SAFETY ANALYSIS REPORT FOR THE 5 MW GEORGIA TECH RESEARCH REACTOR

GEORGIA INSTITUTE OF TECHNOLOGY
ATLANTA, GA 30332-0425

Submitted

APRIL, 1994

Revised

JANUARY 10, 1995

TABLE OF CONTENTS

Table of Contents (cont)

Table of Contents (cont)

Table of Contents (cont)

Table of Contents (cont)

Table of Contents (cont)

TABLES

ILLUSTRATIONS

Figure Number

Illustrations (cont)

I. INTRODUCTION

The Georgia Tech Research Reactor (GTRR) is a heterogeneous, heavy-water moderated and cooled reactor, fueled with highly enriched uranium (HEU) plates comprised of aluminum-uranium alloy, or low enriched uranium (LEU) plates comprised of aluminum - U_3Si_2 alloy. It is designed to produce a thermal flux of more than 10^{14} n/cm^2/sec at a power of 5 MW. The reactor was licensed on December 29, 1964 to operate at one megawatt. On June 6, 1974 the license was amended and the maximum power of the GTRR was increased from one to five megawatts.

Over the years, fuel performance has been satisfactory with no known problems. Engineered safety systems have performed adequately and as intended. No safety problems have been encountered.

Recently the cooling tower was replaced and several upgrades in instruments such as picoammeters and temperature recording devices were implemented.

An application to the NRC to amend the GTRR license to replace high-enriched uranium fuel with low-enriched uranium was submitted January 21, 1993. Currently the NRC is reviewing this request.

2. SUMMARY

2.1 General

The Georgia Tech Research Reactor (GTRR) is located on the campus of the Georgia Institute of Technology, approximately two miles from the center of downtown Atlanta. The two-acre site is on the north end of the campus. The Georgia Tech student body exceeds 13,000. The campus is surrounded by a residential and commercial area. Approximately 30,000 people live within one mile of the site.

The reactor core is approximately 2 feet in diameter, 2 feet high and, when fully loaded, contains provisions for up to 19 fuel assemblies spaced 6 inches apart in a triangular array. The fuel is centrally located in a 6 foot diameter aluminum reactor vessel which provides a 2 foot thick D_2O reflector completely surrounding the core. A cutaway perspective view of the reactor is shown in Figure 2.1.

The reactor vessel is mounted on a steel support structure and is suspended within a thick-walled graphite cup. The graphite provides an additional 2 feet of reflector both radially and beneath the vessel. The core and reflector system is completely enclosed by the lead and concrete biological shield.

The reactor is controlled by means of four cadmium shim-safety blades and one cadmium regulating rod. The four shim-safety blades are mounted at the top of

