7	36.0 N	81.0 W
07	08	1926	5.2	35.9 N	82.1 W
10	27	1927	3.3	36.3 N	76.2 W
11	23	1927	3.3	33.9 N	78.0 W
11	20	1928	3.7	35.8 N	82.3 W
12	23	1928	3.3	35.3 N	80.3 W
01	01	1935	3.8	35.1 N	83.6 W
01	01	1936	2.7	35.1 N	84.0 W
09	06	1936	2.7	35.3 N	80.2 W
12	25	1940	2.7	35.9 N	82.9 W
12	25	1940	3.3	35.9 N	82.9 W
12	26	1940	2.7	35.9 N	82.9 W
05	10	1941	3.3	35.6 N	82.6 W
05	13	1957	4.1	35.8 N	82.1 W
07	02	1957	3.7	35.6 N	82.7 W
11	24	1957	3.9	35.0 N	83.5 W
03	05	1958	3.5	34.2 N	77.8 W
05	16	1958	3.3	35.6 N	82.6 W
01	03	1960	3.3	35.9 N	82.1 W
01	04	1960	2.4	35.9 N	82.1 W
01	20	1964	3.3	35.9 N	82.3 W
11	26	1968	3.3	34.1 N	77.8 W
09	10	1970	3.5	36.0 N	81.4 W
05	29	1971	2.9	36.0 N	82.0 W
05	16	1974	2.7	35.4 N	82.7 W
12	09	1974	2.7	34.2 N	77.2 W
09	25	1977	2.5	36.0 N	82.7 W
02	25	1978	2.2	36.2 N	79.3 W
04	02	1978	1.4	35.6 N	81.2 W
07	09	1978	2.8	35.5 N	82.8 W
08	04	1978	2.3	35.4 N	82.4 W
09	06	1979	3.2	35.3 N	83.2 W
04	09	1980	2.8	34.8 N	79.9 W
04	22	1980	2.2	36.4 N	80.6 W
06	10	1980	2.5	35.5 N	82.8 W
03	04	1981	2.2	35.8 N	79.7 W
04	09	1981	3.3	35.5 N	82.1 W
04	09	1981	2.5	35.5 N	82.1 W
04	10	1981	2.0	35.5 N	82.1 W
05	05	1981	3.1	35.3 N	82.4 W
10	03	1981	1.1	35.6 N	79.4 W

TABLE 2-7 (continued)

Month	Day	Year	Magnitude	Latitude	Longitude
12	04	1981	1.9	35.3 N	84.0 W
07	10	1982	1.1	35.8 N	82.7 W
03	19	1983	1.6	35.3 N	83.3 W
03	25	1983	3.3	35.3 N	82.5 W
09	13	1983	1.7	36.0 N	82.4 W
10	08	1983	1.9	35.2 N	84.2 W
11	15	1983	1.0	36.1 N	79.3 W
11	23	1983	1.7	35.3 N	82.2 W
11	24	1983	0.7	35.8 N	83.0 W
11	27	1983	2.0	36.3 N	81.1 W
11	29	1983	2.7	36.0 N	82.7 W
12	04	1983	1.6	35.6 N	80.0 W
12	05	1983	2.1	35.2 N	82.5 W
02	14	1984	1.8	35.3 N	82.5 W
07	22	1984	1.6	35.2 N	84.2 W
08	27	1984	0.7	35.1 N	84.3 W
09	04	1984	1.5	35.3 N	83.6 W
10	01	1984	2.0	35.3 N	83.6 W
10	22	1984	3.2	36.4 N	81.9 W
12	23	1984	2.2	35.5 N	83.3 W
01	11	1985	1.0	35.3 N	83.3 W
01	11	1985	1.8	35.3 N	83.3 W
03	01	1985	1.2	35.5 N	83.0 W
03	17	1985	1.0	35.2 N	82.5 W
03	19	1985	2.1	35.3 N	82.5 W
04	17	1985	1.4	35.3 N	83.6 W
06	14	1985	1.8	35.3 N	84.0 W
05	18	1986	0.9	35.5 N	83.6 W
06	06	1986	0.8	35.2 N	84.3 W
06	17	1986	2.2	34.9 N	80.3 W
10	01	1986	2.5	35.8 N	80.5 W
01	27	1987	2.1	35.8 N	80.4 W
09	11	1987	2.2	35.1 N	83.0 W
09	19	1987	2.0	35.9 N	82.0 W
10	19	1987	1.0	35.9 N	82.1 W
12	18	1987	2.7	35.1 N	83.0 W
02	18	1988	3.3	35.3 N	83.8 W
03	31	1988	1.7	35.2 N	84.2 W
04	14	1988	0.8	35.5 N	83.4 W
09	10	1988	1.6	35.2 N	84.2 W
04	29	1989	1.9	36.1 N	82.4 W

TABLE 2-7 (continued)

Month	/ Day	Year	Magnitude	Latitude	Longitude
06	18	1989	1.9	35.9 N	82.5 W
06	20	1989	2.0	35.3 N	82.5 W
06	23	1989	0.4	35.1 N	84.2 W
07	01	1989	0.8	35.9 N	82.5 W
08	31	1989	1.1	35.7 N	80.4 W
10	02	1989	0.8	35.5 N	82.9 W
10	08	1989	1.0	36.4 N	81.2 W
10	25	1989	1.7	35.3 N	83.6 W
01	02	1990	2.1	35.1 N	83.0 W
02	22	1990	1.3	36.3 N	81.6 W
03	03	1990	1.8	35.3 N	82.5 W
03	09	1990	1.8	35.9 N	81.7 W
03	16	1990	1.3	35.3 N	83.4 W
03	25	1990	1.2	35.2 N	84.1 W
03	31	1990	0.8	35.3 N	82.5 W
06	11	1990	1.1	35.5 N	83.5 W
08	03	1990	1.6	35.9 N	82.4 W
08	08	1990	0.8	35.9 N	81.7 W
09	18	1990	1.1	35.3 N	83.9 W
09	28	1990	1.9	35.3 N	83.6 W
10	07	1990	1.7	36.1 N	81.9 W
11	07	1990	2.4	35.1 N	83.0 W
11	25	1990	1.5	35.4 N	83.7 W
05	28	1991	2.3	35.3 N	83.6 W
10	07	1991	2.6	36.0 N	82.9 W
12	06	1991	1.8	35.5 N	82.9 W
01	18	1992	2.1	36.0 N	80.1 W
03	12	1992	1.7	35.3 N	83.6 W
06	07	1992	1.9	35.1 N	83.5 W
01	01	1993	3.0	35.9 N	82.1 W
01	28	1993	1.5	35.1 N	84.3 W
07	01	1993	2.3	36.0 N	82.5 W
07	12	1993	2.7	36.0 N	79.8 W
08	06	1994	3.6	35.1 N	76.8 W



TOPOGRAPHIC MAP OF REACTOR SITE

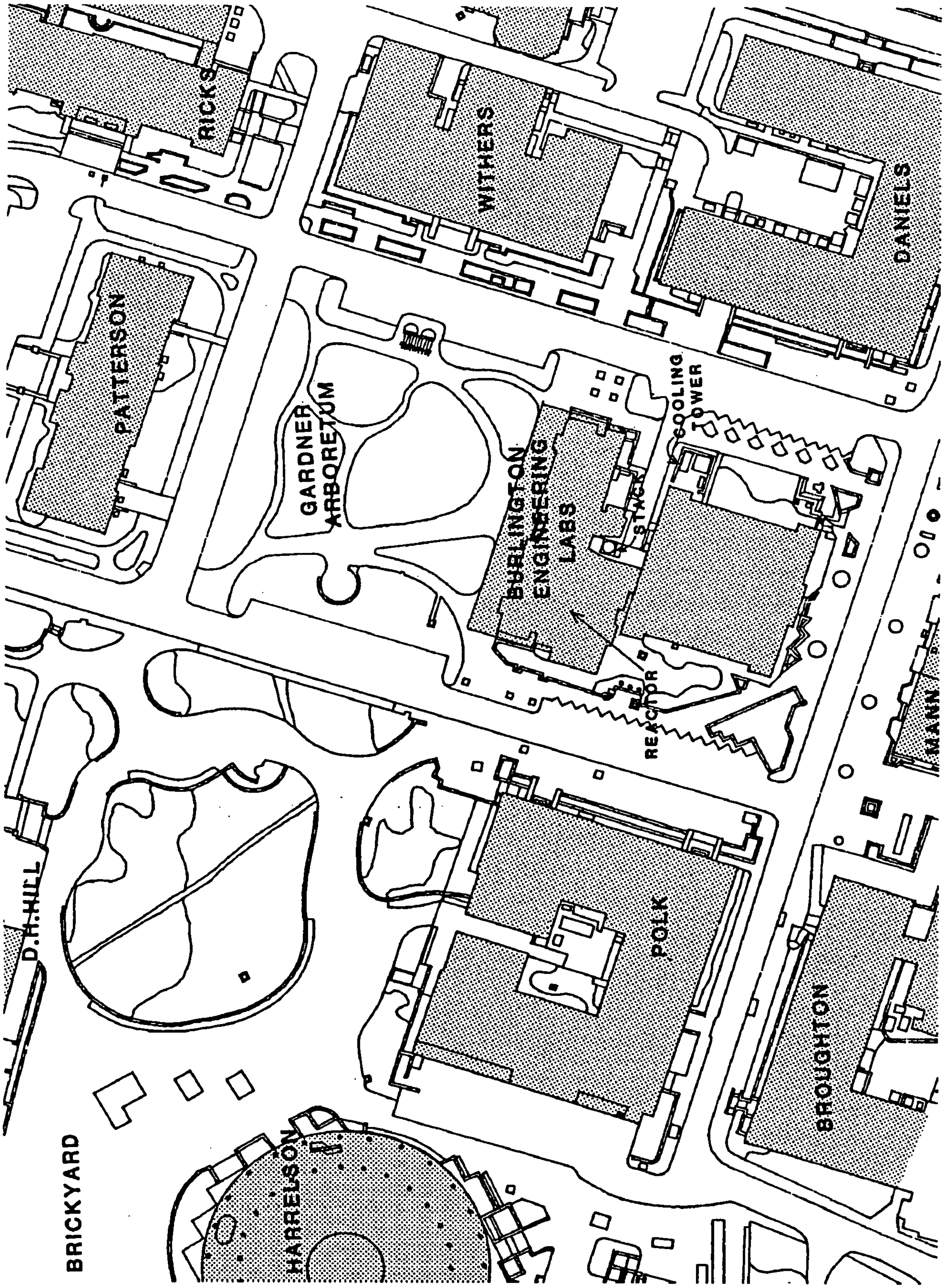
CONTOUR INTERVAL: 20 FEET

SCALE: 1.00 cm = 0.157 miles

NOTE: Shaded Area Denotes Elevation of 400 - 420 feet.

SAR FIGURE 2-2

FIGURE 2-3
Detailed Campus Layout Within Proximity of BEL



RESIDENTIAL: ONE FAMILY TWO FAMILY THREE FAMILY FOUR FAMILY MULTI-FAMILY RES. W/BUSINESS USE	**STORAGE BUILDINGS:** TYPICAL SGL TENANT W/2 OR MORE TENANTS SUPER MARKET DISCOUNT HOUSE DEPARTMENT STORE
BANKS BUILDINGS: BANK BUILDING DRIVE-IN ONLY	**SHOPPING CENTERS:** REGIONAL COMMUNITY NEIGHBORHOOD
BOWLING CENTERS: W/RESTAURANT-LOUNGE W/SNACK BAR ONLY	**STORE COMBINATIONS:** STORE(S) W/APT(S) STORE(S) W/OFFICE(S) STORES/OFCS/APTS TYP-COMBINATIONS OFC/RES. CONVERSION
GARAGE TYPE BLDGS: SALES AND SERVICE SERVICE SHOP OPEN DECK PARKING RAMP GARAGE BUILDINGS CAR WASH WAND CAR WASH	**OTHER FINISHED BLDGS:** AIRPORT TERMINAL CONVENT CLINIC CLUB CHURCH DORMITORY FIRE STATION GYMNASIUM HOSPITAL LIBRARY MOBILE HOME MUNICIPLE BLDG. NURSING HOME POLICE STATION REST HOME SCHOOL THEATRE
GASOLINE STATIONS: TYPICAL STYLE COL., RANCH, ETC. OTHER STYLES	
HOTELS: W/ FULL FACILITIES W/LIMITED FACILITIES	
MOTELS: W/FULL FACILITIES W/LIMITED FACILITIES COTTAGE TYPE	**INDUSTRIAL TYPE BLDGS:** LIGHT MFG. MEDIUM MFG. HEAVY MFG. LIGHT STORAGE MEDIUM STORAGE HEAVY STORGAE COLD STORAGE PLANT BOTTLING PLANT CHEMICAL PLANT DAIRY PLANT FOUNDRY HANGAR POWER HOUSE REFINERY TELEPHONE EXCHANGE TRUCK TERMINAL
OFFICE BUILDINGS: TYPICAL INSURANCE MEDICAL TYPES WITH BANKING FLOOR W/ COMPUTER FLOOR	
RESTARAUNTS: TYPICAL W/LOUNGE SPECIALTY DRIVE-IN PLAIN DRIVE-IN STORE TYPE BLDG	
ROOMING HOUSES: W/ COMMUN KITCHENS W/ LIMITED PRIVILEGES	

SAR FIGURE 2-5

TYPE & USE BREAKDOWN FOR TAX ZONES

U.S. DEPARTMENT OF COMMERCE Economics and Statistics Administration Bureau of the Census

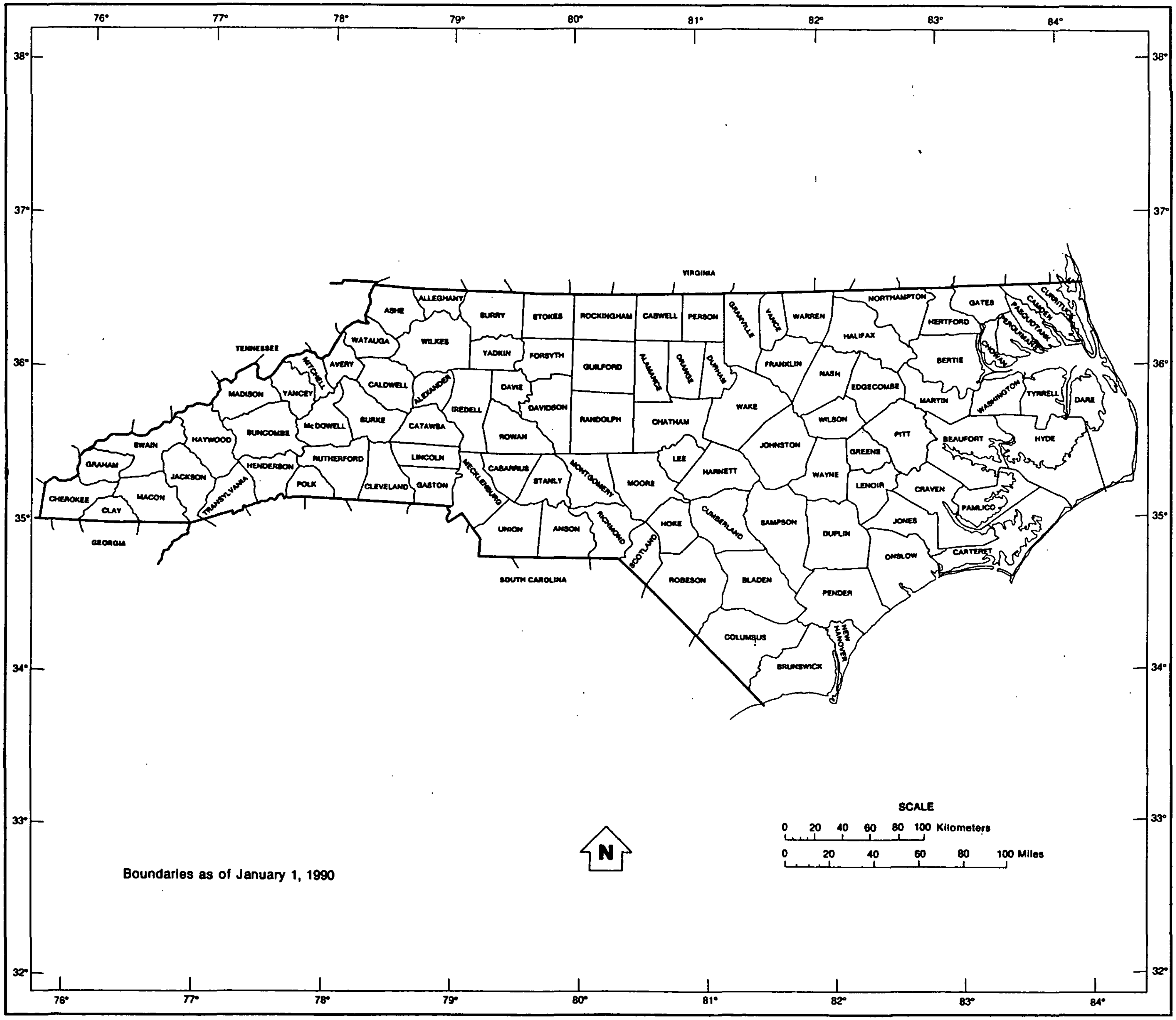
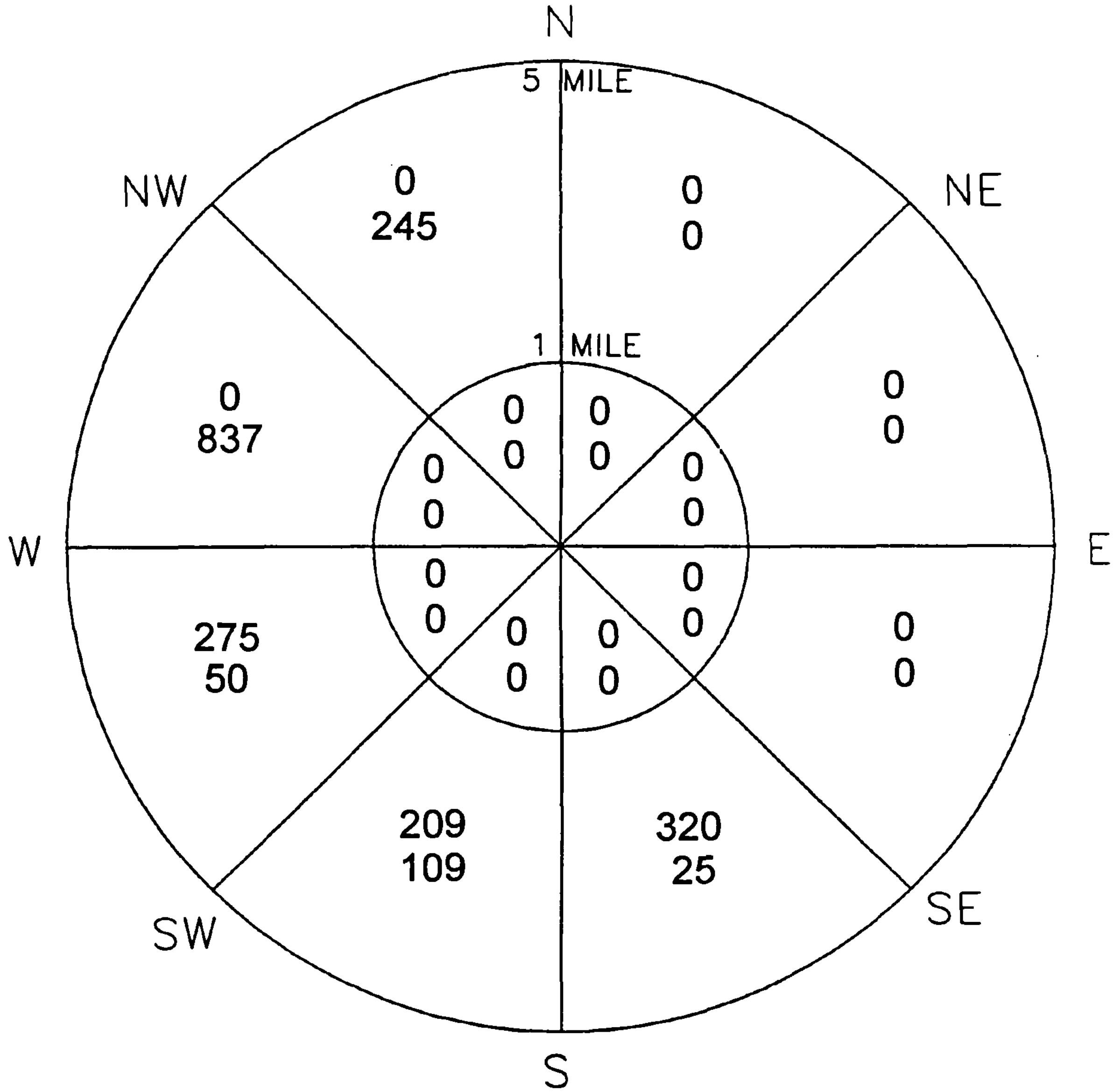


FIGURE 2-6

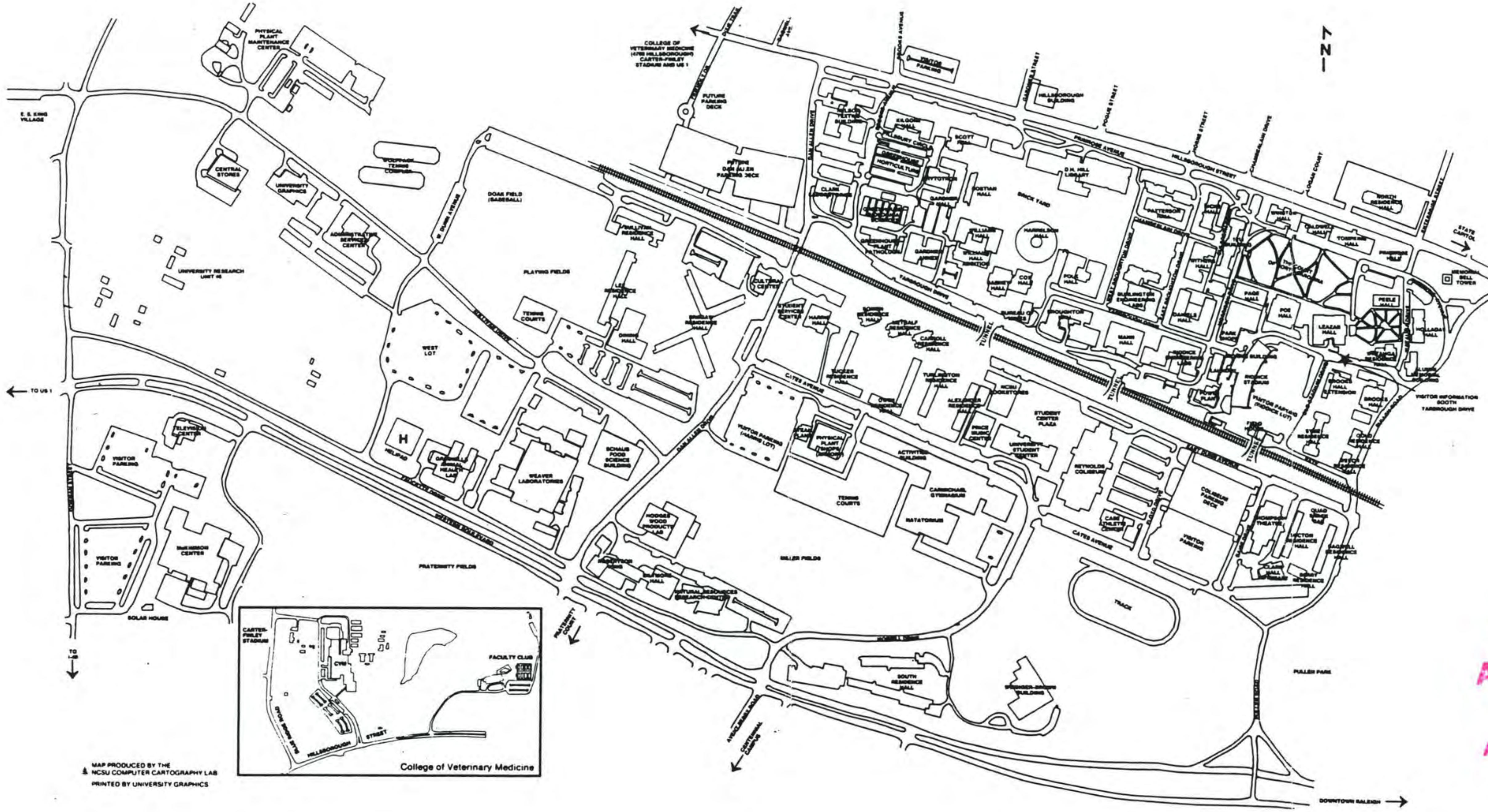
FIGURE 2-7



GRAZING ANIMAL POPULATION DISTRIBUTION

DAIRY CATTLE ONLY

ALL OTHER GRAZING ANIMALS

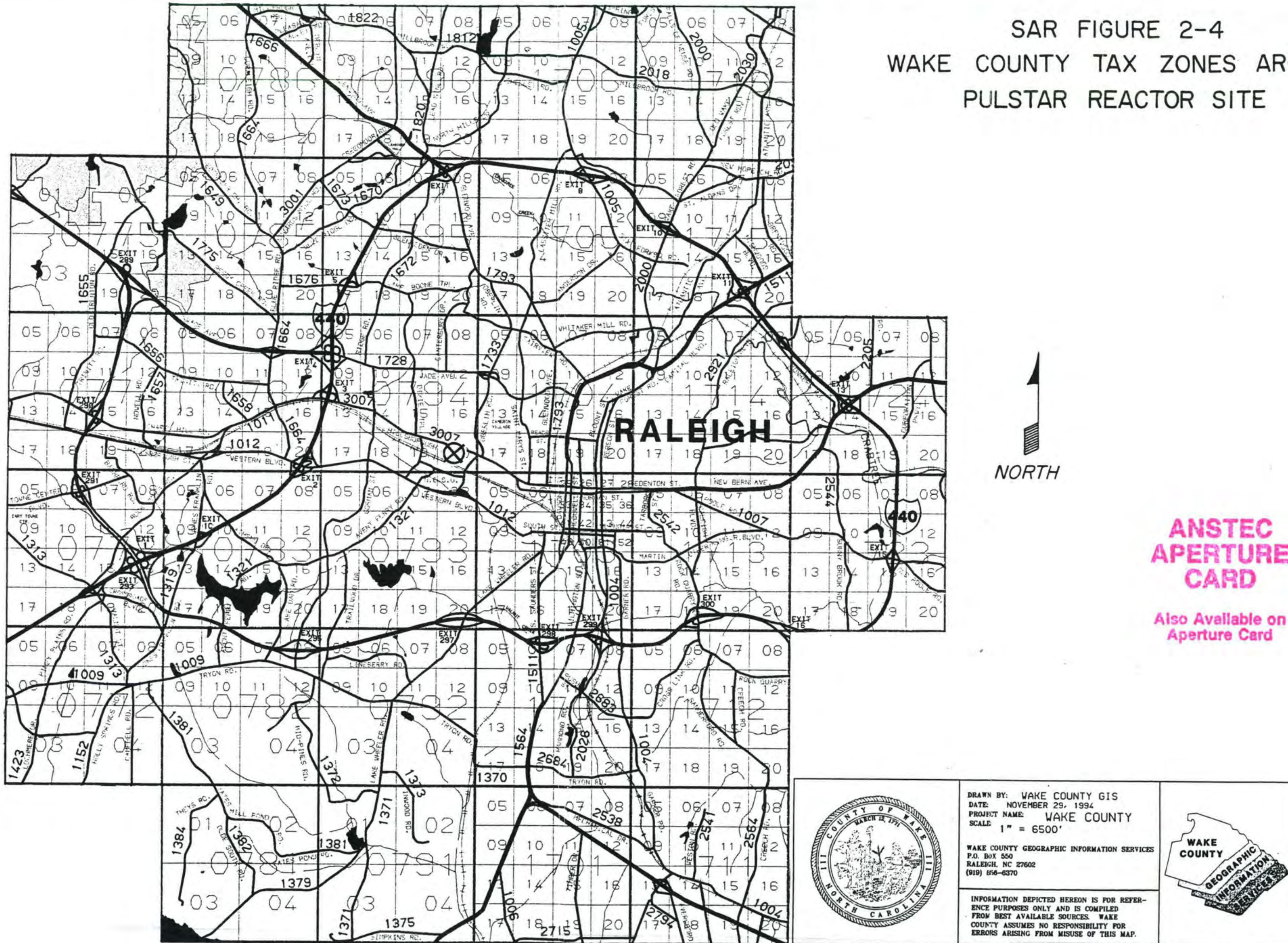


NORTH CAROLINA STATE UNIVERSITY CAMPUS MAP

SAR FIGURE 2-1

9509200199-07

SAR FIGURE 2-4
WAKE COUNTY TAX ZONES AROUND PULSTAR REACTOR SITE



9509200199-08

SAR FIGURE 2-4A

COUNTY OF WAKE EXISTING LAND USE

N
S

ANSTEC APERTURE CARD

Also Available on Aperture Card

LEGEND

AGRICULTURAL		LAKES	
COMMERCIAL		FORESTRY	
INDUSTRIAL		UNKNOWN	
OFFICE		ERROR	
RESIDENTIAL		5 MILE RADIUS	
VACANT			

DATE: MARCH 13, 1995

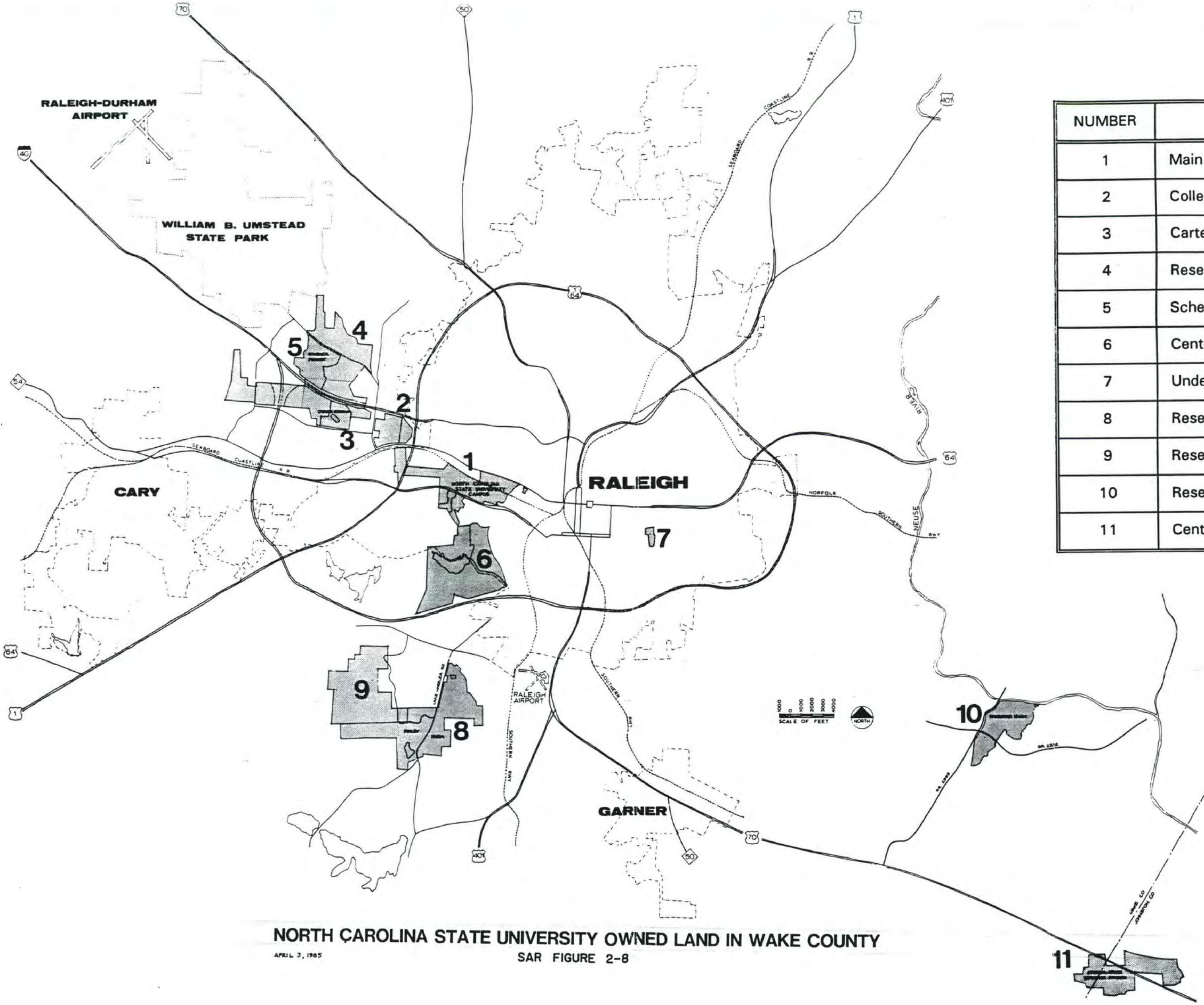
DRAWN BY: WAKE COUNTY
PLANNING DEPARTMENT
P. O. BOX 550
RALEIGH, N.C. 27602

SCALE: 1" = 7,200'

WAKE COUNTY PLANNING DEPARTMENT

INFORMATION DEPICTED HEREON IS FOR REFERENCE PURPOSES ONLY AND IS COMPILED FROM BEST AVAILABLE SOURCES. WAKE COUNTY ASSUMES NO RESPONSIBILITY FOR ERRORS ARISING FROM MISUSE OF THIS MAP.

9509200199-09



KEY

NUMBER	AREA
1	Main Campus
2	College of Veterinary Medicine
3	Carter-Finley Stadium
4	Research Unit 1
5	Schenck Memorial Forest
6	Centennial Campus
7	Undeveloped Land (Gatling Property)
8	Research Unit 2
9	Research Unit 9
10	Research Unit 7
11	Central Crops Research Station

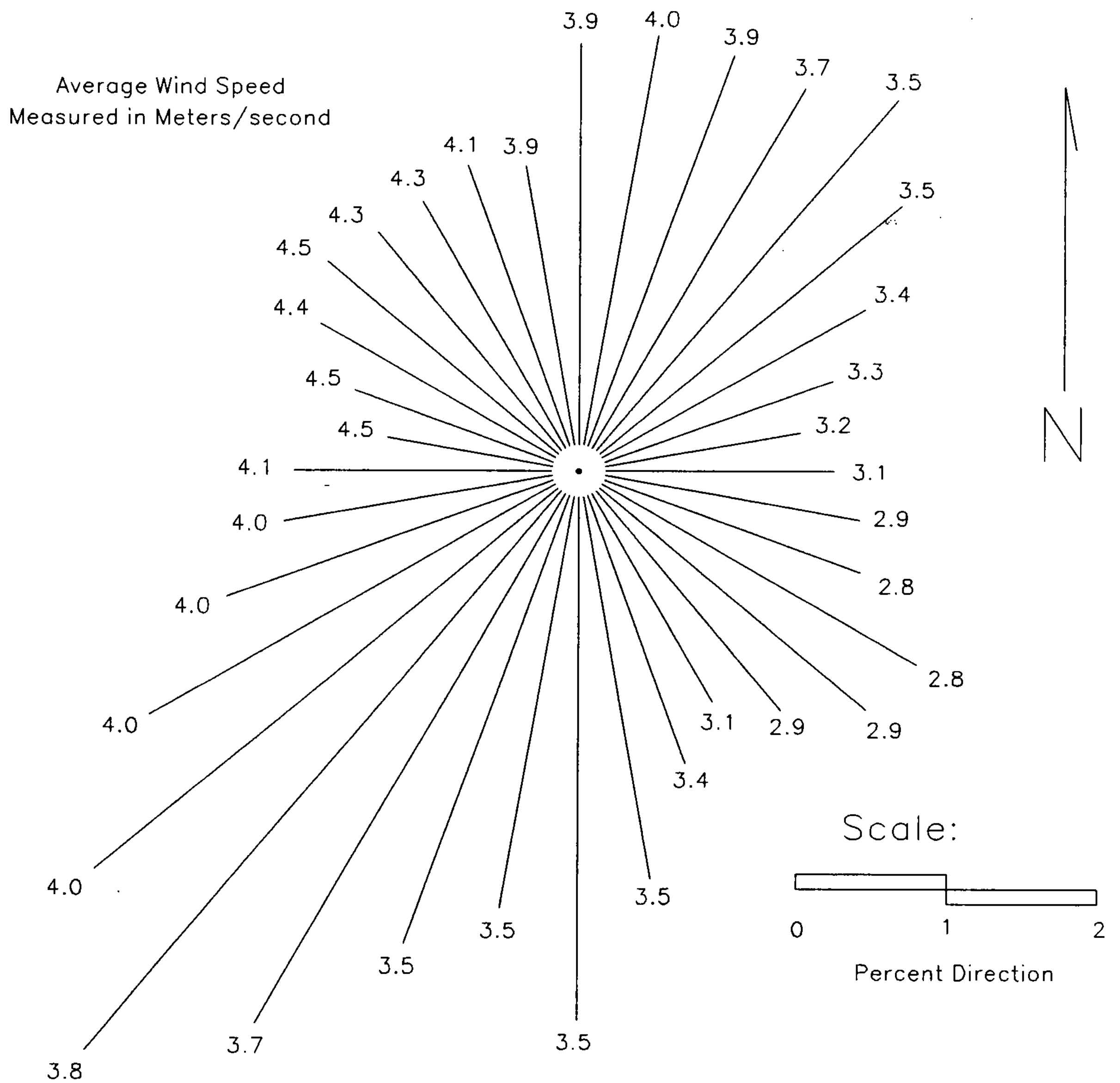
NORTH CAROLINA STATE UNIVERSITY OWNED LAND IN WAKE COUNTY

APRIL 3, 1985

SAR FIGURE 2-8

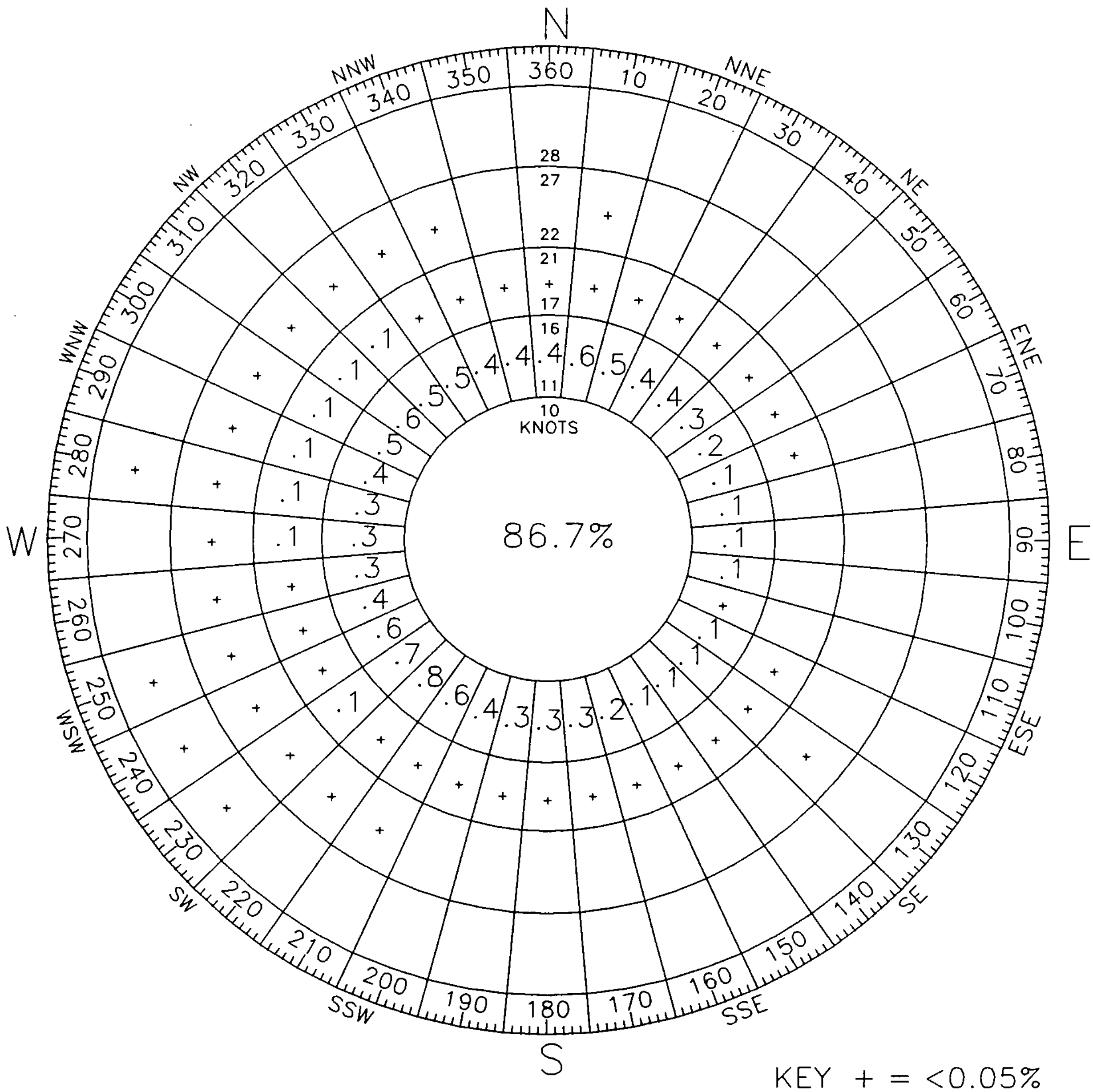
9509200199-10

FIGURE 2-9

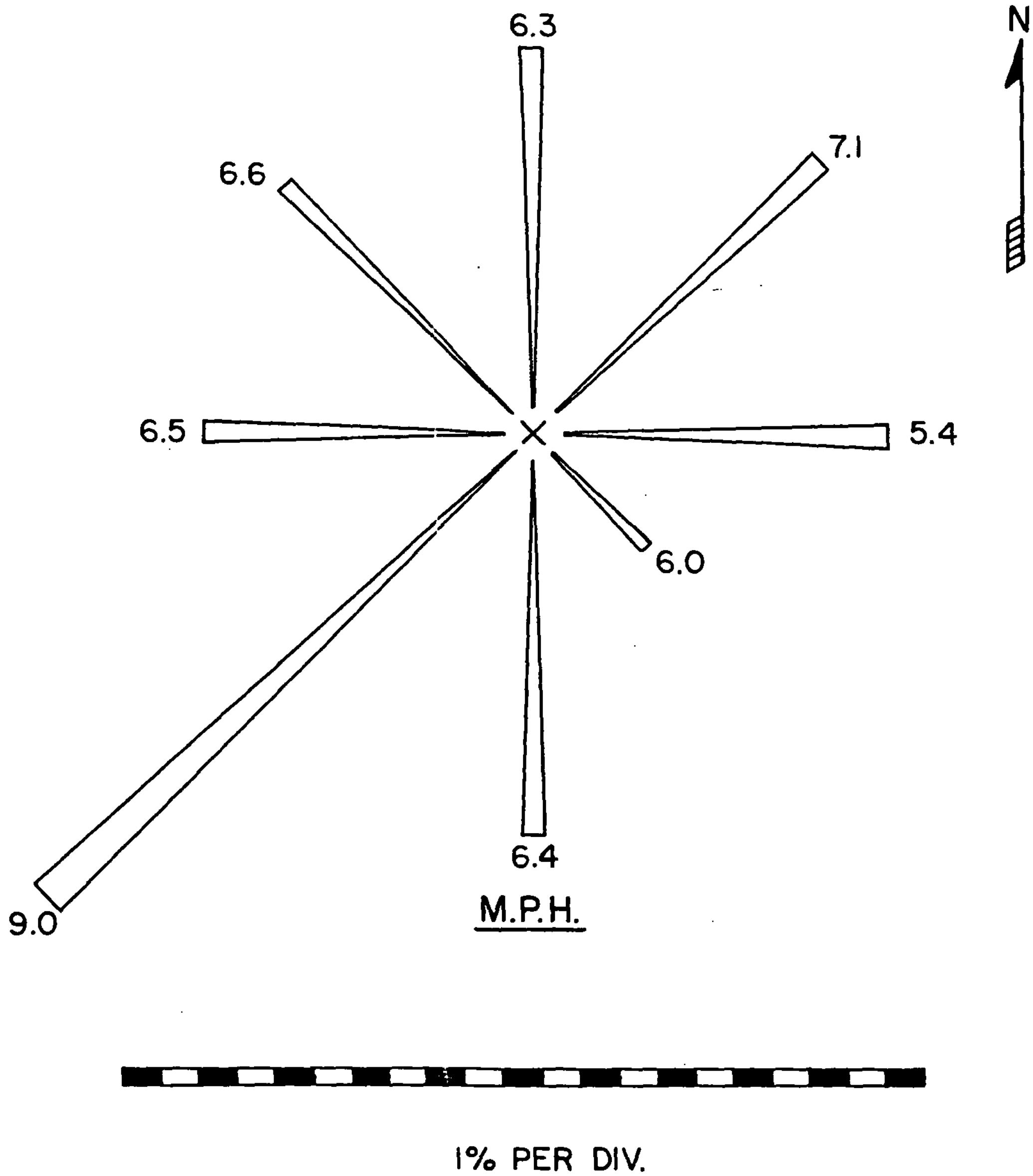


Average Wind Speed and Percent Direction
at Raleigh-Durham International Airport
1990 - 1994

FIGURE 2-9A

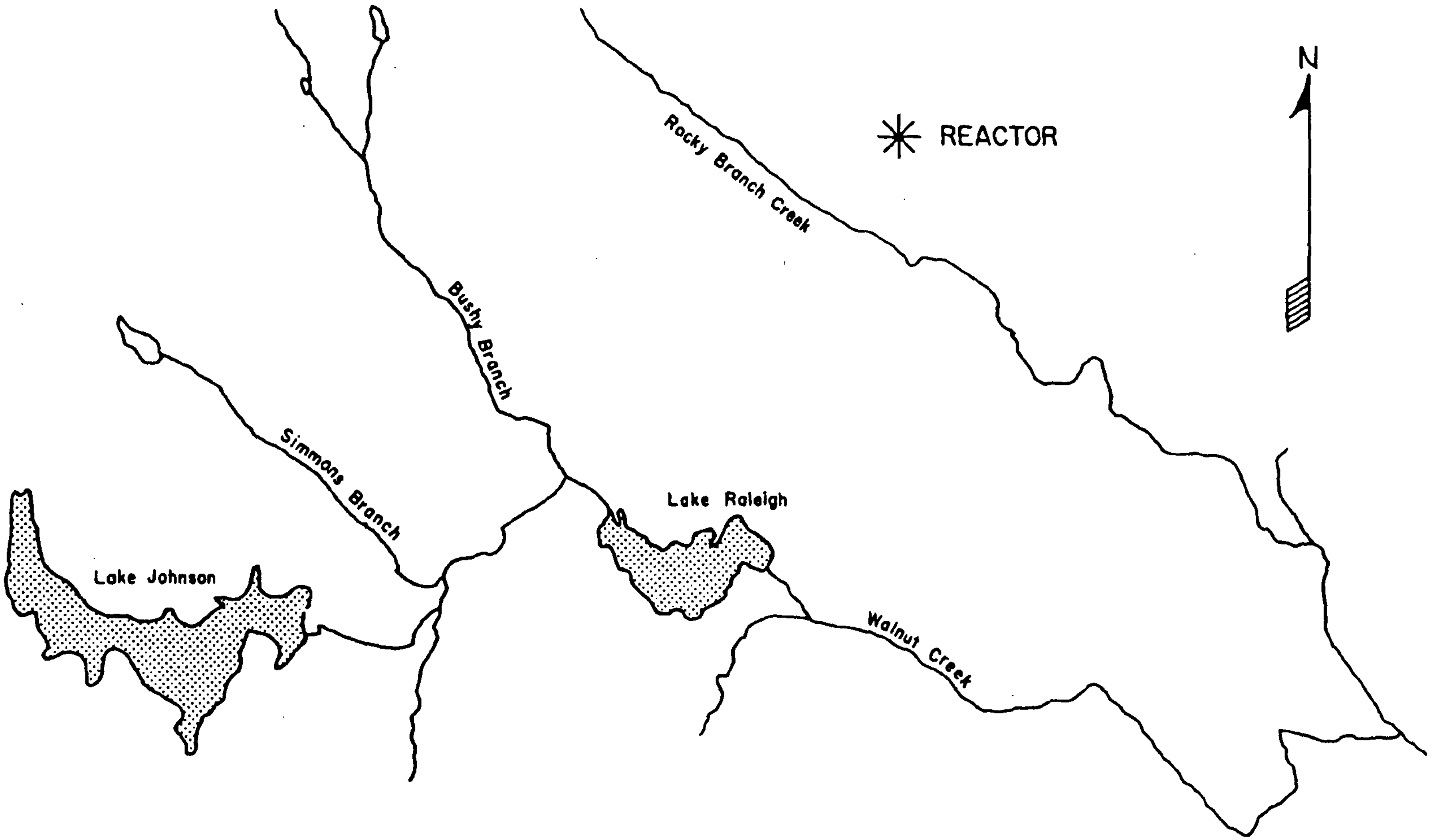


Wind Speed Ranges with Percent Occurrences and Direction
at Raleigh-Durham International Airport
1990 - 1994



AVERAGE WIND SPEED AND % DIRECTION
— OCCURRENCE FOR ALL DATA TAKEN WITH CUP ANEMOMETER AND WIND VANE ATOP RICKS HALL DURING PERIOD 1945 TO 1948.

FIGURE 2-10

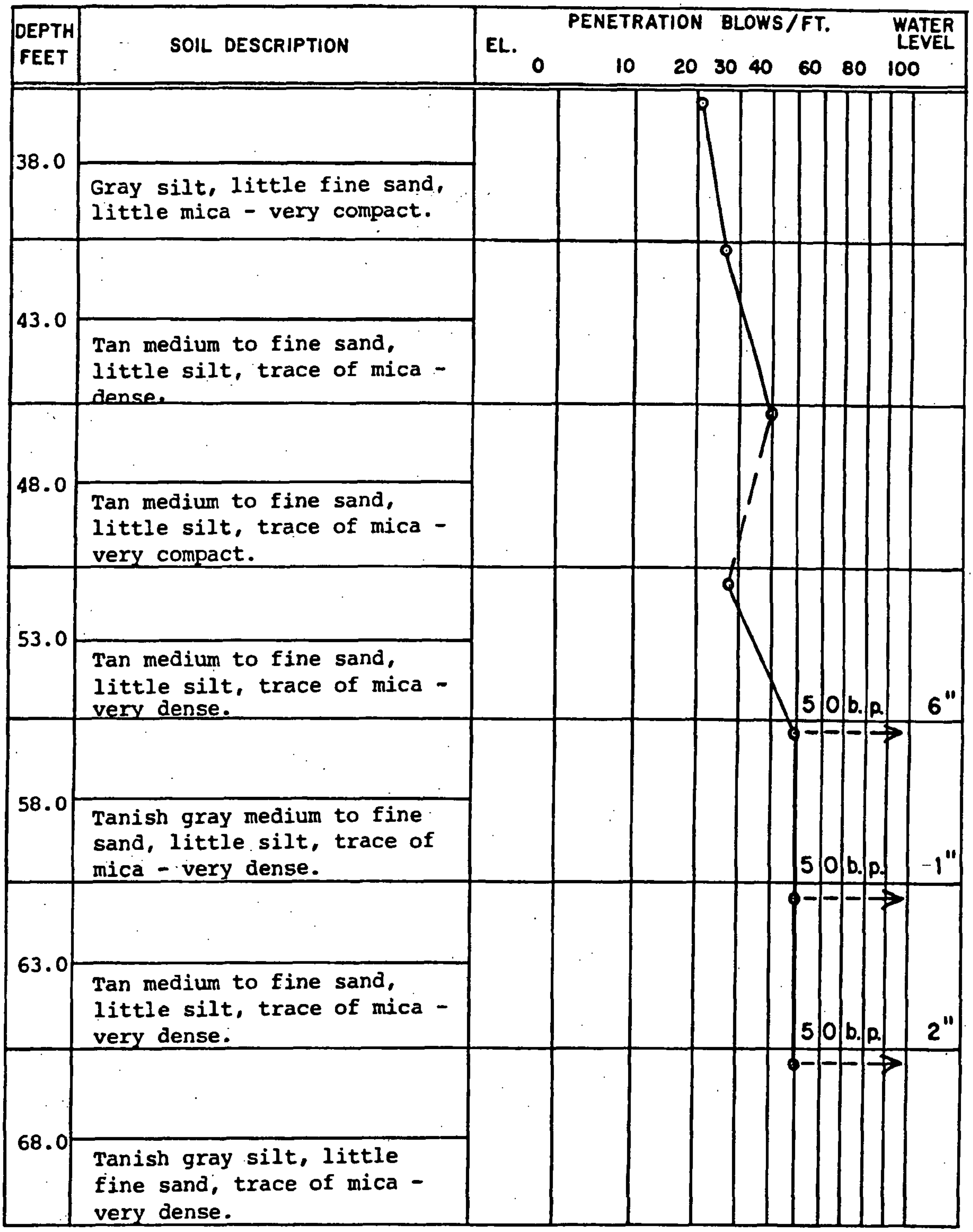


SCALE: 1.0 cm = 1220 feet

SURFACE WATER IN REACTOR SITE VICINITY

FIGURE 2-11

FIGURE 2-12



TYPICAL TEST BORING DATA

3.0 REACTOR

3.1 Design Bases

This section of the Safety Analysis Report shall review start-up test data (as compared to predicted data) for the three core arrangements used during the first licensing interval and address likely changes to the core that will be required in the future to meet reactivity needs as the fuel continues to deplete.

The initial core loading for start-up of the PULSTAR Reactor was the 5 by 5 Standard Core, depicted in Figure 3-8A. This core was comprised of 25 fuel assemblies in a five by five array with four control rods, one of which was a pulsing rod not normally used to control the reactor. As the fuel depleted over the first few years of operation, the core was changed to the 5 by 5 Graphite Reflected Core No. 1, by incorporating five graphite reflectors on the north side of the five by five array of fuel. Figure 3-8B depicts the fuel, graphite and control rod arrangement for the 5 by 5 Graphite Reflected Core No. 1.

The core was rearranged to increase excess reactivity by adding five more graphite reflectors to make the existing 5 by 5 Graphite Reflected Core No. 3, depicted in Figure 3-8C. The specific changes from the 5 by 5 Graphite Reflected Core No. 1 to the 5 by 5 Graphite Reflected Core No. 3 involved:

- Moving the west row of fuel to the east side of the remaining fuel array,
- Addition of five graphite reflectors in the vacant west row on the grid plate, and
- Construction of a "grid plate extension" on the thermal column nosepiece to allow repositioning of the wet exposure ports.

3.1.1 Performance Objectives

The bases upon which the design of the reactor was initially established were that it should be capable of operation at a steady-state power level of 1 MW and be capable of pulsing to a peak power of 2200 MW throughout the reactor lifetime without exceeding any physical or nuclear limitation which could jeopardize either operations personnel, public safety, or the fuel assembly integrity. Twenty years of successful operation of the PULSTAR Reactor at steady-state power levels up to 1 MW have been achieved. The highest pulse peak power achieved has been 980 MW with a corresponding total energy release of 35.4 MW·sec.

The NCSU PULSTAR Facility has elected to discontinue pulsing as of Amendment No. 9 and references to pulsing in this SAR only remain when in support of Section 13 accident scenarios and reactivity limits for experiments.

3.1.2 Design Limits

3.1.2.1 Nuclear Limits

The following nuclear design limits are selected to ensure the safe operation of the reactor under normal or abnormal operating conditions, such that Safety Limits are not exceeded. These nuclear limits are set forth in the following table:

Table 3-1
Nuclear Limits

Nuclear Limit	Values
Reactivity Insertion Mode:	Reactivity Insertion Rate:
Pulse	Not Allowed
Steady-state	0.1%Δk/k/sec (100 pcm/sec)
Square-wave	Not allowed
Power Level	1.3 MW (LSSS)

The shutdown margin for any geometry shall be at least 0.40%Δk/k (400 pcm) with the highest worth control rod completely withdrawn from the reactor and the other two control rods fully inserted into the core. In addition, excess reactivity for the PULSTAR core shall be limited to 3.97%Δk/k (3970 pcm) to further ensure that the 0.40%Δk/k (400 pcm) shutdown margin limit is always satisfied. The operational requirements for the PULSTAR are typically as follows:

Table 3-2
Operational Reactivity Requirements

	Reactivity %Δk/k (pcm)
Xenon (typical operations)	0.60 (600)
Temperature Coefficient	0.14 (140)
Power Defect	0.33 (330)
Total	1.07 (1070)

The 3.97%Δk/k (3970 pcm) limit on excess reactivity is obtained by adding 2.9%Δk/k (2900 pcm) allowance for experiments and 1.07%Δk/k (1070 pcm) for operational requirements.

As for meeting the 0.40%Δk/k (400 pcm) shutdown margin limit, the existing 5 by 5 Graphite Reflected Core No. 3 has a total Gang worth of 9.105%Δk/k (9105 pcm) (see Figure 3-32) and a highest rod worth (Regulating Rod) of 4.185%Δk/k (4185 pcm). With the maximum allowed excess reactivity of 3.97%Δk/k (3970 pcm), the minimum possible shutdown margin would calculate to be:

$$9.105 - 4.185 - 3.970 = 0.950\%\Delta k/k \text{ (950 pcm)}$$

therefore, adequate shutdown margin is assured. No credit is allowed for the pulse rod (also referred to as the shim rod, since pulsing is not permitted) in the shutdown margin calculation. The air supply lines to the pulse rod are disconnected and permanently capped to prevent pulse rod ejection as of Amendment 9. The control rod drive speed is limited to 7.5 inches per minute (0.3 $cm \cdot s^{-1}$) based upon a $\leq$ 0.1%Δk/k/sec ($\leq$ 100 pcm/sec) reactivity insertion limit.

3.1.2.2 Thermal and Hydraulic Limits

The maximum steady-state power level is limited to 250 kW with natural convection cooling (Limiting Safety System Setting). Section 3.2.4 contains additional details on convection cooling. For power levels above 250 kW, the primary coolant pump must be operating with a volumetric flow rate of 500 gpm (31.5 $\ell \cdot s^{-1}$).

The average heat flux at 1 MW for the 25 fuel assembly core is 22,000 Btu/hr-ft^2 (6.94 $w \cdot cm^{-2}$). The maximum heat flux in the core shall not exceed 85,904 Btu/hr-ft^2 (27.1 $w \cdot cm^{-2}$). This criterion limits the maximum average hot spot factor allowed to 3.90. Figures 3-23 and 3-24 form the basis for establishing Safety Limits on the NCSU PULSTAR core. The Safety Limit is actually a function which must be considered over the total power versus volumetric flow range. It must accommodate the entire course of the most adverse anticipated transient. The limiting power level at full primary flow is 3.8 MW. Correcting the Safety Limit measurement and instrument uncertainties results in establishing the upper limit for the "Limiting Safety Systems Settings" (see Appendix 3B). The resistivity of the primary coolant moderator shall be maintained greater than 500 $k\Omega \cdot cm$.

Determination of the hot spot factor for any core configuration employed shall be accomplished prior to operation of the core at 1 MW. This is performed via flux wire activation or miniature fission chamber flux maps. Furthermore, hot spot verification is made when the core geometry is changed by the addition of major in-core facilities.

3.1.2.3 Mechanical Limits

The mechanical limits for the core are such that no fuel pins will have a lateral bow greater than 0.050 inches (0.127 cm) in any 6 inch (15.2 cm) segment, or an increase in length greater than 0.06 inches (0.15 cm) or a diametric increase greater than 0.005 inches (0.013 cm). These limits are set to ensure integrity of the fuel.

Total burn-up on the NCSU PULSTAR fuel shall be limited to 20,000 MWD/tonne U. The BMRC PULSTAR has successfully depleted fuel to a burn-up of 20,000 MWD/tonne

following a successful inspection program at Buffalo involving disassembly and inspection of fuel with burn-ups of 15,000 MWD/tonne [3-22].

The foregoing nuclear, reactivity control, thermal and hydraulic, and mechanical limits set for the core have been based upon the PULSTAR Test which was performed at the Buffalo Materials Research Center, Inc (BMRC) at Buffalo, New York. Results of these tests are published in reports WNY-017 through WNY-023.

3.2 Reactor Design

3.2.1 Summary

The North Carolina State University PULSTAR core is a heterogenous system of light water and UO_2 fuel which provides a source of neutrons for research purposes. The core is immersed under twenty feet (approximately six meters) of water in an open pool, surrounded on the sides and bottom by concrete shielding. Experimental access to core neutrons is provided by five horizontal beam tubes, one through tube, and a thermal column. Core access is available by direct insertion of samples through the pool water into flooded or dry exposure tube on the core periphery. Refer to SAR Figures 3-1 and 3-2 for elevation and plan views of the PULSTAR Reactor.

Control of the power level is accomplished by variable positioning of neutron absorbing rods within the core. These control rods as well as the various neutron detecting chambers used for power level measuring, are suspended from a bridge which spans the reactor pool. The reactor is operated from a console located in a control room near the bridge.

Core heat is removed by the forced circulation of pool water through the core. Before being returned to the pool, this water is cooled and held up for nitrogen-16 decay. A 20 gpm (1.26 $\ell \cdot s^{-1}$) bypass flow of primary water is continuously demineralized and filtered before being returned to the main primary stream flowing to the pool.

The NCSU PULSTAR has a number of design features which make it an extremely safe research reactor. The first of these features is the use of low enriched UO_2 in pellet form as a fuel. This provides an inherent safety shutdown feature due to the Doppler broadening effect.

The heat capacity of UO_2 is quite large, permitting a large release of energy in the core under transient conditions without exceeding the melting point of the cladding. The low thermal diffusivity of UO_2 leads to a long thermal time constant for the fuel (approximately 4 seconds). The long time constant prevents the explosive formation of steam experienced in plate-type metallic reactors undergoing severe reactor transients. The time constant for a typical plate-type aluminum alloyed fuel assembly for instance, is on the order of a few milliseconds.

Considerable experience and information has been gathered on the characteristics of UO_2 under irradiation conditions. The chemical and radiation stability of UO_2 is known to be

excellent. The ability of UO_2 to retain fission products is also excellent and provides a strong motivation for the use of UO_2 aside from the obvious advantages in performance. Use of the fuel in pellet form ensures against the rapid release of energy which could occur through loss of clad integrity, and the subsequent dispersion of fuel into the coolant if the fuel were in powder form.

The nuclear power industry has generated a significant quantity of information on the properties of zircaloy and on operating experience with zirc-clad UO_2 fuel. Therefore, the clad material may be considered a proven material for reactor use. The high melting temperature of zircaloy may be considered a distinct advantage for use in a core designed to be pulsed.

One of the strongest arguments for the inherent safety of the PULSTAR core is its similarity to the SPERT[3-2] oxide core, which has undergone extensive testing. Transients releasing as much as 100 MW·sec of energy have been initiated in the SPERT core without causing damage. The use of sintered pellets instead of powdered fuel is an added safety feature. The only serious defect discovered in SPERT tests was the presence of a double peaked pulse caused by coherent bowing of the fuel pins. The coherent bowing was eliminated when the pins were supported in a fashion that provided an 18 inch (45.7 cm) unsupported length of pin instead of the unsupported length of 6 feet (1.83 m). The PULSTAR fuel is designed to provide an unsupported length of only eight inches so that coherent bowing is not a problem. These supports also serve as wear surfaces to prevent damage to the fuel pin cladding.

A PULSTAR Reactor has been operated at the BMRC in Buffalo, New York, from 1964 to 1995 with fuel burn-ups exceeding 15,000 MWD/tonne U. The performance of the NCSU PULSTAR core in the steady-state and pulsing modes has sufficient backup experience to permit reliable prediction for total energy release and the hot spot factors. Table 3-11 shows the design parameters for the current NCSU PULSTAR Facility.

3.2.2 Mechanical Design

3.2.2.1 Reactor Pool and Shield

The reactor core is positioned in a water filled pool, open at the top and surrounded by concrete at the sides and bottom as shown in Figures 3-1 and 3-2 and fully described in SAR Sections 4 and 10.

3.2.2.2 Experimental Facilities

The experimental facilities are summarized below and are fully described in Section 14 of the SAR.

- One twelve-inch square beam tube
- One eight-inch circular beam tube with a cave type access opening
- Three six-inch circular beam tubes

- One six-inch circular through tube
- One thermal column with movable door
- One 2-inch pneumatic tube with provisions for a second in the future
- Five thermal column access ports (4 horizontal & 1 tangential)

High density concrete plugs are available for all beam tubes and through tube facilities as needed.

3.2.2.3 Core and Support Structure

Fuel and reflector assemblies are positioned in a machined aluminum grid plate. The holes are bored on a square pitch in a 6 by 6 array. The grid plate and fuel are mounted on the plenum chamber, which channels the coolant flow from the fuel assemblies to the outlet pipe located near the center of the pool floor. This pipe serves as a support for the entire plenum and core structure. It also provides a guide rack for the neutron power detectors. An aluminum support structure, resting on the pool floor, supports a Thermal Column extension. Figure 3-3 shows a graphite reflector assembly.

3.2.2.4 Fuel

The reactor core is comprised of twenty-five fuel assemblies. Each assembly as shown in Figure 3-4, contains twenty-five fuel pins. Each pin consists of a zircaloy 2 tube of 0.02 inch (0.051 cm) wall thickness, filled with sintered UO_2 pellets and sealed at the top and bottom. The uranium is enriched to 4 percent by weight in the ^{235}U isotope. Each pellet measures about 0.42 inches (1.07 cm) in diameter and 0.6 inches (1.52 cm) in length. The finished fuel pin is 0.47 inches (1.20 cm) in diameter and 26 inches (66 cm) long. Approximately 20.5 grams of ^{235}U are contained in each pin.

Twenty-five pins are fastened mechanically into bundles and are placed in a zircaloy-2 box open at the top and bottom with a cross section measuring about 2.7 inches (6.86 cm) by 3.2 inches (8.13 cm). The upper end fittings and the lower end fitting (nosepiece) are attached, bringing the overall length of the assembly to about 38 inches (96.5 cm). A bail is inserted between side plates at the top of the assembly to serves as a handle for moving the assembly. There are also two alignment holes in the shoulder of the lower end fitting which mate with pins on the grid plate to prevent misalignment of the assemblies in the core. Openings are provided at the sides of each fuel assembly box to allow coolant flow if the top of the fuel assembly should become blocked by a foreign object. The fabrication of the PULSTAR fuel assemblies is in accordance with AMF Specification APR-1.

3.2.2.5 Control Rod and Pulse Rod Assemblies

Control of the reactor is attained through manipulation of three control rods and one pulse rod. Each is actuated by means of an in-line drive mechanism mounted on the bridge over the core. A control rod assembly, including the control rod drive mechanism (CRDM), is shown in Figure 3-5. The drive package consists of a motor, a gear reduction system, and an Acme screw drive. The limits of the stroke are set by adjustable, cam operated, switches

mounted on the inside of the CRDM. The standard stroke for the rod drive is twenty four inches. Control Rod Drive Mechanisms (CRDM) can be positioned in a variety of locations on the core grid plate. New positions will conform to the Technical Specifications.

Each control rod is connected to a CRDM by an electromagnet which can be de-energized in less than 50 milliseconds after a SCRAM demand.

(1) Safety and Regulating Rods

One of the three control rod assemblies has been selected as the regulating rod, while the remaining two serve as safety rods. All rods are suspended from the bridge directly over the core and extend downward into the core where they are fitted with flat absorber blades of silver-indium-cadmium alloy (80%-15%-5%) approximately 0.18 inches (0.46 cm) thick by 24 inches (61 cm) long. Out of water type scram magnets are provided at the upper end of each rod extension. A drawing of the control rod drive package is shown in Figure 3-6. A hollow shroud is restrained vertically by a tube which is concentric with the extension shaft, and which is positively attached to the CRDM mounted on the bridge. The shock absorber assembly is mounted in the CRDM as shown in Figure 3-5.

The three control rods are manually actuated from the control console either individually, in groups of two, or as a gang of three. The regulating rod may be positioned automatically in response to a power demand setting of the Automatic Control, provided the rod is beyond a specified height.

Lights mounted on the control console illuminate as each drive is coupled to the gang switch and rod contact indication is provided by a switch in the magnet, which is also displayed on the console. Position indication is generated by a transmitter and receiver synchro system and is displayed on the console by individual dials.

The average reactivity rate of insertion in the core, which can be accomplished by the gang withdrawal of the three control rods is approximately 0.046%Δk/k/sec (46 pcm/sec) for the existing 5 by 5 Graphite Reflected Core No. 3. The maximum reactivity insertion rate for the control rods is limited to 0.10%Δk/k/sec (100 pcm/sec), and a start-up accident resulting from a continuous gang rod withdrawal is analyzed under Section 13.2.2.2. Design speed is 7.5 inches/minute (0.3 $cm \cdot s^{-1}$) for the control rods.

(2) Pulse Rod

A pulse rod is provided in addition to the three control rods. It is also driven by a CRDM, shown in Figure 3-7 similar to that used for the other control rods, however, it is not magnetically coupled to the drive. Its original function was to initiate a pulse, and consequently, the portion of the shaft extension above water is fitted with a piston which is moved by a pneumatic cylinder. Air pressure originally ejected the rod at high speed to the full-out position where three shock absorbers stopped it at the top

of the stroke. The pulse rod was returned to the down position by gravity when the exhaust valve opened. An additional downward force was exerted by the shock absorbers. An attachment on the CRDM supports the pulse rod shaft extension by engagement with a collar attached to the shaft.

Since pulsing is no longer allowed, the air lines to the pulse rod from the solenoid valves are disconnected and permanently capped. In addition, the pulse rod is generally withdrawn from the core and is only inserted in the core with Senior Reactor Operator approval for infrequent operations as control rod calibrations. The potential for an accidental pulse rod ejection is nonexistent with these definitive hardware modifications.

3.2.2.6 Experimental Utilization

One or more irradiation spaces are available for insertion of samples directly through the pool water and into the core region. The reactivity worth of experiments or the rate of reactivity change shall not exceed the values indicated in the following table:

Table 3-3
Steady State Mode Experiment Reactivity Limits

Experiment Type	Reactivity Limit
Movable	0.30%Δk/k or ≤ 0.10%Δk/k/sec (300 pcm or 100 pcm/sec)
Non-secured	1.0%Δk/k (1000 pcm)
Secured	1.7%Δk/k (1700 pcm)

The reactivity worth of experiments in the pneumatic rabbit shall not exceed 0.30%Δk/k (300 pcm). Analysis of the reactor kinetics, coupled with the reactivity control system design, confirms that this amount of reactivity can be controlled by the rods and the Doppler effect. Further, the start-up rate generated by this limited value of reactivity, the time available for reactor operator response, and the effect of the control rods actions ensure that the reactor Limiting Safety System Settings (LSSS) are not exceeded. In recognition of the slight prompt jump in power due to the initial transient when the above limiting reactivity change is approached, the reactor power level will momentarily be reduced prior to rabbit insertions such that the nominal operating, full power level is not exceeded.

The total reactivity worth of all experiments is no greater than 3.0%Δk/k (3000 pcm) (absolute value). Irradiation baskets may be either open to pool water or voided. Thermal neutron fluence rates of up to 1.0 (10^{13}) n/cm^2·sec are available in core locations during steady-state operation at 1 MW for the 5 by 5 Graphite Reflected Core No. 3.

The neutrons provided by the beam tubes can be used for neutron diffraction studies, neutron spectroscopy, and time of flight measurements. The beam tubes can also be used as dry irradiation chambers for small samples in radiation effects studies. Samples can be placed at the face of the core and can be monitored easily without the water-proofing precautions required if the samples were placed in the pool. Neutron fluence rates of up to approximately 1 (10^{12}) n/cm^2·sec are available at the core ends of these beam tubes for the 5 by 5 Graphite Reflected Core No. 3.

A twelve-inch square chamber is provided for the dry irradiation of large samples. Again, a flux of approximately 1 (10^{12}) n/cm^2·sec is available at the core end of this beam tube for the 5 by 5 Graphite Reflected Core No. 3.

The graphite thermal column provides a relatively pure source of thermal neutrons for irradiation of large samples. In addition to the bulk irradiation space provided between the graphite and the door, graphite stringers are available with pockets for receiving samples for irradiation. These stringers can be inserted and removed through the Thermal Column door penetrations. Additional Thermal Column sample insertion is provided via a tangential access to the Thermal Column. A vertical access to the void space of the Thermal Column is dedicated as a cable conduit.

3.2.2.7 Stress Analysis

As stated earlier, pulsing is no longer permitted as of license amendment 9. However, the analysis demonstrates the ability of the fuel to withstand a pulse remains in Section 3 in support of an accidental transient as postulated in Section 13.

(1) Thermal Stress in Cladding During a Pulse

Stress analysis calculations [3-3] performed for the Buffalo PULSTAR establish the ability of the cladding to withstand thermal stresses expected at the design pulse. For the Buffalo core involving a 40 MW·sec pulse with forced convection cooling, the maximum stress (compression) was calculated to be 12,000 psi (83 MPa) and the temperature difference across the cladding was found to be 300°F (149°C). For the same pulse with natural convection cooling, the stress was calculated to be 25,000 psi (172 MPa) and the temperature difference about 700°F (371°C). The yield strength for hot rolled zircaloy-2 tubing for the latter case is about 37,000 psi (255 MPa) and for the former case, it is above 40,000 PSIA (276 MPa). The 40 MW·sec pulse at Buffalo represents a maximum specific energy release of 490 watt·sec/gram. Since this is 58 percent higher than the original design maximum specific energy release of 310 watt·sec/gram for the NCSU PULSTAR, and since the actual test program has verified the ability of the cladding to operate with no distortion through a 38 MW·sec pulse with forced convection cooling, it is concluded that the NCSU PULSTAR will also show no distortion if pulsed at the 38 MW·sec design level. The calculated total energy for a 2200 MW pulse of 38 MW·sec, with a maximum energy density of 310 watt·sec/gram, is based on a power distribution factor of 2.92 and a fuel loading of 359 kg of UO_2.

(2) Pressure in Annuli During a Pulse

During welding of the fuel pins and caps, the atmosphere surrounding the pellets is replaced by pure helium at slightly greater than atmospheric pressures but not in excess of 20 PSIA (0.14 MPa). During operation of the core, gaseous fission products will diffuse from the fuel into the annulus. The total pressure in the annulus is taken as the sum of the contribution of the helium and the contribution of the fission product gases.

The initial volume free space in a single fuel pin is 3.62 ml. The additional pressure caused by the gaseous fission products leaking into the annulus is negligible when compared to the assumed pressure of 20 PSIA (0.14 MPa).

At a temperature of 100°F (37.8°C) the length of the annulus is 24.5 inches (62.2 cm) and the diametral gap averages 0.0085 inches (0.022cm). In order to quantify the pressure rise in the annuli during a pulse, a 58 Mw·second energy release is considered. At this energy release, using a total peaking factor 2.92 at the hot spot location, the adiabatic fuel temperature rise calculates to be 2707°F (1486°C). To allow for the bulk coolant temperature, a conservative value of 3000°F (1649°C) for the average fuel temperature is taken. The average gas temperature in the annuli is calculated as the mathematical average between the average fuel temperature and the bulk coolant temperature, yielding a conservative value of 1600°F (1144 K) for further analysis.

A linear thermal expansion coefficient for UO_2 of $2.54(10^{-5})$ cm/°C is assumed[3-18]. The UO_2 pellet diameter reduces the diametral gap to 0.0015 inches (0.004 cm). The total length of the column of fuel pellets becomes 24.4 inches (62.0 cm), and the clearance above the pellets is reduced to 0.1 inches (0.25 cm). The expansion of zircaloy-2 cladding is neglected. Final gap volume is then calculated to be 0.716 cm^3.

Using the analysis above, the average gas temperature is calculated to rise to approximately 1600°F (1144 K). Thus, the volume of the annulus is reduced from 3.62 cm^3 to 0.716 cm^3 and the temperature rises from 100°F (310.8 K) to 1600°F (1144 K). The maximum pressure in the annulus during the pulse will then be 372 PSIA (2.66 MPa). The burst pressure of the zircaloy-2 cladding was found to be 4500 PSIA (30.92 MPa) in a hydrostatic test at room temperature for a cladding thickness of 15 mils (0.038 cm) and reduces to 1080 PSIA (7.42 MPa) at 1600°F (1144 K). Therefore, it can be concluded that pressure increases during pulsing will not cause clad failure.

This analysis involves considerable conservatism, including the scenario that fuel temperature rises adiabatically during the pulse.

3.2.3 Nuclear Design and Evaluation

3.2.3.1 Introduction

The North Carolina State University PULSTAR Reactor has been designed to operate for extended periods of time at a 1 MW power level, in a manner similar to the pool type

reactors fueled with plate type elements. The core is fueled by what is basically a power reactor fuel, i.e., 4 percent enriched UO_2 pellets. The inherent shutdown mechanism, which permits the core to be pulsed to high power, is the increased capture due to Doppler Broadening in the uranium-238 as the temperature of the fuel rises. The feasibility of using such a shutdown mechanism for pulsing operation has been proven in a series of experiments run at the SPERT Facility and at the PULSTAR Facility designed by AMF for the Buffalo Materials Research Center (BMRC), and which now operates on the campus of the State University New York at Buffalo.

3.2.3.2 Comparison with Buffalo PULSTAR

The Buffalo PULSTAR has a 20 fuel assembly core of 6 percent enrichment, and was licensed to operate at a steady-state power level of 2 MW and to pulse routinely with a total energy release of 40 MW·sec, which is equivalent to a maximum specific energy release of 490 watt·sec/gram. During the Buffalo PULSTAR license renewal in 1983, the facility elected to remove pulsing from their license. Hence the use of Buffalo's experience in pulsing is based on their pulsing prior to this licensing change. The North Carolina State University PULSTAR, a modified version of the Buffalo PULSTAR, is designed to operate with a twenty five assembly core of 4 percent enrichment, at a steady-state power level of 1 MW, and to a pulse with an originally estimated total energy release of 38 MW·sec, which, because of the larger core, is equivalent to specific energy release of 310 watt·sec/gram, based on a hot spot factor of 2.92 and a 25 fuel assembly core.

A comparison of the significant operating parameters for the Buffalo and NCSU design is as follows:

Table 3-4
Comparison of Operating Parameters for the Buffalo and NCSU PULSTAR Reactors

	Buffalo	NCSU
Design Steady State Power (MW)	2	1
Mass UO_2 per core (kg)	287	359
Uranium Enrichment	6	4
Maximum Specific Energy Release	490	310
Original Design Pulse Peak Power (MW)	2000	2200
Original Design Pulse Energy Release (MW sec)	40	38

As with the Buffalo PULSTAR, the NCSU PULSTAR Facility has elected to discontinue pulsing as of Amendment 9.

As part of the Buffalo test program, a test pin was located at a point where the flux was a factor of 1.8 higher than the peak flux in the fuel region. The test pin thereby served to

lead the core in peak power and to provide a warning of any abnormality as the test program pulses were progressively increased to the design level. The core physics design for the NCSU PULSTAR is analyzed from the viewpoint that the actual test results obtained from the very similar Buffalo PULSTAR core served as the most accurate basis for predicting the NCSU PULSTAR core performance. The applicable results of the Buffalo test program are therefore referenced frequently in the following sections and serve as a point of departure in providing design modifications for the NCSU requirements.

3.2.3.3 Core Configuration

The NCSU PULSTAR Reactor has a flexible core in which the geometric arrangement of fuel assemblies can be modified to satisfy various experiment requirements. The existing core loading is depicted in Figure 3-8C, while earlier core arrangements are depicted in Figure 3-8A for the original 5 by 5 Standard Core, and Figure 3-8B for the 5 by 5 Graphite Reflected Core No. 1. In each of these figures, the core, control rods, thermal column nosepiece, and beam tubes are detailed.

3.2.3.4 Nuclear Parameters

Table 3-12 summarizes the nuclear parameters for the existing PULSTAR 5 by 5 Graphite Reflected Core No. 3.

3.2.3.5 Calculation Methods

3.2.3.5.1 Reactivity and Power Distribution

For the original license of the NCSU PULSTAR, the method used to calculate the reactivity and power distribution for the 5 by 5 Standard Core followed the flow diagram of Figure 3-9. This section of the SAR shall continue to summarize the methodology used in predicting the nuclear characteristics of the original 5 by 5 Standard Core (for historical purposes) and in addition, provide measured nuclear data for this core, the 5 by 5 Graphite Reflected Core No. 1 and the existing 5 by 5 Graphite Reflected Core No. 3.

The original methodology used the SPAR[3-4] code for the P_3 approximation to the Boltzman Equation for cell calculations, and FORM[3-5] and TEMPEST[3-6] for the few group diffusion parameters. DMM[3-7] was used for the original diffusion calculation. For each computation a 6 percent enriched core and a 4 percent enriched core were compared so that the deviations from the BMRC core could be determined and a comparison of experiment with the predictions for the NCSU PULSTAR made.

The cylindrical fuel pins in a rectangular array were approximated by an equivalent cylindrical cell model which was required by the SPAR code. This was done on the basis of equivalent areas. The model is depicted in Figure 3-10.

The code assumed symmetry at the pin center and outer boundary and was given as an input, a slowing down source proportional to the slowing down density of the various

regions. The code required that the flux be unchanged across the gap. This is a valid assumption for the small gap size. The results of the P_3 computation are shown in Figure 3-11 for the original 5 by 5 Standard Core.

3.2.3.5.1.1 Generation of Nuclear Parameters

All neutron diffusion calculations for the original 5 by 5 Standard Core were based on the use of four lethargy groups. The range of these groups, termed fast, intermediate, epithermal and thermal, are given below:

Table 3-5
Four Group Lethargy Groups for Diffusion Calculations

Group	Lethargy Range	Energy Range
1 Fast	0 to 2.5	10 MeV to 821 keV
2 Intermediate	2.5 to 7.5	821 keV to 5.53 keV
3 Epithermal	7.5 to 16.588	5.53 keV to 0.625 eV
4 Thermal	na	< 0.625 eV

Fast cross sections were generated for the 5 by 5 Standard Core with the aid of the FORM code (a variation of MUFT). The code solves the one-dimensional Boltzman equation, after removing the spatial dependence by taking the Fourier transform. The multi-group (54 group) cross sections used were those proposed by Henry [3-8], and the lethargy widths of the four groups are those proposed by Ombrellaro [3-9]. The most critical portion of the fast group analysis is that connected with the resonance self-shielding due to lumping of ^{235}U. Since the FORM code assumed a homogeneous mixture, the effect of self-shielding was accounted for by the selection of a parameter "L". In the derivation of the "L" factor, the suggestions of Strawbridge & Barry [3-10] were followed which found application in the design of Indian Point 2.

Thermal Diffusion parameters were determined for the original 5 by 5 Standard Core with the aid of the SPAR code for flux weighing of number densities, and the TEMPEST II code for the generation of the actual parameters. The TEMPEST II code can compute the Wigner-Wilkins Spectrum, the Wilkins Spectrum, or the Maxwellian Spectrum, O/(E); the thermal constants were averaged using the Wilkins Spectrum.

3.2.3.5.1.2 Diffusion Calculations

The DMM geometrical model used in these calculations for the original 5 by 5 Standard Core were based on a slab geometry as shown in Figures 3-12 and 3-13, in order to utilize the same model for BMRC and NCSU, the fifth row of fuel assemblies in the 5 by 5 core was represented by two half assemblies symmetrically located.

The result of the 4 percent computation is shown in Figures 3-14. The point-to-average ratio was also computed and is shown in Figure 3-15. This figure demonstrates that the peak power in the 4 percent core was predicted to be slightly lower than the peak power observed in the 6 percent core. In Figures 3-14 and 3-15, the control rod water gaps are between 7.048 and 9.122 cm from the centerline and the water reflector starts at 19.888 cm.

The DMM k_{eff} predicted value for the 4 percent 25 fuel assembly core was 1.019, while the value during the original start-up tests for the 5 by 5 Standard Core was 1.01786. Refer to Figure 3-25 for the excess reactivity versus fuel burn-up behavior and the gain in reactivity associated with past and planned core changes at the PULSTAR Facility during the first license interval. In addition to the DMM calculations, a series of PDQ-2 [3-11] two dimensional diffusion calculations were run to determine core hot spot factors, rod worth, fuel assembly worth, and fluxes. The PDQ-2 calculation agreed well with the one-dimensional DMM calculations. Figure 3-16 shows the core layout utilized in the PDQ-2 calculations for the 5 by 5 Standard Core.

3.2.3.5.1.3 Predicted Neutron Fluxes and Power Distribution

Figure 3-17 shows the fission source distribution calculated for the 5 by 5 Standard Core array. The PDQ-2 calculation indicated that 76 percent of the power generated in the core resulted from thermal fissions. The numbers in the corners of Figure 3-17 are the relative source at those points in the PDQ calculation and include epithermal and thermal fissions. The numbers with bars represent the average source in the fuel pin.

The data given in Figure 3-17 were utilized to determine the hottest fuel pin in the core and the hottest point in this fuel pin. The axial hot spot factor for the core was 1.51. Using the hot spot factor from Figure 3-17 of 1.94 and the axial hot spot factor of 1.51, yields an overall hot spot factor of 2.92. This hot spot factor is used to determine the location of the maximum energy density. The hottest fuel pin is located adjacent to the control rod channel as indicated in Figure 3-17 for the 5 by 5 Standard Core.

The value of 2.92 as an overall hot spot factor defines a bounding value for analysis with regard to power densities. The measured performance of the 5 by 5 Standard Core, 5 by 5 Graphite Reflected Core No. 1 and 5 by 5 Graphite Reflected Core No. 3 all indicated an actual value of less than 2.92. Specifically, comprehensive flux mapping was performed on the 5 by 5 Standard Core, 5 by 5 Graphite Reflected Core No. 1 and the 5 by 5 Graphite Reflected Core No. 3 during the respective start-up test with the actual hot spot factor measured to be 2.80 for the 5 by 5 Standard Core [3-24], 2.53 for the 5 by 5 Graphite Reflected Core No. 1 [3-23] and 2.12 for the 5 by 5 Graphite Reflected Core No. 3 [3-21]. Figure 3-31 summarizes the results of the flux mapping for the existing 5 by 5 Graphite Reflected Core No. 3 as measured during the start-up test.

3.2.3.5.2 Control Rods

Control of the reactor is accomplished by using three Ag-In-Cd control rods. The worth of the three control rods was originally calculated using the PDQ-2 nuclear code for the

Standard Core array. For this core, the worth of the three control rods was calculated to be 7.4%Δk/k (7400 pcm) or an average rod worth of 2.4%Δk/k (2400 pcm). The actual measured total rod worth (gang) for the three cores utilized at the PULSTAR are 8.50, 9.45, and 9.105%Δk/k (8500, 9450, and 9105 pcm), for the 5 by 5 Standard Core, 5 by 5 Graphite Reflected Core No. 1, and 5 by 5 Graphite Reflected Core No. 3, respectively.

The control rods have a withdrawal rate of 7.5 inches per minute (0.3 cm s^{-1}). Each rod can be scrammed into the core in approximately 500 milliseconds. A single rod can be selected to be used as the regulating rod. This regulating rod is used to maintain the core at criticality at a selected power using the automatic controller. The location of the three control rods is variable. A safety feature prevents the repositioning of the rods unless the four adjacent fuel assemblies are unloaded from the core.

The maximum rate of reactivity insertion by the ganged control rods is limited to 0.1%Δk/k/sec (100 pcm/sec). The measured average insertion rates for the three cores are 0.047, 0.052, and 0.048%Δk/k/sec (47, 52, and 48 pcm/sec) for the 5 by 5 Standard Core, 5 by 5 Graphite Reflected Core No.1, and 5 by 5 Graphite Reflected Core No.3, respectively. Each of these core arrangements has a maximum rate of reactivity insertion (using gang rod withdrawal) of less than 0.10%Δk/k/sec (100 pcm/sec).

During the initial checkouts of the reactor, calibration curves were prepared for the three control rods, in gang and as individual rods. The rod calibration curves generated for the existing 5 by 5 Graphite Reflected Core No. 3 during its start-up test are included as Figures 3-26 through 3-30.

The calibration curves are made for each new core loading arrangement (or new fuel replacement in a fixed geometry) to ensure that the design criteria for the core shutdown margin is met.

The design criteria for shutdown margin is that the core excess reactivity shall never exceed that excess reactivity which would permit the core to be made critical by withdrawal of the control rod with the maximum worth to the fully withdrawn position with the other two rods in the full down position. The k_{eff} with the maximum worth rod totally withdrawn is no greater than 0.996.

3.2.3.5.2.1 Pulse Rod (or Shim Rod)

The pulse rod uses a Ag-In-Cd blade which was designed to be ejected from the core in approximately 200 milliseconds. The ejection of the pulse rod was accomplished by using a pneumatic cylinder connected to the absorber section by a rod extension. The positioning of the pulse rod is accomplished with an in-line drive package. The rod speed for positioning the pulse rod is 7.5 inches per minute (0.3 cm·s^{-1}). The air supply lines to the pulse rod are disconnected and permanently capped to prevent pulse rod ejection. The pulse rod is also referred to as the "shim rod", since it no longer has a pulsing capability and is used primarily for calibration of the remaining control rods.

3.2.3.5.3 Reactivity Effects

This section discusses the reactivity requirements for 1 MW steady-state operation. The fuel assembly design is the same as in the Buffalo PULSTAR except for the ^{235}U enrichment. Measured data for the three cores employed are provided below along with the original predictions for the 5 by 5 Standard Core.

3.2.3.5.3.1 Temperature Effect

The reactivity loss due to increased temperature of the fuel and moderator when operating the PULSTAR core at 1 MW results primarily from two effects. The first is the temperature rise in the water moderator and reflector going from the cold to hot condition. The second is the rise in temperature in the UO_2 while going from the cold to hot condition.

The moderator/reflector temperature coefficient in the BMRC core for the temperature range of 65°F (18.3°C) to 130°F (54.4°C) was measured to be -1.44(10^{-2})% Δk/k/°C (-8 pcm/°F). The actual measured values for the NCSU PULSTAR are:

-5.76(10^{-3})%Δk/k/°C (-3.2 pcm/°F) for the 5 by 5 Standard Core [3-24]
-6.12(10^{-3})%Δk/k/°C (-3.4 pcm/°F) for the 5 by 5 Graphite Reflected Core No. 1 [3-20]
-7.02(10^{-3})%Δk/k/°C (-3.9 pcm/°F) for the 5 by 5 Graphite Reflected Core No. 3 [3-21]

The reactivity loss due to the temperature rise in the UO_2 was calculated by using the Doppler Coefficient and the average UO_2 temperature for 1 MW steady-state operation. The average Doppler Coefficient was calculated to be -5.13(10^{-3})%Δk/k/°C (-2.85 pcm/°F) for the 5 by 5 Standard Core based upon a pool temperature of 100°F (37.8°C) and the average UO_2 rise at 1 MW operation of 193°F (89.4°C). This yielded a predicted reactivity loss due to the Doppler Broadening of 0.55%Δk/k (550 pcm) at 1 MW. The measured values of Doppler Coefficient for the NCSU PULSTAR are:

-2.52(10^{-3})%Δk/k/°C (-1.4 pcm/°F) for 5 by 5 Standard Core[3-24]
-3.24(10^{-3})%Δk/k/°C (-1.8 pcm/°F) for the 5 by 5 Graphite Reflected Core No. 1[3-20]
-2.88(10^{-3})%Δk/k/°C (-1.6 pcm/°F) for the 5 by 5 Graphite Reflected Core No. 3[3-21]

The total predicted reactivity loss for steady-state 1-MW operation of the original 5 by 5 Standard Core was -0.60%Δk/k (-600 pcm) while the measured was 0.29%Δk/k (-290 pcm). The corresponding reactivity loss for the 5 by 5 Graphite Reflected Core No. 1 and the 5 by 5 Graphite Reflected Core No. 3 are -0.37%Δk/k (-370 pcm) and -0.33%Δk/k (-330 pcm), respectively.

3.2.3.5.3.2 Void Effect

The Table 3-14 lists the reactivity effects for voids in the reflector adjacent to the core, including predicted and measured values for the 5 by 5 Standard Core and measured values for the 5 by 5 Graphite Reflected Core No. 1 and the 5 by 5 Graphite Reflected Core No.

3. The predicted values for the 5 by 5 Standard Core were estimated from Buffalo measured data [3-12]. The measured void coefficient for the PULSTAR cores are:

$-1.03(10^{-3})\%\Delta k/k/cm^3$ (-1.03 pcm/cm^3) for the 5 by 5 Standard Core [3-24]
$-1.60(10^{-3})\%\Delta k/k/cm^3$ (-1.60 pcm/cm^3) for the 5 by 5 Graphite Reflected Core No. 1 [3-20]
$-1.60(10^{-3})\%\Delta k/k/cm^3$ (-1.60 pcm/cm^3) for the 5 by 5 Graphite Reflected Core No. 3 [3-21]

3.2.3.5.3.3 Xenon Effect

The reactivity loss due to buildup of xenon-135 poison for 1 MW operation was calculated to be -1.05%Δk/k (-1050 pcm) for equilibrium and -0.25%Δk/k (-250 pcm) for xenon poison after 8 hours of operation. Both these calculated values apply to the original 5 by 5 Standard Core. For this core loading, the xenon peak occurring after a shutdown from 1 MW operation was calculated to be -1.06%Δk/k (-1060 pcm), assuming xenon equilibrium prior to reducing the reactor power. Figure 3-18A shows the measured xenon buildup behavior for a 24 hour operation at 1 MW for the original 5 by 5 Standard Core. Extended reactor operation to reach equilibrium xenon was not accomplished with this core.

For the 5 by 5 Graphite Reflected Core No. 1, the measured xenon behavior for a xenon equilibrium operation is shown on Figure 3-18B. In addition, Figure 3-18C depicts the xenon behavior for a more typical operation of 4 hours at 1 MW, also for the 5 by 5 Graphite Reflected Core No. 1.

Figure 3-18D depicts the measured xenon behavior for the 5 by 5 Graphite Reflected Core No. 3 for a typical operation of 4 hours at 1 MW.

3.2.3.5.3.4 Fission Product Accumulation and ^{235}U Burn-up

The long term reactivity change for the 4 percent enriched UO_2 core was calculated to be $9.2(10^{-4})\%\Delta k/k$/MWD/tonne-U (0.92 pcm/MWD/tonne U), excluding xenon-135 fission product poison. Using this rate of reactivity loss, due to burn-up of ^{235}U, buildup of plutonium-239 and accumulation of fission products, a 1.0%Δk/k (1000 pcm) reactivity allowance was calculated to be sufficient to operate the reactor at 1 MW for 1087 MWD/tonne U. This is equivalent to approximately 344 days operation at 1 MW for 24 hours per day (based on a 25 fuel assembly loading, 359 kg of UO_2 and 316 kg U). Figure 3-25 depicts the measured reactivity loss that has occurred since the start-up of the 5 by 5 Standard Core, following through with the changes associated with the 5 by 5 Graphite Reflected Core No. 1 and finally the existing 5 by 5 Graphite Reflected Core No. 3. Measurements indicate an average reactivity loss rate to fuel burn-up of approximately $9.6(10^{-4})\%\Delta k/k$/MWD/tonne-U (0.96 pcm/MWD/tonne).

The initial conversion ratio for the core was calculated to be 0.468. Calculations indicate that throughout the core life, the buildup of plutonium-239 will not have a significant effect on the effective delayed neutron fraction for the core. The Buffalo PULSTAR has operated for more than 40,000 MW·hours at 2 MW with intermittent pulsing without experiencing any significant changes in pulsing characteristics due to buildup of plutonium-239.

3.2.3.5.3.5 Maximum Observed Fuel Assembly Worth

For the 5 by 5 Standard Core, the maximum observed fuel assembly worth was 1.5%Δk/k (1500 pcm) in grid position 3-C. For the 5 by 5 Graphite Reflected Core No. 1, the maximum observed fuel assembly worth was 1.38%Δk/k (1380 pcm), also in grid position 3-C. For the existing 5 by 5 Graphite Reflected Core No. 3, the maximum fuel assembly worth occurs in grid position 4-D and has a value of 1.13%Δk/k (1130 pcm) at a core average burn-up of 438 MWD/MTU.

3.2.3.5.4 Pulse Reactivity Requirements

The following equations can be used to calculate the characteristics of the core for transient (pulsing) operation. The equations are based on a step input of reactivity. These equations take into account the fact that the pulse shape is not symmetrical due to a slightly higher energy release in the tail of the pulse [3-16].

$$E_{total} = \frac{F_S \times \rho_o}{b}$$

where,

E_{total} = total energy released in pulse (MW·sec)
ρ_o = excess reactivity above prompt critical ($)
b = constant of proportionality
F_S = Symmetry factor

The constant, b, equals the compensating reactivity. In the PULSTAR core this compensating or shutdown mechanism is provided by the Doppler effect in the UO_2 fuel.

$$b = \frac{\alpha}{C}$$

where α is the Doppler Coefficient and C is the heat capacity of the UO_2 fuel. The expression for E_{total} can be re-written as:

$$E_{total} = \frac{F_S \times \rho_o \times C}{\alpha} \quad and \quad E_{peak} = \frac{E_{total}}{F_S}$$

From the In-hour equation,

$$\tau = \frac{\ell^*}{\rho_o}$$

where ℓ^* is the neutron generation time (sec) and τ is the stable reactor period (sec). During the initial start-up test for the 5 by 5 Standard Core, the ratio of β/ℓ was determined using neutron density fluctuations. Using a β_{eff} value of 0.0073, the value of ℓ was measured to be 26.8 microseconds.

The peak power for a pulse can be obtained from the following expressions:

$$P_{max} = P_o + \frac{\rho_o^2}{F_S \times b \times \ell^*} = P_o + \frac{E_{peak}}{F_S \times \tau}$$

$$P_{max} = P_o + \frac{E_{total}}{F_S^2 \times \tau}$$

If the pulse is initiated from a low power level, P_o is negligible compared P_{max}, and

$$P_{max} = \frac{E_{total}}{F_S^2 \times \tau}$$

The above relationships were used to calculate the pulse characteristics for the BMRC PULSTAR core. The calculated results compared well with the measured data [3-1, 3-13]. However, pulse repeatability is limited by the accuracy involved in positioning the pulse rod (ρ_o), core initial conditions, and the rod ejection rate.

The original calculated reactivity input necessary to produce the design pulse of 38 MW·sec was 1.72%Δk/k ($2.36). Actual measurements showed the same energy release could be achieved with less input reactivity. As stated earlier, the highest pulse achieved prior to this license amendment involved an energy release of 35.4 MW·sec and a peak power of 980 MW. This pulse was achieved with 1.170%Δk/k ($1.60) (Figure 3-20).

3.2.3.5.4.1 Doppler Coefficient

The Doppler Coefficient calculated for the 5 by 5 Standard Core over the fuel temperature range of 100°F (37.8°C) to 293°F (145°C) was -5.13(10^{-3})%Δk/k/°C (-2.85 pcm/°F). The calculated total change associated with escalating power to 1 MW was -0.55%Δk/k (-550 pcm). Measured Doppler Coefficients and reactivity changes associated with full power operation are listed below for various cores.

Table 3-6
Doppler Coeficients and Reactivity Defects for PULSTAR Cores

Core	Doppler Coefficient (%Δk/k/°C (pcm/°F))	Doppler Defect to 1 MW (Δk/k (pcm))
Standard	-2.52(10^{-3}) (-1.4)	0.270 (270)
G.R. Core No. 1 †	-3.24(10^{-3}) (-1.8)	0.348 (348)
G.R. Core No. 3 †	-2.88(10^{-3}) (-1.6)	0.309 (309)

† G.R. signifies Graphite Reflected Core

3.2.3.5.5 Pulsing Characteristics

Pulsing is no longer permitted at the NCSU PULSTAR Facility, and therefore, there are no reactivity requirements associated with pulsing. However, pulsing results for all NCSU PULSTAR core configurations are presented herein in support of the accident analysis in Section 13.

The design total energy for a pulse of 38 MW·sec was calculated to require a reactivity input of 1.72%Δk/k as follows:

$$\rho_o = \left[\frac{0.0172}{0.0073}\right] - 1.0 = \$1.4$$

$$C_{core} = C_{UO_2}(100^oF) \times M_{core} = 250\,\frac{watt\,sec}{kg\,^oC} \times 359\,kg = 8.98(10^4)\,\frac{watt\,sec}{^oC}$$

$$\alpha = \frac{5.13(10^{-5})}{0.00730} = \$7.03(10^{-3})^oC^{-1}$$

$$E_T = \frac{F_S \times C_{core} \times \rho_o}{\alpha} = \frac{2.1 \times 8.975(10^4) \times 1.4}{7.03(10^{-3})}$$

$$E_T = 37.6\,MW sec \cong 38\,MW sec$$

Start-up testing identified a deviation from the expected total energy release and the measured value (Figure 3-20). The deviation was primarily attributed to differences between the calculated and measured doppler coefficient:

Difference Between Calculated and Measured Doppler Coefficients

Doppler Coefficients	
Calculated	Average Measured Value †
-5.3(10^{-3})%Δk/k/°C	-2.88(10^{-3})Δk/k/°C

† Table 3-6 averages for all PULSTAR cores

The difference between the calculated and measured Doppler Coefficient would have yielded a pulse total energy of approximately 67 MW·sec for the design step reactivity input of 1.72%Δk/k.

Pulsing reactivity was limited to 1.60%Δk/k ($2.19) to prevent exceeding the 58 MW·sec safety limit:

$$\alpha = \frac{2.88(10^{-5})}{0.00730} = \$3.95(10^{-3})°C^{-1}$$

$$E_T = \frac{F_S \times C_{core} \times \rho_o}{\alpha} = \frac{2.1 \times 8.975(10^4) \times \rho_o}{3.95(10^{-3})} = 58\,MW sec$$

$$\rho_o = \$1.22 \;\; and \;\; \rho = \$2.22 \;\; (1.62\%\Delta k/k)$$

Peak pulse power measurements at NCSU showed lower than expected peak powers (Figure 3-21). This difference is attributed to a lower than design pulse rod accumulator pressure. The change was made to reduce the mechanical shock to the control rod support structure.