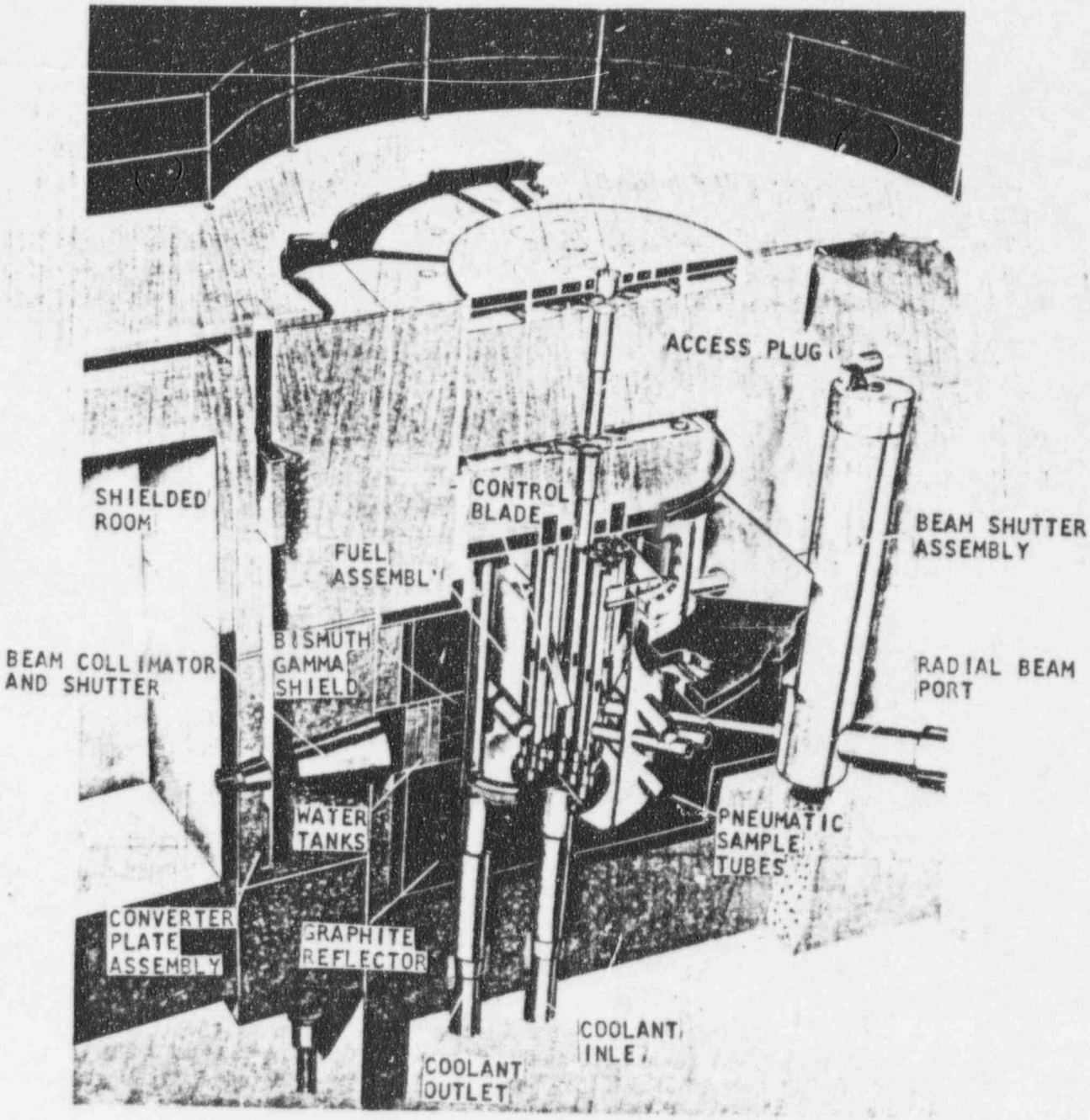


Figure 2.1. Cutaway Perspective View of the GTRR.

the reactor vessel and swing downward through the core between adjacent rows of fuel assemblies. The regulating rod is supported on the reactor top shield and extends downward into the radial D_2O reflector region. This rod moves vertically between the horizontal midplane and the top of the core.

The reactor is provided with a heat removal system, D_2O purification system, shield cooling system, D_2O storage system, radiolytic gas recombination system, and a ventilating system. The heat removal system is composed of a primary heavy water system and a secondary light water system.

The heavy water system includes the reactor vessel, the primary D_2O coolant pumps, the D_2O makeup pump, the heat exchangers and the associated valves and piping. All components in contact with the D_2O are fabricated of stainless steel or aluminum. The light-water secondary system is comprised of the circulating water pumps, the cooling tower, and the associated valves and piping. The secondary coolant system is fabricated primarily of carbon steel.

Since the GTRR is intended for research applications, it is provided with a variety of experimental facilities which will permit a wide range of research investigations. Experiments requiring high intensity neutron or gamma-ray beams can be accommodated as well as those requiring a uniform thermal neutron flux throughout a large volume. The tangent through-tubes are particularly well suited for engineering tests requiring the circulation of a coolant. Irradiations of short duration and requiring rapid sample recovery can also be accomplished. Further discussions of these and other applications are contained in Section 4. The design

includes ten horizontal beam tubes, two horizontal tangent through-tubes, two horizontal pneumatic shuttles, two horizontal irradiation tunnels, twenty vertical irradiation thimbles and two vertical fast flux tubes. In addition, the reactor face contains a thermal column and a bio-medical irradiation facility. Detailed specifications of the various components are contained in Section 4.4.6.