3.2.4 Thermal and Hydraulic Design

3.2.4.1 Steady-state Heat Transfer

3.2.4.1.1 Design Criteria

The steady-state heat transfer design of the NCSU PULSTAR reactor is based on three criteria:

(1) Under forced convection cooling with downward flow, no coolant bulk boiling is allowed in any channel

(2) The ratio of the calculated heat flux at the point of departure from nucleate boiling (DNB) to the maximum steady-state heat flux is greater than 2.0

(3) The maximum temperature of the fuel pellet is less than the melting point of UO_2

It was found in the general thermal-hydraulic analysis of the core that under certain combinations of flow rate and power level, instabilities could occur. It was assumed that these instabilities would eventually lead to DNB. The criterion for selecting a Safety Limit is taken as the fuel cladding temperature. The analysis of a loss of flow transient is presented in Appendix 3B and 3C.

3.2.4.1.2 Design Objectives

The design calculations performed for the NCSU PULSTAR Reactor have an objective which is to determine the limits of operation beyond which the design criteria are violated. In this way, reactor operation is restricted to levels established from these limits. The safety

margin created by such an action will protect the reactor from abnormal occurrences which could violate the design criteria.

The philosophy behind the NCSU PULSTAR design objectives is given more explicitly in the Commission's "Guide to the Content of Technical Specifications." It was felt that the core thermal hydraulic design could not be separated from the bases required for the Technical Specifications. Therefore these analyses are the framework for determining the core Safety Limits (SL) and Limiting Safety System Settings (LSSS) as presented in the Technical Specifications. Revised analysis shall be made for any new core configurations, along with tests to confirm adherence of reactor operation to established Safety Limits and Limiting Safety Systems Settings.

3.2.4.1.3 Design Evaluation

3.2.4.1.3.1 Hot Channel Factors and Operating Parameters

The NCSU PULSTAR can operate in two flow modes, namely, forced and natural convection. The important parameters with respect to the thermal hydraulics of the core are the power level, flow rate, coolant channel inlet temperature (i.e., pool temperature) and coolant pressure as determined by the pool depth above the core midplane. A complete thermal hydraulic analysis of the core requires that the operating parameters be examined to determine their worst or most adverse operating limit since this will bear upon the design limits of the core. This philosophy is reflected in the Commission's "Guide for the Content of Technical Specifications."

For the NCSU PULSTAR thermal hydraulic analysis, the key independent process variable was chosen to be power level. In other words, the parameters of flow, pressure, and inlet temperature were established and then the power level was varied until the design limit was reached. The most adverse values of coolant temperature and pressure were chosen for the analysis and the flow rate was varied even though for the forced convection condition, only one flow is used (31.5 $\ell \cdot s^{-1}$ (500 gpm)). The objective here was to present an analysis which was as general as possible. The following table lists the values used in the analysis.

Table 3-7
Parameters and Values Used in the Hot Channel Factor Calculation

Parameter	Nominal Value	Most Adverse Value
Coolant Flow Rate	31.5 $\ell \cdot s^{-1}$ (500 gpm)	0 to 31.5 $\ell \cdot s^{-1}$ (0 to 500 gpm)
Pool Bulk Temperature	40.6°C (105°F)	120°F (48.9°C)
Pool Depth above Mid-core	21 ft (6.4 m)	14 ft (4.3 m)

The hot channel factors, F_{bulk}, F_{film}, and F_{flux} are products of individual factors for the following effects.

Fuel Loading

The fuel specification allows a ± 2 percent variation in fuel loading between pins. This results in 2 percent higher heat generation rate, and therefore, increases the bulk temperature rise and film temperature directly.

Pin Spacing

The fuel pin tolerance is $\pm$ 0.0015 inch ($\pm$ 0.004 cm) and the fuel assembly box tolerance is $\pm$ 0.005 inch (0.013 cm). The coolant velocity is proportional to the 2/3 power of the hydraulic diameter (d). The bulk temperature rise (t) is inversely proportional to the product of the velocity and flow area (A).

$$t/t_{nom} = (A_{nom}/A)(d_{nom}/d)^{2/3} = 1.012(1.014)^{2/3} = 1.021$$

The heat transfer coefficient varies directly with the cube root of the hydraulic diameter; therefore, the film-temperature ratio is as follows:

$$\phi/\phi_{nom} = (d_{nom}/d)^{1/3} = 1.005$$

Heat Transfer Coefficient

The heat transfer coefficient data correlate to within $\pm$ 20 percent. The film temperature difference increases by a factor of 1.2 since the heat transfer coefficient and film temperature difference are directly proportional.

Instrument Error

The neutron detection instrumentation which monitors the power level is accurate to within $\pm$ 7 percent. Therefore, the bulk temperature and film temperature differences both increase by a factor of 1.07 since they are directly proportional to the power level.

Flow Reduction

The SCRAM signal which initiates reactor shutdown occurs when the flow is less than 90 percent of nominal flow (LSSS). The resultant bulk temperature rise is 1.11 times the nominal, since it is inversely proportional to flow. The film temperature difference increases by a factor of 1.088 since the heat transfer coefficient varies as the 0.8 power of velocity.

Power Increase

A rod reverse signal is initiated at a power level of 10 percent above the nominal reactor power (SSS). Therefore, the film temperature and bulk temperature differences increase by a factor of 1.10 since they are directly proportional to the power level increase.

Power Distribution

The original PDQ calculations yielded a ratio of the maximum local power to the average core power of 2.92 for the 5 by 5 Standard Core. Calculations also predicted the axial peak power generation is 1.51 times the axial average. Therefore the radial peak-to-average was predicted to be 1.94 times the radial average.The following table lists measured total peaking factors for PULSTAR cores.

Table 3-8
Measured Total Power Peaking Factors for PULSTAR Cores

Core	F^N_{total}
Standard	2.80
G.R. Core No. 1	2.50
G.R. Core No. 3	2.12

The 2.92 total value will continue to be used for the bounding value of maximum local power to average core power. Use of the larger 2.92 value in this SAR allows flexibility in changing the core configuration in the future with a possibility that the resultant total peak-to-average ratio might be slightly higher than observed with the existing 5 by 5 Graphite Reflected Core No. 3.

The overall hot channel factors are the product of individual factors. However, it should be noted that when determining the limits of operation to avoid violating the design criteria, the power increase, instrument error, and flow reduction factors are ignored. These elements enter the calculations as reduction factors to the Safety Limits to account for instrument uncertainties and desired operating margins. The overall factors are summarized below.

Table 3-9
Bulk, Film and Flux Uncerntainty Factors

Parameter	F_{bulk}	F_{film}	F_{flux}
Fuel Loading	1.02	1.02	1.02
Pin Spacing	1.021	1.005	1.017
Heat Transfer Coefficient	-	1.20	-
Flow Maldistribution	1.05	1.05	1.05
Radial Peaking	1.94	1.94	1.94
Axial Peaking	-	1.51	1.51
Total	2.12	3.77	3.18

3.2.4.1.3.2 Natural Convection Mode

Under natural convection conditions, bulk boiling can occur resulting in undesirable releases of nitrogen-16 activity following possible bubble rise in the pool water, although bubble collapse would occur at some level due to subcooling. To eliminate this possibility, a limiting criterion for operation can be established such that no bulk boiling is allowed.

Reference 3-12 describes convection power tests performed at the Buffalo Materials Research Center (BMRC), Inc. Coolant temperature rise data across a test fuel assembly in the BMRC PULSTAR core are presented in Figure 5 of Appendix 3C, which contains an analysis of related data for natural convection flow. The boundaries on the data are not linear since a greater heat input will result in a greater coolant flow. The temperature difference shown was determined to be the greatest measured with four outlet thermocouple. This difference should be fairly constant for all channels in the fuel assembly since a higher heat input causes more flow resulting in a uniform temperature difference.

From the data, one can calculate the natural convection coolant flow rate as a function of heat input. The results of this calculation are presented in Table 1 of Appendix 3C. Analysis of the data reveals that for power levels below 1.4 MW, the hot spot clad temperature for the NCSU PULSTAR core remains below 273°F (134°C) which is far below the clad melting temperature or DNB. The hot spot will experience nucleate boiling which is acceptable, and the bulk coolant temperature in the average channel will be limited to less than 220°F (104°C) at the outlet of the channel. The power-flow curve for the hot channel factor will not be much different than for the average channel since the greater heat input will draw more flow. Neglecting for conservatism the fact that more heat input draws more flow, one can calculate a hot channel delta temperature as a function of power level using the data reported by BMRC.

The data were based on a hot pin radial factor of 1.53 which was the same as used for the core. The average flux in the test assembly was 0.82 times the average flux in the core. Using these numbers, a net heat input multiplying factor of 1.87 (i.e., 1.53/0.82) for the hot channel in the test element is derived. Figure 3-22 represents the product of the WNY data curve for "no chimney" operation and the 1.87 factor.

Assuming the pool temperature to be at its maximum allowable peak of 120°F (48.9°C) and the pool level at its minimum of 14 feet (4.3 m), the minimum power level to cause bulk boiling at the hot channel outlet can be determined. This power level is given on Figure 3-22 as 680 kW.

The value of maximum power level quoted in the Technical Specifications for natural convection conditions is 250 kW (LSSS). Therefore a large margin exists for operational safety. It should also be noted that the data chosen for Figure 3-22 are based on "no chimney" which gives a lower average flow for any given input.

3.2.4.1.3.3 Forced Convection Mode

A detailed thermal hydraulic analysis was performed to determine the operating limits under forced convection conditions. This analysis is presented in Appendix 3B. An additional analysis is presented in Appendix 3C which examines clad temperature effects in more detail than Appendix 3B. The objective of the analysis in Appendix 3B was to determine the effects of variation in core coolant flow, system pressure, and core coolant inlet temperature on the steady-state burnout level, flow stability, and coolant bulk boiling in the channel of the core having the greatest heat input.

The core was analyzed under the most limiting conditions of coolant pressure and core inlet temperature. The results of the analysis are presented in Figures 3-23 and 3-24. These Figures show that at low flows (i.e., below about 50 percent of full core flow 500 gpm (31.5 $\ell \cdot s^{-1}$)), the steady-state power is limited by flow stability considerations while for higher flows, bulk boiling at the hot channel exit is limiting.

The normal steady-state operating condition for the NCSU PULSTAR is with a single flow of 500 gpm (31.5 $\ell \cdot s^{-1}$). A flow scram prevents operation below a flow of 90% of 500 gpm (31.5 $\ell \cdot s^{-1}$) (LSSS). However, during a loss of flow transient the coolant will drop to zero, reverse direction, and then increase. It is important to examine the fuel clad temperature during such a transient since this is the one variable which will represent the condition of the clad barrier transient.

Appendix 3C reviews the data generated in the analysis of Appendix 3B which treated a flow reversal transient. Specifically, the clad temperature at the hot spot is calculated assuming nucleate boiling which will be present during the flow reversal. The initial conditions for the analysis were a steady-state power level of 1.4 MW and a flow rate of 80% of full flow (400 gpm (25.2 $\ell \cdot s^{-1}$)). The assumption on initial flow is relatively unimportant since the analysis assumed an instantaneous decrease to zero.

The results of the analysis show that following a flow transient, the maximum clad temperature will be 273°F (134°C) which corresponds to the initial power level of 1.4 MW.

3.2.4.2 Transient Heat Transfer Analysis

3.2.4.2.1 Design Criteria

As of Amendment 9 to the SAR, the NCSU PULSTAR Reactor will no longer operate in the pulse mode. This section, however, dealing with the energy release of a pulse in the fuel, shall remain in support of Section 13 accident scenarios involving an accidental pulse (such as from a fuel loading accident). The transient heat transfer design is based on the following criteria:

(1) Departure from nucleate boiling does not occur during pulsing,

(2) Neither the clad or fuel temperature exceed their melting points.

3.2.4.2.2 Operating Parameters

The operating parameters of the NCSU PULSTAR reactor core for pulsing are listed in the following table:

Table 3-10
Pulsing Parameters

Number of Fuel Assemblies	25
Mass of UO_2	359 kg
Original Design Pulse Energy	38 MW sec
Hot Spot Power Peaking Factor	2.92
Pulse Energy Safety Limit (Forced Convection)	58 MW·sec

3.2.4.2.3 Design Evaluation

Reference 3-1 presents the results of PULSTAR pulse tests conducted at the BMRC. These tests were performed with both natural convection and forced convection. The natural convection results are directly applicable to the NCSU PULSTAR since the same flow geometry exist in both reactors resulting in nearly identical natural convection flows. The pulse data for the natural convection conditions demonstrated that the limits of pulse performance for all practical purposes is dependent only upon the maximum energy density in the core. The density was calculated to be 400 watt·sec/gram. The NCSU PULSTAR core should be subject to the same threshold for DNB. Based on this assumption, DNB will not occur in the NCSU PULSTAR under natural convection conditions for pulse energies of less than:

$$E_{max} \text{ (in MW·sec)} = W(400)10^{-3}/F_p$$

where:

$$W = \text{weight of } UO_2 \text{ in core} = 359 \text{ kg}$$

then,

$$E_{max} = 359(400)10^{-3}/2.92 = 49.3 \text{ MW·sec}$$

The hot spot Factor, F_p, is determined from the core peaking factors and for the NCSU PULSTAR core is 2.92. Therefore, pulse tests reported in Reference 3-1 were actually conducted with energy densities as high as 872 watt·sec/gram with natural convection. Although examination showed that test pins had become deformed, the integrity of the cladding had been maintained. Therefore, indications are that the true limit on energy density lies significantly above the 400 watt·sec/gram value. Forced convection pulse test are described in Reference 3-1. It was the clear conclusion that forced flow increased the pulse capability. Test results were evaluated and a correlation curve of pin performance under forced convection cooling was developed. To use this curve a pin correlation factor

must be calculated. The correlation factor is discussed in Reference 3-1 and is related to changes in hydraulic diameter, perimeter ratio, etc. If the corner pin which will be the most limiting is considered, the film coefficient at burnout can be calculated from Bernath's correlation. This calculation follows, and the reader is referred to Table III in Reference 3-1:

$$h_{BO} = [10{,}890(De)/(De + Di)] + [48(V)]/[De^{0.6}]$$

where:

V = velocity (ft/sec)
De = hydraulic diameter (ft)
Di = heated diameter (ft)

For the corner pin, h_{BO} is as follows (where the nominal flow velocity of the inner channel is adjusted to 1.8 ft/sec (0.55 $m \cdot s^{-1}$)) as exists in the NCSU PULSTAR core at 500 gpm (31.5 $\ell \cdot s^{-1}$):

$$h_{BO} = [10{,}890(0.1512)/(0.1512 + 0.4725)] + [48(1.8)(0.63)]/[(0.1512/12)^{0.6}]$$
$$h_{BO} = 3390 \text{ Btu/hr}\cdot\text{ft}^2\cdot{}^\circ\text{F } (1.93 \text{ w}\cdot\text{cm}^{-2}\ {}^\circ\text{C}^{-1})$$

Therefore, since the calculation of E_{max} in Reference 3-1 is based on a flow of 5 ft/sec, h_{BO}=6782 Btu/hr·ft^2·°F (3.85 w·cm^{-2} °C^{-1}), a flow correction is derived as follows:

$$cf_{flow} = 3390/6782 = 0.502$$

Appendix 3C contains a review of pulse heat transfer data which must be accounted for when establishing a limiting value on pulse energy. It was assumed in Appendix 3C that the lower limit was 400 gpm (25.2 $\ell \cdot s^{-1}$) which includes a 50 gpm (3.2 $\ell \cdot s^{-1}$) differential for the flow scram and 50 gpm (3.2 $\ell \cdot s^{-1}$) for flow measurement uncertainty. Repeating the above calculation for 400 gpm (25.2 $\ell \cdot s^{-1}$) yields a cf_{flow} of 0.478.

The perimeter ratio for the corner pin is 0.663 giving a net correlation factor of:

$$cf = (0.478)(0.663) = 0.317$$

Using the corner pin correlation factor of 0.317 and Figure 3 (BMRC correlation curve of film boiling) in Reference 3-1, one obtains a maximum energy density of 470 watt·sec/gram of UO_2. Using this value, one obtains a maximum energy input of:

$$E = 359(470)(10^{-3})/(2.92) = 58 \text{ MW}\cdot\text{sec}$$

The above value of total energy input during a pulse is the criterion used to establish a safety limit for pulse reactivity insertion. It should be noted however that the BMRC pulse tests indicate that limited DNB can occur without significant fuel clad damage. In fact, pulse tests were conducted to much higher energy densities than that corresponding to the value of 58 MW·sec.

Table 3-11
Tabulated Reactor Data

Fuel	Pin Type, 4% enriched pellets with Zircaloy-2 cladding
Initial Core Loading	25 Pins/Assembly, 25 Assemblies per core, 2-foot active fuel length, 12.59 kg ^{235}U
Present Core Loading	Same as Initial Core Loading with the addition of 10 Graphite Reflectors
Design Steady-state Power	1 Megawatt
Moderator	Light Water
Reflector	Light Water and 10 Graphite Reflectors
Coolant	Light Water
H_2O/UO_2 ratio	1.135
Primary Coolant Flow Rate	500 gpm (31.5 $\ell \cdot s^{-1}$)
Fuel:	
Material	UO_2
Form	Sintered Pellets
Enrichment (weight % ^{235}U)	4.0%
5 by 5 core (kg UO_2)	359 kg
Density UO_2	10.4 gm/cc
Pellet Diameter	0.423 inches (1.07 cm)
Diametrical Gap (Helium filled)	0.0085 inches (0.022 cm)
Zircaloy 2 Cladding Thickness	0.0205 inches (0.052 cm)
Fuel Pin Outside Diameter	0.4725 inches (1.200 cm)
Fuel Pin Lattice	Rectangular
Center to Center Distance	0.606 inches x 0.524 inches (1.54 cm x 1.33 cm)
Length of Active Fuel	24 inches (61 cm)
Fuel Pins per Assembly	25
Fuel Pins in a 25 Assembly Core	625
Box Assembly:	
Material	Zircaloy-2
Inside Dimensions	2.62 inches x 3.03 inches (6.65 cm x 7.70 cm)
Wall Thickness	0.060 inches (0.152 cm)
Clearance Between Boxes	0.040 inches (0.102 cm)
Clearance Between Control Rod Shroud and Box	0.060 inches (0.152 cm)
Moderator:	
Material	H_2O
Temperature	
Cold	70°F (21.1°C)
Hot	nominal 105°F (40.6°C)
Control Rods	
Number	3 Safety Rods and 1 Pulse Rod
Absorber Material	Ag-In-Cd (80%-15%-5%)
Guide Shroud Material	Aluminum

Table 3-11 (Continued)
Tabulated Reactor Data

Control Rods (continued)	
Shape	Rectangular
Transverse Dimensions:	
Guide	6.34 inches x 0.44 inches (16.1 cm x 1.12 cm)
Absorber	4.85 inches x 0.18 inches (12.3 cm x 0.46 cm)
Clearance Absorber to Guide	0.0625 inches (0.16 cm)
Absorber Clad Material	Tin-Nickel
Withdrawal Speed:	
Safety	7.5 inches/minute (0.3 $cm \cdot s^{-1}$)
Withdrawal Speed:	
Pulse Rod	7.5 inches/minute (0.3 $cm \cdot s^{-1}$)
Neutron Source	5 Curies Pu-Be (1.85(10^{11} Bq))

Table 3-12
Nuclear Parameters

Core

Overall dimension — 15 in. x 15 7/8 in. x 24 in. (38.1 cm x 40.3 cm x 61 cm)

Volume Fractions

Material	Lattice	Element	Core
UO_2	0.44256	0.3963	0.3823
Gap	0.01798	0.0161	0.0155
Cladding (warts incl.)	0.09287	0.0832	0.0803
Lattice Water	0.44659	0.4000	0.3858
Boxes		0.0780	0.0753
Water between assemblies		0.0264	0.0255
Control guides			0.0128
Water inside guides			0.0226
	1.00000	1.0000	1.0000

Totals		
UO_2	0.3963	0.3823
Gap	0.0161	0.0155
Water	0.4264	0.4339
Zr-2	0.0831	0.0802
Aluminum	0.0781	0.0881
	1.0000	1.0000

Volume Ratios

H_2O/UO_2	1.041	1.135
H_2O/U	1.89	2.06

Worth of a single fuel element† — 500 pcm to 1130 pcm (0.50 to 1.13%$\Delta k/k$)

Xenon-135 Reactivity†

Equilibrium	800 pcm (0.800%$\Delta k/k$) (estimated)
8 Hr.	292 pcm (0.292%$\Delta k/k$)
Peak after shutdown	825 pcm (0.825%$\Delta k/k$) (estimated)

Reactivity for 1 MW steady-state†

Temperature + Doppler	330 pcm (0.330%$\Delta k/k$)
Xenon-135 (8 hrs)	292 pcm (0.292%$\Delta k/k$)

Table 3-12 (continued)

Reactivity limits for single experiment in core or external to core	1700 pcm (1.7%Δk/k)
Control Rod Worth†	
Avg. Rod Worth	2770 pcm (2.77%Δk/k)
Worth of 3 safety rods	8310 pcm (8.31%Δk/k)
k_{eff} - Core†	
5 by 5 core (cold clean rods out)	1.0172 (at 13,600 MWD/tonne-U)
5 by 5 core (cold clean rods in)	0.938 (at 13,600 MWD/tonne-U)
k_{eff} - Fuel Storage	
13 assemblies (no poison built in)	0.6363
Neutron Temperature	0.0509 eV
Disadvantage factors	
ϕ Mod/ ϕ Fuel	1.243
ϕ clad/ ϕ Fuel	1.125
Age (τ)	43.8 cm^2
Thermal Diffusion Area (L^2)	2.215 cm^2
Reflector Savings H_2O	7.81 cm
Bucklings (B^2)	
width	0.003420
length	0.003160
height	0.001680
Total	0.008260
Minimum Critical Mass Cold Clean	20 fresh fuel assemblies with no graphite reflectors
Doppler Coefficient†	-1.6 pcm/°F (-2.88(10^{-3})%Δk/k/°C)

Table 3-12 (continued)

Neutron Lifetime		
Slowing down time		1.32×10^{-5} sec
Diffusion time		2.28×10^{-5} sec
	Total	3.6×10^{-5} sec
β_{eff}		0.00730
Temperature Coefficient†		-3.9 pcm/°F (-7.02(10^{-3})%Δk/k/°C)
Void Coefficient†		-1.60 pcm/cc (-1.6(10^{-3})%Δk/k/cc)

Experimental Facilities Reactivity Effects†

Excess reactivity of 1.7%Δk/k (1700 pcm) is established with following conditions: Beam tubes water-filled, thermal column nosepiece in place, no experiments, clean, cold and core in position indicated in Figure 3-8C.

(going from Water filled to Air filled condition)

(2)	6" dia. Beam Tubes		- 130 pcm total for both tubes
(1)	8" dia. Beam Tube		- 30 pcm
(1)	6" dia. Through Tube		0 pcm
(1)	12" x 12" Facility		- 160 pcm
		Total	- 320 pcm (-0.320%Δk/k)

Thermal Column Nosepiece	0 pcm

† Reactivity information is for the 5 by 5 Graphite Reflected Core No. 3

Table 3-13
Unperturbed Neutron Fluxes for 1 MW Steady-State Operation
Predicted for the 5 by 5 Standard Core

Group	Energy	ϕ = nv		nvt
		Steady State	Pulse	Pulse
1	10 MeV to 821 keV	$7.04(10^{12})$	$1.55(10^{16})$	$2.68(10^{14})$
2	821 keV to 5.53 keV	$6.12(10^{12})$	$1.35(10^{16})$	$2.33(10^{14})$
3	5.53 keV to 0.625 eV	$5.45(10^{12})$	$1.20(10^{16})$	$2.08(10^{14})$
4	less than 0.625 eV	$1.69(10^{13})$	$3.72(10^{16})$	$6.42(10^{14})$

Table 3-14
Water Reflector Void Effects

Facilities		Reactivity (%Δk/k)			
No.	Description	Standard [1]	Standard [2]	G.R.C. No. 1 [3]	G.R.C. No. 3 [4]
3	6" Beam Tubes	0.20	0.17,0.25,0.16	0.21,0.30,0.25	0.25,0.07,0.06
1	8" Beam Tube	0.18	0.45	0.02	0.03
1	6" Through Tube	0.20	0.70	0.06	0
1	12" Beam Tube	1.00	0.20	0.15	0.16

(all values are the negative reactivity change in pcm going from water to air filled)

[1] - 5 by 5 Standard Core (predicted)

[2] - 5 by 5 Standard Core measured (3-19)

[3] - 5 by 5 Graphite Reflected Core No. 1 measured (3-20)

[4] - 5 by 5 Graphite Reflected Core No. 3 measured (3-21)

3.3 References

3-1 Hall, W.F., et al, PULSTAR Pulse Tests III & IV Comparison of Manual and Forced Convection Cooling, Report No. WNY-023, page 3, May 26, 1966.

3-2 Grund, J.E., Experimental Results of Potentially Destructive Reactivity Additions to an Oxide Core, IDO 17028, December 1964.

3-3 AMF Atomics, The Advanced Pulse Reactor (APR), pages 46 through 49, 1963.

3-4 White, C. SPAR A P_3 Flux Distribution Code, GEAP ED, 1958.

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3-8 Henry, A.F., 54 - Group Library for P_1 Programs, WAPD-TM-224, April, 1960.

3-9 Ombrellaro, P.A., Effective Fast Group Cross Sections in Four Group Theory, WAPD-TM-63, 1967.

3-10 Strawbridge, L.E. and Barry, R.F., Nuclear Science and Engineering, 23, 58-73, 1965.

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3-12 Hall, W.F., Private Communication with WNYNRC, 1966.

3-13 Lumb, R.F., et al, PULSTAR Pulse Tests, WNY-017, page 49, October 1964.

3-14 Mirshak, S., et al, Heat Flux at Burnout, DP-355, February 1959.

3-15 Lowdermilk, W.H., et al, Investigation of Boiling Burnout and Flow Stability for Water Flowing in Tubes, NACA-TN-4382, September 1958.

3-16 H. Soodak, Editor, Reactor Handbook, Volume III, Part A, "Physics", John Wylie and Sons, 1962.

3-17 J.B. Conway, et al, The Thermal Expansion and the Heat Capacity of UO_2 to 2200°F, American Nuclear Society Transaction Volume 6, No.1 (June 1963).

3-18 J. Belle, UO2 Properties and Nuclear Applications, USAEC, Naval reactor Division, USGPO.

3-19 PULSTAR Startup Tests, 1972.

3-20 PULSTAR 5 by 5 Graphite Reflected Core No. 1 Startup Tests, 1976.

3-21 PULSTAR 5 by 5 Graphite Reflected Core No. 3 Startup Tests, 1978.

3-22 Henry, L., Buffalo PULSTAR at the Buffalo Materials Research Center, Private Communication, July 1988.

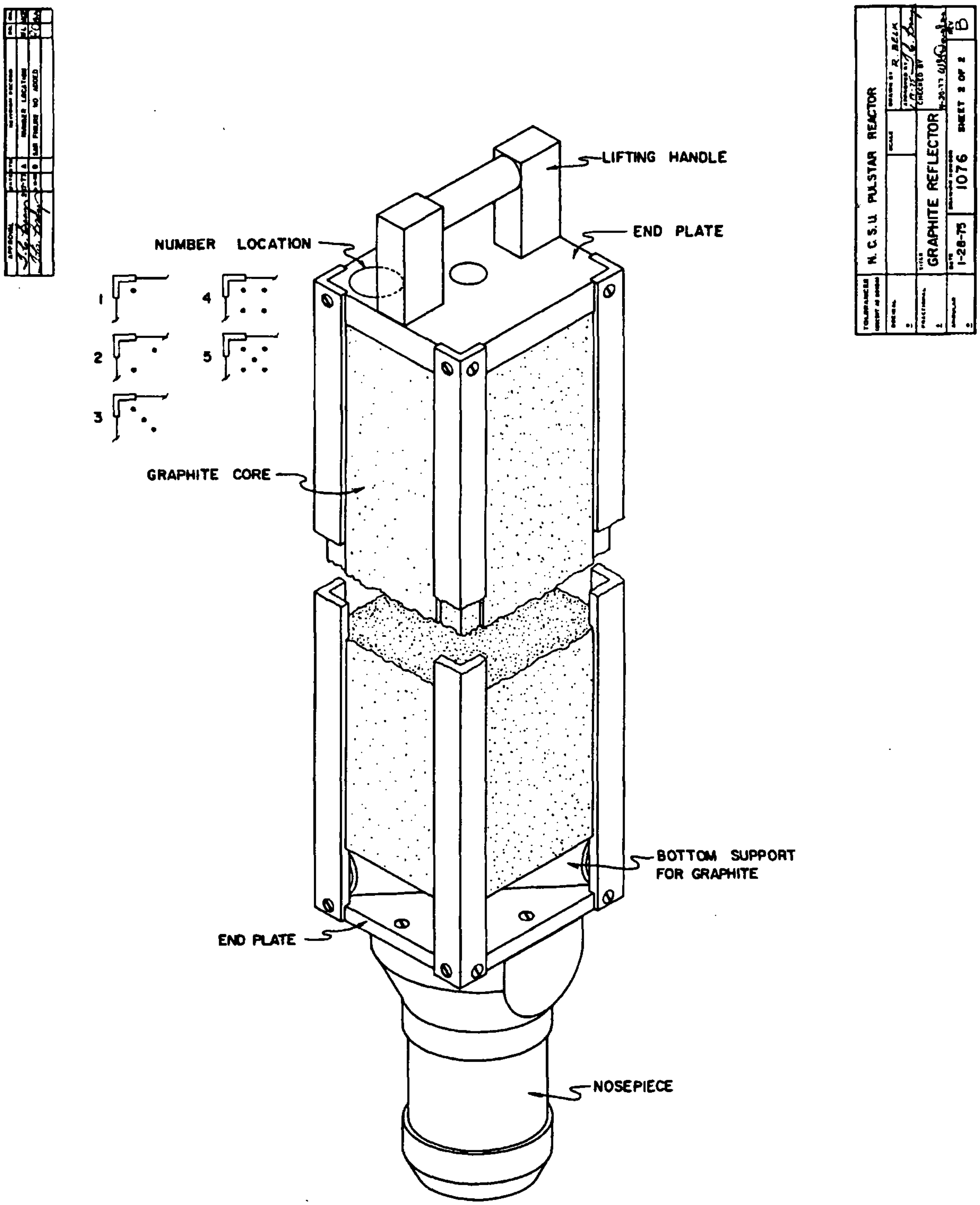
3-23 Nuclear Reactor Program, Technical Note No. 2, 26 January 1976.

3-24 Nuclear Reactor Program, Technical Note No. 1, 15 February 1974.

Figure 3-1

Figure 3-2

Figure 3-3


LIFTING HANDLE
END PLATE
NUMBER LOCATION
1
2
3
4
5
GRAPHITE CORE
BOTTOM SUPPORT FOR GRAPHITE
END PLATE
NOSEPIECE
N. C. S. U. PULSTAR REACTOR
GRAPHITE REFLECTOR
1-28-75
1076
SHEET 2 OF 2
B

9509200199-11

Figure 3-5

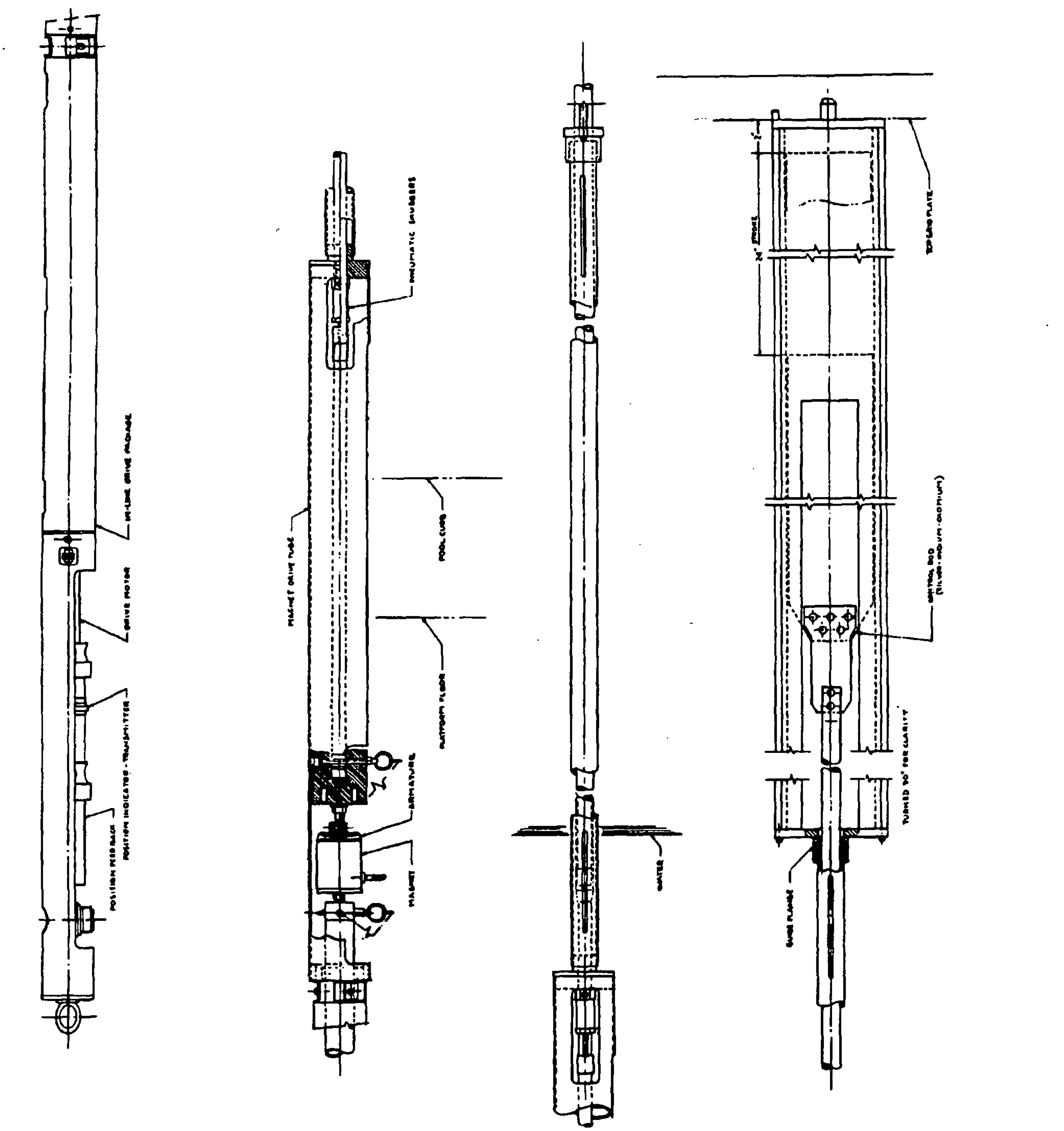

IN-LINE DRIVE PACKAGE
DRIVE MOTOR
POSITION INDICATOR - TRANSMITTER
POSITION FEEDBACK
PNEUMATIC SNUBBERS
MAGNET DRIVE TUBE
POOL CURB
PLATFORM FLOOR
ARMATURE
MAGNET
WATER
TOP GRID PLATE
24" STROKE
CONTROL ROD
TURNED 90° FOR CLARITY
GUIDE FLANGE

Figure 3-6

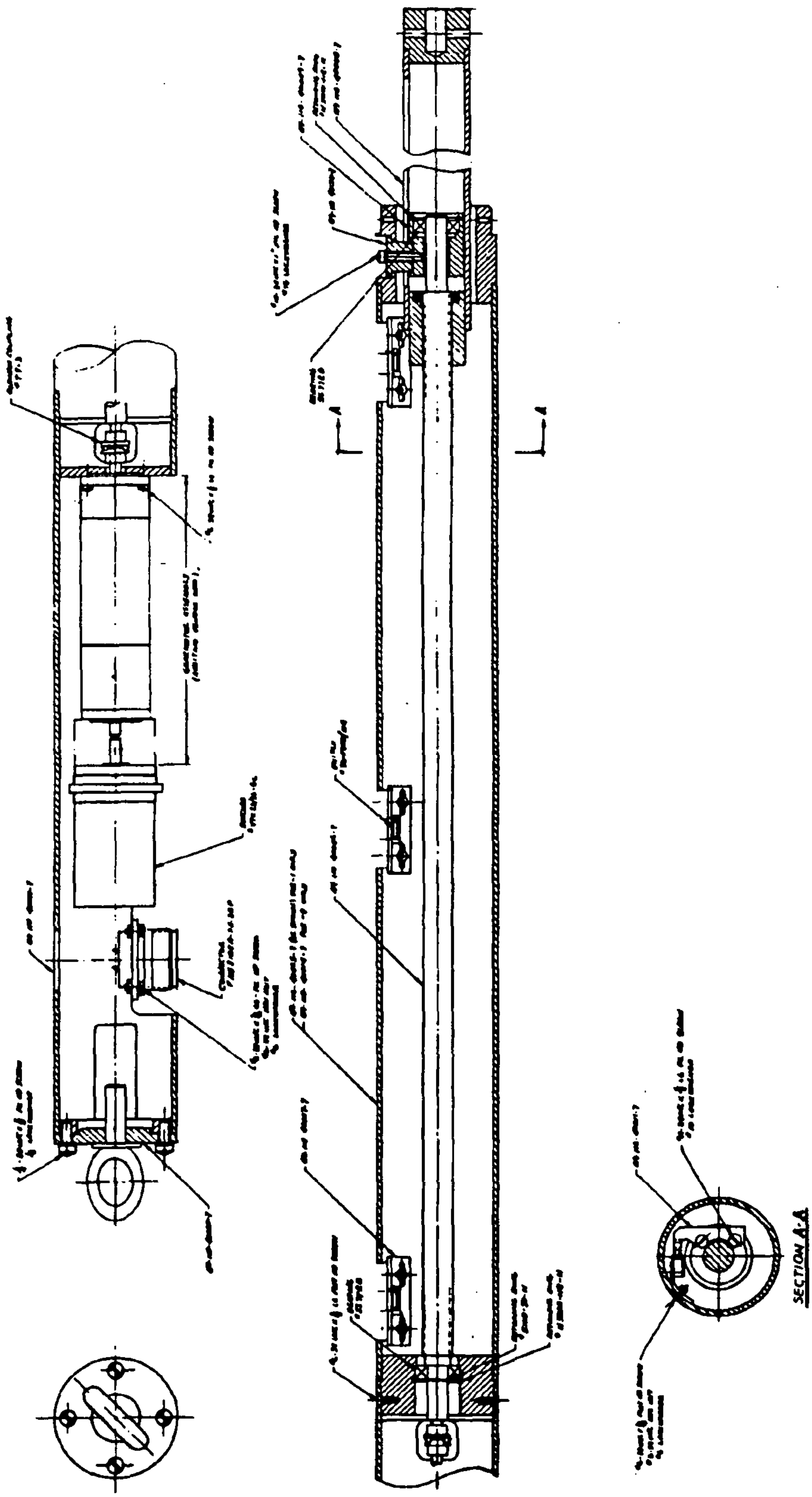
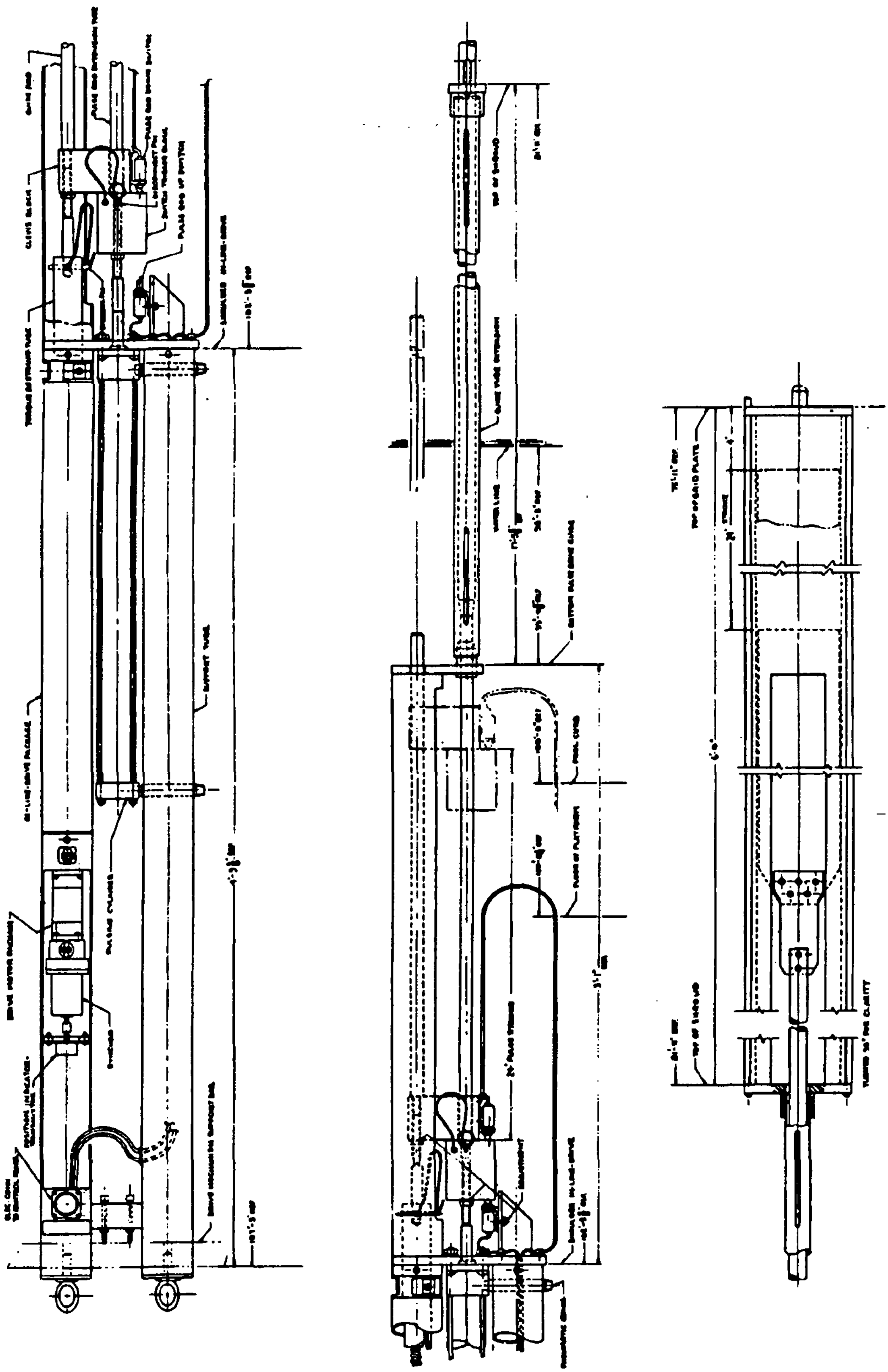
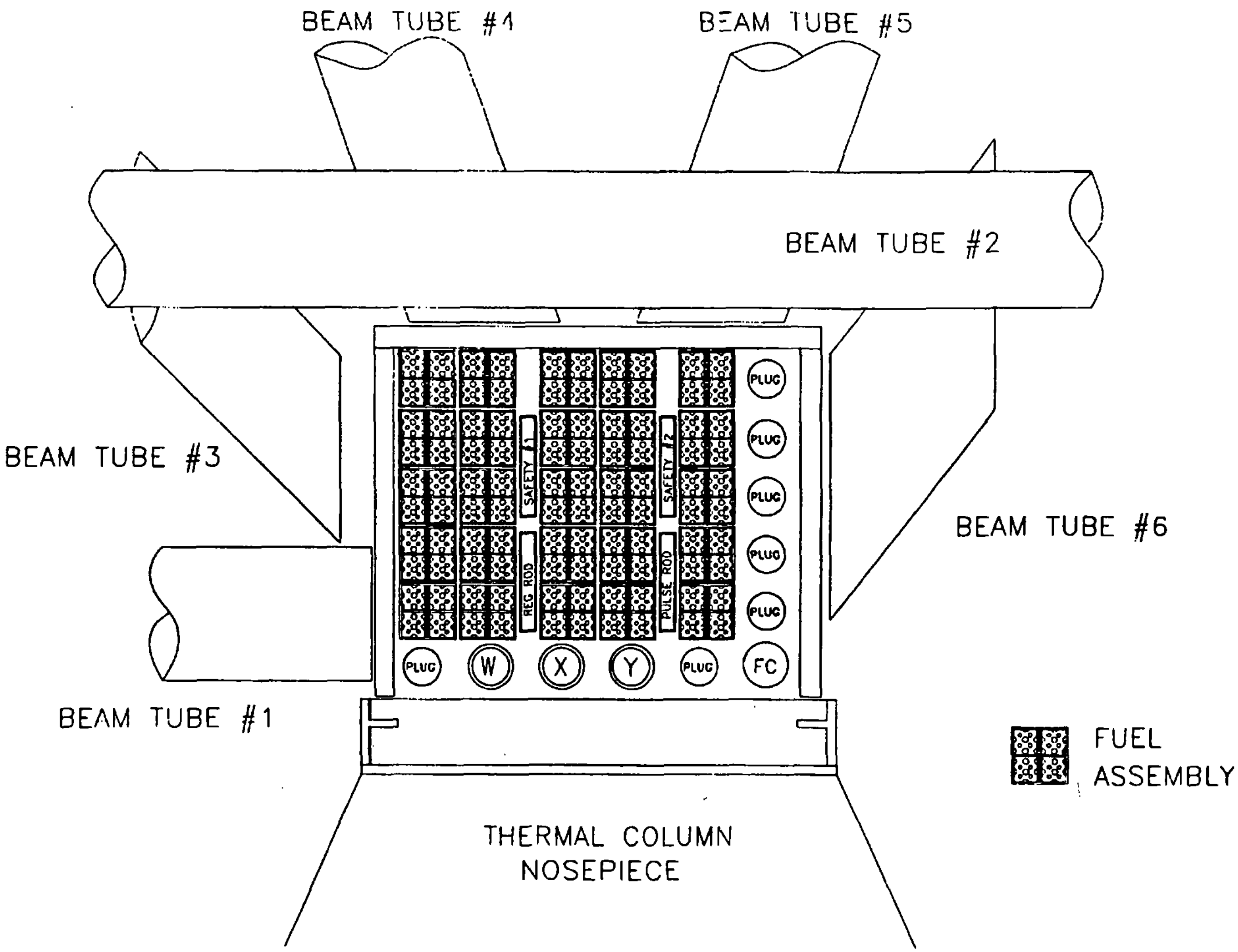


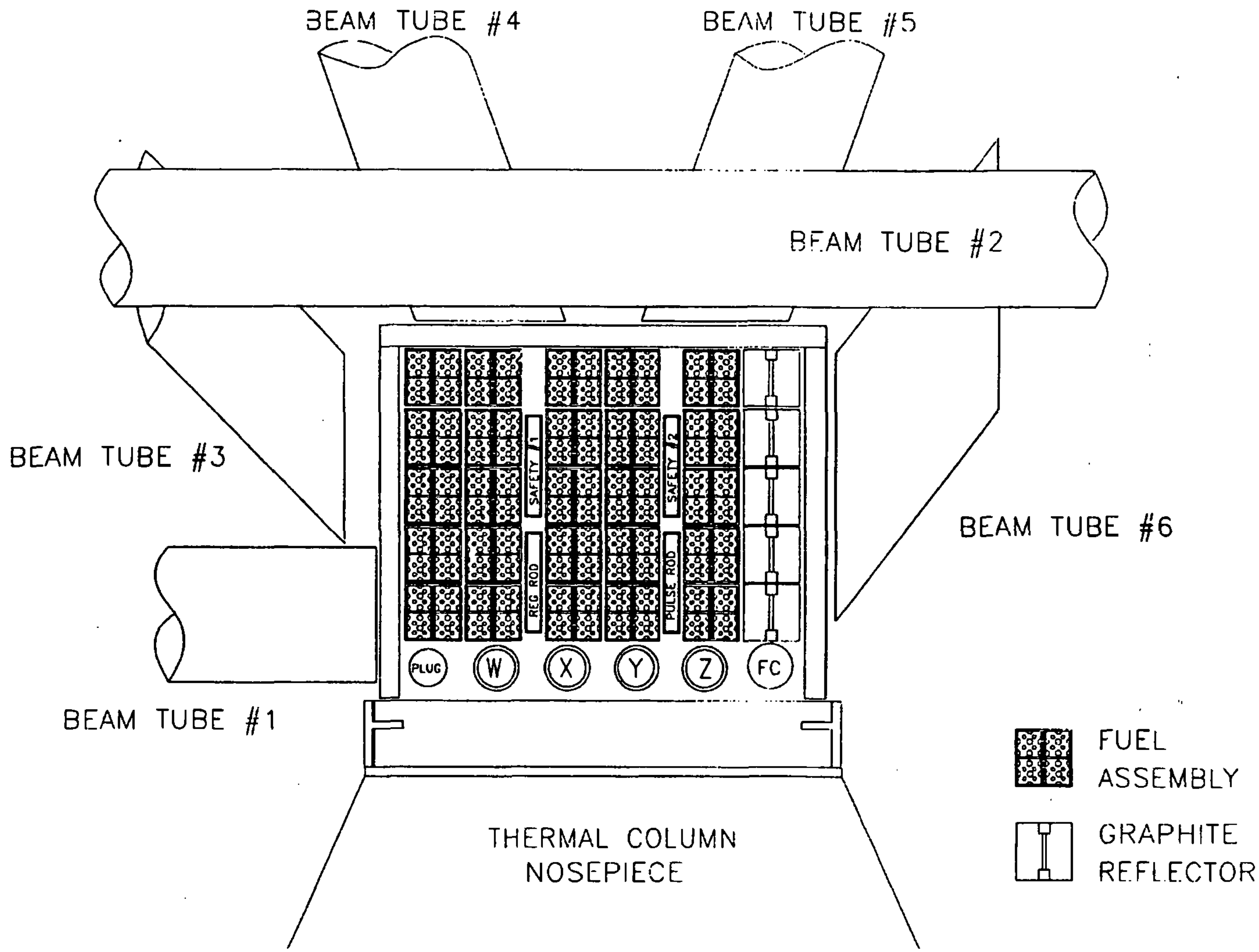
Figure 3-7





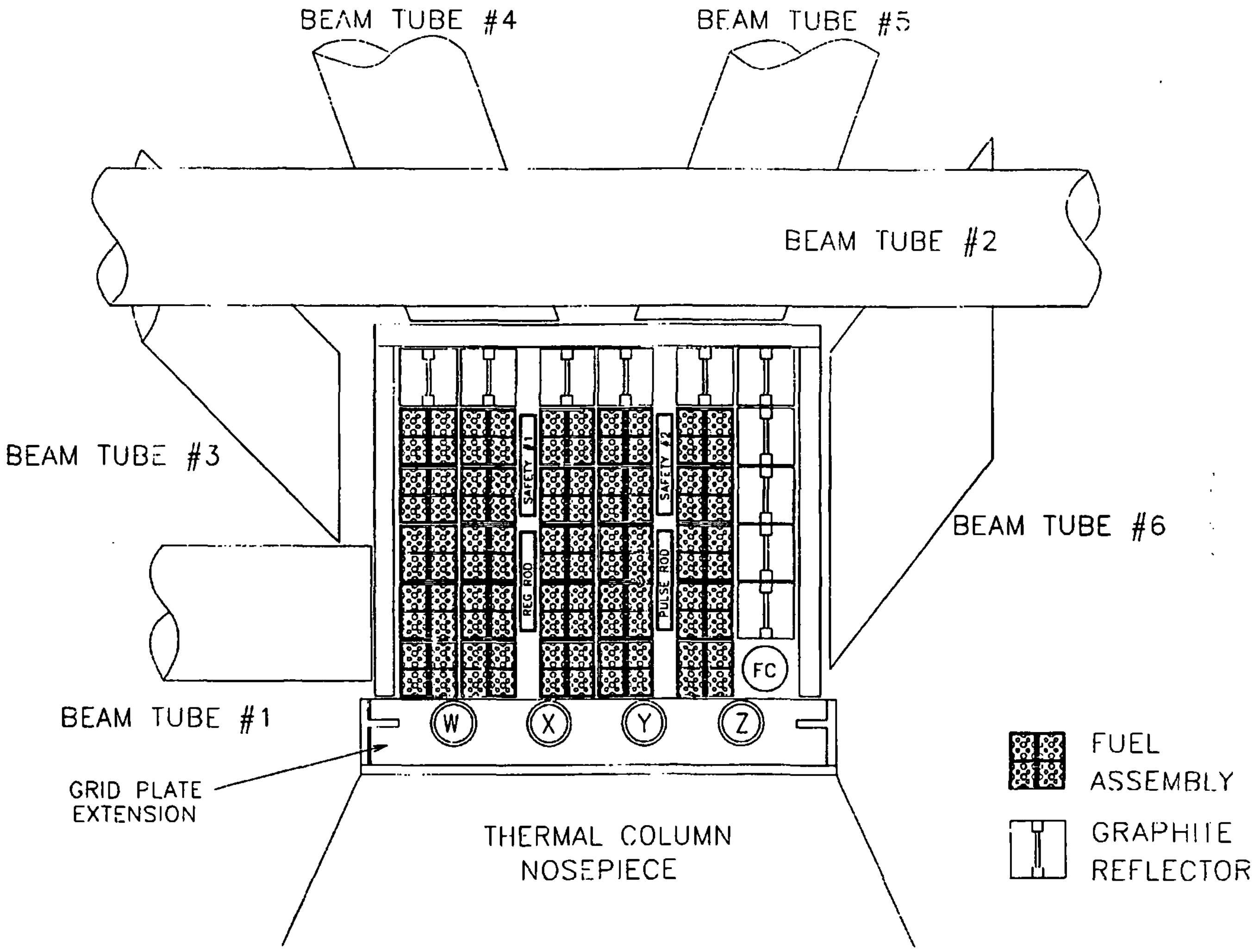
PULSTAR NUCLEAR REACTOR
5 X 5 STANDARD CORE

Figure 3-8A



PULSTAR NUCLEAR REACTOR
5 X 5 REFLECTED CORE # 1

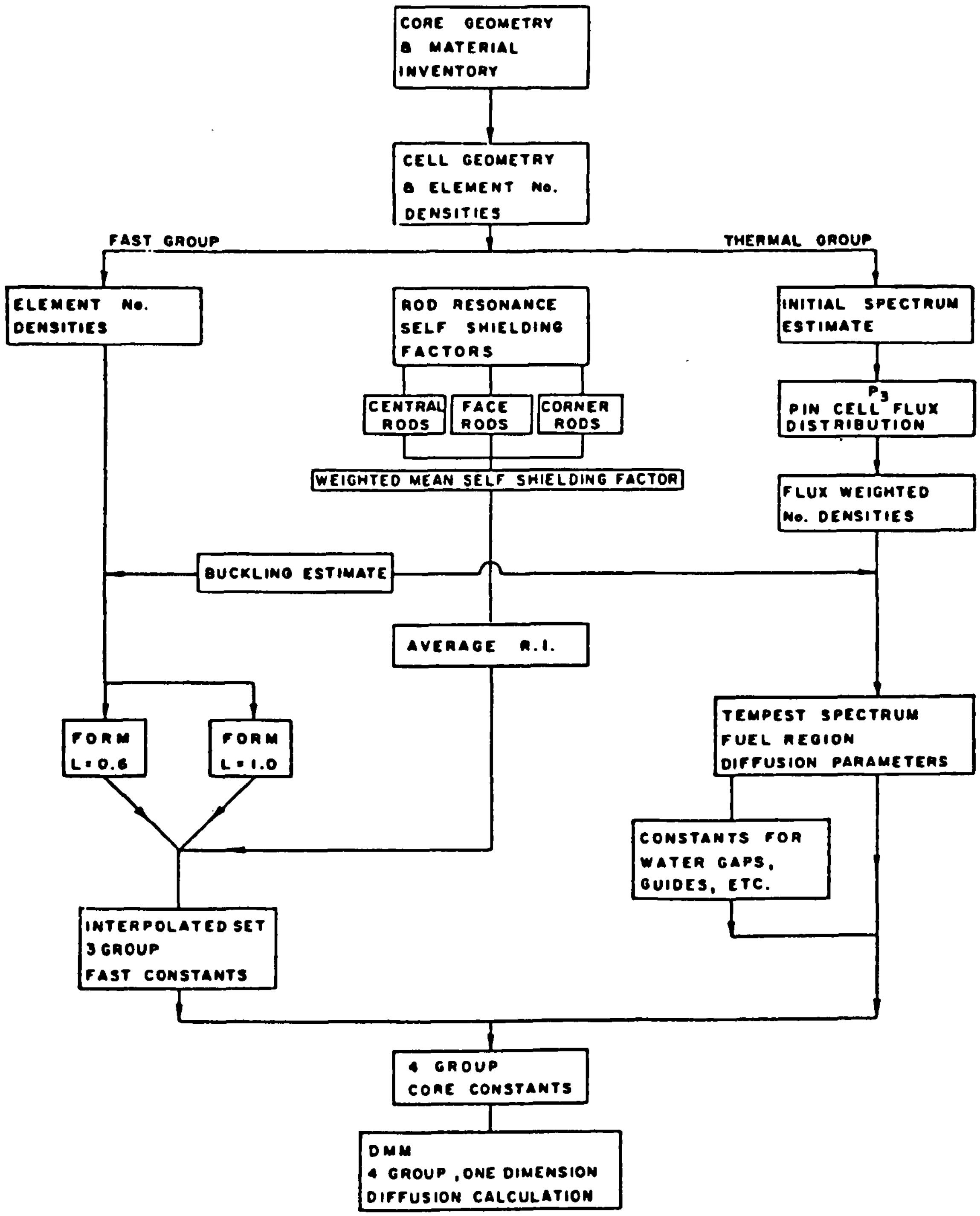
Figure 3-8B



PULSTAR NUCLEAR REACTOR
5 X 5 REFLECTED CORE # 3
(NOT TO SCALE)

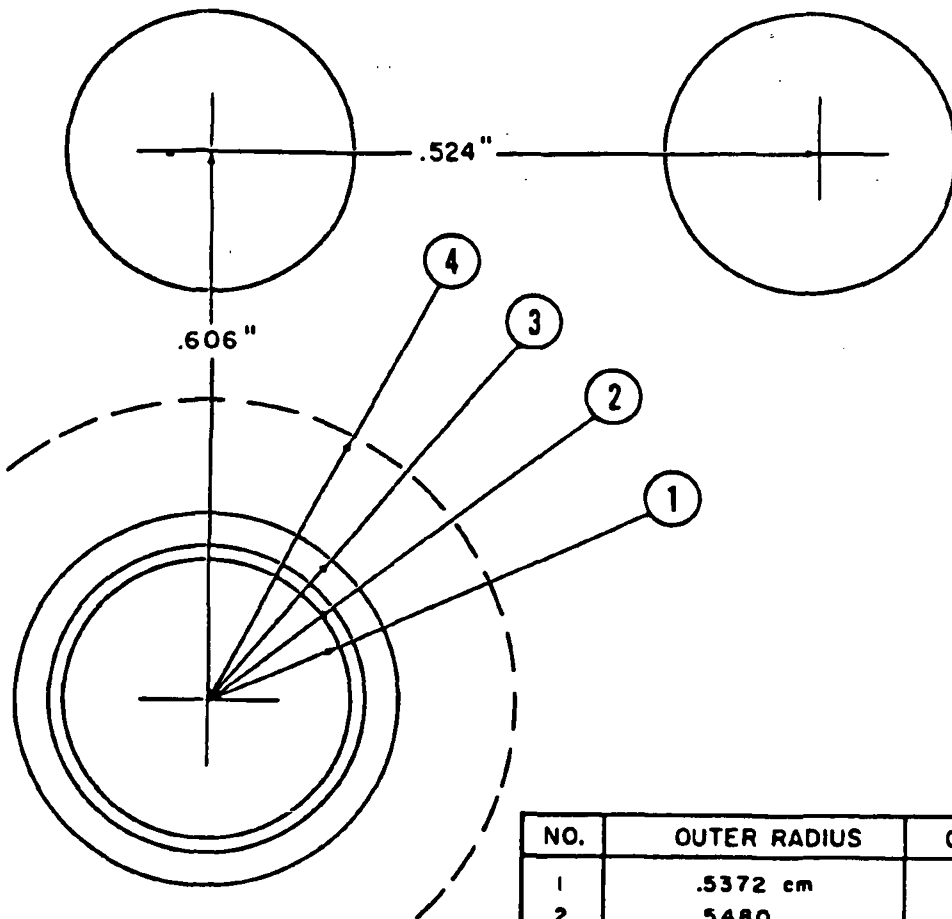
Figure 3-8C

Figure 3-9



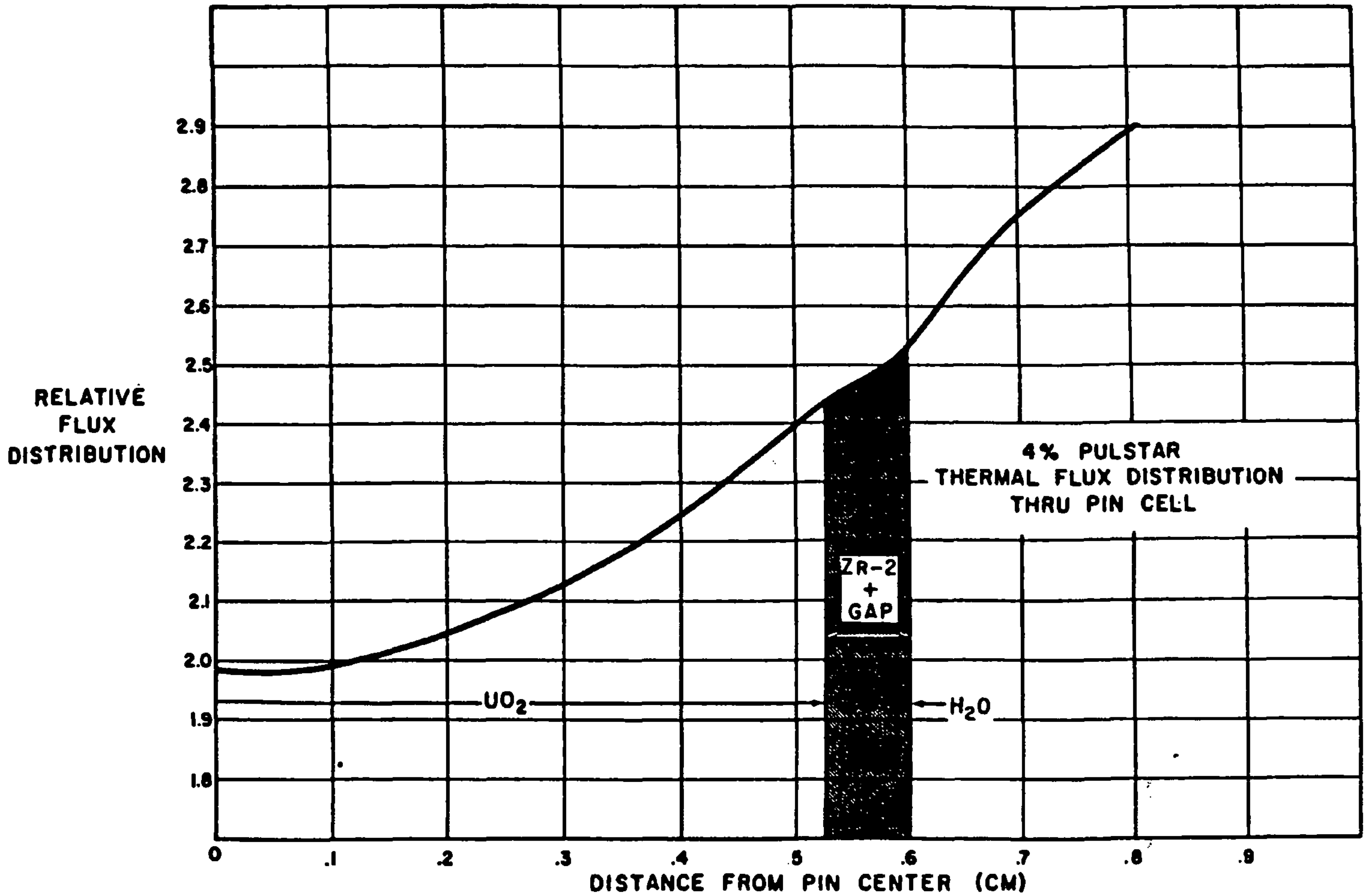
LOGICAL FLOW DIAGRAM

Figure 3-10



NO.	OUTER RADIUS	COMPOSITION
1	.5372 cm	UO_2
2	.5480	GAP
3	.6001	ZR CLAD
4	.8076	H_2O

MODEL FOR SPAR CELL CALCULATIONS



THERMAL FLUX DISTRIBUTION THROUGH PIN CELL
(4% ENRICHMENT)

Figure 3-11

Figure 3-12

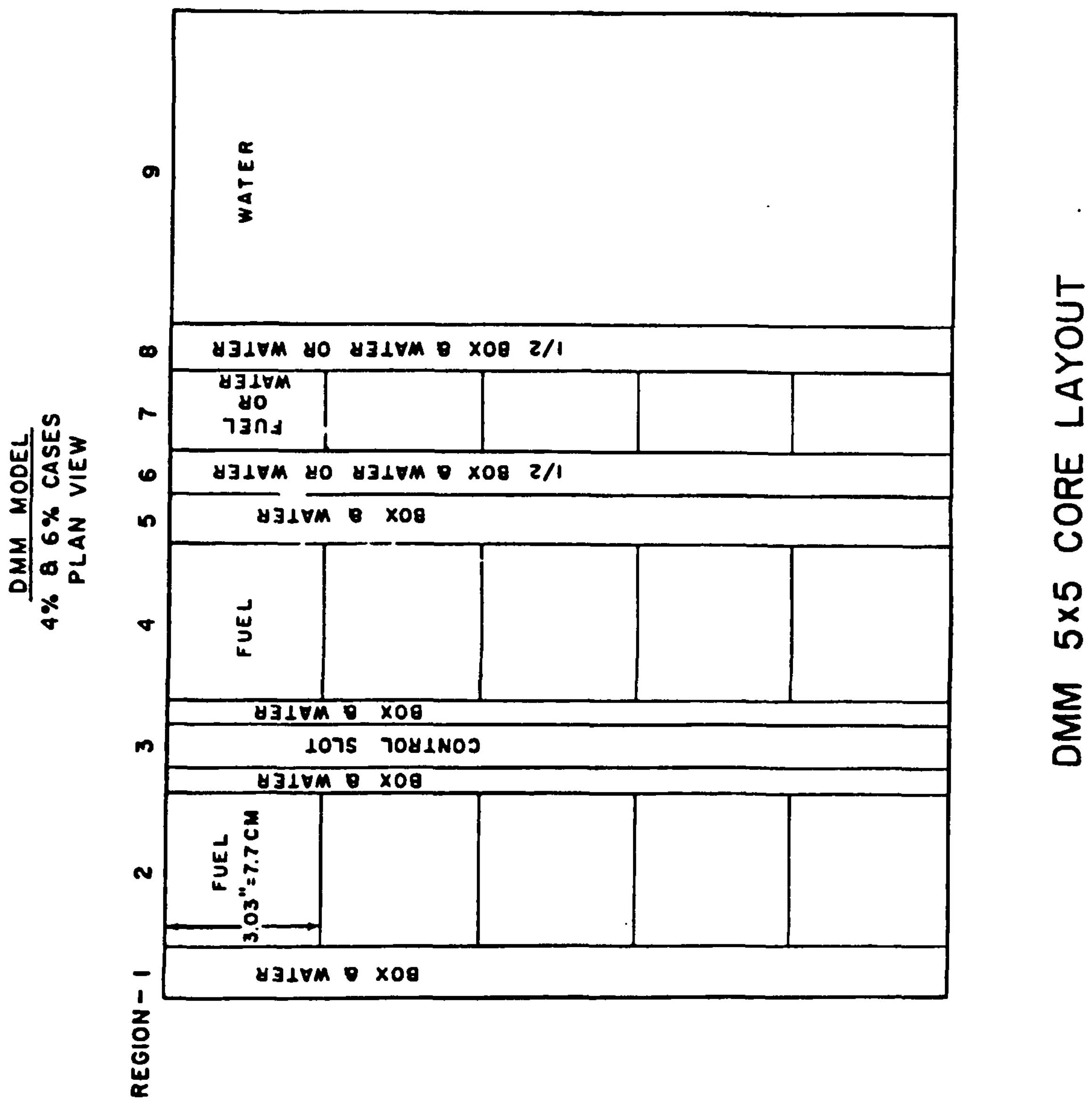

DMM MODEL
4% & 6% CASES
PLAN VIEW
REGION – 1
2
3
4
5
6
7
8
9
BOX & WATER
FUEL
3.03"=7.7CM
BOX & WATER
CONTROL SLOT
BOX & WATER
FUEL
BOX & WATER
1/2 BOX & WATER OR WATER
FUEL OR WATER
1/2 BOX & WATER OR WATER
WATER
DMM 5x5 CORE LAYOUT

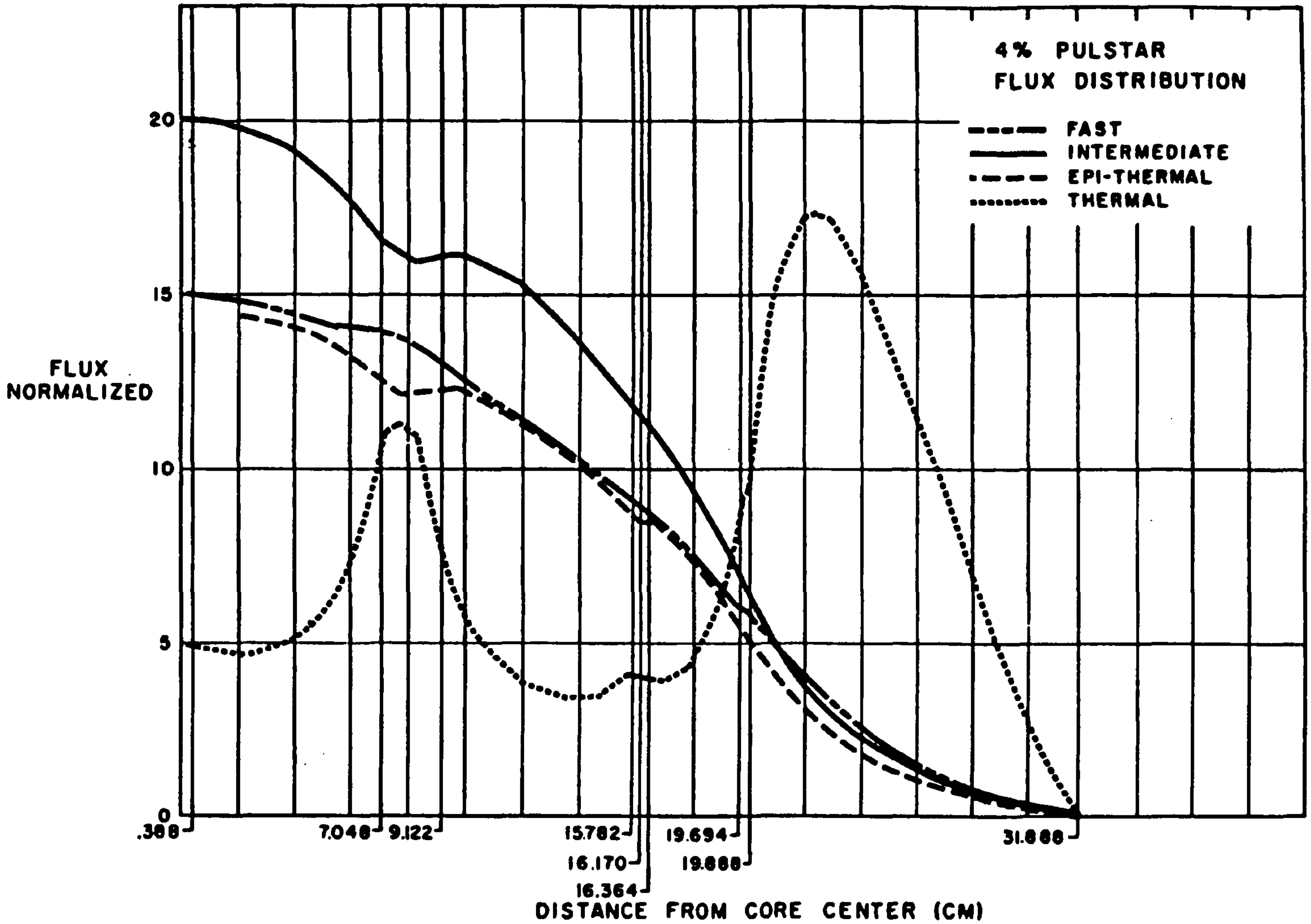
Figure 3-13

4% & 6% CASES
GEOMETRY — SLAB — DISCRETE BOXES

REGION	1	2	3	4	5	6	7	8	9
MATERIAL: 4% CASE	BOX & WATER	FUEL	CONT SLOT & 2 BOX & WAT	FUEL	BOX & WATER	1/2 BOX & WATER	1/2 FUEL	1/2 BOX & WATER	WATER
MATERIAL: 6% CASE	"	"	"	"	"	WATER	WATER	WATER	WATER
Δx, cm	.388	6.660	2.074	6.660	.388	.194	3.330	.194	12.000
X, cm 0	.388	7.048	9.122	15.782	16.170	16.364	19.694	19.888	31.888
NUMBER OF ZONES	2	8	4	8	2	2	4	2	10

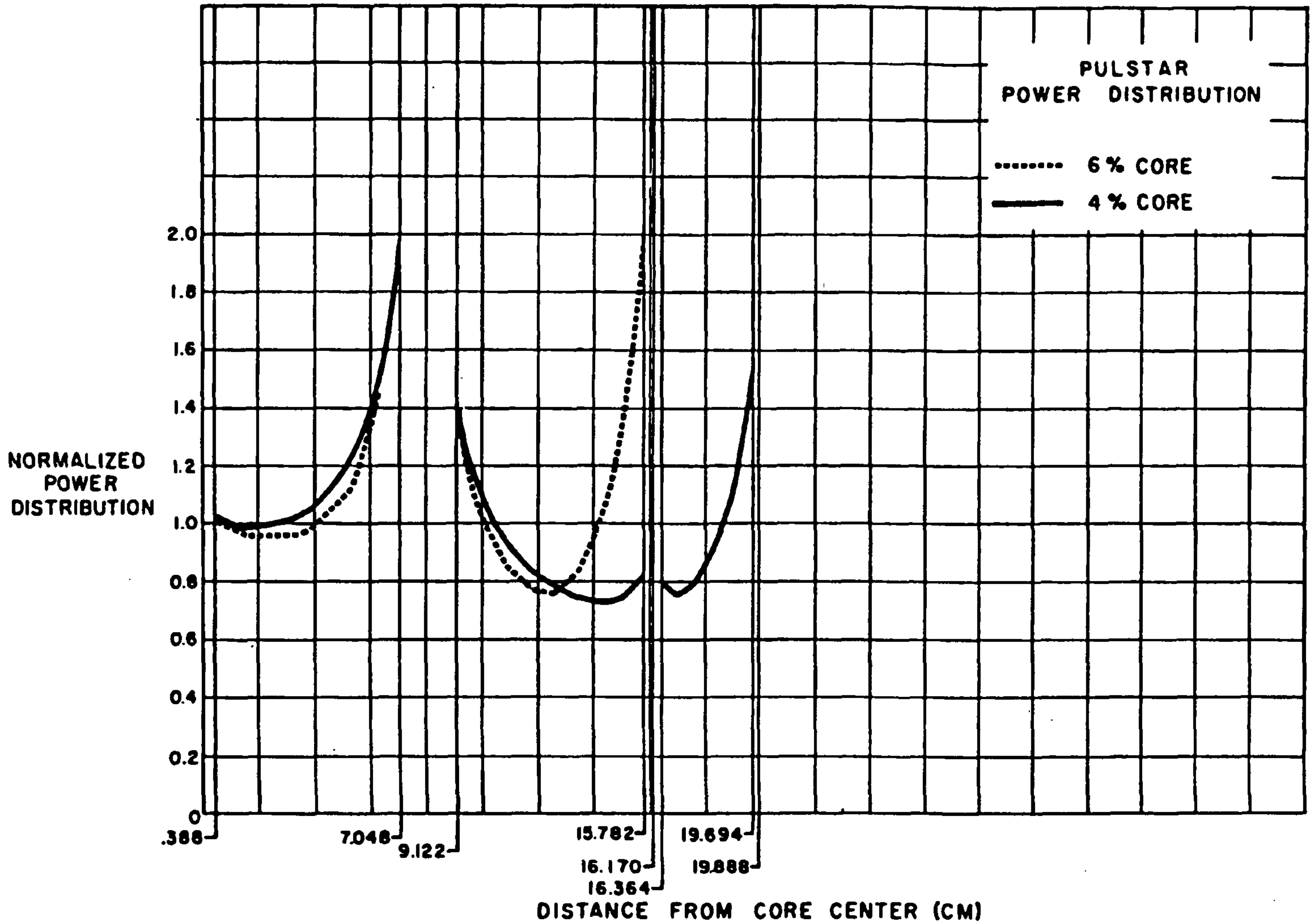
SYMMETRY AXIS

DMM MODEL



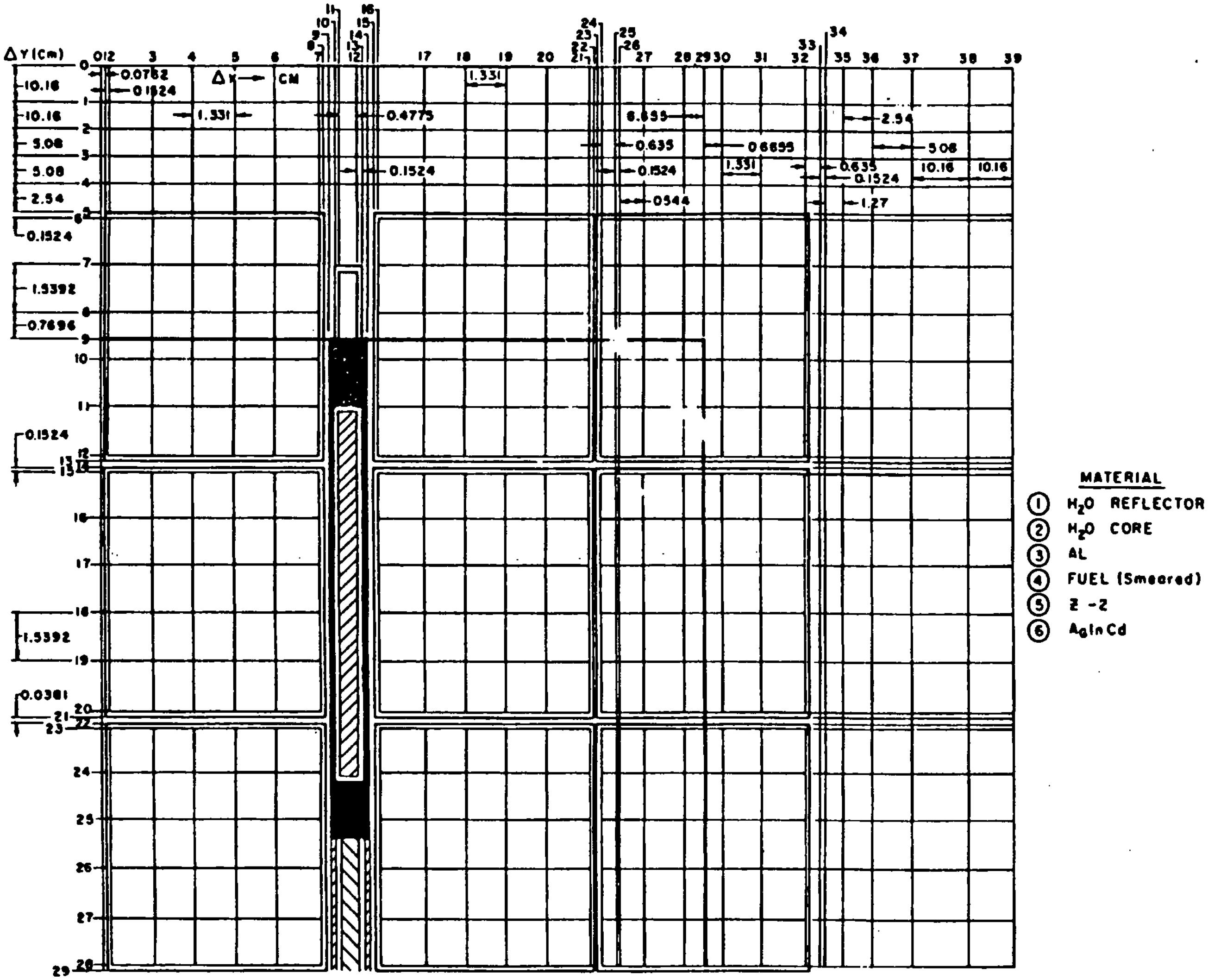
4% FOUR-GROUP NEUTRON FLUXES

Figure 3-14



PULSTAR 5x5 CORE POWER DISTRIBUTION

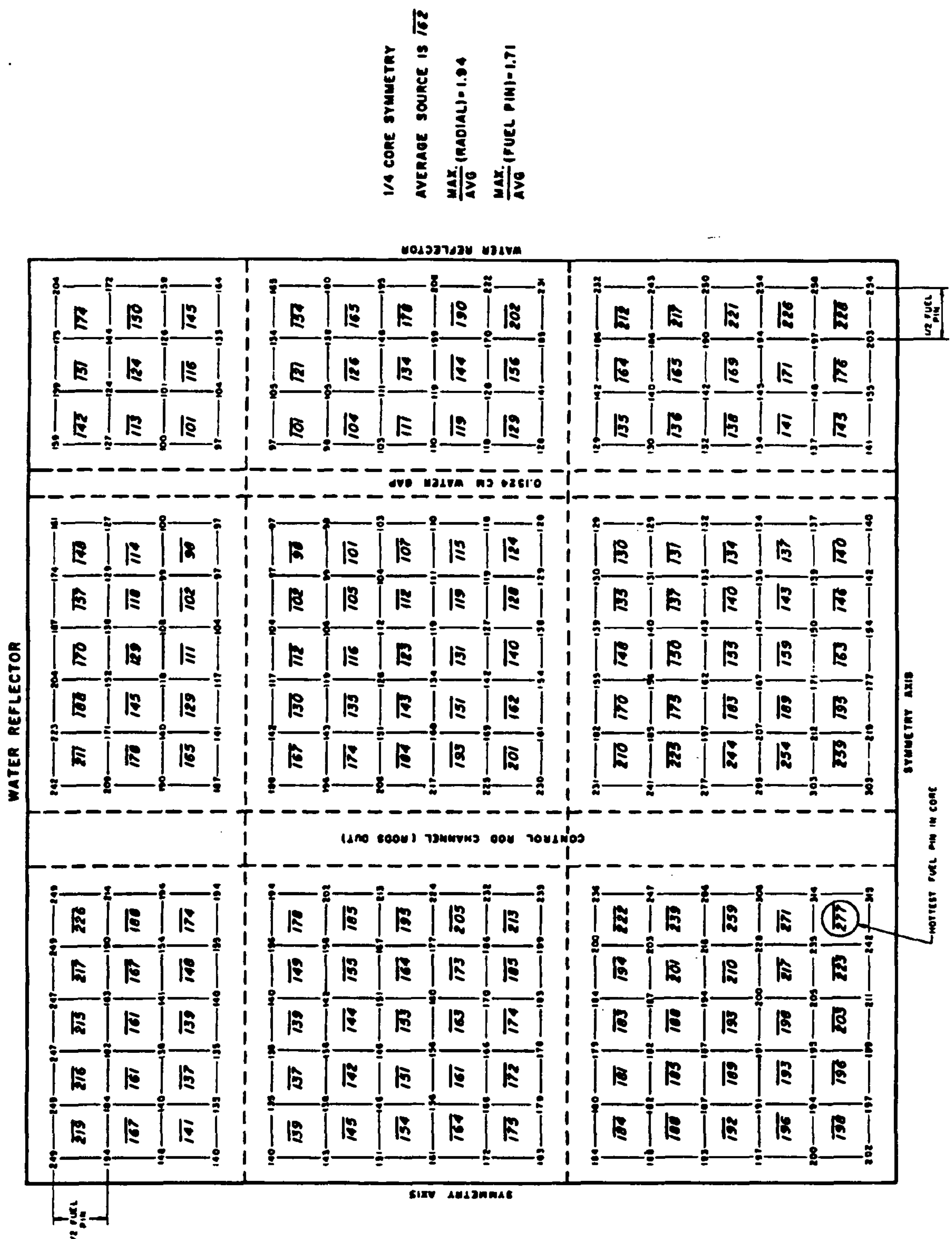
Figure 3-15



PDQ-2 FOR 1/4 CORE LAYOUT

Figure 3-16

Figure 3-17



PULSTAR POWER DISTRIBUTION IN 5x5 CORE

5 X 5 STANDARD CORE STARTUP TEST DATA

XE BUILDUP & DECAY FOR A 24 HR OPERATION AT 1 MW

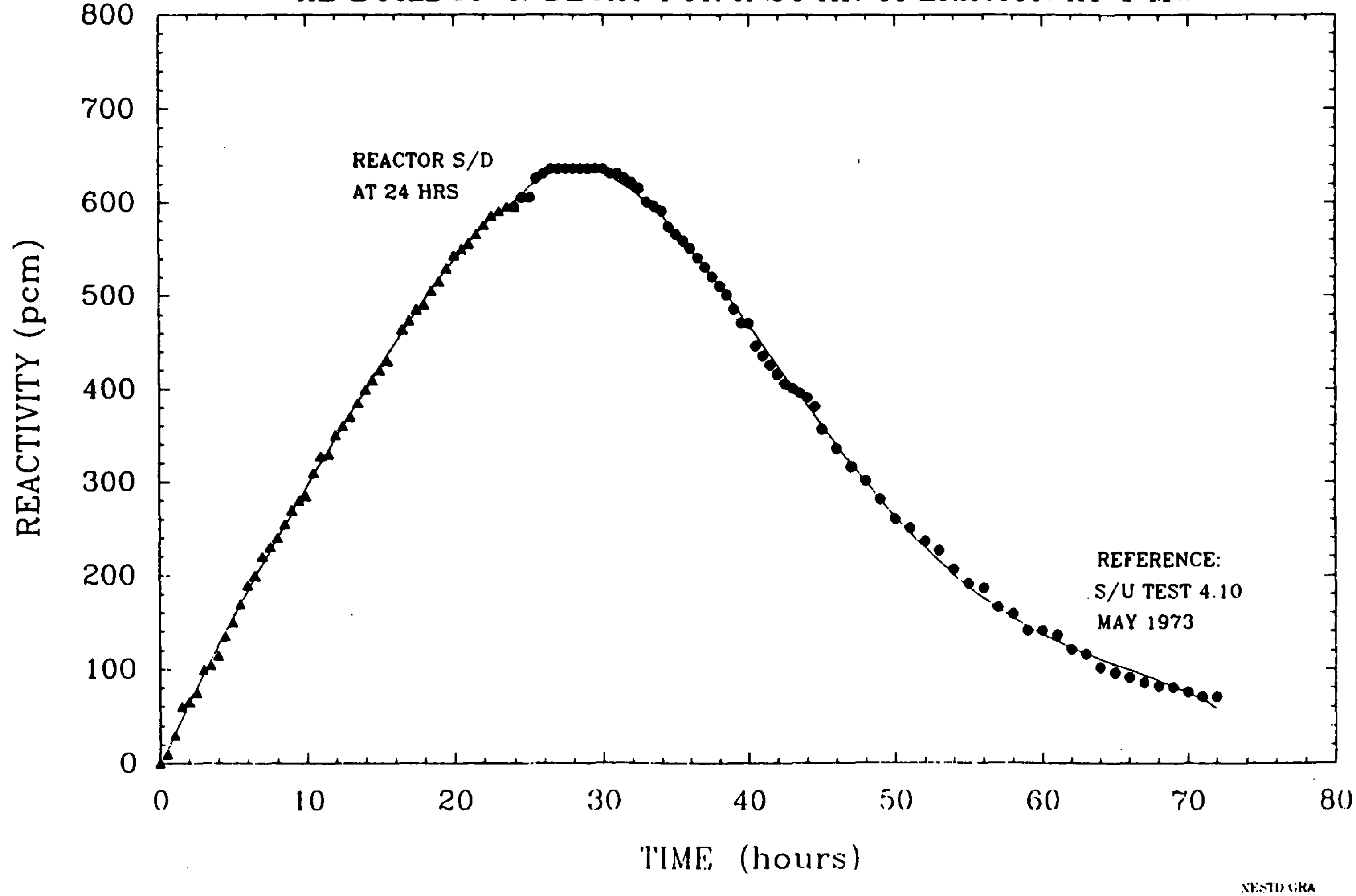


Figure 3-18A

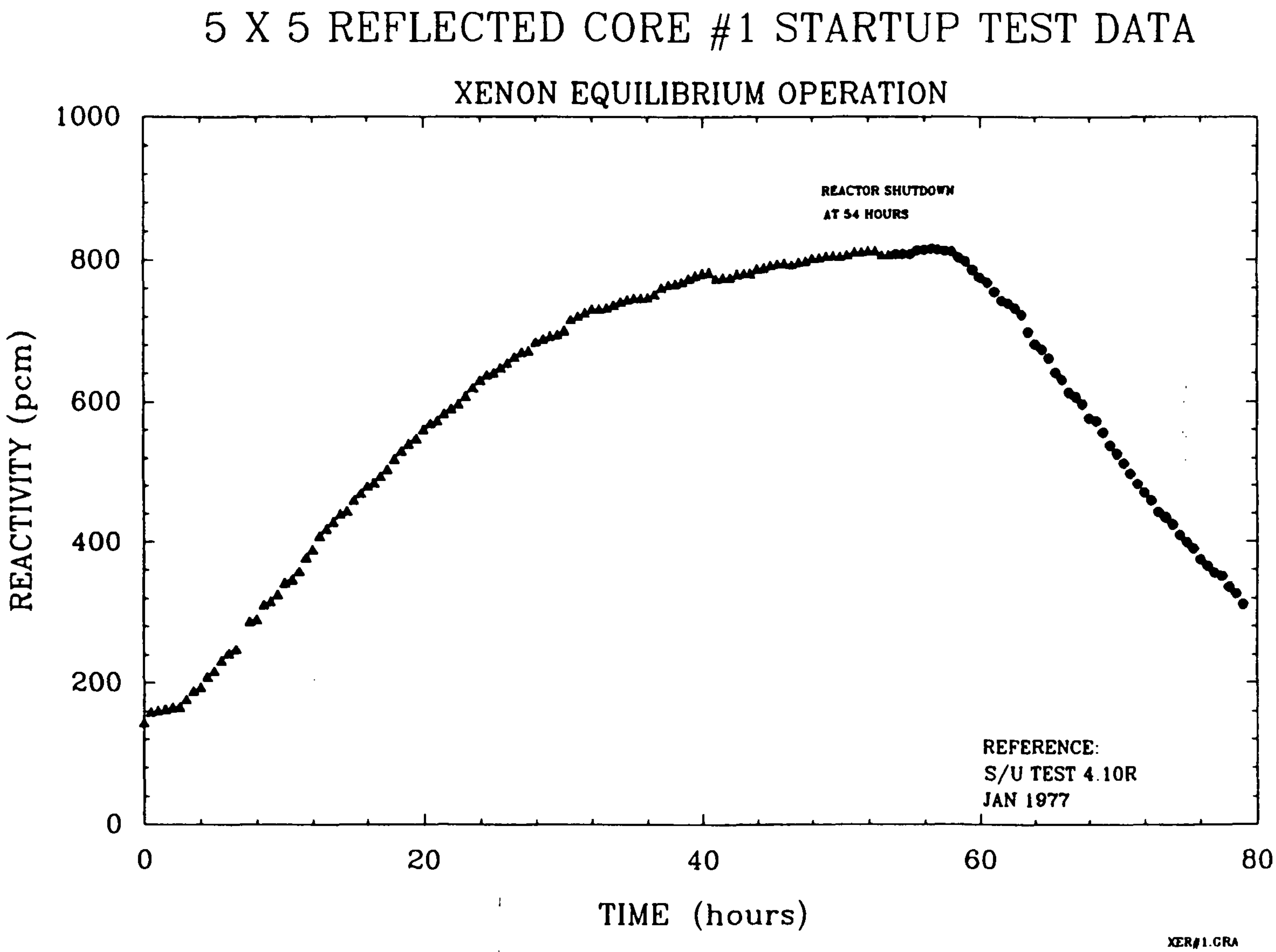

5 X 5 REFLECTED CORE #1 STARTUP TEST DATA
XENON EQUILIBRIUM OPERATION
REACTOR SHUTDOWN
AT 54 HOURS
REFERENCE:
S/U TEST 4.10R
JAN 1977
REACTIVITY (pcm)
0
200
400
600
800
1000
TIME (hours)
0
20
40
60
80
XER#1.GRA

Figure 3-18B

5 X 5 REFLECTED CORE #1 STARUP TEST DATA

XE BUILDUP & DECAY FOR A 4 HR OPERATION AT 1 MW

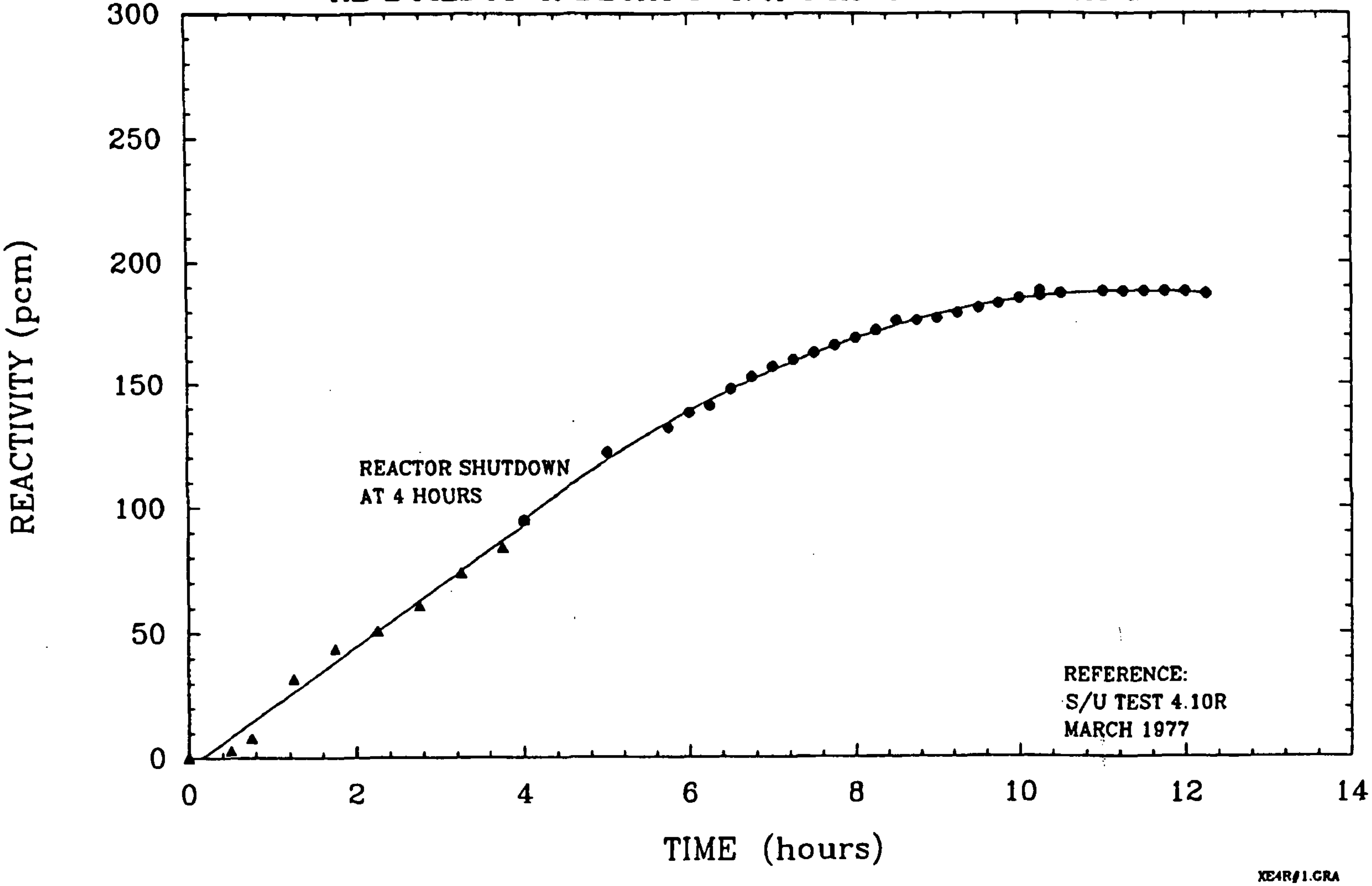


Figure 3-18C

5 X 5 REFLECTED CORE #3 STARTUP TEST DATA

XE BUILDUP & DECAY FOR A 4 HR OPERATION AT 1 MW

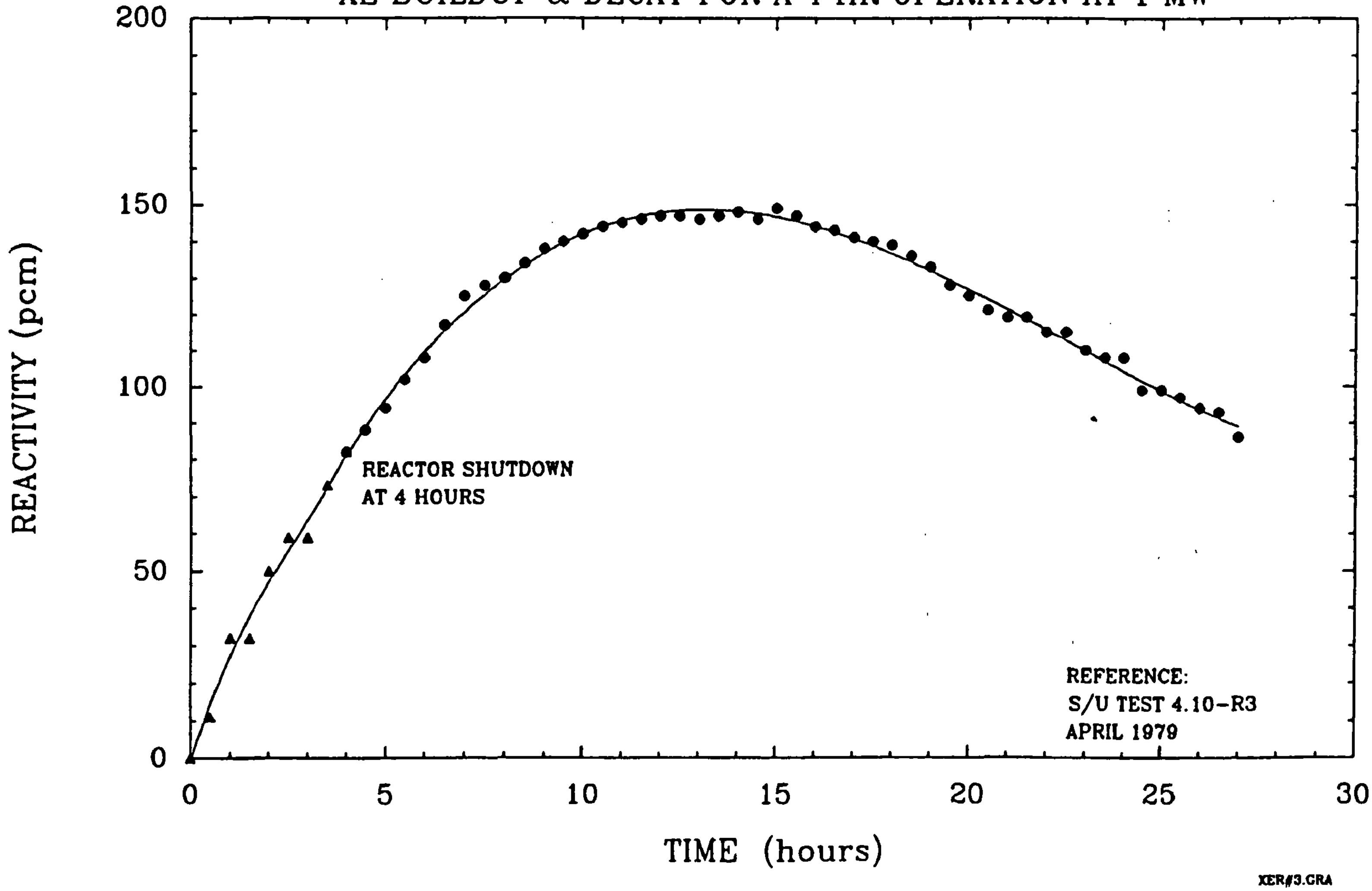


Figure 3-18D

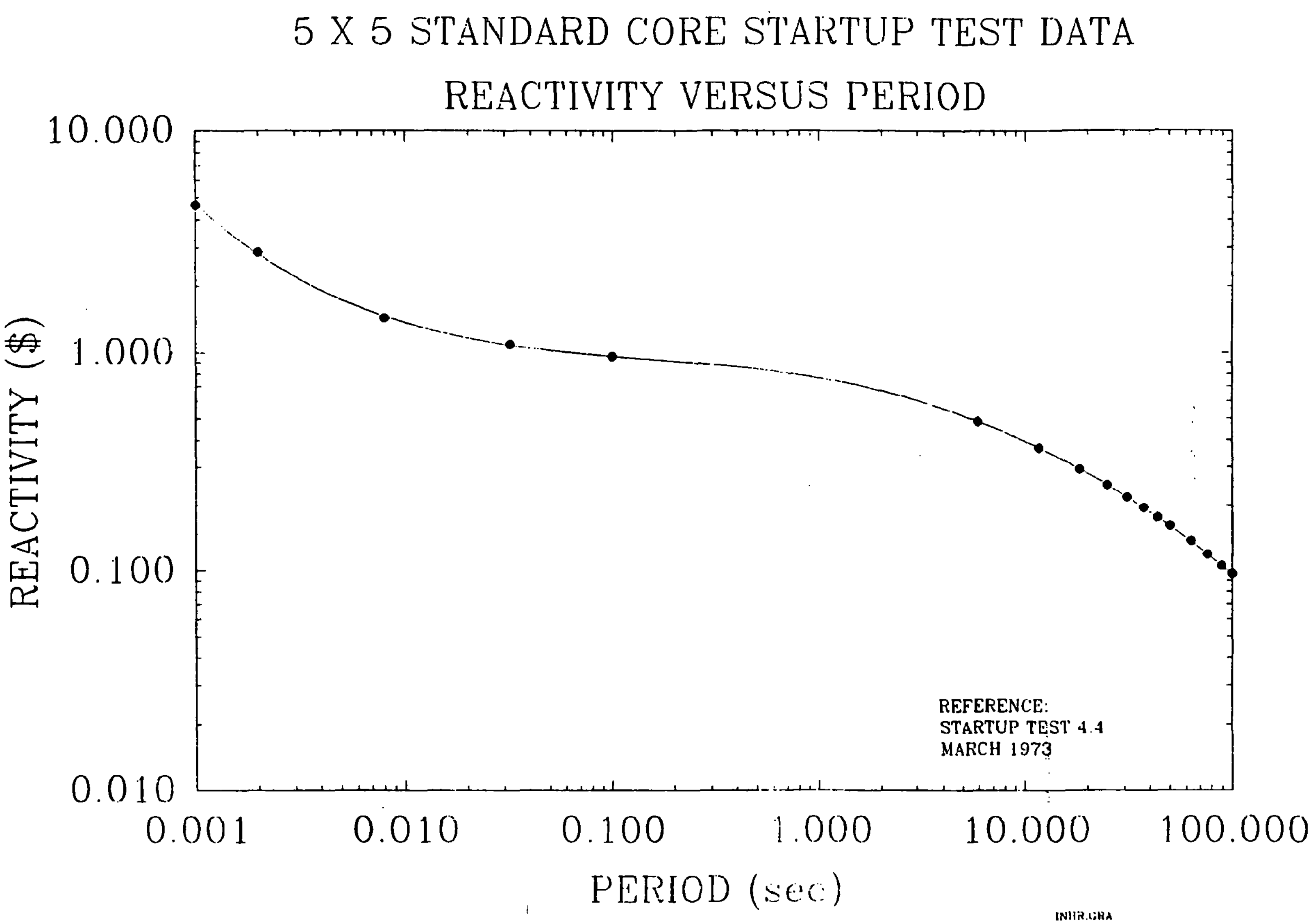

5 X 5 STANDARD CORE STARTUP TEST DATA
REACTIVITY VERSUS PERIOD
REACTIVITY ($)
10.000
1.000
0.100
0.010
0.001
0.010
0.100
1.000
10.000
100.000
PERIOD (sec)
REFERENCE:
STARTUP TEST 4.4
MARCH 1973
INHR.GRA

Figure 3-19

Pulse Energy Release Measured by N−16 and Foils

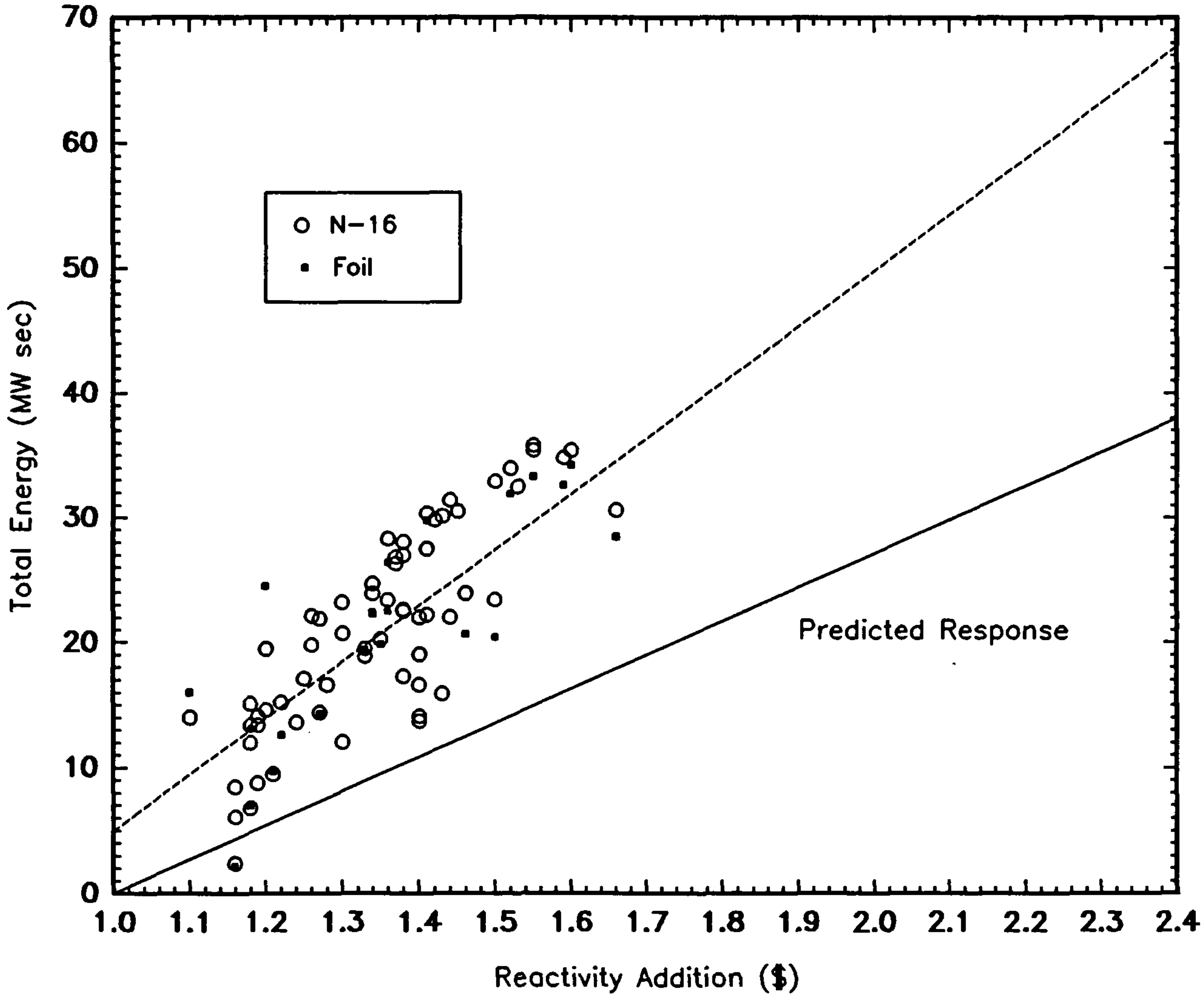


Figure 3-20

Measured Pulse Peak Power (MW)

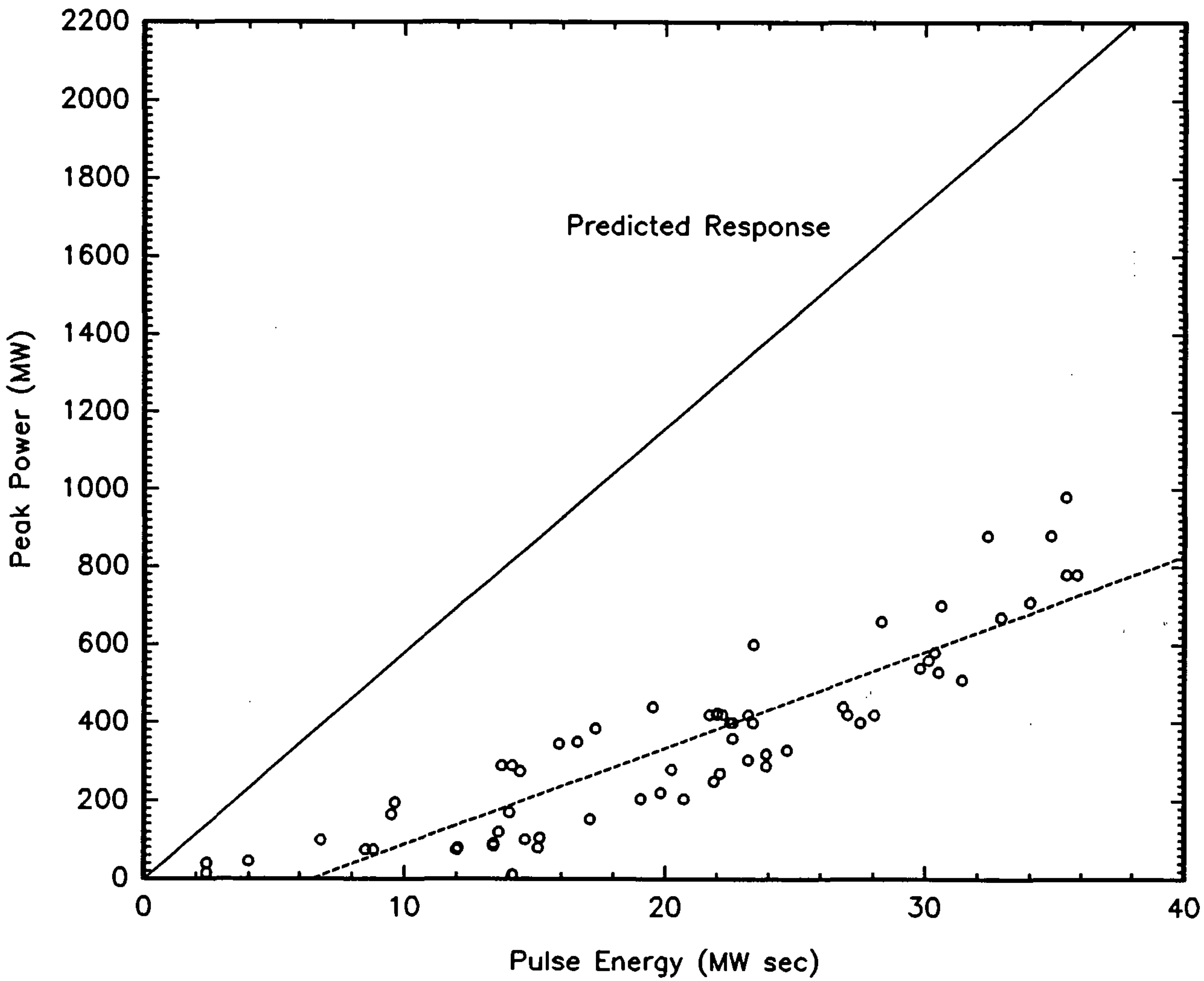


Figure 3-21

AVERAGE & HOT CHANNEL TEMPERATURE RISE vs. POWER
WITH NATURAL CONVECTION FLOW

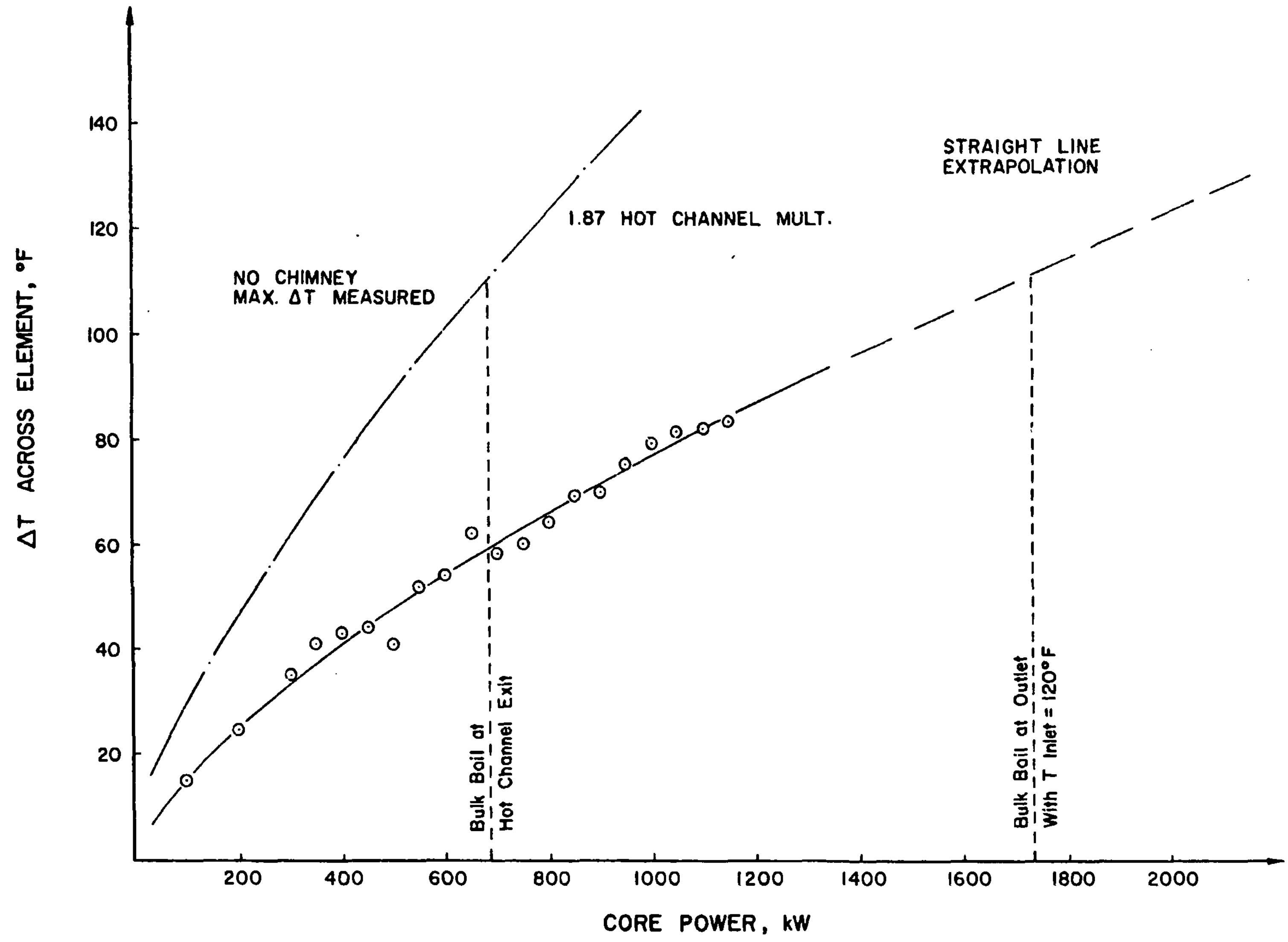


Figure 3-22

Figure 3-23

POWER LEVELS REQUIRED TO REACH DNB OR BULK BOILING IN THE HOT CHANNEL

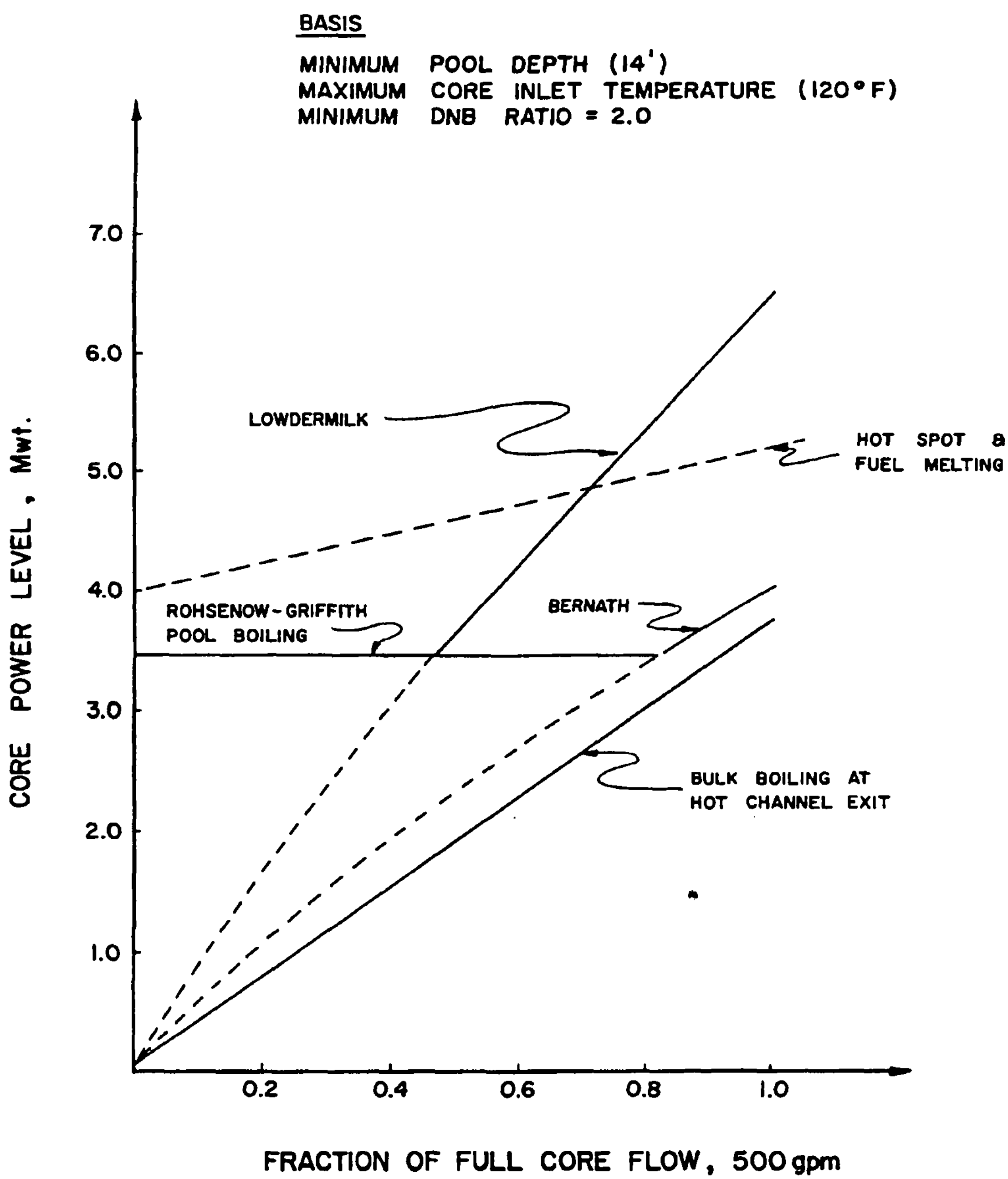
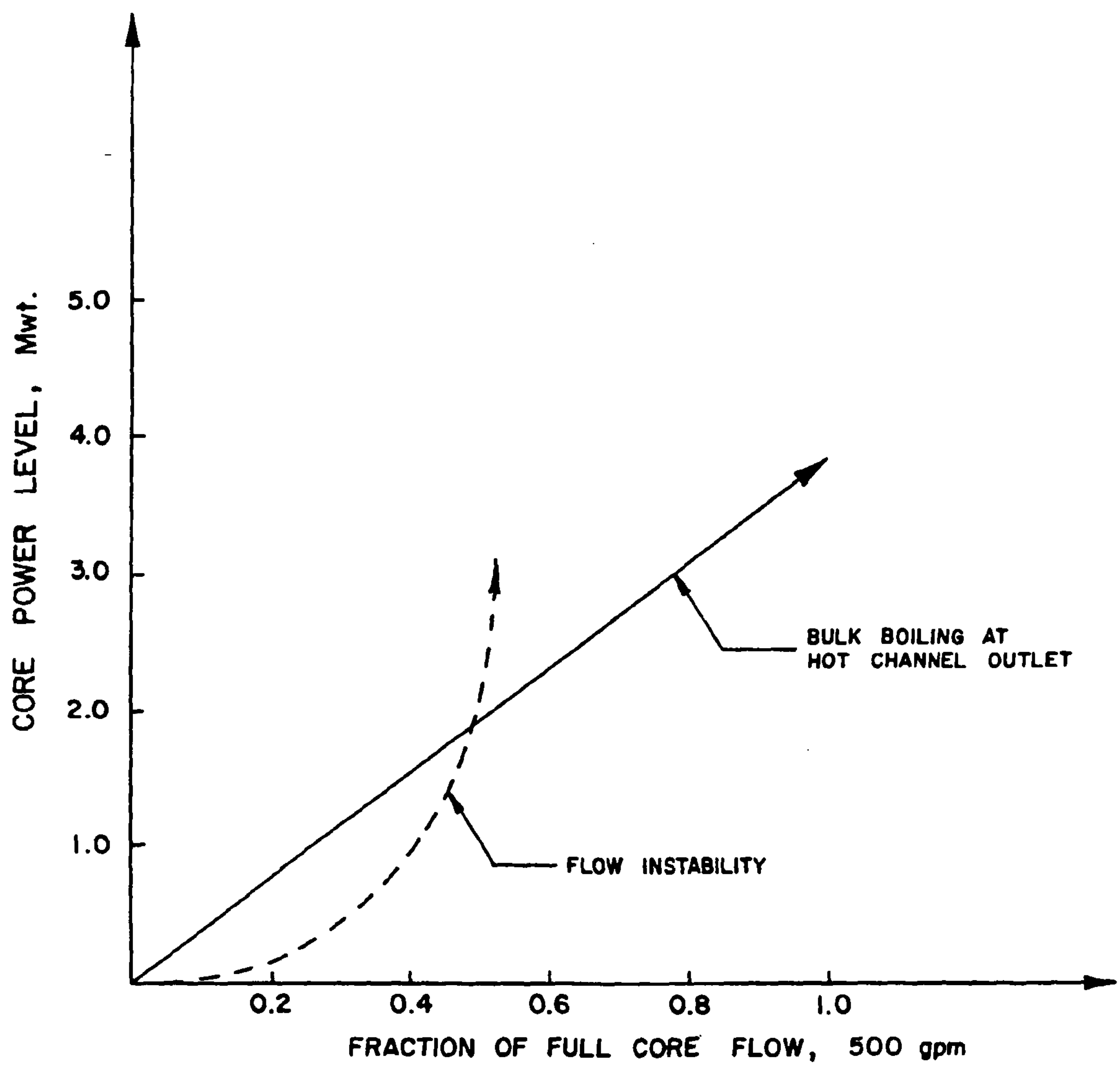


Figure 3-24

POWER LEVELS REQUIRED TO REACH BULK BOILING OR FLOW INSTABILITY IN THE HOT CHANNEL.



North Carolina State University PULSTAR Reactor
Core Excess Reactivity vs. Power History

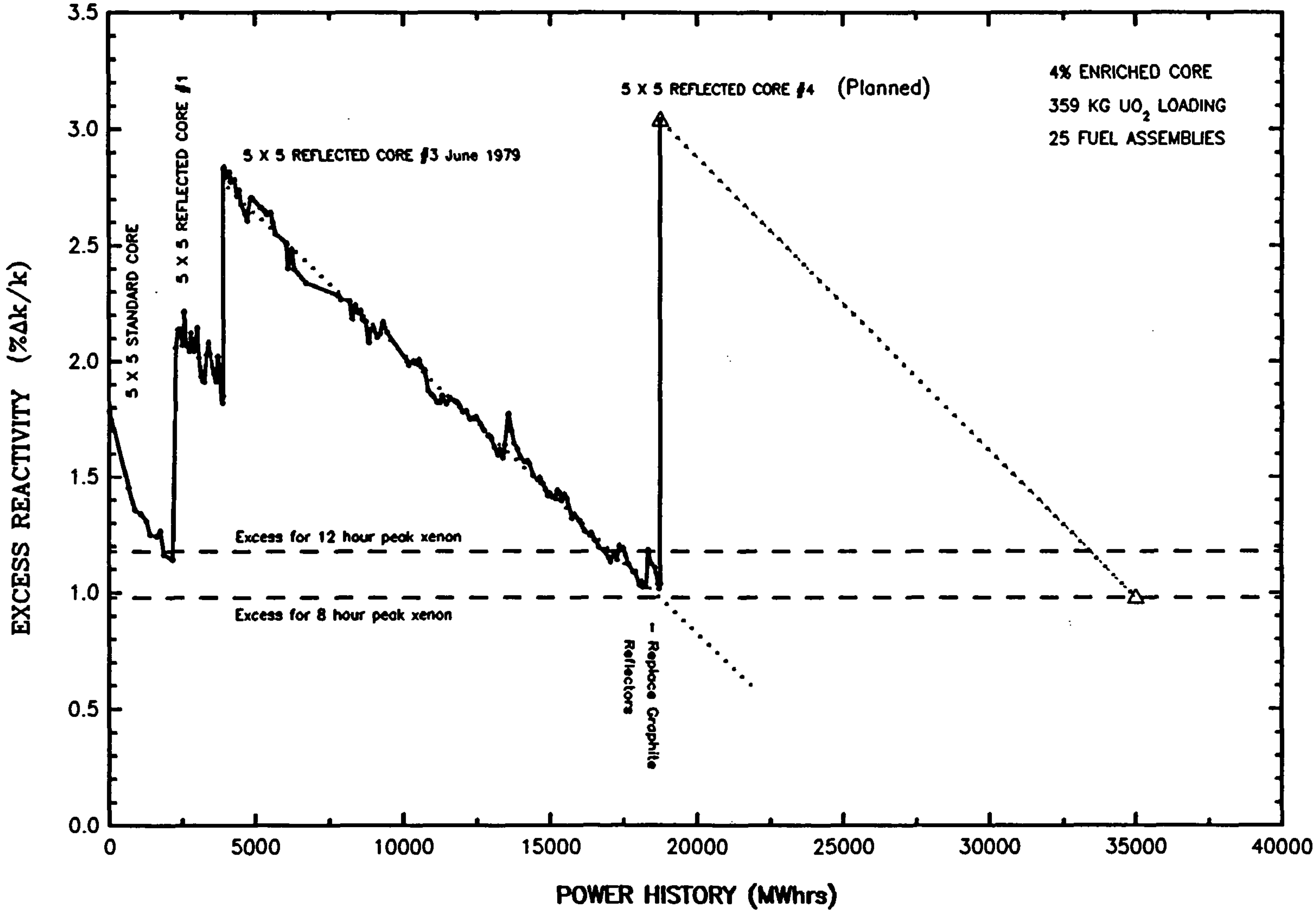


Figure 3-25

5 X 5 REFLECTED CORE #3 STARTUP TEST DATA

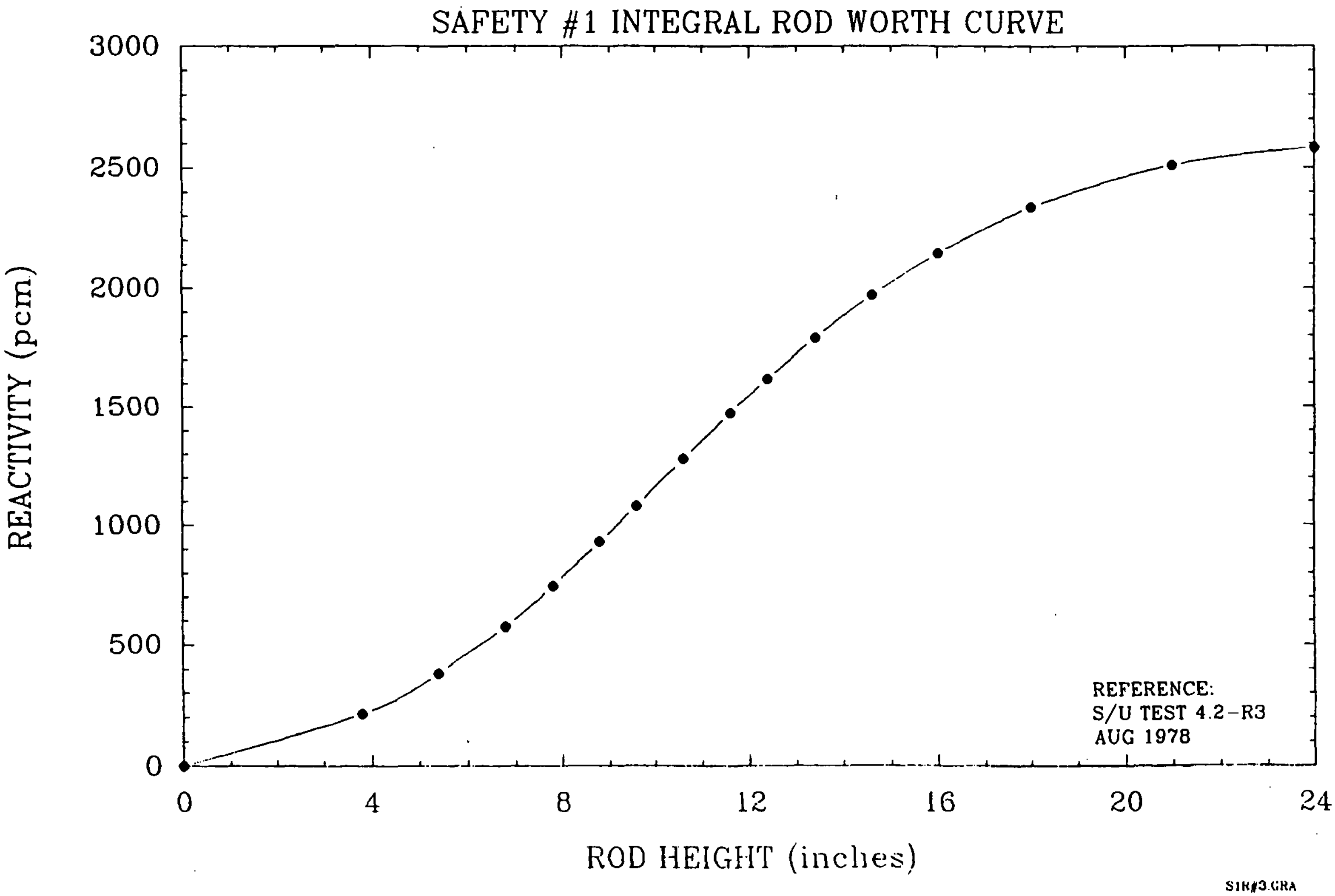


Figure 3-26

5 X 5 REFLECTED CORE #3 STARTUP TEST DATA

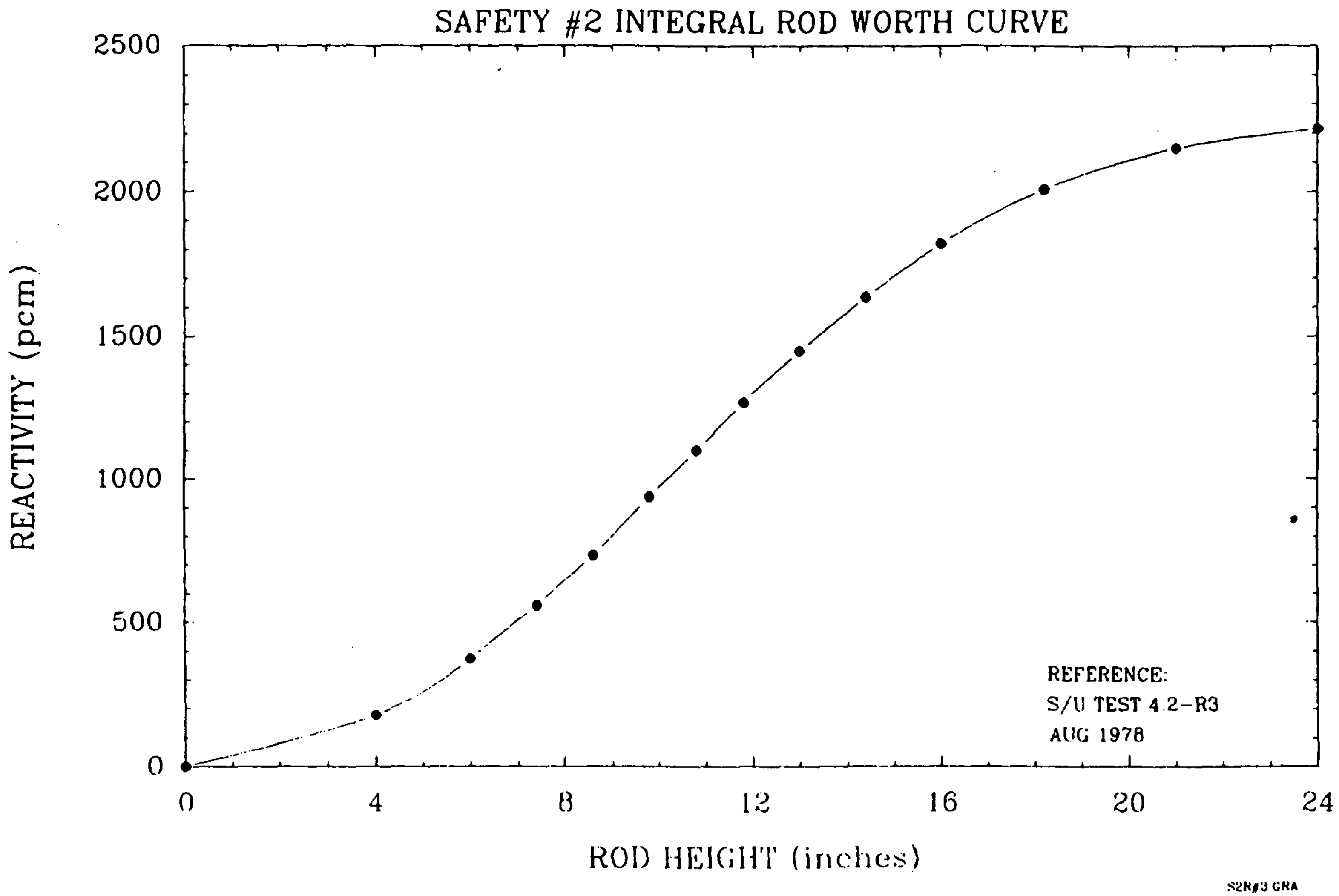


Figure 3-27

5 X 5 REFLECTED CORE #3 STARTUP TEST DATA

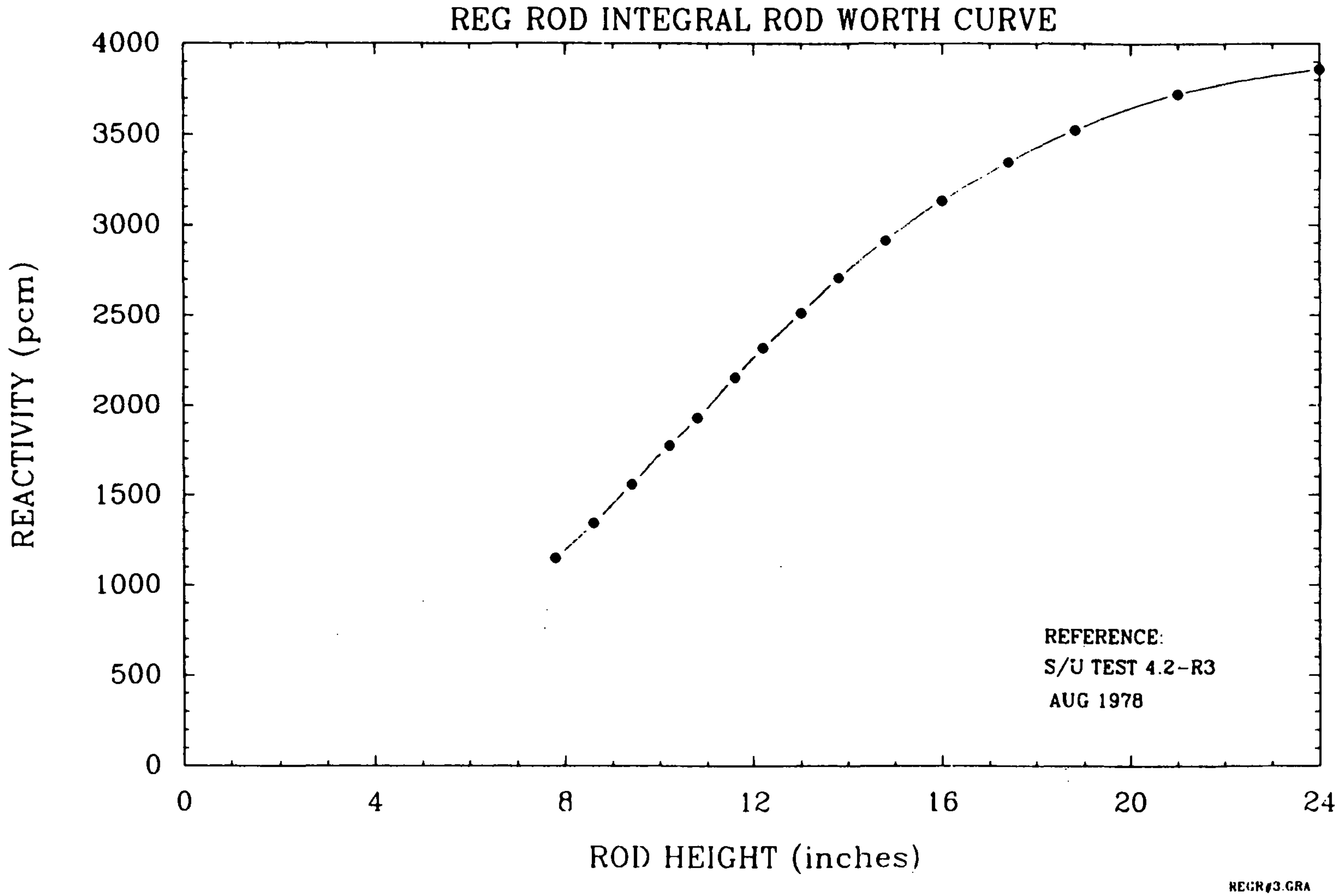


Figure 3-28

5 X 5 REFLECTED CORE #3 STARTUP TEST DATA

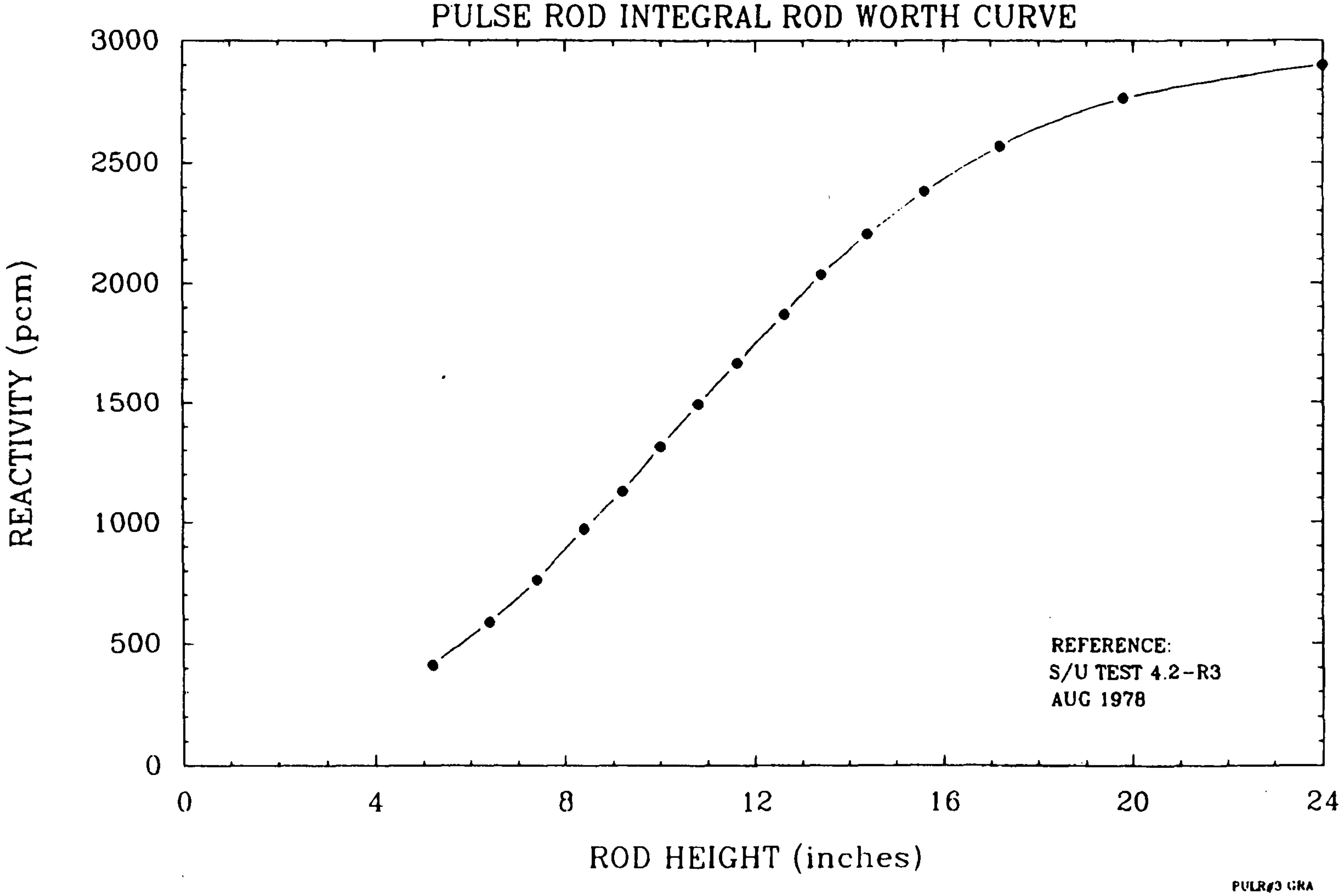


Figure 3-29

5 X 5 REFLECTED CORE #3 STARTUP TEST DATA

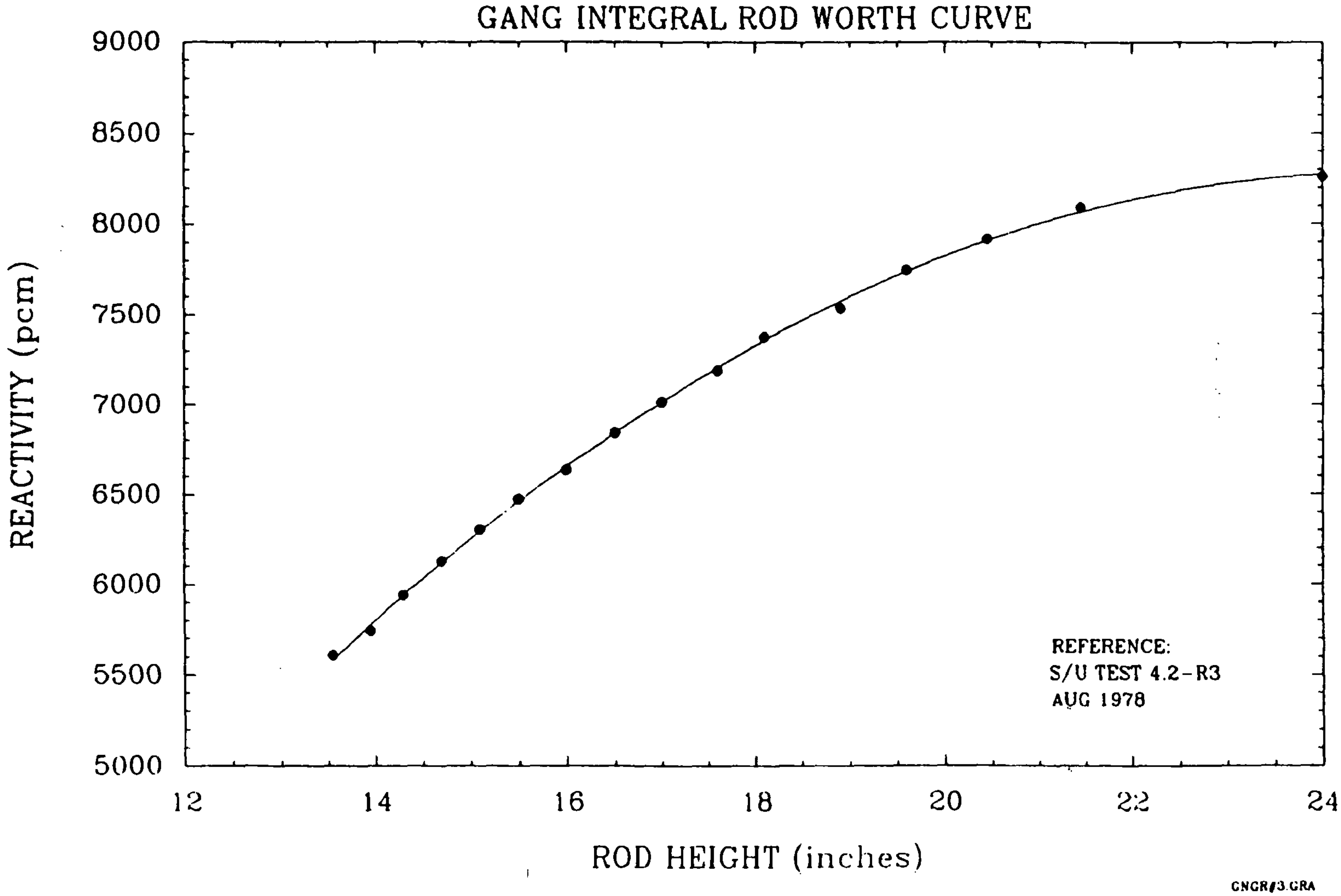


Figure 3-30

Figure 3-31

	1	2	3	4	5	6	
	GRAPHITE REFLECTOR	GRAPHITE REFLECTOR	GRAPHITE REFLECTOR	GRAPHITE REFLECTOR	GRAPHITE REFLECTOR	GRAPHITE REFLECTOR	A
	0.664	0.828	0.977	0.980	0.887	GRAPHITE REFLECTOR	B
	0.753	0.936	1.153	1.078	1.018	GRAPHITE REFLECTOR	C
	0.863	1.027	1.229	1.326	1.263	GRAPHITE REFLECTOR	D
	0.801	0.949	1.125	1.216	1.078	GRAPHITE REFLECTOR	E
	0.690	0.949	1.146	1.088	0.972	GRAPHITE REFLECTOR	F

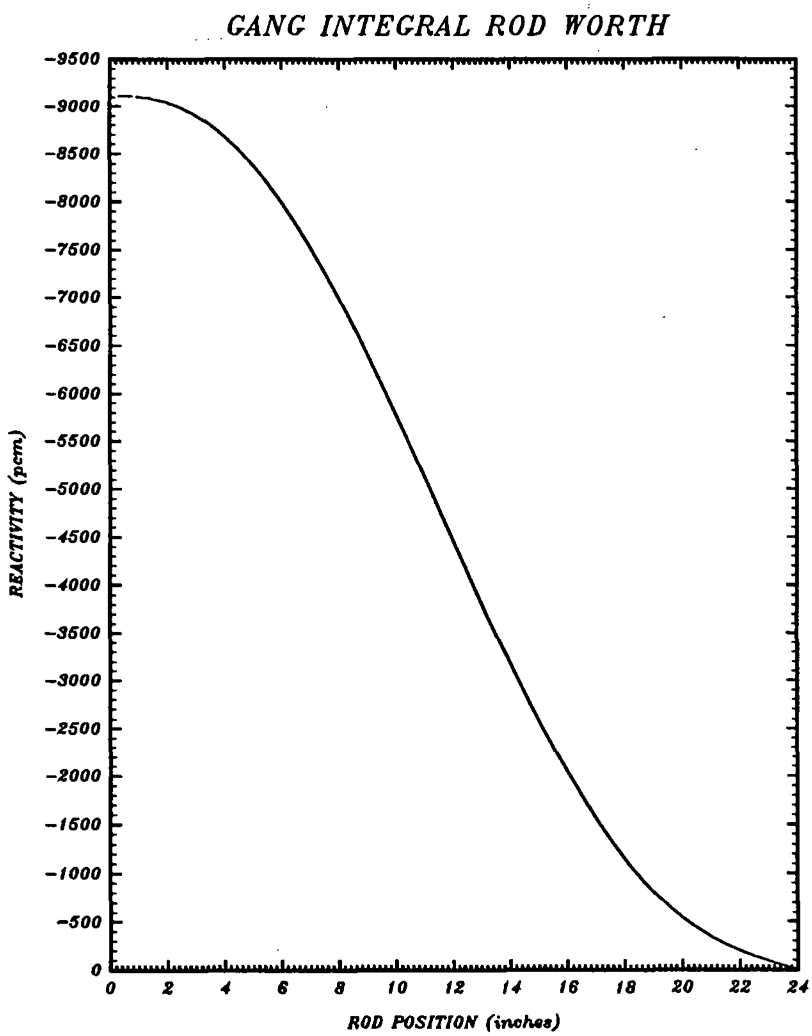
5 X 5 REFLECTED CORE # 3 FLUX DISTRIBUTION SPECIFICS:

1. RELATIVE ASSEMBLY POWER IS DEPICTED ABOVE.
2. PEAK LOCAL FLUX WAS OBSERVED IN COOLANT CHANNEL A-1 OF GRID POSITION 3-E WITH A TOTAL PEAK TO AVERAGE RATIO OF 2.116.
3. UPPER HALF OF CORE PRODUCES 35.9% OF TOTAL POWER, LOWER HALF PRODUCES 64.1% OF TOTAL POWER.
4. MAXIMUM ASSEMBLY TOTAL POWER OCCURS IN GRID POSITION 4-D.

Reference: 5 X 5 REFLECTED CORE # 3 STARTUP TEST, 1978.

5 X 5 REFLECTED CORE # 3 FLUX DISTRIBUTION

Figure 3-32


GANG INTEGRAL ROD WORTH
REACTIVITY (pcm)
-9500
-9000
-8500
-8000
-7500
-7000
-6500
-6000
-5500
-5000
-4500
-4000
-3500
-3000
-2500
-2000
-1500
-1000
-500
0
0
2
4
6
8
10
12
14
16
18
20
22
24
ROD POSITION (inches)

4 REACTOR COOLANT AND SUPPORT SYSTEMS

4.1 Design Bases

4.1.1 Functional Requirements

The Reactor Coolant System, using demineralized water, is designed to remove one megawatt of heat from the PULSTAR Reactor operating in the steady-state mode with forced convection cooling and have sufficient capacity to operate at power levels up to 150 kW with natural convection cooling. Equally important is the ability to cool the reactor during reverse flow upon failure of the primary coolant pump. The analyses in support of this design are found in Section 3 with pertinent features of the system design found in the following description.

4.2 System Design and Operation

The Primary Coolant System is a closed loop type operating at atmospheric pressure. The heat of fission is absorbed by the coolant as it passes down through the reactor core and is transferred to the Secondary Cooling System through a tube and shell type heat exchanger. The heat is then removed from the Secondary System by a standard cooling tower. The coolant system and its supporting components are shown schematically on Figures 4-1A through 4-1F.

4.2.1 Primary System

The reactor pool consists of a 0.25 inch (0.64 cm) aluminum liner which is surrounded by normal and special high density reinforced concrete structure for shielding purposes.

Two cylindrical fuel storage pits,
below the bottom of the pool liner. Each fuel storage pit has
locations in a subcritical configuration. To prevent stagnation of the water in the fuel storage pits, a small amount of water is directed from the inlet pipe to the bottom of the pits through aluminum tubing. An additional fuel storage locations are provided by two linear racks of the pool in a subcritical configuration.

There are two 10 inch (25.4 cm) pipe penetrations at the bottom of the pool liner for the primary coolant inlet and outlet connections. There is a 2 inch (5.1 cm) pipe attached to a weir about 2 feet (61 cm) from the top of the pool liner. This pipe directs any primary overflow to the Liquid Radioactive Drain System. In addition to these penetrations for the passage of primary coolant, there are a number of liner penetrations which accommodate the beam tubes located on three sides of the core. Two sleeves, located near the top of the

pool liner, allow for the passage of the pneumatic transfer system.

In forced convection (refer to Figure 4-1A) the primary water flows downward from the reactor pool through the reactor core and then flows into the core plenum. The outlet plenum is bolted above the 10 inch (25.4 cm) outlet pipe at the bottom center of the pool liner. It serves as a transition from the square reactor core grid to the round outlet pipe. It also serves as the support for the reactor core, and during forced convection, directs the flow of water from the reactor core to the coolant outlet pipe. The plenum is 3 feet (91 cm) high, 22 inches (55.9 cm) by 20.5 inches (52.1 cm) at the top, and 10 inches (25.4 cm) diameter at the bottom. A flapper valve, which is a 15.75 inch (40.0 cm) flat disc, on the side of the plenum is manually shut prior to initiation of forced flow and is held shut by the differential pressure created by the downward flow through the plenum. The flapper is counter-balanced to fall to a 30° open position upon loss of forced flow. The flapper valve may be manually operated by means of a reach rod from the reactor bridge area.

The primary coolant flows from the plenum by means of a 10 inch (25.4 cm) stainless steel pipe, through the pool liner and through a 10 inch (25.4 cm) manual isolation valve (P-1) in the valve pit. All of the stainless steel pipe that is embedded in concrete is bituminous coated and felt wrapped to prevent corrosion. The coolant (hot leg) passes through a tunnel to the N-16 delay tank located in the Primary Piping Vault (PPV). The PPV is below grade at the south side of the reactor building. The N-16 tank is constructed of stainless steel and has internal baffling to delay the primary coolant for approximately one minute at a 500 gpm (31.5 ℓs^{-1}) flow rate. The one minute delay allows the high energy nitrogen-16 gamma rays produce as the coolant flows through the core to decay to near background levels. To allow for system venting and draining, the tank has a 1 inch (2.5 cm) manual gate valve drain (P-12) and a 1 inch (2.5 cm) manual gate valve vent (P-11) located in the vault.

After the coolant leaves the PPV it re-enters the Reactor Building in the Mechanical Equipment Room (MER) where it passes through another 10 inch (25.4 cm) manual gate isolation valve (P-2) just before being reduced to a 6 inch (15.2 cm) pipe and entering the suction side of a single stage horizontal centrifugal pump. The primary pump is located in the Mechanical Equipment Room. The pump is constructed of stainless steel and provides flow at 500 gpm (31.5 ℓs^{-1}) with a discharge pressure of 25.0 psig (268.9 kPa).

Flowing at 500 gpm (31.5 ℓs^{-1}) the coolant passes through a 6 inch (15.2 cm) manual globe throttle valve (P-3) and into the inlet of the heat exchanger. The heat exchanger is a water to water shell and tube type heat exchanger. The primary water makes four passes through the stainless steel tubes and the secondary water makes a single pass in the carbon steel shell section. After being cooled in the heat exchanger, a small portion of the coolant is shunted to the primary demineralizer for purification.

The pipe diameter expands back to 10 inches (25.4 cm) and coolant continues through flow straightening tubes, to reduce turbulence, before entering the flow measuring equipment. The coolant once again exits the Reactor Building and travels a short distance before reentering the PPV. The coolant (cold Leg) then enters the tunnel parallel to the hot leg

travels through the tunnel through a 10 inch (25.4 cm) manual gate isolation valve (P-5). The coolant flows through a 90° elbow which directs the water away from the core where it mixes with the bulk of the coolant in the pool.

The temperature of the pool is a nominal 105°F (40.6°C) when operating at one megawatt except during the hottest days of the year, with the temperature rise across the core being 13.8°F (7.67°C). This results in a nominal pool outlet temperature of 118.8°F (48.2°C). The core inlet temperature (pool) is controlled by regulating the temperature of the secondary coolant.

In the natural convection mode of operation, the Primary Pump is secured and forced flow ceases. The cessation of flow through the pool outlet plenum results in a loss of the differential pressure across the flapper, and the flapper falls open due to the force of gravity. Water from the pool can now enter the outlet plenum through the open flapper valve and flow upward through the reactor core by thermal convection, thus cooling the reactor.

In addition to the main coolant loop, there are several auxiliary components. There is an overflow weir located near the top of the reactor pool. Pool water entering the weir flows by gravity to the radioactive sump in the Mechanical Equipment Room. The Primary Coolant System contains four drain lines (P-7, P-12, P-23, P-29, P-30), two vent lines (P-11, P-15), and two test connections (P-6, P-15). Each of the lines has a threaded pipe cap or a self-sealing quick disconnect to prevent accidental loss of primary coolant due to a valve operating error.

4.2.1.1 Primary Heat Exchanger

The Primary Heat Exchanger is a four pass shell U-Tube type heat exchanger. The heat exchanger has 234 tubes, with an outside diameter of 0.75 inches (19.05 millimeters) x 18 B.W.G. The primary side of the heat exchanger is constructed of 304 stainless steel and the secondary shell side of the heat exchanger is constructed of carbon steel.

A Heat exchanger pressure boundary breach is analyzed in Section 13 of the SAR.

4.2.2 Secondary System

The Secondary Coolant System consists of a pump similar in design to the primary pump, a one megawatt heat exchanger, a three-way diverting valve located in the Mechanical Equipment Room, and a 233 ton cooling tower located on the east side of Burlington Engineering Laboratories.

Cooled water is brought from the cooling tower, at 700 gpm (44.2 ℓs^{-1}), by the secondary pump which has a discharge pressure of 33 psig (328.9 kPa). The Secondary System pump is a horizontally mounted, single stage, centrifugal pump located in the Mechanical Equipment Room.

From the secondary pump the water flows through a 6 inch (15.2 cm) manual glob throttle valve (S-3) and to the inlet of the shell side of the heat exchanger. The coolant leaves the heat exchanger and flows through a 6 inch (15.2 cm) manual isolation valve (S-4) to the three-way pneumatic operated valve (S-5) where it is diverted to the cooling tower or back to the suction of the secondary pump. The Secondary Temperature Controller operates a pneumatic actuator on the three-way valve, thus controlling Secondary System temperature at the heat exchanger. A temperature sensor on the pump suction header measures secondary water temperature. This temperature is monitored on a temperature control panel mounted in the Control Room. When the water temperature is low, the valve is positioned to divert flow back to the pump suction, thus bypassing the cooling tower. As system temperature increases, an increasing portion of the flow is sent through the cooling tower until the system is fully loaded and all of the flow is directed to the cooling tower. With a decreasing thermal load the reverse would occur until virtually all flow bypasses the cooling tower. On loss of control air, the valve fails to the position which provides maximum flow to the cooling tower. Control air to this system is supplied by the Reactor Air compressor.

The coolant temperature entering the heat exchanger is a nominal 87°F (30.6°C). At full power and with a nominal secondary flow rate of 700 gpm (44.2 ℓs^{-1}), the temperature rise across the heat exchanger is a nominal 9.9°F (5.5°C).

The cooling tower is a standard counterflow design with manually controlled dampers to limit cooling in cold weather. The basin is equipped with a water level controller and a thermostatically regulated 5.5 kW heater to prevent freezing during the winter. The fans are mounted on a single shaft which is rotated by a 25 HP, 480 volt weatherproof motor. The blower motor receives power from Motor Control Center No. 2. The fans are electrically interlocked with the secondary pump so that they will not operate without flow in the secondary system and are also thermostatically controlled by a sensor located in the MER. The thermostat is set at a nominal 70°F (21.1°C) which limits the cycling of the cooling tower fans, especially during cold, dry weather.

The chemical control system maintains the desired chemical balance in the secondary water to minimize corrosion and the fouling of heat transfer surfaces in the heat exchanger. This is accomplished by adding treatment chemicals and simultaneously bleeding the system whenever the system resistivity falls below the preset value. The major components of the chemical control system are a chemical addition pump, which is interlocked with the secondary pump, a chemical holding tank, a flow-through conductivity probe, a resistivity controller, and a solenoid operated bleed valve (S-8). The chemical addition pump is actuated by the resistivity controller in the Control Room which also actuates the bleed solenoid valve at the cooling tower. The pump takes a suction on the chemical holding tank and discharges through a check valve (S-17) and a stop valve (S-18) into the secondary pump discharge pressure gauge connection. The chemical holding tank is a 50 gallon (189 ℓ) polyethylene drum located in the Mechanical Equipment Room. The flow type conductivity probe is installed near the secondary pump. A small amount of flow is diverted from the discharge side of the secondary pump, through the probe, and back to the suction

side of the pump. Isolation valves (S-15 and S-16) are installed on the inlet and outlet of the probe. The resistivity controller, which is mounted in the Auxiliary Equipment Rack in the Control Room, receives the signal from the conductivity probe and actuates the solenoid drain valve and chemical pump at a predetermined resistivity value. Water samples are taken on a routine bases to measure the anti-corrosion chemical content in the water.

All Secondary System pipes and valves are constructed of carbon steel except for the lines to and from the conductivity probe, which are copper, and the discharge line from the chemical pump, which is polyethylene. The piping external to the reactor building is insulated and has heater tapes installed between the pipe and the insulation to prevent freezing.

4.2.3 Purification System

The Purification System for the primary coolant is located in the MER and the Purification System pump takes a suction from the 6 inch (15.2 cm) Primary System piping between the heat exchanger and valve P-4.

The Purification System pump, providing flow at 20 gpm (1.26 ℓs^{-1}), is a close-coupled, single stage, stainless steel, centrifugal pump located in the Mechanical Equipment Room.

A 2 inch (5.1 cm) gate isolation valve, 0.5 inch (1.3 cm) influent sample connection and a wye-type strainer are installed on the pump suction line. The purpose of the wye-type strainer is to prevent relatively large foreign objects from damaging the pump impeller. The strainer will prevent passage of particles greater than 0.06 inch (1.52 mm) in size. The pump discharge goes through a 25-micron filter to the demineralizer which uses a mixed ion exchange resin bed. The demineralizer is a closed stainless steel cylinder, 20 inches (50.8 cm) in diameter and 48 inches (1.22 m) tall. In addition to the inlet and outlet connection, a vent on the top and a resin drain on the bottom are provided. All wetted surfaces are stainless steel. The demineralizer holds 8 cubic feet (22.6 cm^3) of non-regenerable nuclear grade resin. A retention element at the bottom of the column prevents resin beads from entering into the system. The water leaving the demineralizer flows through an isolation valve, a 25-micron effluent filter, past a conductivity cell, and through another isolation valve to a check valve (PD-15). From the check valve, the purified water flows past another conductivity cell and returns to the Primary System downstream of valve P-4 through a flow adjusting valve. A 0.5 inch (1.3 cm) effluent sample connection is provided between the check valve and the flow adjusting valve.

The effluent of the system is extremely high quality water with a pH of nearly 7.0 and a resistivity of greater than 10 Megohm·cm. Use of the purification system is an effective method to control corrosion. Samples of pool liner material have been suspended in the pool since startup to document corrosion of the liner. Measurements are taken periodically and have yielded no detectable weight loss. Operation of the purification system to maintain high quality water will enable the continued operation of the reactor during the nest twenty years without corrosion problems.

Since primary coolant passes through the reactor core, it is obvious that high levels of radioactivity and contamination can be concentrated in the demineralizer. Special Procedures are used when changing the resin to minimize exposure levels and the spent resin is treated as radioactive waste. If a fuel assembly were to start leaking, higher than normal radiation levels would be detected in the demineralizer resin or the stack gas monitoring system. A fuel pin clad failure is analyzed in Section 13 of the SAR.

4.2.4 Liquid Radioactive Drain System

The Liquid Radioactive Drain System is designed to receive liquid waste water from designated spaces in Burlington Engineering Laboratories which may be contaminated with radioactive material. From two sumps, one in the MER and the other in the PPV, the waste is pumped to the three waste holding tanks located in an underground concrete vault outside the Reactor Building. The tanks are denoted Waste Tanks No. 1 through 3 consecutively, with tank No. 1 closest to the Reactor Building. Each tank has a gamma scintillation detector and liquid level sensor mounted inside. The tanks are provided with both local manual valves and remotely controlled pneumatic valves to either retain the waste or release it in a controlled manner to the sanitary sewer system.

The sump consists of a rectangular concrete pit in the floor of the Mechanical Equipment Room. The dimensions of the sump are 6 feet (1.8 m) deep by 4 feet (1.2 m) long by 1.5 feet (45.7 cm) wide. It has a special coating on the walls to aid any necessary decontamination operations. The sump pump is a vertically mounted 55 gpm (3.5 ℓs^{-1}) pump with adjustable cut-on and cut-off points actuated by a float switch.

The three waste holding tanks are identical in design. They are right circular cylinder, 904 gallon (3413 ℓ) fiberglass tanks equipped with removable detector wells which extend to a depth corresponding to the tank centerline. Each tank has a liquid level sensor with a range from 0 - 100% full with a readout meter on the Liquid Waste Control Panel in Room B103. An adjustable alarm contact is provided on each percent full meter which will actuate an audible alarm if the level in a given waste tank reaches that value. Pneumatically operated valves are located on the inlet and outlet side of each tank and are controlled from the Liquid Waste Control Panel. In addition, manually operated valves are provided to completely isolate the holding tank system or to by pass the pneumatically operated valves when necessary. These valves can also be aligned to allow transfer of waste from one tank to another through an overflow line.

4.2.5 Instrumentation

Instrumentation for the Reactor Coolant System and related support systems is shown in Figures 4-1A through 4-1F. The reactor pool is equipped with high and low water level alarms and a low water level reactor SCRAM. Abnormal Pool Level alarms annunciate in the Control Room when the water level varies from the zero reference level by $\pm$ 6 inches ($\pm$ 15.2 cm). If the water level drops to -36 inches (-91.4 cm) and the reactor is operating, a SCRAM signal will shut down the reactor and annunciate in the Control Room. The pool

level gauge is located on the console. The loss of pool water is discussed further in Section 13.

The resistivity of the primary coolant is measured by a sensor suspended in the reactor pool and a second one in the outlet of the primary demineralizer. These values are displayed on a resistivity bridge mounted on the left side of the console. The demineralizer decontamination factor, which is a measure of the resin's effectiveness, is checked monthly by Health Physics personnel.

The temperature of the primary coolant is measured at five locations; pool, hot leg, cold leg, and the inlet and outlet of the primary side of the heat exchanger. Each sensor is a resistance temperature detector (RTD) connected to a temperature transmitter which sends a 4-20 mA signal to the temperature recorder mounted on the right side of the console. All of the temperatures (except T-5 and T-6) will annunciate in the control room when they reach a nominal 116°F (46.7°C).

There is an additional temperature monitor suspended in the reactor pool that annunciates in the Control Room when the coolant reaches a nominal 114°F (45.6°C).

The temperature of the secondary coolant is measured at the inlet and the outlet of the heat exchanger with the temperature transmitters identical to those employed for the primary coolant instrumentation. There are also sensors in the Secondary Coolant System which position the three-way diverting valve. The temperature control cabinet is mounted in the west wall of the Control room beneath the large window.

The primary coolant flow rate is measured using an orifice plate mounted in a straight section of the 10 inch (25.4 cm) pipe near the south wall of the MER and a pneumatic differential pressure transmitter mounted nearby. Just before the orifice plate is a section of pipe that contains flow straightening vanes which helps to improve the accuracy of measurement. The flow rate, in gallons per minute, is displayed by a gauge on the south wall of the MER and on the right side of the console. If the reactor is operating above 150 kW, a SCRAM will be annunciated in the Control Room when the primary flow drops below 475 gpm (30 ℓs^{-1}).

The flow rate of the Purification System is 20 gpm (1.26 ℓs^{-1}) and is locally displayed on the demineralizer. There are additional gauges to show the pressure drop across the demineralizer resin and the filter cartridges.

Pressure gauges are mounted on the discharge side of each of the pumps. Indicator lights to show the status of the three pumps are mounted on the local start/stop switch, the Motor Control Center, and the right side of the control console.

The sump located in the MER has a high sump water level sensor which is connected to the annunciator panel in the Reactor Control Console. This alarm is actuated by a float switch which is set above the pump cut-on point. If the pump fails to start, the water level in the

sump will rise to the alert setpoint and cause the annunciator panel to sound along with the illumination of a High Sump Level Alarm warning light.

4.2.6 Materials of Construction

All materials in the Primary System, exclusive of the aluminum pool liner, are type 304 or 304L stainless steel. Exposed piping is Schedule 10S while Schedule 40S is used for the embedded piping and that in the PPV. All of the Secondary System piping is Schedule 40 carbon steel.

All of the embedded primary piping and that in the PPV was full penetration welded, radiographed and dye penetrant tested. Piping connections to equipment including the delay tank are flanged for maintenance purposes.

The reactor biological shield is barytes concrete from its base to 10 feet (3 m) from the top, with the top section being standard concrete. The shield is poured around an aluminum liner which insures water tightness and provides a surface that is easily decontaminated. The surface of the liner next to the concrete is coated to prevent concrete induced corrosion. The welds were radiographed and the liner was filled with water for a leak test prior to pouring the biological shield. Galvanic action between the aluminum liner and the stainless steel piping is prevented by the use of BI-PRO couplings.

The three waste holding tanks are identical in design. The tanks are fiberglass and the sump discharge line to the holding tanks is constructed of Fuseal or similar chemical resistant polypropylene pipe.

4.3 System Design Evaluation

The coolant system is designed to remove one megawatt of heat from the PULSTAR Reactor and maintain an inlet temperature of 105°F (40.6°C) with a 500 gpm (31.5 ℓs^{-1}) flow rate on the primary side. The secondary coolant system is designed to keep the primary system inlet temperature at 105°F (40.6°C) on all but the hottest days of the year.

The purification system is designed to maintain high resistivity and purity of the primary coolant by continuously filtering and demineralizing a 20 gpm (1.26 ℓs^{-1}) bypass flow of the coolant.

The reactor has no requirement for auxiliary power for the coolant pumps or special coolant sources. The reactor is capable of natural convection cooling to several hundred kilowatts without any damage. Analysis of emergency cooling requirements is covered in Section 13.

4.4 Tests and Inspections

The following tests and inspections are performed as required by the Technical Specifications[4-1], and internally generated PULSTAR surveillance files:

1. The resistivity of the primary coolant leaving the demineralizer is checked to determine if the resin is spent.

2. The pressure drop across the purification system filters are checked for loading.

3. All alarms or alerts associated with the coolant system are checked to determine operability.

4. Samples of the reactor pool water are checked periodically for radioactivity, sodium and chlorine content, and pH. Samples also are taken from the inlet and the outlet of the primary Purification System to determine the decontamination factor (DF). Any very small leaks in a fuel pin would be identified from these samples.

5. All of the Primary System mechanical and measuring equipment is periodically checked or calibrated.

6. All 10-inch piping bounded by P-1, P-5, P-2 and P-4 is hydrostatically pressure tested.

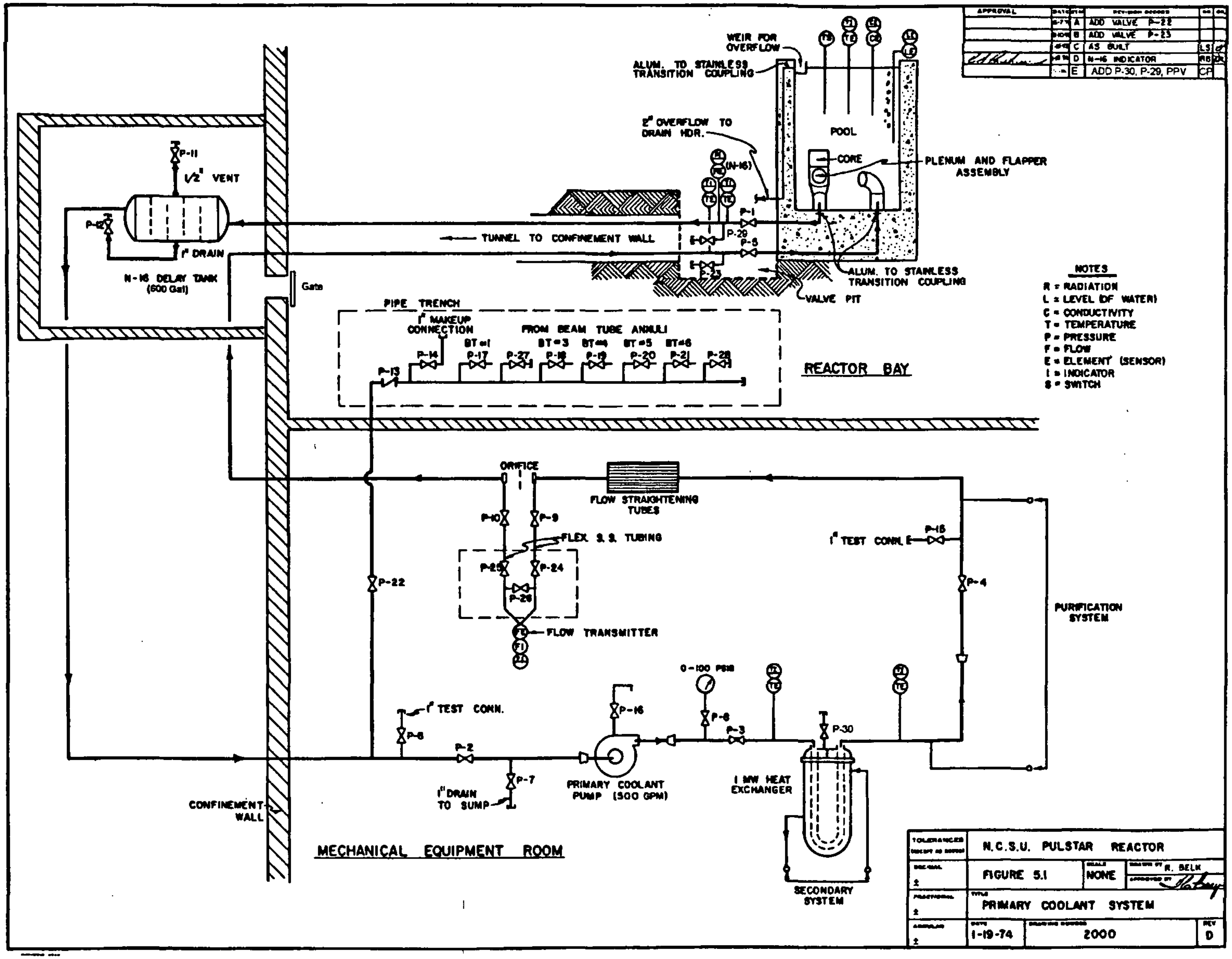
7. A leak surveillance procedure has been established which provides for the earliest possible detection of any significant leakage from the Primary System.

8. The radioactive liquid drain pipe from the sump pump to the waste tank vault is tested hydrostatically to insure system integrity.

9. Pool liner test coupons are periodically weighed to measure material loss due to corrosion.

4.5 References

4-1 "Technical Specifications for the North Carolina State University PULSTAR Reactor", Department of Nuclear Engineering, Nuclear Reactor Program.



WEIR FOR OVERFLOW
ALUM. TO STAINLESS TRANSITION COUPLING
POOL
CORE
PLENUM AND FLAPPER ASSEMBLY
2" OVERFLOW TO DRAIN HDR.
(N-16)
P-1
P-29
P-5
P-23
TUNNEL TO CONFINEMENT WALL
ALUM. TO STAINLESS TRANSITION COUPLING
VALVE PIT
P-11
1/2" VENT
P-12
1" DRAIN
N-16 DELAY TANK (600 Gal)
Gate
PIPE TRENCH
1" MAKEUP CONNECTION
FROM BEAM TUBE ANNULI
BT=1
BT=3
BT=4
BT=5
BT=6
P-13
P-14
P-17
P-27
P-18
P-19
P-20
P-21
P-28
REACTOR BAY
NOTES
R = RADIATION
L = LEVEL OF WATER
C = CONDUCTIVITY
T = TEMPERATURE
P = PRESSURE
F = FLOW
E = ELEMENT (SENSOR)
I = INDICATOR
S = SWITCH
ORIFICE
FLOW STRAIGHTENING TUBES
P-10
P-9
FLEX. S.S. TUBING
P-25
P-24
P-26
FLOW TRANSMITTER
P-22
1" TEST CONN.
P-15
P-4
PURIFICATION SYSTEM
0-100 PSIG
1" TEST CONN.
P-6
P-16
P-8
P-3
P-30
P-2
P-7
1" DRAIN TO SUMP
PRIMARY COOLANT PUMP (500 GPM)
1 MW HEAT EXCHANGER
SECONDARY SYSTEM
CONFINEMENT WALL
MECHANICAL EQUIPMENT ROOM
APPROVAL
A ADD VALVE P-22
B ADD VALVE P-23
C AS BUILT
D N-16 INDICATOR
E ADD P-30, P-29, PPV
N.C.S.U. PULSTAR REACTOR
FIGURE 5.1
NONE
R. BELK
PRIMARY COOLANT SYSTEM
1-19-74
2000
D

Figure 4-1A

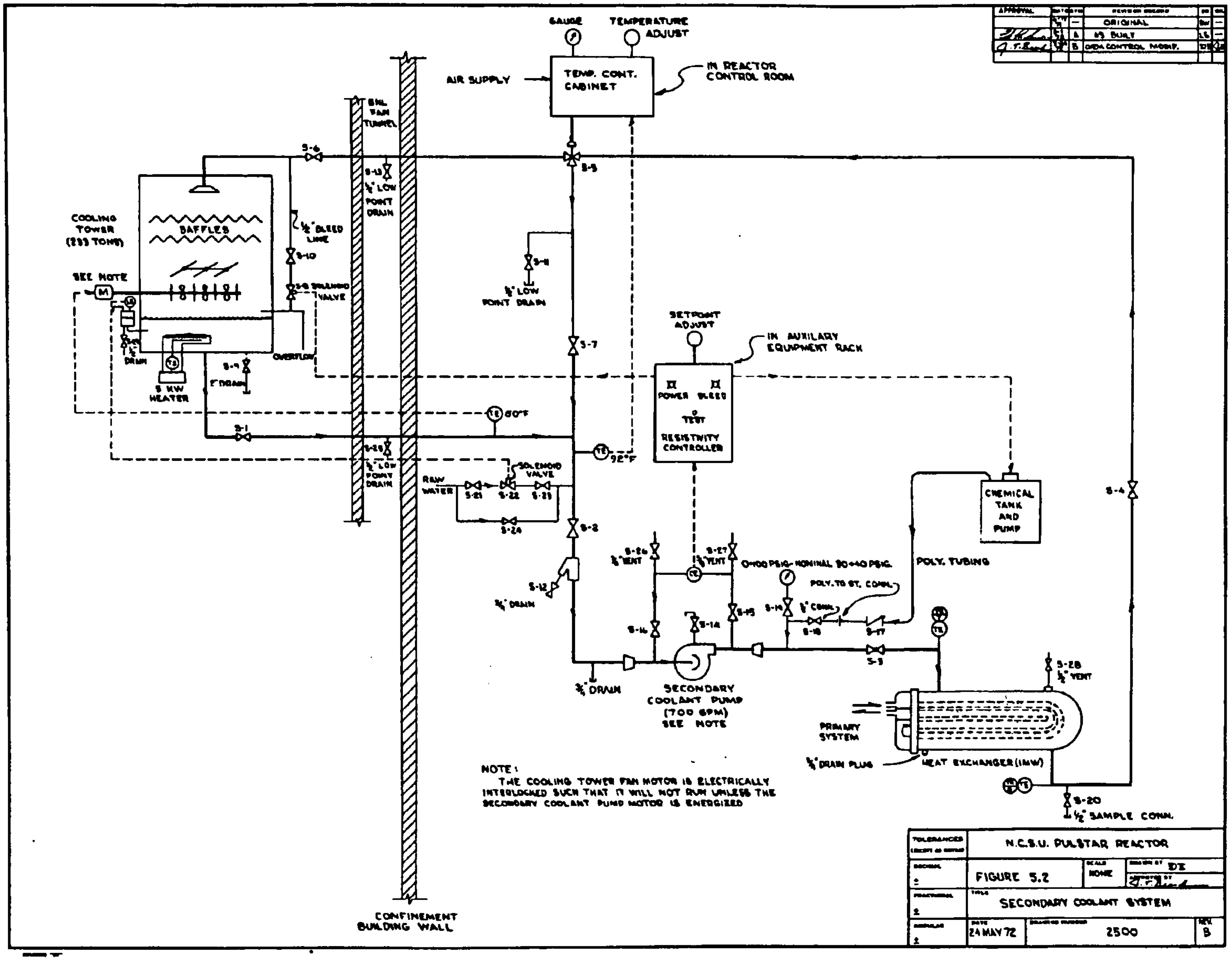
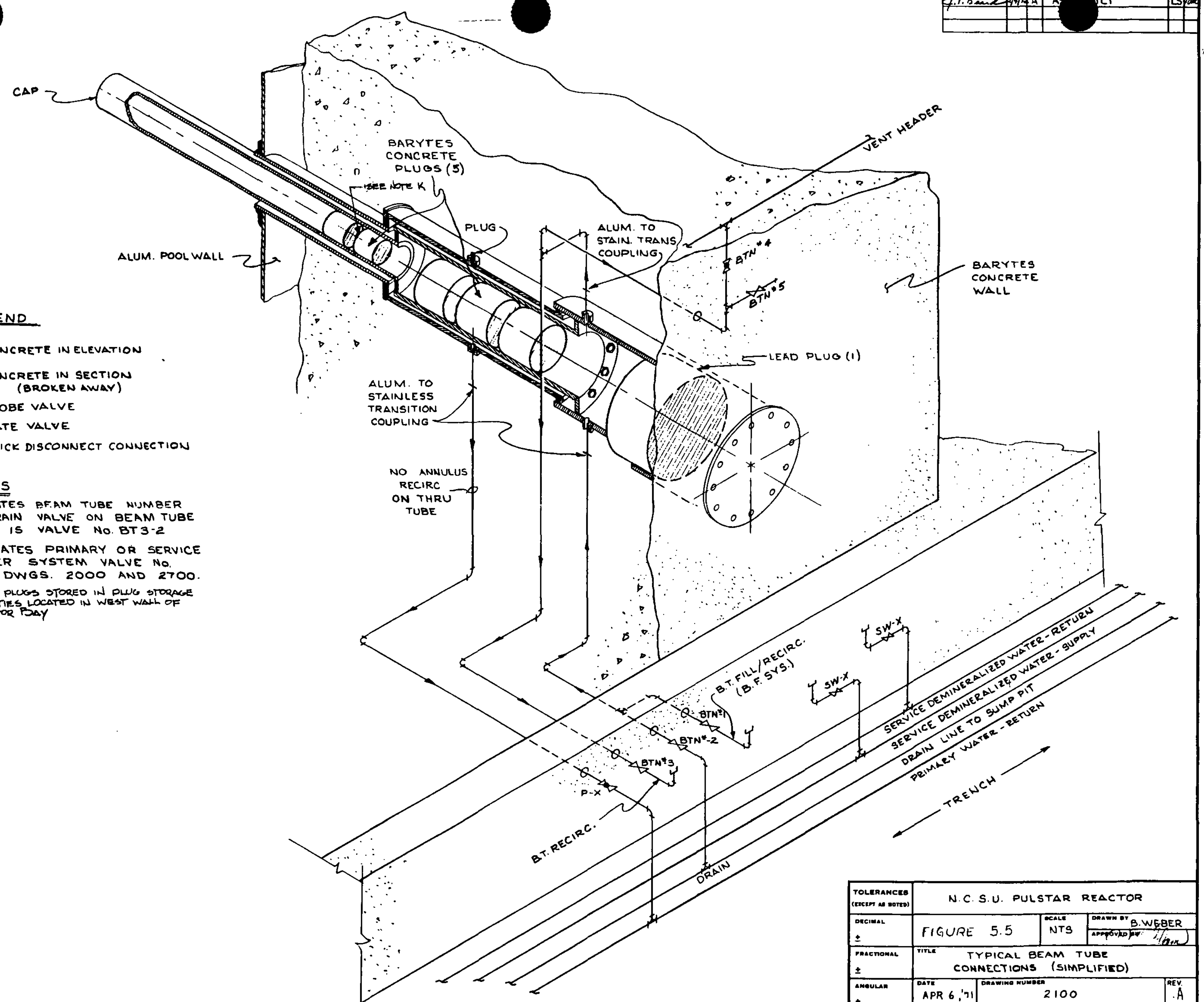

GAUGE
TEMPERATURE ADJUST
AIR SUPPLY
TEMP. CONT. CABINET
IN REACTOR CONTROL ROOM
COOLING TOWER (233 TONS)
BAFFLES
SEE NOTE
5 KW HEATER
OVERFLOW
½" BLEED LINE
SOLENOID VALVE
½" LOW POINT DRAIN
SETPOINT ADJUST
IN AUXILARY EQUIPMENT RACK
POWER BLEED
TEST
RESISTIVITY CONTROLLER
60°F
92°F
RAW WATER
CHEMICAL TANK AND PUMP
POLY. TUBING
0-100 PSIG- NOMINAL 30±10 PSIG.
POLY. TO ST. CONN.
¾" DRAIN
SECONDARY COOLANT PUMP (700 GPM) SEE NOTE
PRIMARY SYSTEM
¾" DRAIN PLUG
HEAT EXCHANGER (1MW)
½" SAMPLE CONN.
NOTE:
THE COOLING TOWER FAN MOTOR IS ELECTRICALLY INTERLOCKED SUCH THAT IT WILL NOT RUN UNLESS THE SECONDARY COOLANT PUMP MOTOR IS ENERGIZED
CONFINEMENT BUILDING WALL
ORIGINAL
AS BUILT
N.C.S.U. PULSTAR REACTOR
FIGURE 5.2
NONE
SECONDARY COOLANT SYSTEM
24 MAY 72
2500
B

Figure 4-1B

SAR FIG. 4-1C OPERATIONS MANUAL FIG. 5.5

APPROVAL	DATE	SYM	REVISION RECORD	DR.	CK.
J.T. Saml	7/4/72	A	A [illegible] ILT	LS	[illegible]



LEGEND

- CONCRETE IN ELEVATION
- CONCRETE IN SECTION (BROKEN AWAY)
- GLOBE VALVE
- GATE VALVE
- QUICK DISCONNECT CONNECTION

NOTES

N* INDICATES BEAM TUBE NUMBER eg. DRAIN VALVE ON BEAM TUBE No. 3 IS VALVE No. BT3-2

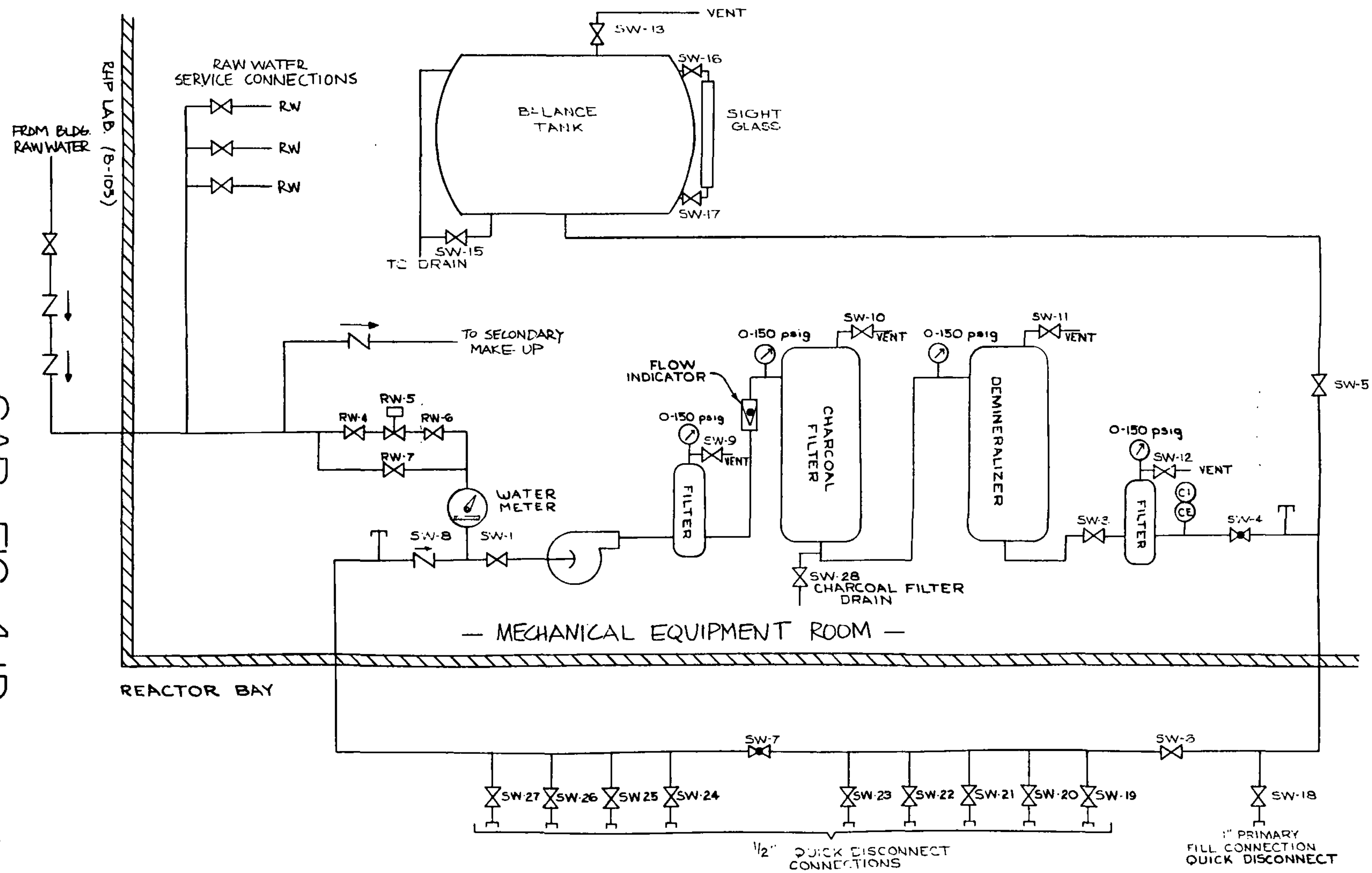
X INDICATES PRIMARY OR SERVICE WATER SYSTEM VALVE No. SEE DWGS. 2000 AND 2700.

K SMALL PLUGS STORED IN PLUG STORAGE FACILITIES LOCATED IN WEST WALL OF REACTOR BAY

TOLERANCES (EXCEPT AS NOTED)	N.C.S.U. PULSTAR REACTOR		
DECIMAL ±	FIGURE 5.5	SCALE NTS	DRAWN BY B. WEBER APPROVED BY [illegible]
FRACTIONAL ±	TITLE TYPICAL BEAM TUBE CONNECTIONS (SIMPLIFIED)		
ANGULAR ±	DATE APR 6, '71	DRAWING NUMBER 2100	REV. A

APPROVAL	DATE	SYM	REVISION RECORD	DR.	CK.
		A	BUILT	LS	
		B	TER METER	AP	
		C	D.O. 79-4	EB	

SAR FIG. 4-1D
OPERATIONS MANUAL FIG. 5.4



WATER METER SPECS
AMERICAN WATER METER
HERSHEY AMETROL DIVISION
MODEL 430, SER. # D121595

TOLERANCES (EXCEPT AS NOTED)	N.C.S.U. PULSTAR REACTOR		
DECIMAL ±	FIGURE 5.4	SCALE NTS	DRAWN BY L SWANSON
FRACTIONAL ±	TITLE SERVICE WATER AND REACTOR BAY RAW WATER SYSTEM		
ANGULAR ±	DATE SEPT 17, 71	DRAWING NUMBER 2700	REV B

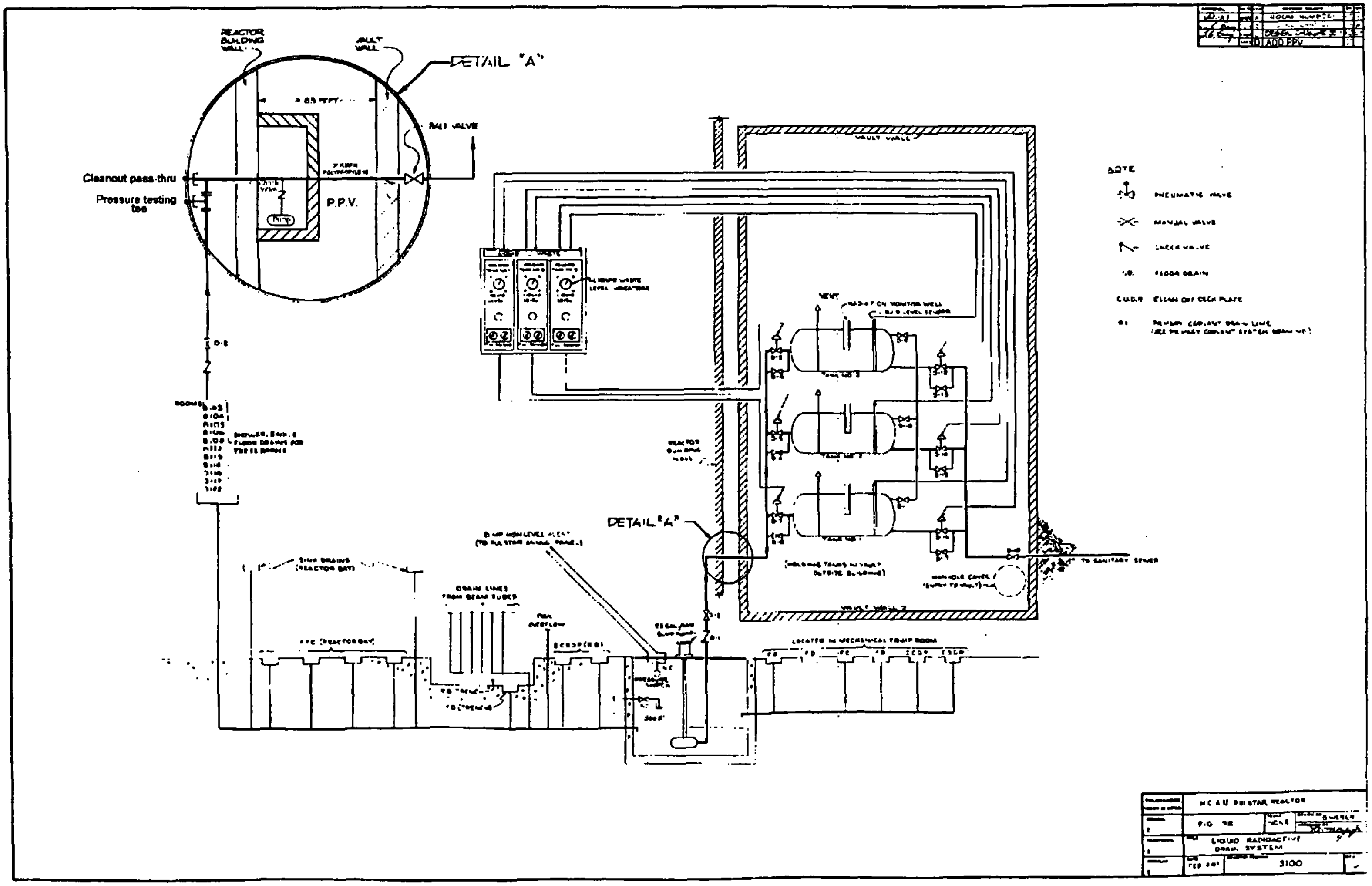
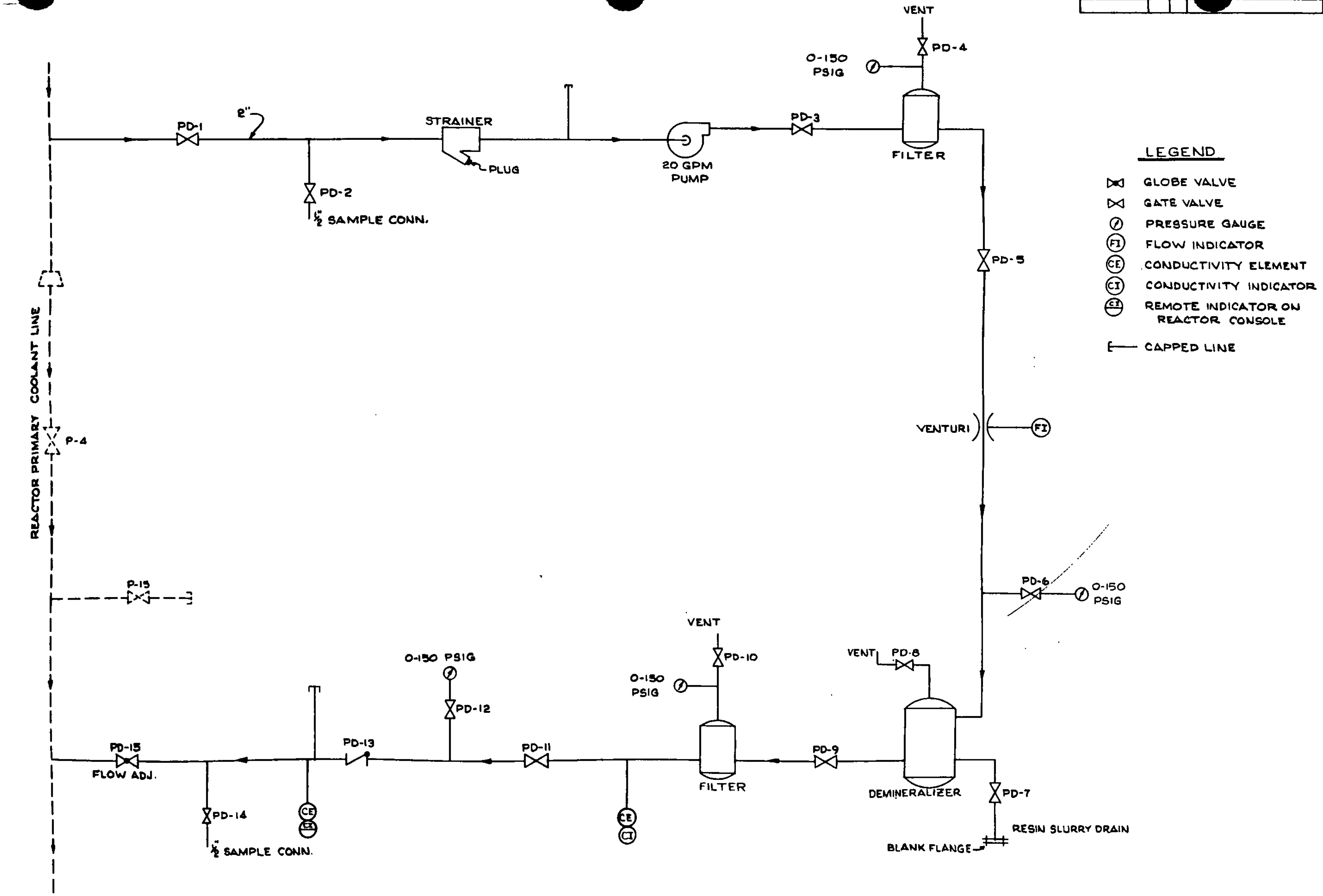

REACTOR BUILDING WALL
VAULT WALL
DETAIL "A"
BALL VALVE
Cleanout pass-thru
Pressure testing tee
P.P.V.
VAULT WALL
VENT
DETAIL "A"
TO SANITARY SEWER
ADD PPV
N C S U PULSTAR REACTOR
LIQUID RADIOACTIVE DRAIN SYSTEM
3100

Figure 4-1E

SAR FIG. 4-1F

OPS MAN. FIG. 5.3



TOLERANCES (EXCEPT AS NOTED)	N.C.S.U. PULSTAR REACTOR		
DECIMAL ±	FIGURE 5.3	SCALE NTS	DRAWN BY B. WEBER
FRACTIONAL ±	TITLE PURIFICATION SYSTEM		
ANGULAR ±	DATE JUN 23, '71	DRAWING NUMBER 2200	REV.

APPROVAL	DATE	SYM	REVISION RECORD	DR.	CK.

5 CONFINEMENT SYSTEM

5.1 Confinement System Structure

5.1.1 Design Bases

The Confinement System design is based on the requirements dictated by the postulated design bases accident and its associated releases as discussed in Section 13. The results of Section 13 analyses confirm that the release of fission products is most improbable, while the amounts involved, if released, would be well within the capabilities of the confinement system. Further, the self-regulating properties of the PULSTAR Reactor, along with the intrinsic safety features of the low enriched fuel design, limit the source and amount of materials which might be released into the confinement structure. By housing the NCSU PULSTAR Reactor in a relatively airtight concrete confinement structure called the Reactor Building, the use of confinement fans with high efficiency filters, charcoal absorbers, and automatically closing dampers, the release of any radioactive material will be controlled within the limits established by appropriate federal and state regulations consistent with part 10 CFR 20.[5-1]

5.1.2 Confinement Structure Design

The Reactor Building for the NCSU PULSTAR Reactor includes the Reactor Bay, a Mechanical Equipment Room (MER), the primary piping vault, and the Control Room as shown in Figures 1-3, 1-4, 1-5, 1-6 and 1-7. The lower east entrance, the loading dock entrance on the east side, the northwest entrance, and the Control Room entrance from the second floor are the confinement doors. , . The Primary Piping Vault (PPV) hatch is also a confinement boundary. l. The basement laboratory area on the north side of the Reactor Bay is a restricted area but is not part of the Reactor Building. These walls are : with a brick veneer on the exterior and a painted surface on the interior. All penetrations of the Reactor Building are sealed to minimize leakage and the possibility of releasing contamination to the environment. Penetrations are sealed in the following manner:

1. All pipes and ducts passing through the walls are sealed to the wall.

2. Electrical conduits are sealed to the walls and the wires are sealed to the inside of the conduit.

3. Major process pipes and ducts have isolation valves or dampers.

4. All doors leading into the Reactor Building are gasketed. They are also self-closing and self-latching.

5. The PPV hatch is gasketed with an internal security barrier.

6. All spare pipes and conduits are plugged or capped.

The Reactor Building is maintained at a negative pressure with respect to the outside (atmosphere) and a slightly smaller negative pressure with respect to the lower nuclear laboratories. The differential pressure (dp) between the Reactor Building and atmosphere shall be at least 0.2 inches of water (49.0 Pa) when the main H&V system is running or at least 0.1 inches of water (24.5 Pa) with either confinement fan running.

Doors may be opened by authorized personnel for less than five minutes for personnel and equipment transport provided audible and visual indication is available for the reactor operator to verify door status. If differential pressure is lost for greater than five minutes, an alarm in the control room will inform the reactor operator. Reactor operation may continue after a loss of dp (with main HVAC operating) for up to thirty minutes, while the loss of dp is investigated and corrected.

5.2 Confinement Initiation

5.2.1 Design Bases

Reactor Building confinement will automatically be initiated as a result of radiation levels in the effluent of the building ventilation or of radiation fields within the confinement area exceeding preset levels. These preset values are based on regulatory[5-2] or emergency[5-3] requirements, and can be found in the latest revision of the PULSTAR Emergency Plan[5-3]. Confinement can also be initiated by a manual switch or by the loss of power to the Radiation Alarm Panel.

The location of radiation detectors and isolation dampers are such that any release of radioactive material as postulated in Section 13 will be detected early enough to ensure that automatic initiation of confinement will mitigate any uncontrolled releases.