The reactor and associated systems are housed within a steel containment building eighty feet in internal diameter. In order to reduce the probability of significant environmental contamination following a release of radioactivity within the building, the containment shell is designed to restrict leakage of its contents to less than 1/2% of its volume per day per psi overpressure. Repeated tests have shown the actual leak rate to be less than half of the design value. To reduce the direct radiation exposure of people outside the building following such an unlikely event as a fission-product release within the containment shell, the shielding capacity of the steel walls is supplemented by the addition of 12 inches of concrete.

The containment building has three levels. The basement contains process and ventilating equipment and space for experimental equipment. The main floor is largely unobstructed and provides space for installation of experiments. The control room is located at the level of the top of the biological shield. The main floor and reactor top are serviced by a twenty-ton capacity polar crane. When the reactor is not operating, access to the building is permitted through a large truck entrance. During operation, access is restricted to an air lock connected to the adjoining laboratory building, and to an air lock leading to the outside.

Among the facilities in the adjoining 24,000 square foot, two-story air conditioned laboratory building are the following:

Two hot cells equipped with master-slave manipulators.

Fuel element storage and handling pool connected to one of the hot cells.

Radiochemistry laboratory containing two radioisotope hoods.

Decontamination and active waste packaging room.

Change room isolating the above facilities from the remainder of the building.

Counting rooms.

Laboratories for low level chemistry, health physics, nuclear chemistry, metallurgy, physics, radiobiology, electronics and biochemistry.

Facilities for disposal of liquid radioactive wastes.

Dark room.

Machine shop.

Animal quarters.

Viewing gallery which permits visitors to observe activities within the reactor building and hot cell service area without actually entering either area.

2.2 Reactor Design Data

The important reactor design characteristics are contained in Table 2.1 for highly enriched uranium plates. The design characteristics for low enriched core are listed in Table 2.1.1.

TABLE 2.1

REACTOR DESIGN DATA FOR HIGHLY ENRICHED URANIUM CORE

Reactor

Type	Heterogenous, D_20 moderated and cooled, highly enricned
Thermal power (MW)	5
Operating pressure (psia)	15
Reactor Outlet temperature, moderator (°F)	132
Active core volume (ft^3)	7.3
Length (ft - in)	2-0
Equivalent diameter (ft - in)	2-4
Power density, average core (kw/l)	24.3
Power density, average moderator (kw/l)	26.6
Power density, average coolant (kw/l)	171
Specific power, average (kw/kg U-235)	1660
Fuel	Uranium-aluminum alloy
U-235 content (kg)	3.01
Cladding	Aluminum
Volume composition of active core	
Uranium (%)	0.076
Aluminum (%)	8.83
D_2 (%)	91.09
Weight composition of active core	
U-235 fuel (kg)	3.01
D_2O (lbs)	462
Aluminum (lbs)	110
Fuel assemblies	
Number in reference design core	16
Coolant flow area per assembly (ft^2)	.0322
Total U-235 per assembly (grams)	188

Fuel plates

Number per assembly	16
Plate width, overall (in)	2.854
Plate thickness, overall (in)	0.050
Plate length, overall (in)	25
Plate length, fuel (in)	23.5
Face clad thickness (in)	0.015
Edge clad thickness (in)	0.204

Shim-safety blades

Number in core	4
Shape	Rectangular
Dimensions (in)	5.5 x 1 x 45.5
Composition	Aluminum-clad cadmium

Regulating rod

Number in core	1
Shape	Tubular
Dimensions (in)	1.38 I.D. x 1.42 O.D. x 24 long
Composition	Aluminum-clad cadmium

Coolant flow

Total flow area in core (ft^2)	0.515
Total weight flow entering core (lbs/hr)	982,000
Total volume flow entering core (gpm)	1,800
Inlet velocity, average coolant within assembly (ft/sec)	7.8
Inlet velocity, maximum coolant within assembly (ft/sec)	8.6

Temperatures (°F)

Coolant entering core	114
Coolant leaving core	132
Plate surface, average	153
Plate surface, maximum	193
Fuel centerline, average	160
Fuel centerline, maximum	211