5.2.2 System Design

Confinement can be initiated by any of the following:

1. Manual confinement switch in the Control Room
2. Manual evacuation switch in the Control Room
3. Manual evacuation switch in the basement nuclear laboratory hallway
4. Control Room area monitor
5. Over-the-Pool area monitor
6. West Wall area monitor
7. Stack Gas monitor
8. Stack Particulate monitor
9. Auxiliary GM monitor
10. Loss of power to the Radiation Alarm Panel

Area monitors are located in the Control Room, over the pool, the west Reactor Bay area, and the Primary Demineralizer. The particulate and stack gas monitors will detect radiation levels in the air being exhausted from the Reactor Building through the 100 foot (30.5 m) high air stack. In order to get a representative sample, the air is withdrawn isokinetically at an approximate 10 cfm (0.005 m^3/sec) rate, 30 feet (9.1 m) downstream from the Exhaust Fan. The Auxiliary GM monitor will give an indication of the gross radiation levels in the air stream.

The radiation levels detected by the area and stack monitors will be displayed on indicators and recorded on a multi-point recorder located in the Control Room. Confinement is automatically initiated as part of the evacuation sequence when any of the monitors No. 4 through No. 9 listed above reaches the alarm setpoint. Alerts and alarms will annunciate in the Control Room.

Each of the monitors confinement signal may be bypassed with a separate key on the Radiation Alarm Panel. When a circuit is bypassed, a "Bypass" indication for that circuit will be energized on the radiation control panel. To ensure proper use of the bypass, authorization by the Senior Reactor Operator (SRO) is required. Lights are provided for key circuits in the Radiation Alarm Panel logic to indicate that each relay is in its normal state. The loss of this indication indicates either that a component failure has occurred, or an abnormal condition exists.

To prevent unnecessary initiation of the evacuation and confinement systems. Specific radiation monitoring channels evacuation signal may be bypassed for the following times:

1.	Process Monitors	Less than one minute immediately after starting the pneumatic blower.
2.	Over-the-Pool Monitor	Less than two minutes during the return of a pneumatic rabbit capsule.
3.	Over-the-Pool Monitor	Less than five minutes during removal of experiments from the reactor pool.

Upon initiation of confinement or evacuation, the Supply Fan and the Exhaust Fan shut down and their isolation dampers close. The Main dampers will indicate closed when the damper has gone full travel. This essentially closes all free paths of air into the Reactor Building. At the same time, Confinement Fan No. 1 will start. After a nominal 55 seconds, should Fan No. 1 fail to start, Confinement Fan No. 2 will start. While either fan is running, air is purged from the Reactor Building at a 600 cfm (0.3 m^3/sec) rate passing through all of the normal filters, through a 99.97% (removal efficiency) High Efficiency Particulate Absorber (HEPA) filter, and a charcoal absorber.

The isolation dampers are electrically activated, air operated devices that can also be manually operated if necessary. All of the isolation dampers fail closed on loss of electrical

power or air pressure. There is an electrically driven damper motor on each of the confinement fans that is powered from the same circuit as the fan motor. In the event of loss of commercial power, the confinement fan and damper motor may be powered by the Auxiliary Generator.

Fan status indicators, damper position indicators, and differential pressure readings (Magnehelic Gauges) provide the reactor operator with the necessary information to determine the status of the Confinement System. Fan status and damper status lights are provided in the control room. The confinement fan dampers are designed to be normally closed and will not indicate open until the damper has gone full travel.

Radioactive material generated by and potentially released by routine operation, or during accidents contain gamma emitting radionuclides. Therefore during maintenance to any required radiation monitoring channel, one of the installed channels may be replaced for up to ninety days, with a gamma-sensitive instrument which has its own alarm, or is observable by the reactor operator or reactor operator assistant. This time limit was chosen in order to allow sufficient time for procurement and testing of specialized equipment. In order to maintain a permanent record of radiation levels, during maintenance to the Radiation Rack Recorder for up to ninety days, all monitoring channels which cause evacuation will be recorded manually at a nominal interval of 30 minutes while the reactor is not shutdown.

5.3 Ventilation System

5.3.1 Design Bases

Several potential sources of radioactive gas and particulate releases exist at the PULSTAR Reactor. These vary from the production of argon-41 gas in the beam tubes or similar facilities to a failed experiment or a ruptured fuel pin. The handling of these potentially radioactive effluent during both normal and confinement conditions requires an adequate ventilation system capable of minimizing uncontrolled releases to the environment and providing the basic requirement of adequate ventilation for personnel and equipment. The size of the ventilation system is based on the magnitude of the release of potential sources and the functional requirements for operation are discussed in Section 13.

5.3.2 System Design

The ventilation system is shown schematically in Figure 5-1. Outside air for the Reactor Building is supplied at 9,650 cfm (4.6 m^3/sec) through the intake cubicle located between the two wings of Burlington Engineering Laboratories. The cubicle is protected by locked grating and the air must pass through a screen, a louver, and a manual damper before entering the air handling equipment in the MER. The air is filtered, heated when necessary, and distributed to the MER and the Reactor Bay. The Control Room receives its air through a separate duct branching from the main duct in the MER. It is separately filtered and conditioned to maintain personnel comfort and electronic equipment stability.

Air from the Control Room is discharged into the Reactor Bay through louvers in the door. Air from the MER and the Reactor Bay is drawn through a pre-filter and the main filters, monitored for radioactivity, and then discharged to the atmosphere through the 100 foot (30.5 m) stack. There is a separate exhaust duct for the beam tubes and the thermal column utilizing a booster fan and an absolute filter. This air is discharged into the exhaust plenum prior to the pre-filter. Two other sources of air entering the exhaust plenum are the bay hood and the pneumatic transfer system. These also have their own booster fans which are operated as required.

The MER contains three exhaust fans, each with a filter train. Normally, all the air discharged from the Reactor Building passes through the exhaust plenum containing a pre-filter (Roll-Kleen) which is 83% efficient by NBS (Cottrell Precipitate) type testing, a main filter with an average efficiency of 85% by the NBS (Atmospheric Dust) type testing, the exhaust fan, and up the stack at a 10,050 cfm (4.7 m^3/sec) rate. If the confinement mode is initiated, the air being discharged is diverted from the main exhaust fan, which is shut down, to one of the confinement fan trains. The air is now discharged at a 600 cfm (0.3 m^3/sec) rate through a 99.97% HEPA filter and a charcoal absorber. The two confinement fans are interlocked so only one can be running at a time.

There are actually two exhaust stacks. The one that is visible from the outside, discharges air from the Burlington Engineering Laboratories south wing. The Reactor Building's stack is located concentrically inside the original stack to within 10 feet (3.0 m) of the top, which prevents back-flow, and the flow from the south wing is used to dilute the air exhausted from the Reactor Building.

5.4 System Design Evaluation

The confinement system is designed to function automatically. It contains manual backups to insure confinement operability. Confinement integrity is accomplished by sealing all Reactor Building penetrations and using industry proven components. An electric generator fueled with natural gas provides an auxiliary source of power when manually started to perform an orderly shutdown of the reactor and to operate either one of the confinement fans.

The location of sensors and dampers insures early detection and maximum isolation between the Reactor Building and the environment. Redundant relays and other design features are incorporated in the logic and actuation circuits in the confinement system. The overall system is designed to operate fail-safe.

5.5 Tests and Inspections

Confinement system tests and inspections are specified in the Technical Specifications [5-4], and in the internally generated PULSTAR surveillance files.

5.6 References

5-1 United States Nuclear Regulatory Commission RULES and REGULATIONS Title 10, Chapter 1, Code of Federal Regulations, Part 20, "STANDARDS FOR PROTECTION AGAINST RADIATION", September 30, 1994

5-2 "Analysis/Calculation For Radiation Monitor Set Point Basis and Calculation" (Internal document NRP 94-001), March 21, 1994

5-3 "PULSTAR Emergency Plan" North Carolina State University Department of Nuclear Engineering, Nuclear Reactor Program. Revision 2, March 36, 1993

5-4 "Technical Specifications for the North Carolina State University PULSTAR Reactor", Department of Nuclear Engineering, Nuclear Reactor Program.

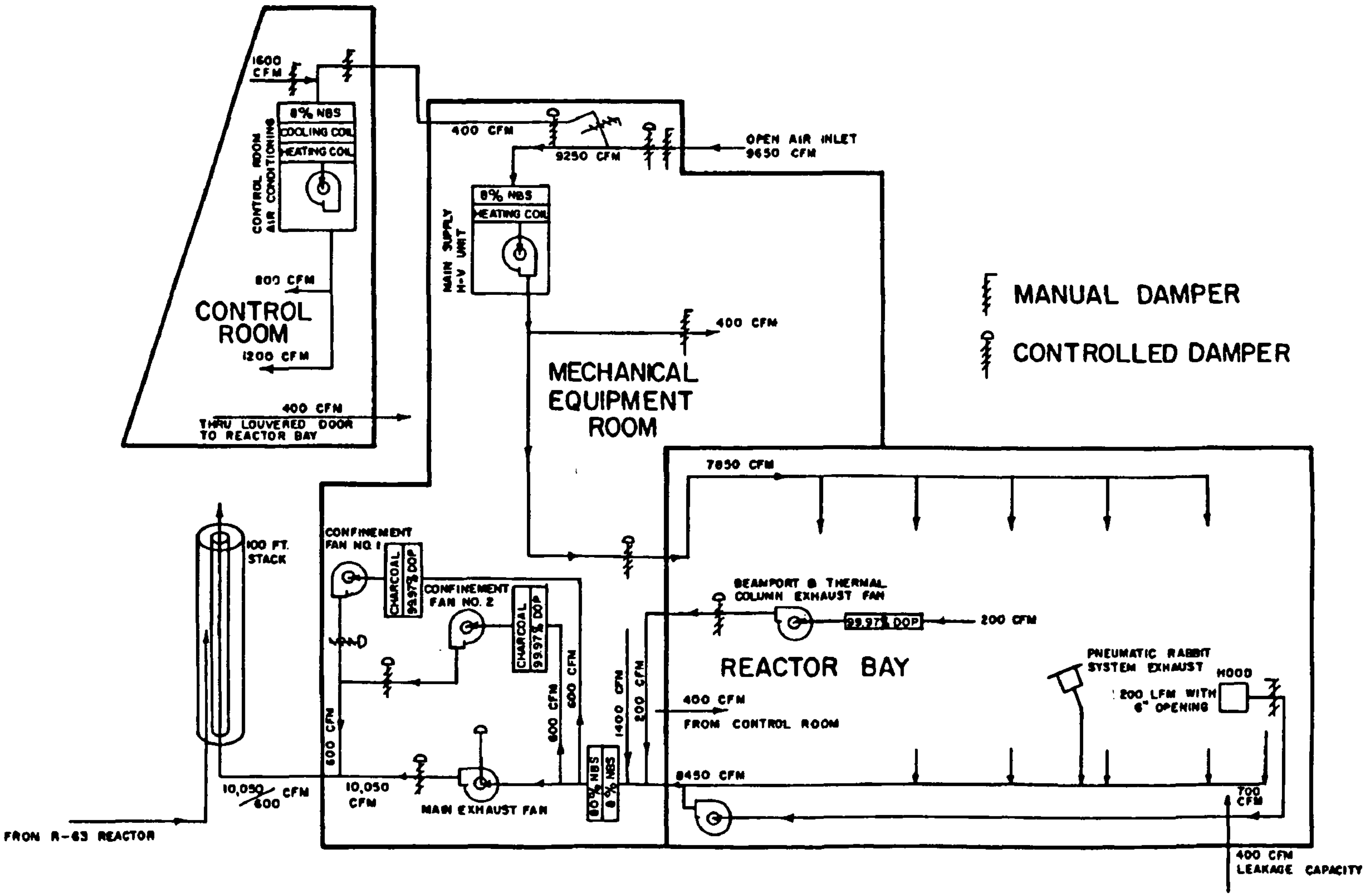

1600 CFM
8% NBS
COOLING COIL
HEATING COIL
CONTROL ROOM AIR CONDITIONING
800 CFM
CONTROL ROOM
1200 CFM
400 CFM
THRU LOUVERED DOOR TO REACTOR BAY
400 CFM
9250 CFM
OPEN AIR INLET 9650 CFM
8% NBS
HEATING COIL
MAIN SUPPLY H-V UNIT
400 CFM
MECHANICAL EQUIPMENT ROOM
MANUAL DAMPER
CONTROLLED DAMPER
7850 CFM
100 FT. STACK
CONFINEMENT FAN NO. 1
CHARCOAL
99.97% DOP
CONFINEMENT FAN NO. 2
CHARCOAL
99.97% DOP
BEAMPORT & THERMAL COLUMN EXHAUST FAN
99.97% DOP
200 CFM
REACTOR BAY
PNEUMATIC RABBIT SYSTEM EXHAUST
HOOD
200 LFM WITH 6" OPENING
400 CFM
FROM CONTROL ROOM
600 CFM
600 CFM
600 CFM
1400 CFM
200 CFM
80% NBS
8% NBS
8450 CFM
700 CFM
10,050/600 CFM
10,050 CFM
MAIN EXHAUST FAN
FROM R-63 REACTOR
400 CFM LEAKAGE CAPACITY

Figure 5-1

6 ENGINEERED SAFETY FEATURES

6.1 Design Bases

Several engineered safety features (ESFs) have been incorporated into the design of the PULSTAR Reactor Facility. The primary purpose of these ESFs by equipment design is to ensure that projected radiological exposures from accidents are kept below the regulatory limits. The major safety features which have been incorporated into the facility design have resulted from the experience gained during the operation of pool type research reactors. The design bases for the following ESFs are that safety actions must be automatic and/or passive and require no reactor operator action to be initiated.

6.1.1 Natural Circulation Cooling

During normal operation of the reactor at power levels above 100 kW, core cooling is accomplished with a 500 gpm (31.5 ℓs^{-1}) primary coolant flow rate. In the event that the primary flow is interrupted by loss of the pump or other causes, a flapper valve on the plenum, which is located directly under the grid plate, will open and provide a path for natural convection cooling to be established within the reactor pool. The flapper valve is held in a closed position by the differential pressure created by the coolant flow through the core and the pool static head at the plenum level. This is the normal operating position during forced flow cooling.

Tests have been performed which indicate the PULSTAR core could be operated in a natural convection cooling mode at a power level of at least 1 MW without departure from nucleate boiling (DNB) occurring in the core.[6-1] Calculation of heat transfer characteristics for the PULSTAR core indicates that flow reversal which occurs during the transition from forced cooling to natural convection cooling does not result in a DNB condition for steady state power levels up to 1 MW.

In the event the coolant flow is disrupted or the Flapper is not fully shut, and the power level is above 150 kW, a SCRAM signal will occur for low primary flow and the flapper will open from the decreased amount of pressure holding it shut. Only a small amount of thermal energy must be removed at the time the flow transition goes through the reduced flow condition.

6.1.2 Cooling of a Partially Plugged Fuel Assembly

One type of accident which has resulted several times in partial meltdown of MTR plate type fueled cores has been due to blockage of the fuel assembly flow by foreign objects such as gaskets, plastic sheets, etc. The PULSTAR fuel assembly has an ESF in its design to eliminate the blocked flow type of accident. This safety feature consists of 4 one-inch (2.5 cm) diameter holes in the zircaloy box just below the fuel pin support plate located in each fuel assembly box. These holes have approximately the same flow area as the upper portion of the fuel assembly. Since all 25 fuel pins in a fuel assembly have a common flow channel,

the four holes will provide a path for the coolant in the unlikely event that a foreign object falls over the upper portion of the fuel assembly.

Flow distribution for the core will change by less than 7 percent due to holes on peripheral fuel assemblies having a less restrictive path since they are facing the lateral water reflector. The pressure drop across the upper support plate was calculated to be only 7 percent of the total pressure drop across the PULSTAR fuel assembly. The flow variation between the fuel assemblies will be such that the fuel assemblies with the higher flow rates will also be the same fuel assemblies which have the higher power generation rate.

6.1.3 Confinement Fans

The Reactor Building main ventilation system has two redundant 600 cfm (1,271 ℓs^{-1}) confinement exhaust fan system with separate filter and carbon adsorber banks. This redundancy will ensure that in the event of airborne contamination in the Reactor Building the main ventilation will shut down and the confinement exhaust fan will maintain the Reactor Building at a negative pressure and purge the contaminated air through filters prior to discharge from the 100 foot (30.5 m) exhaust stack. In addition, the confinement fans may be powered by the Auxiliary Generator.

6.1.4 Control Rod Hold Down

The control rods for the PULSTAR Reactor design are engineered so that it is impossible to remove the complete control rod assembly without first unloading the four fuel assemblies adjacent to the control rod position. This is accomplished by small projections on the bottom of each control rod guide on which adjacent fuel assemblies rest when seated in the grid plate.

6.1.5 Shielded Storage for Radioactive Fuel

The pool design incorporates the use of two fuel storage pits. The spacing and capacity of these storage pits are designed to have a multiplication factor of less than 0.8 when loaded with new fuel assemblies.

Being the low point in the pool, the storage pits are always water filled to provide cooling by natural convection.

In the event repairs must be made near the core location, these fuel storage pits would be used to store the fuel while repair work is executed. Additional shielding can be provided by positioning a lead slab or similar shield over the top of the loaded fuel storage pit. The shielding would be positioned in such a manner as to not obstruct cooling to the irradiated fuel assemblies stored in the pit.

An additiona fuel storage locations are provided

of the pool in a subcritical configuration.

6.2 Routine Tests and Inspections

For the ESFs given in Section 6.1.1 through 6.1.5, the only safety features which would require routine tests and inspections are the confinement exhaust fan and filter system, and the flapper valve operation. Tests and inspections are covered in the Technical Specifications, and by internally generated PULSTAR surveillance files.[6-2]

6.3 References

6-1 Orlosky, P.M. et al, Convection Power Test Part I, WNYNRC Technical Note No. J-435, December, 1966

6-2 "Technical Specifications for the North Carolina State University PULSTAR Reactor", Department of Nuclear Engineering, Nuclear Reactor Program.

7 INSTRUMENTATION AND CONTROL

7.1 General Description

The instrumentation for the NCSU PULSTAR Reactor includes both nuclear and non-nuclear channels using electronic as well as pneumatic signals. Also included are the SCRAM Logic Unit and associated trip circuits that make up the Reactor Safety System. A combination of alarms, interlocks, drive inhibits and reverse drive functions are provided for the safe and efficient operation of the Reactor. In this Instrumentation and Control section, "trips" are referred to frequently as "fail safe". This means that upon loss of electrical power to an instrumentation channel, all trip circuits contained therein will act to limit reactor power or initiate reactor shutdown. Refer to the PULSTAR Technical Specifications[7-1] for the current values of trip points, power level settings, and flow rates and to the PULSTAR Operations Manual for the normal operating levels.

7.2 Nuclear Instrumentation System

The nuclear instrumentation consists of four separate channels to measure neutron flux in the reactor and initiate protective action for specific conditions. The system consists of a Source Range Channel, Log and Linear Channel, Linear Channel, and the Safety Channel. Their ranges overlap sufficiently to accurately monitor reactor power (neutron flux) from a few neutrons per second to ten megawatts. A Nitrogen-16 Channel (N-16) also provides an indirect measurement of neutron flux in the core by monitoring decay gamma radiation produced by the activation of the oxygen in the primary coolant. The overall functions of the system are to measure the power level and rate of change, to provide the reactor operator with meter and recorder outputs of the power level, and to provide usable signals to the Reactor Safety System and control circuits. Refer to Figure 7-1 for the Channel Block Diagrams.

7.2.1 Source Range Channel

The Source Range channel (SR) measures neutron flux levels in the reactor from the source range to approximately 10 watts of power. The SR Channel consists of a uranium lined fission chamber detector, drive assembly, Source Range Monitor (SRM) and the Source Range/Log N recorder. The SRM contains the High and Low Voltage Power supplies, a Pulse Preamplifier, Discriminator and By-pass Filter, Test Generator and a SR Log Count Rate and Period circuit. The Log Count Rate Meter (LCRM) and the Startup Rate meter are located on the front panel of the SRM. The LCRM has a range of six decades from 0.1 cps to 10^5 cps. Because the detector is mounted on an adjustable drive mechanism, the SRM can be used over a wide range of reactor power. The startup rate scale indicates a period from -30 seconds through infinity to +3 seconds.

At very low power levels it is necessary to count neutron pulses rather than measure the current as is done with an ionization chamber. The highly enriched uranium coating on the inside of the detector readily fissions by incoming neutrons and the fission products produced easily ionize the gas within the detector, which is biased by high voltage,

generating pulses of electrical current.

The detector is mounted in a watertight canister suspended just above the reactor core by a drive mechanism attached to the reactor bridge along with the control rod drive mechanisms. The height of the detector can be adjusted over a 24 inch (61 cm) travel by means of a drive switch on the console. Detector position is indicated by a synchro receiver and up and down limit lights. Coarse adjustment of the fission chamber height can be made at the upper extension coupling just below the drive mechanism.

The SRM receives pulses produced not only from fission products, but also from gamma radiation and noise. In general, the pulses produced by fission products are of much greater amplitude than the other pulses and can be electronically discriminated or separated by pulse height in the SRM. The Discriminator and By-pass Filter provides a pulsed signal to the Test Generator circuit which buffers the signal prior to sending the signal to the SR Log Count Rate and Period circuit. The SR Log Count Rate and Period circuit provides a level signal and a rate output signal, which drives the front panel meters. The SR Log Count Rate circuit provides a level signal to the Source Range/Log N recorder through an isolator. The SR Log Count Rate and Period circuit also provides a Transistor Transistor Logic (TTL) buffered output which drives the SR Audio and the Experimental Rack.

The SRM contains bistable trip circuits which produce a rod withdrawal inhibit whenever a non-operate condition exist or, if count rate is <2 cps or $>9 \times 10^4$ cps. The <2 cps inhibit ensures that there is sufficient subcritical multiplication taking place in the core. The $>9 \times 10^4$ cps inhibit ensures the channel is not saturated by an excessively high count rate. The non-operate inhibit is caused whenever, the $\pm$ 15 volt power supply voltage is low, the high voltage power supply voltage is low, or CHANNEL-ON-TEST and any other front panel pushbuttons is activated. All of the trip circuits are of fail-safe design so that in the event of loss of power to the channel, a SR Inhibit will be generated. A test circuit provides inputs at 12.2 Hz, 100 kHz and +3 seconds to functionally check the drawer prior to startup.

7.2.2 Log and Linear Channel

The Log and Linear channel is used to measure the reactor power from less than one watt to 10 megawatts on one continuous scale and to provide startup rate data during the approach to full power. The channel consists of a compensated ion chamber, high voltage, low voltage, compensation voltage power supply, log amplifier, linear and period amplifiers, bistable trips, and the Source Range/Log N recorder.

The detector for this channel is a neutron sensitive, gamma compensated ionization chamber (CIC). The CIC is constructed with two volumes having a common collector electrode. One volume is sensitive only to gamma radiation. By using a coating of boron, the other volume is sensitive to neutrons as well as gamma radiation. The two volumes are connected electrically with power supplies of opposite polarities so that the resulting output current is due to neutrons alone. The detector is surrounded by a sleeve one inch (2.5 cm) thick, a bottom plate one and one half inches (3.8 cm) thick and a top plate one half inch (1.3 cm) thick, all made of lead. The detector is mounted in a waterproof aluminum canister

suspended from the bridge near the core with a screw mechanism for adjusting the final height of the detector. Water is prevented from leaking into the detector canister by pressurized nitrogen supplied from portable tanks.

The log amplifier measures the current produced by the CIC. As the name implies, the output of the log amplifier is a voltage proportional to the logarithm of the input current. This output voltage is used to drive the display driver, period amplifier, linear amplifier and the bistable trip circuits. The log amplifier provides an isolated output to the Source Range/Log N recorder.

The period amplifier provides startup rate information on the front panel meter ranging from -30 seconds through infinity to +3.0 seconds. The linear amplifier receives an input from the Log amplifier and uses this signal to provide a linear indication of power from 0% to 125% reactor power.

The log amplifier contains two bistable trip circuits. The first is a downscale trip set at 4 watts. The lighted switch illuminates when the circuit has been reset. The other is an upscale trip that enables the Low Primary Flow and Flapper Open SCRAM circuits to produce an automatic SCRAM when the power level exceeds 150 kW and there is no primary coolant flow. Both of these trip circuits are fail-safe.

Calibration circuits are built in to functionally check the trip circuits, to check the accuracy of the log, linear and period amplifiers. The following test signal are provided to test the channel: at 1 mA, 0.1 pA, 10 pA, 3 Seconds, Non-Operate, Log Test, Period Test and Linear Test. All the front panel test pushbuttons are of the spring return to the operate position design which prevents test information being displayed when the pushbutton is released.

7.2.3 Linear Channel

The Linear Power Channel provides the reactor operator with accurate information on reactor power level starting in the source range through 125% of full power. The channel consists of a CIC identical to the one used in the Log and Linear Channel as described in Section 7.2.2, high voltage and compensating voltage power supplies, a multi-range monitor, and the Linear/Safety Recorder. The monitor has 16 ranges in multiples of 1 and 3 from 30 mW to 1 MW. The monitor can be set to range automatically or manually at the operator's discretion. Normally the monitor is set to range automatically. The maximum range to which the monitor will automatically range up to can be set using the RANGE SELECT Mode Switch to set the RANGE LIMIT to any one of the sixteen ranges between 30 mW and 1 MW. The monitor provides an isolated output to the Linear/Safety Power Recorder. Built in test circuits are provided to functionally check the trip circuits and the accuracy of the monitor prior to operations.

The Flux Controller which is an integral part of the monitor is the major component of the Automatic Channel. The interlocks that must be satisfied prior to engaging the Automatic Channel are discussed further in Section 7.5.1.

Four trip circuits in the channel are of fail-safe design and provide trips for the Linear Over-Power Reverse, the Linear Channel Over-Power SCRAM and the Flux Controller absolute deviation (FC ABS DEV). The reverse (Trip #1) will cause Safety #1, Safety #2 and the Reg Rod to drive in when reactor power exceeds 110% (-2.0%, +0.0%). The Linear Channel Overpower SCRAM (Trip #2) occurs when reactor power exceeds 120% (-2.0%, +0.0%). The FC ABS DEV (Trip #4) prevents Automatic Channel operation when the Demand Pot setting exceeds actual reactor power by more than ±9%. Trip #3 is a spare and not presently used.

7.2.4 Safety Channel

The Safety Channel is the redundant channel for the Linear Channel which also measures and indicates the reactor power level. This channel consists of an Uncompensated Ion Chamber (UIC), a high voltage power supply, a multi-range monitor and the Linear/Safety recorder. The monitor has two trip circuits that initiate automatic protective action if any preset levels are exceeded.

Because this channel is used primarily in the power range where current produced by gamma radiation is negligible compared to neutrons, compensation is not necessary. The UIC is mounted in a waterproof canister identical to those used in the Log and Linear and Linear Channels without the lead shielding. As with the other detectors, vertical position adjustment and nitrogen pressurization are provided. The high voltage and signal cables going to the detector from the PULSTAR Control Room are run in conduit that is routed physically separate from the conduit carrying the cables to the other detectors.

The monitor has 16 ranges in multiples of 1 and 3 from 30 mW to 1 MW. The monitor can be set to range automatically or manually. Normally the monitor is in manual set to the 1 MW range. The monitor provides an isolated output to the Linear/Safety Recorder. Built in test circuits are provided to functionally check the trip circuits and the accuracy of the monitor prior to operations.

The two trip circuits in the monitor are of fail-safe design and provide trips. A latching trip that enables the Low Primary Flow and Flapper Open SCRAM circuits to produce an automatic SCRAM when the power level exceeds 150 kW and there is no primary coolant flow. The second produces a Safety Channel Overpower SCRAM at 1.2 MW.

7.2.5 N-16 Channel

The N-16 Channel is used to indicate the reactor power level as a function of the decay of N-16 gammas produced when the coolant passes through the operating core. The channel consists of a special detector, a high voltage power supply, and an electrometer. This channel is used as a reference to determine power level during the transition between 900 kW and full power. It is also used as an auxiliary indication of the power level while operating at steady state and when the detectors have to be repositioned due to core changes. The N-16 Channel has shown that it can be used quite successfully to detect fuel leaks like the one experienced at the BMRC PULSTAR. This channel may provide the first

indications of a fuel pin failure.

The detector is a xenon filled, gamma sensitive ionization chamber. The detector output is linear with respect to actual reactor power, assuming a constant coolant flow rate through the core, and is not appreciably affected by primary coolant temperature or rod shadowing. The detector is mounted on the 10 inch (25.4 cm) primary coolant hot leg in the valve pit.

The output of the N-16 detector is measured and displayed by an electrometer mounted in the control console. The electrometer is set on a current range (1 x 10^{-8} amps) which produces a current equivalent to reactor power. There is an auxiliary 0 to 1 volt output which drives an analog current meter. This analog output is provided so the operator can discern a rate of change on the N-16 Channel. There are no trips associated with the N-16 Channel.

7.3 Non-Nuclear Instrumentation

7.3.1 Flow Measuring Channel

The flow of the primary coolant is determined by measuring the pressure drop across a calibrated orifice. The channel consists essentially of a certified orifice plate, a differential pressure transmitter, two flow gauges, one mounted on the wall in the MER near the transmitter and the other on the Control Console, a pressure regulator, and pressure-electric (P-E) switches. Refer to Figure 9-1. A permanently installed inverted "U" tube manometer is mounted beside the flow transmitter for calibration purposes.

The pressure regulator reduces the 100 psi (687 kPa) air from the Reactor Air Supply to 22 psi (151 kPa) for use by the flow transmitter. A normally closed P-E switch monitors the output of this regulator. Should the outlet pressure increase above a setpoint, a Low Primary Flow SCRAM will be generated if the reactor is operating above 150 kW or below 150 kW if the enables were not reset. Refer to Figure 7-2.

The flow transmitter produces a pneumatic signal proportional to the square root of the pressure drop across the orifice mounted in the 10 inch (25.4 cm) primary coolant pipe. This pressure signal is sent to the two gauges with square root scales which are labeled in gallons per minute. A normally open P-E switch mounted in the console will generate a Low Primary Flow SCRAM, if the coolant flow rate drops below 475 gpm (29.97 ℓs^{-1}). Loss of air pressure will produce the same results.

7.3.2 Flow Monitoring Channel

The Flow Monitoring Channel uses an independent method to detect the loss of flow through the reactor core. This channel consists of a counter-weighted circular metal disk (Flapper) covering an opening on the plenum below the core grid plate, a push-rod, and a microswitch.

The Flapper is held closed when there is sufficient flow down through the core to produce a differential pressure between the plenum and the pool. As long as the flow keeps the Flapper closed, a long push-rod actuates a normally open microswitch on the bridge. If the Flapper opens for any reason and the reactor power level is greater than 150 kW, a Flapper Open SCRAM is generated. Since the Flapper is a monitoring channel, there are no other indications other than the annunciator panel.

7.3.3 Pool Level Measuring Channel

The height of the water above the reactor core is continuously measured by a bubbler system and displayed on a pressure gauge (with level scaling) on the control console. A bubbler system basically measures the pressure required to force a constant flow of air to a given depth of water. This pressure is displayed on a gauge scaled from -36 inches (-91.4 cm) to +6 inches (15.2 cm) of water referenced to 20 feet (6.1 m) of water above the core.

Air is supplied from the 100 psi (687 kPa) Reactor Air Supply shown in Figure 9-1. A filter/regulator mounted in the control console reduces this air pressure to 25 psi (172 kPa). The air then passes through a flow regulator also mounted in the console at 0.08 scfm (0.14 m^3/sec) and then to the bubbler pipe through copper and plastic tubing. The level gauge, calibrated in inches of water, indicates the pressure required to force air through the bubbler pipe against the hydrostatic head of the pool water.

Two P-E switches monitor the air pressure (pool level). One generates an Abnormal Pool Level Alarm on the console if the pool level changes $\pm$ 6 inches ($\pm$15.2 cm) from the zero reference level which is the bottom of the overflow weir. The other P-E switch generates a Low Water Level SCRAM, and a trip of the primary pump if the pool level drops to -36 inches (-91.4 cm) which corresponds to 17 feet 0 inches (5.2 m) above the core.

It should be noted that the two radiation monitors mounted over the pool serve as diverse channels to monitor the loss of pool water. These other channels are discussed in Sections 5.2.2 and 10.2.2.

7.3.4 Temperature Measuring Channel

The Temperature Measuring Channel measures the water temperature at seven locations in the Primary and Secondary Coolant. The channel consists of Resistance Temperature Detectors (RTD) mounted in thermal wells, power supply, 4-20 mA temperature transmitters, and a temperature recorder mounted in the Control Console. Temperature is measured in the pool, at the Hot and Cold Leg in the valve pit, at the Heat Exchanger Primary inlet and outlet, at the Heat Exchanger Secondary inlet and outlet, and at a test transmitter. With the exception of the Hot Leg and the Heat Exchanger Primary inlet, all of the RTDs generate a High Temperature Alarm when 116°F (46.7°C) is exceeded. The channel will generate a Pool Temperature SCRAM at a pool temperature (T2) of 116°F $\pm$ 0.5°F.

7.3.5 Pool Temperature Monitoring Channel

The temperature of the pool water is monitored by a thermal switch suspended in the pool near the core which generates a High Pool Temperature Alarm on the console when 114°F (45.6°C) is reached. A normally closed push-button switch is mounted on the console in series with the thermal switch to test the circuit.

7.3.6 Primary Coolant Resistivity

The resistivity of the primary coolant is measured at two locations. The first is in the pool near the core and the other is at the outlet of the Primary Demineralizer. A spring return switch normally selects the pool. A Low Resistivity Alarm is generated when the resistivity of the water drops below 500 kohm·cm.

7.3.7 Rod Position Indicators

The positions of the Control Rod Drive Mechanisms (CRDM) are indicated by synchro receivers mounted on the Control Console. The indicators are calibrated from 0 to 24 inches (61 cm) of travel out of the core. Each CRDM also has up-limit and down-limit lights. Normally the position of three of the control rods is the same as the CRDM when they are magnetically coupled. If the control rods are not coupled, an Off-Magnet light is illuminated on the console, and if they are within one inch (2.5 cm) of being full down, a Seated light is illuminated. The fourth control rod, the Shim Rod (previously referred to as the Pulse Rod), is mechanically coupled and is normally fully withdrawn.

7.3.8 Reactor Air System

Reactor Air is supplied by a standard air compressor and moisture separator located in the Mechanical Equipment Room (MER). The 100 psi (687 kPa) air supply is used for all pneumatic instrumentation and control dampers within the Reactor Building. Refer to Figure 9-1 for details.

The air compressor receives 480 volt, three phase, 60 Hertz power from MCC No. 1. The compressor is normally energized and cycles automatically to maintain between 80 (550 kPa) and 100 psi (687 kPa) receiver pressure. It can be started and stopped from the Control Console. If electrical power is lost to the air compressor and then restored, the air compressor will automatically start after 5 minutes. This is due to an adjustable time delay relay which is set for a 5 minute delay and it is located in the MER on the left side of MCC #1. If the compressor should fail while the reactor is operating, a high pressure nitrogen tank and regulator can be connected to the air supply line by a valve to permit approximately one half hour of additional run time. Should the need arise, the ventilation control dampers can be connected to the Burlington Engineering Laboratories (BEL) air compressor by valves. In addition to supplying air to the ventilation control dampers in the MER, the air supply line header runs across the ceiling of the MER and passes through the wall into the Reactor Bay where it divides. One of the branch lines terminates in a valved, capped supply for experimental use and a dirt leg to exhaust airborne debris. The other

branch line goes up and into the control room were it is filtered and regulated for the Flow Measuring Channel (Section 7.3.1) and the Pool Level Measuring Channel (Section 7.3.3). There is also an unused regulator with valves in the bay along the east wall just before the supply line enters the control room which can be used as a service air supply.

7.4 Reactor Safety System

A reliable reactor protection system functions to ensure that all modes of reactor operation are safe, therefore, the protective action of the system is designed to automatically terminate operations should safe operating conditions cease to exist.

Due to the inherent characteristic of some reactor variables, safe operation is sufficiently ensured, should these variables deviate from preset limits, if operation is manually terminated by the licensed operator. The NCSU PULSTAR Reactor Safety System (RSS) features predominately automatic shutdown mechanisms. The Reactor Safety System is defined as that specified combination of instrumentation channels and associated circuitry which either provides the automatic protective action or provides the alarm which requires that manual protective action be taken. Specifically, the Reactor Safety System consists of the SCRAM Logic Unit with the magnet current circuits, the protective instrumentation channels less secondary readouts, and the associated circuitry. The protective instrumentation channels which are specified to be part of the Reactor Safety System are listed in Table 7-1.

7.4.1 SCRAM/Alarm Control Circuits

The SCRAM/Alarm circuits are energized through the Power circuit breaker while the Protective Channels receive their electrical power through the Instrumentation circuit breaker. Both breakers are mounted on the right side of the console. Any demand for protective action (SCRAM demand) activates the appropriate SCRAM/Alarm circuit. Typical SCRAM/Alarm circuits are shown in Figure 7-2. The SCRAM relays are normally energized and any open circuit condition, either an actual SCRAM demand, loss of electrical power, or relay failure will cause the SCRAM relay to de-energize and remove the input signal from the SCRAM Logic Unit. Upon the loss of this signal, the SCRAM Logic Unit interrupts power to the control rod magnets and causes an automatic shutdown of the reactor. Simultaneously, the SCRAM annunciator horn sounds and lights illuminate on the control console to notify the reactor operator which channel initiated the SCRAM. When either a SCRAM or an alarm is received the following is the sequence that takes place:

a. The activated annunciator panel SCRAM or alarm light(s) will be in the "Fast Flash" mode, and the alarm will be energized.

b. Pressing the Acknowledge push button silences the alarm and the activated alarm goes to "Slow Flash".

c. Pressing the "First Reset" preforms the following:
   1) The alarm will lock in solid if the alarm condition still exist.

2) The alarm clears automatically when the alarm condition clears.

d. After a SCRAM condition clears the SCRAM must be cleared by pressing the "SCRAM Reset" pushbutton.

7.4.2 SCRAM Logic Unit

The SCRAM Logic Unit is the major item in the Reactor Safety System. This unit contains the power supply for the magnets and the system level logic for all protective channels initiating an automatic SCRAM. A block diagram of the SCRAM Logic Unit is shown in Figure 7-3.

7.4.2.1 Basic Operation

The power supply for the magnets is energized only if the Power and Instrumentation circuit breakers are closed and the Reactor Keyswitch is in the ON position. The output of this supply is +40 volts dc and is capable of supplying an adjustable current up to 125 mA to each magnet. The SCRAM Logic Unit has ten input channels; however, only seven are active. The remaining inputs serve as spares and are jumpered. A supply of +12 volts dc is derived from the magnet power supply and is used for the logic voltage. This voltage is fed through the closed SCRAM relay contacts and returned to the SCRAM Logic Unit for each protective channel involved. As long as the circuit is closed and +12 volts is present at the logic circuit input, a logic 0 is generated and there is no SCRAM demand. Conversely, if the +12 volts is interrupted, a logic 1 is generated and a SCRAM demand occurs.

The logic circuits control the current to the magnets in two ways. Solid state circuits will electronically turn off the current to each magnet if a SCRAM demand occurs, and, for redundancy, a relay contact will also interrupt the magnet current bus. Additionally, a set of contacts from a relay in each of the input circuits is connected in series with the magnet power bus. This requires that the seven inputs be at logic 0 to have magnet power available to the control rods. Once the power to the magnets has been interrupted by a SCRAM demand, a lockout circuit ensures that the current cannot be restored until the reactor operator manually resets the SCRAM Logic Unit.

7.4.2.2 Bypass Circuits

The SCRAM Logic Unit has provisions which prevent Linear and Safety channel SCRAMs from occurring for a predetermined amount of time after a set of contacts closes. These circuits are no longer used. Neither the Manual SCRAM nor the Reactor Keyswitch are bypassed by any means. In order to test the protective action of the Flow Measuring Channel and the Flow Monitoring Channel independently, a SCRAM bypass switch for each channel is mounted on the control console. The two switches are spring return with safety covers and are wired so that only one channel can be bypassed at a time, and if both switches are operated, neither channel is bypassed. This ensures that there is at least one channel active for protection against loss of coolant flow.

7.5 Operating Features

In order to provide for the safe and efficient operation of the reactor, several operating features are provided in the design. Among these features are Interlocks, Alarms, Drive Inhibits, Reverse Drives, and Rod Drop Time.

7.5.1 Interlocks

An interlock is a mechanical or electrical device that will prevent a particular action from occurring until all prerequisites for that action are satisfied. The PULSTAR Reactor has the following interlocks: fission chamber movement, up-drive power for the individual control rod drive mechanisms, up-drive power for the gang control of the drive mechanisms, magnet power, and automatic power control.

Fission Chamber movement is allowed only if the Gang Drive switch is in the mid position. As with the CRDM there is an Up-limit and a Down-limit switch to stop the motors. Up-drive power for the individual CRDM requires that no Source Range Channel Inhibit or Reverse be present and that the Ganged Insert switch be in the OUT position. The "Low shutdown Margin" alarm informs the operator that the reactor may be critical with less than the minimum shutdown reactivity from the control rods available. Switches on the CRDM determine this margin when the Log N Operative switch has been depressed after reaching 4 watts.

The fourth control rod (the Shim Rod) requires that the Reactor Keyswitch be in the ON position and that no Source Range Channel Inhibit or Reverse be present for motion in either direction. Any future references to control rods does not necessarily apply to this CRDM. Up-drive power for the CRDM by the Gang Drive switch requires the Ganged Insert switch be in the OUT position and no Reverse be present with the Reactor Keyswitch OFF, and no Reverse or Source Range Channel Inhibit be present with the Reactor Keyswitch ON. There are no interlocks that prevent the drive mechanisms from being moved in the down direction. Magnet power is energized if electricity is available to the console, the Reactor Keyswitch is ON, and there are no SCRAM demands present.

Automatic control of the Regulating Rod is allowed if the CRDM is withdrawn beyond 13.5 inches (34.4 cm) rod height, the Mode Keyswitch is in the STEADY STATE position, FC ABS DEV is within $\pm 9\%$, and the Gang Drive switch is in the neutral (mid) position.

7.5.2 Annunciator Alarms

An Alarm is the annunciation indicating that a setpoint of a less serious nature than a SCRAM has been exceeded and may require action by the reactor operator. Refer to Section 7.4.1 for the sequence of operation of the Panalarm annunciator. Refer to Table 7-2 for the Annunciator Alarms and Setpoints.

7.5.3 Annunciator SCRAM Alarms

A SCRAM alarm is the annunciation indicating that a safety system setpoint has been exceeded. This requires the reactor to be automatically shutdown by the SCRAM Logic Unit or a manual SCRAM by the reactor operator. Refer to Section 7.4.1 for the sequence of operation of the Panalarm annunciator. Refer to Table 7-3 for the SCRAM Alarms and Setpoints.

7.5.4 Drive Inhibits

An Inhibit is the prohibition of any control rod withdrawal. Inhibits are generated by the conditions listed in Section 7.2.1 and discussed in 7.5.1 during a reactor startup and approach to power.

7.5.5 Reverse Drives

One of the functions of the Reverse drive is to automatically decrease power without unnecessarily dropping all the control rods by a protective action SCRAM. If power were to slowly exceed 110% (-2.0%, +0.0%) on the Linear Channel, the three control rods would drive toward the 0 inch (0 cm) position as long as the condition was not acknowledged. The Ganged Insert switch drives the control rods into the core as long as the switch is in the IN position. This is the normal method of shutting down the reactor. When ever Magnet Power is lost (SCRAM) and the Reactor Keyswitch is ON, a Reverse drive is generated to ensure that in the event a control rod did not fully seat, the CRDM would push it down.

7.5.6 Rod Drop Time

The time referred to as Rod Drop Time is the interval between the instant of a SCRAM demand to the SCRAM Logic Unit and the instant that the Rod Seated light illuminates. This definition is consistent with the safety analysis of Appendix 3B. Rod Drop Times will be measured at intervals to verify that drop time is within the limits specified in the Technical Specifications[7-1].

7.6 Control Systems

All activity is centralized at the Control Console in the PULSTAR Control Room, and switches, gauges, and dials that control most of the equipment associated with the reactor are within easy reach of the operator. The layout of the console is shown in Figure 7-4A and 7-4B. The instruments in this console provide the reactor operator with all the information necessary for safe and efficient manipulation of the controls.

7.6.1 Steady State Controls

The reactor is controlled by positioning the neutron absorbing rods in the space between two rows of fuel assemblies. Two of the control rods are labeled Safety No. 1 and Safety No. 2; the third is the Regulating or Reg Rod which may be operated by an Automatic

Channel to maintain the reactor at a specified power. The Automatic Channel consist of a Flux Controller, which is integral to the Linear Monitor, Power Demand Potentiometer mounted on the console, and Auto and Manual pushbuttons. Figure 7-5 shows the interlocks and switches involved in withdrawing the Control Rod Drive Mechanisms. As stated previously, there are no interlocks that will prevent driving the control rods into the core.

The control rod magnets are energized when the Reactor Keyswitch is in the ON position and there are no SCRAM demands present. The rods may be withdrawn either individually or in a gang of any combination of rods. The worth of the rods and their withdrawal speed are such that the total reactivity insertion rate of all the rods on gang will not be greater than 100 pcm/second. The positions of the Control Rod Drive Mechanism are continuously indicated on the synchro receivers mounted on the console. The position of the control rod is inferred by the combination of Off Magnet Light and Rod Seated Light.

Once a steady power level has been attained manually, this level may be maintained by the Automatic Channel if the interlocks are satisfied. When an interlock is no longer satisfied after automatic control is achieved, the Automatic Channel will disengage and an annunciator on the console will activate. The Automatic Channel may be used to continually position the Reg Rod to maintain reactor power at a specified level. The error signal is generated by the Flux Controller which is the difference between actual power and the power demand potentiometer. The Flux Controller produces a series of pulses that actuate either an up drive or a down drive relay supplying electrical power to the CRDM motor that will bring the power level back to the desired point. For safety purposes, the maximum deviation is limited by the FC ABS DEV (Trip #4). If this value is exceeded, the Automatic Channel will disengage and an annunciation is generated. The Automatic Channel is not required for reactor operations and may be used at the discretion of the reactor operator.

7.6.2 Process Control

As a convenience to the reactor operator, eight lighted push-button sets and two single switch assemblies are mounted on the right side of the console to control the following equipment: Primary Pump, Secondary Pump, Reactor Air Compressor, Main H&V Fans, Primary Demineralizer Pump, Confinement Fans 1 and 2, P-N Blower, Beam Tube and Thermal Column Exhaust Fan (BT&TC), and the Auxiliary Generator.

7.7 References

7-1 "Technical Specifications for the North Carolina State University PULSTAR Reactor", Department of Nuclear Engineering, Nuclear Reactor Program.

7-2 "PULSTAR Emergency Plan" North Carolina State University Department of Nuclear Engineering, Nuclear Reactor Program, Revision 4, September 17, 1996

TABLE 7-1
PROTECTIVE CHANNEL

Protective Channel	SSS	LSSS	Required Protective Action
1. "Reactor On" Keyswitch	-	-	Manual SCRAM
2. Manual SCRAM Button	-	-	Manual SCRAM
3. Startup Channel	<2 cps	-	Inhibit when less than minimum count rate (when <4 watts).
	$>9x10^4$ cps	-	Inhibit when greater than maximum count rate (when <4 watts).
4. Log and Linear Channel	150 kW	250 kW	Enable Flow/Flapper SCRAM.
5. Linear Channel	1.2 MW	1.3 MW	Automatic SCRAM.
6. Safety Channel	150 kW	250 kW	Enable Flow/Flapper SCRAM.
	1.2 MW	1.3 MW	Automatic SCRAM.
7. Flow Measuring Channel	475 gpm ($29.97\ \ell s^{-1}$)	450 gpm ($28.39\ \ell s^{-1}$)	Automatic SCRAM when enabled.
8. Flow Monitoring Channel	-	-	Automatic SCRAM when enabled.
9. Flow and Flapper SCRAM Bypass Test Switches	-	-	Only one switch operative at one time.
10. Pool Level Measuring Channel	17' (5.2 m)	14' 2" (4.3 m)	Automatic SCRAM.
11. Over-the-Pool Monitor	100 mRem/hr	-	Alarm/Manual SCRAM
12. Pool Temperature Measuring Channel	116°F (46.7°C)	117°F (47.2°C)	Automatic SCRAM
13. Pool Temperature Monitoring Channel	114°F (45.6°C)	117°F (47.2°C)	Alarm/Manual SCRAM

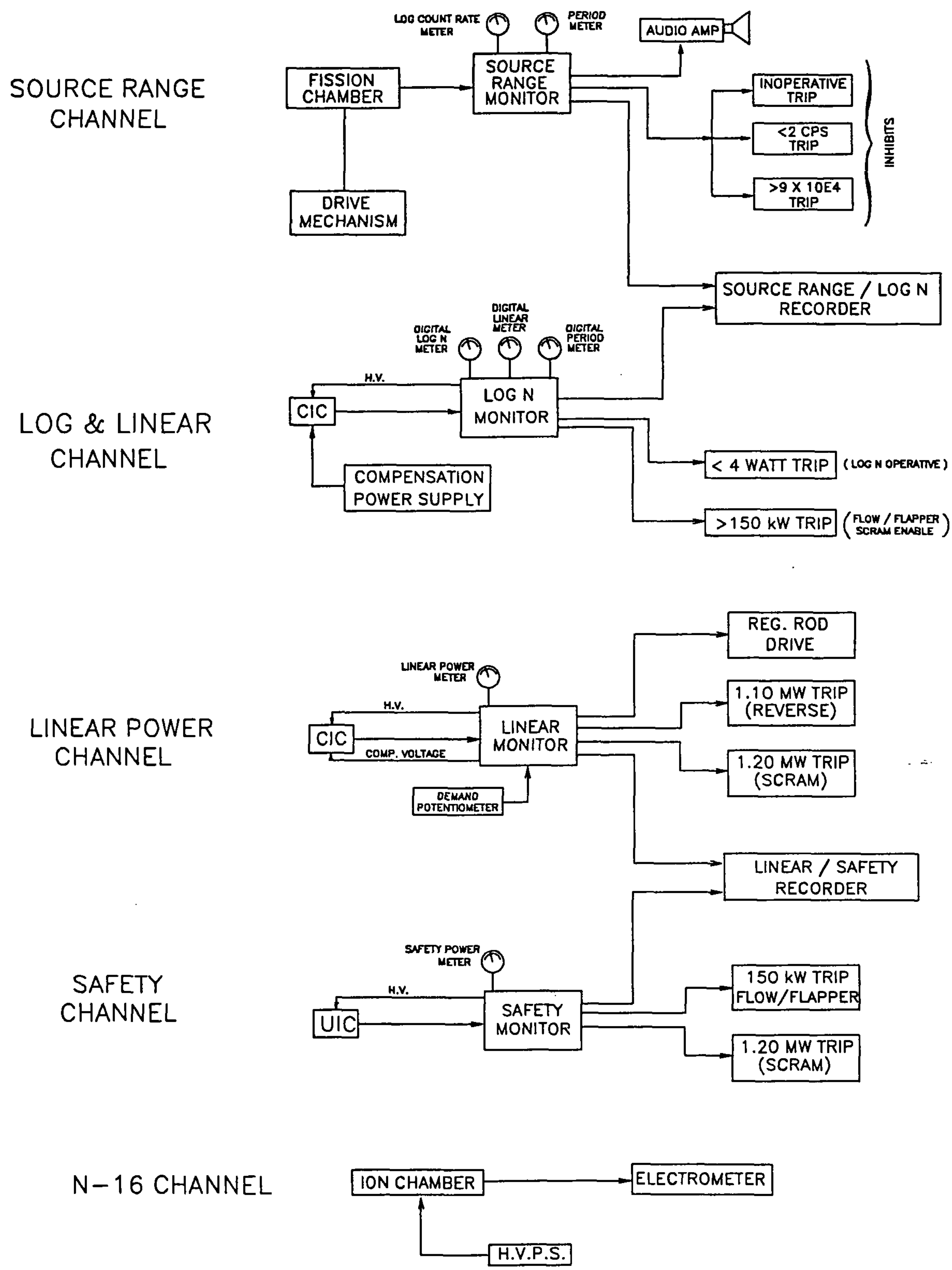
TABLE 7-2
ANNUNCIATOR ALARMS

ANNUNCIATOR ALARM	SETPOINT
Control Room Radiation Alert	See Emergency Plan Table 5[7-2]
Over the Pool Radiation Alert	See Emergency Plan Table 5[7-2]
West Wall Radiation Alert	See Emergency Plan Table 5[7-2]
Demineralizer Radiation Alert	See Emergency Plan Table 5[7-2]
Stack Gas Radiation Alert	See Emergency Plan Table 5[7-2]
Stack Particulate Radiation Alert	See Emergency Plan Table 5[7-2]
Auxiliary GM Radiation Alert	See Emergency Plan Table 5[7-2]
Filter GM Radiation Alert	See Emergency Plan Table 5[7-2]
R-63 Fans Not Operating	R-63 Fans Secured
Low Natural Gas Pressure	Loss of Natural Gas Pressure
Low Shutdown Margin	See SAR Section 7.5.1
Low Pool Resistivity	See SAR Section 7.3.6
Source Range Inhibit	See SAR Section 7.2.1
Low Reactor Building Diff. Pressure	≤0.2"Bay vs Atmos.(≥4.5 min.)
High Coolant Temperature (RTD)	See SAR Section 7.3.4
PRI Piping Vault Door Not Closed	PPV Door (hatch) or Gate Not Closed
Linear Channel Over-Power Reverse	See SAR Section 7.2.3
Pneumatic Sample in Reactor	PN Shuttle in Reactor Core
Auto Channel Disengaged	See SAR Section 7.6.1
High Pool Temperature (Switch)	See SAR Section 7.3.5
Abnormal Pool Level	See SAR Section 7.3.3
High Sump Level	> -40 in. (-102 cm) nominal Referenced from floor level
Ramp Door Not Closed	Ramp Door Not Closed
Loading Dock Door Not Closed	Loading Dock Door Not Closed

TABLE 7-3
ANNUNCIATOR SCRAM ALARMS

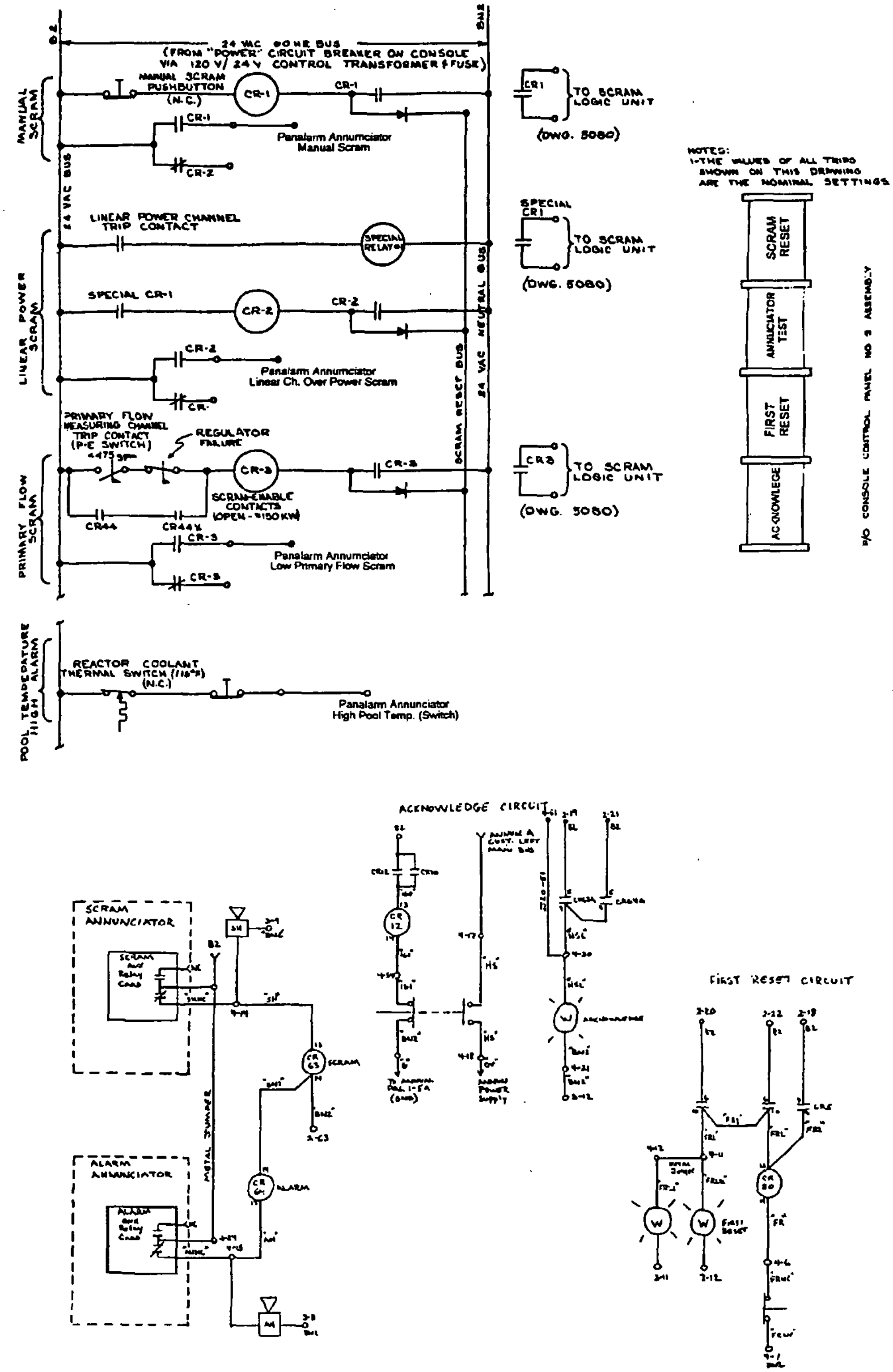
SCRAM ANNUNCIATOR	SETPOINTS
Linear Channel Over-Power	See SAR Section 7.2.3
Safety Channel Over-Power	See SAR Section 7.2.4
Low Primary Flow	See SAR Section 7.3.1
Flapper Not Closed	See SAR Section 7.3.2
Low Pool Level	See SAR Section 7.3.3
Pool Temperature	See SAR Section 7.3.4
Manual	Operator Decision

Figure 7-1



Block Diagram - Nuclear Instrumentation

Figure 7-2



Typical SCRAM/Alarm Control Circuits

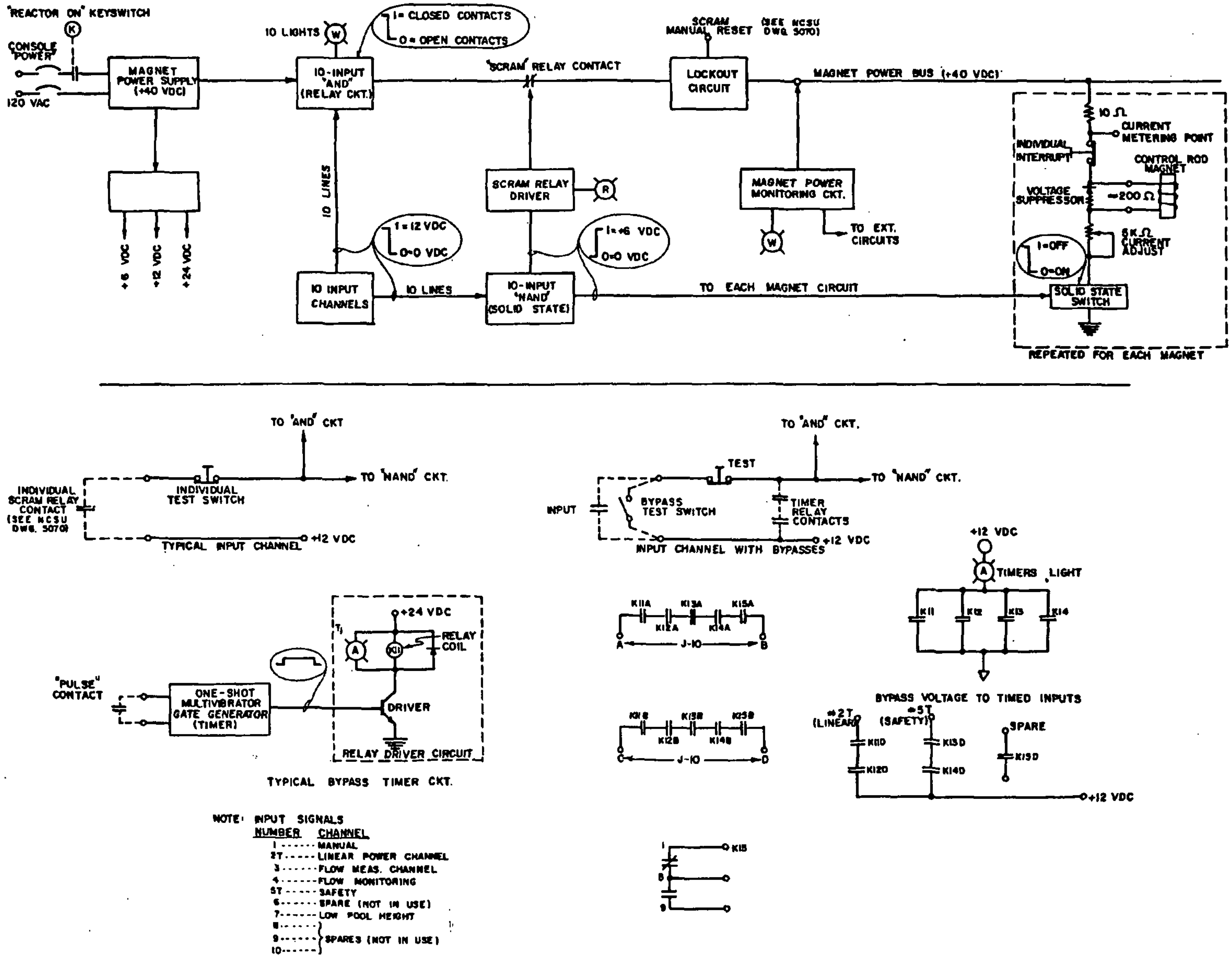


Figure 7-3

SCRAM Logic Unit Block Diagram

KEY

A - SCRAM Logic Unit
B - Source/Log and Linear Recorder
C - N-16 Monitor
D - Annunciator Panel
E - Safety Channel Monitor
F - Primary Coolant Flow Gauge
G - Pool Water Level Gauge

KEY

H - Temperature Recorder
I - Linear and Safety Recorder
J - Shim Rod Position Indicator
K - Flux Controller Demand Potentiometer
L - Linear Channel Monitor
M - Log and Linear Channel Monitor
N - Control Rod Position Indicators
O - Source Range Monitor

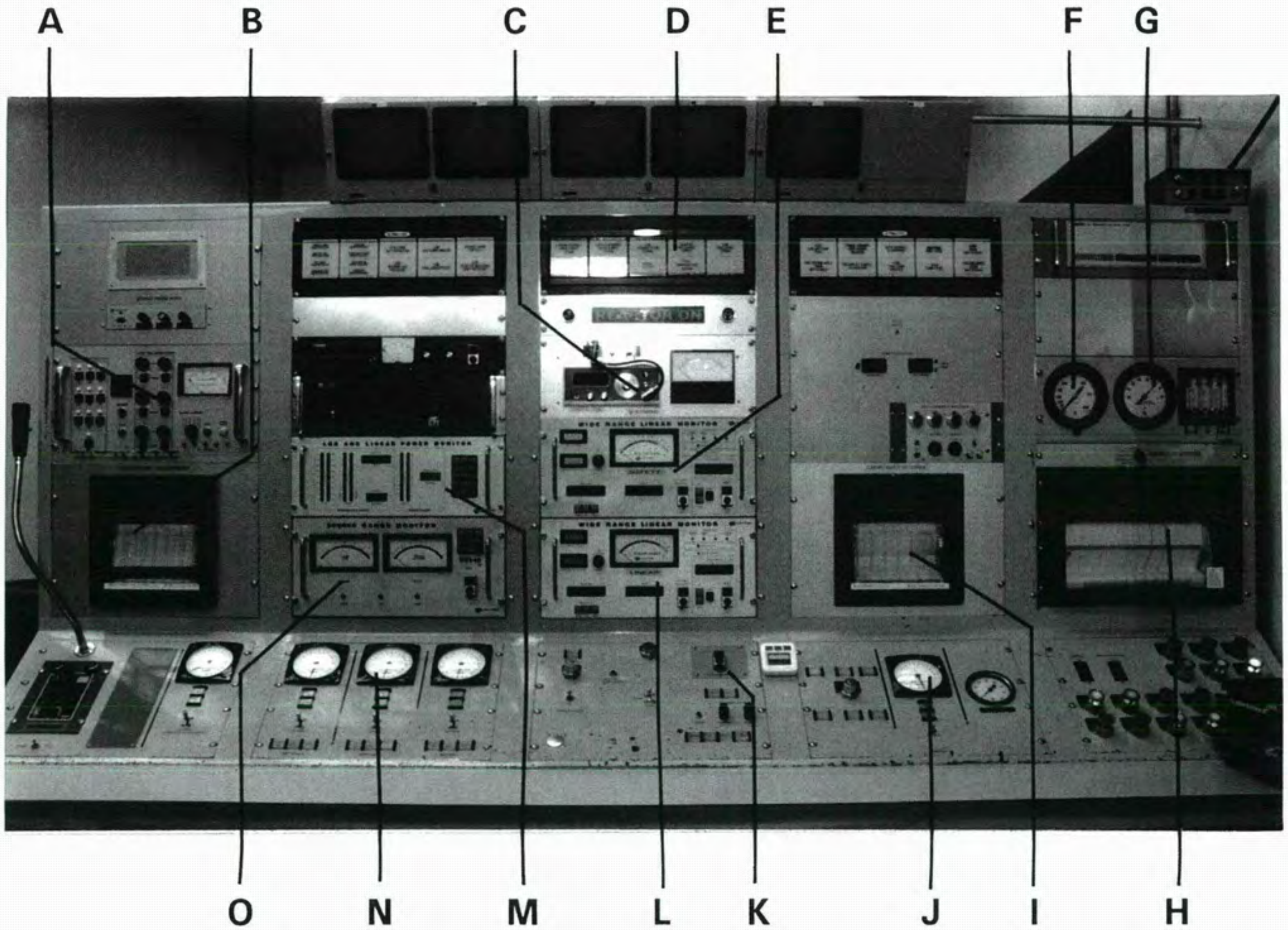
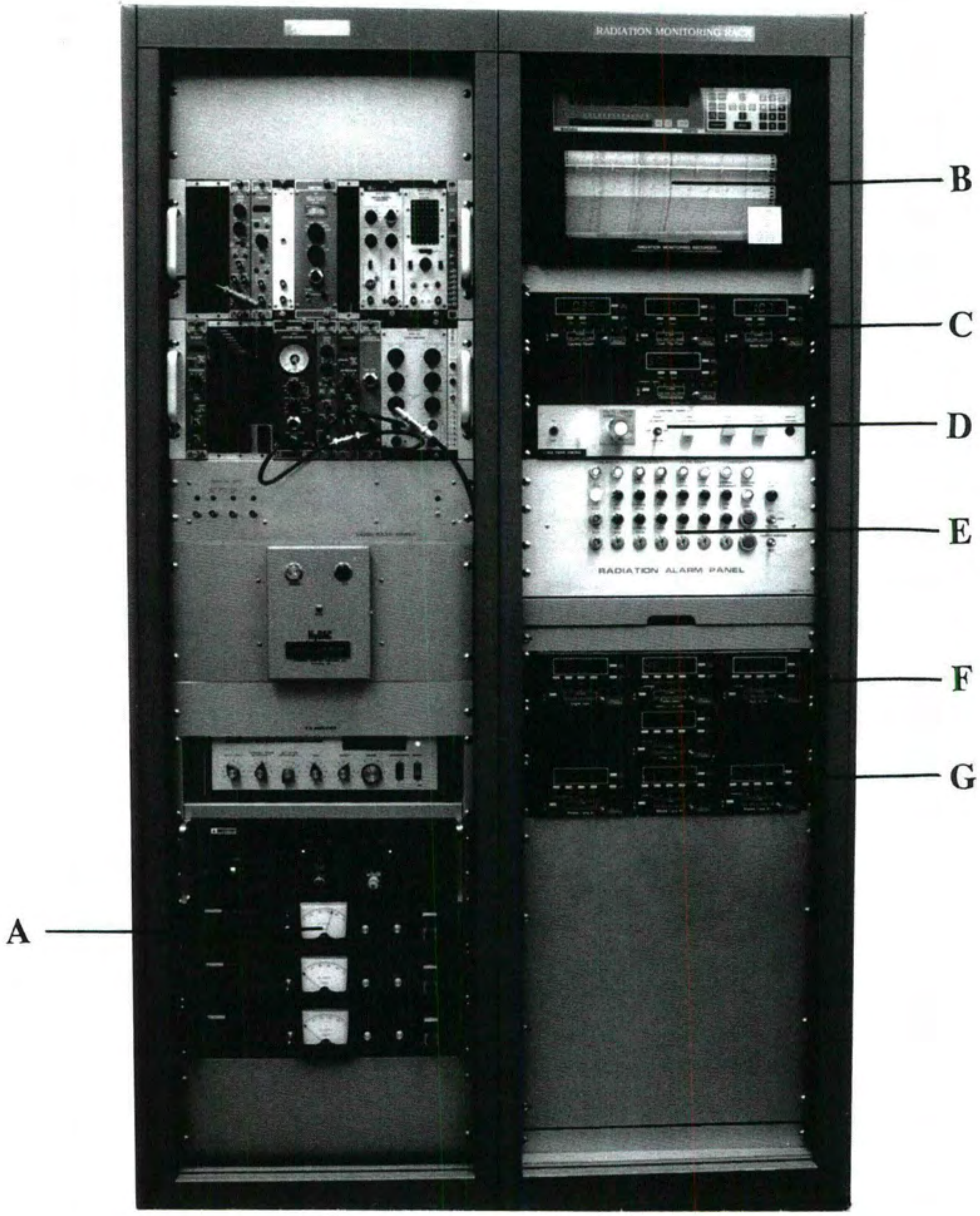


Figure 7-4A

PULSTAR Control Console

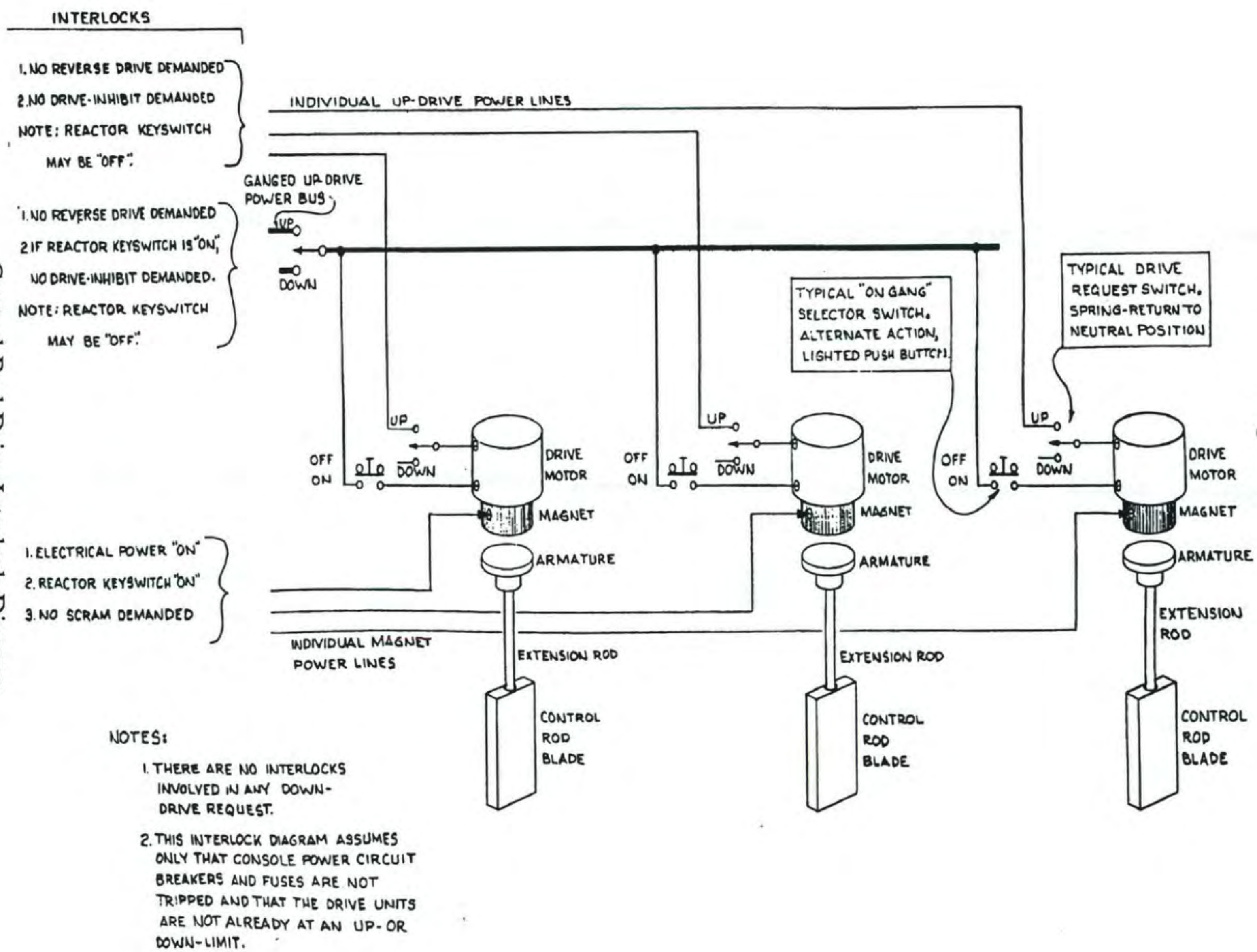
Figure 7-4B



KEY

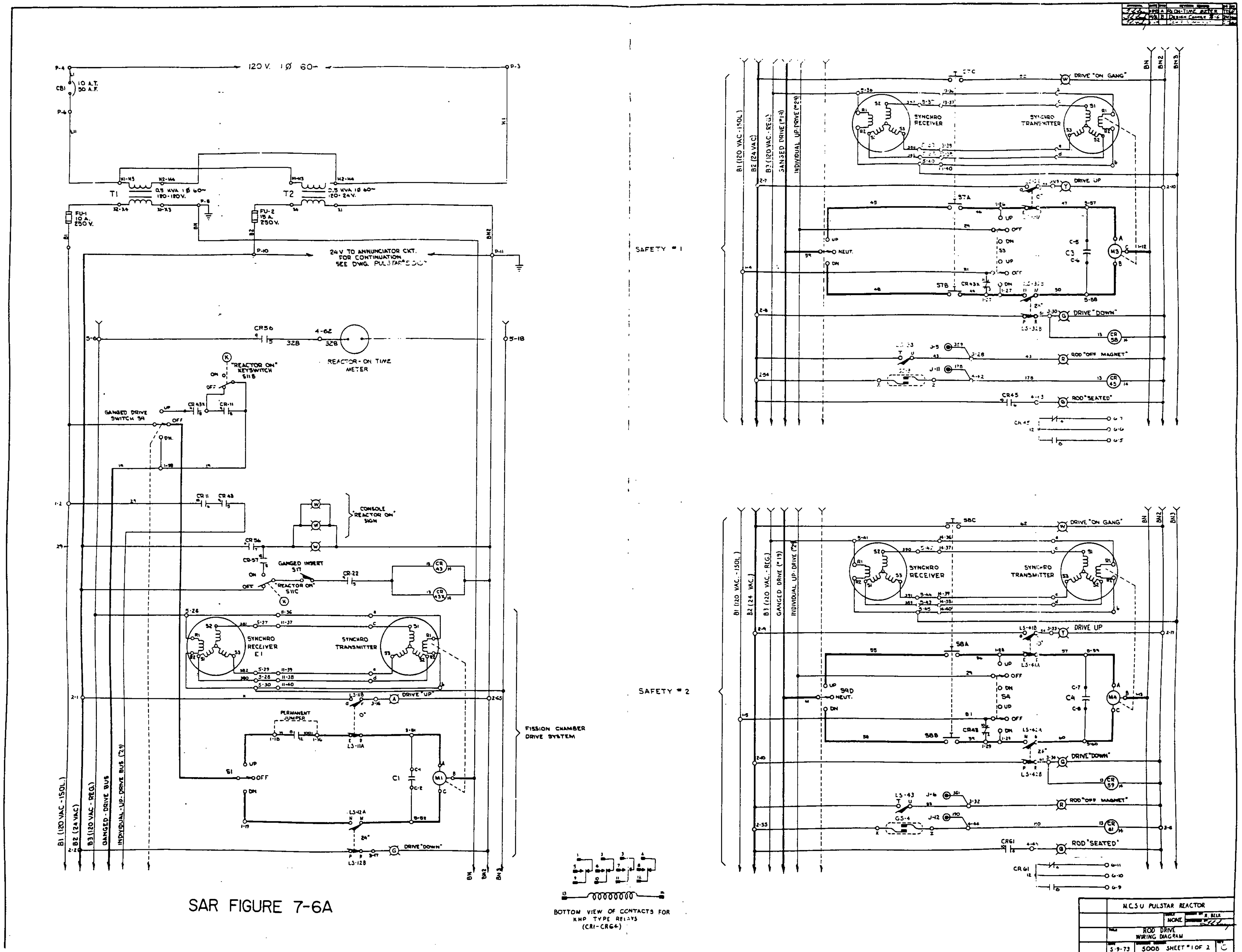
A - NI High Voltage Power Supply
B - Radiation Monitoring Recorder
C - Area Monitor (4)
D - Stack Sample Unit
E - Radiation Alarm Panel
F - Process Monitor (4)
G - Waste Tank Monitor (3)

PULSTAR Radiation Monitoring Rack



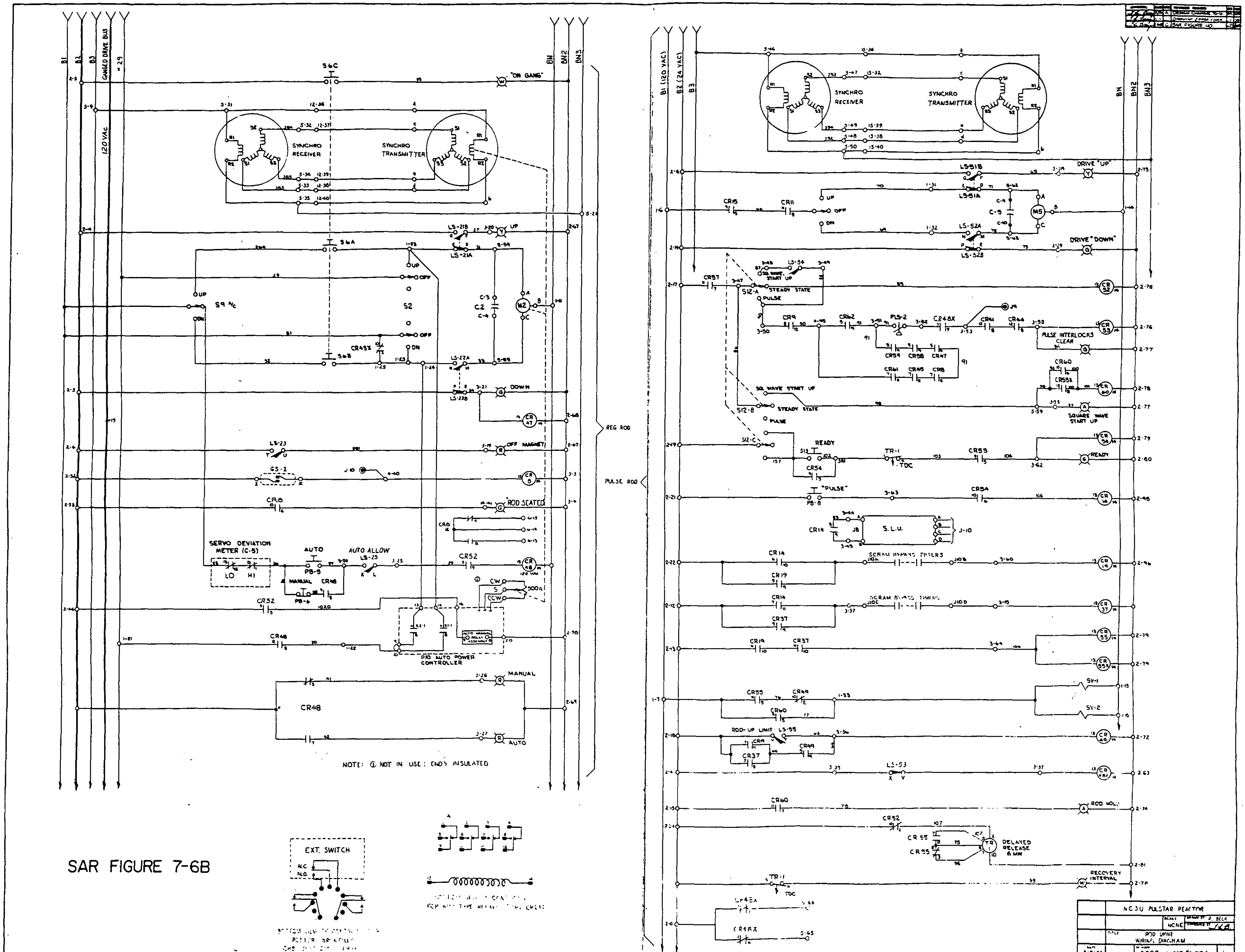
Control Rod Drive Interlock Diagram

Figure 7-5



SAR FIGURE 7-6A

9610090225-01



SAR FIGURE 7-6B

ANSTEC APERTURE CARD

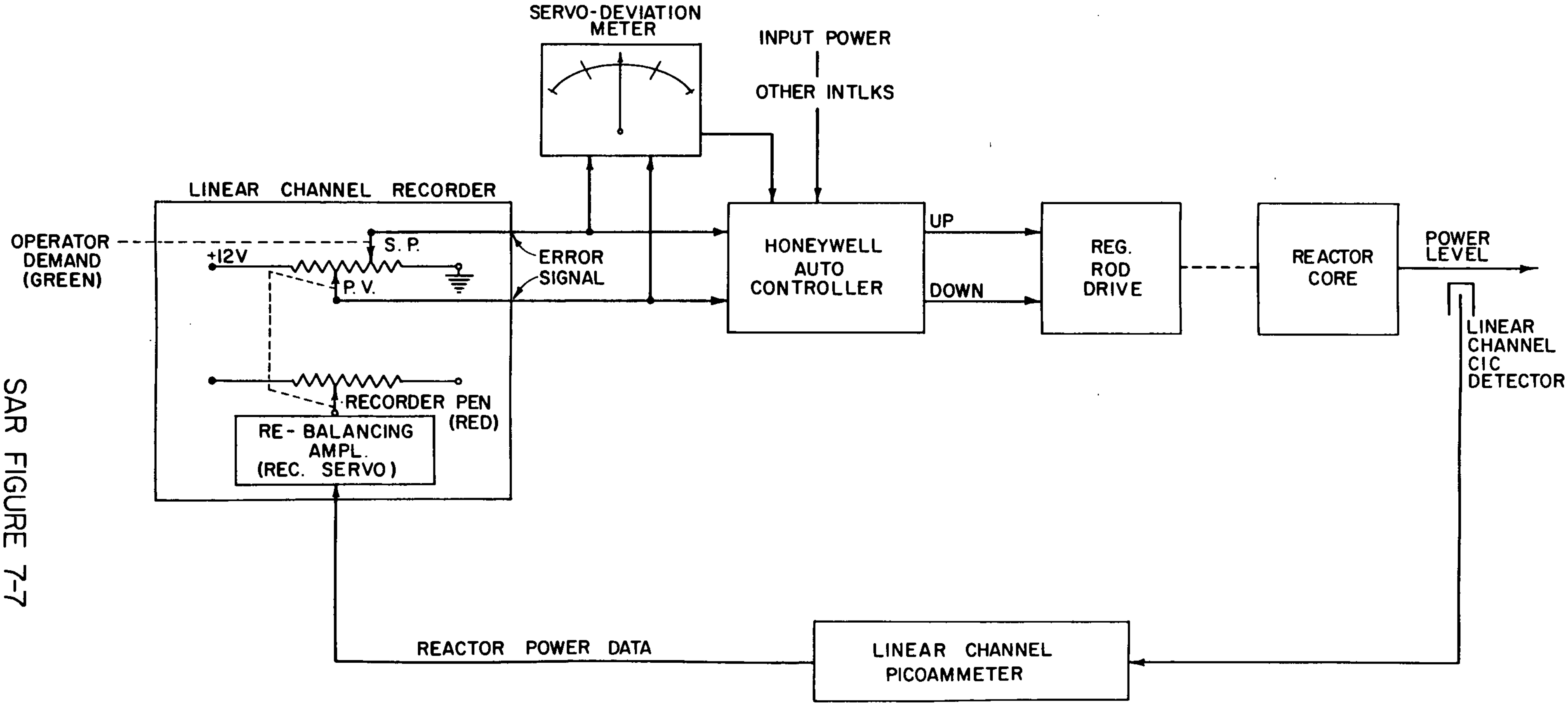
Also Available on Aperture Card

9610090225-02

APPROVAL	DATE	SYM	REVISION RECORD	DR	CK
[illegible]	2-13-74	A	AS BUILT	RB	[illegible]
[illegible]	2-22-74	B	CHANGE	RB	[illegible]
[illegible]	5-30-[illegible]	C	NO CHANGE	[illegible]	[illegible]

SAR FIGURE 7-7

OPERATIONS MANUAL FIG. 4.5



S. P. = SET POINT

P. V. = PROCESS VARIABLE (RX POWER)

TOLERANCES (EXCEPT AS NOTED)	N. C. S. U. PULSTAR REACTOR		
DECIMAL ±	FIG. 4.5	SCALE NONE	DRAWN BY R. BELK; APPROVED BY [illegible]
FRACTIONAL ±	TITLE AUTOMATIC POWER LOOP		
ANGULAR ±	DATE 2-13-74	DRAWING NUMBER 5130	REV C

8 ELECTRICAL SYSTEM

8.1 Introduction

The electrical power for Burlington Engineering Laboratories is supplied from the University distribution system. In the event that commercial power is lost, emergency lighting is supplied by backup batteries and selected radiation monitors are supplied by an Uninteruptible Power Supply. The PULSTAR Reactor is equipped with an auxiliary electric generator to ensure a more orderly shutdown in the event that commercial power is lost.

8.2 Commercial Electrical System

Burlington Engineering Laboratories (BEL) receives its power from the University 12 kV, three phase, 60 hertz underground electrical service. The voltage is stepped down to 480/277 volts by an outdoor, pad mounted, oil filled transformer located on the The transformer is connected to the switchgear in by a weatherproof bus. All feeder circuits in BEL originate from this point. The 400 ampere, three phase feeder provides electrical power to the Reactor Building Motor Control Center (MCC #1) and is used to de-energize electrical power to all reactor equipment in the event of fire or other emergency conditions. Refer to the electrical schematic diagram shown in Figure 8-1A.

8.3 Auxiliary Electrical System

Commercial 480 volt, 3 phase electrical power for the Control Console and for the two Confinement Fans is routed through special devices called Load Transfer Controls (LTC). Also routed to each LTC is 480 volt, three phase power from the 12 kW Auxiliary Generator located in Room 1103. The conduits distributing commercial power and generator power are routed for maximum practical separation to minimize a single event rendering both sources of power inoperable.

Commercial power is routed through each LTC's normally closed set of line contacts. Also connected to each LTC is a Transfer Control Switch which is mounted on the control room wall to the right of the Control Console. Specifically, the loads transferred by the switches are: Confinement Fan No. 1, Confinement Fan No. 2, and the Control Room Distribution Panel. Upon loss of commercial power, the generator must be started and one, two, or three Transfer Switches must be operated to allow the line contacts in each LTC to open and the generator contacts to close supplying electrical power from the generator to their respective loads. The line and generator contacts are mechanically interlocked so that only one set of contacts can be closed at a time. When commercial power is restored, the Control Room Distribution LTC instantly switches back to the line contacts regardless of the Transfer Switch position. The two Confinement Fan LTCs will switch back to the line contacts after approximately one to two minutes. It should be noted that only one of the Confinement Fans can operate at a time due to electrical interlocks in the fans' magnetic starters even though both LTCs will switch to the generator contacts.

8.3.1 Motor Control Center #1

Motor Control Center #1, located in the to PULSTAR Reactor equipment. MCC #1 has a manually operated 400 ampere, three phase, fused disconnect switch. All pump and fan motors supplied by MCC #1 are 480 volt, three phase, 60 Hertz induction motors.

Figure 8-1A shows MCC #1 and all of the fused disconnects and combination magnetic starter with fused disconnects for each piece of electrical equipment. Each magnetic starter has its own 480 to 120 volt stepdown transformer to operate the starter control and indicator lamp circuits. Figure 8-1A shows the location of the start/stop controls for the starter circuits. Refer to Table 8-1 for MCC #1 load list and ratings.

8.3.2 Motor Control Center #2B

Motor Control Center #2B, , supplies 480/277 volt power to the cooling tower fans and basin heater. Refer to Table 8-2 for load list and ratings.

8.3.3 Auxiliary Generator Control Panel

The auxiliary generator has its own control panel located inside the west end of the housing unit. This panel contains the generator voltmeter with a phase selector switch, 3 AC ammeters, a running time meter, and a frequency meter. The panel is also equipped with a water temperature gauge, an oil pressure gauge, and a battery charge-rate ammeter. The panel is equipped with 3 circuit breakers.

Mounted on the control panel is a "Run-Stop-Remote" switch. This switch allows the generator to be started locally, by placing the switch in "Run." By placing this switch in "Remote," the generator may be started remotely from the reactor console. This switch is normally in the "Remote" position.

The generator also has an emergency latch relay with reset and indicator light. This latch relay opens the starting and/or run circuitry of the generator if low oil pressure or high water temperature trips occur.

8.3.4 Auxiliary Distribution Panel

The Auxiliary Distribution Panel, located adjacent to the Auxiliary Generator, houses the feeder breakers that supply auxiliary power to the two confinement fans and the Control Room Distribution Panel. The panel also houses the control transformers that supply the coils used in the load transfer system and the generator power available light.

The Auxiliary Distribution Panel is connected to the Auxiliary Generator through an automatic transfer switch. This switch is spring loaded to remain in the open position. As the generator comes up to speed and voltage, the generator output voltage works against the spring tension to close the switch and apply power to the Auxiliary Distribution Panel.

Upon securing the Auxiliary Generator, the output voltage is removed and the transfer switches automatically opens. Refer to Figure 8.1B for the load distribution.

8.3.5 Auxiliary Panels

The Control Room Distribution Panel (CRDP) is located under the south window in the Control Room and receives power from a 480/120v transformer located just outside the MER below the Beam Tube Thermal Column Exhaust Fan. Refer to Figure 8-1B for load distribution.

The Reactor Bay Lighting Panel (RBLP) is located just outside the MER below the Beam Tube Thermal Column Exhaust Fan and receives power from MCC #1 through a 480/120v stepdown transformer located just outside the MER below the Beam Tube Thermal Column Exhaust Fan. Refer to Figure 8-1B for load distribution.

The Primary Piping Vault Panel (PPVP) is located in the Primary Piping Vault on the north wall and receives power from the RBLP breakers 25 and 27. The panel supplies power for the vault lighting, receptacles and sump pump. Refer to Figure 8-1B for load distribution.

All electrical work is installed in accordance with the latest revision of the National Electrical Code and applicable local codes.

TABLE 8-1
MCC #1 LOAD LIST

Primary Pump Motor

Location:	Southwest corner of Mechanical Equipment Room between secondary pump motor and sump pump motor.
Horsepower:	7-1/2
Operating Points:	1. Process Control Panel on Reactor Control Console 2. Motor Control Center #1 3. Locally at Motor 4. Reactor Bridge
Interlocks:	Local Lockout Device for maintenance

Secondary Pump Motor

Location:	South wall of Mechanical Equipment Room - East of Primary Pump Motor
Horsepower:	20
Operating Points:	1. Process Control Panel on Reactor Control Console 2. Motor Control Center #1 3. Locally at Motor
Interlocks:	Local Lockout Device for maintenance

Prim. Demin. Pump Motor

Location:	Near east end of heat exchanger in Mechanical Equipment Room
Horsepower:	1
Operating Points:	1. Process Control Panel of Reactor Control Console 2. Motor Control Center #1 3. Locally at Motor
Interlocks:	Lockout Device for maintenance

Service Demin. Pump Motor

Location:	East wall of mechanical Equipment Room near service water resin tanks
Horsepower:	1
Operating Points:	1. Motor Control Center #1 2. Locally at Motor
Interlocks:	Local Lockout Device for maintenance

Table 8-1 continued

Sump Pump Motor

	Location:	Above sump pit along west wall of Mechanical Equipment Room
	Horsepower:	3/4
	Operating Points:	1. Motor Control Center #1 2. Locally at Motor
	Interlocks:	Local Lockout Device for maintenance

Main Supply Fan Motor

	Location:	Southeast corner of the Mechanical Equipment Room
	Horsepower:	10
	Operating Points:	1. Process Control Panel on the Reactor Control Console (start/stop) 2. Automatically (stop) by Radiation Alarm Panel

Main Exhaust Fan Motor

	Location:	Northeast corner of the Mechanical Equipment Room
	Horsepower:	15
	Operating Points:	1. Process Control Panel on the Reactor Control Console (start/stop) 2. Automatically (stop) by Radiation Alarm Panel

Beam Tube and Thermal Column Exhaust Fan Motor

	Location:	Above the Reactor Building Lighting Transformer outside the Mechanical Equipment Room
	Horsepower:	1/2
	Operating Points:	1. Process Control Panel on the Reactor Control Console (start/stop) 2. Locally at Motor 3. Automatically (stop) by Radiation Alarm Panel

Table 8-1 continued

Control Room Air Conditioning Fan Motor

Location:	Above the Control Room Ceiling
Horsepower:	2
Operating Points:	1. Locally at Motor 2. Honeywell Control Panel near the Reactor Control Console (Start/Stop) 3. Automatically (stop) by Radiation Alarm Panel

Confinement Fan Motor

Location:	East wall of the Mechanical Equipment Room
Horsepower:	5
Operating Points:	1. Process Control Panel on the Reactor Control Console (start/stop) 2. Automatically (start) by Radiation Alarm Panel

TABLE 8-2
MCC #2B LOAD LIST

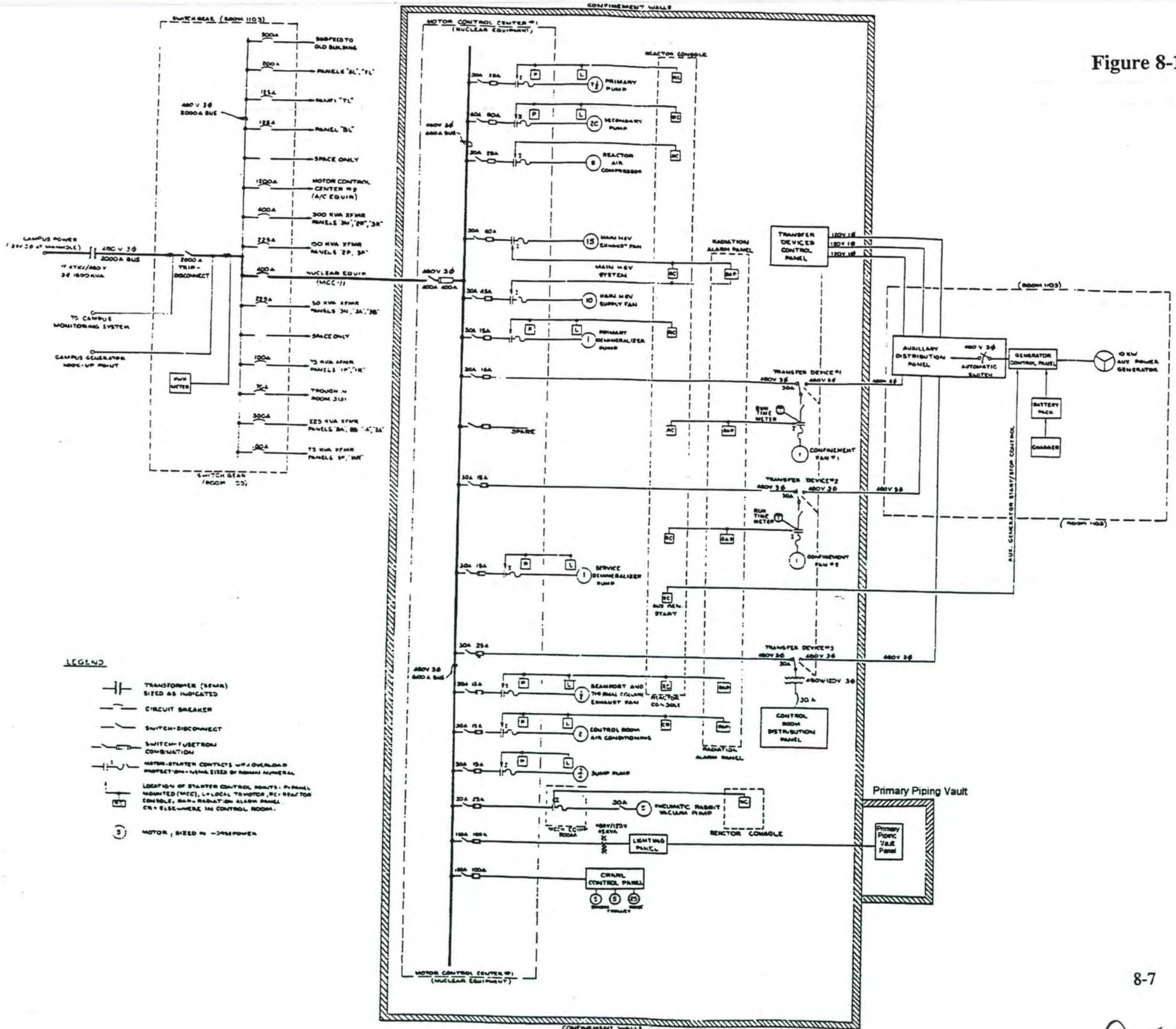
Cooling Tower Fan Motor

Location:	At the Reactor Cooling Tower on the East side of Burlington Labs.
Horsepower:	20
Operating Points:	1. MCC #2B 2. Locally at the Cooling Tower.
Interlocks:	None

Cooling Tower Basin Heater

Location:	In the Reactor Cooling Tower on the East side of Burlington Labs.
Operating Points:	1. MCC #2B 2. Locally at the Cooling Tower.
Interlocks:	None

Figure 8-1A



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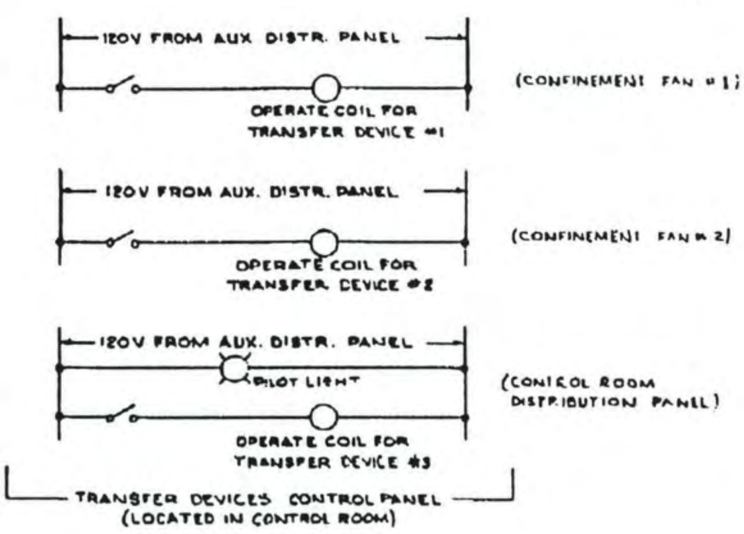
AUXILLARY DISTRIBUTION PANEL (LOCATED IN ROOM 1103)
(FORMERLY EMERGENCY DISTRIBUTION PANEL)

SERVICE	WATTS	CKT NO.			CKT NO.	WATTS	SERVICE
TO TRANSFER DEVICE CONTROL #1 (CONF. FAN #1)		1	480V		2		SPARE
SPARE		3		480/120V	4		TO TRANSFER DEVICE CONTROL #3 (CONTROL ROOM DIST PANEL)
SPARE		5		480/120V	6		TO TRANSFER DEVICE CONTROL #2 (CONF FAN #2)
TO TRANSFER DEVICE #1 (CONF. FAN #1)			15A	30A			TO TRANSFER DEVICE #3 (CONTROL RM DIST PANEL)
GENERATOR RADIATOR DAMPER MOTOR			15A	15A			TO TRANSFER DEVICE #2 (CONF. FAN #2)
SPACE							SPACE

A B C N

480V 3Ø 4W SERVICE FROM AUX. GENERATOR



CONTROL ROOM DISTRIBUTION PANEL
(FORMERLY GENERATOR DISTRIBUTION PANEL)

SERVICE	WATTS	CKT NO.			CKT NO.	WATTS	SERVICE
REACTOR CONSOLE (POWER)	1250	1	20	20	2		EMER LIGHTS EQUIP. RM - H/V AREA
SPARE		3	20	20	4		SPARE
REACTOR CONSOLE (INSTRUMENTATION)	1250	5	20	20	6		EMER LIGHTS EQUIP RM - PUMP AREA
EVACUATION HORNS & R.A.P.		7	20	20	8		RECEPTACLES (CONTROL RM)
STACK SAMPLER		9	20	20	10		STACK SAMPLER VACUUM MOTOR
EMER LIGHTS CONTROL ROOM		11	20	30	12		EMERG LIGHTS (MER)
AUX RACK		13	20	20	14		CONSOLE FANS
SPACE		15			16		SPACE
UNINTERUPTIBLE POWER SUPPLY		17	20	20	18		COMPUTER POWER SUPPLY

A B C N

120V 3Ø 4WIRE SERVICE FROM TRANSFER DEVICE #3

PRIMARY PIPING VAULT PANEL

SERVICE	WATTS	CKT NO.			CKT NO.	WATTS	SERVICE
LIGHTS		1	20		2		DELAY TANK SUMP PUMP
RECEPTACLE DELAY TANK		3	20		4		
RECEPTACLE EXIT SIGN		5	20	40	6		SPARE
SPACE		7			8		SPACE
SPACE		9			10		SPACE

A B C N

120V 3Ø 4WIRE SERVICE

LIGHTING PANEL SCHEDULE (LOCATED ON NORTH SIDE OF LOADING DOCK)

SERVICE	NO. OF OUTLETS	WATTS	CKT. NO.			CKT. NO.	WATTS	NO. OF OUTLETS	
LTG-REACTOR BAY	2	2000	1	20	20	2	1800	7	LTG-CONTROL RM
LTG-REACTOR BAY	2	2000	3	20	20	4	1300	7	LTG-CONTROL RM
LTG-REACTOR BAY	3	1200	5	20	20	6	1000	5	LTG-EQUIP. RM
LTG-REACTOR BAY	4	1600	7	20	20	8	900	3	LTG-EQUIP RMS.
ROLL-KLEEN FILTER MOTOR	1	530VA	9	20	20	10	450VA	1	BLDG HAV COMP AFTER COOLER
REACTOR AIR COMPRESSOR DRYER	1	1340 VA	11	20	30	12	700	4	RECEPTACLES-UNDER CATWALK
RECEPTACLES-EQUIP RM	9	1600	13	20	20	14	—	—	RECEPTACLES-ON BRIDGE
RECEPTACLES-REACTOR BAY	4	1500	15	20	20	16		1	SHIELD RECEPTACLES
RECEPTACLES-REACTOR BAY	4	1500	17	30	20	18		1	SHIELD RECEPTACLES T/C LIGHTS
RECEPTACLES-CONTROL RM	—	—	19	20	20	20	—	—	LTG- REACTOR BAY MERCURY
LTG. REACTOR BAY	—	—	21	20	20	22	—	—	LTG- REACTOR BAY MERCURY
RABBIT CONTROL VOLTAGE	1	1800	23	20	20	24	—	—	LTG- REACTOR BAY MERCURY
PRIMARY PIPING VAULT			25	50	15	26			REACTOR BAY HOOD FAN
			27			28			
SPACE			29			30			

A B C N

120/208 3Ø 60~ 4 WIRE SERVICE FROM MOTOR CONTROL CENTER #1

Figure 8-1B

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9 AUXILIARY SYSTEMS

9.1 Service Systems

The PULSTAR Reactor utilizes several auxiliary systems for routine and emergency operations. These are Reactor Air, Service Water, Reactor Bay Raw Water, Beam Tube and Thermal Column Ventilation, Radioactive Liquid Waste, Personnel Warning, Communications, Auxiliary Electrical, and Reactor Bay Crane.

9.1.1 Reactor Air

Reactor Air is supplied by a standard air compressor and moisture separator located in the Mechanical Equipment Room (MER). The 100 psi (690 kPa) air supply is used for all of the pneumatic instrumentation (pool level and primary flow) and all of the ventilation control dampers within the Reactor Building. A high pressure nitrogen tank and regulator is connected by a valve to the air supply line so that reactor operations can continue for approximately one half hour if the air compressor fails. There is also a method for valving the Burlington Engineering Laboratories (BEL) air compressor to the air supply line to operate the ventilation control dampers. See Figure 9-1 for details.

9.1.2 Service Water

A source of high purity demineralized water is available in the MER and in the valve trench around the biological shield. The Service Water System supplies demineralized organic-free water for make-up to the Primary System, filling beam tubes, and for other uses requiring pure water. See figure 4-1D for system drawing.

The major components of the service water system are a close-coupled stainless steel centrifugal pump with a capacity of 10 gpm (0.6 ℓs^{-1}) circulation pump, a balance tank constructed of stainless steel with a capacity of approximately 70 gallons (265 ℓ), a demineralizer package, water meter, and associated piping and valves.

The demineralizer package is comprised of an inlet filter, a charcoal bed, resin bed, an outlet filter, flow meter, and a resistivity cell with a readout module. The inlet and outlet filters are essentially identical. The inlet filter has six replaceable cartridges, while the outlet filter has only three. Normally, 25 micron filter cartridges will be installed in both filters. The carbon filter is a stainless steel closed cylinder with an exterior PVC coating. The column holds approximately three cubic feet of washed, activated carbon and is used to remove organic materials from the water. The demineralizer column is identical in construction to the carbon filter with an H-OH mixed bed resin. The water is purified by an ion exchange process. Positive impurity ions are exchanged for H^+ ions and negative impurity ions are exchanged for OH^- ions. This process yields demineralizer effluent pure water with a pH of about 7.0.

During normal operation, water is recirculated through the service water system by the pump at a nominal 3 gpm (0.2 ℓs^{-1}). The two filters, charcoal bed, and resin bed establish and maintain the purity of the water in the system. Pure water may be withdrawn from the system by use of a hose attached to any of ten quick-disconnect fittings.

9.1.3 Reactor Bay Raw Water System

The Reactor Bay Raw Water System supplies make-up water to the Secondary System and raw water for utility use in the Reactor Bay area. The Reactor Bay Raw Water system consists of two in-line check valves, a three-valve manifold, a pressure regulating valve, and connections to the service water and Secondary Systems. The pressure regulating valve reduces the raw water pressure from city water pressure to 30 psig (308 kPa) for use in the Service Water System. See figure 4-1D for system drawing.

9.1.4 Beam Tube and Thermal Column Ventilation

The vent lines from the Beam Tubes and the Thermal Column are connected to a header through normally open valves around the perimeter of the first lift of the Biological Shield. This in turn is connected to a HEPA filter and the suction of the Beam Tube and Thermal Column Exhaust Fan. Just prior to the HEPA filter is a water separator and drain leg with a "S" trap.

9.1.5 Radioactive Waste

All liquid waste in BEL that is potentially radioactive collects in the sump in the MER. Liquid from the sinks and the floor drains of some of the third floor laboratories, all of the basement nuclear laboratories, and the Reactor Building, including all vents and drains on the Primary and Secondary coolant pipes and the Service Water in the Reactor Building, flow to the sump by gravity drain lines. As the sump fills, the pump cycles to transfer the liquid waste to the three holding tanks located in an underground vault near the southwest corner of the Reactor Building. There it is held for sampling, monitoring and if necessary, dilution before being discharged into the sanitary sewer system. See Figure 4-1E for details of the liquid waste handling system.

Radioactive gases are filtered and monitored before being discharged to the atmosphere from the Reactor Building through the 100 foot (30.5 m) high stack (see Figure 5-1). Radioactive solids are collected, monitored, dried, and shipped out for burial. All radioactive waste discharges and shipments are controlled (as described in Section 10) so that they are within the limits prescribed by 10 CFR 20, 10 CFR 61, 10 CFR 70 and other pertinent regulations.

9.1.6 Personnel Warning

There are two conditions that require warning building personnel of danger. The first is fire. In case of fire heat sensors located in the Reactor Control Room, MER, or south wing

of Burlington Nuclear Laboratories (BEL), will automatically activate the Fire Alarm System prompting a response from the City of Raleigh Fire Department. In the north wing of Burlington Nuclear Laboratories, manual pull stations must be activated to initiate the alarm utilizing two types of bells. The first type is located at various places on each floor of the main wing and will chime for four coding sequences and then go silent. The other bell is located in the central stairway and will continue to chime until it is reset by Public Safety personnel.

The other condition is building evacuation in response to radiological reasons. A cyclic horn indicates that personnel must leave the Reactor Building by the proper evacuation route, and if the situation demands, the entire BEL complex may be evacuated.

9.1.7 Communications

Several methods of communication are available to personnel in the Reactor Building. Telephones are located in the Control Room and on the floor level of the Reactor Bay. An intercom connects the Control Room with the Associate Director, the Reactor Operations Manager, Operations, the Health Physicist, the Reactor Bridge, the Mechanical Equipment Room, the Primary Piping Vault, the Change Room, and the Radiochemistry Lab. A public address system is available in the Control Room that can be heard in the MER, the Reactor Bay, and the basement laboratory hallway. There are five portable 4 watt UHF transceivers that operate on a clear frequency in the Raleigh area, available to operations personnel. These radios are cycled between a charging mount on the south wall of the PULSTAR Control Room and the emergency equipment locker located in the foyer. A battery powered bull horn kept in the Control Room is also available for use as required.

9.1.8 Auxiliary Electrical System

Electric power is available for the Control Room and the two Confinement Fans from a 12 kW standby generator. See Section 5, and Section 8 for a detailed description.

9.1.9 Emergency Lighting

Two types of battery powered emergency lighting have been installed in various locations throughout the reactor building to assist personnel in the event of loss of lighting. The lights sense the loss of power and automatically energize. Two emergency lighting fixtures, one in the control room and the other in the reactor bay are powered from the BEL complex. These lights will remain energized for approximately 45 minutes upon loss of power to the entire BEL. In order to supply emergency lighting due to a loss of power to only the reactor building, and to provide additional personnel safety, three self-contained emergency lights have been installed. The self-contained batteries are designed to last a minimum of 90 minutes.

9.1.10 Reactor Bay Crane

The Reactor Bay Crane, a ten ton capacity unit, is used to handle heavy equipment within the reactor bay. All drive systems are electrically powered, floor operated, and are equipped with positive self-locking devices to prevent motion in the event of power failure. The crane is driven by three separate motors. One motor is used to drive the bridge assembly, another motor to operate the trolley and a third motor to raise and lower the hoist. The bridge motor drives the travelling crane at a maximum speed of 175 fpm (0.89 m/s). The unit has 3 speeds and is equipped with travel limit switches. The trolley motor drives the hoist assembly from north to south on the bridge at a maximum speed of 125 fpm (0.64 m/s) The trolley controls are 5-speed and are equipped with travel limit switches. The hoist motor has a maximum speed of 40 fpm (0.2 m/s) and is equipped with 5-speed controls. The crane receives power from MCC #1 through a fused disconnect switch. All the controls for the crane are mounted in a pushbutton station that is suspended by cable and chain from the trolley. The height of the controls is adjustable between the Reactor Bridge and Bay floor.

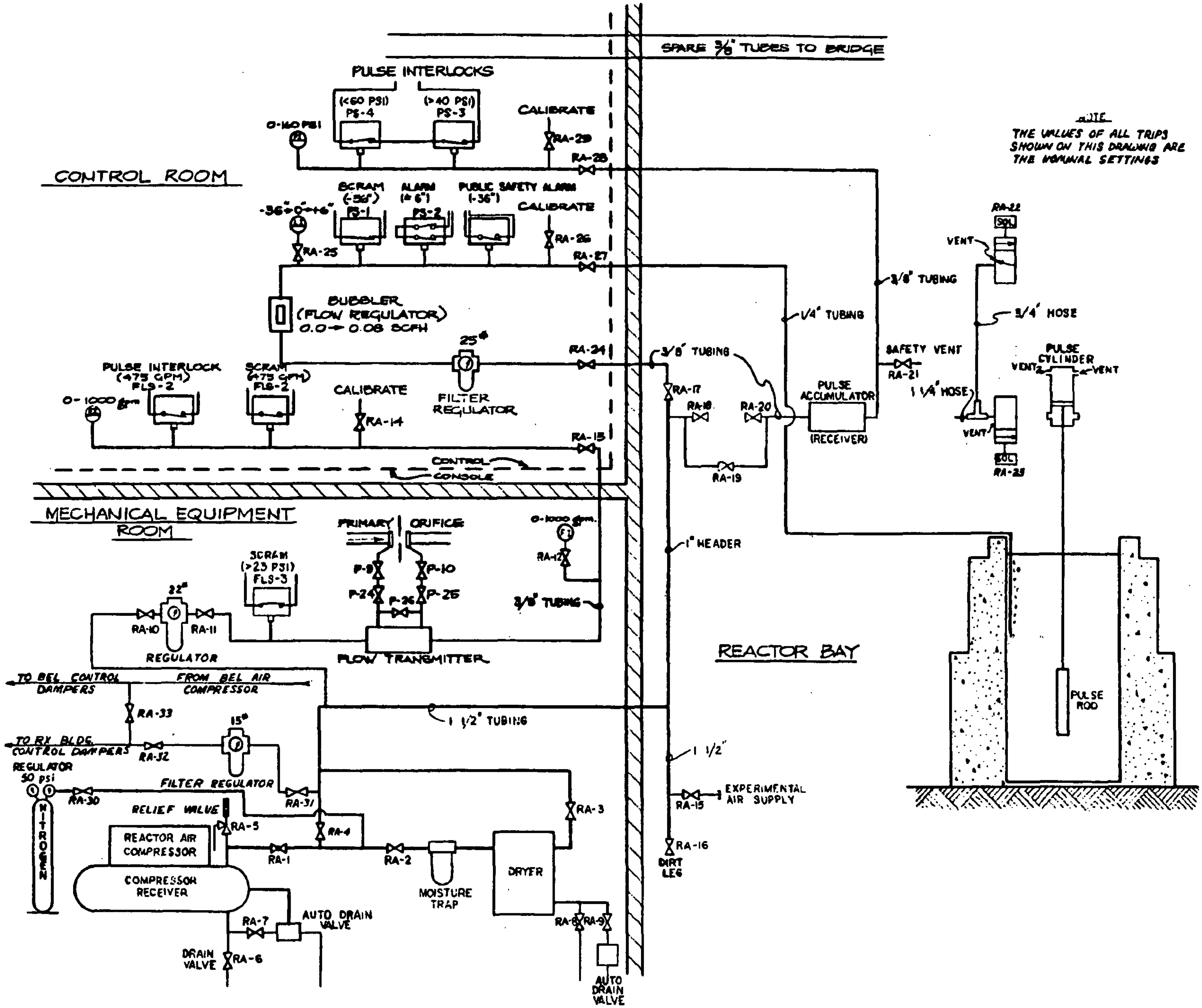

SPARE 3/8" TUBES TO BRIDGE
PULSE INTERLOCKS
(<60 PSI) PS-4
(>40 PSI) PS-3
0-160 PSI
CALIBRATE
RA-29
RA-28
CONTROL ROOM
SCRAM (-36") PS-1
ALARM (±6") PS-2
PUBLIC SAFETY ALARM (-36")
CALIBRATE
-36"←0→+6"
RA-25
RA-26
RA-27
BUBBLER (FLOW REGULATOR) 0.0 → 0.08 SCFH
25#
FILTER REGULATOR
RA-24
PULSE INTERLOCK (475 GPM) FLS-2
SCRAM (475 GPM) FLS-2
0-1000 gpm
CALIBRATE
RA-14
RA-15
CONTROL CONSOLE
MECHANICAL EQUIPMENT ROOM
PRIMARY ORIFICE
P-9
P-10
P-24
P-26
P-25
0-1000 gpm
RA-12
3/8" TUBING
SCRAM (>23 PSI) FLS-3
22#
RA-10
RA-11
REGULATOR
FLOW TRANSMITTER
TO BEL CONTROL DAMPERS
FROM BEL AIR COMPRESSOR
RA-33
15#
TO RX BLDG. CONTROL DAMPERS
RA-32
REGULATOR 50 psi
FILTER REGULATOR
RA-30
RA-31
1 1/2" TUBING
RELIEF VALVE
RA-5
RA-4
RA-3
NITROGEN
REACTOR AIR COMPRESSOR
COMPRESSOR RECEIVER
RA-1
RA-2
MOISTURE TRAP
DRYER
RA-7
AUTO DRAIN VALVE
DRAIN VALVE
RA-6
RA-8
RA-9
AUTO DRAIN VALVE
3/8" TUBING
RA-17
RA-18
RA-20
RA-19
1/4" TUBING
3/8" TUBING
PULSE ACCUMULATOR (RECEIVER)
SAFETY VENT
RA-21
1" HEADER
1 1/2"
RA-15
EXPERIMENTAL AIR SUPPLY
RA-16
DIRT LEG
REACTOR BAY
NOTE
THE VALUES OF ALL TRIPS SHOWN ON THIS DRAWING ARE THE NOMINAL SETTINGS
RA-22
SOL
VENT
3/4" HOSE
PULSE CYLINDER
VENT
1 1/4" HOSE
RA-23
PULSE ROD

Figure 9-1

10.0 RADIOACTIVE WASTES AND RADIATION PROTECTION

10.1 Radioactive Wastes

10.1.1 Design Bases

Radioactive wastes from the use of a reactor facility for teaching and research purposes are slight compared to those of a test reactor, nuclear power reactor, or fuel processing facility. Low level liquid wastes are usually from draining beam tubes, primary pump seal leakage, radiochemical by-products, or decontamination activities. Solid wastes consist mostly of de-watered resins, filter media, discarded small items, and irradiated materials on occasion. No outstanding sources of gaseous wastes exist except the usual production of Argon-41 by neutron activation of air. Design bases for radioactive wastes at the PULSTAR reactor include a liquid waste handling system and stack effluent monitoring equipment. Efforts to minimize the generation of wastes include corrosion control, demineralization of service water and primary water, identification and repair of leaking components, nitrogen purge of the Pneumatic System when it is not in use, minimization on use and decontamination of materials, and review of proposed experimental projects by Operations and Health Physics. Experimental project reviews include estimates of the types and amount of waste expected, what administrative controls and design features are to be used to minimize waste generation, and waste disposal methods (refer to Section 14).

10.1.2 Liquid Waste Handling

Radioactive liquids from the Reactor Building and from associated nuclear laboratories located in the Burlington Engineering Laboratory (BEL) are collected by the liquid waste handling system and are estimated at about 12,000 gallons (4.5×10^4 ℓ) per year. This system is depicted in Figure 4-1E of this document. Sources of low level liquid waste in the Reactor Building are from such operations as beam tube draining, flushing the primary demineralizer resins, and leaking components. Liquid waste from the nuclear laboratories will consist of low level liquids from cleaning contaminated glassware or similar items since authorization to use radioactive material by the North Carolina State University Radiation Protection Committee (RPC) is issued with the provision that no high level or biological wastes be placed in the drain system. Rather, high level liquid or biological wastes are stored for collection by the University Radiation Protection Division (RPD) and prepared for disposal in accordance with 10 CFR 20, 10 CFR 61, and other applicable regulations.

The liquid waste from the Reactor Building and associated nuclear laboratories are collected in the sump in the floor of the Mechanical Equipment Room (MER). Another sump is located in the lower elevation of the Primary Piping Vault. Both sumps pump liquid waste to the waste handling tanks located outside the Reactor Building in the waste tank vault. Each waste tank is monitored by a gamma scintillation detector. These detectors indicate the gross activity of the waste water and provide an alarm in the PULSTAR Reactor Control Room if their set point is exceeded. Refer to Figure 10-1 for details of the waste tank radiation monitoring instrumentation. Liquid level indicators are also equipped for each tank and provide remote indication at the BEL.

Valve controls for filling the liquid waste tanks and releasing their contents are located at the BEL. These controls are under the administrative control of the Reactor Health Physicist (RHP) who keeps them locked to prevent inadvertent operation. Liquid waste tank contents are circulated by an external pump arrangement prior to sampling to ensure a representative sample is obtained for analysis. Releases take place only after the contents of the tank have been analyzed to determine that the total radioactivity and concentration present in the waste liquid are within appropriate regulatory limits. Analysis include gross beta-gamma concentration, tritium concentration, and concentration of principle gamma emitting nuclides. The gross beta-gamma concentration shall be less than 2×10^{-5} μCi/ml as was determined from the minimum average daily campus water use volume of 671,000 gal per day and an assumed daily liquid waste discharge volume of 3300 gallons:

$$2\times10^{-5}\mu Ci/ml \approx 1\times10^{-7}\mu Ci/ml \times \frac{671{,}000gal}{3300gal}$$

where, 1×10^{-7} μCi/ml is taken from 10 CFR 20 Appendix B Table 3
671,000 gal per day is taken from Reference 10-8

Isolation of liquids discharged from the Reactor Building may be accomplished, if necessary, by either of two valves located between the sump and the three waste tanks or by disconnecting power to the sump. The sump in the Primary Piping Vault is operated only when needed.

10.1.3 Solid Waste Disposal

All radioactive solids from the Reactor Building and associated nuclear laboratories are collected, monitored, and stored for transfer to the RPD for disposal. Solid waste consist of de-watered resins, filter media, and dry active waste such as discarded small items and irradiated materials which are no longer useful. Average annual volume of solid waste based on previous operational history is approximated at 15 cubic feet of de-watered resins and 30 cubic feet of compacted dry active waste. Compaction of dry active waste is performed by the RPD. Disposal is in accordance with RPD policies, 10 CFR 20, 10 CFR 61, and State of North Carolina Regulations for Protection Against Radiation.

10.1.4 Gaseous Waste Processing

Gaseous waste processing is discussed in Section 5.

10.2 Radiation Protection

10.2.1 Shielding and Corrosion Control

Dose rates in excess of 2.5 mrem/h (6.9×10^{-9} Sv/s) do not exist at any point on the concrete Biological Shield at a power level of 1 MW by design. In accordance with this criterion, and
above the Reactor Bay floor are employed for shielding in a horizontal

direction. From above the floor, this horizontal shielding is reduced to . The ordinary concrete is further reduced to . Technical Specification Limiting Safety System Setting for water level above the reactor core is 14 feet and 2 inches (4.3 m) which is associated the most limiting anticipated transient of a loss of forced convection flow. Therefore this represents a minimum water cover above the reactor core available for radiation shielding.

The Reactor Building wall and roof materials provide sufficient shielding to meet dose limits for individual members of the public given in 10 CFR 20.

The RHP performs routine radiation surveys to verify dose rates near the biological shield wall and other areas within the Reactor Building and adjacent areas outside the reactor building are within regulatory and design limits. Within the Reactor Building, local shielding may be used as specified by the RHP. Shield surveys were part of the initial reactor testing phase and are discussed in Section 12.

With adequate demineralization, the principal activity in the primary water leaving the core is caused by Nitrogen-16 (^{16}N) activity. In the forced circulation mode of cooling, a delay tank in the primary system allows for decay of ^{16}N. This delay tank is located in the buried Primary Piping Vault (PPV). Radiation dose rates in unrestricted areas outside the PPV with the reactor in the forced convection cooling mode at full power have been measured up to a net value of 0.04 mrem/h. However, most unrestricted areas outside the PPV are at normal background radiation levels. Typically an occupancy factor of 1/16 is used to assess dose to members of the public in areas subject to pedestrian and vehicular traffic[10-9]. Based on this occupancy factor and the measured dose rate, an annual dose of 22 mrem to an individual member of the public is calculated from continuous operation at 1 MW, or 8760 MW·h per year. Therefore, sufficienct shielding is in place around the PPV to meet dose limits for individual members of the public given in 10 CFR 20.

To decrease the generation of radioactive materials associated with water impurities, a Service Water demineralizer is used to treat the city water before it is added to the pool. Impurities present in the primary water are controlled by a bypass demineralizer (refer to Section 4) where a portion of the primary flow is filtered and passed through a mixed bed ion exchange column. Corrosion control is accomplished by these demineralizer systems, which maintain the resistivity of the water greater than 500 kohm·cm. Very low levels of radioactivity normally exist downstream from the delay tank in the Primary Coolant System. The primary demineralizer accumulates much of this radioactivity. Replacement of the primary demineralizer resin is performed if the pool resistivity meter indicates less than 1 Mohm·cm.

Primary water is periodically sampled to determine the pH, resistivity, activity concentration, and effectiveness of corrosion control. pH is maintained between 5.5 and 7.5. Resistivity is maintained greater than 500 kohm·cm. Activity concentration is determined gross beta analysis of the dried residue of a one liter sample or by gamma spectroscopy of a liquid

sample. The effectiveness of corrosion control is performed by neutron activation analysis of an aliquot of primary water and comparison to a known standard solution. Also, the effectiveness corrosion control and fuel integrity are determined by the detection of unusual radionuclides by gamma spectroscopy.

Secondary water is periodically sampled to detect the presence of activated corrosion products, activated coolant products, or fission products by gamma spectroscopy. A heat exchanger leak would be indicated if any of these radioactive nuclides are detected.

10.2.2 Installed Radiation Monitoring Instrumentation

The Reactor Bay area and stack radiation monitors that can automatically initiate confinement are described in Section 5. There are several other additional installed radiation monitors. An ion chamber monitor is located next to the primary demineralizer and provides local indication at the Mechanical Equipment Room entrance and remote indication, alarms, and annunciation in the Control Room. A monitor using a GM tube is located adjacent to the roughing filters in the main exhaust system and provides remote indication, alarms, and annunciation in the Control Room. Each liquid waste tank is monitored by a gamma sensitive scintillation detector and provides remote indication and alarms for each waste tank in the Control Room. (Refer to Section 5 and Figure 10-1). Set points bases for installed radiation monitoring instrumentation are given in Table 10-1.

Other radiation monitors with local indication and alarms may be placed within the Reactor Bay for general area monitoring to supplement the installed radiation monitoring system. These monitors may be moved based on operational needs by the RHP. A constant airborne particulate monitor located in the Reactor Bay may be used to assess airborne particulate activity. Other gamma-sensitive instruments or air monitors may be placed at locations specified by the RHP as well. Maintenance and by-pass of the installed radiation monitors which initiate confinement is described in Section 5. Calibration, operation, and maintenance of installed radiation monitoring instrumentation and other radiation monitors are performed periodically using approved procedures or vendor manuals.

10.2.3 Access

Personnel access to the Reactor Building laboratories is controlled by
; doors. Normal access for personnel to the Reactor Bay is through the Control Room and Northwest doors. Personnel monitoring stations are located near these two doors. There are additional doors leading to and from the Reactor Building. Personnel access through these doors is controlled by the RHP or Reactor Operations personnel. These doors are used for momentary equipment transport and in emergency situations as required. Monitoring of personnel and materials is required prior to exiting through these doors. Because of potentially High Radiation Areas as defined in 10 CFR 20, personnel access to the MER and Primary Piping Vault may be controlled by locked doors or gates.

10.2.4 Health Physics

The RHP develops, maintains, and implements the PULSTAR radiation protection program approved by the University RPC at the reactor facility. This program meets the requirements given in 10 CFR 20 and emphasizes the ALARA (As Low As Reasonably Achievable) philosophy in all areas of operation that affect radiation safety at the reactor facility.

Table 10-1
Installed Radiation Monitoring Instrumentation Set Point Bases

Radiation Monitor	Set Point Bases
Control Room	Alert: Restricted area dose rate limit. Alarm: Habitation of the Control Room by occupational personnel.
Over the Pool	Alert: Abnormal radiation level from noble gas or external source. Alarm: High Radiation Area as defined in 10 CFR 20.
West Wall	Alert: Abnormal radiation level from noble gas or external source. Alarm: High Radiation Area as defined in 10 CFR 20.
Deminer-alizer	Alert: High Radiation Area as defined in 10 CFR 20. Alarm: Greater than Alert.
Stack Gas	Alert: Fraction of the alarm set point. Alarm: Fraction of the value associated with the "Notification of Unusual Event" emergency classification.
Stack Particulate	Alert: Fraction of the alarm set point. Alarm: Fraction of the value associated with the "Notification of Unusual Event" emergency classification.
Auxiliary GM	Alert: Fraction of the alarm set point. Alarm: Fraction of the value associated with the "Notification of Unusual Event" emergency classification.
Filter GM	Alert: Fraction of the alarm set point. Alarm: Fraction of the value associated with the "Notification of Unusual Event" emergency classification.
Waste Tanks (3)	No bases is given because these monitors only provide trending information. Alert is set as a fraction of the alarm.

The RHP is located in the BEL and is the primary individual responsible for radiation safety and radiological controls at the reactor facility. Reactor Operations and all other trained individuals are responsible for following radiation safety policies and procedures.

An extensive inventory of personnel dosimeters, portable GM, ion chamber, neutron, and charged particle detectors are maintained by the RHP. Facilities and equipment for counting smear samples, performing air and water radioassays, calibrating instruments, and initiating and controlling radiological emergency actions are available. Gamma spectroscopy analysis of radioactive contamination is available in the Neutron Activation Analysis Laboratory. Health Physics at the reactor facility is assisted by the University RPD. The RPD may perform confirmatory surveys and informs the RHP about results of such surveys. The RPD also provides personnel monitoring services, schedules special bioassay examinations as necessary, and performs portable instrument calibrations as necessary.

A Change Room, with a showers and personnel and clothing decontamination equipment is maintained to handle traffic to and from the reactor facility. Personnel monitoring stations are located in close proximity to the Change Room exit.

An emergency plan has been developed and is on file with the Nuclear Regulatory Commission to insure the proper response of all personnel in case of an emergency. The necessary radiological and support equipment is maintained in a state of readiness as described in the approved Emergency Plan.

10.2.5 Personnel Dose

Annual personnel dose history as monitored from external sources to occupational personnel employed at the reactor facility are given in Table 10-2. As indicated in Table 10-2, most of the external radiation dose to personnel has been monitored as less than 0.1 rem. External dose for calendar year periods yield similar results.

Table 10-2
Annual External Personnel Dose Histories

Whole Body Dose (rem)	Number of Individuals in Each Range									
	FY 85-86	FY 86-87	FY 87-88	FY 88-89	FY 89-90	FY 90-91	FY 91-92	FY 92-93	FY 93-94	FY 94-95
Not Detected	16	1	3	2	5	2	15	9	1	11
Less than 0.1	10	28	25	27	22	22	8	14	22	14
0.1 to 0.25	1				1	1			2	
0.25 to 0.50	1									
More than 0.50										

NOTE: Whole Body dose is deep dose-equivalent and FY is Fiscal Year

Monitoring of internal dose from intakes of radioactive materials has not been required. However, internal dose is estimated to be negligible based on air sampling and bioassay results. Air sampling results have indicated only low concentrations on a few occasions and bioassay results have not indicated retention of any radioactivity by any individual worker. Personnel dose monitoring records from external sources have also been reviewed for other dose categories. Dose to the skin and lens of the eyes are essentially equal to the whole body dose. Extremity dose also has been typically below 0.5 rem. Therefore, all personnel doses have been demonstrated to be well within annual regulatory limits.

10.3 Argon-41 and Nitrogen-16

This section describes the levels of Argon-41 and Nitrogen-16 that have been documented during reactor operations.

10.3.1 Nitrogen-16

10.3.1.1 Forced Convection Flow

The dose rate of ^{16}N traveling through the pool outlet piping with the reactor at full power and primary system at rated flow was measured at 8 rem/h (2.2 x 10^{-5} Sv/s) on contact. This measurement was taken in the valve pit adjacent to the biological shield. Normally this area is not accessible due to installed shield plugs. These shield plugs can only be removed with the overhead crane. Radiation levels from ^{16}N in the Primary Piping Vault range from a few mrem/h (2.8 x 10^{-9} Sv/s) at the vault access point in the Reactor Bay to 2 rem/h (5.6 x 10^{-6} Sv/s) adjacent to the delay tank. Supplemental shielding has been erected to reduce dose rates inside the Primary Piping Vault and to keep dose rates within 10 CFR 20 limits for unrestricted areas. High Radiation Area controls as required by 10 CFR 20 are implemented for personnel entry into the Primary Piping Vault while at power.

^{16}N is carried downward through the core region, core plenum, and outlet piping assembly to the ^{16}N Delay Tank while the reactor is operating with forced convection flow. At this point internal baffling of the Delay Tank allows sufficient time for the primary coolant to decay. The transit time of the primary coolant in this section of the primary piping is at least 1 minute, allowing for more than 8 half-lives of ^{16}N decay. Measurements taken during extended full power operations with standard survey instruments indicate radiation levels up to 2 rem/h (5.6 x 10^{-6} Sv/s) from ^{16}N activity in the Primary Piping Vault and up to 5 mrem/h (1.4 x 10^{-8} Sv/s) from other coolant activation products inside the MER.

At 95% power, typical dose rate over the pool during forced flow is approximately 1 mrem/h and originates primarily from the fission process with a minor contribution from ^{41}Ar. The ^{16}N that originates around the periphery of the core, which is not drawn into the core and the outlet piping, migrates about the core and slowly rises, decaying as it goes, and does not contribute significantly to the pool surface dose rate.

Radiation measurements taken underwater at 95% power with forced convection flow conditions at 14 feet 2 inches (4.3 m) were 110 mrem/h (3.1×10^{-7} Sv/s). If the pool level was reduced to 14 feet 2 inches (4.3 m) under forced convection flow at 95% power, the estimated dose rate at the normal pool level of 21 feet (6.4 m) above the centerline of the reactor core (indicated 0 inches), would be 50 mrem/h (1.4×10^{-7} Sv/s). Note that the underwater measurements used to provide this extrapolated value are conservative since back-scattered radiation from above the 14 feet 2 inch (4.3 m) level was detected. Limiting Safety System Setting for water height above the reactor core under conditions of forced convection flow is 14 feet 2 inches (4.3 m). With the Safety System Setting (SSS) for pool level at 17 feet (5.2 m) radiological conditions associated ^{16}N and ^{41}Ar are manageable.

10.3.1.2 Natural Convection Flow

In natural convection flow at 10% power the pool surface dose rate is due to the fission process and coolant activation products including ^{16}N rising with the convection water currents. Total dose rate is approximately 3 mrem/h (8.3×10^{-9} Sv/s).

Radiation measurements taken underwater at 10% power with natural convection flow at 14 feet 2 inches (4.3 m) were 250 mrem/h (7.0×10^{-7} Sv/s). If the pool level was reduced to 14 feet 2 inches (4.3 m) under natural convection flow at 10% power, the estimated dose rate at the normal pool level of 21 feet (6.4 m) above the centerline of the reactor core (indicated 0 inches), would be 114 mrem/h (3.1×10^{-7} Sv/s). Note that the underwater measurements used to provide this extrapolated value are conservative since back-scattered radiation from above the 14 feet 2 inch (4.3 m) level was detected. With the Safety System Setting (SSS) for pool level at 17 feet (5.2 m) radiological conditions associated ^{16}N and ^{41}Ar are manageable.

10.3.2 Argon-41 Production

^{41}Ar production is associated with routine operation of the PULSTAR Reactor. This section analyzes off-site consequences using actual measured values for routine ^{41}Ar production. Measurements indicate that the routine ^{41}Ar production is by:

- Air dissolved in pool water (9.4%)
- Beam Tubes and the Thermal Column (7.1%)
- Pneumatic transfer system (83.5%).

In keeping with the concept of ALARA and therefore minimizing ^{41}Ar production and release, the pneumatic system is only operated when required for sample transfer and is otherwise purged with nitrogen to remove the air whenever the reactor is above 0.5 MW. Beam tube experiments are designed to minimize ^{41}Ar production and release and thus have not contributed significantly to the overall ^{41}Ar production.

The PULSTAR reactor building maintains a negative pressure differential with respect to the atmosphere at all times for either normal or confinement ventilation. Therefore, the only release point considered is the top of the 100 foot (30.5 m) exhaust stack.

10.3.2.1 Dose Rate from Argon-41 in Unrestricted Areas

Calculation NRP 91-01 [10-4] was performed to determine the annual off-site dose for ground level receptors from routine releases of ^{41}Ar. A brief description of the source term and calculation method used in Calculation NRP 91-01 are provided in Appendix 13-1. Similar calculations are reported here for ground and elevated receptor locations of interest.

PULSTAR facility surveillance records have demonstrated ^{41}Ar production at an average rate of 5 Ci (1.9 x 10^{11} Bq) per 1000 MWh of operation. The analysis performed in this section makes the assumption of continuous reactor operation for 8760 hours producing 43.8 Ci (1.6 x 10^{12} Bq) of ^{41}Ar. The analysis further assumes continuous occupancy at each off-site location and a constant plume direction.

Solutions for each stability class were determined for two ground receptor off-site locations and two elevated receptor off-site locations of interest surrounding the PULSTAR reactor:

(i) The upper floors of the D.H. Hill Library that are at the same elevation as the top of the ventilation stack, located 140 meters northwest of the stack

(ii) The nearest campus student housing, Carroll Hall, located 260 meters southwest of the stack, particularly the upper floors that are at the same elevation as the top of the stack

(iii) The nearest permanent residence, located at least 200 meters from the stack

(iv) The maximum off-site ground location at approximately 50 meters from the stack

The dose calculation for a ground receptor includes the sky shine contribution from the overhead plume. Elevated receptor dose calculations are given by the semi-infinite cloud submersion model. ANSI/ANS-15.7-1977 (N379) provides the following guidance for atmospheric conditions to be used in calculating offsite dose for periods greater than 4 days:

33.33% Pasquill class C at 3.0 m/s
33.33% Pasquill class D at 2.0 m/s
33.33% Pasquill class F at 2.0 m/s

Table 10-3 presents the annual doses calculated for ^{41}Ar releases from the PULSTAR reactor facility assuming 8760 MW•h of operation.

The dose that could occur for continuous reactor operation if the wind maintained a constant direction toward a particular off-site location for the entire year with equal periods of stability classes C, D, and F, from typical ^{41}Ar releases is calculated to be within 10 CFR 20 limits. In addition, no credit for shielding by surrounding buildings was taken and the calculations further assume that the individual stays at the reference location continuously all year. Therefore, this calculation conservatively estimates the radiological consequences of ^{41}Ar releases from the PULSTAR reactor.

Table 10-3
Estimated Annual Argon-41 Doses For Continuous Reactor Operation at Full Power

Location (m)	Description	Whole-Body (mrem)	Skin (mrem)
50	Maximum Ground Location	1.80	1.8
200	Nearest Housing	0.50	0.50
140†	D.H. Hill Library	25.1	8.8
260†	Carroll Hall Dormitory	8.6	3.1

† Elevated receptor height of 30 m (stack height)
NOTE: 1 mrem equals 1×10^{-5} Sv

10.3.2.2 Dose Rate from Argon-41 in Restricted Areas

^{41}Ar concentration in the restricted areas of the reactor facility is estimated to be 5×10^{-8} μCi/ml (1.9×10^{3} Bq/m^3). This is determined by using the estimated 43.8 Ci (1.6×10^{12} Bq) activity produced by continuous operation and assuming that only 16.5% of the activity is uniformly dispersed in the total volume of air exhausted by the normal ventilation system during the year. The remaining 83.5% of the activity produced is assumed to be exhausted directly to the stack by the Pneumatic transfer system.

$$5 \times 10^{-8}\ \mu\text{Ci/ml} = \frac{(43.8 \times 10^{6}\ \mu\text{Ci})(0.165)}{(10{,}050\ \text{ft}^3/\text{min})(28{,}317\ \text{ml/ft}^3)(5.26 \times 10^{5}\ \text{min/y})}$$

Dose rate from ^{41}Ar is approximated by using the method discussed in Regulatory Guide 8.34 [10-6] as follows:

$$4 \times 10^{-2}\ \text{mrem/h} = \frac{(5000\ \text{mrem})(5 \times 10^{-8}\ \mu\text{Ci/ml})}{(2000\ \text{h})(3 \times 10^{-6}\ \mu\text{Ci/ml})}$$

where, 3×10^{-6} μCi/ml is the Derived Air Concentration (DAC) given in 10 CFR 20
Appendix B Table 1
5000 mrem is associated with 2000 DAC-h of exposure

Dose rate from ^{41}Ar is also approximated by using the total body dose conversion factor given in Regulatory Guide 1.109[10-7] as follows:

$$5 \times 10^{-2}\ \text{mrem/h} = \frac{(8.84 \times 10^{-3}\ \text{mrem/y})(1\text{y}/8760\text{h})(5 \times 10^{-8}\ \mu\text{Ci/ml})}{(1\ \text{pCi/m}^3)(1\ \mu\text{Ci}/\ 1 \times 10^{6}\ \text{pCi})(1\ \text{m}^3/1 \times 10^{6}\text{ml})}$$

Therefore, the whole body dose rate from ^{41}Ar for locations within the restricted area of the reactor facility based on typical and continuous reactor operation is calculated to be less than 1 x 10^{-1} mrem/h (3 x 10^{-10} Sv/s).

10.4 References

10-1 Slade, David H., Meteorology and Atomic Energy, TID-24190, Air Resources Laboratories, Washington, D.C., Chapter 3, 1968.

10-2 Lamarsh John R., Introduction to Nuclear Engineering, Addison-Wesley Publishing Company, Reading, Massachusetts, 1975.

10-3 ANSI/ANS-15.7-1977 (N379), Research Reactor Site Evaluation, American Nuclear Society, La Grange Park, Illinois.

10-4 North Carolina State University internal document, Calculation NRP 91-01 "PULSTAR Reactor Facility Offsite Dose Calculation", 1991

10-5 NUREG 0851, "Nomograms for Evaluation of Doses From Finite Noble Gas Clouds", US Nuclear Regulatory Commission, 1983

10-6 Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses", US Nuclear Regulatory Commission, July, 1992

10-7 Regulatory Guide 1.109, "Calculation of Annual doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50 Appendix I", Revision 1, October, 1977

10-8 Letter from the Assistant Director for Operating Reactors, NRC to the University Radiation Protection Officer, dated May 17, 1973

10-9 Nation Council on Radiation Protection and Measurements Report No.49, "Structural Shielding Design and Evaluation for Medical Use of X-Rays and Gamma Rays of Energies Up to 10 MeV", September 15, 1976

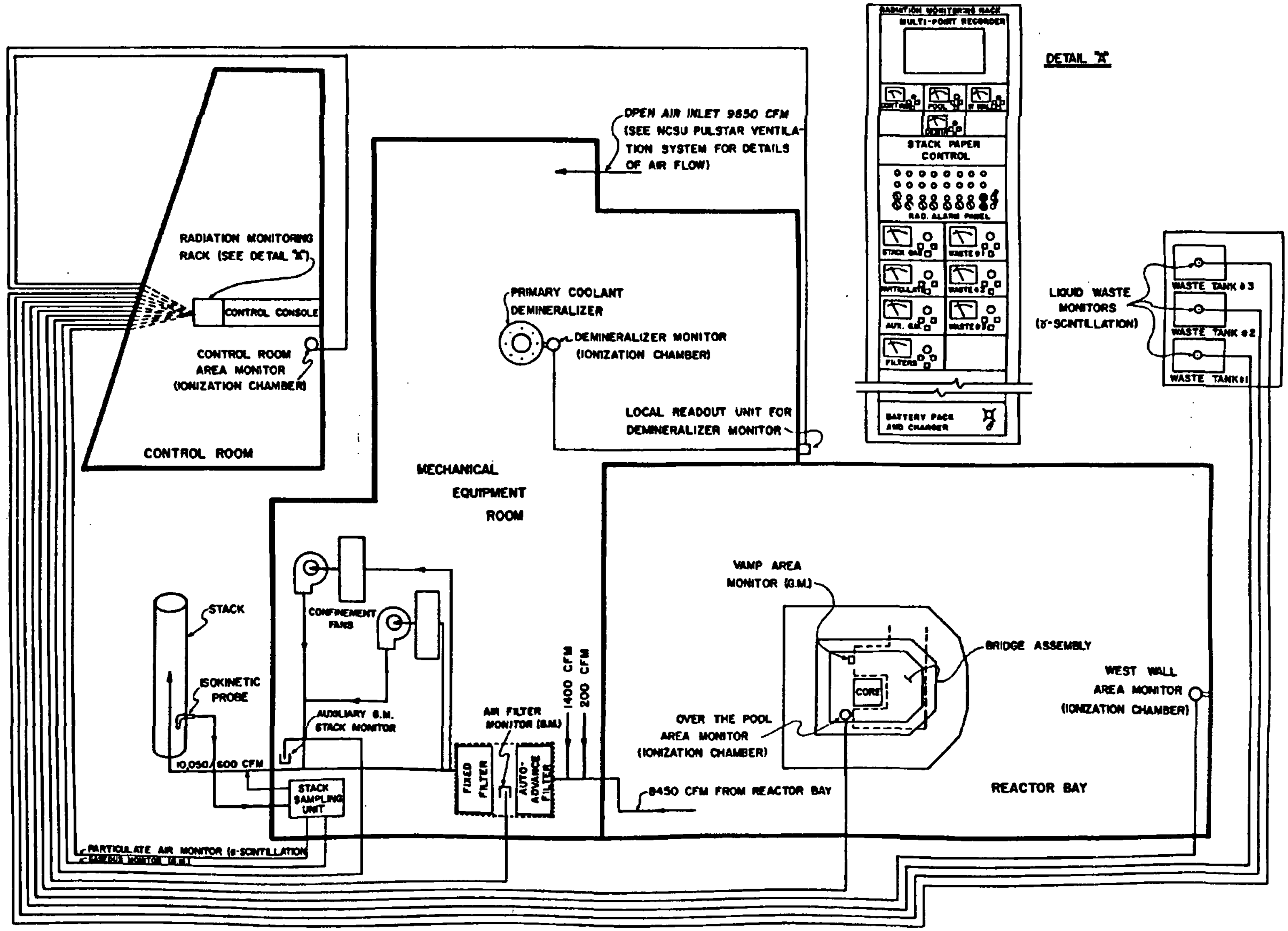

RADIATION MONITORING RACK
MULTI-POINT RECORDER
STACK PAPER CONTROL
RAD. ALARM PANEL
BATTERY PACK AND CHARGER
DETAIL "A"
LIQUID WASTE MONITORS (γ-SCINTILLATION)
WASTE TANK #3
WASTE TANK #2
WASTE TANK #1
OPEN AIR INLET 9850 CFM (SEE NCSU PULSTAR VENTILATION SYSTEM FOR DETAILS OF AIR FLOW)
RADIATION MONITORING RACK (SEE DETAIL "A")
CONTROL CONSOLE
CONTROL ROOM AREA MONITOR (IONIZATION CHAMBER)
CONTROL ROOM
PRIMARY COOLANT DEMINERALIZER
DEMINERALIZER MONITOR (IONIZATION CHAMBER)
LOCAL READOUT UNIT FOR DEMINERALIZER MONITOR
MECHANICAL EQUIPMENT ROOM
STACK
CONFINEMENT FANS
ISOKINETIC PROBE
AUXILIARY G.M. STACK MONITOR
10,050/600 CFM
STACK SAMPLING UNIT
AIR FILTER MONITOR (G.M.)
FIXED FILTER
AUTO-ADVANCE FILTER
1400 CFM
200 CFM
8450 CFM FROM REACTOR BAY
PARTICULATE AIR MONITOR (β-SCINTILLATION)
VAMP AREA MONITOR (G.M.)
BRIDGE ASSEMBLY
CORE
OVER THE POOL AREA MONITOR (IONIZATION CHAMBER)
WEST WALL AREA MONITOR (IONIZATION CHAMBER)
REACTOR BAY

Figure 10-1

11 CONDUCT OF OPERATION

11.1 Organization and Responsibilities

The North Carolina State University (NCSU) administration and PULSTAR Reactor Operations staff recognize a basic responsibility to operate the PULSTAR Reactor safely and efficiently, and to provide experimental facilities for various teaching, research, and service programs. The achievement of these objectives, under the guidance of the Director of the Nuclear Reactor Program, requires a proficient staff, advanced planning, and adherence to basic guidelines and regulations from the Nuclear Regulatory Commission (NRC), the North Carolina State Department of Environment, Health, and Natural Resources, and University administration.

11.1.1 Description of Organizational Structure

The following line and functional descriptions reflect the administrative controls for the routine operations of the PULSTAR facility. North Carolina State University at Raleigh is one of sixteen constituent institutions of the Consolidated University of North Carolina. The University's line organization is shown in Figure 11-1.

11.1.1.1 Facility Startup Operations

This section remains in the Safety Analysis Report for historical reference. The initial startup included acceptance and integrated systems test, initial core loading and criticality, approach to power, and the confirmation of reactor parameters. These tests were carried out by the Department of Nuclear Engineering's regular staff under the direction of the Nuclear Operations Administrator (now referred to as the Director). Supporting and advising the Nuclear Operations Administrator was a PULSTAR qualified reactor contractor (AMF) field engineer, a cold licensed (pre-critical) Senior Reactor Operator, and a cold licensed Reactor Operator from the Department. The operators had either completed qualification training on a similar reactor prior to initial startup or had previous research reactor experience. After initial operations and a sufficient period of training, a staff of operators were qualified by the Department.

11.1.1.2 Present Operations

After the startup testing, initial operations, and licensing of additional Department personnel, the organizational structure for PULSTAR Operations developed into that shown in Figure 11-1. The normal operating staff for the PULSTAR consists of the facility Director, Associate Director, a licensed Reactor Operations Manager, a licensed Chief Reactor Operator, a licensed Chief of Reactor Maintenance, and a Reactor Health Physicist who reports to the Nuclear Engineering Department Head. The minimum staffing when the reactor is not secured consists of a licensed reactor operator present in the control room, a facility qualified reactor operator assistant (ROA) capable of being at the reactor

facility within five minutes upon request of the reactor operator on duty, and a Senior Reactor Operator readily available on call [11-1]. The described staffing follows the guidance of ANSI/ANS 15.1 [11-1] and meets the requirements of 10 CFR 50.54 (k) and (m(1)) [11-2]. The methods and procedures for operating and maintaining the PULSTAR are based on manuals provided by the equipment vendors and specific operating procedures developed by the PULSTAR staff.

11.1.1.3 Safety Review

In addition to a formal, well defined, and conscientious safety program by the PULSTAR staff, outside "third party" safety committees have been established to oversee and audit the manner in which operations are carried out. The University maintains two such permanent committees for reviews, the Radiation Protection Committee and the Reactor Safety and Audit Committee. Appointed by University administration they are composed of persons from such fields as reactor analysis, design, operations, instrumentation, other engineering and scientific fields, and are active in and concerned with, safety analysis.

11.1.1.3.1 Radiation Protection Committee (RPC) [11-3]

Initially established by the Chancellor, the Radiation Protection Committee (RPC) is the central agency on campus responsible for establishing procedures and policies for the procurement, use, and disposal of all radioactive materials and sources of ionizing radiation. The Committee determines the adequacy of training of personnel for performing experiments with ionizing radiation and maintains a system to inventory radioactive materials through the campus Radiation Protection Office. The Committee approves, from the standpoint of safety and health, the proposed usage, procedures, and facilities associated with PULSTAR Reactor experiments. It is responsible for ensuring that utilization and disposal of all radioisotopes and equipment are in accordance with regulations from the Nuclear Regulatory Commission (NRC), the North Carolina State Department of Environment, Health, and Natural Resources, and University administration. The Committee can demand cessation of manipulations or actions which are felt to compromise safety of the PULSTAR Reactor. The RPC exercises oversight over the University Radiation Protection Program and performs final review of the actions of the Reactor Safety and Audit Committee (RSAC).

11.1.1.3.2 Reactor Safety and Audit Committee (RSAC) [11-3]

The RSAC has the primary responsibility to assist the RPC in ensuring that the reactor is operated in compliance with the facility license and all applicable regulations. The RSAC makes periodic, independent, objective appraisals of reactor procedures and operations specified in Section 6 of the Technical Specifications. At the discretion of the RSAC, professionals from other universities or outside organizations may be invited to assist in these appraisals.

In addition to the audit function of the RSAC, the committee also provides technical review and analysis in areas such as design changes, procedure changes, license amendments, etc. related to the PULSTAR Reactor.

11.1.1.4 Personnel

11.1.1.4.1 Titles and Qualifications

Director of the Nuclear Reactor Program (NRP) (Level 1) [11-4] - The Director is a member of the Department of Nuclear Engineering faculty with responsibilities for the long range development of the Nuclear Reactor Program and for the general conduct of Program operations. The Director evaluates new service and research applications for the PULSTAR Reactor, recruits new users of the facilities, supervises the development and expenditure of Program budgets, develops department and university support for needed capital investment. The Director is responsible for reactor performance, personnel matters, and reactor safety. The Director works through the Associate Director to monitor daily operations and the Reactor Health Physicist to monitor radiation safety.

Associate Director (Level 2) [11-4] - The Associate Director is responsible for facility development, safe, and efficient operation of the PULSTAR Reactor Facility. In matters pertaining to the operation of the facility and the Technical Specifications, the Associate Director reports to the Director of the Nuclear Reactor Program. The Associate Director consults with the Director in support of long term development of the facility and on academic utilization of the reactor facility. The Associate Director also holds the title of Reactor Engineer and performs those calculations or reviews necessary to ensure optimum utilization of reactor fuel and experimental facilities. The Associate Director works closely with the Reactor Health Physicist on matters of emergency preparedness and radiation safety.

Reactor Operations Manager (Level 3) [11-4] - The Reactor Operations Manager (ROM) is responsible for assuring that operations are conducted in a safe manner and within the limits prescribed by the facility license, all applicable Nuclear Regulatory Commission regulations, and experiments are conducted in agreement with the provisions granted by the Radiation Protection Committee. The ROM is responsible for routine supervision and management of the PULSTAR Reactor facility and will have the associated experience in reactor principles, operation, and maintenance. In planning operational activities, the ROM ensures proper coordination between operations and maintenance activities. The Reactor Operations Manager works closely on a daily basis with the Associate Director and Reactor Health Physicist. The ROM position maintains a Senior Reactor Operator license.

Chief Reactor Operator (Level 4) [11-4] - The Chief Reactor Operator (CRO) supports and assists the ROM in planning operational requirements and activities of the reactor, support equipment, and associated irradiation facilities. The CRO maintains the PULSTAR Operations Manual including changes for approval, reviews operating logs and records,

maintains fuel burn-up records, schedules outages, and makes operational reports to the ROM. The CRO is responsible for reactor operator training and requalification programs. These programs include the periodic updating of operators and technicians in nuclear safety standards and operating practices.

Supporting the CRO will be licensed reactor operators and reactor operator assistants. The CRO will have at least three years experience in the operation of a research and test reactor or equivalent. The CRO position maintains a Reactor Operator license.

Chief of Reactor Maintenance (Level 4) [11-4] - The primary responsibility of the Chief of Reactor Maintenance is to support and assist the ROM in the specific area of reactor preventative and corrective maintenance. The CRM routinely inspects the reactor complex and plans the periodic surveillance of all appropriate reactor systems and maintains records of repairs and upkeep. Occasionally the CRM will be assisted by the ROM, CRO, and Reactor Operators. In these instances, the CRM will train and supervise the work of others who provide assistance as necessary.

The CRM maintains sufficient inventory of repair equipment and supplies to assure minimal delays in reactor availability. The CRM is familiar with the facility as-built drawings and renders assistance to outside maintenance contractors who may be required to service equipment in the Reactor Building. The CRM shall have at least three years of formal maintenance training or the professional equivalent. The CRM position maintains a Reactor Operator license.

Reactor Health Physicist (RHP) - The Reactor Health Physicist (RHP) performs a staff function for the Nuclear Reactor Program and Department of Nuclear Engineering. The RHP may be assisted by technicians trained in health physics principles. The Radiation Protection Division of the Environmental Health and Safety Center (SHSC) is also available upon request in health physics matters. The RHP develops, manages, and directs the Health Physics program to ensure personnel safety during operation of the reactor and related experimental facilities. The RHP is available upon request to departmental laboratory experimenters. The RHP conducts radiation and industrial safety training programs for the PULSTAR facility on-site, formulates policies and procedures for all health physics functions, serves as fuel and source accountability officer, and plans release or disposal of radioactive material with the RPO. In addition, the RHP serves as advisor on health physics and nuclear services requirements and trends.

11.1.1.4.2 Maintenance and Technical Support

The Department of Nuclear Engineering has access to University machine shops with the capability of advanced fabrication, milling, and assembly techniques. These shops routinely produces complex experimental equipment for laboratory use within the College of Engineering; therefore, shop personnel fully understand the necessity of accuracy and precision when providing services for the reactor.

Technical support for engineering and scientific topics related to the reactor facility are conveniently available through the Department of Nuclear Engineering faculty and other experienced professional personnel in other departments on the NCSU campus.

11.2 Training

11.2.1 Nuclear Reactor Program Staff

The training program for the NCSU PULSTAR Operations staff is designed to meet the needs of each person appointed, depending on background, previous experience, and training. The phases of training include basic reactor theory, reactor facilities and design, and operation of the PULSTAR under the supervision of a licensed reactor operator.

All operators will undergo a selection, training and certification program prior to unsupervised operation of the PULSTAR reactor. All licensed operators will participate in a requalification program, which will be conducted over a period not to exceed two years. The requalification program will be followed by successive two year programs. PULSTAR training and records shall continue to meet the requirements of 10 CFR 55 and 20, as amended.

The Chief Reactor Operator (Program Administrator) is exempt from written, oral, and demonstrational examinations.

Besides the qualification and licensing of the reactor operators, the maintenance staff will qualify by demonstration and/or examination.

11.2.2 Replacement Personnel

Replacement personnel will require training and qualification. These are handled as described in 11.2.1 above.

11.2.3 Emergency Plans and Procedures

An Emergency Plan for the PULSTAR Reactor has been developed and is on file with the Nuclear Regulatory Commission. From the approved Emergency Plan, Emergency Procedures were developed and are implemented to maintain a state of readiness in case of an emergency.

11.3 Procedures

11.3.1 General

Preliminary procedures were furnished by the contractors during the startup of the facility. The staff then developed and augmented each procedure required for PULSTAR

operations. A PULSTAR Operations Manual details reactor operating procedures while Health Physics Procedures, Special Procedures, and Surveillance Procedures have all been developed to handle the operation and utilization of the facility.

11.3.2 Changes

Procedures will periodically be reviewed and revised as necessary to address both normal and emergency operating conditions. Such changes will be done in accordance with Special Procedure 2.1.

11.4 Records and Reports

Operating records required administratively and by the Technical Specifications are needed to ensure adequate surveillance of reactor operations, provide sufficient repair history and recorded symptoms to detect malfunctions and perform remedial maintenance. The use of standard log book entries, periodic recording of plant parameters, and following detailed check lists are most important. Activities such as shutdowns, malfunctions, maintenance performed, research activities, and sample irradiations are recorded. Data from these logs are used to summarize and render long term evaluations of the facility operation. Included in the records on file will be routine operating logs, preventive maintenance and malfunction reports, and equipment history.

11.5 Administrative Control

Administrative controls have been established to insure that all operations, tests, and emergencies are handled in accordance with written procedures. These procedures are reviewed and approved in accordance with Special Procedure 2.1.

Informal audits are performed on a frequent basis by the Associate Director and Reactor Health Physicist, independent of the Radiation Protection Committee or Reactor Safety and Audit Committee in order to review operations and assist with any problems which might arise.

There are four sets of keys which are controlled by the Reactor Operations Manager (ROM); one reactor console, one reactor mode key, six radiation alarm bypass, and one confinement/evacuation reset key. Authority for use of these keys is based on the training of the pertinent individual in accordance with approved written procedures.

11.6 References

11-1 ANSI/ANS-15.1, The Development of Technical Specifications for Research Reactors, December 7, 1990, page 8.

11-2 Title 10, Code of Federal Regulations, Part 50, Section 54 (k) and (m(1)), August 31, 1993

11-3 Radiation Protection Manual, Sixth Edition, 1994, issued by the Environmental Health and Safety Center, NCSU.

11-4 ANSI/ANS-15.1, The Development of Technical Specifications for Research Reactors, December 7, 1990, page 8.

NORTH CAROLINA STATE UNIVERSITY
PULSTAR REACTOR ORGANIZATIONAL CHART

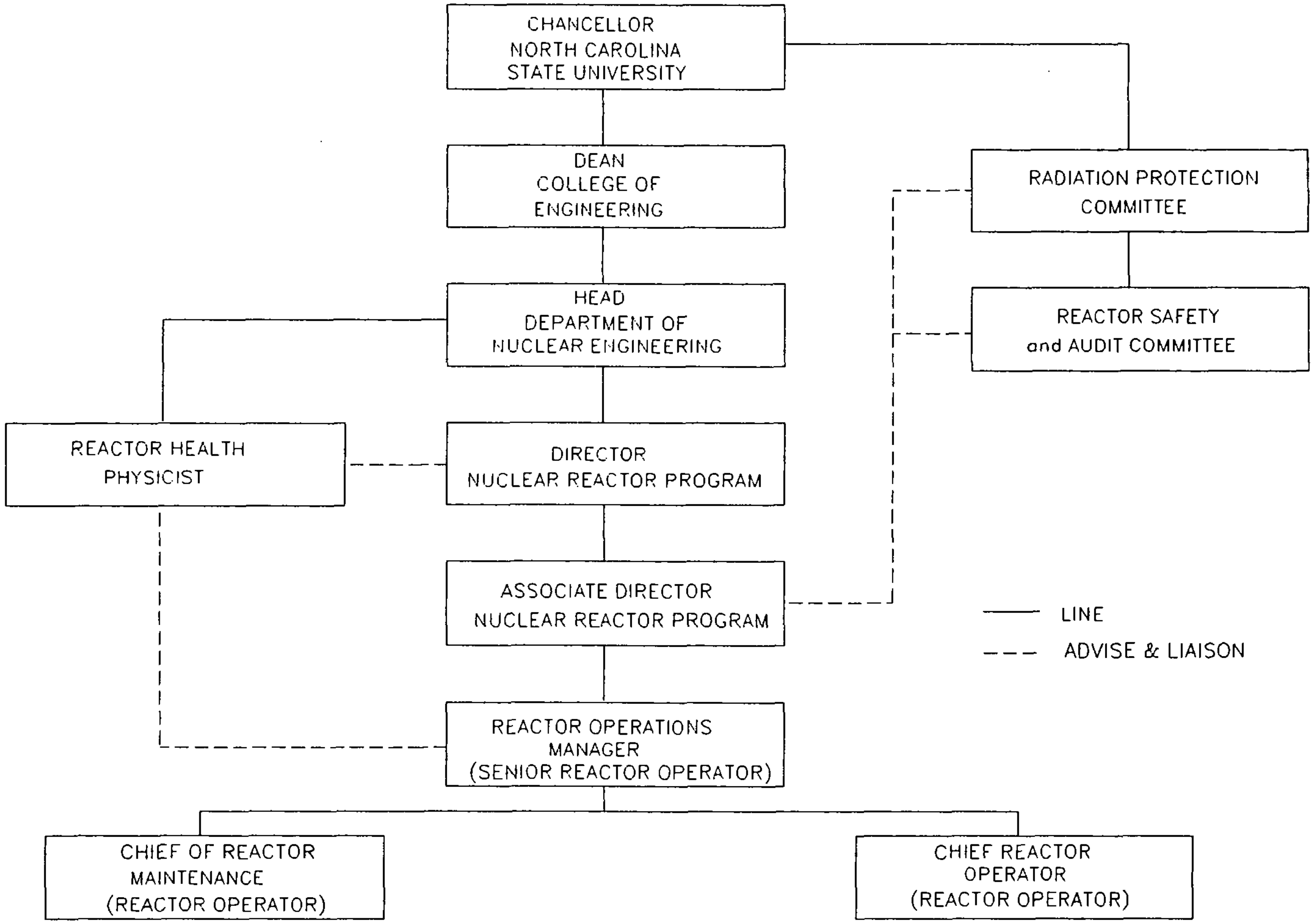


Figure 11-1

12 INITIAL TESTS AND OPERATIONS

12.1 Introduction

An initial test and operation program was performed on the reactor and auxiliary equipment for ensuring proper operation and to verify the operational characteristics of the core. For historical purposes, this section will continue to summarize the initial battery of tests required in support of the initial operating license of the PULSTAR reactor facility.

12.2 Tests Prior to Initial Fuel Loading

The following pre-critical tests were performed prior to loading fuel in the core. Each test is followed by a written statement of the purpose of the test. The procedures followed in these tests are contained in the facility "Startup Manual."

12.2.1 Reactor Building Ventilation Test

The ventilation system capacity was tested under normal and confinement modes.

12.2.2 Radiation Monitoring System

The radiation monitoring system was tested using radioactive sources and the alert and alarm setpoints were set.

12.2.3 Primary and Secondary Coolant System

The Primary Coolant System flow, flapper valve, and tank level alarm setpoints were adjusted along with a verification of successful pump operation. The Secondary System pump and cooling tower fan operability was verified. In addition, the primary demineralizer flow rate was adjusted and resistivity control verified.

12.2.4 Console and Nuclear Instrumentation

All console alarm settings were made and acceptable performance verified. The neutron detectors were source checked. A signal generator was used for checkout of the Pulse Channel response time. The various interlocks for different modes of operation were also checked.

12.2.5 Control and Pulse Rod Drive Packages

The alignment, drop time, rod drive speeds, pulse ejection time and operation were tested.

12.2.6 Evacuation Procedure

A practice evacuation was made to verify the adequacy of the system and to identify any potential problems and/or improvements with the system.

12.3 Initial Critical Test Program

After the tests identified in section 12.1 were successfully completed, the initial criticality test program proceeded as detailed in this section. The initial critical test program covered the initial loading of fuel through the zero power tests. The tests are listed below along with a brief statement of the test purpose. Detailed procedures for these tests are contained in the PULSTAR Startup Manual.

12.3.1 Initial Approach to Criticality

The reactor was loaded with fuel in a deliberate, safe and controlled manner. After criticality was attained and initial power determination was made, nuclear instrumentation was checked for proper operation.

12.3.2 Excess Reactivity Measurement

Upon completion of initial criticality, the 5 X 5 Standard Core loading was completed by loading additional fuel, and control rod calibrations were performed to determine excess reactivity. The results of these tests are detailed in section 3 of this document. Shutdown Margin with a stuck rod criteria was verified for this initial core loading.

12.3.3 Flux Measurements

Flux measurements were made for the 5 X 5 Standard Core loading by copper wire irradiations. This allowed the determination of the hot spot factor.

12.3.4 Experimental Facilities Worth

The reactivity effect of the various experimental facilities was measured. Beam tube reactivities were determined by the change in critical rod height associated with draining and filling the beam tube.

12.3.5 Pulse Rod Calibration

The pulse rod was calibrated for both the 5 X 5 Standard Core and the 5 X 5 Pulsing Core (see Section 3) with the pneumatic air supply disconnected.

12.3.6 Power Calibration

The flux measurements referenced in section 12.2.3 above were used as an initial estimate for adjusting the power detectors' position.

12.4 Approach to 1 MW Steady State Operation

12.4.1 Natural Convection Cooling

Using the 5 X 5 Standard Core loading, the reactor power was increased in 50 kW increments up to 250 kW. Control rod positions were recorded to determine the power defect associated with the power level.

12.4.2 Forced Cooling

After the core was operated at 100 kW with natural convection cooling, the power was reduced to zero and the primary pump started to establish forced cooling. The power level was then increased in 100 kW increments. A power check was made at each power level using the flow times delta temperature technique. Nuclear instruments were then adjusted to agree with this power level check. Doppler and power coefficients were verified to be negative and their values estimated on this power increase.

12.4.3 1 MW Shielding and Building Survey

After achieving the 1 MW power level, a shield survey was made to verify the adequacy of the Biological Shield. Radiation readings throughout the Reactor Bay were measured and the Mechanical Equipment Room (MER) was also surveyed.

12.4.4 Xenon Effect

During the shield survey power operation, control rod positions were recorded to determine the xenon reactivity behavior on the reactor core.

12.4.5 Automatic Control

After 1 MW operation was completed, the reactor power was reduced to 500 kW for a checkout of the automatic control system.

12.4.6 Square Wave Test

It was initially envisioned that square wave startup testing would be completed during the initial checkout. However, this testing was not completed and the PULSTAR Facility has no plans to perform a square wave startup.

12.5 Pulse Tests

A pulse test program was executed to reach the design pulse. The test consisted of incremental increases in the reactivity step input to the core. Calibration of the Pulse Measuring Channel had already occurred as a part of the steady state tests. After each reactor pulse, the peak power and total energy release was recorded and used to develop pulsing curves.

12.5.1 Approach to Design Pulse

During the approach to the design pulse, auxiliary equipment was used to determine the pulse shape, peak power, peak energy release and total energy release. The data was taken from a Visicorder. The reactor period was then determined from the pulse traces.

12.5.2 Measurement of $1/\beta_{eff}$

The data measured during the approach to the design pulse was plotted to determine the $1/\beta_{eff}$ for the reactor core loading.

12.5.3 Design Pulse

After reaching the reactivity insertion for the design pulse, this amount was repeated ten times to check repeatability of the pulse.

12.5.4 Fuel Pin Inspection

The hottest fuel pin identified in the 5 X 5 Standard Core Loading was examined following the ten design pulses to verify no physical changes had resulted from the design pulsing.

12.6 Post-critical Tests

Following completion of the above test program, the operation data was reviewed to identify any improvements that could be made to increase the safety of the reactor. In addition, normal operating background radiation levels versus their respective alarm points were checked to review their difference.

12.7 Operating Restrictions

Through out the initial test program, the operating restrictions as identified in the PULSTAR Technical Specifications were adhered to.

13.0 SAFETY ANALYSIS

13.1 Postulated Accidents

Postulated accidents considered at the NCSU PULSTAR research reactor facility may be grouped into the following two categories:

Non-Excursion Accidents
Loss of Primary Flow
Water Logging
Loss of Pool Water
Fuel Pin Clad Failure
Heat Exchanger Pressure Boundary Breach
Failed Fueled Experiment

Excursion Accidents
Fuel Loading Accident
Start-up Accident
Pulse Rod Fails to Return
Experiment Failure
Control Rod Failure
Cold Primary Coolant Slug
Fuel Storage

13.2 Accident Analysis

In the following discussion, the cause, effect, and consequence of each of the accidents listed above are considered. In cases where hazardous conditions are predicted, a description is given of preventive or corrective measures incorporated into the PULSTAR Facility.

13.2.1 Non-Excursion Accidents

13.2.1.1 Loss of Primary Flow

In the event of failure of the primary coolant pump or accidental shutoff of the discharge line, either the low flow or flapper open condition will automatically scram the reactor. The flapper will open on the side of the plenum below the core region and will allow a flow path for natural convection cooling.

Appendix B, Section IV, contains the loss of flow analysis assuming conservative initial conditions and low flow scram initiation delays of 2 and 20 seconds. The ultraconservative assumption of a 20 second scram results in a minimum DNB ratio of 8, hot channel fuel centerline temperature of 1240°F (671°C), and a hot channel clad temperature of 380°F (193°C). These results demonstrate that the core will not be damaged during the flow reversal period, nor will any other adverse effect occur. Additionally, tests[13-1 & 13-2] have been run in a 20 assembly core which show that the core can be operated at 1 MW with natural convection cooling.

13.2.1.2 Waterlogging

An important consequence of the use of unbonded fuel assemblies is the possibility of a waterlogging failure. Such failure could occur in the event of a defect or leak in the fuel cladding, which would permit in-leakage of water during low power operations or at shutdown. During a subsequent pulse, pressure would be generated in the annulus between fuel pellets and cladding and due to the inability of the steam to escape rapidly through the defect, a failure of the pin could result.

Two cases of such waterlogging failures have been documented,[13-3] both of which occurred in conjunction with experiments in which holes were drilled through the cladding to accommodate thermocouples. The first case, in which a series of holes were drilled in a line, a pulse having a period of 7.5 milliseconds resulted in a rupture 12 inches (30.5 cm) in length, which followed the line of drilled holes. In a similar case involving only one thermocouple hole, in which the epoxy sealant deteriorated and allowed water to enter the tube, a two centimeter vertical crack developed in the center of a small blister which formed after a pulse.

Since pulsing the NCSU PULSTAR Reactor is no longer permitted as of Amendment 9, failure of a waterlogged fuel pin during pulsing is not applicable. It should be noted, however, that if a defect in the cladding should occur, the fission products would likely escape during the original in-leakage of water, and should serve to indicate the presence of the defective pin.

13.2.1.3 Loss of Pool Water

A complete loss of pool water in a short period of time would uncover the core while a significant amount of heat is still being generated in the fuel pins from fission product decay heat. While investigating the consequences of such an accident, it was assumed that the reactor had been operating with a 25 fuel assembly core at a power level of 1 MW for an infinite time, and that fission product concentrations had attained equilibrium.

The loss of water coverage over the core could occur [redacted] permitting the pool to drain into the Reactor Bay. The loss of pool water would not result in failure of the fuel pin cladding since the heat removal is sufficient, with the core in air, to remove shutdown heat generated in the core. Calculations, based upon experiments at the Livermore Test Reactor (LPTR)[13-4] in which loss of water was used as a shut down mechanism and in which the fuel plate temperatures were measured, were used to determine the maximum surface temperature of a fuel pin. At the LPTR, the maximum plate temperature was reached approximately one hour after uncovering the core.

Fuel plate temperature data from the LPTR at 1.0 MW were adjusted to reflect differences in hydraulic and heat transfer characteristics. The resulting maximum surface temperature

becomes 765°F (407°C), which is well below the melting point of zircaloy; therefore, no clad damage would occur. Another analysis calculated the maximum fuel average temperature to be 1040°F (560°C) for a 48 hour continuous full power and 1270°F (688°C) for 12,960 hours of continuous operation at full power. The fuel average temperatures remain significantly below thresholds that would lead to either clad melt or fuel melting[13-5].

As detailed in Section 3, the NCSU PULSTAR has incorporated 10 graphite reflectors to increase the excess reactivity of the core. The potential for a large release of Wigner energy and/or a graphite fire following a loss of pool water has been analyzed by Brookhaven National Laboratories for all research reactors in the United States[13-6]. This study as documented in NUREG/CR-4981 concluded that there is no new evidence associated with either the Windscale Accident or the Chernobyl Accident that indicates a credible potential for a graphite burning accident in any of the U.S. research reactors, such as the PULSTAR. Furthermore, the study indicates there is no new evidence that suggests that detailed case-by-case safety analyses of the role of graphite in NRC licensed reactors are warranted.

The major hazard associated with the complete loss of pool water are the high radiation levels from the uncovered core. The corrective measure would be to repair the leak and refill the tank without subjecting personnel to excessive radiation dose. The time required to drain the tank through tank liner penetrations varies from 2.4 minutes for the 12 inch x 12 inch (30.5 cm x 30.5 cm) beam tube to 12.3 minutes for a 6 inch (15.2 cm) beam tube. In the event the loss of water were to occur during working hours, leak repair at the area outside the biological shield could be accomplished and pool refilling initiated to maintain water shielding over the reactor core. Large scale refilling would be accomplished by using a local fire hydrant and fire department hoses spraying up and over the pool parapet from the Reactor Bay loading dock doors.

Assuming the complete draining of the pool did occur, the radiation levels at the top of the pool near the Control Room has been calculated by Monte Carlo techniques (and bench-marked with LPTR data)[13-7] to be approximately 250 mrem/h ($6.9(10^{-7})$ Sv/s) at 10 minutes after shutdown from full power. In addition, the dose rate predicted for the Reactor Bay floor during the loss of water scenario is 175 mrem/h ($4.9(10^{-7})$ Sv/s) (also 10 minutes after shutdown). Dose rates outside the Reactor Building against the Reactor Bay walls is predicted to be on the order of 4 mrem/h ($1.1(10^{-8})$ Sv/s) due to sky-shine. The radiation level on the Reactor Bay floor would result in personnel dose being below 10 CFR 20[13-8] annual limits for repair of the leak and starting of pool refill operations.

. These storage pits are designed to ensure a sub-critical condition i. If the loss of water accident were to occur, the accident would be handled as stated above. After plugging the failed beam tube or piping, Operations personnel would refill the pool and the fuel could then be moved to the storage pit and covered with additional shielding as needed. The storage pit would be water filled at all times. The use of the storage pits would thus allow additional repair with the tank partially drained if needed.

Because of the high radiation levels in an upward direction from the uncovered reactor core, this accident is considered to be the maximum credible non-excursion accident. Alarms and administrative controls would clear affected areas of personnel in ample time to prevent excessive personnel exposure.

13.2.1.4 Fuel Pin Clad Failure

The Buffalo PULSTAR fuel pins have operated for at least 40,000 MW·h and have experienced more than 250 pulses without failure of the zircaloy-2 cladding. Even though a mechanism for the release of the fission products in the fuel pin annuli cannot be assigned to the MCA by any realistic or practicable mode, an analysis of such a postulated release on reactor operations is presented.

After any mode of operation, the failure of fuel pin cladding is assumed to occur by mechanical damage to the clad by impacting an assembly on its side. The design basis for the release of fission products released is the failure of three pins by this mode. Any slower mode of fission product release would be detected by routine water chemistry and radioanalysis on the primary coolant, by the radiation monitors over the reactor pool, by the Auxiliary GM, Stack Gas, and Demineralizer Radiation Monitoring Channels, and by the continuous air monitor (CAM) in the Reactor Bay.

The 500 gpm (31.5 ℓs^{-1}) primary coolant flow passes downward through the core, since forced flow is required for full power. At the time of clad failure, the fission product inventory[13-9] in the core is based on 6320 MW·days of operation at 1.0 MW (or equivalent to 20,000 MWD/MTU). The activities of the fission products which would be found in the fuel pin annulus gap between the zircaloy-2 cladding and the UO_2 pellets are given in the Table 13-1.

For calculational purposes, assume that three fuel pins suffer a clad rupture and all the fission product gases contained in the annulus and upper chamber of the three fuel pins are released into the pool water at a depth of 20 feet (6.1 m). Only the gaseous fission products are considered since the other fission products would remain in the reactor pool. Of the gaseous fission products released into the pool water, 97 percent of the iodine and bromine isotopes are assumed to be retained in the reactor water[13-10].

For the purpose of assessing the consequences of released fission products, it is assumed that the halogens that are not retained in the water and all the noble gases escape into the Reactor Bay air space. However, in the forced flow cooling condition, gases would tend to be carried by the forced flow out the pool hotleg piping, through the delay tank and heat exchanger, and back to the pool.

Table 13-2 shows the gaseous fission product activity which would be released into the Reactor Bay air space in the event three fuel pins were to rupture under the pool surface.

Table 13-1
Activities of Fission Products in the Fuel Pin Annuli Gaps

Isotope	λ (s^{-1})	Annuli Ci/Core	Annuli Ci/Pin
Kr-88	6.78(10^{-5})	0.78	1.26(10^{-3})
Xe-133	1.53(10^{-6})	11.3	1.81(10^{-2})
Xe-138	8.15(10^{-4})	0.040	6.40(10^{-5})
I-131	9.98(10^{-7})	5.33	8.52(10^{-3})
I-133	9.26(10^{-6})	6.80	1.09(10^{-2})
Br-83	8.05(10^{-5})	11.9	1.90(10^{-2})
Br-84	3.63(10^{-4})	0.10	1.60(10^{-4})
Cs-137	7.33(10^{-10})	0.69	1.10(10^{-3})
Cs-138	3.59(10^{-4})	0.548	8.76(10^{-4})
Cs-139	1.22(10^{-3})	23.2	3.72(10^{-2})
Sr-89	1.59(10^{-7})	21.9	3.50(10^{-2})
Sr-90	7.55(10^{-10})	0.575	9.20(10^{-4})
Sr-91	2.02(10^{-5})	0.753	1.20(10^{-3})

NOTE: 1 Curie (Ci) equals 3.7(10^{10}) Bq

Table 13-2
Gaseous Fission Product Release to the Reactor Building Air Space
From a Failure of Three Fuel Pins Below the Pool Surface

Isotope	Ci
Kr-88	3.78(10^{-3})
Xe-133	5.43(10^{-2})
Xe-138	1.92(10^{-4})
I-131	7.67(10^{-4})
I-133	9.80(10^{-4})
Br-83	1.72(10^{-3})
Br-84	1.43(10^{-5})

The anticipated sequence of events which would occur for a postulated three fuel pin failure would be as follows:

(a) Fission products which escape from the fuel cladding rupture would be carried by the coolant downward, through the ^{16}N delay tank, heat exchanger, and primary pump and then returned to the bottom of the pool.

(b) Solubility and volatility reduce the iodine and bromine fission products available for escape into the Reactor Building to a factor of $3(10^{-2})$ [13-10]. The radiation level increase in the Stack Gas or Auxiliary GM would most likely cause the normal ventilation system to automatically shutdown, placing the Reactor Building into confinement and thus preventing release of unfiltered air. The Radiation Monitoring System alarm automatically initiates evacuation of the Reactor Bay area. When the normal ventilation system shuts down, a 600 cfm ($0.3\ m^3 \cdot s^{-1}$) confinement fan will automatically start and pass the Reactor Building air through a filtration system prior to its discharge from the 100 foot (30.5 m) stack. The filtration system has a High Efficiency Particulate Absorber filter with a removal efficiency of 99.97% and an activated charcoal filter with a removal efficiency of 99%. Prior to use and as a routine surveillance after installation, the charcoal filters are tested for iodine removal efficiency. After passing through the confinement filters, and prior to discharge from the stack, the 600 cfm ($0.3 m^3 \cdot s^{-1}$) flow would normally be diluted again with exhaust flow from the ventilation system serving the south wing of the Burlington Engineering Laboratories (a factor of 20 dilution), however, no allowance for this additional dilution is assumed in this analysis.

(c) The Reactor Bay free air volume is approximately $2.4(10^9)\ cm^3$. The fission product gases escaping from the pool are assumed to be uniformly distributed throughout the Reactor Bay. Table 13-3 shows the activity contained in the 600 cfm ($0.3\ m^3 \cdot s^{-1}$) Reactor Building air exhaust after uniform distribution inside the Reactor Bay and passage through the confinement filter system occurs.

(d) Based on the assumptions under item (c) above, the fission product release scenario is calculated to last 2.4 hours with no allowance for decay. Various off-site locations downwind of the stack would be exposed to the plume during this time. Therefore, the release has a constant concentration as detailed in Table 13-3, with no allowance for the south wing ventilation dilution and a total duration of 2.4 hours. Since the stack exhaust exit is elevated and not accessible by the public, a dispersion factor for various off-site locations is calculated to quantify radiation dose to the public associated with the three fuel pin failure scenario.

Table 13-3
Air Activity Exhausted by Stack After Filtration for the Three Fuel Pin Failure

Isotope	Stack Exhaust After Filtration μCi/ml	AEC† Value μCi/ml	Average Release Rate Ci/s	Total Release Ci
Kr-88	$1.57(10^{-6})$	$9.00(10^{-9})$	$4.44(10^{-7})$	$3.78(10^{-3})$
Xe-133	$2.26(10^{-5})$	$5.00(10^{-7})$	$6.42(10^{-6})$	$5.43(10^{-2})$
Xe-138	$8.00(10^{-8})$	$2.00(10^{-8})$	$2.27(10^{-8})$	$1.92(10^{-4})$
I-131	$3.39(10^{-9})$	$2.00(10^{-10})$	$9.61(10^{-10})$	$7.67(10^{-6})$
I-133	$4.08(10^{-9})$	$1.00(10^{-9})$	$1.16(10^{-9})$	$9.80(10^{-6})$
Br-83	$7.16(10^{-9})$	$9.00(10^{-8})$	$2.03(10^{-9})$	$1.72(10^{-5})$
Br-84	$5.96(10^{-11})$	$8.00(10^{-8})$	$1.69(10^{-11})$	$1.43(10^{-7})$

† Effluent air concentration for unrestricted areas given in 10CFR20 App. B, Table 2
Notes: Multiply μCi/ml by $3.7(10^{10})$ for Bq/m^3 and multiply Ci by $3.7(10^{10})$ for Bq

The assumption of mixing in the Reactor Bay area is conservative since no dilution is assumed for the 600 cfm ($\sim 0.3\ m^3 \cdot s^{-1}$) of fresh air which must leak into the Reactor Building as makeup. Based on the Reactor Bay volume of $2.4(10^9)\ cm^3$ and a purge rate of 600 cfm ($\sim 0.3\ m^3 \cdot s^{-1}$), this analysis makes the additional conservative assumption that the entire fission product inventory is removed from the Bay in approximately 2.4 hours corresponding to the time for a complete air volume exchange in the bay. This process would actually take a considerably larger amount of time due to the continual dilution by in-leakage and the slow purge rate. Total activity removed, however, remains the same regardless of duration since the assumed source term is fixed. Additionally, it is assumed that no radioactive decay takes place during the release interval.

Four off-site locations are considered for the dose rate calculation to the unrestricted area surrounding the PULSTAR reactor:

(i) The upper floors of the D.H. Hill Library that are at the same elevation as the top of the ventilation stack, located 140 meters northwest of the stack

(ii) The nearest campus student housing, Carroll Hall, located 260 meters southwest of the stack, particularly the upper floors that are at the same elevation as the top of the stack

(iii) The nearest permanent residence, located at least 200 meters from the stack

(iv) The maximum off-site ground location at approximately 50 meters from the stack

The whole body, skin, and thyroid doses at the various off-site locations of interest are determined using the dosimetry models from Regulatory Guide 1.109[13-11].

Fuel Pin Annuli radioiodine and radiobromine inhalation dose calculations

The annual organ dose from inhalation of radioiodines and radiobromines in air were assessed using the following equation as given in Regulatory Guide 1.109:

$$D_{ja}^{A} = R_a \sum_i \chi_i(r,\theta) DFA_{ija} \quad and \quad \chi_i(r,\theta) = 3.17(10^4) Q_i \left[\frac{\chi}{Q}\right](r,\theta)$$

where:

D_{ja}^{A} = the dose to organ j of an individual in the age group a at a location (r,θ) due to inhalation (mrem/y)

DFA_{ija} = the inhalation dose factor from radionuclide i, organ j, and age group a (mrem/pCi)

R_a = the annual air intake for individuals in the age group a (m^3/y)

$\chi_i(r,\theta)$ = annual average concentration of radionuclide i in the air at location (r,θ) with units (pCi/m^3)

Q_i = the release rate of nuclide i to the atmosphere (Ci/y)

$[\chi/Q]^D(r,\theta)$ = the calculated atmospheric dispersion factor (s/m^3)

3.17×10^4 = the number of pCi/Ci divided by the number of seconds in a year.

Since the area immediately around the facility is a university campus and business community, the "adult class" was used in the non-noble gas inhalation dose calculations, and therefore:

$$R_a = 8000 \ m^3/y$$

Radioiodine and radiobromine inhalation doses are presented in Table 13-4 for the four off-site locations of interest.

Table 13-4

Radioiodine and Radiobromine Off-Site Inhalation Doses From Three Fuel Pin Failure

Inhalation Doses (mrem)			
Location (m)	Description	Total Body	Thyroid
50	Maximum Ground Location	$1.2(10^{-6})$	$6.4(10^{-4})$
200	Nearest Housing	$1.1(10^{-6})$	$5.8(10^{-4})$
260 †	Nearest Dormitory	$2.1(10^{-5})$	$1.1(10^{-2})$
140 †	D.H. Hill Library	$5.0(10^{-5})$	$2.7(10^{-2})$

† Receptor elevation is 30 meters
Note: Multiply mrem by $1(10^{-5})$ for Sv

Fuel Pin Annuli Noble Gas Inventory Whole Body and Skin Dose Calculations

The results presented in this section include the dose rate to a receptor in the unrestricted area from a hypothetical rupture of three fuel pins. This calculation was performed using the methodology described in Appendix 13-1. The dose calculation for a ground receptor includes the contribution from the overhead plume. Elevated receptor dose calculations reduce to the semi-infinite cloud submersion model. The limiting meteorological conditions from the ^{41}Ar results presented in FSAR Section 10 are applied to this calculation.

The fission noble gas inventory collected in the annuli of the fuel rod was previously presented in Table 13-2 and only includes ^{88}Kr, ^{133}Xe, and ^{138}Xe. Whole body and skin doses from the combined fission noble gas inventory are presented in Tables 13-5 through 13-8. The fuel pin failures were assumed to occur simultaneously leading to a release duration of approximately 2.4 hours.

The analysis has addressed the dose to a receptor in the unrestricted areas proximate to the NCSU PULSTAR Reactor facility. The gaseous effluents considered were the fission noble gas inventory of a postulated failure of three fuel pins. The parametric results offers sufficient combinations of meteorological conditions and receptor locations to answer the question of dose to the unrestricted area.

Worse case results of the NCSU analysis from all radionuclides are summarized in Table 13-9 and show that postulated releases are a small fraction of the 10 CFR 20 dose limits and are considered to be radiologically insignificant.

TABLE 13-5

SELECTED WHOLE-BODY FISSION NOBLE GAS DOSES (mrem)
FOR A 2.4 HOURS RELEASE
GROUND RECEPTOR AT 50 M (Heff Off - Confinement Mode)

WINDSPEED (M/S)	PASQUILL CLASS	D^{γ} (Xe-133) ($*10^{-2}$)	D^{γ} (Xe-138) ($*10^{-2}$)	D^{γ} (Kr-88) ($*10^{-2}$)	D^{γ} (Total) ($*10^{-2}$)
0.5	A	0.170	0.090	0.213	0.473
0.5	B	0.170	0.105	0.249	0.524
1.0	A	0.092	0.047	0.107	0.246
1.0	B	0.108	0.053	0.125	0.286
2.0	A	0.046	0.024	0.054	0.124
2.0	B	0.054	0.028	0.063	0.145
2.0	C	0.074	0.038	0.086	0.198
2.0	E	0.148	0.076	0.171	0.395
2.0	F	0.170	0.114	0.257	0.541
4.0	B	0.027	0.014	0.031	0.072
4.0	C	0.037	0.019	0.043	0.099
4.0	D	0.057	0.030	0.066	0.153
4.0	E	0.074	0.038	0.086	0.198
6.0	C	0.025	0.013	0.029	0.067
6.0	D	0.038	0.020	0.044	0.102
6.0	F	0.074	0.038	0.086	0.198
9.0	C	0.017	0.009	0.019	0.045
9.0	D	0.025	0.013	0.029	0.067
10.0	F	0.044	0.023	0.051	0.118

Note: Multiply mrem by $1(10^{-5})$ for Sv

TABLE 13-6

SELECTED WHOLE-BODY FISSION NOBLE GAS DOSES (mrem)
FOR A 2.4 HOURS RELEASE
GROUND RECEPTOR AT 200 M (Heff Off - Confinement Mode)

WINDSPEED (M/S)	PASQUILL CLASS	D^{γ} (Xe-133) (*10^{-2})	D^{γ} (Xe-138) (*10^{-2})	D^{γ} (Kr-88) (*10^{-2})	D^{γ} (Total) (*10^{-2})
0.5	A	0.059	0.040	0.117	0.216
0.5	B	0.059	0.028	0.080	0.167
1.0	A	0.039	0.031	0.080	0.150
1.0	B	0.035	0.016	0.040	0.091
2.0	A	0.020	0.008	0.040	0.068
2.0	B	0.018	0.009	0.020	0.047
2.0	C	0.024	0.012	0.027	0.063
2.0	E	0.045	0.022	0.051	0.118
2.0	F	0.059	0.032	0.076	0.167
4.0	B	0.009	0.005	0.010	0.024
4.0	C	0.012	0.006	0.014	0.032
4.0	D	0.018	0.009	0.020	0.047
4.0	E	0.022	0.011	0.026	0.059
6.0	C	0.008	0.004	0.009	0.021
6.0	D	0.012	0.006	0.013	0.031
6.0	F	0.022	0.011	0.026	0.059
9.0	C	0.005	0.003	0.006	0.014
9.0	D	0.008	0.004	0.009	0.021
10.0	F	0.013	0.007	0.015	0.035

Note: Multiply mrem by $1(10^{-5})$ for Sv

TABLE 13-7

SELECTED SKIN FISSION GAS DOSES (mrem)
FOR A 2.4 HOURS RELEASE
RECEPTOR AT 50 M (Heff Off - Confinement)

WINDSPEED (M/S)	PASQUILL CLASS	D^6 (Xe-133) ($*10^{-2}$)	D^6 (Xe-138) ($*10^{-2}$)	D^6 (Kr-88) ($*10^{-2}$)	D^6 (Total) ($*10^{-2}$)
0.5	A	0.176	0.093	0.220	0.489
0.5	B	0.176	0.109	0.257	0.542
1.0	A	0.096	0.048	0.110	0.254
1.0	B	0.112	0.056	0.129	0.297
2.0	A	0.048	0.025	0.055	0.128
2.0	B	0.056	0.029	0.065	0.150
2.0	C	0.077	0.039	0.089	0.205
2.0	E	0.152	0.078	0.177	0.407
2.0	F	0.176	0.117	0.264	0.557
4.0	B	0.028	0.014	0.032	0.074
4.0	C	0.038	0.020	0.044	0.102
4.0	D	0.059	0.030	0.068	0.163
4.0	E	0.076	0.040	0.088	0.211
6.0	C	0.026	0.013	0.030	0.074
6.0	D	0.039	0.020	0.045	0.113
6.0	F	0.076	0.040	0.088	0.220
9.0	C	0.017	0.009	0.020	0.051
9.0	D	0.026	0.014	0.030	0.078
10.0	F	0.046	0.024	0.048	0.137

Note: Multiply mrem by $1(10^{-5})$ for Sv

TABLE 13-8

SELECTED SKIN FISSION GAS DOSES (mrem)
FOR A 2.4 HOUR RELEASE
RECEPTOR AT 200 M (Heff Off - Confinement Mode)

WINDSPEED (M/S)	PASQUILL CLASS	D^{6} (Xe-133) ($*10^{-2}$)	D^{6} (Xe-138) ($*10^{-2}$)	D^{6} (Kr-88) ($*10^{-2}$)	D^{6} (Total) ($*10^{-2}$)
0.5	A	0.098	0.052	0.141	0.291
0.5	B	0.099	0.040	0.102	0.241
1.0	A	0.060	0.039	0.092	0.191
1.0	B	0.055	0.024	0.052	0.131
2.0	A	0.030	0.011	0.047	0.088
2.0	B	0.028	0.013	0.026	0.067
2.0	C	0.032	0.015	0.032	0.079
2.0	E	0.046	0.022	0.053	0.121
2.0	F	0.061	0.033	0.079	0.173
4.0	B	0.014	0.007	0.013	0.034
4.0	C	0.016	0.008	0.016	0.040
4.0	D	0.018	0.009	0.021	0.048
4.0	E	0.023	0.012	0.027	0.062
6.0	C	0.011	0.005	0.011	0.027
6.0	D	0.012	0.006	0.014	0.032
6.0	F	0.023	0.012	0.026	0.061
9.0	C	0.007	0.004	0.007	0.018
9.0	D	0.008	0.004	0.009	0.021
10.0	F	0.014	0.007	0.016	0.037

Note: Multiply mrem by $1(10^{-5})$ for Sv

Table 13-9

Estimated Doses from Postulated Three Fuel Pin Failure

Fission Gas Doses (mrem)				
Location (m)	Description	Whole-Body	Thyroid	Skin
50	Maximum Ground Location	$5.4(10^{-3})$	$6.4(10^{-4})$	$5.6(10^{-3})$
200	Nearest Housing	$2.2(10^{-3})$	$5.8(10^{-4})$	$2.9(10^{-3})$
260 †	Nearest Dormitory	$7.8(10^{-3})$	$1.1(10^{-2})$	$7.8(10^{-3})$
140 †	D.H. Hill Library	$2.2(10^{-2})$	$2.7(10^{-2})$	$2.2(10^{-2})$

† Receptor elevation is 30 meters
Note: Multiply mrem by $1(10^{-5})$ for Sv

Maximum Dose and Corresponding Ground Location

The relatively low stack elevation of the PULSTAR facility results in the maximum dose location to be near the stack at the site boundary. However, the Burlington Engineering Laboratory provides shielding from the plume. The maximum ground dose location will be at approximately 50 meters from the stack in the absence of attenuation from permanent structures.

Table 13-9 details the doses that would be expected to occur at the various off-site locations of interest as a result of a three fuel pin failure scenario. This table also indicates that the off-site consequences are quite low, even when all of the conservative assumptions are included, i.e., no allowance for radioactive decay, no in-leakage during the purging of the Reactor Bay, and no credit for dilution by the south wing ventilation system. In summary, the postulated failure of three fuel pins (at their maximum burn-up level) with the release of their annuli gases results in off-site doses that are low relative to naturally occurring and other man-made doses.

Activity in the Reactor Bay air also represents a source of direct radiation. The direct radiation from the Reactor Bay air was calculated assuming that all the fission product gases in the annular gap of 63 fuel pins (approximately 10 percent of the core loading) are released into the Reactor Bay. For this calculation, it is assumed the sources remain in the Reactor Bay and distributed within the bay air volume. The Reactor Bay walls were assumed to be equivalent to 6 inches (15.2 cm) of ordinary concrete. Using the above assumptions, the direct radiation level outside the Reactor Bay wall was calculated to be 0.4 mR/h (4 μSv/h). This calculation was made using computer code Microshield 4[13-12] which utilizes the point kernel method.

The calculation of direct radiation dose from gaseous fission products released from 63 fuel pins indicates that radiation levels to individual members of the public outside the Reactor Building are within 10 CFR 20 limits.

Radiation Doses Inside Reactor Bay from Three Fuel Pin Failure Accident

Estimates of external doses and internal doses associated with the release of fission products into the reactor bay air volume from the three fuel pin failure accident may be made based on methodologies given in EPA 400[13-13], Regulatory Guide 8.34[13-14], or Regulatory Guide 1.109. These methodologies were modified for the conditions assumed in the analysis of this postulated accident. External dose from submersion within the radioactive cloud inside the reactor bay may be estimated by the following equations:

From EPA 400:

$$H=\sum_i DCF_i \times C_i \times T \times 10^3$$

where, H is effective dose in mrem, 10^3 converts rem to mrem
DCF_i is Dose Conversion Factor for radionuclide i given in EPA 400 Table 5-3
C_i is the concentration of radionuclide i in μCi/ml
T is duration in hours

From Regulatory Guide 8.34:

$$H=\sum_i \frac{C_i}{DAC_i} \times \frac{T}{2000} \times 5 \times 10^3$$

where, 5 is effective dose in mrem, 10^3 converts rem to mrem
DAC_i is the Derived Air Concentration for radioactive noble gas i given in 10 CFR 20 Appendix B Table 1 for effective dose

From Regulatory Guide 1.109:

$$H = S_F \times \sum_i DFB_i \times C_i \times T$$

$$H = 1.11 \times S_F \times \sum_i DF_i^{\gamma} \times C_i \times T + \sum_i DFS_i \times C_i \times T$$

where, S_F is the shielding factor from surrounding structures which does not apply for this case and is therefore equal to unity

DFB_i, DF_i^{γ}, and DFS_i are the dose factors for radioactive noble gas i given in Regulatory Guide 1.109 Table B-1

C_i is concentration in pCi/m^3

T is duration in years

Note: 1 μCi equals 10^6 pCi and 1 m^3 equals 10^6 ml

These equations were solved for the three fuel pin failure accident and results are given in Table 13-10. Duration, T, of the event has been reported previously as 2.4 h. C_i values for these equations were determined by dividing data from Table 13-2 by $2.4(10^9)$ ml (free air volume of the reactor bay) and then multiplying by 10^6 to convert Ci to μCi.

Table 13-10
External Dose Within Reactor Bay From Submersion By Three Fuel Pin Failure

NUCLIDE	EPA 400 (mrem)	Reg Guide 8.34 (mrem)	Reg. Guide 1.109 Body (mrem)	Reg. Guide 1.109 Skin (mrem)
Kr-88	4.8	4.6	6.2	3.5
Xe-133	1.1	1.3	1.8	1.9
Xe-138	1.4	1.2	1.9	1.3
I-131	0.18	NA	NA	NA
I-133	0.34	NA	NA	NA
TOTAL	7.7	7.1	9.9	6.7

Note: Multiply mrem by 10^{-5} for Sv

Therefore, external dose to an individual inside the reactor bay as a result of the three fuel pin failure accident are conservatively calculated to be 10 mrem for both effective dose (whole body dose or deep dose-equivalent) and shallow dose-equivalent (skin dose).

Internal dose to an individual inside the reactor bay from inhalation of radioiodines and radiobromines may be estimated by the following equations:

From EPA 400:

$$H = \sum_i DCF_i \times C_i \times T \times 10^3$$

where, DCF_i is given in EPA 400 Table 5-4 for effective dose and committed doses

From Regulatory Guide 8.34:

$$H = \sum_i \frac{C_i}{DAC_i} \times \frac{T}{2000} \times 5 \times 10^3$$

$$H = \sum_i \frac{C_i}{DAC_i} \times \frac{T}{2000} \times 50 \times 10^3$$

where, 5 is effective dose in rem and DAC_i is given in 10 CFR 20 Appendix B Table 1 for effective dose for radionuclide i, 10^3 converts rem to mrem

50 is committed dose in rem and DAC_i is given in 10 CFR 20 Appendix B Table 1 for committed dose for radionuclide i, 10^3 converts rem to mrem

From Regulatory Guide 1.109:

$$H = \sum_i C_i \times B(t) \times DFA_{ija} \times T$$

where, B(t) is the adult working breathing rate of derived from Regulatory Guide 8.34 in ml/y rather than the adult resting value given in Regulatory Guide 1.109 Table E-5; B(t) = $1.2(10^6)$ ml/h = [$2.4(10^9)$ ml per y / 2000 h per y]

DFA_{ija} is the dose factor for radionuclide i, adult age group a, and organ j from inhalation given in Regulatory Guide 1.109 Table E-7

These equations were solved for the three fuel pin failure accident and results are given in Table 13-11. Duration, T, of the event has been reported previously as 2.4 h. C_i values for these equations were determined by dividing data from Table 13-2 by $2.4(10^9)$ ml (free air volume of the reactor bay) and then multiplying by 10^6 to convert Ci to μCi.

Finally, external deep dose is added to the internal doses to give total effective dose (whole body) and committed dose to the thyroid. Therefore, total doses to an individual inside the reactor bay as a result of the three fuel pin failure accident are conservatively calculated to be 50 mrem effective dose and $2(10^3)$ mrem committed dose to the thyroid. Use of respiratory protection devices, engineering controls, and decay were not been considered in the calculation of these doses. Comparison of calculated doses to 10 CFR 20 limits is made in Table 13-12. The calculated doses inside the reactor bay associated with the postulated three fuel pin failure accident are a small fraction of the occupational regulatory limits.

Table 13-11
Internal Dose Within Reactor Bay By Inhalation From Three Fuel Pin Failure

NUCLIDE	EPA 400 Effective Dose (mrem)	EPA 400 Committed Dose to the Thyroid (mrem)	Reg Guide 8.34 Effective Dose (mrem)	Reg Guide 8.34 Committed Dose to the Thyroid (mrem)	Reg. Guide 1.109 Thyroid Dose (mrem)
I-131	31	$1.0(10^3)$	25	$1.0(10^3)$	$1.4(10^3)$
I-133	7.1	$2.1(10^2)$	6.0	$2.4(10^2)$	$3.1(10^2)$
Br-83	NA	NA	0.14	NA	NA
Br-84	NA	NA	$1.8(10^{-3})$	NA	NA
TOTAL	38	$1.2(10^3)$	31	$1.2(10^3)$	$1.7(10^3)$

Note: Multiply mrem by 10^{-5} for Sv

Table 13-12
Calculated Doses Inside the Reactor Bay From Three Fuel Pin Failure

10 CFR 20 DOSE QUANTITY	CALCULATED VALUE (mrem)	10 CFR 20 LIMIT (mrem)
Deep Dose-Equivalent (external)	10	5000
Shallow Dose-Equivalent (external skin or extremity)	10	50,000
Eye Dose-Equivalent (external)	10 (inferred from deep and shallow dose)	15,000
Committed Effective Dose-Equivalent (internal)	40	5000
Committed Dose-Equivalent (internal)	2000 (thyroid)	50,000
Total Effective Dose-Equivalent (external + internal)	50	5000
Total Organ Dose-Equivalent (external + internal)	2000 (thyroid)	50,000

Note: Multiply mrem by 10^{-5} for Sv

13.2.1.5 Heat Exchanger Pressure Boundary Breach

The PULSTAR reactor cooling system utilizes a tube-shell water-to-water heat exchanger to transfer the reactor energy from the primary system to the secondary system. The heat exchanger is described in Section 4. Water chemistry is controlled in both the primary and secondary systems to minimize corrosion and impurities. Primary system integrity is monitored by primary and secondary water sampling along with a primary system hydrostatic test. The sampling program has demonstrated primary water radionuclide concentrations to be well within the 10 CFR 20 limits for release to the sanitary sewer and require dilution by approximately a maximum volumetric factor of four to meet unrestricted release limits. Table 13-13 presents typical radiochemistry analysis results for the PULSTAR primary water at 1 MW.