Heat transfer

Area in core (ft^2)	209
Heat flux, average (Btu/ft^2-hr)	78,450
Heat flux, maximum (Btu/ft^2 -hr)	191,000
Thermal conductivity, U-Al (Btu/hr-ft-°F)	110
Thermal conductivity, Al (Btu/hr-ft-°F)	125

Reactor Vessel

Design pressure (psig)	9
Design temperature (°F)	150
Diameter (nominal outside diameter, ft-in)	6-0
Diameter (maximum at opening, ft-in)	6-6
Height of vessel (ft-in)	10-4
Wall thickness (base metal, nominal, in)	0.375
Composition	Type 1100 A1
Design strength at 150°F (psig)	2350
Lower head	
Shape	Dished
Thickness (in)	0.50

Nozzles

Coolant outlet nozzle	1 - 10 in
Coolant inlet nozzle	1 - 10 in
Liquid level	1 - 3 in
Experimental facilities	3 - 6 in
	8 - 4 in
	1 - 2 1/2 in x 5 in (round)
	1 -2 in x 6 in (rectangular)
Over-pressure relief vent	1 - 6 in
Shim-safety drives	4 - 3 in

Approximate vessel weight (lbs) 2000

Shielding

Thermal shielding (in)	0.25 boral plus 3.5 lead
Annular concrete shield (ft - in)	4 - 9

Reactor Containment Building

Shape	Cylindrical with torispherical top and flat bottom
Shell diameter, inside (ft - in)	82 - 2
Shell composition	ASTM-A201 Grade B
Shell thickness (in)	Bottom and sides 7/16
	Top side 1-3/4
	Top center 5/8
Maximum expected pressure (psig)	2.1

Design pressure (psig)	2.0
Safety factor	3
Test pressure (psig)	2.0
Maximum expected temperature at 2.1 psig (°F)	109
Air locks	2
Truck door	1

Physics

Coolant void coefficient of reactivity, core center (%/cc)	-3.4×10^{-4}
Void coefficient of reactivity, core average (%/c c)	-2.6×10^{-4}
Temperature coefficient of reactivity, (%/° C)	-0.02
Reactivity, cold to hot (%)	0.5
Reactivity, Xe plus Sm (%)	3.4
Reactivity, experiments (%)	2.0
Reactivity, burnup allowance (%)	3.6
Reactivity, control allowance (%)	2.4
Maximum reactivity to be controlled (%)	11.9
Reactivity controllable by 4 shim-safety blades (%)	25

Max/avg thermal flux radial (5 MW calculated, 16 assemblies)	1.26
Max/avg thermal flux, axial (1 MW experimental, 13 assemblies)	1.41

TABLE 2.1.1

REACTOR DESIGN DATA FOR LOW ENRICHMENT URANIUM CORE

Reactor Type	Heterogeneous, D_2O moderated and cooled, low enriched
Thermal power (MW)	5
Operating pressure (psia)	15
Reactor Outlet temperature, moderator (°F)	131
Active core volume (ft^3)	
Length (ft - in)	2-0
Equivalent diameter (ft - in)	2-4
Power density, average core (kw/l)	24.2
Power density, average moderator (kw/l)	32.8
Power density, average coolant (kw/l)	173.7
Specific power, average (kw/kg U-235)	1389
Fuel	U_3Si_2
U-235 content (kg)	3.6
Cladding	Aluminum
Volume composition of active core	
U_3Si_2	2.97
Aluminum (%)	9.20
D_2O coolant	14.00
D_2O moderator	74.03
Weight composition of active core	
U-235 fuel (kg)	3.6
D_2O (lbs)	422
Aluminum (lbs)	107
Fuel assemblies	
Number in reference design core	16
Coolant flow area per assembly (ft^2)	0.0302
Total U-235 per assembly (grams)	225

Fuel plates

Number per assembly	18
Plate width, overall (in)	2.854
Plate thickness, overall (in)	0.050
Plate length, overall (in)	25
Plate length, fuel (in)	23.5
Face clad thickness (in)	0.015
Edge clad thickness (in)	0.204

Shim-safety blades

Number in core	4
Shape	Rectangular
Dimensions (in)	5.5 x 1 x 45.5
Composition	Aluminum-clad cadmium