Table 13-13
Primary Water Radioactivity Analysis

Nuclide	Average Concentration μCi/ml	Unrestricted † Area Limit μCi/ml	Fraction of Limit
Ag-110m	$3.0(10^{-6})$	$6.0(10^{-6})$	0.5
Cl-38	$1.0(10^{-4})$	$3.0(10^{-4})$	0.3
Co-60	$2.0(10^{-6})$	$3.0(10^{-6})$	0.7
Fe-59	$3.0(10^{-6})$	$1.0(10^{-5})$	0.3
H-3	$2.0(10^{-4})$	$1.0(10^{-3})$	0.2
Mn-54	$7.0(10^{-7})$	$3.0(10^{-5})$	0.02
Na-24	$6.0(10^{-5})$	$5.0(10^{-5})$	1.0
Zn-65	$3.0(10^{-6})$	$5.0(10^{-6})$	0.6
		Σ	3.6

† 10CRF20 Appendix B, Table 2
Note: 1μCi equals $3.7(10^4)$ Bq and 1 ml equals 10^{-6} m^3

The PULSTAR Pool water elevation is higher than the secondary cooling water basin creating a passive potential energy gradient. This gradient provides the driving force in a pressure boundary breach for a primary side to secondary side water leak with or without forced primary flow. However, operating the secondary pump causes the shell side pressure of the heat exchanger to exceed the primary side pressure. A heat exchanger pressure boundary breach while the secondary pump is operating would result in a mass transfer from the secondary side to the primary side.

The design of the facility prevents the core from becoming uncovered due to a heat exchanger pressure boundary breach. This is due to the cooling tower basin elevation which is approximately 10 feet (3.3 m) higher than the PULSTAR core.

A heat exchanger leak is unlikely due to the construction and materials of the heat exchanger, primary and secondary water chemistry, and a surveillance program. If a pressure boundary breach were to suddenly appear during power operations, secondary water would enter the primary system. This type of leak is easily identified since secondary water contaminants would be activated and collected in the primary water demineralizer system ion-exchange resin causing an abnormal radiation level increase. There would be no primary system inventory leaving the PULSTAR restricted area for this scenario.

A pressure boundary breach during low power operation when the secondary system is not operating or while the reactor is secured would result in a primary to secondary leak. Detection of the primary boundary breach would be from a decrease in pool water level. A double-ended guillotine break of one tube in the heat exchanger with the primary pump operating would result in an estimated leak rate of 4.5 gpm ($2.8(10^{-1})\ell \cdot s^{-1}$). Pool water level will decrease at a maximum rate of 6 inches$\cdot h^{-1}$ ($4.2(10^{-3})$ cm$\cdot s^{-1}$). The abnormal (low) pool water level alarm set-point at -12 inches (15.2 cm) would be reached in two hours. The low pool level scram/primary pump trip set-point at -36 inches (91.4 cm) would be reached 5 hours following the breach. The estimated three hours between the first abnormal pool level indication and the second is sufficient time for the PULSTAR staff to respond from off-site, investigate, and mitigate the leak. Thus, the maximum duration for this event would be five hours.

A primary to secondary system leak over a five hour period releases approximately 1080 gallons ($4.09(10^3)\ \ell$) containing the typical radionuclide concentrations in Table 13-13. These concentrations are below 10 CFR 20 limits for releases to the sanitary sewer. However, the cooling tower basin overflow drains to the storm sewer where unrestricted release limits apply. The campus storm sewer discharges into the Rocky Branch Creek and ultimately into the Neuse River. The minimum volumetric flow rate of the Neuse River is $8.7(10^7)$gallons per day ($3.8(10^3)\ \ell \cdot s^{-1}$) from FSAR Section 2. The release would then be conservatively diluted by the following factor:

$$Dilution\ \ Factor \simeq \frac{3.8(10)^3\ \ell \cdot s^{-1} \times 5h \times 3600\ s \cdot h^{-1}}{4.09(10)^3\ \ell} \simeq 1.7(10)^4$$

Internal radiation doses associated with this event may be calculated based on the methodologies given in Regulatory Guide 8.34 or Regulatory Guide 1.109. These methodologies were modified for the conditions assumed in the analysis of this postulated event. Internal dose calculations were made using the radionuclide concentrations given in Table 13-13, assumed dilution factor, release volume, and event duration using following equations for the drinking (potable) water pathway:

From Regulatory Guide 8.34:

$$H = \sum_i 5 \times 10^3 \times \frac{I_i}{ALI_i}$$

$$I_i = C_i \times M_p \times e^{-\lambda_i t_p} \times 7.3(10^5) \times T$$

where, H is effective dose in mrem, 10^3 converts rem to mrem
I_i is intake activity of radionuclide i in μCi
C_i is release concentration from Table 13-13 in μCi/ml
ALI_i is the Annual Limit on Intake of radionuclide i associated with a committed effective dose of 5 rem given in 10 CFR 20 Appendix B Table 1
λ_i is the decay constant, in h^{-1}
t_p is the transit time between release and ingestion, which is assumed to be 5 h for this event
M_p is the reciprocal of the dilution factor

$$M_p = 1/1.7(10^4) = 5.9(10^{-5})$$

T is assumed to be 1 year
$7.3(10^5)$ ml is the annual water consumption of reference man

From Regulatory Guide 1.109:

$$R_{apj} = 1100 \times \frac{M_p \times U_{ap}}{F} \times \sum_i Q_i \times D_{aipj} \times e^{-\lambda_i t_p}$$

$$H = R_{apj} \times T$$

where, R_{apj} is the annual dose in mrem to an individual from age group a from pathway p to organ j
1100 is a unit conversion constant
U_{ap} is the intake rate by an individual of age group a associated with pathway p, in ℓ/y, given in Table E-5 for the maximum exposed individual
F is the liquid effluent flow rate, in ft^3/s
$F = (4.03(10^3)\ \ell)(1000\ ml/\ell)(1\ ft^3/28{,}318\ ml)/(5h \times 3600\ s/h)$
$F = 7.9(10^{-3})\ ft^3/s$ for this event
Q_i is the activity release rate of radionuclide i in Ci/y
D_{aipj} is the ingestion dose factor specific to a given age group a, radionuclide i, potable water pathway p, and organ j in mrem/pCi, given in Regulatory Guide 1.109 Tables E-11, E-12, E-13, and E-14
H is total dose in mrem

These two equations were solved for the heat exchanger pressure boundary breach event for each age group. Results from the method described in Regulatory Guide 8.34 are given in Table 13-14. Results from the method described in Regulatory Guide 1.109 are given in Table 13-15. Table 13-15 results exclude short-lived ^{38}Cl.

Results given in Table 13-14 and Table 13-15 assume that the sole source of drinking water is the Neuse River and that the total annual consumption of water occurs five hours after the postulated event occurs. Comparison of Table 13-14 and Table 13-15 indicates that Regulatory Guide 8.34 is yields conservative, but negligible, results. Table 13-15 indicates that the GI-LLI is limiting for each age group and that the adult GI-LLI receives the maximum dose. Therefore, it can be concluded that a heat exchanger primary pressure boundary breach will not result in core uncovery and the dosimetric consequences from the postulated primary water release is insignificant.

Table 13-14

Internal Doses From Drinking Water Pathway From Heat Exchanger Pressure Boundary Breach Event As Analyzed Using Regulatory Guide 8.34

NUCLIDE	DECAY FRACTION	INTAKE (μCi)	ORAL ALI (μCi)	Committed Effective Dose (mrem)
Ag-110m	1	$1.3(10^{-4})$	$5(10^{2})$	$1.3(10^{-3})$
Cl-38	$3.7(10^{-3})$	$1.6(10^{-5})$	$3(10^{4})$	$2.7(10^{-6})$
Co-60	1	$8.6(10^{-5})$	$2(10^{2})$	$2.2(10^{-3})$
Fe-59	1	$1.3(10^{-4})$	$8(10^{2})$	$8.0(10^{-4})$
Mn-54	1	$3.0(10^{-5})$	$2(10^{3})$	$7.5(10^{-5})$
Na-24	0.79	$2.0(10^{-3})$	$4(10^{3})$	$2.6(10^{-3})$
Zn-65	1	$1.3(10^{-4})$	$4(10^{2})$	$1.6(10^{-3})$
H-3	1	$8.6(10^{-3})$	$8(10^{4})$	$5.4(10^{-4})$
TOTAL		$1.1(10^{-2})$	$5(10^{2})$	$9(10^{-3})$

Note: Multiply mrem by 10^{-5} for Sv

Table 13-15

Internal Doses to Various Organs and Age Groups From Drinking (Potable) Water From Heat Exchanger Pressure Boundary Breach Event As Analyzed Using Regulatory Guide 1.109

ORGAN	INFANT (mrem)	CHILD (mrem)	TEEN (mrem)	ADULT (mrem)
BONE	$6.3(10^{-6})$	$6.2(10^{-6})$	$2.4(10^{-6})$	$2.6(10^{-6})$
LIVER	$1.0(10^{-5})$	$8.8(10^{-6})$	$4.1(10^{-6})$	$4.5(10^{-6})$
BODY	$8.2(10^{-6})$	$7.7(10^{-6})$	$3.2(10^{-6})$	$3.5(10^{-6})$
THYROID	$5.9(10^{-6})$	$5.3(10^{-6})$	$2.2(10^{-6})$	$2.5(10^{-6})$
KIDNEY	$7.0(10^{-6})$	$6.6(10^{-6})$	$2.9(10^{-6})$	$3.2(10^{-6})$
LUNG	$6.4(10^{-6})$	$5.7(10^{-6})$	$2.4(10^{-6})$	$2.7(10^{-6})$
GI-LLI	$1.0(10^{-5})$	$1.0(10^{-5})$	$8.4(10^{-6})$	$1.2(10^{-5})$

Note: Multiply mrem by 10^{-5} for Sv

13.2.1.6 Failure of a Fueled Experiment

Experiments containing fissile material may be performed in the experimental facilities of the PULSTAR reactor. These experiments shall be performed in accordance with the limitations and specifications of this safety analysis to ensure that a total failure of the experiment shall be bounded by the fuel pin clad failure analysis described previously in this section. That is, the fission product inventory in a fueled experiment shall be bounded by the Fuel Pin Clad Failure analysis.

Source Term

A fueled experiment at the NCSU PULSTAR facility shall be limited to contain only uranium-235 as the fissile material or fuel. The fission products directly from the fission process in the experiment constitute the source term for this analysis. A fueled experiment may be performed at a dry or wet experimental facility. The fueled experiment failure analysis presented in this section will be bounded by the fuel pin clad failure analysis.

The source term given in Table 13-16 is directly from the Fuel Pin Clad Failure analysis. It should be noted that radioiodines and radiobromines activities correspond to an underwater radionuclide release and were corrected for retention in water. The application of the fuel pin clad failure source term in this analysis results in two conservative conditions:

- Lower noble gas inventories would be produced by limiting the fission product production rate to the radioiodine and radiobromine production rates

- An underwater failed fueled experiment would result in reduced airborne radioiodine and radiobromine concentrations by retention in water of these nuclides

Table 13-16

Summary of Noble Gas and Halogen Production from Fuel Pin Clad Failure Analysis

Isotope	Pin Annuli Activity (Ci)†	λ (s^{-1})†	Cumulative Yield (%)††
Br-83	$1.72(10^{-3})$	$8.05(10^{-5})$	0.51
Br-84	$1.43(10^{-5})$	$3.63(10^{-4})$	0.90
Kr-88	$3.78(10^{-3})$	$6.78(10^{-5})$	3.57
I-131	$8.14(10^{-4})$	$9.98(10^{-7})$	3.10
I-133	$9.80(10^{-4})$	$9.26(10^{-6})$	6.50
Xe-133	$5.43(10^{-2})$	$1.53(10^{-6})$	6.62
Xe-138	$1.94(10^{-4})$	$8.15(10^{-4})$	5.74

† Table 13-1 and 13-2 values
†† ANL-5800, Reactor Physics Constants, 2nd Edition, 1963, and Reference 13-6

A calculation for the amount of fissile material and total power generated in the fueled experiment can now be performed having the source term established with Table 13-16.

Mass of Uranium-235 and Total Power Calculation

The fission of one Uranium-235 atom produces two fission products. The production rate of a specific fission product is given by:

$$\frac{dN_i}{dt} = Formation\ Rate \quad - \quad Destruction\ Rate$$

The nuclide *i* will be assumed to be produced directly from thermal fission of Uranium-235 with a yield γ_i and the only destruction mechanism assumed will be decay; that is, transmutation is not considered in order to maximize the radionuclide concentration of interest.

$$\frac{dN_i}{dt} = \gamma_i N_f \sigma_f \phi_t \; - \; \lambda_i N_i$$

$$N_i(t) = \frac{\gamma_i N_f \sigma_f \phi_t}{\lambda_i}\left(1 - e^{-\lambda_i t}\right) \quad or \quad A_i(t) = \gamma_i N_f \sigma_f \phi_t \left(1 - e^{-\lambda_i t}\right)$$

The analysis assumes a fueled experiment containing 400 milligrams of uranium-235 is subjected to thermal neutron fluence rates ranging from 10^{10} to 10^5 neutrons·cm^{-2}·sec^{-1} for continuous total exposure times ranging from 10^0 to 10^5 seconds. A sample calculation for the Br-84 activity produced in the fueled experiment is provided below. Table 13-17 summarizes the total activity generated for the isotopes in Table 13-16 for the sample calculation. Figure 13-1 provides the acceptable operating range for a 400 milligram fueled experiment.

$$A(10^5) = 0.009 \times 10^8 \times 540(10^{-24}) \times 1.025(10^{21}) \times \left(1 - e^{-3.63(10^{-4}) \times (10^5)}\right)$$

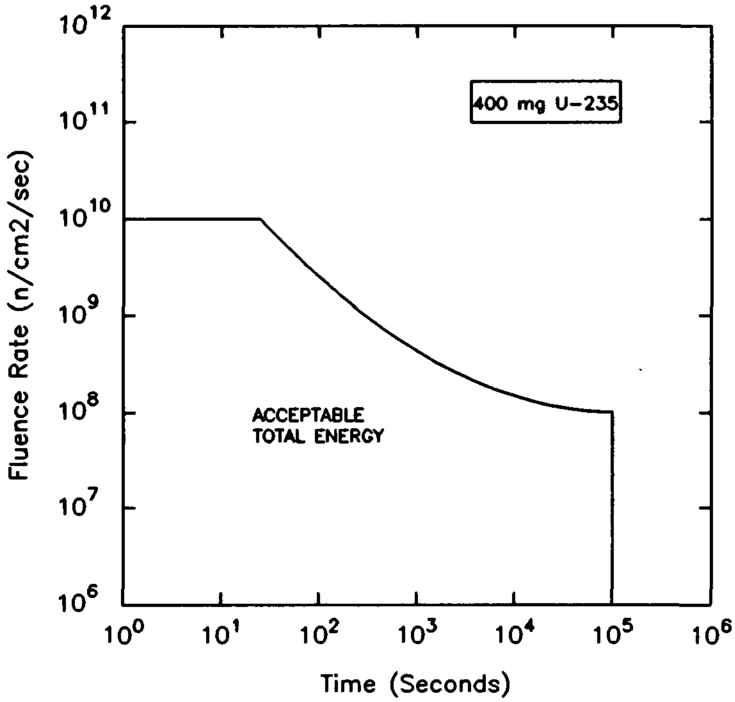
$$A(10^5) = 4.98(10^5)\; dps \;=\; 1.35(10^{-5})\; Curies$$

Table 13-17

Calculated Activities for a 400 milligram Fueled Experiment at a Fluence Rate of 10^8 n·cm^{-2}·s for 10^5 seconds

Isotope	Activity (Ci)	Activity Limit (Ci)
Br-83	7.63(10^{-6})	1.72(10^{-3})
Br-84	1.35(10^{-5})	1.43(10^{-5})
I-131	4.40(10^{-6})	8.14(10^{-4})
I-133	5.87(10^{-5})	9.80(10^{-4})
Xe-133	1.40(10^{-5})	5.43(10^{-2})
Xe-138	8.59(10^{-5})	1.94(10^{-4})
Kr-88	5.33(10^{-5})	3.78(10^{-3})

Figure 13-1



Total Energy Release and Power

The fueled experiment energy release rate or power and total energy release for a continuous 10^5 seconds run is obtained below.

$$\begin{aligned} \textit{Fission rate} &= N_F \sigma_f \phi_t \\ &= 1.025(10^{21}) \times 540(10^{-24}) \times 10^8 \\ &= 5.54(10^7)\ \textit{fissions} \cdot s^{-1} \end{aligned}$$

$$\begin{aligned} \textit{Energy release rate} &= 200\ MeV \cdot \textit{fission}^{-1} \times 5.54(10^7)\ \textit{fissions} \cdot s^{-1} \\ &= 1.11(10^{10})\, MeV \cdot s^{-1} \\ &= 1.77(10^{-3})\ \textit{watt} \end{aligned}$$

$$\textit{Total energy released} = 1.77(10^{-3})\, J \cdot s^{-1} \times 10^{5}\, s$$
$$= 1.77(10^{2})\, \textit{Joules}$$

The analysis demonstrates that fueled experiments may be performed in experimental facilities of the PULSTAR reactor with the following conditions and limitations:

- o The maximum mass of uranium-235 is limited to 400 milligrams.
- o The thermal power (or fission rate) generated in the experiment is not greater than 5.5×10^{8} fissions per second (1.77 milliwatt).
- o The total exposure of the material is not greater than the limits set in Figure 13-1.
- o The reactor shall not be operated with a fueled experiment unless the ventilation is operated in the confinement mode.
- o The specifications pertaining to reactor experiments, detailed in Section 3.7 Limitations of Experiments, apply to fueled experiments.

13.2.2 Excursion Accidents

13.2.2.1 Fuel Loading Accident

The fuel loading accident is the MCA for excursion type accidents associated with the NCSU PULSTAR Reactor. The maximum worth of a fuel assembly for the existing 5 by 5 Graphite Reflected Core No. 3 has been measured to be 1.130%Δk/k (1130 pcm) (see Section 3.2.3.5.3.5). It is estimated that a twenty five assembly core of PULSTAR fuel is capable of absorbing a step input of more than 1.59%Δk/k (1590 pcm) without failure of fuel pin cladding. The 1.59%Δk/k (1590 pcm) reactivity insertion would yield an estimated 58 MW·s pulse (see Figure 3-20) which has been established as the Safety Limit for a pulse as detailed in Section 3.2.4.2.3.

To demonstrate this, it is assumed that erroneous handling of the fuel does occur under the following conditions:

(1) The core has been loaded in the optimum configuration.

(2) The reactor is critical.

(3) A fuel assembly is dropped from a height of two feet above the core and subsequently enters the optimum location.

The optimum position for the 5 by 5 Graphite Reflected Core No. 3 has a measured worth of 1.13%Δk/k (1130 pcm). To get the upper limit of this accident, it is assumed the

reactivity is a step input. The resulting pulse has an estimated energy release of 40 MW·s (see Figure 3-20) and a maximum specific energy production of 325 watt·s/gram. This is less than the specific energy density of 400 watt·s/gram design criterion. A fuel loading accident therefore does not jeopardize the safety of operating personnel or the general public.

For future core arrangements, the maximum fuel assembly worth shall be measured during start-up testing and limited to 1.59%Δk/k (1590 pcm).

13.2.2.2 Start-up Accident

The following accident analysis was made to determine the results of the continuous rod withdrawal of all three control rods from a subcritical core through criticality up to shutdown by the high level neutron flux scrams. For this accident the following sequence of events is assumed:

(1) Console operator has failed to return the Linear Power Channel range selector to the most sensitive or mid-scale position. This would mean that the high neutron flux scrams would not be set at the most sensitive or on-scale position prior to withdrawal of control rods to bring the core critical, and thus a high neutron flux scram would occur at the top of this power range.

(2) It is assumed that the three control rods are withdrawn in gang and the gang rate at which reactivity is inserted in the core is 0.1%Δk/k/s (100 pcm/s) as the core attains criticality. This reactivity rate is approximately a factor of two greater than that observed on the 25 fuel assembly core. Refer to Section 3 for further details.

(3) The power level of the core is assumed to be 1 milliwatt at the time the reactor reaches criticality.

The shortest period which could result from a ramp rate of reactivity insertion of 0.1%Δk/k/s (100 pcm/s) was calculated, using Newton's Inequality Equation, to be greater than 29 milliseconds. Therefore, at the time the reactor power reaches 1.2 MW and the reactor is scrammed, the reactor period is calculated to be 29 milliseconds. During the <50 millisecond delay time from when the scram signal is received until the control rods start into the core, a negligible amount of reactivity is added to the core. The power would continue to rise on a period of 29 milliseconds during this time from 1.2 MW to 6.7 MW.

The time required for the addition of β_{eff} excess is approximately 7 seconds. In the following second, sufficient prompt excess reactivity is added to give the reactor a period of 29 milliseconds. The power rise during the sub-prompt time can be neglected, and the time for the power to increase from 1 milliwatt to 1.2 MW is 0.6 seconds using the period of 29 milliseconds. Using the time, the rods are being withdrawn after criticality is reached, to be 8.6 seconds, the excess reactivity of approximately 0.85%Δk/k (850 pcm) is added to the core up to the time the three control rods start dropping under the force of gravity. Using a reactivity rate of 800 pcm/inch, which is equal to the 0.1%Δk/k/s (100 pcm/s) insertion

rate and a rod speed of 7.5 inches/minute (0.3 cm$\cdot$s^{-1}), the three control rods would have to drop 1.06 inches (2.69 cm) to reduce the reactor to the point of criticality. The time required for the rods to drop 1.06 inches (2.69 cm) is less than 0.077 seconds. The peak power reached will be less than 120 MW and the total energy release less than 17 MW·s.

The results of this analysis indicates that continuous withdrawal of the three control rods in gang will result in a peak power less than 120 MW and a total energy release less than 17 MW·s. This analysis does not take into account the 9(10^4) cps inhibit on the Start-up Channel, the reverse (run-back) at 1.1 MW or any shutdown mechanism except scramming the reactor at 1.2 MW. If the inherent shutdown mechanism, the Doppler effect, were included, the magnitude of the peak power and energy release would be reduced.

This accidental power excursion in this analysis is significantly less than the nominal original design pulse for the reactor core, and the means of excursion is prevented by a combination of interlocks, reverse, operator controls, and reactor physics.

13.2.2.3 Pulse Rod Fails to Return to Starting Position

Pulsing is no longer permitted for the NCSU PULSTAR Reactor on approval of this amendment, and hence this scenario no longer applies.

13.2.2.4 Experiment Failure

A hypothetical excursion accident which can be postulated and estimated for the PULSTAR core is a step input of 1.0%Δk/k (1000 pcm) occurring if an experiment were to fail. It is based upon the restriction that no single experiment shall exceed a worth of 1.0%Δk/k (1000 pcm). This excursion accident scenario is of the same type analyzed in Section 13.2.2.1 for the fuel loading accident, but is less severe and falls within the step input analyzed therein.

13.2.2.5 Control Rod Failure

Failure of a single control rod to drop after a scram will not result in a hazard to reactor operations since positive shutdown of the core can be made by dropping any single rod. Sufficient cooling of the Ag-In-Cd control rods is provided to prevent melting.

In the event a control rod does not fall under the force of gravity to its down limit due to sticking following a Reactor SCRAM, the control rod drives which automatically drive down to the down limit after a scram will drive the stuck rod into the core.

13.2.2.6 Cold Primary Coolant Slug

During operation of the reactor, it is conceivable that a slug of cold pool water might pass through a critical core resulting in a positive reactivity input. For this analysis all modes of operation are considered.

In the natural convection cooling mode of operation, the steady state power is limited to 150 kW. Using the test results[13-1], the average temperature difference across the core at 150 kW is approximately 27°F (15°C). In this accident analysis, it is assumed that the core is operating in the natural convection mode at 150 kW when the flapper is closed, and the primary coolant pump is assumed to be operating. Closing the flapper valve will start forced convection cooling with cold pool water such that the 27°F (15°C) temperature rise across the core will no longer exist. Using a moderator temperature coefficient of $-7.02(10^{-3})$ $\%\Delta k/k/°C$ (-3.9 pcm/°F) and the 27°F (15°C) temperature change, a positive reactivity of $0.105\%\Delta k/k$ (105 pcm) would occur. This reactivity addition would result in a positive reactor period of 60 seconds. The reactor power would rise on the 60 second period until the Doppler coefficient and temperature coefficient reduced the $0.105\%\Delta k/k$ (105 pcm) excess reactivity to zero. Using the measured Power Coefficient of $-0.33\%\Delta k/k/MW$ (-330 pcm/MW) for the 5 by 5 Graphite Reflected Core No. 3, the reactor power would level at less than 320 kW.

For analysis of the effect of a cold water slug passing through a critical core operating in the forced convection mode, a temperature difference of the cold water slug and the bulk pool water temperature was assumed to be 27°F (15°C). It is doubtful that a variation of this size in the pool water temperature could exist under normal conditions. The result of the change in the core coolant temperature of 27°F (15°C) would result in a positive reactor period of 60 seconds. If this were to occur during 1 MW operation, the reactor would be scrammed by the high neutron flux level scrams at 1.2 MW.

The above analysis indicates that no hazard exists from a cold water slug accident.

13.2.2.7 Fuel Storage

Improper storage configuration of PULSTAR fuel could cause an excursion. To eliminate this potential, subcritical wet storage is provided in the PULSTAR pool in two storage racks, one with a capacity of 13 assemblies and the other with a capacity of 7 assemblies. The k_{eff} for the thirteen assembly rack is less than 0.3[13-14]. These two racks augment two 13-assembly capacity round fuel storage pits at the bottom of the pool liner. Therefore, one finds no critical array will exist, and all rack and storage assemblies would be significantly subcritical. Finally, during initial loading of the wet storage racks and pool storage pit, confirmation of the assembled k_{eff} was made.

13.2.3 Natural Hazards

As discussed in the Site Description Section 2, there are no significant natural phenomena which would contribute to a radiation hazard from the reactor. The surrounding area appears quite safe from the view point of meteorology, hydrology, geology, and seismology.

13.3 Conclusions

A review of those accidents which could possibly occur, both in the non-excursion and excursion categories, indicates that the maximum credible accident is of the non-excursion

category and consists of a loss of pool water due to a ruptured inlet or outlet pipe line. Even in this event, there is no core melting or loss of cladding integrity. The hazard associated is related only to the vertical radiation beam emanating from the unshielded shutdown core. Corrective measures can be taken to plug the leak and refill the tank without danger from the vertical radiation beam.

The maximum credible excursion accident is of even less severity than the postulated non-excursion accident. It consists a fuel loading accident whereby up to 1.59%$\Delta k/k$ (1590 pcm) step input is added to the reactor core. This value reactivity step input is found to produce a lower specific energy release density than that experienced by the PULSTAR fuel pins used in the initial Buffalo PULSTAR pulse test core.

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14.0 NCSU EXPERIMENTAL PROGRAM

14.1 Introduction

The program of operating a research reactor is aimed at the ultimate goal of providing maximum use for the experimentalist. As in the case of reactor operation, planning, design, and operation of experiments requires equal skill, attention and rigor. The emphasis in these areas is dictated by the type of experiments planned by the Faculty and University departments.

The NCSU PULSTAR Reactor shall continue to be used for the traditional university activities of teaching and research. In addition, it shall continue to provide specialized nuclear services to state and federal agencies and industry.

Administrative procedures and review criteria for reactor experiments will be summarized later in this section. In addition, the experimental facilities that are expected to be used on a routine basis are discussed.

The beam tubes can be used for time of flight and diffraction type experiments and radiative capture gamma spectroscopy. In addition, selected material irradiations can be conducted for nondestructive testing by neutron radiography. Finally, radiation damage effects on electronic materials, components and subsystems can be performed.

Pneumatic tubes are traditionally used for neutron activation analysis, isotope production, radiochemistry and synthesis, and nuclear physics studies.

There are four Rotating Exposure Ports (REP) which allow for placement of samples at peripheral locations to the reactor core. Samples are encapsulated and then loaded into irradiation containers. Rotation of the loaded irradiation container within the REP provides for uniform irradiation of encapsulated samples. Typical uses of the REP include neutron activation analysis, production of radioisotopes, and radiation damage studies.

Dry Exposure Ports (DEP) allow for placement of samples at peripheral locations to the reactor core or various pool locations. Samples are loaded into irradiation containers and placed in the DEP. Typically, DEP is used for measurement of reactor parameters using various types of detectors.

The thermal column provides a thermal neutron flux environment for diffusion studies and other work requiring a low-energy neutron source. Graphite bars (or stringers) are stacked inside the thermal column and are arranged to provide axial and tangential access to the interior of the column.

With regard to the reactor proper as an object of experimental investigation, undergraduate and graduate laboratory classes have convenient access for reactor operating parameter studies. Unique or exclusive-use type experiments by students and researchers may also be easily accommodated. To illustrate the areas of applications, work will continue in neutron

activation analysis, fission product behavior, radiation damage, reactor noise techniques, safety engineering, tracer techniques, reactor kinetics, health physics and dosimetry, and nuclear medicine research.

14.2 Experimentation Guidelines

At the PULSTAR Facility, several methods of performing experiments are employed. The experimenter is encouraged to operate his own experiment, with conditions imposed only for reasons of safety. Careful definition of safety limits and a close relationship between the reactor operation staff and the experimenter is necessary. Usually, operations personnel insert and remove experiments from the irradiation facilities. An experimenter, however, can be certified by the operations personnel to use a routine facility such as the pneumatic tube.

All experiments performed in or with use of the reactor must be evaluated and approved by the appropriate PULSTAR staff. The Request for Reactor Operations (Run Sheet) defines the format, requirements, and conditions for reactor use. Should there be a safety issue related to a reactor experiment or should a new, untried experiment be proposed, the University Radiation Protection Committee (RPC) must review and approve the requested experiment prior to actual performance.

For a proposed experiment, a member of the operations staff, or in the case of a student, his particular faculty sponsor, is assigned the responsibility of following and assisting with the planning of the experiment.

Administrative procedures and controls are developed which give sufficient definition of and checks on the performance of experiments. The PULSTAR staff review shall ensure that procedures are written by the experimenter that will provide detailed and specific controls of an experiment should that be deemed necessary. No changes shall be made in experimental procedures without prior review by the pertinent PULSTAR staff.

14.3 Limitations of Experiments

There are many limitations associated with experiments in the PULSTAR Facility. Irradiation of explosives, unstable mixtures, and dynamic devices which could compromise the reactor integrity are limited by the Technical Specifications. Refer to the approved Technical Specifications for limitations associated with reactivity, chemical and/or physical integrity, radiation hazards, fissionable material, etc.

14.4 Irradiation Facilities Operation Description

14.4.1 Core Experiments

In a typical irradiation one or more spaces are available adjacent to the reactor core to take advantage of the high thermal neutron flux region. These facilities may be either vertical, water-filled exposure tubes, or the pneumatic transfer system. The total worth of

experiments placed in the reactor core is limited to a maximum 3000 pcm (absolute worth). The worth of each experiment will not exceed 1000 pcm (absolute value). The thermal neutron flux in these exposure ports is approximately 1 x 10^{13} n/cm^2·sec.

14.4.2 Beam Tubes

The beam tubes are provided to extract neutrons from the core for use in diffraction studies, spectroscopy, radiography, and time-of-flight measurements. The basic tube assembly consists of an embedded aluminum sleeve, a concentric closed-end aluminum chamber, and a set of interior shielding plugs of canned borated barytes concrete and lead. The tubes can also be used as dry irradiation chambers for small samples in radiation effect studies. Samples can thus be placed at the core face and easily monitored in a dry environment. Thermal fluxes up to approximately 1 x 10^{12} n/cm^2·sec are available at the core end of these beam tubes. Refer to Figure 14-1 for beam tube details.

Loading and unloading beam tubes is performed in accordance with written procedures approved by the Radiation Protection Committee (RPC). Beam tube plug storage ports are provided in the West wall of the Reactor Bay.

14.4.3 Pneumatic Transfer System (PN)

A 2-inch (5.1 cm) pneumatic tube is provided for the rapid transfer of samples to and from the reactor. This tube is used for the production of isotopes and for neutron activation analysis in a wide variety of engineering and scientific areas. In the future, a high speed pneumatic system could be installed in one of the beam tubes and/or thermal column ports as required. The routing of the piping and blower for the Pneumatic System has been chosen such that syphoning of the pool water and subsequent uncovering of the core cannot occur. Refer to Figure 14-2 for additional details on the Pneumatic System.

14.4.4 Rotating Exposure Ports (REP)

Four 2.625 inch (6.67 cm) diameter REP are provided for irradiation of samples at peripheral locations to the reactor core. Each port is capable of irradiating one container holding encapsulated samples. The port, irradiation container, and encapsulated samples are submersed in the reactor pool. Refer to Figure 14-3 for details.

Standard irradiation containers are maintained by Reactor Operations but an experimenter may construct an irradiation container specific for an approved PULSTAR project if needed. Irradiation containers are identified by a unique number, handled using nylon string or fishing line, and secured in the reactor pool during use and storage. These containers are only briefly removed from the reactor pool during loading and unloading of samples. Construction of the irradiation containers includes weights at the bottom to ensure placement is maintained during use and storage, screw top cap or sealed cap, and holes in the body to allow for filling and draining of reactor pool water.

Samples loaded into the irradiation container are uniquely identified by the experimenter. Loading and unloading of samples is performed by qualified personnel who are knowledgeable in radiation protection and physical reactor controls. All sample materials, capsules, and associated items (markings, string, etc.) to be irradiated must meet approved PULSTAR project requirements.

14.4.5 Dry Exposure Port (DEP)

DEP is generally constructed of semi-rigid, curved aluminum or plastic tubing with a water tight end cap. Leak test and sample passage are conducted before the DEP is used. During use the water tight end of the DEP is secured in position in the reactor pool and the open end is secured above the pool surface. Radiation streaming from the reactor core is prevented by virtue of the DEP tube curvature. Detectors or encapsulated samples are lowered inside the DEP to the desired position for the experiment or measurement. Radiation surveys designated by the Reactor Health Physicist are performed as necessary while the DEP is in use. After use the DEP may be secured away from the reactor core and stored in the pool or may be removed from the pool and stored or disposed. All materials and associated items to be irradiated must meet approved PULSTAR project requirements. All irradiated materials and the DEP are surveyed upon removal from the pool.

14.4.6 Thermal Column

The graphite Thermal Column provides a relatively pure source of thermal neutrons. Figure 14-4 illustrates the location of the column with respect to the reactor core. The graphite column dimensions are 4 feet wide (1.2 m) x 4 feet high (1.2 m) x 5 feet long (1.5 m), composed entirely of 4 inch (10.2 cm) x 4 inch (10.2 cm) graphite bars (stringers). These stringers are of two lengths; 24 inches (61 cm) and 36 inches (91.4 cm). Access to the thermalized neutron flux may be through four (4) axial ports in the high density concrete-filled shielding door at the far end of the column or through a tangential port which makes a perpendicular intersection with the column three feet from the core end. One or more neutron detectors are frequently installed in the Thermal Column for monitoring thermal neutron levels.

14.5 Administrative Procedures for Use

14.5.1 Request for Project Approval

Section 14.2 earlier introduces the basic plan for requesting reactor use and describes the need for review of proposed experiments from the standpoint of safety. The request for reactor service is made through the PULSTAR Operations staff who maintains custody of all necessary forms and essential records. An experimenter must furnish in standardized format a description and purpose of the reactor use desired. Consideration must also be given to the irradiation facility required, necessary flux, total exposure, target activation, and core reactivity effects of the experiment. Questions of safety for untried experiments require

that thermodynamic unknowns be addressed, target chemical form, flammability and toxicity be discussed and special emergency procedures related to the foregoing be detailed.

On completion of the document and approval by the Associate Director or his designee, the request is reviewed and approved by the Radiation Protection Committee (RPC). On approval, a PULSTAR Project Number is assigned. Special conditions or constraints may be dictated by the RPC as deemed necessary.

On returning to operations, the experimenter proceeds to prepare for his experiment under the surveillance of his sponsor (i.e., a Faculty member for instance). In most cases, the use of standard instrumentation, techniques, equipment, and practices are employed to ensure success of the experiment.

14.5.2 Review Criteria

The leading criteria to be used in determining these potential hazards and effects associated with an experiment are as follows:

a. Chemical

The following materials if used within the reactor water system (in other than trace quantities) may result in hazardous conditions; therefore, they are avoided or treated with special precautions:

Mercury	Sodium
Copper and alloys	Potassium
Carbon or Organics	Hydrogen
Iron or Carbon Steel	Liquid Oxygen
Lithium	

In addition, chemical compounds decompose, silver solders may cause contamination, powders may escape encapsulation, and boron or cadmium containing materials affect reactivity.

b. Encapsulation

All experiments shall be contained unless it can be shown that the absence of such does not create a hazard. Any system which operates under positive pressure (excluding typical nuclear detectors such as an ion chamber), or may develop pressure due to an accident, and contains or is expected to contain amounts of releasable radioactivity or toxins which would jeopardize personnel safety, must have double encapsulation.

c. Heat Transfer

Systems transferring heat generated by gamma-heating, exothermic reactions, or fission within the experiment, must be designed for thermal stress generated by the

reactor operation at 120 percent of nominal power as well as stresses induced by fast startup rates or sudden shutdowns.

d. Mechanical Integrity

The choice of materials must be of the type which is structurally suitable within the test environment, and resulting radioactivity must be considered in view of handling facilities. Finally, the selection is always sensitive to corrosion problems, either of the material or effects induced in reactor components.

e. Radiation

No experiment, during normal operation, shall result in a direct radiation level in accessible work areas which cannot be reasonably controlled to insure compliance with 10 CFR 20. If a credible failure scenario would result in higher levels, special radiation monitoring shall be provided.

f. Instrumentation and Control

Sufficient instrumentation must be included to measure all parameters which may relate to a potential hazard, and automatically control the experiment if needed and practical. This would include such items as status lights for beam shutter positions in semi-permanent facilities.

g. Experiment Operation

All experiments will be operated in accordance with the approved procedures on the PULSTAR Project Form. PULSTAR operations personnel render assistance as required to scientists and engineers in planning and executing experiments on the PULSTAR Facility. Close contact between experimenter and operations personnel will continue to insure a workable and safe plan for each experiment.

h. Interference with Reactor

Experiments should be arranged so that they will cause little nuclear or physical interference with operation of the reactor. Installation and removal should normally be possible within a reasonable time, even in the case of experiment failure. It is necessary to make sure that reactivity effects of moving experiments do not exceed Technical Specification and/or administrative limits.

i. Manning of Experiments

In most cases the manning of experiments is not required. However, those experiments and operations which do require direct supervision or monitoring shall be specified in the approved PULSTAR Project.

FIGURE 14-1
BEAM TUBE

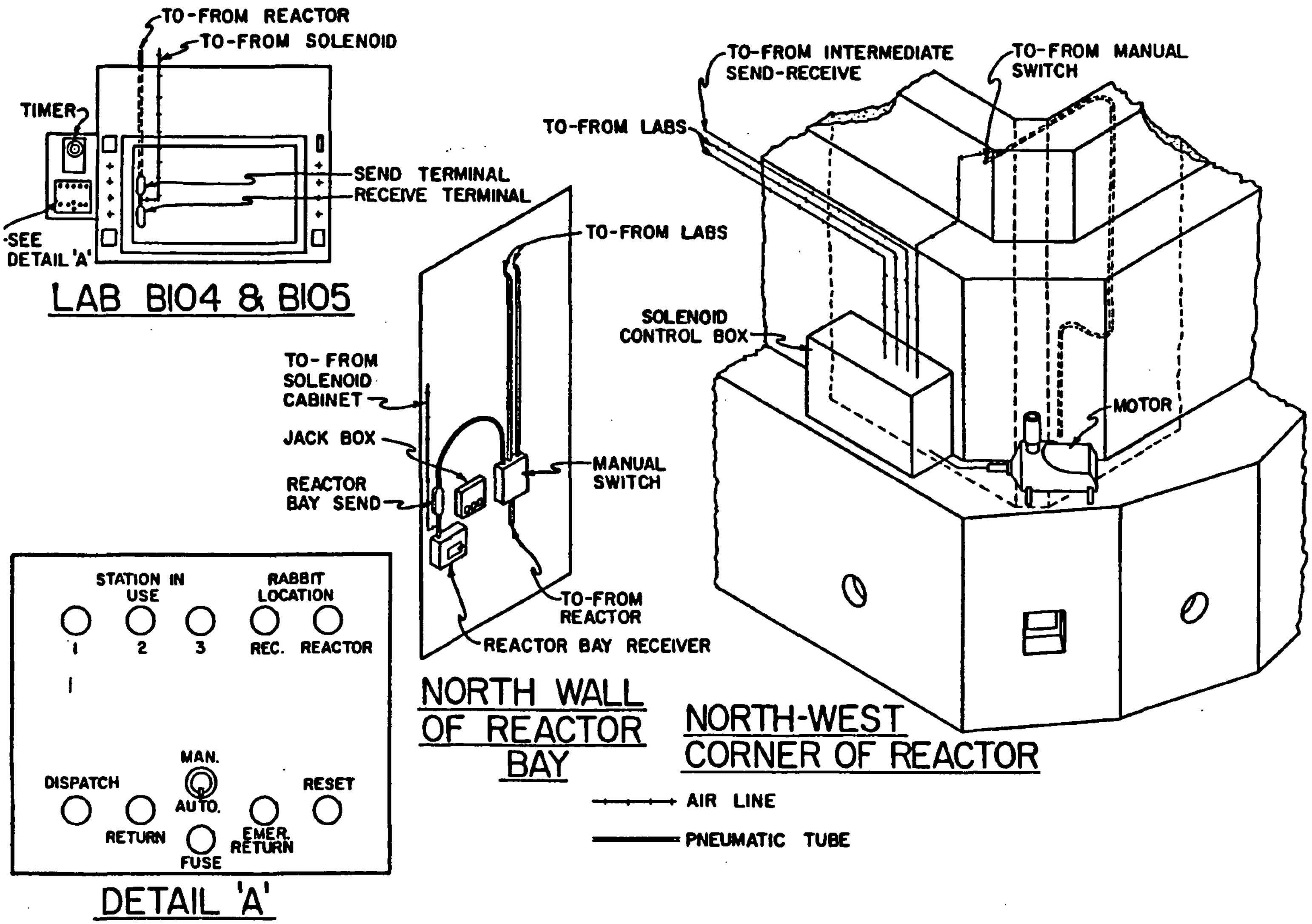


FIGURE 14-2
PNEUMATIC SYSTEM

FIGURE 14-3
ROTATING EXPOSURE PORTS

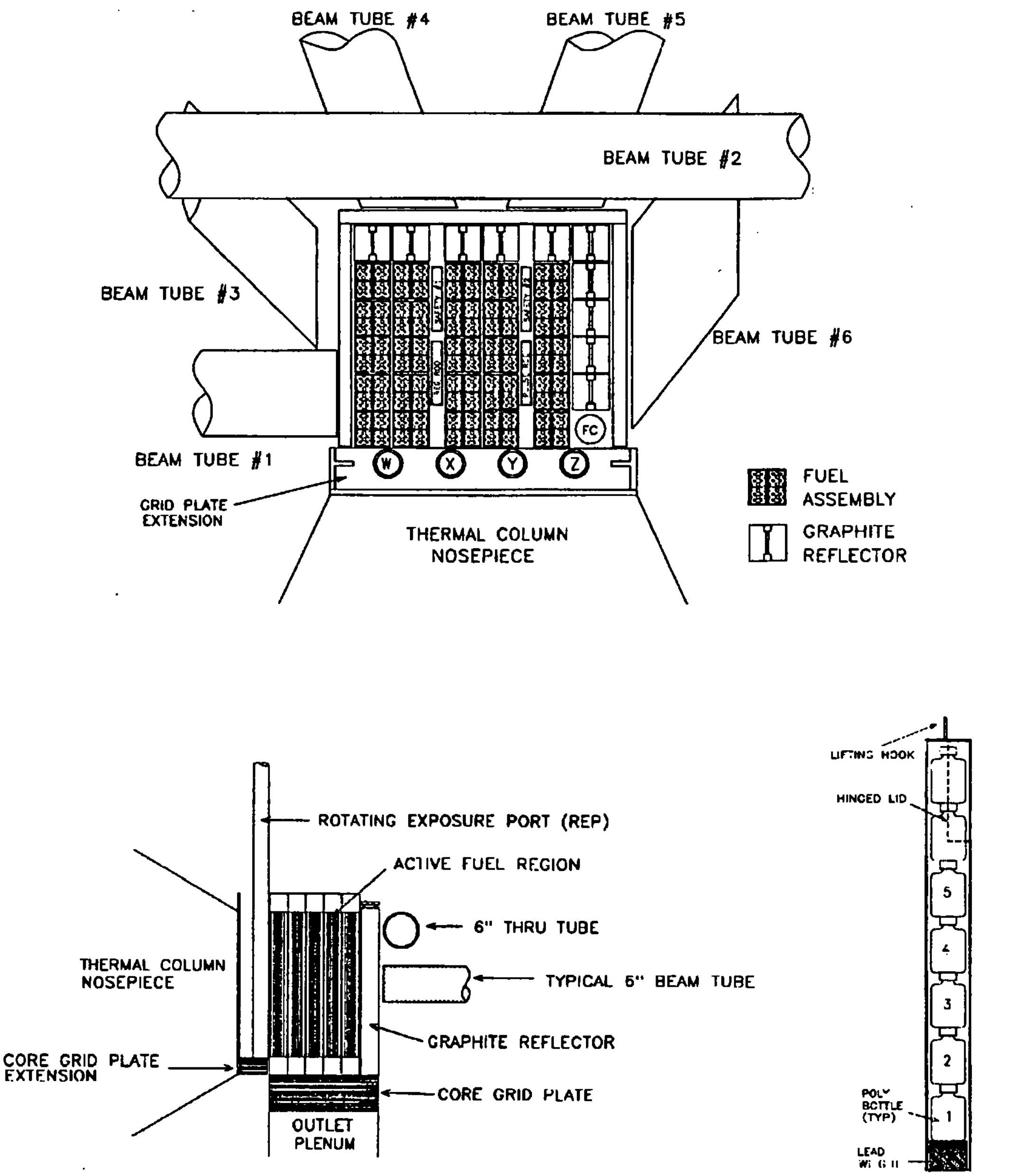
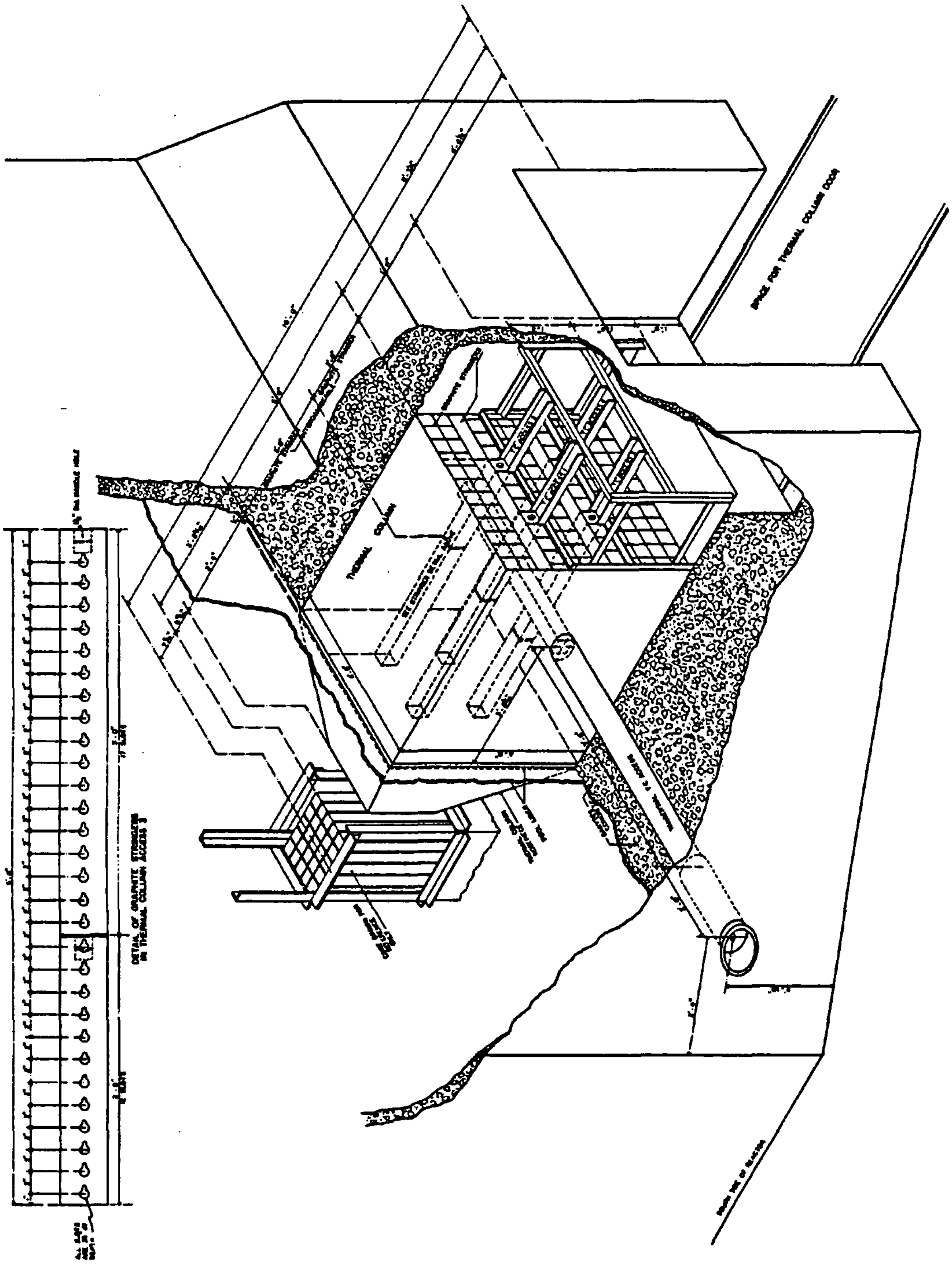


FIGURE 14-4
THERMAL COLUMN



APPENDIX 3-A
AVERAGE FUEL TEMPERATURE RISE CALCULATION

This appendix provides a calculation for the determination of the average and maximum fuel temperature rise associated with 1 MW operation. First, the following variables apply:

Q'_{avg} = **average** heat generated per ft of fuel rod = 2730.4 Btu/hr-ft°F

Q'_{hot} = **maximum** heat generated per ft of fuel rod = 2.92 Q'_{avg} = 7973 Btu/hr-ft°F

r_0 = outer radius of fuel pellet = 0.2115 inches = 0.017625 ft

r_1 = inner radius of clad = 0.2157 inches = 0.01798 ft

r_2 = outer radius of clad = 0.2362 inches = 0.01969 ft

k_{Fuel} = thermal conductivity of UO_2 fuel = 2 Btu/hr-ft°F

k_{He} = thermal conductivity of He fill gas = 0.1 Btu/hr-ft°F

k_{Zr} = thermal conductivity of zircaloy clad = 8.2 Btu/hr-ft°F

h_s = heat transfer coefficient of coolant (values follow in analysis)

The value for UO_2 thermal conductivity is a strong function of fuel temperature and ranges from 2 Btu/hr-ft°F at temperatures expected during a pulse to values over 3 Btu/hr-ft°F at the typical steady state operating temperatures. To provide a conservative result, the lower value shall be assumed. The Q'_{hot} value would appear in the hot channel, where the total peak to average of 2.92 is expected to occur.

The centerline fuel temperature is given by the following terms:

$$T_{centerline} = T_{coolant} + \Delta T_{film} + \Delta T_{clad} + \Delta T_{gap} + \Delta T_{fuel}$$

$$T_{centerline} = T_{coolant} + [Q'/\Pi]\{[1/2r_2h_s] + [(1/2k_{Zr})(\ln(r_2/r_1))] + [(1/2k_{He})\ln(r_1/r_0)] + [1/4k_{fuel}]\}$$

$$T_{centerline} = T_{coolant} + [Q'/2\Pi]\{[r_2h_s]^{-1} + [(k_{Zr})^{-1}(\ln(r_2/r_1))] + [(k_{He})^{-1}\ln(r_1/r_0)] + [2k_{fuel}]^{-1}\}$$

For example, at 1 MW with a nominal flow of 500 gpm (yielding a h_s=700 Btu/hr-ft^2-°F as detailed in Appendix 3B, page II-V), the <u>average</u> fuel centerline temperature is

calculated to be:

$$T_{centerline} = 105°F + [(2730.4)/(6.283)]\{[(0.01969)(700)]^{-1} +$$

$$[(8.2)^{-1}\ln(0.2362/0.2157)] + [(0.1)^{-1}\ln(0.2157/0.2115)] + [2(2)]^{-1}\}$$

$$T_{centerline} = 105°F + 434.5\{[0.07255] + [0.01108] + [0.1966] + [0.25]\}$$

$$T_{centerline} = 105°F + 31.5°F + 4.8°F + 85.4°F + 108.6°F$$

$$T_{centerline} = 335°F$$

The average temperature across the fuel pellet is assumed to be 1/2 of the temperature rise across the fuel pellet since the temperature distribution is parabolic. Therefore one can readily calculate this value for the prescribed conditions:

$$T_{fuel\ avg} = 105°F + 31.5°F + 4.8°F + 85.4°F + 0.5(108.6)°F$$

$$T_{fuel\ avg} = 281°F$$

For the maximum centerline fuel temperature at a nominal 1 MW and 100% flow, the value of Q'_{hot} is used:

$$T_{centerline} = 105°F + [(7973)/(6.283)]\{[(0.01969)(700)]^{-1} + [(8.2)^{-1}\ln(0.2362/0.2157)] + [(0.1)]^{-1}\ln(0.2157/0.2115)] + [2(2)]^{-1}\}$$

$$T_{centerline} = 105°F + 92.1°F + 14.0°F + 249.5°F + 317.2°F$$

$$T_{centerline} = 778°F$$

and calculating for the average fuel temperature at the hot spot location:

$$T_{fuel\ avg} = 105°F + 92.1°F + 14.0°F + 249.5°F + 0.5(317.2°F)$$

$$T_{fuel\ avg} = 619°F$$

APPENDIX 3B

THERMAL-HYDRAULIC ANALYSIS OF THE NORTH CAROLINA STATE UNIVERSITY PULSTAR CORE

HIT-486

March 1971

Prepared For
The American Machine & Foundry Company

HITTMAN ASSOCIATES, INC.
COLUMBIA, MARYLAND

LEGAL NOTICE

FOREWORD

This report was prepared under Purchase Order Number Y-86733 with the American Machine and Foundry Company, and is the Final Report of a Thermal-Hydraulic Analysis of the North Carolina State University PULSTAR Reactor.

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LIST OF FIGURES

I. INTRODUCTION

This report summarizes the steady state and transient thermal-hydraulic analyses performed for the North Carolina State University (NCSU) PULSTAR reactor. The effort described herein was performed by Hittman Associates for the American Machine and Foundry Company (AMF) in connection with licensing technical assistance. The purpose of this effort was to determine the steady state thermal capability of the NCSU PULSTAR core with respect to margin to DNB, bulk boiling, and flow instability, and to develop curves which would form the basis for establishing the Safety Limits and Limiting Safety System Settings for inclusion in the new form Technical Specifications. In addition, a transient analysis was performed to determine the effects of a loss of core flow incident with subsequent flow reversal on steady state thermal-hydraulic margins.

II. SUMMARY

A thermal-hydraulic analysis for the NCSU PULSTAR core was performed to determine the effects of variations in core coolant flow, system pressure, and core coolant inlet temperature on the steady state burnout power level, flow stability, and coolant bulk boiling in the channel of the core having the greatest heat input.

The core was analyzed under the most limiting conditions of coolant pressure and core inlet temperature. The results of the analyses are presented in Figures III-4 and III-7. These figures show that at low flows (i.e., below about 50 percent of full core flow), the steady state power level is limited by flow stability considerations while for higher flows, bulk boiling at the hot channel exit is limiting.

Figures III-4 and III-7 formed the basis for establishing Safety Limits on the NCSU PULSTAR core. The Safety Limit curve is given in Figure III-8. Since this reactor only has one pump and one design flow condition, the Safety Limit is in actuality a point function and is the limiting power level at full flow, namely 3.8 Mwt. Correcting the curve in Figure III-8 for measurement and instrument uncertainties results in establishing the upper limit for the Limiting Safety System Setting at 3.4 Mwt. This is referred to as an upper limit since the corrections do not include an allowance for safety system action.

A transient analysis was performed to determine the effects of a loss of flow incident with subsequent flow reversal on the required allowance for safety system action. It was shown that even for very conservative assumptions with respect to the severity of the accident and the response of the safety system, little thermal overshoot occurs. As a result, the overpower limitations for the NCSU PULSTAR core can be based on steady state rather than transient conditions.

III. STEADY STATE THERMAL-HYDRAULIC ANALYSIS

A. Model Description

1. Limiting Criteria for Core Safety

In establishing the technical specifications and normal operating conditions for the NCSU PULSTAR reactor, it is necessary to examine the conditions which might lead to a disruption of fuel integrity. Such conditions can be brought about as a result of burnout or due to an excessive flow instability and subsequent burnout. Since only downflow was considered in the analysis, burnout or a flow instability was assumed to occur if two-phase flow resulted at the channel exit. This condition is hereafter referred to as the no bulk boiling criterion at the channel exit.

Flow instabilities can also occur in a channel due to the effects of thermal buoyancy and increased pressure loss with subcooled nucleate boiling. It was assumed in the analysis that this flow instability would lead to burnout. The occurrence of such an instability was, therefore, considered to be a limiting criterion for operation and hereafter is referred to as the flow instability criterion.

Departure from nucleate boiling (DNB) and centerline fuel melting were the final two limiting criteria assumed in the analysis. The minimum allowable DNB ratio was established at a value of 2.0 and the fuel melting point is 5080°F.

2. Parameter Variation

The fuel cladding is considered to be the principle physical barrier in the NCSU PULSTAR facility which separates the radioactive materials produced and used in the fuel from the environs. It was with this barrier in mind that the limiting criteria for core safety were established.

The process variables that affect this cladding barrier include coolant pressure, as related to the depth of the pool water above the core, flow rate, power level, and coolant inlet temperature. Regulations for the determination of Safety Limits recognize that susceptibility to burnout will vary with the principal process parameters mentioned above. Therefore, the determination of the Safety Limit in the chosen variable was treated parametrically.

For the NCSU PULSTAR, the choice of core power level as the variable upon which to place the safety limit was the most logical selection. For the purpose of generality, the core flow rate was assumed to be variable from full flow down to a no flow condition. This, in reality, is not the case since system operation does not provide for variable controlled flow.

The safety limit is determined from analyses which consider all other process variables to be at the worst upperbound of their operating range. For coolant pressure, this upperbound occurs when the pool depth is at its minimum allowable value as specified in the technical specifications. The limit on coolant inlet temperature is at its maximum value which corresponds to the maximum allowable pool temperature as specified in the technical specifications.

3. Thermal-Hydraulic Model

Due to the effects of increased pressure loss with subcooled nucleate boiling and adverse buoyancy from downflow, an excessive hydraulic instability can occur in pool reactor hot channels.

The mechanics of a flow instability can be better understood if one considers the following equation:

$$P_2 - P_1 = \Delta P_{elev} - \Delta P_{flow} \qquad \text{(III-1)}$$

where:

P_1, P_2 = static pressure above and below the core, respectively

ΔP_{elev} = difference in elevation head between the top and bottom of the core

ΔP_{flow} = coolant flow pressure drop due to acceleration, entrance and exit effects, and channel friction across the core

If one were to plot Equation III-1 for the core average coolant channel as a function of coolant flow rate, a curve similar to that shown in Figure III-1 would result.

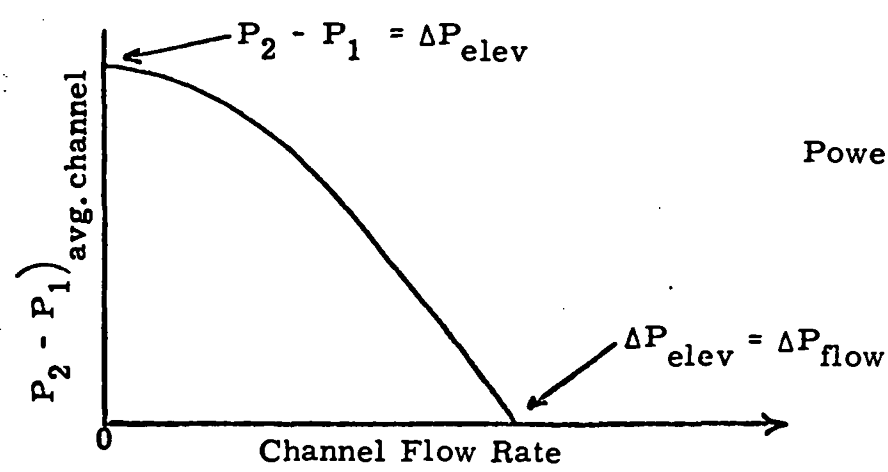


Figure III-1. Average Channel Pressure Drop - Flow Curve

Figure III-1 indicates that for a no-flow condition, the static pressure difference is due entirely to the elevation head. At some high flow condition, the flow associated pressure losses will have increased enough to balance the elevation head and at this point the static pressure difference across the average channel will be zero. Further flow increases will result in a situation such that $P_1 > P_2$.

One can construct a similar pressure drop-flow curve for the hot channel; the difference in elevation head will be less than for the average channel for any given flow rate due to the higher average coolant temperature (hence, a lower average density) in the hot channel. In the NCSU PULSTAR core, this is very important since the flow pressure drop terms are a very small part of the total due to the very low nominal flow velocity. Considering the above, the hot channel would have a pressure drop-flow curve similar to that shown in Figure III-2.

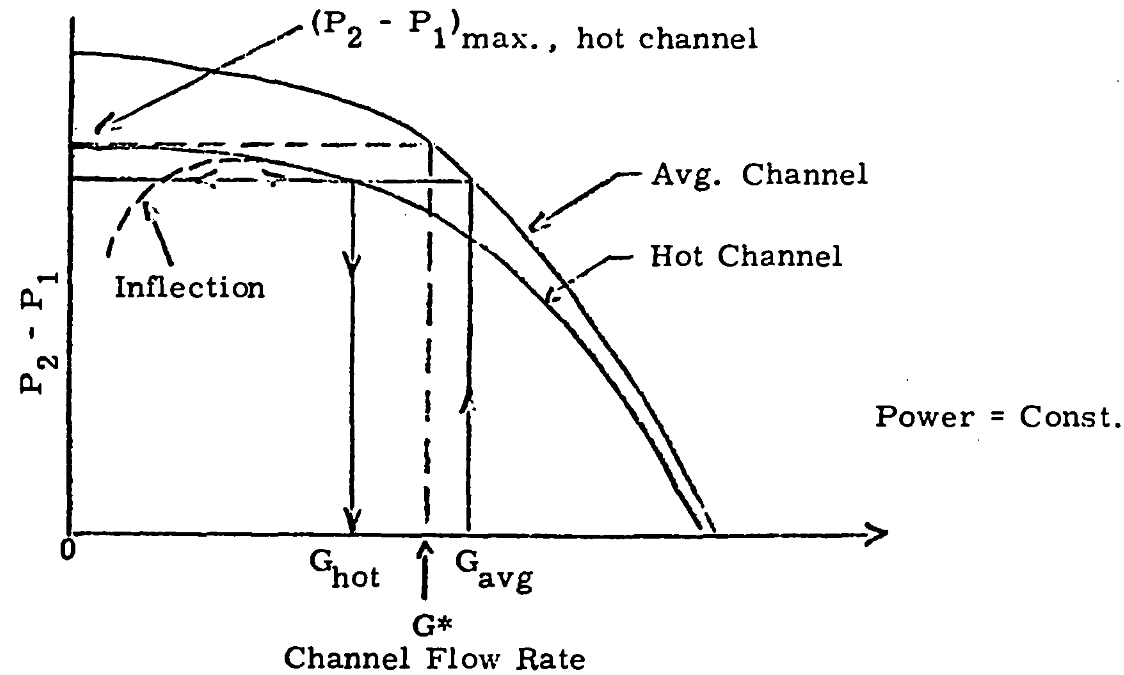


Figure III-2. Hot Channel Pressure Drop - Flow Curve

If we neglect the effects due to flow mixing, then the total pressure drop across all channels must be the same. Therefore, using Figure III-2, for any core flow rate, one can determine $(P_2 - P_1)_{avg}$ and then determine the flow in the hot channel (as shown by the arrows). However, if$(P_2 - P_1)_{avg}$ exceeds the maximum value shown for the hot channel (which could also result from an inflection in the hot channel curve as shown) then a flow instability will occur since no steady state solution can be obtained. Mathematically, the flow in the hot channel will try to reverse in order to change the sign of the flow pressure loss terms so that $P_2 - P_1$ can increase. The average channel flow at which this occurs (e.g., G* in Figure III-2), in combination

with the power level, represents a limiting point of operation with respect to flow instability. A locus of these points can be constructed by examining flow instability under various combinations of flow and power level.

The channel pressure losses were calculated from the following:

$$\Delta P_{total} = \Delta P_{acceleration(1)} + \Delta P_{acceleration(2)} + \Delta P_{elevation} + \Delta P_{friction} + \Delta P_{entrance\ and\ exit} \qquad \text{(III-2)}$$

where:

$$\Delta P_{acc(1)} = \frac{G^2}{g_c}\left(\frac{1}{\rho_{out}} - \frac{1}{\rho_{in}}\right) \qquad \text{(Spatial)} \qquad \text{(III-3)}$$

$$\Delta P_{acc(2)} = \frac{G^2}{2g_c}\left[\left(\frac{\sigma_{out}^2 - 1}{\rho_{out}}\right) - \left(\frac{\sigma_{in}^2 - 1}{\rho_{in}}\right)\right] \qquad \text{(Configuration Change)} \qquad \text{(III-4)}$$

$$\Delta P_{elev} = \bar{\rho} L \qquad \text{(III-5)}$$

$$\Delta P_{fric} = \frac{G^2 L}{2g_c D_e \bar{\rho}}\left(\frac{f}{f_{iso}}\right) f_{iso} \qquad \text{(No Local Boiling)} \qquad \text{(III-6)}$$

$$\Delta P_{e\ \&\ e} = \frac{G^2}{2g_c}\left(\frac{K_c}{\rho_{in}} + \frac{K_e}{\rho_{out}}\right) \qquad \text{(III-7)}$$

and where:

G = the mass velocity in the channel, lb/hr-ft^2

g_c = the acceleration of gravity, 32.2 ft/sec^2

ρ_{in}, ρ_{out} = the density of the fluid in or out of the channel, lb/ft^3

$\bar{\rho}$ = the average density of the fluid in the channel

L = the length of the channel, ft

K_c, K_e = the unrecoverable entrance or exit coefficient

$\sigma_{in}, \sigma_{out}$ = area ratio at entrance or exit (ratio of smaller to larger area

D_e = the channel hydraulic diameter

f_{iso} = isothermal friction factor

f/f_{iso} = the friction factor ratio for heating

Experiments have shown that for isothermal flow, rectangular channel friction factors are close to the Moody curve for smooth tubes (Ref. 1). The isothermal friction factors on the Moody curve are well approximated by the following in the Reynolds number range of interest in the NCSU PULSTAR (Ref. 2):

$$f_{iso} \cong \frac{0.316}{Re^{1/4}} \qquad \text{(III-8)}$$

f_{iso} is always evaluated at the bulk fluid temperature. However, to account for factor reduction due to decreased viscosity near the wall due to heating, we have (Ref. 3):

$$\frac{f}{f_{iso}} = \left(\frac{\mu_{wall}}{\mu_{bulk}}\right)^{0.14} \qquad \text{(III-9)}$$

With the occurrence of subcooled nucleate boiling in channels, friction pressure losses increase. A correlation proposed by Reynolds (Ref. 4) was used to calculate subcooled nucleate boiling pressure loss in the hydraulic analysis. This correlation is given by Tong (Ref. 5) as:

$$\Delta P_{fric}\Big)_{\substack{local\\boiling}} = \left(\frac{dp}{dl}\right)_o \frac{G\,D_e C_p\,(T_{sat} - T_{LB})}{4aq''} \sinh\left[\frac{4aq''\ell}{GD_eC_p(T_{sat} - T_{LB})}\right] \qquad \text{(III-10)}$$

where:

a = $(4.6 \times 10^{-6})\, q'' + 1.2$

ℓ = length of local boiling zone, ft

$\left(\frac{dp}{dl}\right)_o$ = isothermal non-local-boiling pressure gradient at location of beginning of local boiling, psi/ft

T_{LB} = coolant bulk temperature at location of beginning of local boiling, ^oF

T_{sat} = coolant saturation temperature, ^oF

C_p = coolant specific heat, Btu/lb-^oF

Therefore, when nucleate boiling occurs in the channel, the relation given in Equation III-10 replaces that given in Equation III-6 over the length of the channel where nucleate boiling is occurring. Figure III-3 shows the clad surface and bulk coolant temperatures in a channel where nucleate boiling occurs.

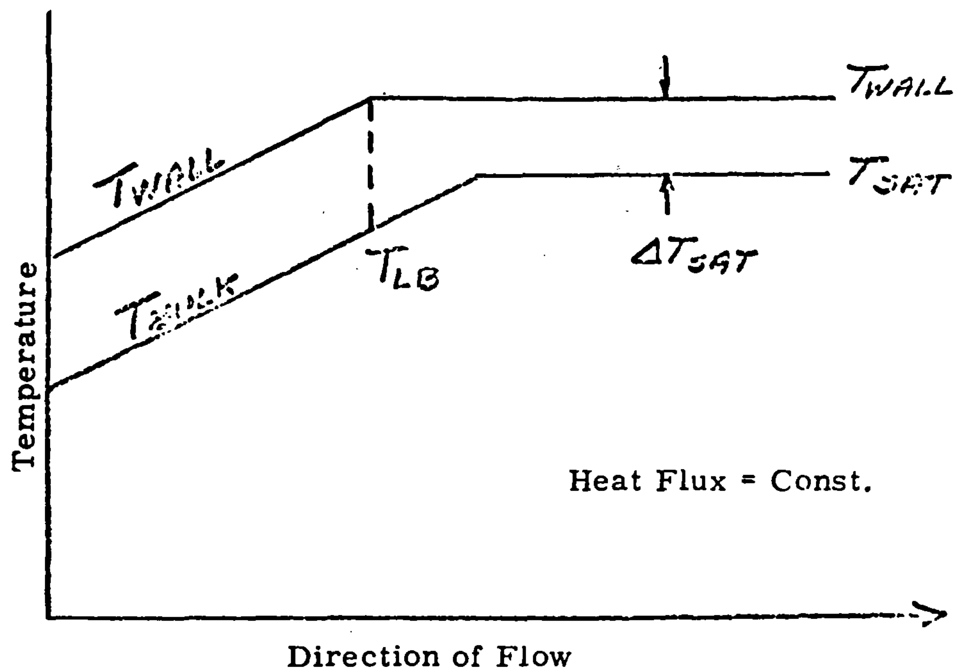


Figure III-3. Clad and Coolant Temperature Profile Along Coolant Channel

If T_{LB} can be determined, then the location of the onset of nucleate boiling can be found from relations which give the bulk coolant temperature rise with axial position. The onset of local boiling can be predicted as follows:

$$T_{LB} \cong T_{sat} + \Delta T_{sat} - \left(\frac{\bar{q}''}{\bar{h}}\right) \qquad \text{(III-11)}$$

where:

$\bar{q}''$ = average heat flux over the length of the channel, Btu/hr-ft^2

$\bar{h}$ = average film heat transfer coefficient over the length of the channel up to the point of nucleate boiling, Btu/hr-ft^2-^{o}F

ΔT_{sat} = $T_w - T_{sat}$, at the point of initiation of nucleate boiling, ^{o}F

Bergles and Rohsenow (Ref. 6) have developed a criterion for the beginning of subcooled nucleate boiling by solving graphically the bubble growth equation to give:

$$q''_{LB} = 15.60\, p^{1.156} (T_w - T_{sat})^{2.3/p^{0.0234}} \qquad \text{(III-12)}$$

where:

q''_{LB} = local boiling heat flux, Btu/hr-ft^2

p = coolant pressure, psia

Using Equation III-12, one can then obtain ΔT_{sat} at the point of initiation of local boiling based on $\bar{q}''$ and then from Equation III-11 obtain T_{LB}.

The film heat transfer coefficient, h, can be calculated using the Hausen equation (Ref. 7) which accounts for such factors as small channel length to diameter ratios of the order of 60 to 100 (PULSTAR has an L/D of about 60) and flows in the transition Reynolds number range. However, a more conservative approach was used in the analysis. It was assumed that the film coefficient could be calculated using the relationship which follows (Ref. 5):

$$h = 0.023\left(\frac{k}{D_e}\right)Re^{0.8}\,Pr^{0.4} \qquad \text{(III-13)}$$

The use of the above results in a lower value of h than would be given by the Hausen equation. A lower film coefficient will result in nucleate boiling being initiated earlier and hence an increase in the pressure loss due to local boiling. This, in turn, has an adverse effect upon the flow stability in the channel.

The bulk coolant temperature rise in the channel is given by the following:

$$T(z) - T_{in} = \frac{4\bar{q}''z}{GD_eC_p} \qquad \text{(III-14)}$$

where:

z = axial position along the channel, ft

T(z) = bulk coolant temperature at any position z, ^{o}F

The above relation was used to determine at what power level,for any given flow,bulk boiling occurred at the outlet of the hot channel. The heat flux was modified by the hot channel factor for enthalpy rise in the hot channel. The hot channel factors used in the analysis are discussed in Section III. B. 2.

4. Burnout Heat Flux Correlations

Numerous correlations have been proposed for predicting subcooled and quality burnout. Unfortunately, just about all of these correlations are not applicable for the parameter range of the NCSU PULSTAR reactor.

Selection of burnout correlations should be made on the basis of burnout data that fall within the pertinent parametric range. A literature search failed to uncover any data exactly covering the geometry, pressures, and velocities characteristic of the NCSU reactor. It was concluded, therefore, that several correlations would be evaluated and compared in order to obtain a reasonable estimate of the burnout heat flux.

Two burnout correlations which were examined are based on forced convection conditions in a channel. The first correlation is the Lowdermilk correlation (Ref. 8) given as follows:

$$\text{For} \quad \frac{G}{(L/D_e)^2} < 150 \qquad q''_{crit} = \frac{270\ G^{0.85}}{D_e^{0.2}\left(\frac{L}{D_e}\right)^{0.85}} \tag{III-15}$$

The test conditions for which this correlation was developed are given in Table III-1.

TABLE III-1. TEST CONDITIONS FOR LOWDERMILK CORRELATION

Test section	Round tube
Pressure	14.7-100 psia
$\Delta T_{subcooling}$, inlet	0-140°F
Tube diameter	0.00425-0.0157 ft
L/D	25-250
q" critical	0.9×10^6-13.2×10^6 Btu/hr-ft^2

The conditions given in Table III-1 are satisfactory with respect to the NCSU reactor except for the range on critical heat flux. Calculations using the Lowdermilk correlation resulted in critical heat fluxes lower than the range of correlated data. In addition, the low flow conditions present in the NCSU core are also somewhat outside the range of data since the Lowdermilk correlation indicates that the critical heat flux approaches zero as the flow approaches zero. This is not the case since for very low flows, the critical heat flux will approach a value given by pool boiling correlations.

The second forced convection type burnout correlation examined was Bernath's correlation (Ref. 9). This correlation recognizes that the two-phase coolant near the heated surface is highly turbulent and well mixed at the boiling crisis. The mixture may thus be considered homogeneous, and the heat transfer mechanism may be viewed as convection through this two-phase mixture layer as given in the equation:

$$q''_{crit} = h_{crit}(T_{w,crit} - T_{bulk}) \tag{III-16}$$

Equations developed by Bernath for h_{crit} and $T_{w,crit}$ are as follows:

$$T_{w,crit} = 1.8\left[57 \ln p - 54\left(\frac{p}{p+15}\right) - \frac{V}{4}\right] + 32 \qquad \text{(III-17)}$$

and

$$h_{crit} = 10,890\left(\frac{D_e}{D_e + D_i}\right) + \frac{48V}{D_e^{0.6}} \quad \text{for } D_e \leq 0.1 \text{ ft} \qquad \text{(III-18)}$$

where:

p	=	system pressure, psia
V	=	coolant velocity, ft/sec
h_{crit}	=	critical film coefficient, Btu/hr-ft^2-^{o}F
$T_{w,crit}$	=	critical wall temperature, ^{o}F
D_e, D_i	=	hydraulic diameter and (heated perimeter)/π, respectively, ft

The ranges of correlated experimental subcooled water data are given in Table III-2.

TABLE III-2. TEST CONDITIONS FOR BERNATH CORRELATION

Test Section	Round tube, rectangular channel and annulus
Pressure	23-3000 psia
Velocity	4.0-54 ft/sec
D_e	0.143-0.66 in.

From the above, one can see that the coolant velocity data are far outside the conditions existing in the NCSU reactor since the flow velocity at nominal core flow is about 1.8 ft/sec in the coolant channel.

In addition to the two forced convection correlations described above, a third correlation based on pool boiling conditions was used. This correlation was chosen so as to provide some estimate of the lower bound on the critical heat flux at very low flows since both forced convection correlations lead to a zero heat flux at zero flow which is not true.

Rohsenow and Griffith (Ref. 10) obtained a correlation of the pool boiling critical flux given as follows:

$$\frac{q''_{crit}}{H_{fg}\rho_v} = 143\left(\frac{\rho_\ell - \rho_v}{\rho_v}\right)^{0.6} \quad \text{ft/hr} \qquad \text{(III-19)}$$

where:

ρ_ℓ, ρ_v = saturated liquid and vapor densities, lb/ft^3

H_{fg} = latent heat of vaporization, Btu/lb

This correlation was used in the analysis and was evaluated at the saturation pressure corresponding to the minimum value of pool depth above the core.

5. Fuel Temperature Analysis

The fuel and clad temperature profiles were calculated using the standard heat transfer relationships for conduction and convection in cylindrical geometry. Heat transfer was assumed to occur in the radial direction only and the thermal conductivities of the fuel and clad were assumed not to vary with position. The gap conductance was based on a helium fill gas atmosphere and the worst manufacturing gap at beginning of life. This corresponds to the lowest value of gap conductance expected over the life of the plant since after operation, the gas composition becomes diluted from helium to a mixture of helium, krypton, and xenon and the gap width decreases due to fuel swelling.

The hot channel factor for film temperature rise was used in the calculations of peak fuel centerline temperature at the hot spot.

B. PULSTAR Thermal-Hydraulic Characteristics

1. Core Description

The NCSU PULSTAR core is described in detail in Section 1.3 of the SAR. The data presented in the referenced section were used in the steady state and transient analyses described in this report.

The steady state analysis assumed that the core flow could be varied down to zero flow. This was done for the sake of generality only since the core design calls for only two conditions of flow, namely, natural convection and forced flow where the forced flow results from single pump operation at 500 gallons per minute.

The axial heat flux distributions in the hot and average channels were assumed to be flat for the pressure drop and flow stability calculations. Enthalpy hot channel factors were applied, however, and total peaking factors were used in the burnout calculations. Neglecting the axial variation in heat flux for the flow instability calculations could be justified since the results indicated that the flow pressure drop terms were small for the flow velocities

found in this core. The prime factor affecting the flow stability was the adverse buoyancy which, for the entire channel, is relatively unaffected by the heat flux distribution.

2. Hot Channel Factors

Three hot channel factors were used in the steady state and transient analyses of the NCSU PULSTAR core. The factors attempted to account for uncertainties in channel bulk coolant temperature rise, film temperature drop, and local heat flux (F_b, F_{film}, $F_{q''}$, respectively). Flow maldistribution in the plenum was considered as a component of the affected hot channel factors through a five percent flow reduction factor.

Table III-3 lists the various components assumed to be incorporated into the three factors. It should be noted that there are differences between these factors and those presented in Section 3.2.4 of the SAR. These differences are due to the fact that certain components of the factors in the SAR are associated with instrument uncertainties which, in the present analyses, are treated separately.

TABLE III-3. OVERALL BULK, FILM, AND HEAT FLUX FACTORS

	F_b	F_{film}	$F_{q''}$
Fuel loading	1.02	1.02	1.02
Pin spacing	1.021	1.005	1.017
Heat transfer coefficient	---	1.2	---
Flow reduction	1.05	1.05	1.05
Radial peaking	1.94	1.94	1.94
Axial peaking	---	1.51	1.51
	2.12	3.77	3.18

C. Results of Steady State Analysis

1. Results of Burnout and Fuel Temperature Analyses

The results of the burnout calculations are presented in Figure III-4. The conditions assumed for these calculations are based on the process variables of pressure and coolant inlet temperature being at their worst values. These values correspond to a minimum of 17 feet of pool water above the core (actually 14 feet to allow for scram setting) and an inlet temperature of 120°F.

Figure III-4 also includes the results of the calculations giving the power level which results in bulk boiling at the outlet of the hot channel.

It can be seen from Figure III-4 that the bulk boiling limitation at the outlet of the hot channel is more restrictive than the MDNBR = 2.0 limitation. This is true even if the forced convection burnout correlations are extended down to a zero flow condition. The pool boiling critical flux calculations resulted in a maximum allowable power level of 3.5 Mwt. This should be a lower bound on reaching burnout under the flow conditions postulated. Note that a minimum allowable DNB ratio of 2.0 is reasonably conservative and provides another margin of safety to avoid coming close to the critical heat flux.

Centerline fuel melting was calculated to occur at the hot spot at a power level of 5.25 Mwt at full flow, as shown on Figure III-4. The fuel centerline calculation results also show that at zero flow (which actually corresponds to some natural convection flow), the reduced film coefficient lowers the power level required to reach possible fuel melting. A conservatively low value of convection film coefficient was chosen for this part of the analysis (see Section IV.B.1 of this report). The results indicate that a large margin exists between burnout or bulk boiling limits and the limits on power imposed by the centerline fuel melting criterion.

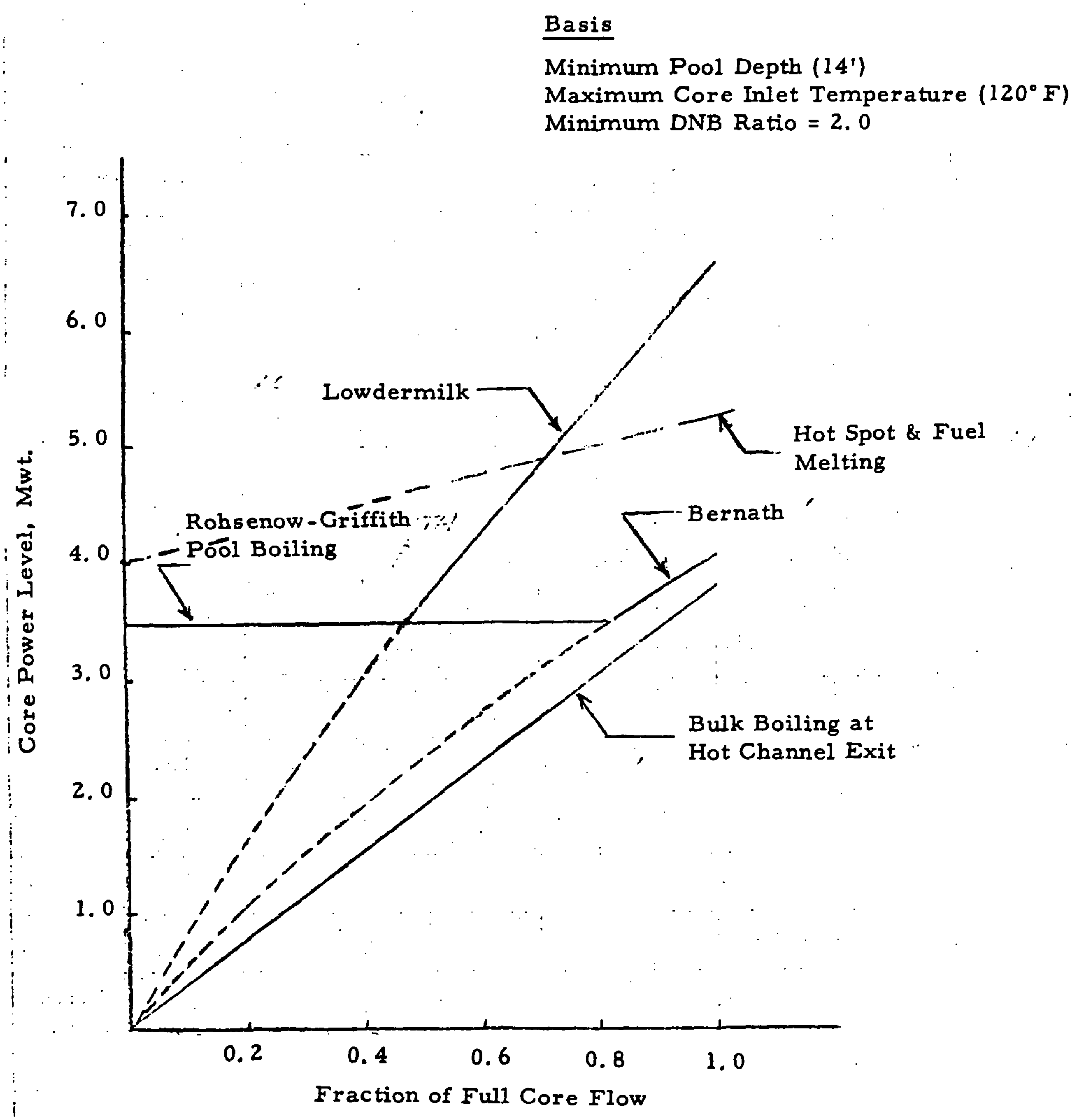
2. Results of Flow Instability Analysis

The initial calculations in the flow stability analysis were related to determining the pressure drop-flow curves for the average and hot channels. The process variables of coolant pressure and inlet temperature were assumed to be the same values as for the burnout analysis since this adversely affected flow stability.

The acceleration pressure drop (both spatial and configuration change) was calculated and found to be negligible compared to the other pressure drop components. This is due to the fact that the nominal core flow velocity is very low in this reactor. The entrance and exit losses, however, were significant. This pressure loss varied significantly with flow rate, as expected, but had negligible variation with power level since small density variations did not have a strong influence in the basic relationships.

FIGURE III-4

POWER LEVELS REQUIRED TO REACH DNB
OR BULK BOILING IN THE HOT CHANNEL



The pressure differences due to elevation head were very significant and represented the large fraction of the total pressure difference across the core. Since the elevation head is directly related to mean density, strong variation of the pressure difference was obtained with both flow rate and power level or heat input.

The friction pressure losses for no nucleate boiling conditions were small, again due to the low flow velocities. The magnitude of the losses was about half of the entrance and exit losses and showed a similar trend of not being affected by power level to any extent. When nucleate boiling occurred in the hot channel, the friction losses increased over the no nucleate boiling loss, but not dramatically. The increases amounted to about 20 percent at most.

The pressure loss-flow curves for the average and hot channels are shown in Figures III-5 and III-6, respectively. It can be seen that for the hot channel, an inflection occurs in the Δ P-flow curves indicating flow instability. This inflection is due primarily to the coolant heating effects resulting in a significant decrease in the elevation head at low flows. The average channel curves give some indication of flow instability at low flows also. However, the hot channel is more severe and would occur first.

Using Figures III-5 and III-6, one can construct the locus of points on the power-flow curve where flow instability will occur. The results are shown in Figure III-7 which also includes the bulk boiling curve for reference. It can be seen that for less than about 50 percent of full core flow, flow instability in the hot channel limits the allowable power level more than bulk boiling at the hot channel outlet. It is also clear that for flows above 50 percent of nominal, the likelihood of flow instability is drastically reduced and is of no concern.

D. Safety Limitations and Safety Margins

The Safety Limit is a value of the chosen variable (e.g., power level) at which one can say with confidence that no serious consequences will occur. The data presented previously in Figures III-4 and III-7 can be combined to establish a Safety Limit curve as a function of core flow rate. This curve is shown in Figure III-8.

Since the NCSU PULSTAR operates at only one forced convection flow condition, the Safety Limit can be established as a single point value. From Figure III-8, this value is 3.8 MWt at 100 percent flow, or 500 gpm. See Appendix 3C for analysis in support of Safety Limit as a continuous function of power and flow.

At a level on the safe side of the safety limit, a limiting safety system setting is selected. The region between this setting and the safety limit should be sufficient to allow for corrective action by the safety system to return the situation to normal or to shut the reactor down before the safety limit would be reached. This means that circuit response times and transient characteristics must be taken into account. These effects are considered in Section IV.B of this report.

FIGURE III-5

AVERAGE CHANNEL PRESSURE DIFFERENCE - FLOW CURVE AS A FUNCTION OF POWER LEVEL

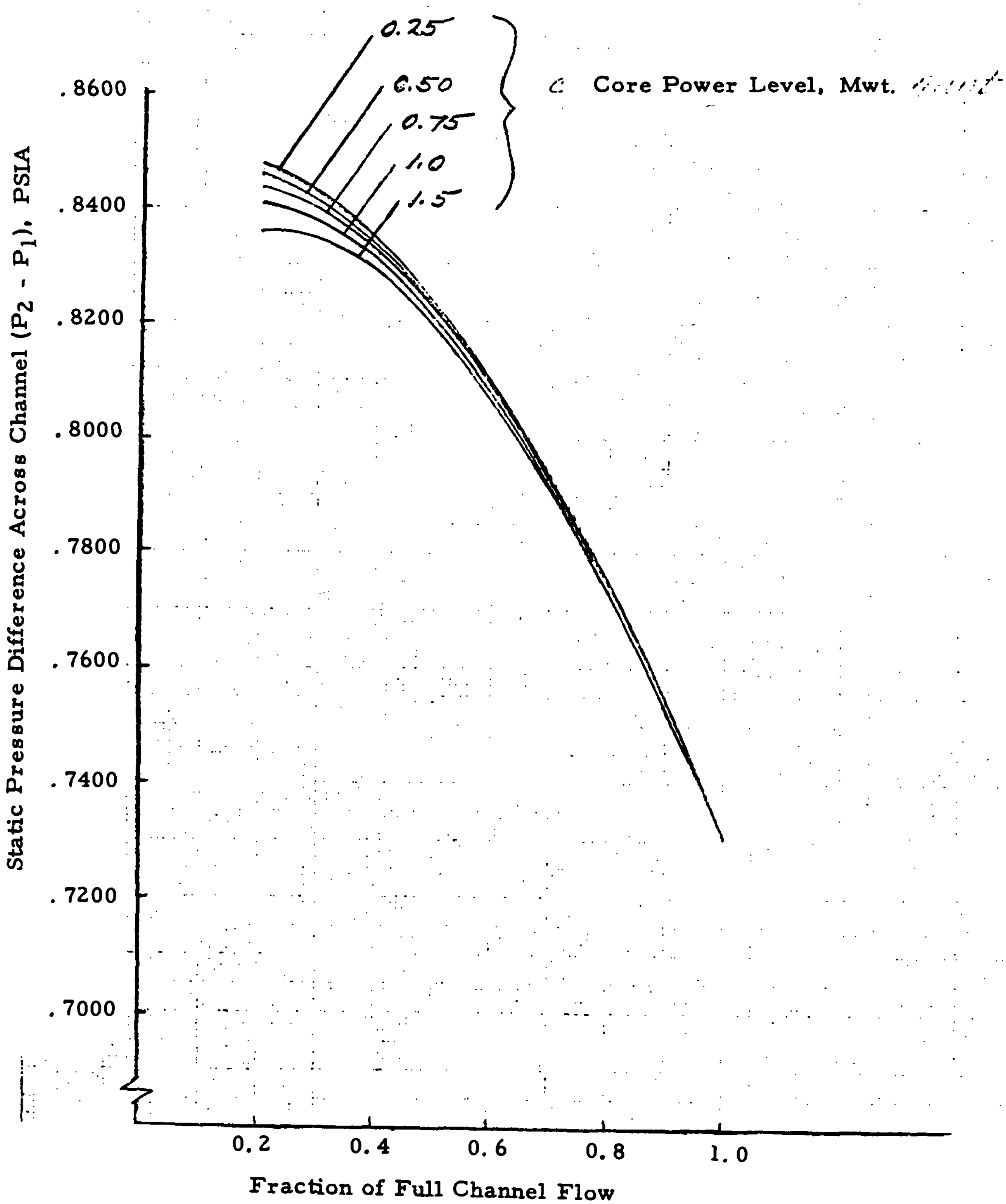


FIGURE III-6

HOT CHANNEL PRESSURE DIFFERENCE - FLOW CURVE AS A FUNCTION OF POWER LEVEL

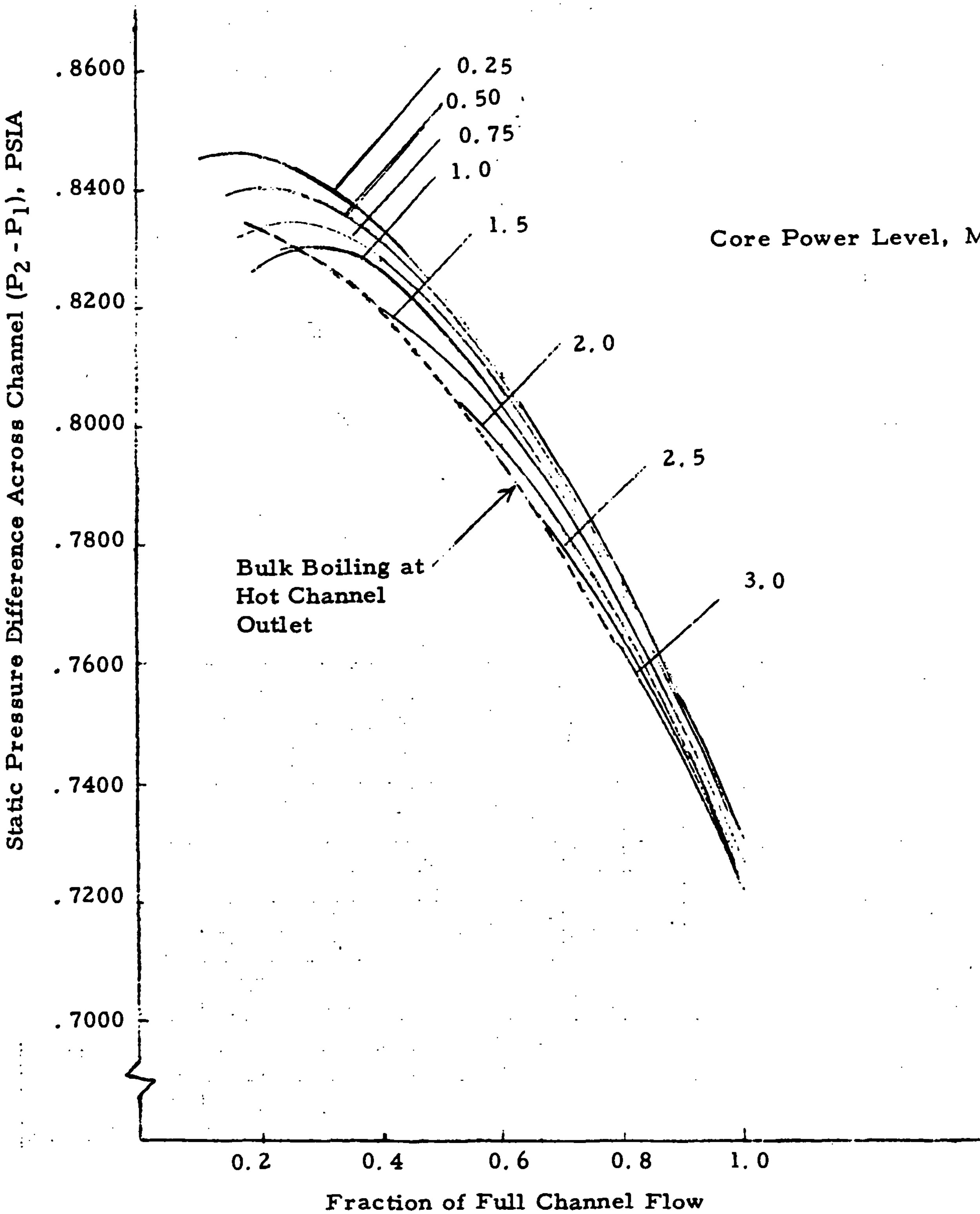


FIGURE III-7

POWER LEVELS REQUIRED TO REACH BULK BOILING OR FLOW INSTABILITY IN THE HOT CHANNEL

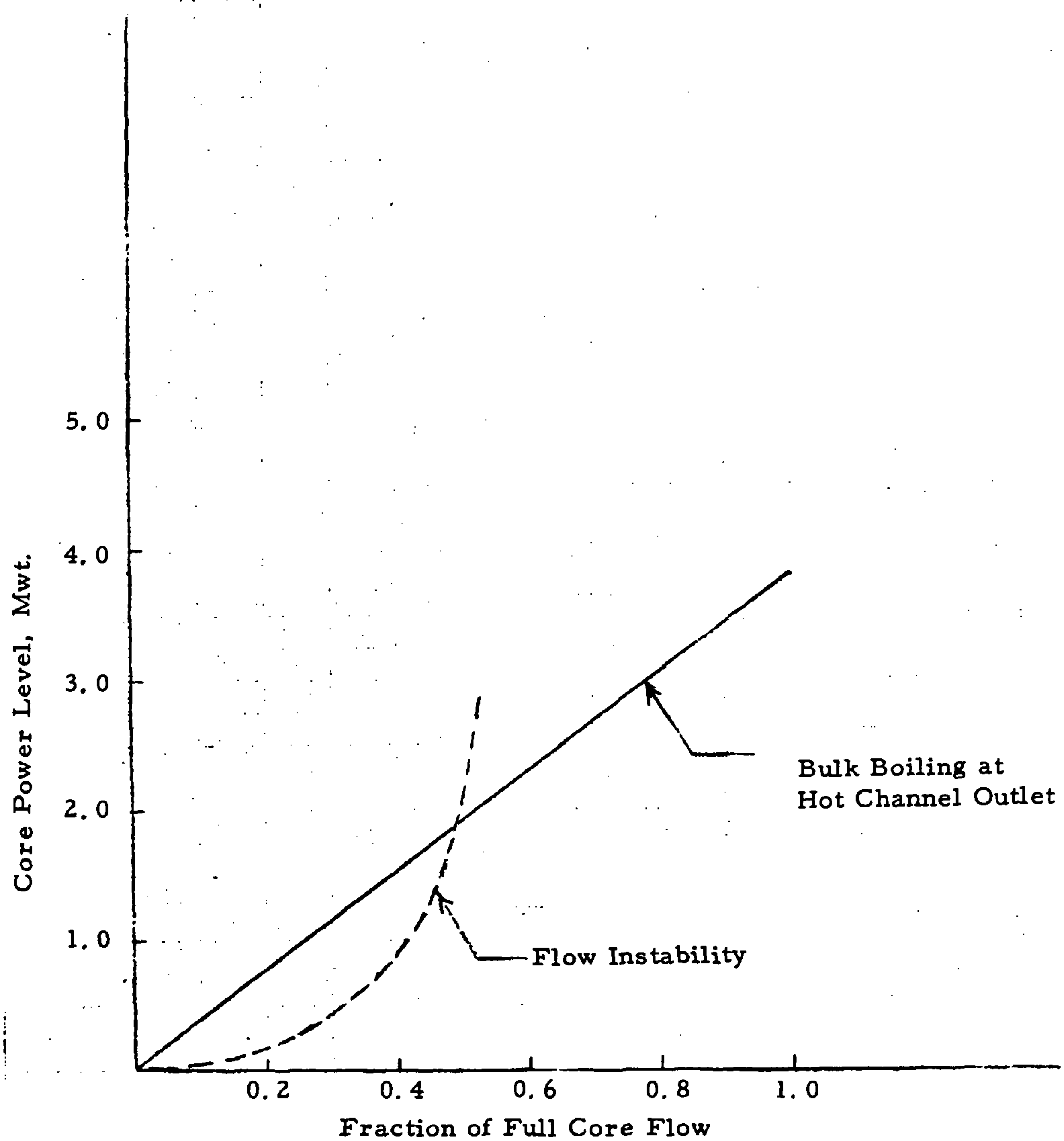
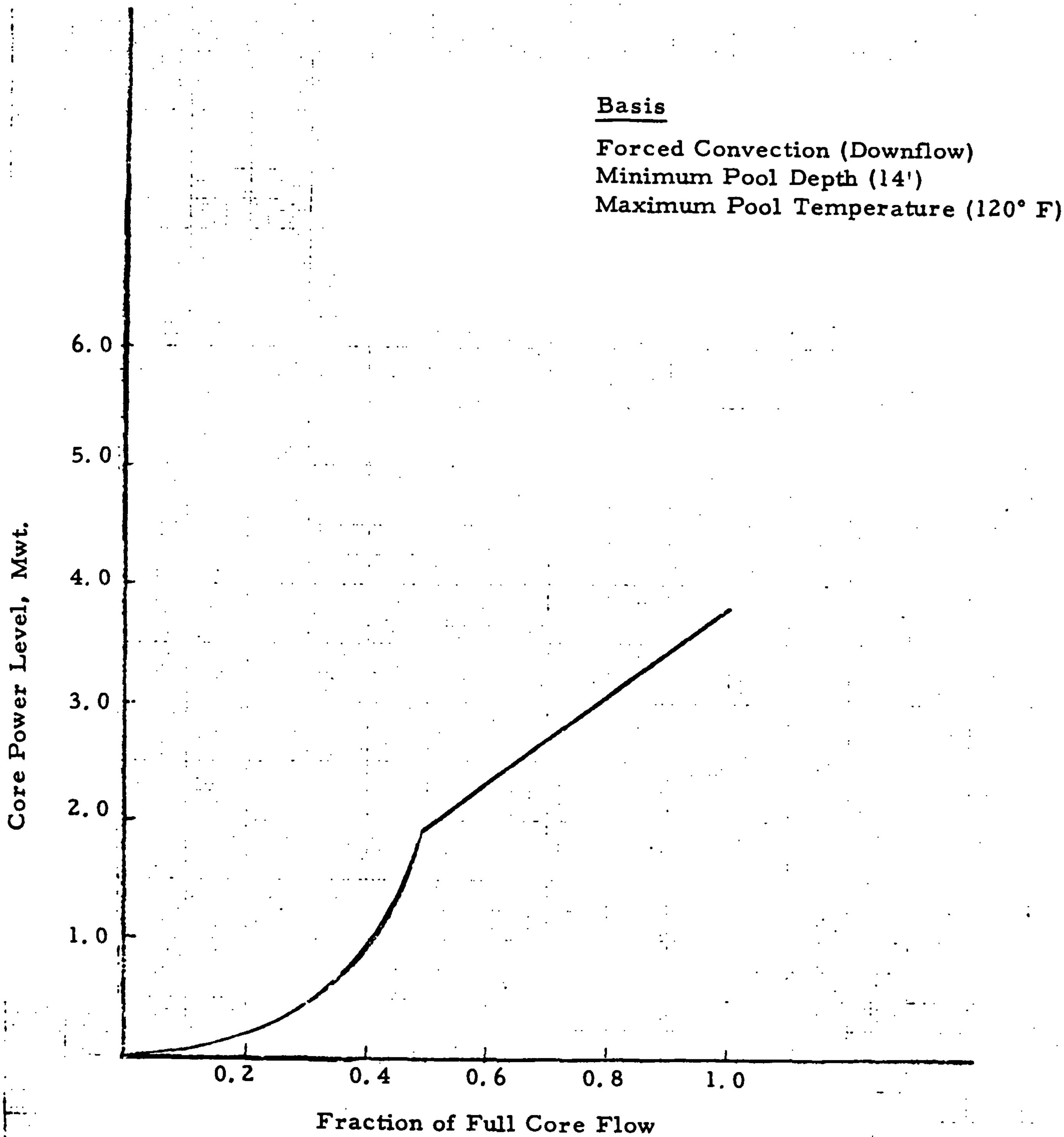


FIGURE III-8

POWER LEVELS REQUIRED TO REACH FLOW INSTABILITY OR BULK BOILING



Also included must be an allowance for calibration uncertainties and instrument inaccuracies. Section 3.2.4 of the SAR indicates a ± 7 percent of full power instrumentation uncertainty for power level measurement. In addition, a flow scram does not occur until the flow is less than 90 percent of nominal. Taking these factors into account and using Figure III-8, one can arrive at a corrected Safety Limit. See Appendix 3C for added analysis in support of the SL and LS^3. 5

IV. TRANSIENT ANALYSIS

A. Transient Model - Loss of Flow Incident

1. Introduction

As discussed in the previous section, the region between the Limiting Safety System Setting or LS^3 and the Safety Limit should be sufficient to allow for corrective action to be taken or the reactor shutdown following an abnormal occurrence. Such an occurrence worthy of analysis due to its possible severity is the loss of core flow due to a loss of pump power or a stuck rotor for example. The following sections deal with the analysis of such an incident.

2. General Relationships and Assumptions

A loss of flow incident in the NCSU PULSTAR reactor would lead to a core flow reversal since the downflow would eventually be terminated under the influence of a natural circulation driving head. The detailed analysis of such an incident is difficult since little is known about the magnitude of the film coefficient during the reversal or the coastdown period itself. In order to perform an analysis, therefore, one must make some conservative assumptions about the transient to assure a safe conclusion.

In the present analysis, it was assumed that the core flow stopped instantaneously at the time of loss of flow. The film coefficient was also assumed to drop to a conservatively low value representative of pool heat transfer conditions. The net effect of these assumptions was to suddenly decrease the heat transfer out of the fuel by a large factor. During the duration of the transient, it was further assumed that the film coefficient remained at this low value and the flow was not affected by natural convection effects which would occur once the flapper opened. In reality, this assumption of a sudden stoppage in core flow rate is not far from reality. The nominal flow velocity is only 1.8 feet per second which does not provide much flow momentum. Therefore, the coastdown period should indeed be quite short, on the order of a few seconds.

The assumption of a constant low film coefficient also is conservative since it neglects the fact that in the hot channel some nucleate boiling will occur. Nucleate boiling substantially increases the heat transfer coefficient and hence the heat transfer out of the fuel. The present assumption therefore will result in the greatest heat retention in the fuel and hence the greatest possibility of approaching high fuel temperatures or melting.

As will be seen, the key to the entire analysis is the control over the heat generation in the fuel. This control is obtained by scramming the reactor upon sensing a loss of flow. Since the nominal heat removal capability of the coolent is not far removed from natural convection conditions, due to the low nominal flow rate, a sudden decrease in flow does not severely inhibit this heat removal capability. In addition, the nominal heat input is low enough that the core coolant does not heat up at a fast rate even if no flow occurs.

A lumped parameter model of the nuclear core was utilized in the analysis. A heat balance on the fuel elements yields,

$$C_f M_f \frac{dT_f}{dt} = P - k_f(T_f - T_s) \tag{IV-1}$$

where:

C_f	=	specific heat of the fuel, Btu/lb
M_f	=	mass of fuel in the channel, lb
T_f	=	average fuel temperature, ^oF
P_f	=	heat generation rate in the fuel, Btu/sec
k_f	=	total conductance in the fuel elements from the radial position having the average fuel temperature to the surface of the fuel element, Btu/sec-^oF
T_s	=	clad surface temperature, ^oF
t	=	time, sec

The variation in heat generation rate in the fuel with time is described by the normal six-group reactor kinetics equations:

$$\frac{dP}{dt} = \left(\frac{\Delta\rho - \beta^*}{\ell^*}\right) P + \sum_{i=1}^{6} \lambda_i C_i \tag{IV-2}$$

$$\frac{dC_i}{dt} = \frac{\beta_i P}{\ell^*} - \lambda_i C_i \tag{IV-3}$$

where:

$\Delta\rho$	=	reactivity
β^*	=	effective neutron delay fraction
ℓ^*	=	effective neutron lifetime, sec
λ_i	=	decay time of the i^{th} group, Btu/sec
β_i	=	effective neutron delay fraction of the i^{th} group
C_i	=	term relating to the concentration of delayed neutrons of the i^{th} group, Btu/sec

Since a prime consideration in the analysis was the possible rise in fuel temperature due to heat storage, it was conservatively assumed that the reactor power was limited only by the reactivity insertion associated with the scram. In other words, Doppler and moderator temperature feedback effects were neglected. The reactivity due to control rod motion was treated as a delayed ramp function as follows:

$$\Delta\rho = f(t) \tag{IV-4}$$

where:

$f(t)$ = function defining the rate of reactivity insertion as a function of time

The fuel element surface temperature was defined by equating the rate at which heat was transferred to and from the cladding, neglecting the storage of heat in the cladding.

$$k_f(T_f - T_s) = h(T_s - T_c) \tag{IV-5}$$

where:

h = total heat transfer conductance from surface of the fuel elements to the coolant, Btu/sec ^oF

T_c = bulk average temperature of the channel coolant, ^oF

As discussed earlier, the channel coolant was assumed to remain stationary upon loss of flow and therefore the average coolant temperature rise is given by:

$$C_c M_c \frac{dT_c}{dt} = h(T_s - T_c) \tag{IV-6}$$

The hot spot and hot channel fuel centerline temperatures were treated through use of the hot channel factors presented in Section III. B. 2 of this report. The same heat transfer equations applied as given above except that the heat generation rate was increased to account for power peaking and related uncertainties, and the film temperature difference was increased through the use of F_{film}.

3. Analog Solution Using MIMIC

The coupled equations described above were solved using an analog simulation computer code designated MIMIC (Ref. 11)(Modified Integration Digital Analog Simulator). MIMIC is a digital computer program whose input language endows the digital computer, from the viewpoint of the user, with the parallel nature of the analog. At the same time, it eliminates the time and amplitude scaling problems associated with analog computation.

B. Results of Transient Analysis

1. Input Conditions

The input conditions used in the transient analyses are given in Table IV-1. The scram reactivity insertion rate was based on a rod drop time of 1 second and a total reactivity insertion of 1.40% $\Delta k/k$. This value was based on the minimum available reactivity worth even if the most reactive assembly is assumed to be stuck in the fully withdrawn position. The 1.40% $\Delta k/k$ value is equal to the reactivity worth assigned to the fuel temperature (Doppler) and moderator temperature variations from cold to hot full-load conditions together with the control margin, the Xe-135 buildup, and minimum shutdown margin allowances.

TABLE IV-1. INPUT CONDITIONS FOR TRANSIENT ANALYSIS

Total scram reactivity insertion	1.40% $\Delta k/k$
Reactivity insertion rate	1.40% $\Delta k/k$/sec
Scram circuitry delay time	0.050 sec
Low flow scram initiation delay	2, 20 sec
Initial core power level	1 Mwt
Coolant inlet temperature at t = o	120^{o}F

The change in the film coefficient was based on typical natural convection coefficients obtained for film temperature differences similar to those in the NCSU core. The nominal full flow film coefficient was about 700 Btu/hr-ft^{2} ^{o}F. It was assumed in the analysis that when loss of flow occurred, the film coefficient dropped instantaneously to a value of 200 Btu/hr-ft^{2} ^{o}F. This value was obtained from a correlation (Ref. 3) developed from data for heated vertical planes and cylinders in a pool. Conservative assumptions were made with respect to the film temperature difference and average film temperature needed for the evaluation of the correlation. It is expected that during the transient, no lower value of film coefficient could be obtained since some flow turbulence caused by a flow reversal, in addition to the onset of limited local boiling, would result in much better heat transfer.

2. Results

The results of the transient analyses are presented in Figures IV-1 through IV-6.

Figure IV-1 presents the core nuclear power level following a flow scram which is assumed to occur 20.05 seconds after the loss of flow. The ultraconservative assumption of a 20-second delay before 90 percent of nominal flow is reached was used to demonstrate the low heat storage in the fuel due to the low heat generation and good heat transfer out into the coolant. Another delay value of 2 seconds was also used since this is more representative of actual conditions. The power scram for the 2-second delay case is exactly the same as that shown in Figure IV-1 except that the curve is shifted to account for the smaller delay time. It should be noted that during the delay period the fuel temperature and moderative temperature will be rising resulting in a decrease in power level and hence heat generation rate due to the negative temperature coefficients of reactivity which in this analysis have conservatively been assumed to be zero.

Figure IV-2 shows the variation with time of local hot spot fuel centerline temperature for the two cases of scram delay time. It is clear from the figure that the large heat capacity of UO_2 in combination with the relatively good heat transfer out of the fuel even during a flow reversal results in a small fuel temperature rise during the transient. It can also be seen that even if the delay time were extended further, the low rate of fuel temperature rise would not result in any large rise in temperature.

Figure IV-3 presents calculated data showing the change in the average and hot channels bulk average coolant temperature with time. This temperature rise is based on zero flow from time zero. The rise therefore is directly proportional to the heat input from the fuel. As can be seen, the rate of temperature rise is relatively low, being only an average of about 1.5°F/sec for the worst condition of scram delay.

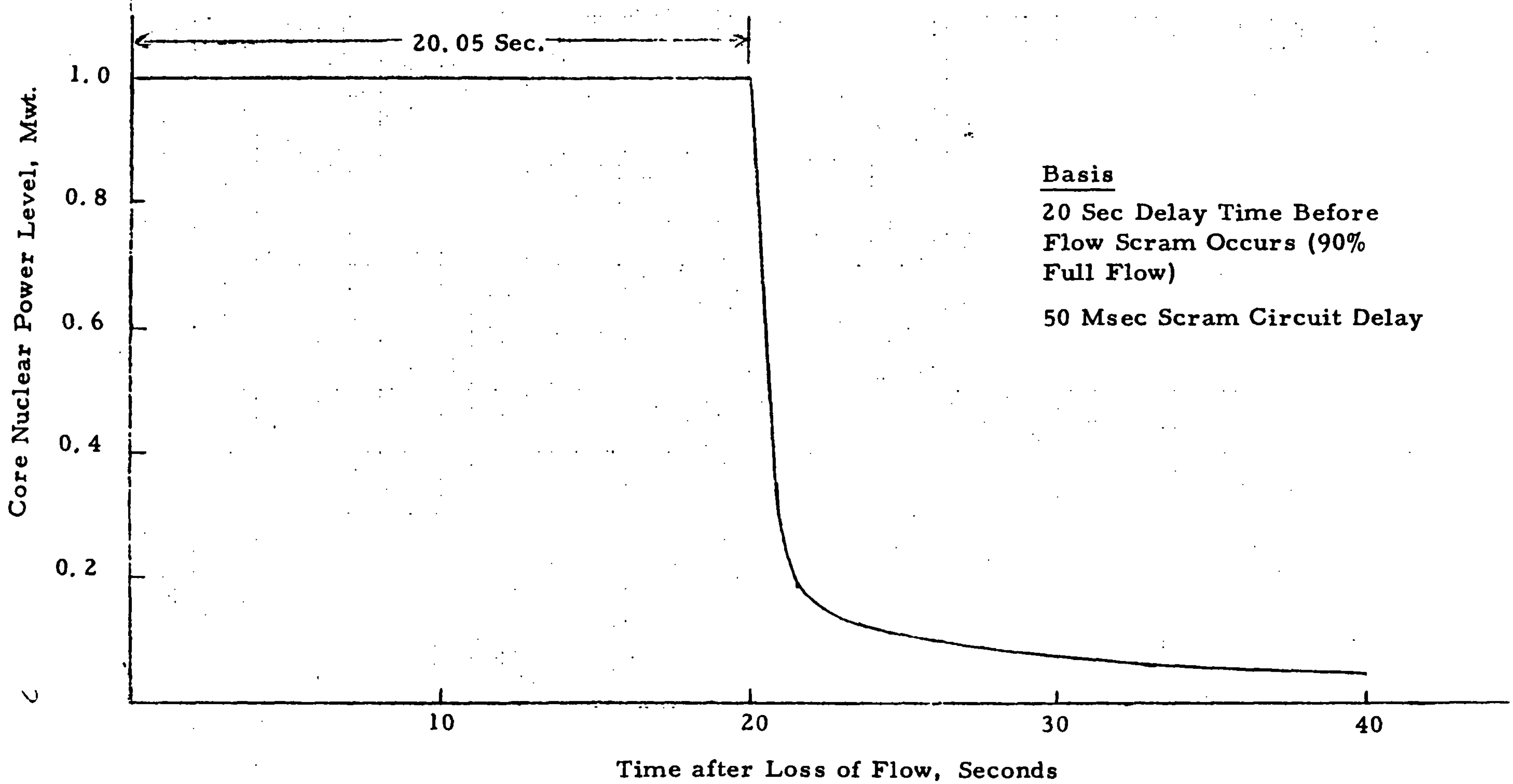
The time dependent heat fluxes for the average and hot channels are shown in Figures IV-4 and IV-5, along with the hot spot heat flux. Figure IV-6 presents curves of the hot spot DNB ratios for both cases of scram delay. The DNB ratio curves were based on a critical heat flux value of 4.9×10^5 Btu/hr/ft^2 as reflected previously in Figure III-4 of Section III. C. 1 for the pool boiling condition.

The analysis of Figure IV-6 indicates that during the course of the transient, the MDNBR remains far above the established 2.0 limitation.

General indications are that even if the postulated transient were initiated from a much higher power level, none of the established limiting criteria would be violated and reactor operation would proceed in a safe fashion.

FIGURE IV-1

CORE NUCLEAR POWER LEVEL FOLLOWING A POWER SCRAM DUE TO LOSS OF FLOW



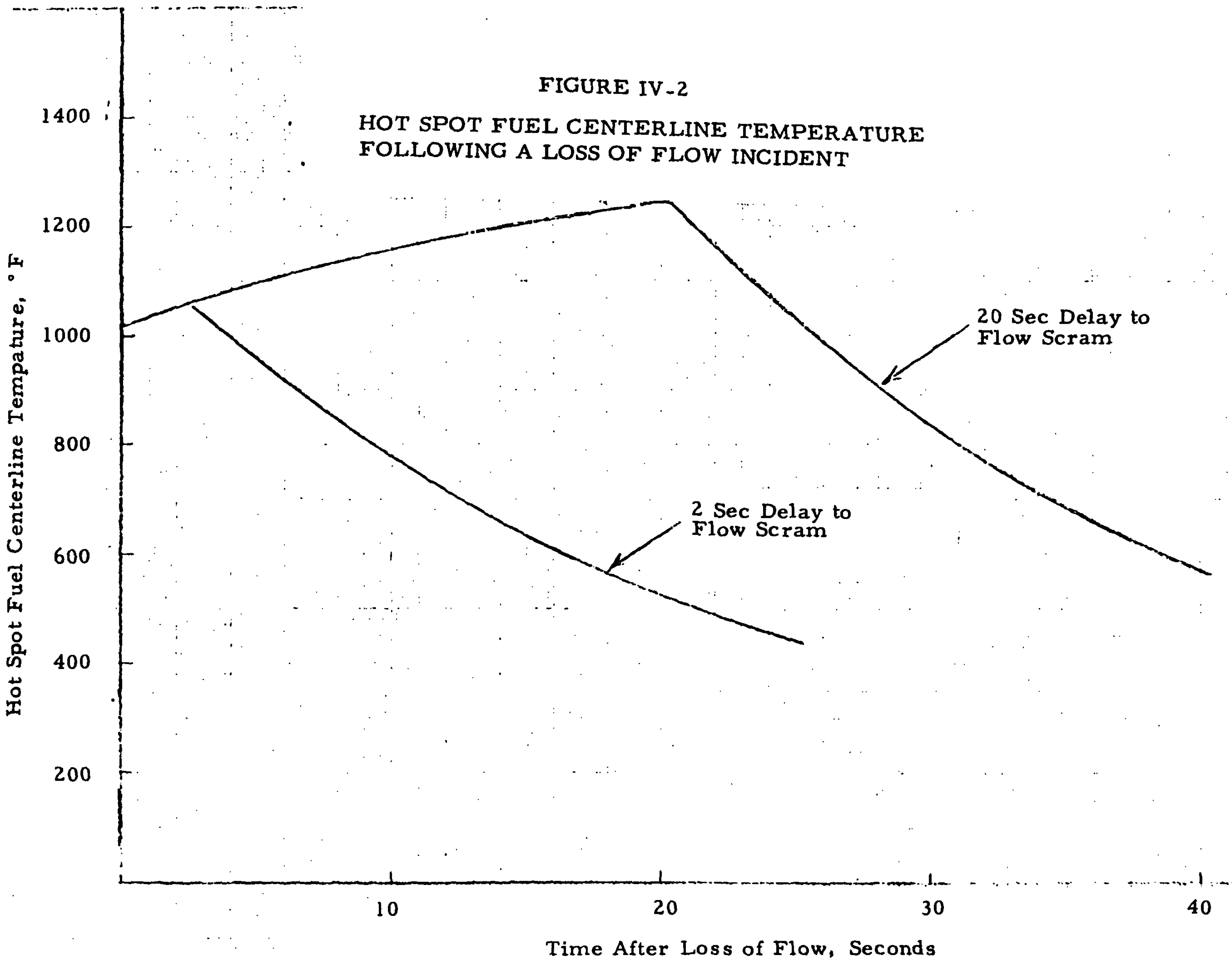
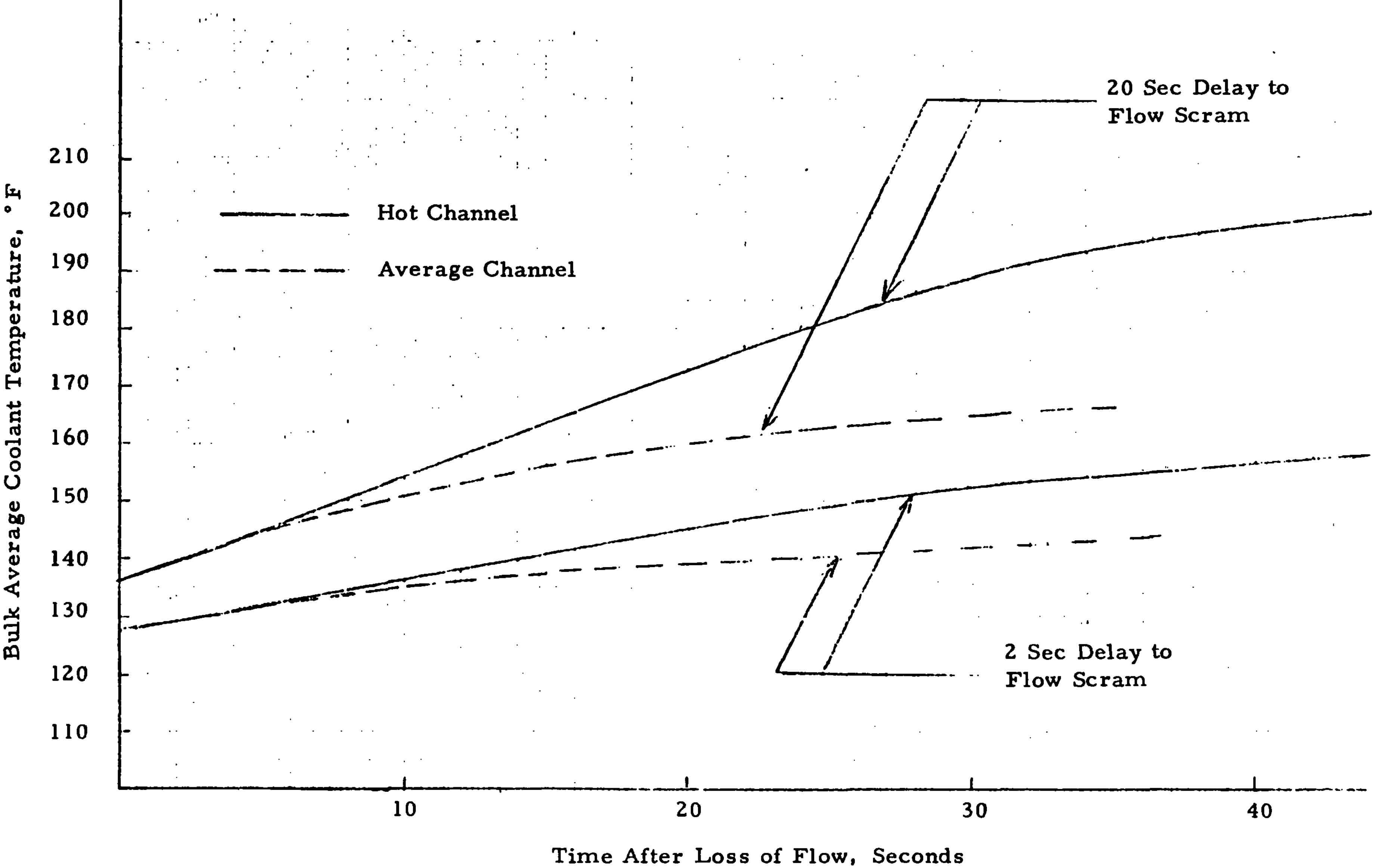


FIGURE IV-2

HOT SPOT FUEL CENTERLINE TEMPERATURE FOLLOWING A LOSS OF FLOW INCIDENT

FIGURE IV-3

AVERAGE AND HOT CHANNEL BULK AVERAGE COOLANT TEMPERATURE FOLLOWING A LOSS OF FLOW INCIDENT



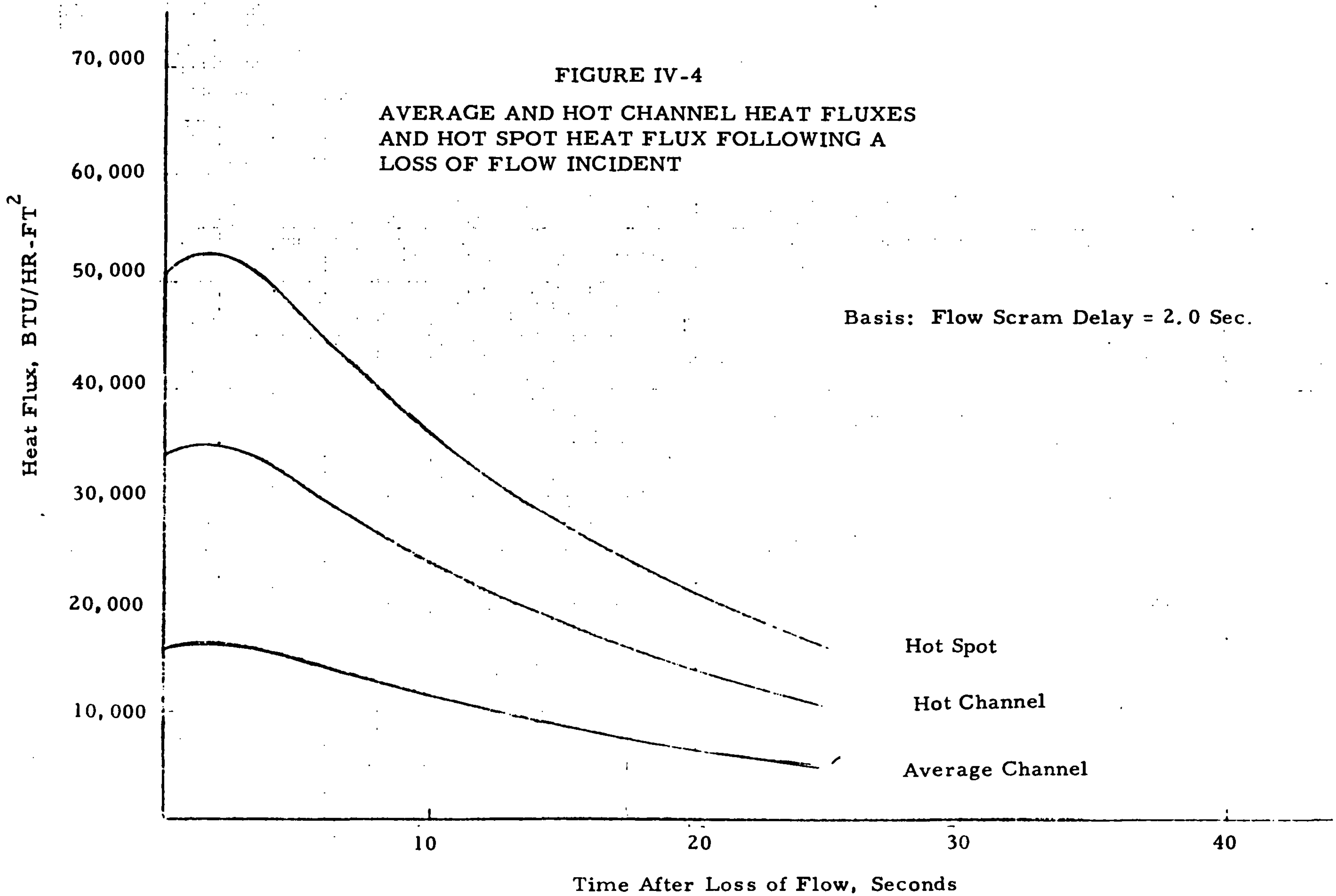


FIGURE IV-4

AVERAGE AND HOT CHANNEL HEAT FLUXES AND HOT SPOT HEAT FLUX FOLLOWING A LOSS OF FLOW INCIDENT

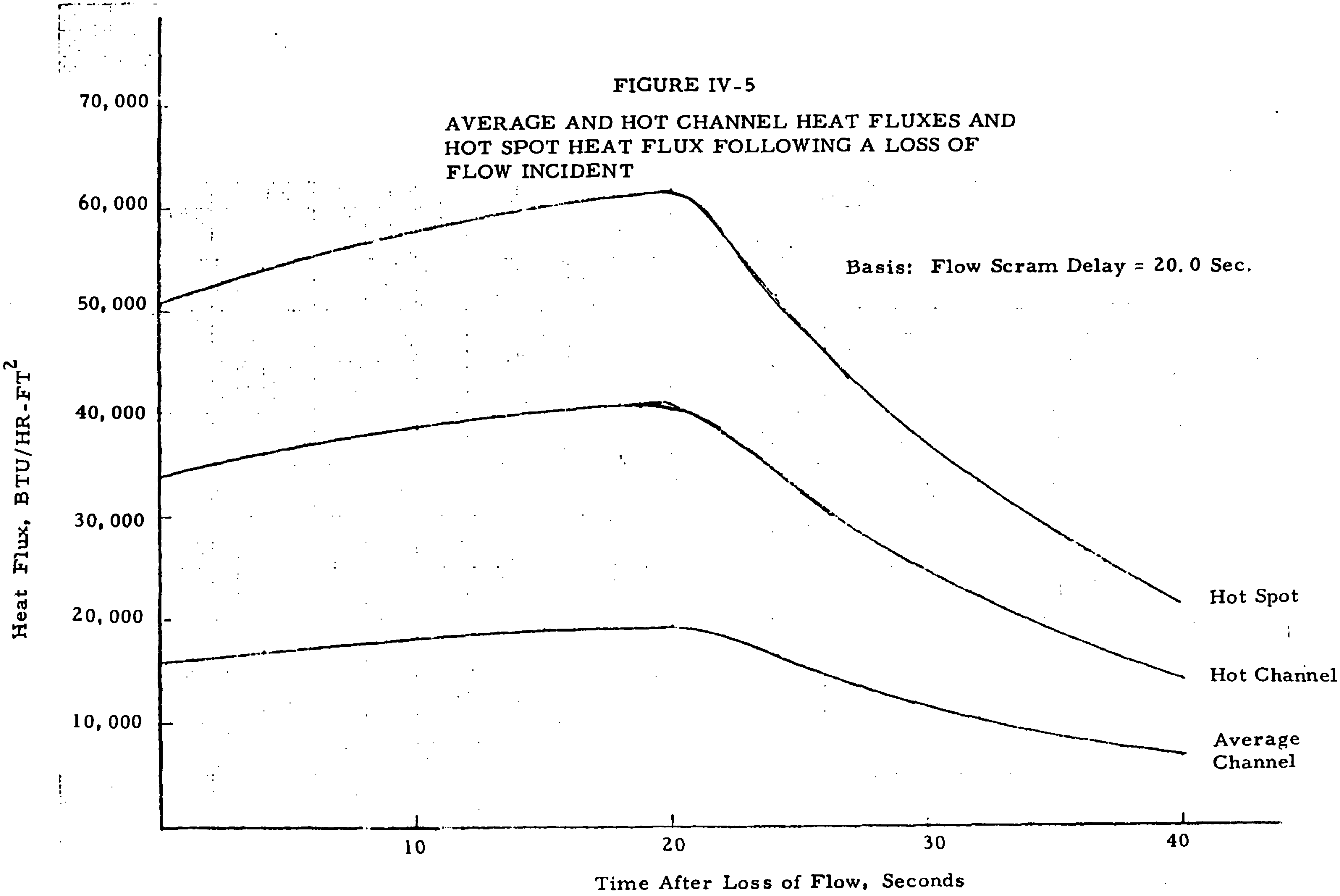


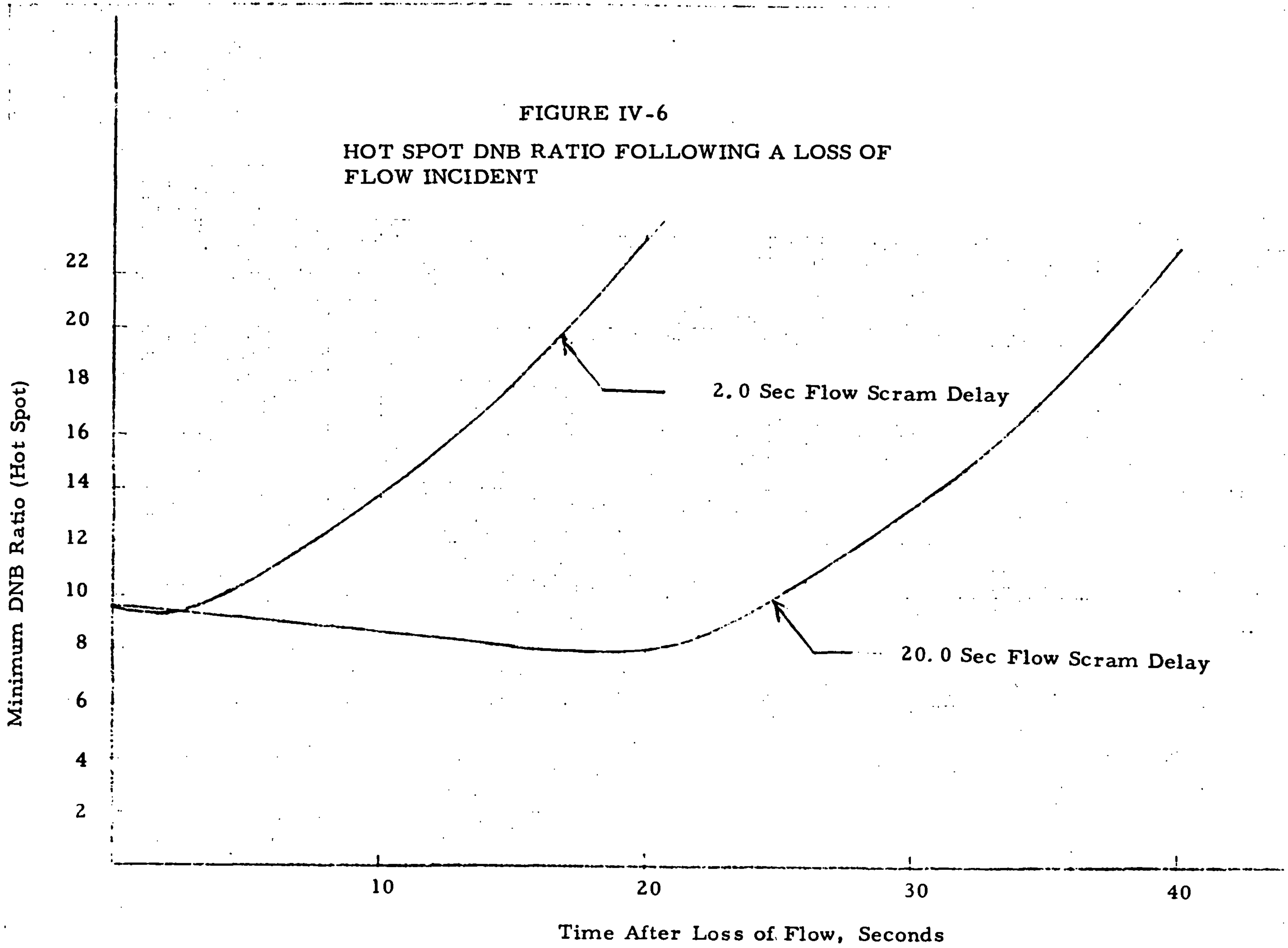
FIGURE IV-5

AVERAGE AND HOT CHANNEL HEAT FLUXES AND HOT SPOT HEAT FLUX FOLLOWING A LOSS OF FLOW INCIDENT

Basis: Flow Scram Delay = 20.0 Sec.

FIGURE IV-6

HOT SPOT DNB RATIO FOLLOWING A LOSS OF FLOW INCIDENT



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Appendix 3C

Additional Information and Analysis in Support of North Carolina State University PULSTAR Core and its Technical Specifications

Docket Number 50-297

(originally submitted in response to AEC request dated
16 November 1971, and now revised for Amendment 9, December 15, 1989)

This Appendix was prepared in response to a request for additional information for the NCSU PULSTAR reactor, reference AEC letter dated 16 November 1971. This additional information has been generated to further support the bases for the revisions of the Technical Specifications. These revisions also make the Technical Specifications more consistent with current regulatory practice and format. This Appendix is further revised in this latest amendment to remove pulsing and square wave related analysis no longer applicable.

To be consistent with the order of the revised Technical Specifications, this report is divided into several sections. Each section presents new data or calculations and/or a summary of old data which is applicable to the specification.

The revised Technical Specifications considers only the steady state mode of operation. In the Steady State mode there are two conditions of flow which are acceptable. These are forced convection and natural convection. The following sections will consider each of these modes and flow conditions from the point of view of establishing Safety Limits and Limiting Safety System Settings.

Steady State Mode - Forced Convection Flow

The original Safety Limit (SL) in the Steady State mode with forced convection flow was placed on power level. See Technical Specifications, April 26, 1971 the value chosen was 3.8 Mw. This power level was obtained from the analysis in Appendix 3B of the FSAR (HIT-486) and Figure III-8 of that report.

This single value approach is not entirely consistent with current regulatory practice and therefore appropriate modifications have been made to consider specifications on other important interrelated process variables. In addition, special considerations must be made with respect to the "application of acceptable criteria over the whole power versus flow range that will accommodate the entire course of the most adverse anticipated transient without exceeding the safety limits."

To accomplish the above it was decided to examine the fuel clad temperatures during the flow reversal transient which was analyzed in Appendix 3B. The MIMIC code was used in the Appendix 3B analysis. The computer output was reviewed for clad temperature data and this information is plotted in Figure 1 for the average and hot channel. The curves show the channel averaged clad temperatures for both a 2 second and a 20 second flow scram delay time. The hot spot clad temperature can be calculated approximately by the following:

$$(T_{clad})_{hot\ spot} = T_{bulk,HC} + F_{film}/F_b\ (T_{clad,HC} - T_{bulk,HC})$$

where:

$T_{bulk,HC}$ = Average bulk coolant temperature in the hot channel, °F

$T_{clad,HC}$ = Average clad temperature in the hot channel, °F

F_{film} = Film temperature rise hot spot factor

F_b = Bulk coolant temperature rise hot channel factor

The values of F_{film} and F_b are 3.77 and 2.12 respectively as given in Appendix 3B. The values of $T_{bulk, HC}$ and $T_{clad, HC}$ are obtained from Figure IV-3 in Appendix 3B and Figure 1 in this report. The results of the calculation of the hot spot clad temperature are shown in Figure 2. It must be noted that these clad temperatures (i.e., Figures 1 and 2) were calculated assuming an instantaneous flow decrease to zero flow. As such, the convection film coefficient was assumed to drop from about 600 (Btu/hr)/ft^2 °F to 200 (Btu/hr)ft^2 °F at time zero in a step fashion. Therefore, the initial values of clad temperature shown on Figures 1 and 2 at time zero are significantly higher than when flow was present due to the assumed step decrease in film coefficient. For example, the steady state value of clad temperature for the average channel is about 160°F. Following a step change in the film coefficient, this clad temperature will step up to 207°F as shown on Figure 1 at time zero. In actuality, the rise in clad temperature will be more gradual since the flow will not instantaneously drop to zero.

The thermal-hydraulic analysis in Appendix 3B did not consider nucleate boiling during the flow reversal analysis. This conservatively drove the clad temperatures higher than would actually be obtianed. This is due to the fact that when nucleate boiling occurs the film convection coefficient increases significantly allowing for greater heat transfer out of the fuel to the coolant.

One can estimate the effect of nucleate boiling on clad temperature rise during the flow reversal transient by looking at the heat flux required to initiate nucleate boiling. This is obtained from the Bergles and Rohsenow equation (see Appendix 3B, p. III-6) which is plotted in Figure 3.

A calculation of the hot spot clad temperature using the equation given previously for 1.0 MWt steady state full flow operation gives a value of 257°F and therefore a (T_{clad} - T_{clad}) of about 26°F where T_{sat} is 231°F for the minimum pool level of 14 feet. From Figure 3 one can see that for a hot spot heat flux of 8.3 x 10^4 Btu/hr-ft^2 nucleate boiling will initiate with a (T_{clad} - T_{sat}) of about 10°F. Therefore, the hot spot is in nucleate boiling and the use of the equation given previously is not valid. If we assume that fully developed nucleate boiling exists at the hot spot, then the McAdams correlation (see reference 1) can be used. This is also plotted in Figure 3. This curve yields a (T_{clad} - T_{sat}) of about 37°F at the hot spot heat flux. This is the upper bound on the wall temperature at the steady state hot spot heat flux.

Following a loss of flow, the nucleate boiling heat transfer coefficient will not change appreciably since it is not affected by flow velocity. Therefore, for the hot spot one would not expect for that change due to the relatively slow increase in bulk coolant temperature.

The average and hot channel clad temperatures are below saturation under steady state conditions. Following a loss of flow, the clad temperatures will rise due to the falling film coefficients. Eventually the critical (T_{clad} - T_{sat}) for nucleate boiling will be reached. At no time during the transient, however, will the heat flux at any point exceed the steady state heat flux at the hot spot. This is due to the fact that initially the heat flux drops due to the lower film coefficient. Thereafter, the flux will begin to rise as the fuel temperature increases due to the stored heat. However, this rise will be quickly

terminated upon initiation of a loss of flow SCRAM. Even without termination, the heat flux out of the fuel will be limited by the heat production in the fuel which for the duration of the transient does not rise above its initial steady state value. It should also be mentioned that this analysis does not take any credit for the negative Doppler coefficient of reactivity which would lower the core power upon a rise in fuel temperature.

For the reasons discussed above, the maximum clad temperature that will occur during a loss of flow transient will be 268°F. This is obtained by adding (T_{clad} - T_{sat}) obtained from Figure 3 for the hot spot heat flux (i.e., 37°F) to the saturation temperature of 231°F.

In order to account for the fact that the Limiting Safety System Setting (LSSS) for the NCSU PULSTAR will be 1.3 MW and that there is a ± 7% uncertainty in this setpoint, the flow transient should be initiated from a power level of 1.4 MW. Applying this 1.4 factor to the hot spot heat flux and using Figure 3 one can obtain the maximum clad temperature as was done for the initial power level of 1.0 MW. The resulting clad temperature is 273°F. This maximum expected value of clad temperature is far below the clad damage point (usually considered the melting point). In actuality, the clad temperature will never reach this value due to the short time prior to a flow scram (less than 3 seconds to reach the scram setpoint of 450 gpm upon loss of flow from 500 gpm) and due to such effects as a negative Doppler coefficient of reactivity which was conservatively neglected in this and the Appendix 3B analysis.

The Safety Limit curve can be constructed as shown in Figure 4. Below 80% of full flow (scram setpoint of 450 gpm plus a 10% uncertainty in flow measurement) the previous analysis shows that the maximum clad temperature reached during the transient will be less than 273°F. This temperature corresponds to a steady state power level of 1.4 MW or 140% of full power. Therefore, this power level was chosen as the Safety Limit on power level which will accommodate the entire course of the flow reversal transient. Above 80% of full flow the Safety Limit on power level is based on the criterion of no bulk boiling in the hot channel as analyzed previously in Appendix 3B.

Steady State Mode - Natural Convection Flow

There was little directly applicable information found on the natural convection mode of operation for the NCSU PULSTAR (e.g., power vs flow curves). However, reference 2 has data for the Buffalo Materials Research Center (BMRC) PULSTAR which may be applied to this analysis.

Section 3.2.4.1.3.2 in the SAR reviews some of the data which is presented in reference 2. The primary conclusion to be drawn from the BMRC data is that the core could be operated to as high as 1.15 MW with natural convection with no warping, bambooing, or other visible damage to the fuel rods. In fact, much higher power levels would have been acceptable but the tests were concluded without exceeding 1.15 MW. Analyses in Appendix 3B indicate that burnout will probably not be reached for power levels of less than 3.5 MW under any flow conditions.

To examine clad temperatures under conditions of natural convection flow we must determine the flow rate as a function of heat input. Figure 5 is a plot of bulk coolant temperature rise across the test element as a function of heat input for the BMRC

natural convection test. The test data was collected with and without the use of a chimney which aids natural convection flow. All of the data, however, falls below the upper curve which will be used to calculate flow. Use of the upper curve will yield the lowest calculated flow and hence is conservative. The formula used to

calculate the flow is:

$$m = Q/c\Delta T$$

which can be rewritten as:

$$GPM = [(kW)(3.413 \times 10^3)] / [(\Delta T)(60)(62.4)(.1337)] = 6.83(kW)/(\Delta T)$$

where simply the flow rate in gpm is given directly from the kW heat input and ΔT across the test element. The results of this calculation are presented in Table 1. It can be seen that the increase in flow rate with heat input is not linear due to the fast increasing pressure losses which are approximately proportional to the square of the flow velocity. It should be noted that the power-flow curve for the hot channel will not be much different than for the average channel since the greater heat input will draw more flow. Examination of Table 1 also shows that even for power levels as high as 2000 kW, the flow is only about 22.6% of full flow. This low flow will result in a low film coefficient of heat transfer and nucleate boiling at the hot spot is a certainty. Using Figure 3 presented earlier in this report we can determine (T_{clad} - T_{sat}) as a function of power level from the McAdams curve. The maximum bulk coolant temperature in a channel can be determined from Figure 5 for different power levels and for the assumption that the maximum coolant inlet temperature will be 120°F. With this information one can construct a curve of hot spot clad temperature versus power level under conditions of natural circulation flow. This curve is shown in Figure 6. In addition, the peak bulk coolant temperature in the average channel is shown and from this one can determine that no bulk boiling in the channel will occur for power levels below 1.77 MW.

The power level at which the hot spot clad temperature is 273°F (i.e., the SL chosen for forced circulation) is 1.4 MW as was determined for forced circulation conditions. Therefore, just as for forced convection, a Safety Limit on power level of 1.4 MW will maintain the clad temperature below the chosen limit of 273°F which is far below the clad melting point. In addition, bulk boiling will not occur for power levels below 1.4 MW.

It might be noted that the reason why one obtains the same power level-clad temperature relation for natural circulation flow as for forced flow is that the McAdams curve for fully developed nucleate boiling (Figure 3) is essentially independent of flow velocity. This is especially relevant for the NCSU PULSTAR due to the low nominal flow velocity at full flow.

During natural convection times, it is desirable to eliminate any bulk boiling in the coolant channels due to the possibility of releasing N-16 via steam bubble rise in the pool. Bulk boiling in natural convection does not affect flow stability since flow is upward. Therefore, any limitation on bulk boiling is not related to fuel clad performance. An analysis of bulk boiling in the hot channel is presented in the SAR, Section 3.2.4.1.3.2. This analysis makes the conservative assumption that the flow in a channel is not affected by heat input. Therefore, the fact that in actuality the hot channel will draw more flow is neglected resulting in the greatest coolant temperature rise. The analysis presented in the SAR shows that bulk boiling at the outlet of the hot channel will occur at 680 kWt.

Pulse Mode (Not Allowed)

Analyses of Pulse Mode operation are presented in the SAR, Section 3.2.4.2.3. As of this amendment, pulsing is discontinued at the NCSU PULSTAR, however, this analysis remains in support of Section 13 accident scenarios.

Conclusions drawn in reference 3 indicate that there is a substantial conservative margin associated with the pulse energy which was found to result in local film boiling at the hot spot. It was concluded for example that fuel pin bowing, which results from film boiling, does not necessarily constitute a hazard since the bowed pins would be effectively pre-stressed, and subsequent repetitive pulses at the same energy release would not exceed yield stress. Moreover, during the natural convection portion of the BMRC program, test pins were repetitively subjected to specific energy releases of 872 watt·seconds/gram at the hot spots. For the NCSU PULSTAR this energy density corresponds to a total pulse energy of about 107 MW·sec, where:

$$E_{total} \text{ (in MW·sec)} = W\,(872)(10^{-3})/F_p$$

where,

W = weight of UO_2 in core = 359 kg

F_p = local power peaking factor = 2.92

A method which can be used to estimate the pulse energy release which will initiate film boiling is based on the correlation curve of pin performance under forced convection cooling as presented in reference 3. From the FSAR Section 3.2.4.2.3, the "corner core" pin in the NCSU PULSTAR core has a correlation factor of 0.317 as opposed to a factor of 0.463 for the BMRC PULSTAR. The difference is due to the fact that the core coolant flow rate is significantly less for the NCSU PULSTAR than the BMRC Core. The correlation factor of 0.317 is based on a core flow rate of 400 gpm which is 80% of full flow. To establish the limiting value one should use a flow lower than the Limiting Safety System Setting of 450 gpm. For this analysis 400 gpm was selected for the Safety Limit on flow for the pulse mode. This provides a 10% of full flow uncertainty between the LSSS and the Safety Limit.

Using the corner core pin correlation factor of 0.317 and Figure 3 in Reference 3, one obtains a maximum energy density of about 470 watt-seconds per gram of UO_2. This value is obtained from the upper curve in Figure 3 of Reference 3, which indicates slight film boiling. This limited film boiling was found acceptable based on examinations of the fuel pins after the tests. The total pulse energy input corresponding to this energy density is 58 MW·sec, as calculated by the equation for E_{total} given above. With no additional information, one must assume that this value of pulse energy input is the limiting value or criterion to avoid possible fuel clad damage due to film boiling effects. In reality, however, the true limit is significantly higher. This is due to the fact that short term, localized film boiling (e.g., during a very brief time following a pulse) is not necessarily harmful or damaging since its time duration for pulse conditions is quite short. In addition, there is data which has been noted earlier which

confirms that limited film boiling does not result in fuel clad damage.

If it is conservatively assumed that a total pulse energy input of 58 MW·seconds is the limit to avoid possible fuel clad damage, then the corresponding reactivity insertion can be found which will produce this limiting pulse. This reactivity insertion is the key process variable upon which a Safety Limit must be placed to avoid transgressing the safety criterion of maximum allowable pulse energy input. Other process variables which must be considered since they are interrelated are the core coolant flow rate which influences worst pin correlation factor and the core initial power level. The bulk coolant temperature is not of great significance during a pulse as long as it is within the general operating range used in other modes of operation. The pulse is so fast that there is essentially no coolant temperature response until much later, and what response there is is quite slow. On the other hand, the initial core power level is important since it determines the initial fuel temperature. If the initial power level were high for example, fuel melting might occur at the hot spot following a pulse. For example, in the NCSU PULSTAR the limit on energy density was given previously as 470 watt·seconds per gram of UO_2. This can be equated to an equivalent adiabatic fuel temperature rise as follows:

$$\Delta T_{fuel} = (470 \text{ watt·sec/gm } UO_2)(454 \text{ gm/lb})(3.413 \text{ Btu/watt·hr})(1 \text{ hr}/3600 \text{ sec})(1°F/0.075 \text{ Btu/lb})$$

$$\Delta T_{fuel} = 2700°F$$

If we assume a limit on fuel temperature of 5080°F (UO_2 melting point) then the initial fuel temperature cannot exceed 5080-2700 or 2380°F at the hot spot centerline. This centerline temperature corresponds to a steady state power level of approximately 2.4 MW. This may be determined from the following:

$$(T_f)_{center,\,HS} = (T_{bulk})_{HC} + (q'')(HSF)/(U)$$

where:

$(T_f)_{center,\,HS}$	= Fuel centerline temperature at the hot spot
$(T_{bulk})_{HC}$	= Average bulk coolant temperature in the hot channel
q''	= Core average heat flux
HSF	= Hot spot factor
U	= Overall heat transfer coefficient

The peak temperature at the hot spot fuel centerline expected during 58 MW·second pulse from 250 kW would be about 3060°F which is well below fuel melting temperatures.

The reactivity insertion which will result in a pulse energy input of 58 MW·seconds can be determined from Figures 3-20 in the SAR.

To summarize, the specified Safety Limit, viz., 58 MW·sec, based on a maximum energy density criterion in the fuel of 470 watt·sec/gm, provides a significant margin of safety and avoids damage from film boiling at the hot spot.

References

1. Tong, C.S., "Boiling Heat Transfer and Two Phase Flow," John Wiley and Sons, 1965.
2. Technical Note J-435, Western New York Nuclear Research Center, Inc., December, 1966.
3. WNY-023, WNYNRC, May 26, 1966.

TABLE 1

NATURAL CONVECTION FLOW RATE AS A FUNCTION OF HEAT INPUT (POWER LEVEL)

Power (kW)	△T max (°F)	Flow (Core) (gpm)	Fraction of Full Flow (500 gpm)
100	20	34	.068
200	32	43	.086
300	40	51	.102
400	48	57	.114
500	54	63	.126
600	60	68	.136
800	71	77	.154
1000	81	84	.168
1200	89	92	.184
1400	97	98	.196
1600	105	104	.208
1800	113	109	.218
2000	121	113	.226

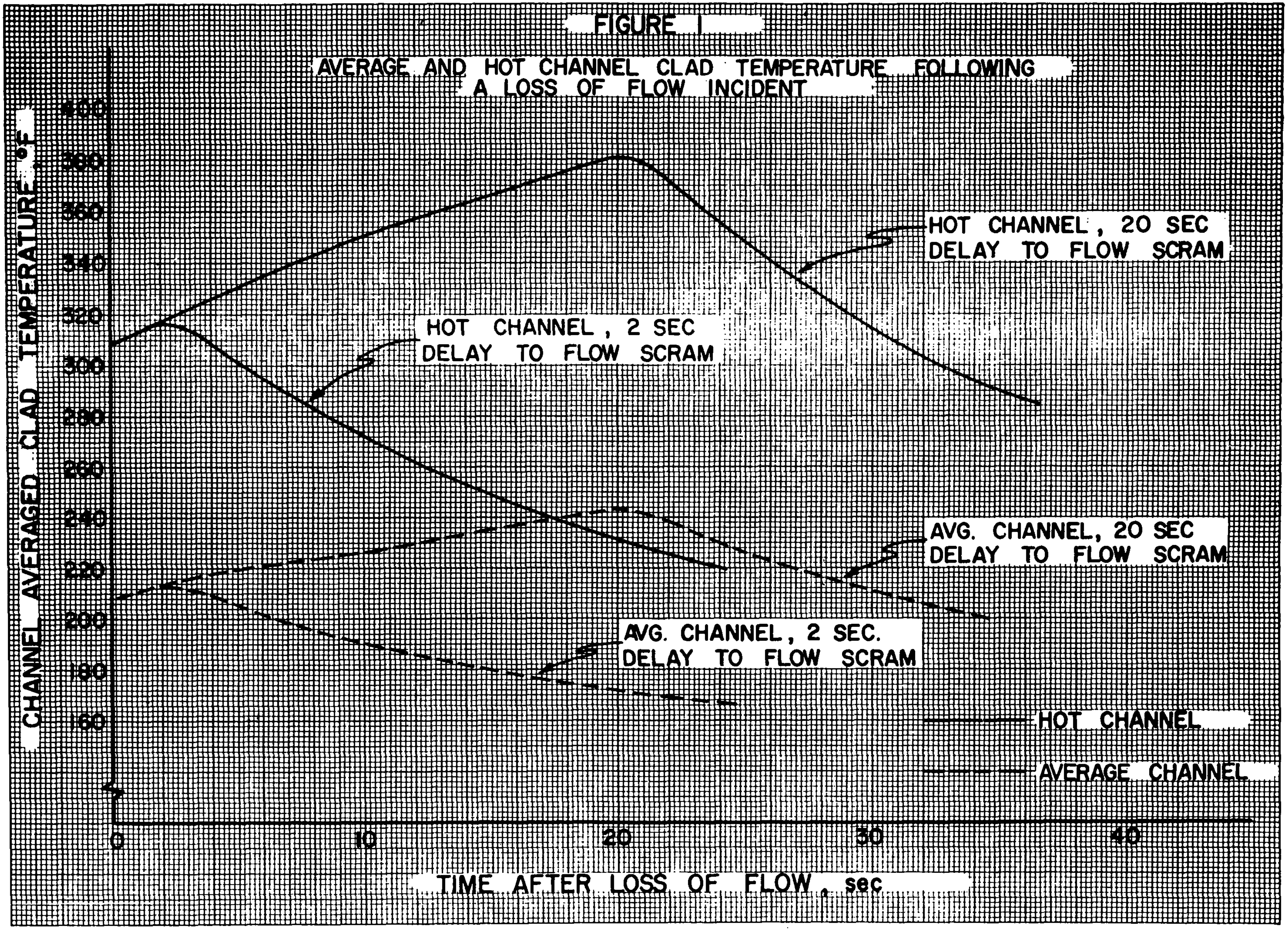


FIGURE I
AVERAGE AND HOT CHANNEL CLAD TEMPERATURE FOLLOWING A LOSS OF FLOW INCIDENT
CHANNEL AVERAGED CLAD TEMPERATURE, °F
400
380
360
340
320
300
280
260
240
220
200
180
160
HOT CHANNEL, 20 SEC DELAY TO FLOW SCRAM
HOT CHANNEL, 2 SEC DELAY TO FLOW SCRAM
AVG. CHANNEL, 20 SEC DELAY TO FLOW SCRAM
AVG. CHANNEL, 2 SEC. DELAY TO FLOW SCRAM
HOT CHANNEL
AVERAGE CHANNEL
0
10
20
30
40
TIME AFTER LOSS OF FLOW, sec

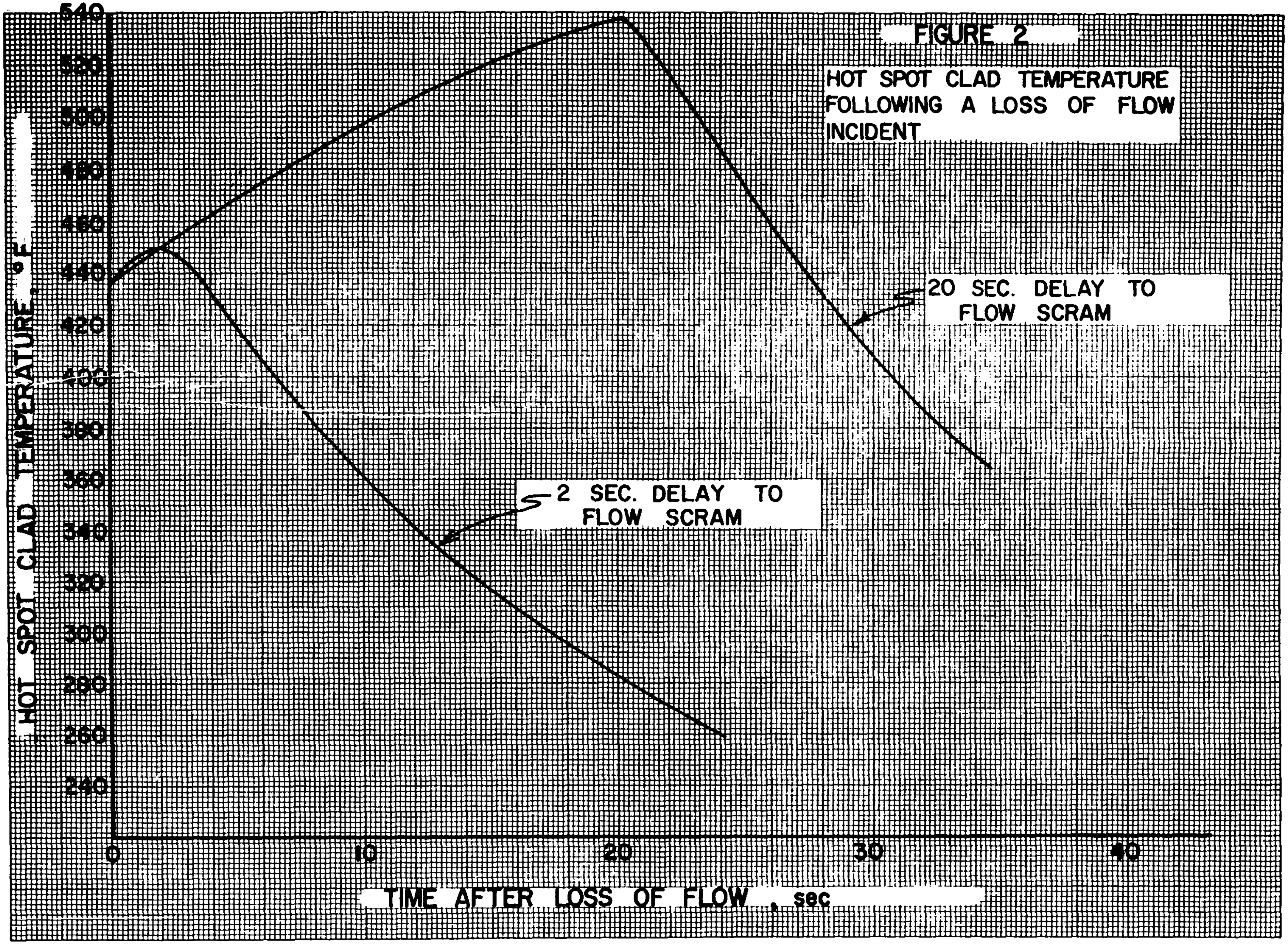


FIGURE 2
HOT SPOT CLAD TEMPERATURE FOLLOWING A LOSS OF FLOW INCIDENT
20 SEC. DELAY TO FLOW SCRAM
2 SEC. DELAY TO FLOW SCRAM
HOT SPOT CLAD TEMPERATURE, °F
540
520
500
480
460
440
420
400
380
360
340
320
300
280
260
240
0
10
20
30
40
TIME AFTER LOSS OF FLOW, sec

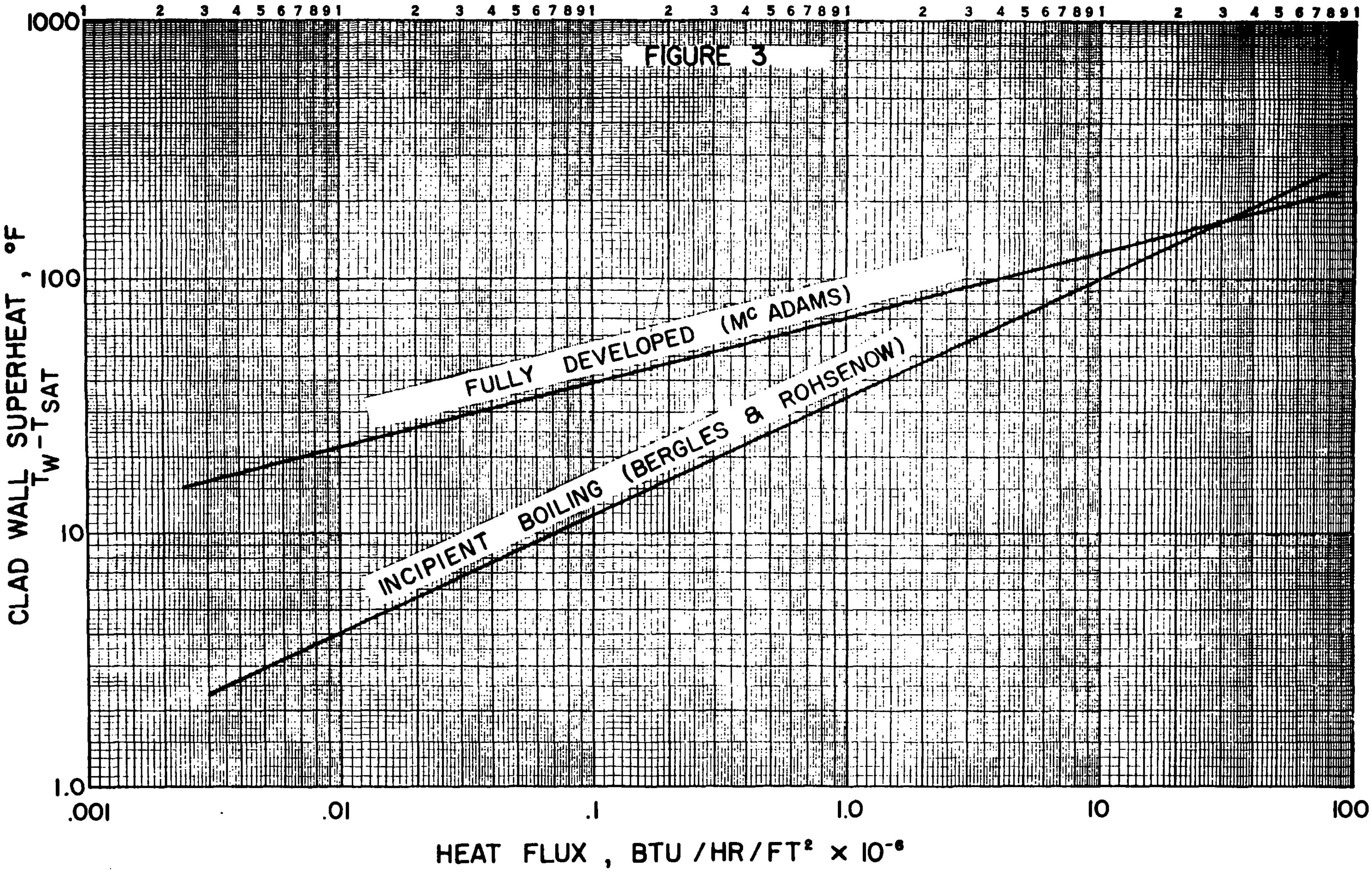


FIGURE 3
FULLY DEVELOPED (Mc ADAMS)
INCIPIENT BOILING (BERGLES & ROHSENOW)
CLAD WALL SUPERHEAT, °F
$T_W - T_{SAT}$
HEAT FLUX, BTU/HR/FT² × 10⁻⁶
1000
100
10
1.0
.001
.01
.1
1.0
10
100

FIGURE 4

POWER-FLOW SAFETY LIMIT CURVE

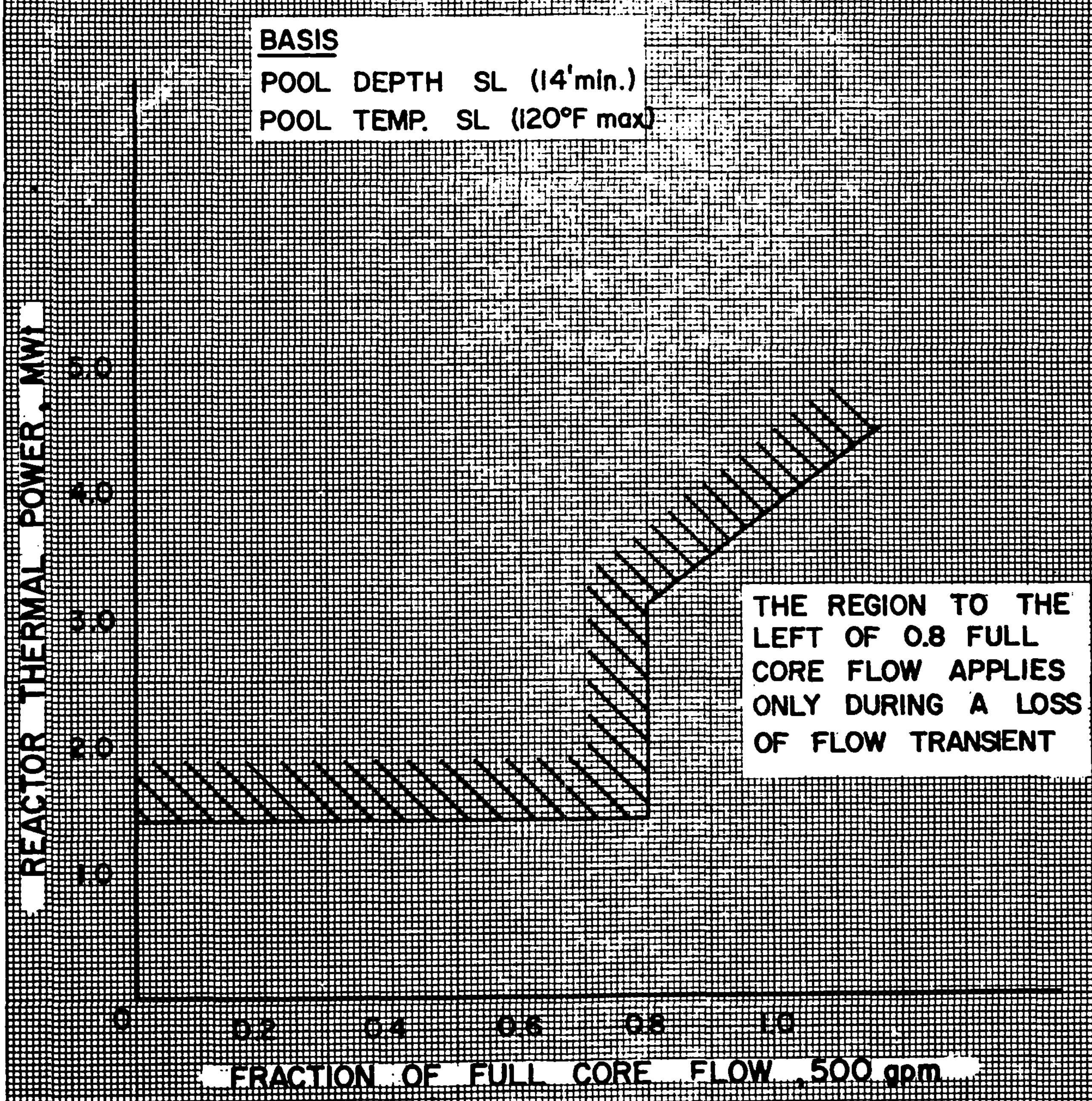
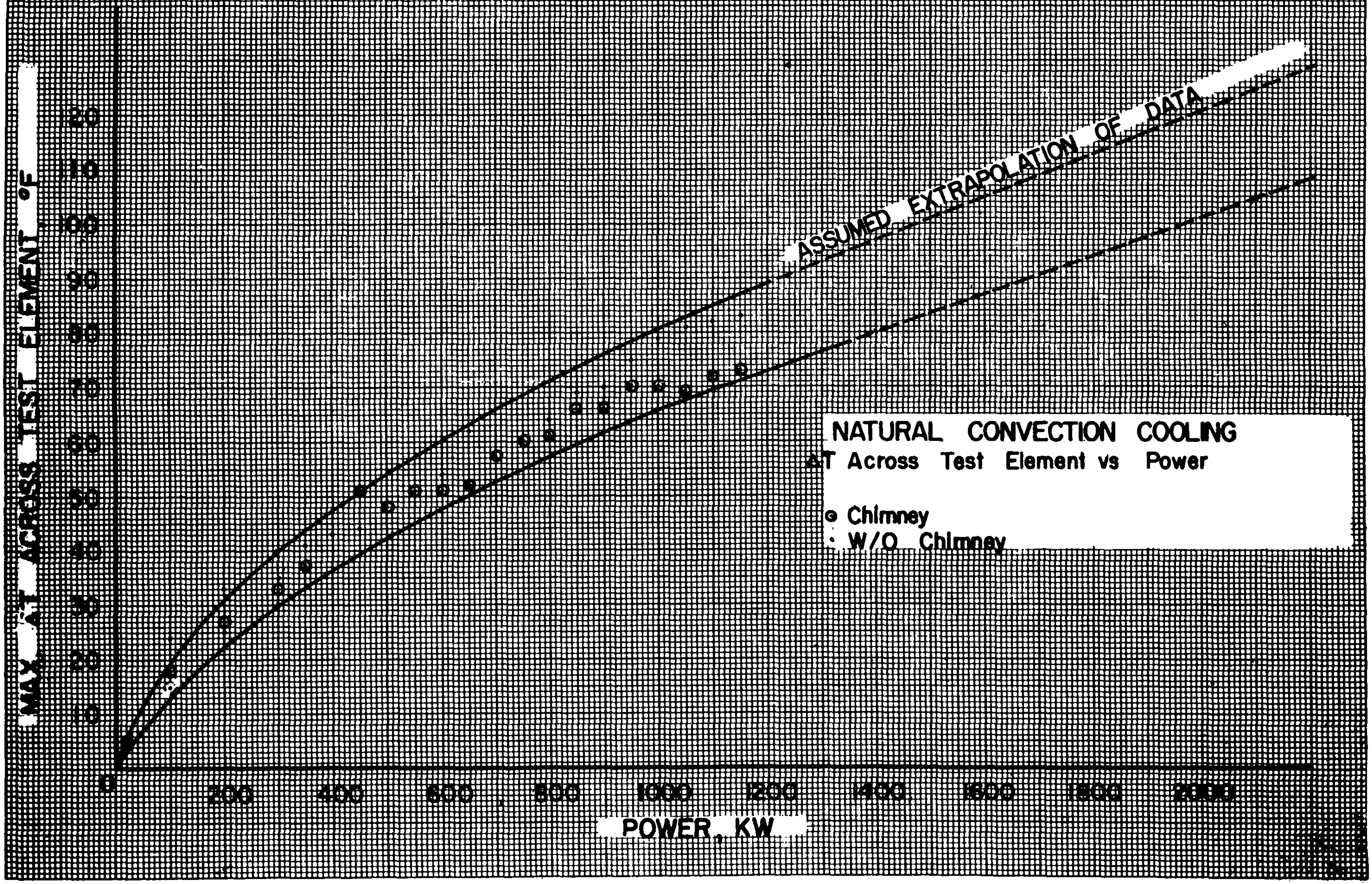


FIGURE 5



MAX. ΔT ACROSS TEST ELEMENT °F
120
110
100
90
80
70
60
50
40
30
20
10
0
200
400
600
800
1000
1200
1400
1600
1800
2000
POWER, KW
ASSUMED EXTRAPOLATION OF DATA
NATURAL CONVECTION COOLING
ΔT Across Test Element vs Power
Chimney
W/O Chimney

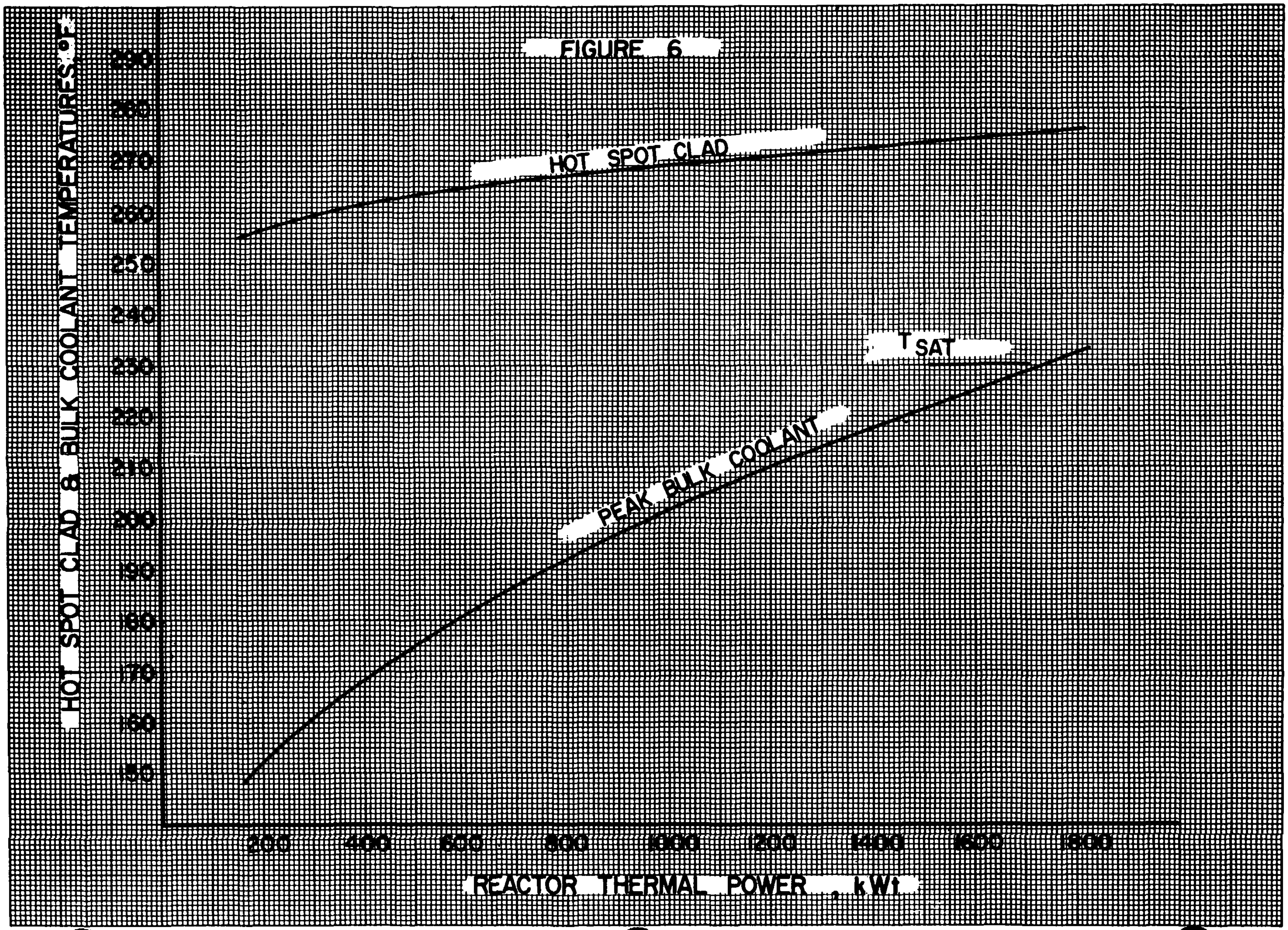


FIGURE 6
HOT SPOT CLAD
T SAT
PEAK BULK COOLANT
HOT SPOT CLAD & BULK COOLANT TEMPERATURES, °F
280
280
270
260
250
240
230
220
210
200
190
180
170
160
150
200
400
600
800
1000
1200
1400
1600
1800
REACTOR THERMAL POWER, kWt

Appendix 13-A

ABSTRACT

The total dose to a receptor in the unrestricted areas proximate to the NCSU PULSTAR Reactor facility is calculated in SAR Chapters 10 and 13 for normal and postulated accident conditions, respectively. Specifically, the total dose at the nearest permanent housing and at a maximum ground location for different meteorological conditions are evaluated in the SAR chapters.

SAR source term for normal operation is the noble gas ^{41}Ar. A hypothetical accident involving the failure of three fuel pins will have as the source term the radioactive noble gas inventory of the gap.

1.0 INTRODUCTION

The NCSU PULSTAR research reactor applied for a facility license (R-120) renewal on August 19, 1988. The Nuclear Regulatory Commission required additional information in the review of the application. This appendix documents the methodology and computer model established to address these questions. The contents of the appendix are based on calculation NRP-91-01, "PULSTAR Reactor Facility Offsite Dose Calculation", prepared by the Nuclear Reactor Program Staff.

The licensing of a nuclear facility requires that doses to unrestricted areas be calculated for normal and accident conditions. The NRC requested that the original PULSTAR calculation be updated to include the "shine" dose contribution from a passing cloud of radioactive constituents.

The question asked is typical for power plants in which a significant radionuclide inventory is present. There have been many detailed studies, both experimental and theoretical, on hypothetical releases from facilities such as nuclear power plants. These facilities share a few common design and site criteria. Typically, power plants are not situated near suburban areas and any gases released would be from stacks of significant heights. The PULSTAR reactor, however, is situated near major structures and has a relatively low stack height. This results in much of the data available for power plant calculations to be inconsistent with our facility's calculation.

It must be made clear that the PULSTAR facility does not represent a radiological threat to the general public. The safety features of the facility and conservative safety factors would prevent any significant radiological releases. Nevertheless, this facility must comply with the federal requirements to calculate offsite doses.

The specific questions which are answered to support the PULSTAR facility re-licensing are the following:

Whole-body doses to a ground receptor due to a plume emitted from the exhaust stack consisting of the following radionuclides;

a. Argon-41
b. Fuel Pin Gap Gas Inventory for Hypothetical Accident

at a maximum ground location and nearest permanent housing (240 meters) using conservative atmospheric conditions.

The models for predicting the dose rates from elevated releases were developed in the referenced literature. The analysis utilizes these models for the PULSTAR relicensing calculations. A computer model was developed specifically to address ^{41}Ar and fuel pin gap noble gas inventory releases from a 30 meter exhaust stack. This appendix also includes the verification and validation of the computer model.

The computer model was utilized in a parametric study in order to obtain conservative whole-body dose estimates. The analysis includes combinations of atmospheric stability classes along with wind velocity to calculate dose to a ground level receptor at various locations from the release point. In addition, the model considers the special case of effluents released from points less than the height of adjacent solid structures.

2.0 METHOD OF SOLUTION

There are no known calculations which involve a 30 meter stack height near a populated area. The references studied to support this analysis involve elevated releases and relatively distant downstream doses. These results can not be directly applied to the PULSTAR calculation.

A computer model was written by the NCSU staff to calculate the dose rate to the unrestricted areas around the PULSTAR facility. The model allows for the actual release height to be used.

2.1 Computer Model

The computer model calculates the average dose rate from releases considering wind shifting and varying atmospheric stability conditions. The model is based on the well known Gaussian distribution given by the following expression [1,2]:

$$\chi(x,y,z) = \frac{Q_x'}{2\pi U \sigma_y \sigma_z} G(z)\, e^{-\left(\frac{y^2}{2\sigma_y^2}\right)}$$

$$G(z) = e^{-\left(\frac{h-z}{2\sigma_z^2}\right)} + e^{-\left(\frac{h+z}{2\sigma_z^2}\right)}$$

χ(x,y,z) is the radionuclide concentration at the receptor location (x,y,z). The source of radioactive material is assumed to be released at a constant rate of Q'_0 (Curies/year) at some height (H) above the receptor located a distance "y" from the release point. The term Q'_x is the specific activity release rate for a given nuclide corrected for decay during travel for a given wind speed "U".

$$Q_x' = Q_0' e^{-\frac{\lambda R}{U}}$$

The lateral and vertical standard deviations for the plume dispersion are give by σ_y and σ_z respectively.

It must be assumed for long duration releases that the wind direction continuously shifts over a horizontal angle θ. It is feasible with this assumption to average the radionuclide concentration across the horizontal sector. This is performed by integrating the Gaussian

distribution from negative to positive infinity in the "y" direction and averaging over the sector width with "R" as the radial equivalent of "x". The mathematical manipulation yields the meandering plume expression [2,5]:

$$\chi\,(R,\theta,z) \;=\; \frac{Q'G(z)}{\sqrt{2\pi}\,\sigma_z\,U R\theta}$$

The dose rate to a receptor on the ground was calculated by assuming a "ring" shaped differential volume dV of the plume centrally located a vertical distance "z" above the receptor as shown in Figure 1.

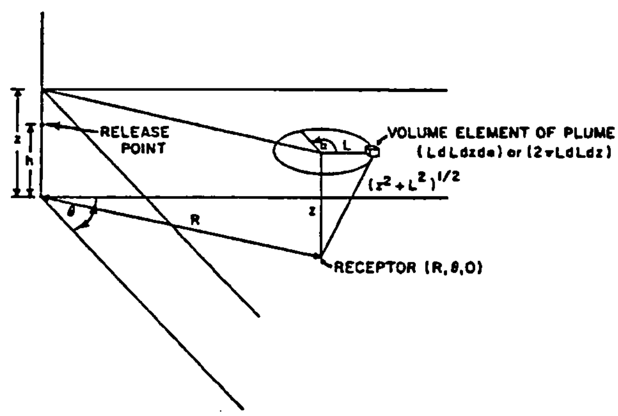


Figure 1 (TID-24190)

The average concentration in the ring will be approximately equal to (R,θ,z) which is the center of the ring. The gamma air dose rate to a ground receptor from this ring due to radioactive emissions from dV is [2]:

$$dD'_{\gamma} = \frac{0.4044}{r^2}\;\mu_a\,\chi(R,\theta,z)\;E_{\gamma}(1+k\mu r)\;e^{(-\mu r)}\,dV$$

where

$$k = \frac{\mu - \mu_a}{\mu_a}$$

$$r = \sqrt{L^2 + z^2}$$

$$dV = 2\pi L\,dl\,dz$$

The literature has shown that the integrated dose is accurately obtained for $\mu R\theta > 2$ by assuming that the plume consists of a series of infinite horizontal planes. The previous differential expression is integrated and appropriate substitutions are made to arrive at the integrated dose in air from the plume [1,2]:

$$D'_\gamma = \frac{260\,\mu_a Q' E_\gamma}{U R \theta}(I_1 + k I_2)$$

where

$$I_1 = \frac{1}{2\sqrt{2}\,\sigma_z}\int_0^\infty G(z)\,E_1(\mu z)\,dz$$

$$E_1 = \int_{\mu z}^\infty \frac{e^{-\mu r}}{\mu r}\,d(\mu r)$$

$$I_2 = \frac{1}{2\sqrt{2}\,\sigma_z}\int_0^\infty G(z)\,e^{-\mu z}\,dz$$

The integrals I_1 and I_2 are functions of plume height, linear attenuation coefficient, and vertical standard deviation. As a result, the integrals are also dependent on wind-related plume-rise effects, gamma energy, and atmospheric stability. The values of I_1 and I_2 were computed numerically utilizing the Newton-Cotes 9-point formula [2,3]. The I_1 and I_2 functions have the following limiting values [1,2]:

(*a*) *for* $\sigma_z \to 0$ $\quad I_1 = \frac{\sqrt{\pi}}{2}E_1(\mu h) \quad (h>0)$

$$I_2 = \frac{\sqrt{\pi}}{2}e^{-\mu h}$$

(*b*) *for* σ_z *large*, $\sigma_z \geq h$; $\sigma_z > (\frac{1}{\mu})$

$$I_1 = \frac{1}{\sqrt{2}\,\mu\,\sigma_z}\,e^{-\frac{h^2}{2\sigma_z^2}}$$

$$I_1 = I_2$$

The limiting value (b) reduces the integrated dose expression to the semi-infinite cloud model with sector averaging. It is important to consider these limiting conditions for the PULSTAR facility analysis for H > 0 because:

(1) The I_1 and I_2 functions over estimate the gamma dose for $\mu R\theta < 2$

(2) σ_z numerically approaches the release height (H) considerably close to the release point for certain atmospheric stability classes.

(3) The receptor location may be very close to the stack resulting in relatively small dispersion coefficients.

The PULSTAR offsite dose analysis requires the special case for effluents released from points less than the height of adjacent solid structures. The PULSTAR facility has neighboring structures (e.g., D.H. Hill library and Poe Hall) which are higher than the facility stack. Dose assessments for receptors in these structures should assume a ground level release ($H_{eff} = 0$) [4].

The computer model includes the limiting cases of the integral solution and a special case for elevated receptors:

(1) $\mu R\theta > 2$
(2) $\sigma_z \rightarrow 0$ and $H > 0$
(3) $\sigma_z \geq H$ and $\sigma_z \gg (1/\mu)$
(4) $h_e = 0$

The model will yield quite accurate values for case (1). However, for $\mu R\theta < 2$ the model over estimates the gamma dose. The limiting conditions (2) and (3) are addressed in the model. Condition (3) is of particular interest to the PULSTAR facility due to the relatively low stack height. The fourth case addresses receptors at an elevation equal to the PULSTAR stack height. Incidently, these doses may be directly obtained from the nomograms in NUREG-0851 for values of R > 250 meters [5].

2.2 ^{41}Ar Source Term Concentration

The questions and answers supporting the relicensing of the PULSTAR have reported an annual average release rate for ^{41}Ar of 6.783 Ci/y with both building exhaust fans operating [SAR 10.3.2] as follows:

PULSTAR Fan:	10,050 ft^3/m
Old Wing Fan:	12,500 ft^3/m
Total Stack Flow:	22,550 ft^3/m

2.3 Fuel Pin Annuli Fission Gas (Noble) Inventory

The fission gas inventory collected in the annuli of the fuel rod was previously calculated in response to a Commission question. TID-14844, **Calculation of Distance Factors for Power and Test Reactor Sites**, was utilized to reach the three fuel pin noble gas inventory:

Isotope	Activity (Ci)
Kr-88	$3.78(10^{-3})$
Xe-133	$5.42(10^{-2})$
Xe-138	$1.94(10^{-3})$

2.4 Mass and Energy Attenuation Coefficients

The values for mass (μ) and energy (μ_a) attenuation coefficients were obtained from Reference 4. The values were numerically fitted and coded as gamma energy dependent functions. Tables 1 and 2 list the graphical and fitted values.

TABLE 1
MASS ATTENUATION COEFFICIENT, μ, IN AIR

Energy (MeV)	μ (1/M) (Graph)	μ (1/M) (Function)	Difference %
0.01	0.6600	0.6595	-0.1
0.02	0.1000	0.1042	4.2
0.03	0.0460	0.0433	-5.9
0.04	0.0320	0.0294	-8.1
0.05	0.0270	0.0248	-8.1
0.06	0.0240	0.0229	-4.6
0.07	0.0230	0.0220	-4.3
0.08	0.0220	0.0212	-3.6
0.09	0.0210	0.0206	-1.9
0.10	0.0200	0.0202	1.0
0.15	0.0175	0.0182	4.0
0.20	0.0160	0.0166	3.8
0.30	0.0140	0.0143	2.1
0.40	0.0125	0.0127	1.6
0.50	0.0115	0.0115	0.0
0.60	0.0105	0.0106	1.0
0.70	0.0098	0.0099	1.0
0.80	0.0090	0.0092	2.2
0.90	0.0086	0.0087	1.1
1.00	0.0080	0.0083	3.8
2.00	0.0058	0.0058	0.0
3.00	0.0046	0.0047	2.2
4.00	0.0040	0.0040	0.0
5.00	0.0035	0.0035	0.0
6.00	0.0032	0.0032	0.0
7.00	0.0030	0.0029	-3.3
8.00	0.0028	0.0027	-3.6
9.00	0.0027	0.0025	-7.4
10.00	0.0026	0.0024	-7.7

TABLE 2
ENERGY ATTENUATION COEFFICIENT, μ_A, IN AIR

Energy (MeV)	μ (1/M) (Graph)	μ (1/M) (Function)	Difference %
0.01	0.6000	0.5999	0.0
0.02	0.0660	0.0673	2.0
0.03	0.0195	0.0185	-5.3
0.04	0.0088	0.0080	-9.1
0.05	0.0052	0.0048	-7.7
0.06	0.0040	0.0037	-7.5
0.07	0.0034	0.0033	-2.9
0.08	0.0032	0.0031	-3.1
0.09	0.0031	0.0031	0.0
0.10	0.0030	0.0031	3.3
0.15	0.0032	0.0034	6.3
0.20	0.0035	0.0036	2.9
0.30	0.0037	0.0037	0.0
0.40	0.0039	0.0038	-2.6
0.50	0.0039	0.0038	-2.6
0.60	0.0038	0.0038	0.0
0.70	0.0038	0.0037	-2.6
0.80	0.0037	0.0037	0.0
0.90	0.0036	0.0036	0.0
1.00	0.0035	0.0036	2.8
2.00	0.0030	0.0031	3.3
3.00	0.0027	0.0027	0.0
4.00	0.0024	0.0024	0.0
5.00	0.0023	0.0022	-4.3
6.00	0.0021	0.0021	0.0
7.00	0.0021	0.0020	-4.8
8.00	0.0020	0.0020	0.0
9.00	0.0019	0.0019	0.0
10.00	0.0019	0.0019	0.0

2.5 Plume Lateral and Vertical Standard Deviations

The vertical (σ_z) and lateral (σ_y) plume standard deviations were obtained from Reference 1 with the exception of the Pasquill G stability class. The values were numerically fitted and coded as a distance and atmospheric stability dependent function. The Pasquill stability class G σ_z and σ_y values were calculated from the class F values as follows [6]:

$$\sigma_z(G) = 2/3\ \sigma_z(F)$$
$$\sigma_y(G) = 3/5\ \sigma_y(F)$$

The Pasquill A - G atmospheric stability methodology was applied in this analysis. The stability classes are categorized as follows:

A - Extremely Unstable
B - Moderately Unstable
C - Slightly Unstable
D - Neutral
E - Slightly Stable
F - Moderately Stable
G - Extremely Stable

Tables 3 through 8 list the graphical and numerical values for atmospheric stability classes A through F. Table 9 lists the numerical values for atmospheric stability class G.

TABLE 3
AXIAL STANDARD DEVIATION, σ_Z
PASQUILL CLASS "A"

Distance (M)	σ_Z (M) (Graph)	σ_Z (M) (Function)	Difference %
100	16	15.5	-3.12
200	31	33.3	7.42
300	55	56.4	2.55
500	120	124.9	4.08
700	240	239.5	-0.21
1000	550	550.0	3.91
1300	1000	987.0	-1.30

TABLE 4
AXIAL STANDARD DEVIATION, σ_Z
PASQUILL CLASS "B"

Distance (M)	σ_Z (M) (Graph)	σ_Z (M) (Function)	Difference %
100	11	11.21	1.91
200	20	20.93	4.65
300	32	30.85	-3.59
400	42	41.30	-1.67
500	53	52.50	-0.94
1000	120	122.14	1.78
2000	360	358.00	-0.56
3000	770	771.70	0.22
3400	1000	999.20	-0.08

TABLE 5
AXIAL STANDARD DEVIATION, σ_Z
PASQUILL CLASS "C"

Distance (M)	σ_Z (M) (Graph)	σ_Z (M) (Function)	Difference %
100	7.6	7.8	2.63
200	15.0	15.2	1.33
300	22.0	22.2	0.91
400	30.0	29.0	-3.33
500	36.0	35.6	-1.11
1000	68.0	65.9	-3.09
2000	120.0	119.4	-0.50
3000	170.0	167.0	-1.77
10,000	420.0	428.0	1.91
20,000	710.0	703.9	-0.86
40,000	1000.0	1101.2	0.11

TABLE 6
AXIAL STANDARD DEVIATION, σ_Z
PASQUILL CLASS "D"

Distance (M)	σ_Z (M) (Graph)	σ_Z (M) (Function)	Difference %
100	5	5.5	10.0
200	9.5	9.4	-1.05
300	13	12.8	-1.92
400	17	15.8	-7.06
500	19	18.6	-2.10
900	30	28.1	-6.33
2000	50	48.2	-3.6
3000	58	62.6	7.93
10,000	130	131.2	0.92
20,000	200	196.5	-1.75
40,000	290	288.9	-0.38
100,000	435	435.1	0.02

TABLE 7
AXIAL STANDARD DEVIATION, σ_Z
PASQUILL CLASS "E"

Distance (M)	σ_Z (M) (Graph)	σ_Z (M) (Function)	Difference %
100	3.0	3.7	1.39
200	6.3	6.6	4.76
300	9.0	9.2	2.22
400	12	11.5	-4.17
500	14	13.6	-2.86
1000	22	22.3	1.36
2000	37	35.0	-5.41
3000	46	44.3	-3.70
5000	60	58.9	-1.83
10,000	84	82.3	-2.02
50,000	150	155.5	3.67
100,000	190	187.3	-1.42

TABLE 8
AXIAL STANDARD DEVIATION, σ_Z
PASQUILL CLASS "F"

Distance (M)	σ_Z (M) (Graph)	σ_Z (M) (Function)	Difference %
10	2.3	2.5	9.57
200	4.0	4.2	5.00
500	8.0	8.3	3.75
1000	14.0	13.5	-3.57
2000	22.0	21.0	-4.55
4000	32.0	31.0	-3.13
6000	39.0	38.0	-2.56
10,000	46.0	48.0	4.35
20,000	58.0	62.8	8.27
60,000	80.0	83.3	4.13
90,000	90.0	86.7	3.67

TABLE 9
AXIAL STANDARD DEVIATION, σ_Z
PASQUILL CLASS "G"

Distance (M)	σ_Z (M) (Function)
100	1.68
200	2.77
300	3.78
500	5.55
900	8.41
4000	20.67
10,000	32.01
40,000	40.00

TABLE 10
LATERAL STANDARD DEVIATION, σ_y
PASQUILL CLASS "A"

Distance (M)	σ_y (M) (Graph)	σ_y (M) (Function)	Difference %
100	22	24.6	-11.82
200	48	49.9	-3.96
300	65	64.1	1.38
500	100	100.0	0
1000	200	183.0	8.50
1300	220	230.3	4.68

TABLE 11
LATERAL STANDARD DEVIATION, σ_y
PASQUILL CLASS "B"

Distance (M)	σ_y (M) (Graph)	σ_y (M) (Function)	Difference %
100	18	20.6	14.44
200	33	37.6	-13.94
300	47	53.8	-14.47
500	82	84.0	-2.44
1000	150	154.0	-2.67
2000	290	281.6	2.90
3000	400	401.3	-0.33
3400	450	447.4	0.58

TABLE 12
LATERAL STANDARD DEVIATION, σ_y
PASQUILL CLASS "C"

Distance (M)	σ_y (M) (Graph)	σ_y (M) (Function)	Difference %
100	14	14.7	-5.00
500	54	60.0	--11.11
1000	105	109.9	-4.67
2000	200	201.1	-0.55
3000	290	286.5	1.21
10,000	840	819.3	2.46
20,000	1600	1500.0	6.25

TABLE 13
LATERAL STANDARD DEVIATION, σ_y
PASQUILL CLASS "D"

Distance (M)	σ_y (M) (Graph)	σ_y (M) (Function)	Difference %
100	8.4	9.0	-7.14
500	38.0	40.0	-5.26
1000	72.0	71.4	0.83
2000	140.0	130.7	6.64
3000	180.0	186.2	-3.44
10,000	540.0	532.5	1.39
20,000	990.0	975.0	1.52

TABLE 14
LATERAL STANDARD DEVIATION, σ_y
PASQUILL CLASS "E"

Distance (M)	σ_y (M) (Graph)	σ_y (M) (Function)	Difference %
100	6.4	7.4	-15.63
500	28.0	30.0	-7.14
1000	56.0	54.9	1.96
2000	110.0	100.6	8.55
3000	150.0	143.3	4.47
10,000	420.0	409.6	2.48

TABLE 15
LATERAL STANDARD DEVIATION, σ_y
PASQUILL CLASS "F"

Distance (M)	σ_y (M) (Graph)	σ_y (M) (Function)	Difference %
100	4.4	4.9	-111.36
500	20.0	20.0	0.00
1000	38.0	36.6	3.68
2000	70.0	67.1	4.14
4000	130.0	122.8	5.54
6000	170.0	174.9	-2.88
10,000	280.0	273.1	2.46
20,000	490.0	500.0	-2.04

TABLE 16
LATERAL STANDARD DEVIATION, σ_y
PASQUILL CLASS "G"

Distance (M)	σ_y (M) (Function)
100	3.28
200	5.99
300	8.54
500	13.33
1000	24.41
10,000	185.1

2.6 Effective Height Model for Stack

The effluent exiting the stack carries momentum and buoyancy which potentially propels the effluent vertically beyond the stack height. This leads to an effective release height which may be larger than the physical stack height. Two models have been chosen from the literature for study.

The first model considers the atmospheric stability class, effluent discharge velocity and thermal energy, and the horizontal windspeed [4,7]:

$$h_{eff} = h_{fixed} + h_{pr} - h_t - c$$

$$h_{pr} = \frac{\alpha V d + \beta \sqrt{Q_h}}{u}$$

$$c = 3d\left(1.5 - \frac{V}{U}\right)$$

The effective stack height (h_{eff}) is obtained from the fixed height (h_{fixed}) by the correction factors h_{pr}, h_t and **c**. The factor h_{pr} is the distance the plume rises above the release point as a result of momentum and buoyancy effects, h_t corrects for terrain differences near the stack, and **c** corrects for down-wash caused by turbulent eddies in the stack wake [7]. The values of α and β are functions of atmospheric conditions and have been determined empirically. Q_h is the thermal energy discharge in watts compared with ambient. The air temperature difference between the PULSTAR bay and ambient is approximately between 10°C (summer) and 45°C (winter) resulting in Q_h values of 6.11E+04 and 1.94E+05 J/s respectively.

The model may not be suitable for research reactor facilities since gaseous discharge rates of radionuclides are usually small and close to the ambient temperature. These limitations allow for h_{pr} to be ignored.

The second model studied is much simpler since it only considers the effluent exit velocity and horizontal windspeed [8]:

$$\Delta H = D\left(\frac{V}{U}\right)^{1.4}$$

This model is applicable for small volume release rates (< 50 m^3/s), having significant effluent exit velocity (> 10 m/s), and small temperature difference (< 50°C above ambient). PULSTAR facility gaseous effluent exit velocity with normal ventilation of 4.72 m^3/s is approximately 24 m/s. The effluent temperature difference is typically less than 50°C.

The second model is better suited for the PULSTAR facility analysis. However, both models will be included in the computer code for all possible conditions. Table 10 shows the effective release heights for the PULSTAR facility stack utilizing both the models discussed. The effective height model is activated through the input section of the computer code.

TABLE 17
EFFECTIVE RELEASE HEIGHT
(with ΔT = 28°C)

H_O = 30 M

Wind Speed (m/s)	**H_{eff} (m) [7]** Stable	Neutral	Unstable	**H_{eff} [8] (m)**
0.5	64.1	108.3	249.1	143.2
1.0	47.8	69.2	139.6	72.9
2.0	38.5	49.6	84.8	46.3
3.0	35.7	43.1	66.5	39.2
4.0	34.3	39.8	57.4	36.2
5.0	33.4	37.8	51.9	34.5
6.0	32.8	36.5	48.3	33.5
7.0	32.4	35.6	45.7	32.8
8.0	32.1	34.9	43.7	32.3
9.0	31.9	34.4	42.2	32.0
10.0	31.7	33.9	41.0	31.7

2.7 Dosimetry Models

The ground-level gamma air dose rates (D_γ') were calculated from the previously introduced expression [1,2]:

$$D_\gamma' = \frac{260\,\mu_a Q' E_\gamma}{U R \theta}\left(I_1 + k I_2\right)$$

The expression calculates the gamma air dose rate for one single gamma emitting radioisotope. Multiple isotopes or gamma decay mechanisms may be evaluated by summing the individual contributions.

The gamma whole-body radiation doses (D_γ^T) from an elevated radioactive cloud consisting of "k" gamma emitters were obtained from the following expression [8]:

$$D^T(R,\theta) = 1.11 \times S_F \times \sum_k D_k^{\gamma}(R,\theta)\, e^{-\mu_a^T(E_k)t_d}$$

S_F is the attenuation factor which reduces the dose to the receptor due to buildings or structures. This analysis does not take credit for the structures around the PULSTAR facility and S_F equals unity. $\mu_a^T(E_k)$ is the tissue energy absorption coefficient in cm^2/gm and t_d is the product of tissue density and depth (5 cm) in gm/cm^2. The factor 1.11 is the average ratio of tissue to air energy absorption coefficients [3].

Beta doses depend mainly on the concentration of beta-emitting radioactive materials in the immediate vicinity of a receptor. The beta dose from a passing cloud may be approximated using the infinite cloud model. It must be assumed that the beta emitting material is uniformly distributed in the cloud. The concentration of the radioactive constituent depends on the size of the cloud, distance from the release point, and windspeed. The ground-level beta skin dose consists of two components, the gamma, and the beta contributions. The skin dose rates (D_β) were calculated from the following expression [3]:

$$D_\beta = 1.11 \times S_F \times D_\gamma + 3.17(10^4)\left(\frac{\chi}{Q}\right) \times DFS$$

The dose factors for exposure to a semi-infinite cloud of noble gases (DFS) were obtained from the literature.

2.8 Point and Line Source Limiting Models

The numerical calculation described above will be conservative for certain conditions. It became necessary to implement certain checks for limiting the conservatism of the integrated cloud results. The computer model employs two simple and classic point and line source approximations to the offsite dose calculation.

A concentrated spatial point containing the total activity released will lead to an unattenuated photon fluence rate given by the following expression:

$$\phi_\gamma(r) = \frac{S}{4\pi r^2}$$

S is the source strength in disintegrations per second and **r** is the distance to the receptor. Similarly, the photon fluence at a point **r** normal to a line source of length **L** containing a total activity **S** is given by:

$$\phi_\gamma(r) = \frac{S_L}{4\pi r}\left[TAN^{-1}\left(\frac{l_1}{r}\right) + TAN^{-1}\left(\frac{l_2}{r}\right)\right]$$

The lengths l_1 and l_2 are the segments of the line perpendicular to the vector **r**. S_l is the activity per unit length.

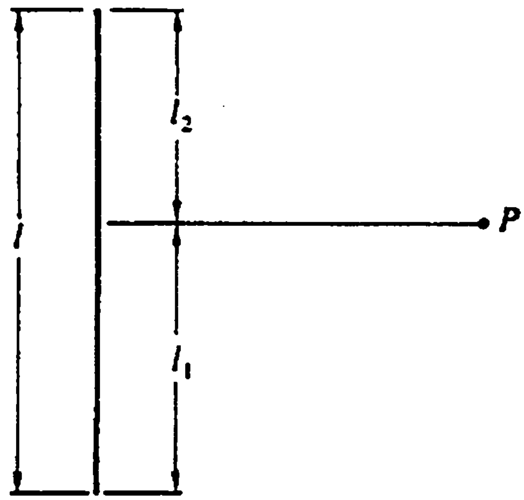


Figure 2

The exposure rate in air due to a fluence rate of photons at a point **r** is obtained from the following expression:

$$X' = \phi_\gamma(r)\, E \left(\frac{\mu_{en}}{\rho} \right) \frac{e}{W}$$

E represents the photon energy (J), e the electron charge (C), μ_{en}/ρ the mass energy absorption coefficient (m^2/kg), and W is the mean energy expended in air per ion pair formed (J). In this form, the exposure rate will be given in C/kg/s and may be converted to R/s by the conversion factor 1 R = 2.58E-04 C/kg.

These simple unattenuated air dose models were incorporated in this analysis to obtain ultra conservative limits of exposure rates; that is, the numerical calculation was limited to these point and line source results.

3.0 RESULTS

The results for the dose rate to a receptor in the unrestricted area from routine releases of Ar-41 and due to the noble gas inventory released in a hypothetical rupture of three fuel pins are included in SAR Sections 10 and 13, respectively. The results included in this appendix is limited to the benchmark effort for the numerical model.

Computer Model Validation and Verification

The computer program validation and verification consisted of benchmarking the model against selected reference literature. The integral solutions I_1, I_2, and I_T were compared to Figures 7.21, 7.22, and 7.23 in TID-24190. These results are presented in Tables 18, 19, and 20. The whole-body doses at 250 and 500 meters for an elevated release of 100 meters were compared to the Ar-41 meandering plume monogram in NUREG-0851. Tables 21 through 22 document this comparison.

The benchmarking effort demonstrated excellent agreement between the calculated and the reference values.

TABLE 18
THE I_1 AND I_2 INTEGRAL FUNCTIONS

$\mu\sigma_Z$	μh	I_1 (Fig. 7-21)	I_1 (Calculated)	I_2 (Fig. 7-22)	I_2 (Calculated)
H = 30 R = 100 μ = 0.00727861					
0.11	0.22	1.20	1.1304	~0.70	0.7052
0.08	0.22	1.10	1.0510	~0.70	0.7043
0.06	0.22	1.05	1.0150	~0.70	0.7034
0.04	0.22	1.05	0.9929	~0.70	0.7026
H = 30 R = 150 μ = 0.00727861					
0.17	0.22	1.20	1.2043	~0.70	0.7000
0.12	0.22	1.20	1.1351	~0.70	0.7050
H = 30 R = 200 μ = 0.00727861					
0.24	0.22	1.60	**1.8539**	0.9	**1.8539**
0.07	0.22	1.00	0.9575	0.7	0.6895
H = 30 R = 250 μ = 0.00727861					
0.32	0.52	0.67	0.6545	0.55	0.5362
0.15	0.52	0.55	0.5012	0.53	0.5261
H = 73 R = 500 μ = 0.00727861					
0.91	0.52	0.62	**0.6561**	0.50	**0.6561**
0.38	0.52	0.70	0.6821	0.55	0.5340
H = 73 R = 100 μ= 0.00727861					
0.11	0.53	0.50	0.4834	0.55	0.5236
0.08	0.53	0.50	0.4736	0.55	0.5220
0.06	0.53	0.50	0.4694	0.55	0.5213
0.04	0.53	0.50	0.4661	0.55	0.5207

TABLE 18 (cont.)
THE I_1 AND I_2 INTEGRAL FUNCTIONS

$\mu\sigma_Z$	μh	I_1 (Fig. 7-21)	I_1 (Calculated)	I_2 (Fig. 7-22)	I_2 (Calculated)
H = 143 R = 250 μ = 0.00727861					
0.320	1.04	0.23	0.2217	0.33	0.3290
0.190	1.04	0.20	0.1927	0.33	0.3181
0.081	1.04	0.18	0.1835	0.33	0.3136
0.058	1.04	0.18	0.1825	0.32	0.3131
H = 143 R = 500 μ = 0.00727861					
0.910	1.04	0.40	0.3937	0.35	0.3562
H = 143 R = 1000 μ = 0.00727861					
4.16	1.04	0.170	**0.1648**	0.1700	**0.1648**
H = 687 R = 250 μ = 0.00727861					
0.32	5.0	0.0010	0.0011	0.0065	0.0063
H = 687 R = 500 μ = 0.00727861					
0.90	5.0	0.0020	0.0019	0.0100	0.0090
H = 687 R = 1000 μ = 0.00727861					
4.16	5.0	0.0800	0.08296	0.0820	0.08376
H = 30 R = 50 μ = 0.00727861					
0.06	0.24	1.00	0.9853	0.650	0.6977

TABLE 19
THE I_T INTEGRAL FUNCTION ($E\gamma$ = 0.7 MeV)
H = 100 M

σ_z (M) (Fig. 7-23)	I_T (Fig. 7-23)	σ_z (M) (Calculated)	I_T (Calculated)
5	0.70	5.5	0.74
10	0.70	9.4	0.75
100	1.00	99.0	1.01
500	0.40	428.0	0.40

TABLE 20
THE I_T INTEGRAL FUNCTION ($E\gamma$ = 0.7 MeV)
H = 50 M

σ_z (M) (Fig. 7-23)	I_T (Fig. 7-23)	σ_z (M) (Calculated)	I_T (Calculated)
5	1.50	5.5	1.40
10	1.50	9.4	1.40
100	1.60	99.0	1.70
500	0.40	428.0	0.44

TABLE 21
WHOLE-BODY DOSE (mRem/Ci)
H =100, R= 250

WINDSPEED (M/S)	ATMOSPHERIC CONDITION	D (Calculated)	D (NUREG-0851)
0.5	A	0.0172	0.0190
0.5	B	0.0155	0.0170
1.0	A	0.0088	0.0090
1.0	B	0.0079	0.0080
3.0	A	0.0030	0.0035
3.0	B	0.0027	0.0030
3.0	C	0.0026	0.0030
3.0	E	0.0026	0.0030
3.0	F	0.0026	0.0030
6.0	C	0.0013	0.0014
6.0	D	0.0013	0.0014
6.0	F	0.0013	0.0014
9.0	C	0.0009	0.0009
9.0	D	0.0009	0.0009
10.0	F	0.0009	0.0009

TABLE 22
WHOLE-BODY DOSE (mRem/Ci)
H = 100, R= 500

WINDSPEED (M/S)	ATMOSPHERIC CONDITION	D (Calculated)	D (NUREG-0851)
0.5	A	0.0106	0.0150
0.5	B	0.0086	0.0150
1.0	A	0.0056	0.0065
1.0	B	0.0045	0.0065
3.0	A	0.0019	0.0015
3.0	B	0.0016	0.0015
3.0	C	0.0014	0.0010
3.0	E	0.0013	0.0010
3.0	F	0.0013	0.0010
6.0	C	0.0007	0.0006
6.0	D	0.0007	0.0006
6.0	F	0.0007	0.0006
9.0	C	0.0007	0.0006
9.0	D	0.0005	0.0004
10.0	F	0.0004	0.0004

4.0 CONCLUSIONS

The computer program has been verified to yield acceptable results. SAR sections 10 and 13 apply the program to calculate the dose in the unrestricted area in the proximity of the PULSTAR reactor for routine operation and postulated fuel failure conditions.

5.0 REFERENCES

1. TID-24190, **Meteorology and Atomic Energy**, D.H. Slade, Editor, July 1968.

2. YAEC-1105, "A Method for Computing the Gamma-Dose Integrals I_1 and I_2 for the Finite Cloud Sector Averaged Model", J.N. Hamawi, 1976.

3. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I", Rev. 1, October 1977.

4. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", Revision 1, July 1977.

5. NUREG-0851, "Nomograms for Evaluation of Doses from Finite Noble Gas Clouds", July 1983.

6. NUREG/CR-3332, **Radiological Assessment: A Textbook on Environmental Dose Analysis**, September 1983.

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8. ANSI/ANS 15.7, "Research Reactor Site Evaluation", 1977.

UPDATED SAFETY ANALYSIS REPORT

APPENDIX A

TECHNICAL SPECIFICATIONS FOR THE NORTH CAROLINA STATE UNIVERSITY PULSTAR REACTOR

FACILITY LICENSE NO. R-120
DOCKET NO. 50-297

ORIGINAL ISSUE DATE: August 25, 1972

AMENDMENT 11 - September 4, 1995

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LIST OF FIGURES

Included in this document are the Technical Specifications and the "Bases" for the Technical Specifications. The bases provide the technical support for the individual technical specification and are included for information purposes only. The bases are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.0 DEFINITIONS

The terms Safety Limit (SL), Limiting Safety System Setting (LSSS), and Limiting Condition of Operation (LCO) are as defined in Paragraph 50.36 of 10 CFR Part 50.

1.1 Channel: A channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

1.2 Channel Calibration: A channel calibration is an adjustment of a channel, such that its output responds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trips and shall be deemed to include a Channel Test.

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

1.4 Channel Test: A channel test is the introduction of a known signal into a channel to verify that it is operable.

1.5 Cold Critical: The condition of the reactor when it is critical, with negligible xenon, and the fuel and bulk water are both at an isothermal temperature of 70°F.

1.6 Confinement: Confinement means a closure on the overall facility which controls the movement of air into and out of the facility through a controlled path.

1.7 Control Rod: A control rod is a neutron absorbing blade having an in-line drive which is magnetically coupled and has SCRAM capability.

1.8 Excess Reactivity: Excess reactivity is that amount of reactivity that would exist if all Control Rods (and Shim rod) were fully withdrawn from the point where the reactor is critical ($k_{eff}=1$).

1.9 Experiment: Any operation, hardware, or target (excluding devices such as detectors) which is designed to investigate reactor characteristics or which is intended for irradiation within the pool, on or in a beam tube or irradiation facility and which is not rigidly secured to a core or shield structure so as to be a part of their design. Specific categories of experiments include:

a. Tried Experiments: Tried experiments are those which have been previously performed in this reactor. Specifically, a tried experiment has a similar size, shape, composition and location of an experiment previously approved and performed in the reactor.

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

c. Non-Secured Experiment: A non-secured experiment is an experiment that does not meet the criteria for being a "Secured" experiment.

d. Movable Experiment: A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

e. Fueled Experiment: A fueled experiment is an experiment which contains fissionable material.

1.10 Experimental Facilities: Experimental facilities are facilities used to perform experiments. They include beam tubes, thermal columns, void tanks, pneumatic transfer systems, in-core facilities at single-assembly positions, out-of-core irradiation facilities, and the bulk irradiation facility.

1.11 Measured Value: The measured value is the value of a parameter as it appears on the output of a channel.

1.12 Operable: Operable means a component or system is capable of performing its intended function.

1.13 Operating: Operating means a component is performing its intended function.

1.14 pcm: A unit of reactivity that is the abbreviation for "percent millirho" and is equal to 10^{-5} $\Delta k/k$ reactivity. For example, 1000 pcm equals 1.0% $\Delta k/k$.

1.15 Reactor Building: The Reactor Building includes the Reactor Bay, Control Room, the Mechanical Equipment Room (MER), and the Primary Piping Vault (PPV). The Nuclear Regulatory Commission R-120 license applies to the areas in the Reactor Building and the Waste Tank Vault.

1.16 Reactor Operation: Reactor operation is any condition when the reactor is not secured or shutdown.

1.17 Reactor Operator: A reactor operator (RO) is an individual who is certified to manipulate the controls of a reactor.

1.18 Reactor Operator Assistant (ROA): An individual who has been certified by successful completion of an in-house training program to assist the licensed reactor operator during reactor operation.

1.19 Reactor Safety System: Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.20 Reactor Secured: The reactor is secured when:

a. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection, or

b. The following conditions exist:

i. All scrammable neutron absorbing control rods are fully inserted, and

ii. The Reactor Keyswitch is in the OFF position and the key is removed from the lock, and

iii. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and

iv. No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding one dollar {0.73% $\Delta k/k$ (730 pcm)}.

1.21 Reactor Shutdown: That subcritical condition of the reactor where the absolute value of the negative reactivity of the core is equal to or greater than the shutdown margin.

1.22 Reportable Event: A Reportable Event is any of the following:

a. Violation of a Safety Limit.

b. Release of radioactivity from the site above allowed limits.

c. Any of the following:

i. Operation with actual Safety System Settings (SSS) for required systems less conservative than the Limiting Safety System Settings (LSSS) specified in these specifications.

ii. Operation in violation of Limiting Conditions for Operation (LCO) established in these Technical Specifications.

iii. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdown. (For components or systems other than those required by these Technical Specifications, the failure of the extra component or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function).

iv. An unanticipated or uncontrolled change in reactivity greater than one dollar {0.73% $\Delta k/k$(730 pcm)}. Reactor trips resulting from a known cause are excluded.

v. Abnormal or significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks), which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

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

1.23 Shim rod: A shim rod is a neutron absorbing rod having an in-line drive which is mechanically, rather than magnetically, coupled and does not have a SCRAM capability.

1.24 Senior Reactor Operator: A senior reactor operator is an (SRO) individual who is certified to direct the activities of reactor operators.

1.25 Shutdown Margin: Shutdown margin means the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition with the most reactive scrammable rod fully withdrawn, the non-scrammable rod (Shim rod) fully withdrawn, and with experiments considered at their most reactive condition, and finally, that the reactor will remain subcritical without further operator action.

1.26 True Value: The true value is the actual value of a parameter.

1.27 Unscheduled Shutdown: An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operations. This does not include shutdowns which occur during testing or check-out operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

2.1.1 Safety Limits (SL) for Forced Convection Flow.

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance with forced convection flow. These interrelated variables are:

P	=	Reactor Thermal Power
W	=	Reactor Coolant Flow Rate
T_{inlet}	=	Reactor Coolant Inlet Temperature
H	=	Height of water above the top of the core

Objective

The objective is to assure that the integrity of the fuel clad is maintained.

Specification

Under the condition of forced convection flow, the Safety Limit shall be as follows:

a. The combination of true values of reactor thermal power (P) and reactor coolant flow rate (W) shall not exceed the limits shown in Figure 2.1-1 under any operating conditions. The limits are considered exceeded if the point defined by the true values of P and W is at any time outside the operating envelope shown in Figure 2.1-1.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

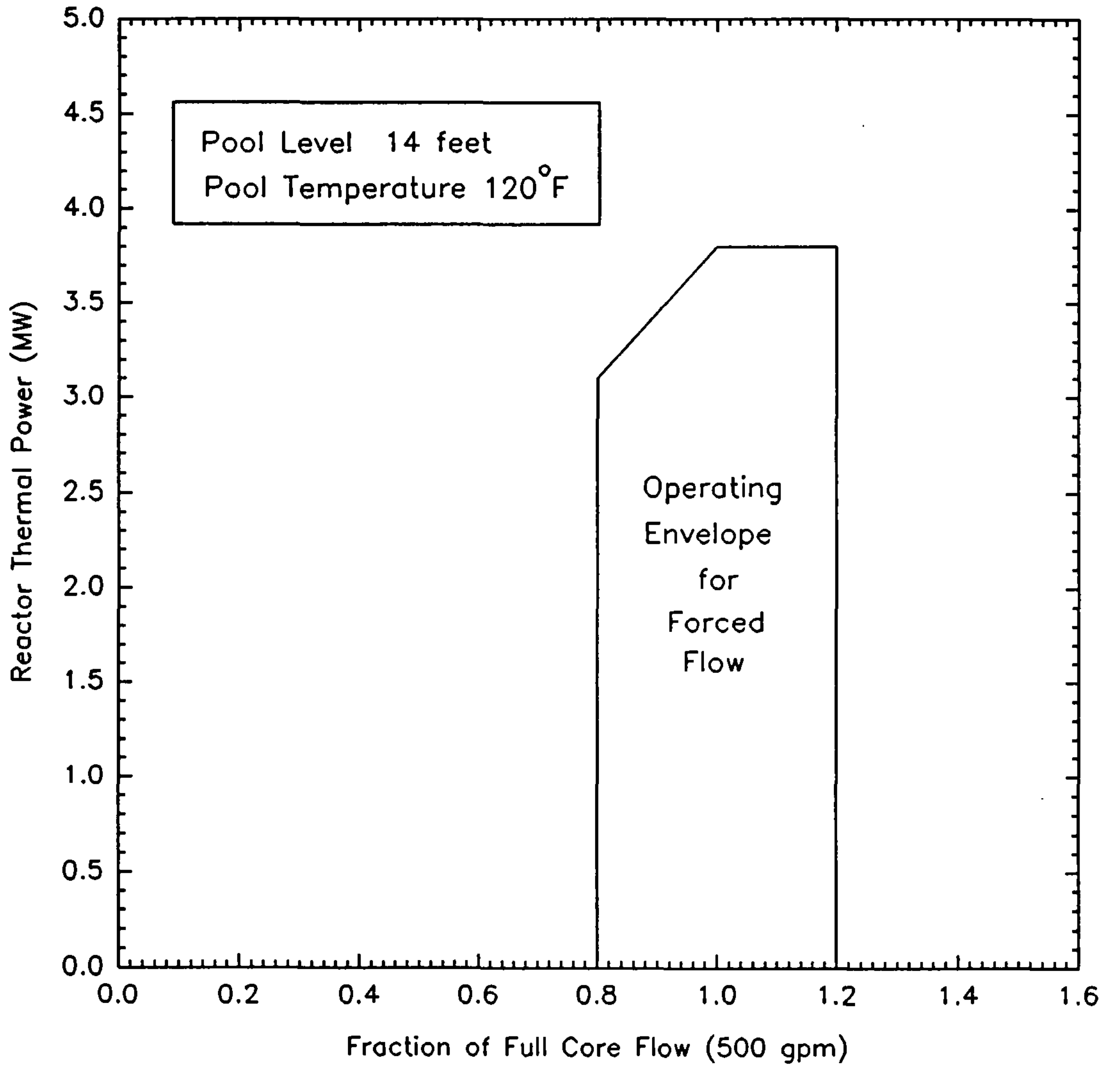
Bases

Above 80 percent of the full core flow of 500 gpm in the region of full power operation, the criterion used to establish the Safety Limit was no bulk boiling at

the outlet of any coolant channel. This was found to be far more limiting than the criterion of a minimum allowable burnout heat flux ratio of 2.0. The analysis is given in the SAR Appendix 3B.

In the region below 80 percent of full core flow, where, under a loss of flow transient at power the flow coasts down to zero, reverses, and then establishes natural convection, the criterion for selecting a Safety Limit is taken as a fuel cladding temperature. The analysis of a loss of flow transient is presented in Appendix 3B of the SAR. For initial conditions of full flow and an operating power of 1.4 MWt, the maximum clad temperature reached under the conservative assumptions of the analysis was 273°F which is well below the temperature at which fuel clad damage could possibly occur. The Safety Limit shown in Figure 2.1-1 for flow less than 80 percent of full flow is the steady state power corresponding to the maximum fuel clad temperature of 273°F with natural convection flow, namely, 1.4 MWt.

Figure 2.1-1


Pool Level 14 feet
Pool Temperature 120°F
Operating Envelope for Forced Flow
Reactor Thermal Power (MW)
5.0
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0.0
0.2
0.4
0.6
0.8
1.0
1.2
1.4
1.6
Fraction of Full Core Flow (500 gpm)

2.1.2 Safety Limit (SL) for Natural Convection Flow.

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance with natural convection flow. The interrelated variables are:

P	=	Reactor Thermal Power
T_{inlet}	=	Reactor Coolant Inlet Temperature
H	=	Height of water above the top of the core

Objective

The objective is to assure that the integrity of the fuel clad is maintained.

Specification

Under the condition of natural convection flow, the Safety Limit shall be as follows:

a. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

b. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

c. The true value of reactor thermal power (P) shall not exceed 1.4 MWt.

Bases

The criterion for establishing a Safety Limit with natural convection flow is established as the fuel clad temperature. This is consistent with Figure 2.1-1 for forced convection flow during a transient. The analysis of natural convection flow given in Appendix 3B and 3C of the SAR shows that at 1.4 MWt the maximum fuel clad temperature is 273°F which is well below the temperature at which fuel clad damage could occur. The flow with natural convection at this power is 98 gpm. This flow is based on data from natural convection tests with fuel assemblies of the same design, performed in the prototype PULSTAR Reactor, as referenced in Section 3 of the SAR.

2.2 Limiting Safety System Settings

2.2.1 Limiting Safety System Settings (LSSS) for Forced Convection Flow

Applicability

This specification applies to the setpoints for the safety channels monitoring reactor thermal power (P), coolant flow rate (W), height of water above the core (H), and pool water temperature (T).

Objective

The objective is to assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specification

Under the condition of forced convection flow, the Limiting Safety System Settings shall be as follows:

P	=	1.3 MWt (max.)
W	=	450 gpm (min.)
H	=	14 feet, 2 inches (min.)
T	=	117°F

Bases

The Limiting Safety System Settings that are given in the Specification 2.2.1 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded during the most limiting anticipated transient (loss of flow). The safety margin that is provided between the Limiting Safety System Settings and the Safety Limits also allows for the most adverse combination of instrument uncertainties associated with measuring the observable parameters. These instrument uncertainties include a flow variation of ten percent, a pool level variation of two inches and a power level variation of seven percent.

The analysis presented in Section 3 of the SAR of a loss of flow transient indicates that if the interrelated variables were at their LSSS, as specified in 2.2.1 above, at the initiation of the transient, the Safety Limits specified in 2.1.1 would not be exceeded.

2.2.2 Limiting Safety System Settings (LSSS) for Natural Convection Flow

Applicability

This specification applies to the setpoints for the safety channel monitoring reactor thermal power (P), the height of water above the core (H), and the pool water temperature (T).

Objective

The objective is to assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specifications

Under the condition of natural convection flow, the Limiting Safety System Settings shall be as follows:

P	=	250 kWt (max.)
H	=	14 feet, 2 inches (min.)
T	=	117°F

Bases

The Limiting Safety System Settings that are given in Specification 2.2.2 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded. The specifications given above assure that an adequate safety margin exists between the LSSS and the SL for natural convection. The safety margin on reactor thermal power was chosen with the additional consideration related to bulk boiling at the outlet of the hot channel. This criterion is not related to fuel clad damage (for these relatively low power levels) which was the criterion used in establishing the Safety Limits (see Specification 2.1.2). It is desirable to minimize to the greatest extent practical, ^{16}N dose at the pool surface which might be aided by steam bubble rise during upflow in natural convection. Analysis of coolant bulk boiling given in SAR, Section 3, indicates that the large safety margin on reactor thermal power assumed in Specification 2.2.2 above will satisfy this additional criterion of no bulk boiling in any channel.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Configuration

Applicability

This specification applies to the reactor core configuration during forced convection or natural convection flow operation.

Objective

The objective is to assure that the reactor will be operated within the bounds of established Safety Limits.

Specification

The reactor shall not be operated unless the following conditions exist:

a. A maximum of twenty-five fuel assemblies in a five by five array configuration.

b. A maximum of ten graphite reflectors located on the core periphery.

c. Unoccupied grid plate penetrations plugged.

d. A minimum of four control rod guides are in place.

Bases

This specification requires that the core be configured such that there is no bypass cooling flow around the fuel through the grid plate.

3.2 Reactivity

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods, shim rod and experiments.

Objective

The objective is to assure that the reactor can be shut down at all times and that the Safety Limits will not be exceeded.

Specifications

The reactor shall not be operated unless the following conditions exist:

a. The shutdown margin, with the highest worth scrammable control rod fully withdrawn, with the shim rod fully withdrawn, and with experiments at their most reactive condition, relative to the Cold Critical condition, is greater than 0.4% $\Delta k/k$ (400 pcm).

b. The excess reactivity is not greater than 4.07% $\Delta k/k$ (4070 pcm).

c. The drop time of each control rod is not greater than 1.0 second.

d. The rate of reactivity insertion of the control rods is not greater than 0.1% $\Delta k/k$ (100 pcm) per second (critical region only).

e. The absolute reactivity worth of experiments or their rate of reactivity change shall not exceed the values indicated in the following table:

Experiment	Limit
Movable	0.3% $\Delta k/k$ (300 pcm) or $\leq$0.1% $\Delta k/k/\text{sec}$ (100 pcm/sec), whichever is more limiting
Non-secured	1.0% $\Delta k/k$ (1000 pcm)
Secured	1.7% $\Delta k/k$ (1,700 pcm)

f. The total reactivity worth of all experiments shall not be greater than 3.0% $\Delta k/k$ (3000 pcm) sum of their absolute values.

Bases

a. The shutdown margin required by Specification 3.2a assures that the reactor can be shut down from any operating condition and will remain shutdown after cooldown and xenon decay, even if the highest worth scrammable rod should be in the fully withdrawn position. Refer to Section 3.1.2.1.

b. The upper limit on excess reactivity ensures that an adequate shutdown margin is maintained.

c. The rod drop time required by Specification 3.2c assures that the Safety Limit will not be exceeded during the flow reversal which occurs upon loss of forced convection coolant flow. The rise in fuel temperature due to heat storage is partially controlled by the reactivity insertion associated with the SCRAM. The analysis of this transient is based upon this SCRAM reactivity insertion taking the form of a ramp function of two second duration. This analysis is found in SAR Section 3.2.4 and Appendix 3B. The rod drop time is the time interval measured between the instant of a test signal input to the SCRAM Logic Unit and the instant of the rod seated signal.

d. The maximum rate of reactivity insertion by the control rods which is allowed by Specification 3.2d assures that the Safety Limit will not be exceeded during a startup accident due to a continuous linear reactivity insertion. Refer to SAR Section 13.

Experiments affecting the reactivity condition of the reactor are commonly categorized by the sign of the reactivity effect produced by insertion of the experiment. An experiment having a large reactivity effect of either sign can also produce an undesirable flux distribution that could affect the peaking factor used in the Safety Limit calculations and the calibration of safety channels.

e. The Specification 3.2e is intended to prevent inadvertent reactivity changes during reactor operation caused by the insertion or removal of an experiment. It further provides assurance that the failure of a single experiment will not result in a reactivity insertion which could cause the Safety Limit to be exceeded. Analyses indicate that the inadvertent reactivity insertion of these magnitudes will not result in consequences greater than those analyzed in the SAR Sections 3 and 13.

f. The total limit on reactivity associated with experiments ensures that an adequate shutdown margin is maintained.

3.3 Reactor Safety System

Applicability

These specifications apply to the reactor safety system channels.

Objective

The objective is to require the minimum number of reactor safety system channels which must be operable in order to assure that the Safety Limits are not exceeded.

Specifications

The reactor shall not be operated unless the reactor safety system channels described in the following table are operable:

	Measuring Channel	Function
a.	Startup Power Level[1]	Inhibits Control Rod withdrawal when neutron count is ≤ 2 cps.
b.	Safety Power Level	SCRAM at ≤ 1.3 MW (LSSS), Enable for Flow/Flapper SCRAMs at ≤ 250 kW (LSSS).
c.	Linear Power Level	SCRAM at ≤ 1.3 MW (LSSS).
d.	Log N Power Level	Enable for Flow/Flapper SCRAMs at ≤ 250 kW (LSSS).
e.	Flow Monitoring[2]	SCRAM when flapper not closed and Flow/Flapper SCRAMs are enabled.
f.	Primary Coolant Flow[2]	SCRAM at ≥ 450 gpm (LSSS) when Flow/Flapper SCRAMs are enabled.
g.	Pool Water Temperature Monitoring Switch	Alarm and Manual SCRAM at ≤ 117°F (LSSS).
h.	Pool Water Temperature Measuring Channel	SCRAM at ≤ 117°F (LSSS).
i.	Pool Water Level	SCRAM at ≤ 14 feet 2 inches.

j.	Manual Button	Manual SCRAM
k.	Reactor Keyswitch	Manual SCRAM
l.	Over-the-Pool[(3)] Radiation Monitor	Alarm (100 mR/hr) and Manual SCRAM

[(1)]Required only for reactor startup when power level is less than 4 watts.

[(2)]Either the Flapper SCRAM or the Flow SCRAM may be bypassed during maintenance testing and/or performance of a startup checklist in order to verify each SCRAM is independently operable. The reactor must be shutdown in order to use these bypasses.

[(3)]Bypassed for less than two minutes during return of a pneumatic rabbit capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

Bases

The Startup Channel inhibit function assures the required startup neutron source is sufficient and in its proper location for the reactor startup, such that a minimum source multiplication count rate level is being detected to assure adequate information is available to the operator.

The reactor power level SCRAMs provide the redundant protection channels to assure that, if a condition should develop which would tend to cause the reactor to operate at an abnormally high power level, an immediate automatic protective action will occur to prevent exceeding the Safety Limit.

The primary coolant flow SCRAMs provide redundant protection channels to assure when the reactor is at power levels which require forced flow cooling that, if sufficient flow is not present, an immediate automatic shutdown of the reactor will occur to prevent exceeding a Safety Limit. The Log N Power Channel is included in this section since it is one of the two channels which enables the two flow SCRAMs when the reactor is above 250 kW (LSSS).

The pool water temperature channel provides for shutdown of the reactor and prevents exceeding the Safety Limit due to high pool water temperature.

The pool water level channel together with the Over-the-Pool (Bridge) radiation monitor, provides two diverse channels for shutdown of the reactor and prevents exceeding the Safety Limit due to insufficient pool height.

To prevent unnecessary initiation of the evacuation and confinement systems during the return of the pneumatic rabbit capsule from the core to the unloading station or during removal of experiments from the reactor pool, the over-the-pool monitor may be bypassed during the specified time interval.

The manual SCRAM button and the Reactor Keyswitch provide two manual SCRAM methods to the reactor operator if unsafe or abnormal conditions should occur.

3.4 Reactor Instrumentation

Applicability

These specifications apply to the instrumentation that shall be available to the reactor operator to support the safe operation of the reactor, but are not considered reactor safety systems.

Objective

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

Specifications

The reactor shall not be operated unless the following channels/systems/components listed are operable:

a. N-16 Power Measuring Channel[1]

b. Control Rod Position Indications (for each Control Rod and the Shim rod)

c. Differential pressure gauge for "Bay with respect to Atmosphere"

[1]Required when reactor power is greater than 500 kW.

Bases

The N-16 Channel provides the necessary power level information to allow adjustment of Safety and Linear Power Channels.

Control rod position indications give the operator information on rod height necessary to verify shutdown margin.

The differential pressure gauge provides the pressure difference between the Reactor Bay and the outside ambient and confirms air flow in the ventilation stream for both normal and confinement modes.

3.5 Radiation Monitoring Equipment

Applicability

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.

Objective

To assure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specification

The reactor shall not be operated unless the radiation monitoring equipment listed in the following table is operable.

a. Three fixed area monitors operating in the Reactor Building with their setpoints as follows:(1)(2)(3)

	ALERT SETPOINT	ALARM SETPOINT
i. Control Room	$\leq$ 2.5 mR/hr	$\leq$ 25 mR/hr
ii. Over-the-Pool	$\leq$ 10 mR/hr	$\leq$ 100 mR/hr
iii. West Wall	$\leq$ 10 mR/hr	$\leq$ 100 mR/hr

b. Particulate and gas building exhaust monitors continuously sampling air in the facility exhaust stack with their setpoints as follows:(1)(3)(4)

	ALERT SETPOINT	ALARM SETPOINT
i. Stack Gas	$\leq$ 900 ^{41}Ar AEC	$\leq$ 1000 ^{41}Ar AEC
ii. Stack Particulate	$\leq$ 900 ^{60}Co AEC	$\leq$ 1000 ^{60}Co AEC

Airborne Effluent Concentrations (AEC) values from 10CFR20 Appendix B, Table 2

c. The Radiation Rack Recorder.(5)

3.6 Confinement and Main HVAC Systems

Applicability

This specification applies to the operation of the Reactor Building confinement and main HVAC systems.

Objective

The objective is to assure that the confinement system is in operation to mitigate the consequences of possible release of radioactive materials resulting from reactor operation.

Specification

The reactor shall not be operated, nor shall irradiated fuel be moved within the pool area, unless the following equipment is operable, and conditions met:

	Equipment/Condition	Function
a.	All doors, except the Control Room, basement corridor entrance and PPV service hatch self-latching, self-closing, closed and locked.	To maintain reactor building negative differential pressure (dp).(1)
b.	Control room and basement corridor entrance door: self-latching, self-closing and closed.	To maintain reactor building negative differential pressure.(2)
c.	PPV service hatch closed, padlocked, and locked.	To maintain reactor building negative differential pressure.
d.	Reactor Building under a negative differential pressure of not less than 0.2" H_2O with the normal ventilation system or 0.1" H_2O with one confinement fan operating.	To maintain reactor building negative differential pressure with reference to outside ambient.(3)
e.	Confinement system	Operable(4)(5)
f.	Evacuation system	Operable(6)
g.	R-3 Ventilation Fans	Operable

(1)Doors may be opened by authorized personnel for less than five minutes for personnel and equipment transport provided audible and visual indication is available for the reactor operator to verify door status. Refer to SAR Section 5.

(2)Doors may be opened for periods of less than five minutes for personnel and equipment transport between corridor area and Reactor Building. Refer to SAR Section 5.

(3)During an interval not to exceed 30 minutes after a loss of dp (with Main HVAC operating) is identified, reactor operation may continue while the loss of dp is investigated and corrected. Refer to SAR Section 5.

(4)Operability also demonstrated with an auxiliary power source.

(5)One filter train may be out of service for the purpose of maintenance, repair, and/or surveillance for a period of time not to exceed 45 days. During the period of time in which one filter train is out of service, the standby filter train shall be verified to be operable every 24 hours during reactor operations with the Reactor Building in normal ventilation.

(6)The public address system can serve temporarily for the Reactor Building evacuation system during short periods of maintenance.

Bases

In the event of a fission product release, the confinement initiation system will secure the normal ventilation fans and close the normal inlet and exhaust dampers. In confinement, a confinement fan system will: maintain a negative pressure in the Reactor Building and insure in-leakage only; purge the air from the building at a greatly reduced and controlled flow through charcoal and absolute filters; and control the discharge of all air through a 100 foot stack on site. Section 5 of the SAR describes the confinement system's sequence of operation.

The allowance for operation under a temporary loss of dp when in normal ventilation is based on the requirement of having the confinement system operable and therefore ready to respond in the unlikely event of an airborne release.

3.7 Limitations of Experiments

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities. Fueled experiments must also meet the requirements of Specification 3.8.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist:

a. All materials to be irradiated shall be either corrosion resistant or encapsulated within a corrosion resistant container to prevent interaction with reactor components or pool water. Corrosive materials, liquids, and gases shall be doubly encapsulated.

b. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected by a factor of 2. Pressure buildup inside any container shall be limited to 200 psi.

c. Cooling shall be provided to prevent the surface temperature of an experiment to be irradiated from exceeding the saturation temperature of the reactor pool water.

d. Experimental apparatus, material or equipment to be inserted in the reactor shall be positioned so as to not cause shadowing of the nuclear instrumentation, interference with control rods, or other perturbations which may interfere with safe operation of the reactor.

e. Concerning the material content of experiments, the following will apply:

   i. No experiment will be performed unless the major constituent content of the material to be irradiated is known and a reasonable effort has been made to identify trace elements and impurities whose activation may pose the dominant radiological hazard. When a reasonable effort does not give conclusive information, one or

more short irradiations of small quantities of material may be performed in order to identify the activated products.

ii. Attempts will be made to identify and limit the quantities of elements having very large thermal neutron absorption cross sections, in order to quantify reactivity effects.

iii. Explosive material[1], shall not be allowed in the reactor. Experiments reviewed by the Radiation Protection Committee in which the material is considered to be potentially explosive, either while contained, or if it leaks from the container, shall be designed to maintain seal integrity even if detonated, to prevent damage to the reactor core or to the control rods or instrumentation and to prevent any change in reactivity.

iv. Each experiment will be evaluated with respect to radiation-induced physical and/or chemical changes in the irradiated material, such as decomposition effects in polymers.

v. Experiments involving flammable[1] or highly toxic materials[1] require specific procedures for handling and shall be limited in quantity as approved by the Radiation Protection Committee. No cryogenic liquids[1] will be allowed within the biological shield of the PULSTAR Reactor.

g. Credible failure of any experiment shall not result in releases or exposures in excess of the annual limits established in 10 CFR 20.

[1]Defined as follows (reference - "Handbook of Laboratory Safety" - Chemical Rubber Company, 4th Ed., 1995, unless otherwise noted):

Toxic: A substance that has the ability to cause damage to living tissue when inhaled, ingested, injected, or absorbed through the skin ("Safety in Academic Chemistry Laboratories" - The American Chemical Society, 1994).

Flammable: Having a flash point below 73°F and a boiling point below 100°F. The flash point is defined as the minimum temperature at which a liquid forms a vapor above its surface in sufficient concentration that it may be ignited as determined by appropriate test procedures and apparatus as specified.

Explosive: Any chemical compound, mixture, or device, the primary or common purpose of which is to function by explosion with substantially simultaneous release of gas and heat, the resultant pressure being capable of destructive effects. The term includes, but is not limited to, dynamite, black powder, pellet powder, initiating explosives, detonators, safety fuses, squibs, detonating cord, igniter cord, and igniters.

Cryogenic: A cryogenic liquid is considered to be a liquid with a normal boiling point below -238°F (reference - National Bureau of Standards Handbook 44).

Bases

Specifications 3.7a, 3.7b, 3.7c, and 3.7d are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure; and, serve as a guide for the review and approval of new and untried experiments by the facility personnel, as well as the Radiation Protection Committee.

Specification 3.7e insures that no physical or nuclear interferences compromise the safe operation of the reactor, specifically, an experiment having a large reactivity effect of either sign could produce an undesirable flux distribution that could affect the peaking factor used in the Safety Limit calculation and/or safety channels calibrations. Review of the experiments using these LCOs and the Administrative Controls of Section 6 will insure the insertion of experiments will not negate the considerations implicit in the Safety Limits and thereby become an Unreviewed Safety Question.

3.8 Operation with Fueled Experiments

Applicability

This specification applies to the operation of the reactor with any fueled experiment.

Objective

To assure that the confinement leak rate and fission product inventory are within the limits used in the PULSTAR Safety Analysis and are consistent with present U. S. Nuclear Regulatory Commission guides and the Code of Federal Regulations.

Specifications

Fueled experiments may be performed in experimental facilities of the PULSTAR reactor with the following conditions and limitations:

(1) The maximum mass of uranium-235 is limited to 400 milligrams.

(2) The thermal power (or fission rate) generated in the experiment is not greater than 5.5×10^8 fissions per second (1.77 milliwatt).

(3) The total exposure of the material is not greater than the limits set in Figure 3.8-1.

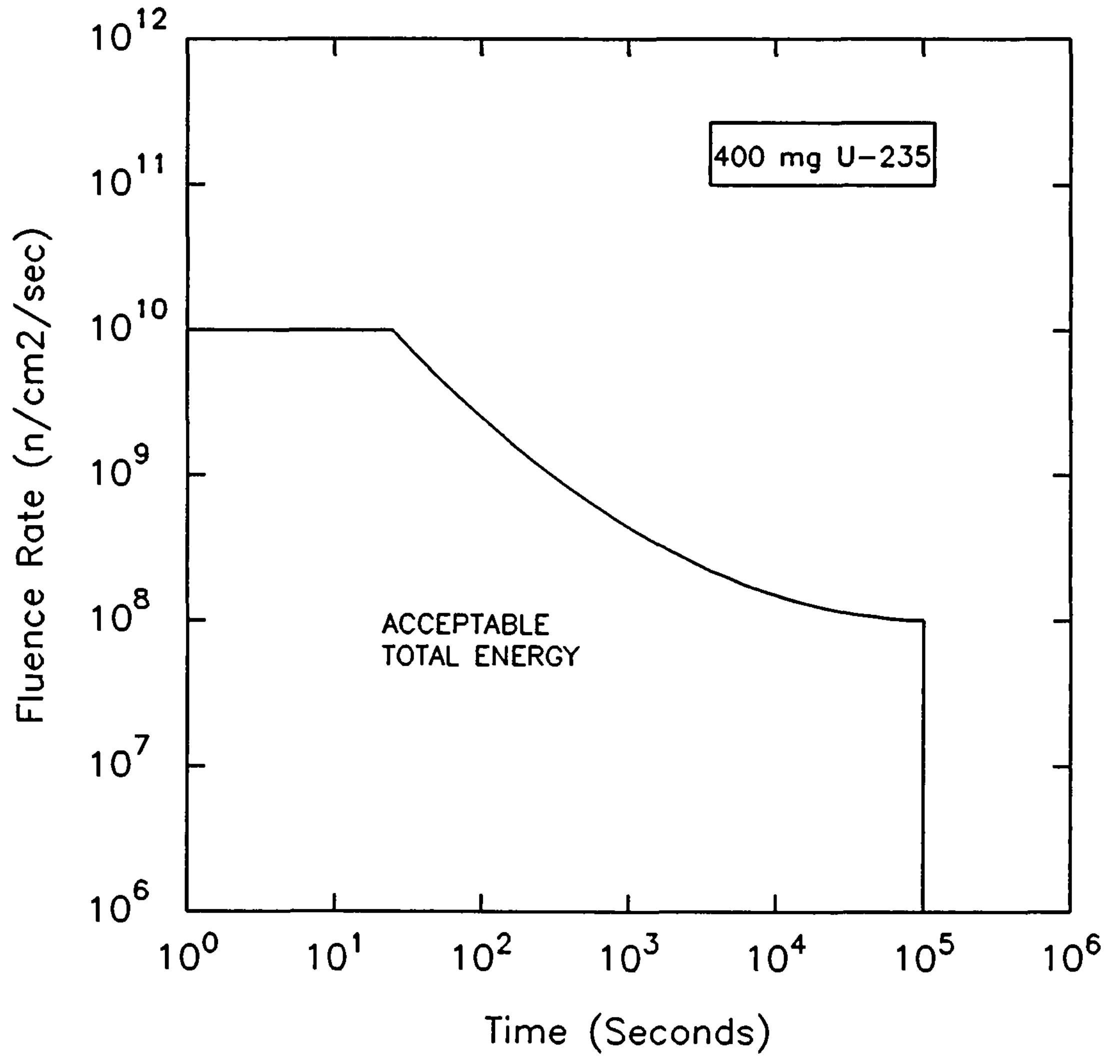
(4) The reactor shall not be operated with a fueled experiment unless the ventilation system is operated in the confinement mode.

(5) The specifications pertaining to reactor experiments, detailed in Section 3.7 Limitations of Experiments, apply to fueled experiments.

Bases

In the event of the failure of a fueled experiment with the subsequent release of fission products the inhalation exposure to these isotopes at any location is bounded by the Fuel Pin Clad Failure Analysis. The failed fueled experiment analysis is described in SAR Section 13.

Figure 3.8-1


400 mg U-235
ACCEPTABLE
TOTAL ENERGY
Fluence Rate (n/cm2/sec)
Time (Seconds)
10^{12}
10^{11}
10^{10}
10^{9}
10^{8}
10^{7}
10^{6}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}

3.9 Primary Coolant

Applicability

These specifications apply to the water quality and flow path of the primary coolant.

Objective

The objective is to ensure that primary coolant quality be maintained to acceptable values in order to reduce the potential for corrosion and limit the buildup of activated contaminants in the primary piping and pool.

Specifications

The reactor shall not be operated unless the pool water meets the following limits:

a. The resistivity shall be $\geq$ 500 k$\Omega\cdot$cm.

b. The pH shall be within the range of 5.5 to 7.5.

Bases

The limits on resistivity are based on reducing the potential for corrosion in the primary piping or pool liner and to reduce the potential for activated contaminants in these systems.

4.0 SURVEILLANCE REQUIREMENTS

The intent of the surveillance interval (e.g., annually, but not to exceed 15 months) is to maintain an average cycle, with occasional extensions as allowed by the interval tolerance.

If it is desired to permanently change the scheduled date of a surveillance (e.g., move an August scheduled date to an April scheduled date because of lower seasonal temperatures), the particular surveillance item will be performed at the earlier date and the associated interval normalized to this revised earlier date. In no cases will permanent scheduling changes (that yield slippage of the surveillance interval routine scheduled date) be made by using the allowed interval tolerance.

Surveillance tests, (except those specifically required for safety when the reactor is shutdown) may be deferred during reactor shutdown; however, they must be completed prior to reactor startup (or immediately after the startup, if and only if reactor operations are required to perform the surveillance item). Surveillance requirements scheduled to occur during operation which cannot be performed with the reactor operating may be deferred until planned reactor shutdown.

4.1 Fuel

Applicability

This specification applies to the surveillance requirement for the reactor fuel.

Objective

The objective is to monitor the physical condition of the PULSTAR fuel.

Specifications

All fuel assemblies shall be visually inspected biennially but at intervals not to exceed thirty (30) months.

Bases

The assemblies are inspected for physical damage including corrosion of endplates, nosepiece and zircaloy box; missing assembly screws; dented and scratched surfaces; and blockage of coolant channels.

The biennial inspection of PULSTAR fuel assemblies in conjunction with the monthly primary coolant analysis has been shown to be adequate for Zr-2 clad assemblies to insure fuel assembly integrity based on a long history of the prototype PULSTAR steady state and pulse operation.

4.2 Control Rods

Applicability

This specification applies to the surveillance requirements for the control rods, shim rod, and control rod drive mechanisms (CRDM).

Objective

The objective is to assure the operability of the control rods and shim rod, and to provide current reactivity data for use in verifying adequate shutdown margin.

Specifications

a. The reactivity worth of the shim rod and each control rod shall be determined annually but at intervals not to exceed fifteen months for the steady state core in current use. The reactivity worth of all rods shall be determined for any new core or rod configuration, prior to routine operation.

b. Control rod drop times[1] and control rod drive times shall be determined: (1) annually but at intervals not to exceed fifteen months, and (2) after a control assembly is moved to a new position in the core or after maintenance or modification is performed on the control rod drive mechanism.

c. The control rods shall be visually inspected biennially but at intervals not to exceed thirty months.

d. The values of excess reactivity and shutdown margin shall be determined monthly, but at intervals not to exceed six weeks, and for new core configurations.

[1]Applies only to magnetically coupled rods.

Bases

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide a means for determining the reactivity worths of experiments inserted in the core. The measurement of reactivity worths on an annual basis provides a correction for the slight variations expected due to burnup. This frequency of measurement has been found acceptable at similar research reactor facilities, particularly the prototype PULSTAR which has a similar slow change of rod value with burnup.

Control rod drive and drop time measurements are made to determine whether the rods are functionally operable. These time measurements may also be utilized in reactor transient analysis.

Visual inspections include: detection of wear or corrosion in the rod drive mechanism; identification of deterioration, corrosion, flaking or bowing of the neutron absorber material; and verification of rod travel setpoints.

Control rod surveillance procedures will document proper control rod system reassembly after maintenance and recorded post-maintenance data will identify significant trends in rod performance.

4.3 Reactor Instrumentation and Safety Systems

Applicability

This specification applies to the surveillance requirements for the Reactor Safety System and other required reactor instrumentation,.

Objective

The objective is to assure that the required instrumentation and Safety Systems will remain operable and will prevent the Safety Limits from being exceeded.

Specifications

a. A channel check of each measuring channel in the RSS shall be performed daily when the reactor is in operation.

b. A channel test of each channel in the RSS shall be performed prior to each day's operation, or prior to each operation extending more than one day.

c. A channel calibration of the ^{16}N Channel shall be made semi-annually, but at intervals not to exceed seven and one-half months. A calorimetric measurement shall be performed to determine the ^{16}N detector current associated with full power operation.

d. A channel calibration of the following channels shall be made semi-annually but at intervals not to exceed seven and one-half months. [1]

   1. Pool Water Temperature (T_2)
   2. Primary Cooling and Flow Monitoring (Flapper)
   3. Pool Water Level
   4. Primary Heat Exchanger Inlet (T_5) and Outlet Temperature (T_6)
   5. Safety and Linear Power Channels

[1] A channel calibration shall also be required after repair of a channel component that has the potential of affecting the calibration of the channel.

Bases

The daily channel tests and checks will assure that the Reactor Safety Systems are operable and will assure operations within the limits of the operating license. The semi-annual calibrations will assure that long-term drift of the channels is corrected. The calorimetric calibration of the reactor power level, in conjunction with the Nitrogen-16 Channel, provides a continual reference for adjustment of the Linear, Log N and Safety Channel detector positions.

4.4 Radiation Monitoring Equipment

Applicability

This specification applies to the surveillance requirements for the area and stack effluent radiation monitoring equipment.

Objective

The objective is to assure that the radiation monitoring equipment is operable.

Specification

(a) The area and stack monitoring systems shall be calibrated annually but at intervals not to exceed fifteen months.

(b) The setpoints shall be verified weekly, but at intervals not to exceed 10 days.

Bases

These systems provide continuous radiation monitoring of the Reactor Building with a check of readings performed prior to and during reactor operations. Therefore, the weekly verification of the setpoints in conjunction with the annual calibration is adequate to identify long-term variations in the system operating characteristics.

4.5 Confinement and Main HVAC System

Applicability

This specification applies to the surveillance requirements for the confinement and main HVAC systems.

Objective

The objective is to assure that the confinement system is operable.

Specifications

a. The confinement and evacuation system shall be verified to be operable within seven days prior to reactor operation.

b. Operability of the confinement system on auxiliary power will be checked monthly but at intervals not to exceed six weeks.(1)

c. A visual inspection of the door seals and closures, dampers and gaskets of the confinement and ventilation systems shall be performed semi-annually but at intervals not to exceed seven and one-half months to verify they are operable.

d. The control room differential pressure (dp) gauges shall be calibrated annually but at intervals not to exceed fifteen months.

e. The confinement filter train shall be tested biennially but at intervals not to exceed thirty months and prior to reactor operation following confinement HEPA or carbon adsorber replacement. This testing shall include iodine adsorption, particulate removal efficiency and leak testing of the filter housing.(2)

f. The air flow rate in the confinement stack exhaust duct shall be determined annually but at intervals not to exceed fifteen months.

g. The air flow rate of the Burlington South Wing (R-3) ventilation fans shall be determined annually, but at intervals not to exceed fifteen months.

(1)Operation must be verified following modifications or repairs involving load changes to the auxiliary power source.

(2)Testing shall also be required following major maintenance of the filters or housing.

Bases

Surveillance of this equipment will verify that the confinement of the Reactor Building is maintained as described in Section 5 of the SAR.

4.6 Primary and Secondary Coolant

Applicability

This specification applies to the surveillance requirement for monitoring the radioactivity in the primary and secondary coolant.

Objective

The objective is to monitor the radioactivity in the pool water to verify the integrity of the fuel cladding and other reactor structural components. The secondary water analysis is used to confirm the boundary integrity of the primary heat exchanger.

Specification

a. The primary coolant shall be analyzed bi-weekly, but at intervals not to exceed 18 days. The analysis shall include gross beta/gamma counting of the dried residue of a 1 liter sample or gamma spectroscopy of a liquid sample, Neutron Activation Analysis (NAA) of an aliquot, and pH and resistivity measurements.

b. The secondary coolant shall be analyzed bi-weekly, but at intervals not to exceed 18 days. This analysis shall include gamma spectroscopy of a liquid sample.

Bases

Radionuclide analysis of the pool water samples will allow detection of fuel clad failure, while neutron activation analysis will give corrosion data associated with primary system components in contact with the coolant. Refer to SAR Section 10.

The detection of activation or fission products in the secondary coolant provides evidence of a primary heat exchanger leak. Refer to SAR Section 10.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

a. The reactor fuel shall be UO_2 with a nominal enrichment of 4% in ^{235}U, zircaloy clad, with fabrication details as described in Section 3 of the Safety Analysis Report.

b. Total burnup on the reactor fuel is limited to 20,000 MWD/MTU.

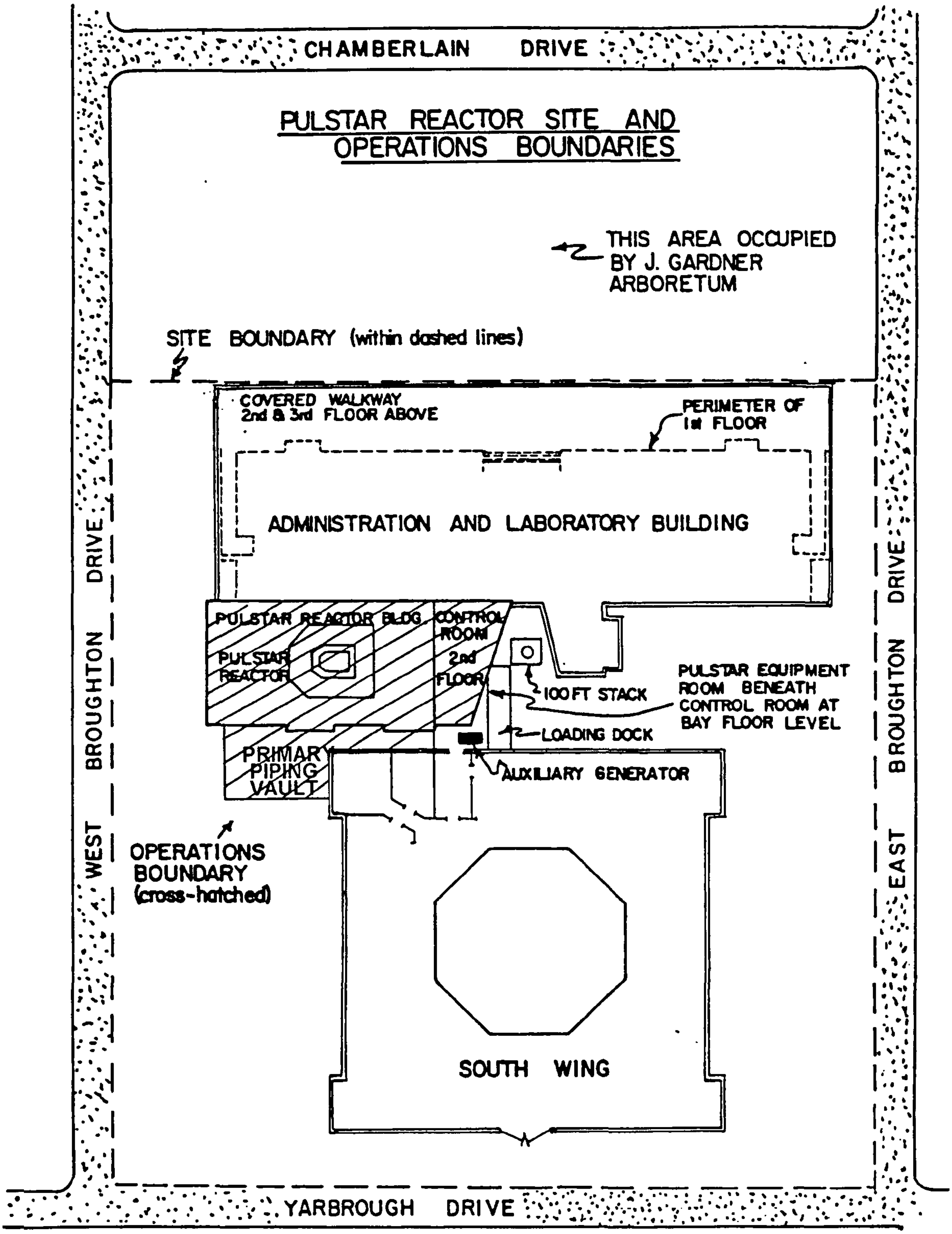
5.2 Reactor Building

a. The reactor shall be housed in the Reactor Building, designed for confinement. The minimum free volume in the Reactor Building shall be 2.25×10^9 cm^3 (refer to SAR Section 13 analysis).

b. The Reactor Building ventilation and confinement systems shall be separate from the Burlington Engineering Laboratories building systems and shall be designed to exhaust air or other gases from the building through a stack with discharge at a minimum of 100 feet above ground level.

c. The openings into the Reactor Building are the truck entrance door, personnel entrance doors, PPV service hatch, and air supply and exhaust ducts.

d. The Reactor Building is located within the Burlington Engineering Laboratory complex on the north campus of North Carolina State University at Raleigh, North Carolina. Restricted Areas as defined in 10 CFR 20 include the PULSTAR Reactor Bay, Mechanical Equipment Room, Primary Piping Vault, and Waste Tank Vault. The PULSTAR Control Room is part of the Reactor Building, however it is also a controlled access area and a Controlled Area as defined in 10 CFR 20. The facility license applies to the Reactor Building and Waste Tank Vault. Figure 5.2-1 depicts the licensed area as being within the operations boundary.


CHAMBERLAIN DRIVE
PULSTAR REACTOR SITE AND OPERATIONS BOUNDARIES
THIS AREA OCCUPIED BY J. GARDNER ARBORETUM
SITE BOUNDARY (within dashed lines)
COVERED WALKWAY 2nd & 3rd FLOOR ABOVE
PERIMETER OF 1st FLOOR
ADMINISTRATION AND LABORATORY BUILDING
PULSTAR REACTOR BLDG
CONTROL ROOM
2nd FLOOR
PULSTAR REACTOR
100 FT STACK
PULSTAR EQUIPMENT ROOM BENEATH CONTROL ROOM AT BAY FLOOR LEVEL
LOADING DOCK
PRIMARY PIPING VAULT
AUXILIARY GENERATOR
OPERATIONS BOUNDARY (cross-hatched)
SOUTH WING
WEST BROUGHTON DRIVE
EAST BROUGHTON DRIVE
YARBROUGH DRIVE

5.3 Fuel Storage

Fuel, including fueled experiments and fuel devices not in the reactor, shall be stored in a geometrical configuration where k_{eff} is no greater than 0.9 for all conditions of moderation and reflection using light water except in cases where a fuel shipping container is used, then the licensed limit for the k_{eff} limit of the container shall apply.

5.4 Reactivity Control

Reactivity control is provided by four neutron absorbing blades. Each control blade is nominally comprised of 80% silver, 15% indium, and 5% cadmium with nickel cladding. Three of these neutron absorbing blades are magnetically coupled and have scramming capability. The remaining neutron absorbing blade is non-scrammable. One of the scrammable rods may be used for automatic servo-control of reactor power. When in use, the servo-control maintains a constant power level as indicated by the Linear Power Channel.

5.5 Primary Coolant System

The Primary Coolant System consists of the aluminum lined reactor tank, a ^{16}N decay tank, a pump, and heat exchanger, and associated stainless steel piping. The nominal capacity of the primary system is 15,600 gallons. Valves are located adjacent to the biological shield to allow isolation of the pool, and at major components in the primary system to permit isolation.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

The reactor facility shall be an integral part of the Department of Nuclear Engineering of the College of Engineering of North Carolina State University. The reactor shall be related to the University structure as shown in Figure 6.1-1. Responsibility for the safe operation of the PULSTAR Reactor shall be with the chain of command established in Figure 6.1-1. Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and these technical specifications, and federal regulations.

6.1.1 Organizational Structure

The following specific organizational levels (as defined by ANSI-15.1-1990) and positions shall exist at the PULSTAR Facility:

LEVEL 1: This level shall include the Chancellor, the College of Engineering Dean, the Nuclear Engineering Department Head, and the Director of the Nuclear Reactor Program.

The Director of the Nuclear Reactor Program is responsible for the long range development of the Nuclear Reactor Program and for the general conduct of Program operations. He evaluates new service and research applications for the PULSTAR Reactor, recruits new users of the facilities, supervises the development and expenditure of Program budgets, develops department and university support for needed capital investment, and works through the Associate Director to monitor daily operations. While the Associate Director supervises daily operations and makes routine decisions related to safety and operation, the Director is responsible for reactor performance and personnel matters. The minimum qualifications for the Director are a M.S. degree in engineering or physical science and at least ten years of reactor experience related to fission reactor technology. A Ph.D. in Nuclear Engineering or Physics may substitute for two years of experience.

LEVEL 2 - Associate Director of the Nuclear Reactor Program: The Associate Director is responsible for the safe and efficient operation of the PULSTAR Reactor Facility. In matters pertaining to the operation of the facility and these Specifications, the Associate Director reports to the Director of the Nuclear Reactor Program. The Associate Director and Director of the Nuclear Reactor Program consult with the Head, Department of Nuclear Engineering on

PULSTAR operations matters as required. The minimum qualifications for the Associate Director are a Bachelor of Science Degree in engineering or physical science, at least five years of reactor analysis or technical support, and at least two years of supervisory experience. A B.S. degree in Nuclear Engineering or Physics may substitute for one year of the reactor analysis and or technical support experience.

LEVEL 3 - Reactor Operations Manager: The Reactor Operations Manager, who shall be qualified as a Senior Reactor Operator, shall be responsible for assuring that operations are conducted in a safe manner and within the limits prescribed by the facility license, all applicable Nuclear Regulatory Commission regulations, and the provisions of the Radiation Protection Committee. The Reactor Operations Manager reports directly to the Associate Director of the Nuclear Reactor Program.

LEVEL 4 - Operating Staff: This level includes the positions of Chief Reactor Operator, Chief of Reactor Maintenance, and the remaining Senior and Reactor operators. Personnel at this level report to the Reactor Operations Manager (for PULSTAR Reactor related matters).

Reactor Health Physicist: The Reactor Health Physicist is responsible for assuring the safety of reactor operations from the standpoint of radiation protection. The Reactor Health Physicist reports directly to the Nuclear Engineering Department Head and shall function independent of the campus Radiation Protection Office as shown in Figure 6.1-1. He shall possess relevant practical experience in the application of health physics principles.

In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon the appropriate qualifications.

6.1.2 Minimum Staffing

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

a. A certified reactor operator (either Senior Operator or Operator) in the Control Room.

b. A Reactor Operator Assistant (ROA), capable of being at the reactor facility within five minutes upon request of the reactor operator on duty.

c. A Senior Reactor Operator. This individual may be referred to as the "Designated Senior Reactor Operator (DSRO)" and shall be readily on call, meaning:

i. Has been specifically designated and the designation known to the reactor operator on duty.

ii. Keeps the reactor operator on duty informed of where he may be rapidly contacted and the phone number.

iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15 mile radius).

d. A Reactor Health Physicist, or his designated alternate. This individual shall also be on call, under the same limitations as prescribed for the Senior Reactor Operator under specification (c).

6.1.3 Senior Reactor Operator Duties

The following events shall require the presence of a Senior Reactor Operator at the facility or its administrative offices:

a. Initial startup and approach to power (i.e., required after each time interval that the reactor is secured).

b. All fuel or control rod relocations within the reactor core or pool.

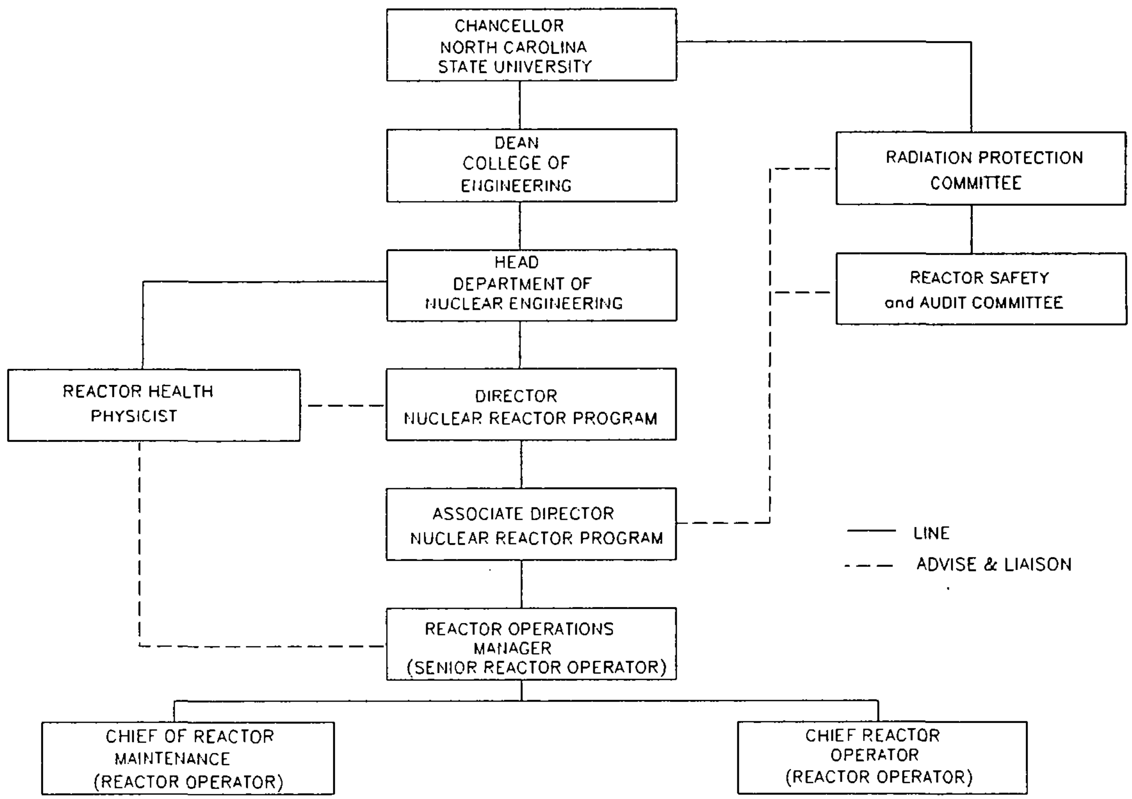
c. Relocation of any in-core experiment with a reactivity worth greater than one dollar.

d. Recovery from unplanned or unscheduled shutdown or significant power reduction (documented verbal concurrence from a Senior Reactor Operator is acceptable only for a known event).

6.1.4 Selection and Training

All operators will undergo a selection, training and certification program prior to unsupervised operation of the PULSTAR reactor. All licensed operators will participate in a requalification program, which will be conducted over a period not to exceed two years. The requalification program will be followed by successive two year programs.

NORTH CAROLINA STATE UNIVERSITY
PULSTAR REACTOR ORGANIZATIONAL CHART



SAR11-1.DC2

Figure 6.1-1

6.2 Review and Audit

6.2.1 Radiation Protection Committee and Reactor Safety and Audit Committee

The Radiation Protection Committee (RPC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices, including the nuclear reactor, is conducted in the safest possible manner with the minimum effect on members of the University community and the general public. The RPC exercises oversight over the University Radiation Protection Program and performs final review of the actions of the Reactor Safety and Audit Committee (RSAC).

RSAC has the primary responsibility to assist the RPC in ensuring that the reactor is operated in compliance with the facility license and all applicable regulations. RSAC performs an annual audit of the operations and performance of the reactor program.

6.2.2 RPC and RSAC Composition and Qualifications

a. RPC shall consist of at least Seven voting members from the general faculty. At least five of these faculty members are appointed to three year staggered terms by the University. Their terms shall be staggered so that no more than three of these five members may be replaced each year. These members shall be selected from faculty who are actively engaged in teaching and/or research involving the use of radiation or who manifest a high degree of expertise in the areas of nuclear science and related fields. One voting member is appointed by the Faculty Senate, whose term shall not exceed three years, and another voting member is appointed from the Department of Nuclear Medicine. Less than a majority of the RPC members shall be from the line organization presented in Figure 6.1-1. Non-voting ex-officio members include the University Radiation Protection Officer and the Director of Environmental Health and Safety. Representatives from the Physical Plant Division, University Research Administration, and Nuclear Reactor Program may serve in a non-voting liaison capacity. The RPC shall prescribe which review items (detailed in 6.2.3) are to be delegated to RSAC.

b. RSAC shall consist of at least five persons who have expertise in one or more of the component areas of nuclear reactor safety. These include Nuclear Engineering, Nuclear Physics, Health Physics, Electrical Engineering, Chemical Engineering, Material Engineering, Radiochemistry, and Nuclear Regulatory Affairs. At least three of the members are appointed from the general faculty by the University upon recommendation by the RPC. These faculty members shall be constituted as follows: Director of the Nuclear Reactor Program shall serve as a permanent

member, one member from an appropriate discipline within the College of Engineering, and one member from the general faculty. Appointments are for three years. The remaining RSAC members are the Reactor Health Physicist and a member from the Radiation Protection Division of the Environmental Health and Safety Center who serve as permanent members. An additional member may represent an outside nuclear related agency. At the discretion of RSAC, specialist from other universities and outside establishments may be invited to assist in its appraisals.

c. A quorum shall consist of not less than a majority of the full RPC or RSAC and shall include the chairman or his designated alternate. Members from the line organization shown in Figure 6.1-1 shall not form a quorum.

d. The RPC shall meet at least six times per year. While the RSAC shall meet at least four times per year, with intervals between meetings not to exceed six months. RSAC may also meet as specifically required by the audit function or upon call of the Chairman.

6.2.3 RPC/RSAC Review and Approval Function

The following items shall be reviewed and approved by the RPC or by referral to RSAC, as needed:

a. Determinations that proposed changes in equipment, systems, test, experiments, or procedures which have safety significance do not involve an unreviewed safety question.

b. All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

c. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

d. Proposed changes to the Technical Specifications or facility license.

e. Violations of technical specifications or license. Violations of internal procedures or instructions having safety significance.

f. Operating abnormalities having safety significance.

g. Reportable Events (as per technical specification definition 1.24).

h. Audit reports.

RPC summaries and meeting minutes shall be provided to the Chancellor, Provost, Vice Chancellor for Research, Vice Chancellor for Business and Finance, Faculty Senate, and University Archives.

A summary of RSAC meeting minutes, reports, and audit recommendations approved by RSAC shall be submitted to Dean of the College of Engineering, Head of the Nuclear Engineering Department, Director of the Nuclear Reactor Program, Associate Director of the Nuclear Reactor Program, the RPC, Director of Environmental Health and Safety, and the RSAC prior to the next scheduled RSAC meeting. Recommendations of the annual audit made by RSAC are forwarded to the RPC for concurrence before being implemented.

6.2.4 RSAC Audit Function

The audit function shall consist of selective, but comprehensive, examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations shall also be used as appropriate. The RSAC, under the authority of the RPC, shall be responsible for this audit function. This audit shall include:

a. Facility operations for conformance to the technical specifications and license, annually, but at intervals not to exceed fifteen months.

b. The retraining and requalification program for the operating staff, biennially, but at intervals not to exceed thirty months.

c. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, annually, but at intervals not to exceed fifteen months.

d. The Emergency Plan and Emergency Procedures, biennially, but at intervals not to exceed thirty months.

e. Radiation Protection.

Deficiencies uncovered that affect reactor safety shall be immediately reported to the Head of the Nuclear Engineering Department, Director of the Nuclear Reactor Program and the Associate Director of the Nuclear Reactor Program, and the RPC.

6.3 Operating Procedures

Written procedures shall be prepared, reviewed and approved prior to initiating any of the following:

a. Startup, operation and shutdown of the reactor.

b. Fuel loading, unloading, and movement within the reactor.

c. Maintenance of major components of systems that could have an affect on reactor safety.

d. Surveillance checks, calibrations and inspections required by the technical specifications or those that may have an affect on reactor safety.

e. Personnel radiation protection, consistent with applicable regulations and that include commitment and/or programs to maintain exposures and releases as low as reasonably achievable (ALARA).

f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

g. Implementation of the emergency plan and security plan.

Substantive changes to the above procedures shall be made effective only after documented review by the RPC (or RSAC as applicable) and approval by the Associate Director of the Nuclear Reactor Program, or his designated alternate.

Minor modifications to the original procedures which do not change their original intent may be made by the Reactor Operations Manager, but the modifications shall be approved by the Associate Director of the Nuclear Reactor Program within 14 days.

Temporary deviations from procedures may be made by the Senior Reactor Operator (on duty as required by specification 6.1.2 c.) or Reactor Operations Manager, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to the Associate Director of the Nuclear Reactor Program, or his designated alternate.

6.4 Review of Experiments

6.4.1 New (untried) Experiments

All new experiments or class of experiments, referred to as "untried" experiments, shall be reviewed and approved by the Associate Director of the Nuclear Reactor Program, Reactor Health Physicist, and the Radiation Protection Committee (or RSAC as applicable), prior to initiation of the experiment.

The review of new experiments shall be based on the limitations prescribed by Technical Specifications 3.7 and 3.8 and other Nuclear Regulatory Commission regulations, as applicable. If the Radiation Protection Committee, the Associate Director of the Nuclear Reactor Program, and the Reactor Health Physicist jointly agree that the experiment can be safely performed within the limitations of the technical specifications and other applicable Nuclear Regulatory Commission regulations, then an approved PULSTAR Project Number can be issued by the RPC for the experiment.

6.4.2 Tried Experiments

All proposed experiments are reviewed by the Reactor Operations Manager and the Reactor Health Physicist (or their designated alternates). Either of these individuals may deem that the proposed experiment is not adequately covered by the documentation/analysis associated with an existing approved PULSTAR Project and therefore constitutes an untried experiment that will require the approval process detailed under Technical Specification 6.4.1. If the Reactor Operations Manager and the Reactor Health Physicist concur that the experiment is a tried experiment, then the request is approved and the experiment can be scheduled within the limitations of the reactor operating schedule.

Substantive changes to previously approved experiments shall be made only after review and approval by the Associate Director of the Nuclear Reactor Program, Reactor Health Physicist, and the Radiation Protection Committee (or RSAC as applicable).

6.5 Action to be Taken in Case of Safety Limit Violation

In the event a Safety Limit is violated:

a. The reactor shall be shut down and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. The Safety Limit violation shall be promptly reported to the Associate Director of the Nuclear Reactor Program, or his designated alternate.

c. The Safety Limit violation shall be reported to the Nuclear Regulatory Commission in accordance with specification 6.7.1.

d. A Safety Limit violation report shall be prepared that describes the following:

   i. Circumstances leading to the violation including, when known, the cause and contributing factors.

   ii. Effect of violation upon reactor facility components, systems, or structures and on the health and safety of facility personnel and the public.

   iii. Corrective action to be taken to prevent recurrence.

The report shall be reviewed by the Radiation Protection Committee and any follow-up report shall be submitted to the Nuclear Regulatory Commission when authorization is sought to resume operation.

6.6 Action to be Taken for Reportable Events (other than SL Violation)

In case of a Reportable Event (other than violation of a Safety Limit), as defined by section 1.22 of these specifications, the following action shall be taken:

a. Reactor conditions shall be returned to normal or the reactor shall be shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Associate Director of the Nuclear Reactor Program, or his designated alternate.

b. The occurrence shall be reported to the Associate Director of the Nuclear Reactor Program, and to the Nuclear Regulatory Commission in accordance with specification 6.7.1

c. The occurrence shall be reviewed by the Radiation Protection Committee at their next scheduled meeting.

6.7 Reporting Requirements

6.7.1 Reportable Event

For Reportable Events as defined by section 1.22 of these specifications, there shall be a report not later than the following work day by telephone to the Nuclear Regulatory Commission Operations Center and the Nuclear Regulatory Commission Region II Regional Administrator, followed by a written report within 14 days that describes the circumstances of the event

6.7.2 Permanent Changes in Facility Organization

Permanent changes in the facility organization involving either Level 1 or 2 personnel (refer to specification 6.1) shall require a written report within 30 days to the Nuclear Regulatory Commission Document Control Desk and the Nuclear Regulatory Commission Region II Regional Administrator.

6.7.3 Changes Associated with the Safety Analysis Report

Significant changes in the transient or accident analysis as described in the Safety Analysis Report shall require a written report within 30 days to the Nuclear Regulatory Commission Document Control Desk and the Nuclear Regulatory Commission Region II Regional Administrator.

6.7.4 Annual Operating Report

An annual operating report is required to be submitted no later than August 31st of each year and will cover the period of July 1st through June 30th. The original is transmitted to the Document Control Desk, Nuclear Regulatory Commission, Washington, with a copy transmitted to the Nuclear Regulatory Commission Region II Regional Administrator. The annual report shall contain as a minimum, the following information:

a. A brief narrative summary:

   i. Operating experience including a summary of experiments performed.

   ii. Changes in performance characteristics related to reactor safety that occurred during the reporting period

   iii. Results of surveillance, tests and inspections.

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

c. The number of emergency shutdowns and unscheduled SCRAMs, including reasons therefore, and corrective actions.

d. Discussion of the corrective and preventative maintenance operations performed during the period, including the effect, if any, on the safety of operation of the reactor.

e. A brief description, including a summary of the analyses and conclusions of changes in the facility or in procedures and of tests and experiments carried out pursuant to Section 50.59 of 10 CFR.

f. A summary of the nature and amount of radioactive effluent released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge, including:

Liquid Waste (summarized by quarter)

i. Radioactivity released during the reporting period:

(1) Number of batch releases.

(2) Total radioactivity released (in microcuries).

(3) Total liquid volume released (in liters).

(4) Diluent volume required (in liters).

(5) Tritium activity released (in microcuries).

(6) Total (yearly) tritium released.

(7) Total (yearly) activity released.

ii. Identification of fission and activation products:

Whenever the concentration of radioactivity in the waste tank at the time of release without dilution exceeds 2×10^{-5} μCi/ml, as determined by a gross beta-gamma count of the dried residue of a one liter sample and shall also be analyzed prior to release for principal gamma emitting radionuclides. An estimate of the quantities present shall be reported for each of the identified nuclides. Refer to SAR Section 10.

iii. Disposition of liquid effluent not releasable to the sanitary sewer system:

Any waste tank containing liquid effluent failing to meet the requirements of 10 CFR 20, Appendix B, reported hereunder, to include the following data:

(1) Method of disposal.

(2) Total radioactivity in the tank (in microcuries) prior to disposal.

(3) Total volume of liquid in tank (in liters).

(4) The dried residue of a one liter sample shall be analyzed for the principal gamma-emitting radionuclides. The identified isotopic composition with estimated concentrations shall be reported. The tritium content shall be included.

<u>Gaseous Waste</u> (summarized on a monthly basis)

i. Radioactivity discharged during the reporting period (in curies) for:

(1) Gases

(2) Particulates, with half lives greater than eight days.

ii. The AEC used and the estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis. (AEC values are given in 10 CFR 20, Appendix B, Table 2.)

<u>Solid Waste</u>

i. The total amount of solid waste packaged (in cubic feet).

ii. The total activity involved (in curies).

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

g. A summary of radiation exposures received by facility personnel and visitors, including pertinent details of significant exposures.

h. A summary of the radiation and contamination surveys performed within the facility and significant results.

i. A description of environmental surveys performed outside the facility.

6.8 Retention of Records

Records and logs of the following items, as a minimum, shall be kept in a manner convenient for review and shall be retained as detailed below. In addition, any additional federal requirement in regards to record retention shall be met.

a. Records to be retained for a period of at least five (5) years:

   i. Normal plant operation and maintenance.

   ii. Principal maintenance activities.

   iii. Reportable events.

   iv. Equipment and components surveillance activities.

   v. Experiments performed with the reactor.

   vi. Changes to Operating Procedures

   vii. Audit summaries

   viii. RPC and RSAC meeting minutes

b. Records to be retained for the life of the facility:

   i. Gaseous and liquid radioactive waste released to the environs.

   ii. Results of off-site environmental monitoring surveys.

   iii. Radiation exposures for all PULSTAR personnel.

   iv. Results of facility radiation and contamination surveys.

   v. Fuel inventories and transfers.

   vi. Drawings of the reactor facility.

c. Records to be retained for at least one training cycle:

   i. Records of retraining and requalification of certified operating personnel shall be maintained at all times the individual is employed, or until the certification is renewed.

North Carolina State University PULSTAR
Docket No.: 50-297
License No.: R-120

Special Procedure 2.6 PULSTAR Operator Requalification Program Revision 6

Special Procedure 2.6 Revision 6
Change Synopsis

A. Section 1.0

1. Pg. 1 — Updated the ANS standard.

2. Pg. 1 — Changed "...justified by the mode of operation at the PULSTAR." to "...justified by the mode of operation and the unique design of the PULSTAR Reactor." This change was made to clearly state why deviations (as allowed by 10 CFR 55.59(7)) from the requirements in 10 CFR 55 are justified.

B. Section 2.0

Pg. 2 — Updated the definitions and added a definition for Designated Senior Reactor Operator.

C. Section 3.0

Pg. 3 — References were updated and ANS 15.1 Development of Technical Specifications for Research Reactors and NUREG-1478 Non-Power Reactor Operator Licensing Examiner Standards added.

D. Section 4.0

Pg. 3 — Requalification interval was changed to the frequency specified in 10 CFR 50.59(a)(1).

E. Section 5.1

Pg. 3 — Changed "Lectures which may be..." to "Lectures which shall be..." This change clearly states the lectures are a requirement as stated in 10 CFR 55.59(c)(2).

F. Section 5.5

Pg. 5 — Changed "...at the earliest possible time following the missed lecture." to "...prior to the next scheduled requalification lecture."

G. Section 7.1

1. Pg. 5 — Written exam interval was changed to the frequency specified in 10 CFR 50.59(a)(1).

2. Pg. 5 — The structure of the written exam was changed to one similar in format to NRC examinations.

H. Section 7.2

Pg. 5 — Because of the changes in paragraph 7.1 ROs and SROs will now take the same type of written exam. Paragraph 7.2 outlines the additional knowledge requirements required of a SRO.

I. Section 8.1

Pg. 6 — The requirements to maintain proficiency were clarified to eliminate confusion. The requirements for DSROs were separated to eliminate confusion.

J. Section 8.2

Pg. 6 — This section was added to clearly state which of the items listed in 10 CFR 55.59(3)(i) apply to the PULSTAR reactor. Furthermore, as allowed by 10 CFR 50.59(4)(iv), this paragraph indicates which items will be accomplished by discussion.

K. Section 8.3

Pg. 7 — Additional forms were added to aid in tracking the completion of the requirements added in Section 8.2

L. Section 9.1

Pg. 7 — The Oral/Demonstrational exam frequency was changed to the recommended interval in ANS 15.1. The operational exam content was changed to indicate which items listed in 10 CFR 55.45(a) apply to the PULSTAR reactor. The structure of the operational exam was changed to one similar to the NRC exams outline in NUREG-1478.

M. Section 10.0

Pg. 10 — The intervals for review of documents was changed to the recommended frequency in ANS-15.1.

N. Section 11.0

Pg. 10 — The requirements for passing the Oral/Demonstrational and the Written exams were separated and expanded to clearly state what is required to pass each type of exam.

J. Section 12.0

Pg. 11 — Clarified the required hours under instruction as specified in 10 CFR 55.53(f)(2) for re-establishing proficiency.

Note: New forms have been added to this procedure as required by the changes and existing form numbers have been changed where necessary.

Special Procedure 2.6
PULSTAR Operator Requalification Program

1.0 INTRODUCTION

The purpose of the PULSTAR Operator Requalification Program is to ensure that the continuous high level of personnel proficiency requisite to the safe and efficient operation of the North Carolina State University PULSTAR Reactor is maintained. This program is intended to conform with the requirements of 10 CFR 55 as applied to research reactors and ANS 15.4, 9 June 1988. Any deviations from the requirements of 10 CFR 55 are justified by the mode of operation and the unique design of the PULSTAR Reactor. More specifically, the conditions conducive to continued Reactor Operator and Senior Reactor Operator proficiency at the NCSU PULSTAR facility are as follows:

(1) Reactivity control manipulations are extremely frequent. Steady power reactor operations in excess of eight hours are uncommon. A startup checklist, reactor startup, power level change, and shutdown are normally performed three to five times a week.

(2) Frequent operations of the reactor for undergraduate teaching laboratories provides the operators with the review of the theory and practice of what might otherwise be uncommon reactor control manipulations.

(3) Formal Reactor Operator Training programs for prospective power plant reactor operators are routinely presented. These programs are presented by licensed PULSTAR staff and require proficiency in all areas of reactor operation. They cover a broad spectrum of experiments designed to bridge the gap between theory and practice. Numerous reactivity manipulations are performed and/or supervised by NCSU licensed personnel during these programs.

2.0 DEFINITIONS

2.1 Class A reactor operator: An individual who is certified to direct the activities of Class B reactor operators. Such an individual is also a reactor operator and is commonly referred to as Senior Reactor Operator (SRO).

2.2 Class B reactor operator: Any individual who is certified to manipulate the controls of the reactor. Such an individual is commonly referred to as Reactor Operator (RO).

2.3 Controls: Apparatus and mechanisms, the manipulation of which, directly affect the reactivity or power level of the reactor.

2.4 Designated Senior Reactor Operator: A specific individual who is assigned the duties of Senior Reactor Operator (DSRO) during reactor operations.

2.5 Lecture: A pre-planned discourse, demonstration, or film program that is presented by a qualified person.

2.6 On-the-Job Training (OJT): The experience gained while manipulating the controls of the reactor, or directing the activities of individuals during plant control manipulations.

3.0 REFERENCES

3.1 NCSU PULSTAR Safety Analysis Report, Section 11.2, 15 December 1971.

3.2 Code of Federal Regulations, 10 CFR 55, Operators' Licenses; 30 April 1994.

3.3 ANS 15.4, Standard for Selection and Training of Personnel for Research Reactors, 9 June 1988.

3.4 Operator Requalification Program, Revision 4, NCSU, PULSTAR Reactor, 23 January 1989.

3.5 ANS 15.1, Development of Technical Specifications for Research Reactors, 07 December 1990.

3.6 NUREG-1478 Non-Power Reactor Operator Licensing Examiner Standards, June 1994.

4.0 SCHEDULE

The Requalification Program shall be conducted over a period not to exceed 24 months, and will be followed by successive two-year programs. The contents of this program are lectures, quizzes, written examinations, and document review. All individuals will be on the same two-year cycle. New operators will join the program at the time of licensing.

5.0 LECTURES

5.1 Requalification lectures will be conducted approximately every two months. Subjects for these lectures will be determined by the overall performance on the last written examinations. Lectures which shall be included, but not limited to, over the two-year period are:

(1) Theory and principles of operation.

(2) General and specific reactor operating characteristics.

(3) Reactor instrumentation and control systems.

(4) Reactor protection systems.

(5) Engineering safety features.

(6) Normal, abnormal, and emergency operating procedures.

(7) Radiation control and safety.

(8) Technical specifications.

(9) Administrative controls.

(10) Applicable portions of Title 10, Chapter I, Code of Federal Regulations.

5.2 Formal research reactor training programs for prospective power plant reactor operators can be substituted for requalification lectures on a one-to-one basis, i.e., one training program can count as one requalification lecture.

5.3 Individual study may also substitute as a lecture; however, no more than two such study periods can be credited over a two year period.

5.4 All regularly scheduled lectures should be attended by all licensed operators. Those licensed operators who achieved an average score of greater than 80% on the last written examination may be excused from lecture attendance at the discretion of the Chief Reactor Operator. Passing an NRC licensing exam will also provide this exemption for the duration of the requalification two-year cycle.

5.5 Personnel missing lectures who were not excused shall be briefed on the topics, given a copy of the lesson plan (if applicable), and be required to take the quiz (if applicable) prior to the next scheduled requalification lecture.

6.0 QUIZZES

Written quizzes may be given on any topics covered by a lecture. The results of these quizzes will be used to evaluate the effectiveness of the given lectures, and to ascertain whether additional training is needed.

7.0 WRITTEN EXAMINATION

7.1 A written examination shall be taken at intervals not to exceed 24 months. The examination should be similar in content to the NRC licensing exams and shall cover the following categories:

(1) Reactor Theory, Thermodynamics and Facility Operating Characteristics (A)

(2) Normal and Emergency Operating Procedures and Radiological Procedures (B)

(3) Plant and Radiation Monitoring (C)

7.2 Personnel holding Reactor Operator licenses and personnel holding Senior Reactor Operator licenses shall be examined on the same categories. In addition, the SRO test shall contain questions covering the following areas: Radioactive Materials Handling Disposal and Hazards, Fuel Handling and Core Parameters, Administrative Procedures, Limiting Conditions For Operation. These additional questions will be included in the appropriate sections listed in 7.1. Written examination results will be documented on the Evaluation of Reactor Proficiency Form 2.6-4.

8.0 ON THE JOB TRAINING

8.1 The primary mechanism for on-the-job training is the routine operation of the PULSTAR Reactor. To maintain proficiency as a reactor operator each

licensed operator shall be required to annually perform at least ten reactivity manipulations in any combination of reactor startups and shutdowns and actively perform a minimum of four hours per calendar quarter on shift as RO. Direct supervision of the reactivity manipulations as DSRO shall be considered equivalent to actual performance.

To maintain proficiency as a senior reactor operator each SRO shall actively perform a minimum of four hours per calendar quarter as DSRO. In addition, each SRO shall be required to annually perform at least ten reactivity manipulations in any combination of reactor startups and shutdowns either as a RO or DSRO. Direct supervision of the reactivity manipulations as DSRO shall be considered equivalent to actual performance.

8.2 In addition to the requirements outlined in 8.1, annually at intervals not to exceed 15 months, each operator shall control the plant during the following:

A. Changes in power levels greater than 10 percent.

* B. Dropped rod.

* C. Rx SCRAM.

* D. Loss of primary coolant.

* E. Loss of Rx Air.

* F. Loss of primary coolant flow.

* G. Loss of a SCRAM logic channel.

* H. Stuck Rod.

* I. Fuel cladding failure or high coolant activity.

* J. Malfunction of the Auto channel.

* K. Loss of a Nuclear Instrumentation channel.

* L. Loss of electrical power.

* Since the primary means of protection for the PULSTAR is to either S/D or SCRAM, it is important for the operator to understand and recognize when either is required. These items will be performed as walk throughs with the Chief Reactor Operator concentrating on indications and actions taken. No operator action will actually be performed on the reactor console.

8.3 To ensure the completion of sections 8.1 and 8.2, the following forms will be used to record completion:

A. Form 2.6-1 Record of RO/SRO Functions

B. Form 2.6-2 Quarterly Record of RO/SRO Functions

C. Form 2.6-3 Annual Record of RO/SRO Functions

A suitable computer generated form may be substituted for each of the forms listed above.

To ensure compliance, these forms will be periodically reviewed by the Chief Reactor Operator.

9.0 ORAL AND DEMONSTRATIONAL

9.1 An annual oral and demonstrational examination shall be conducted for all licensed operators. The interval for this shall not exceed 15 months. Completion of this exam shall be documented on the Evaluation of Reactor Operator Proficiency Form 2.6-4.

This examination shall cover a representative sample from among the areas listed below:

1. Performing a startup checklist.
2. Properly controlling the PULSTAR between S/D and power operations.
3. Identify console annunciators and explain the proper response required for the annunciators.

4. Explain instrumentation on the console and when the instrumentation is required.
5. Properly controlling the Rx during abnormal and emergency conditions.
6. Shifting the electrical systems to the Auxiliary Generator.
7. Placing the Rx Building in the confinement mode.
8. Describe the function and proper operation of the Radiation Monitoring Rack, fixed, and portable radiation monitoring equipment.
9. Demonstrate knowledge of the radiation hazards associated with the PULSTAR, and the procedures used to reduce excessive exposure.
10. Demonstrate appropriate knowledge, based upon license held, of the Emergency Plan including when and if the Emergency Plan should be executed.
11. Demonstrate the ability to safely operate the PULSTAR and work within the organization structure to insure that license limits are not exceeded.

The examination will be open reference and comprised of five tasks which will be used to evaluate the operators ability to satisfactorily operate the facility. An RO is responsible for only RO tasks; an SRO is responsible for all tasks. Should an operator incorrectly perform or fail to perform a critical step, the task may be graded as unsatisfactory if the deficiency jeopardizes the safety of the facility or has significant safety impact on the public. Failure to perform, or incorrectly perform two or more critical steps shall result in failure of the task. Critical steps are those actions which, when not performed correctly or not performed at all, would prevent the system from operating safely or prevent the completion of an essential safety action.

For tasks for which a written procedure exists, it is acceptable to use a copy of the procedure and mark those steps considered critical. If no written procedure exists, a written description of the task, list of expected actions, and critical steps will be generated for the examination. Abnormal and emergency conditions will be simulated and normally the operator will simulate the actions that are taken when responding to these simulated conditions.

At least one of the five tasks must involve a reactivity manipulation. This operation may be part of an overall evolution including more than one task. For example, an operator may perform a reactor startup as the first task, then response to a simulated instrument failure as the second task.

At least one of the tasks performed must involve a response to an emergency situation. At least one of the tasks performed must involve a response to an abnormal event, and at least one of the tasks performed must be performed in the facility. An example of this would be the completion of the bay portions of a startup checklist.

At least two questions to ask each operator will be developed for each of the five tasks. These questions should be based on the task or system being operated and, where appropriate, discriminate between RO/SRO responsibilities. Also, these questions should emphasize knowledge required for task performance or procedure implementations and compliance.

10.0 <u>DOCUMENT REVIEW</u>

10.1 All licensed personnel shall review the contents of all abnormal and emergency procedures semiannually, with intervals not to exceed 7.5 months.

10.2 All significant facility license, technical specifications, design and procedural changes shall be routed to all licensed operators in a timely

manner. The initialed Reactor Operator Routing Sheet, Form 2.6-5, will be kept on file as a record of these reviews.

10.3 All licensed personnel shall review the contents of the PULSTAR Operations Manual annually, but at intervals not to exceed 15 months. Changes in the Operations Manual shall be reviewed prior to an individual's operation of the Reactor.

11.0 EVALUATION

11.1 Persons achieving a score of less then 70% on any area of the written examination will receive additional tutoring in weak areas followed by a re-examination in those areas. An overall score of less the 70% will be considered a failure. In this case, the individual(s) involved will be relieved from all licensed activities and will receive retraining which will be documented on the Reactor Operator Retraining Form 2.6-6. Satisfactory completion of a re-examination will be required prior to resumption of licensed activities. Documentation of this re-examination will be entered into the individual's requalification record on the Reactor Operator Retraining Form 2.6-6.

11.2 To satisfactorily complete the Oral/Demonstrational exam each examinee must satisfactorily complete at least 4 of the 5 tasks on the exam. Also the examinee must complete 70% of the task system/procedure questions. If an individual does not satisfactorily complete the Oral/Demonstrational exam, the individual(s) involved will be relieved from all licensed activities and will receive retraining which will be documented on the Reactor Operator Retraining Form 2.6-6. Satisfactory completion of a reexamination will be required prior

to resumption of licensed activities. Documentation of this reexamination will be entered into the individual's requalification record on the Reactor Operator Retraining Form 2.6-6.

12.0 ABSENCE FROM LICENSED ACTIVITIES

Before resumption of licensed activities, an individual who has not been actively performing licensed functions for a minimum of four hours per calendar quarter shall be required to demonstrate to the Chief Reactor Operator (or designated assistant) that their knowledge and understanding of the operation and administration of the facility is satisfactory. This may be accomplished through written, oral, or operational evaluation or a suitable combination. A minimum of six hours under the direction of a licensed operator or senior operator as appropriate and in the position to which the individual will be assigned must be completed prior to returning to active status. Any deficiencies revealed must be corrected before the individual resumes licensed functions. This demonstration will be documented on the Re-establishment of Reactor Operator/Senior Reactor Operator Proficiency Form 2.6-7, and maintained in the individual's requalification records.

13.0 EXEMPTIONS

The Chief Reactor Operator (Program Administrator) is exempt from the written, oral, and demonstrational examinations. Personnel who are not going to be performing licensed activities over the entire two year period are only required to participate in the requalification program during the time they are performing licensed activities. This exemption would normally apply to student operators who will graduate prior to completion of the two year period.

14.0 RECORDS

The Chief Reactor Operator shall establish and maintain requalification records on all personnel involved in this program. The latter shall include as a minimum, copies of individual written examinations and results, results of oral and demonstrational examinations, evidence of participation in requalification lectures, number of reactivity control manipulations performed, documentation of any additional training administered in areas in which an operator or senior operator has exhibited deficiencies, and the CRO's annual evaluation of the operator.

15.0 ADMINISTRATION

The NCSU Operator Requalification Program will be administered by the Chief Reactor Operator (CRO) under the normal supervision of the Reactor Operations Manager (ROM) and NRP Associate Director (AD). The CRO will be responsible for all testing and scheduling and will maintain complete requalification records.

16.0 APPROVALS:

Reactor Operations Manager: [signature]

Date: 17 Feb 95

Associate Director: [signature]

Date: 17 Feb 95

Radiation Protection Committee: [signature]

Date: 4/24/95

FORM 2.6-1
Record of RO/SRO Functions

	S/U Checks			S/U Performed		S/U	Sample	Shut-	R.O.	S.R.O.	
Date	L.F.	S.K.	K.O.	Routine	T/A	Super	Changes	downs	Hours	Hours	Other
Carry Fwd.											
Total as of											

FORM 2.6-2
Quarterly Record of RO/SRO Functions

Qtr.	S/U Checks			S/U Performed		S/U	Sample	Shut-	R.O.	S.R.O.	Other
	L.F.	S.K.	K.O.	Routine	T/A	Super	Changes	downs	Hours	Hours	
1.											
2.											
3.											
4.											
Total											

FORM 2.6-3
Annual Record of RO/SRO Functions

	Date Completed	Sat/Unsat	CRO initial
A. Changes in power levels > 10 percent.			
*B. Dropped rod.			
*C. Rx SCRAM.			
*D. Loss of primary coolant.			
*E. Loss of Rx Air.			
*F. Loss of primary coolant flow.			
*G. Loss of a SCRAM logic channel.			
*H. Stuck Rod.			
*I. Fuel cladding failure high coolant activity.			
*J. Malfunction of the Auto channel.			
*K. Loss of a Nuclear Instrumentation channel.			
*L. Loss of electrical power.			

* Since the primary means of protection for the PULSTAR is to either S/D or SCRAM, it is important for the operator to understand and recognize when either is required. These items will be performed as walk throughs with the Chief Reactor Operator concentrating on indications and actions taken. <u>No operator action will actually be performed on the reactor console</u>.

Form 2.6-4
Evaluation of Reactor Proficiency

Name____________________ Requal Cycle__________

R.O./S.R.O. License Number__________ Issue Date__________

1. Summary of O.J.T.

Position(s) Held____________________

Totals from O.J.T. Log:

	S/U Checks			S.U. Performed		S/U	Sample	Shut-	R.O.	S.R.O.
Year	L.F.	S.F.	K.O.	Routine	T/A	Super	Changes	downs	Hours	Hours

2. Summary of Operator Requalification Program Written Test Results for Past Calendar Year.

Section	Points Possible	Points Received	Per Cent
A	______	______	______
B	______	______	______
C	______	______	______
TOTAL	______	______	______

Comments: ____________________

Form 2.6-4 (continued)
Evaluation of Reactor Proficiency

Name: ______________________

3. Results of Oral and Demonstrational Examinations

Date: ______________________ Examiner: ______________________

Task #
____

	Task Sat/Unsat
1. ______________________	____________
2. ______________________	____________
3. ______________________	____________
4. ______________________	____________
5. ______________________	____________
4 of 5 Tasks completed satisfactory (Yes/No)	____________

Task System/Procedure Questions.

	Number of Questions answered correctly
1. Task #1 Questions	____________
2. Task #2 Questions	____________
3. Task #3 Questions	____________
4. Task #4 Questions	____________
5. Task #5 Questions	____________
Final Task Question Grade (minimum 70%)	____________
Overall Final Grade (Pass/Fail)	____________

Form 2.6-5
REACTOR OPERATING ROUTING SHEET

SUBJECT: ____

____

DATE INITIATED: ______________________________________

Knowledge of the attached material is essential to your effectiveness as a licensed reactor operator. Please review promptly. Initial and pass on.

RO/SRO/ROA	INITIAL	DATE

REMARKS: ____

____

____

____

RETURN TO: ____________________________

FORM 2.6-6

Reactor Operator Retraining

NAME: ____________________

DATE RELIEVED FROM LICENSED ACTIVITIES: ____________________

EXAM DATE: ____________________

I. Summary of deficient areas (give grades and topics)

II. Schedule of retraining (give dates and topics)

III. Results of re-examination

Date Returned to Licensed Activities: ____________________

CRO Approval: ____________________

FORM 2.6-7

Re-establishment of

Reactor Operator/Senior Reactor Operator Proficiency

NAME: ______________________________

Date: ______________________________

Period of Absence From Licensed Activity: ____________________

Requirements for re-establishing proficiency

A. Review of Documents completed __________

B. Startup Checklist completed __________

C. Reactor Startup completed __________

D. Oral Evaluation completed __________

E. Under Instruction Watch completed __________
(Six hours minimum)

Verification of Proficiency Re-established

Date returned to Licensed Activities: ____________________

CRO: ____________________