Regulating rod

Number in core	1
Shape	Tubular
Dimensions (in)	1.38 I.D. x 1.42 O.D. x 24 long
Composition	Aluminum-clad cadmium

Coolant flow

Total flow area in core (ft^2)	0.483
Total weight flow entering core (lbs/hr)	998,395
Total volume flow entering core (gpm)	1,800
Inlet velocity, average coolant within assembly (ft/sec)	9.45

Temperatures (°F)

Coolant entering core	114
Coolant leaving core	131
Plate surface, average	163
Plate surface, maximum	178
Fuel centerline, average	165

Heat transfer

Area in core (ft^2)	229
Heat flux, average (Btu/ft^2-hr)	71,600
Thermal conductivity, U_3Si_2 (Btu/hr-ft-°F)	51.99
Thermal conductivity, Al (Btu/hr-ft-°F)	104

Reactor Vessel

Design pressure (psig)	9
Design temperature (°F)	150
Diameter (nominal outside diameter, ft-in)	6-0
Diameter (maximum at opening, ft-in	6-6
Height of vessel (ft-in)	10-4
Wall thickness (base metal, nominal, in)	0.375
Composition	Type 1100 A1
Design strength at 150°F (psig)	2350

Lower Head

Shape	Dished
Thickness (in)	0.50

Nozzles

Coolant outlet nozzle	1 - 10 in
Coolant inlet nozzle	1 - 10 in
Liquid level	1 - 3 in
Experimental facilities	3 - 6 in
	8 - 4 in
	1 - 2 1/2 in x 5 in (round)
	1 - 2 in x 6 in (rectangular)
Over-pressure relief vent	1 - 6 in
Shim-safety drives	4 - 3 in

Approximate vessel weight (lbs)	2000

Shielding

Thermal shielding (in)	0.25 boral plus 3.5 lead
Annular concrete shield (ft - in)	4 - 9

Reactor Containment Building

Shape	Cylindrical with torispherical top and flat bottom
Shell diameter, inside (ft - in)	82 - 2
Shell composition	ASTM-A201 Grade B
Shell thickness (in)	Bottom and sides 7/16
	Top side 1 - 3/4
	Top center 5/8
Maximum expected pressure (psig)	2.1
Design pressure (psig)	2.0

Safety factor	3
Test pressure (psig)	2.0
Maximum expected temperature at 2.1 psig (°F)	109
Air locks	2
Truck door	1

Physics

Coolant void coefficient of reactivity, core center %(k/k)/% void	-3.3×10^{-2}
Temperature coefficient of reactivity % (k/k)/°C	-2.3×10^{-2}
Reactivity, cold to hot (%)	0.3
Reactivity, Xe plus Sm (%)	-3.8
Reactivity, experiments (%)	2.0
Reactivity, burnup allowance (%)	3.6
Reactivity, control allowance (%)	2.4
Maximum reactivity to be controlled (%)	11.9
Reactivity controllable by 4 shim-safety blades (%)	25

2.3 Reactor Safety

The GTRR is a heavy-water moderated reactor similar in design to the MIT reactor and the CP-5 reactor which was located at Argonne National Laboratory. As such, it has all of the inherent safety characteristics of water-moderated reactors in general and those of the CP-5 and MIT types in particular. Chief among these are negative moderator void and temperature coefficients which enable the reactor to absorb reactivity additions by consequent changes in the moderator density. This ability provides the reactor with a self-limiting mechanism which tends to stabilize power in the event of unusual and possibly hazardous operating difficulties.

Abnormal operating conditions can arise as a result of physical plant malfunctions or operator errors. The first category includes pump failures, loss of electrical power, instrumentation failures, safety or regulating control element failures and loss-of-coolant accidents. In all of these situations, a reactor scram will be initiated by no less than two, and possibly as many as ten, separate and independent trip circuits. Decay heat is removed satisfactorily by convection circulation of D_2O after a scram from power or pump failure. Increased decay heat following 5 MW operation is removed through an emergency cooling system described in Section 4.4.8.3.

Errors of omission or commission made by the operating staff may impose sudden additions of reactivity upon the reactor. These errors could include improper fuel loading, improper startup, or failure to replace the graphite plugs in idle experimental facilities (which might lead to a sudden reduction in moderator void volume). The effect of sudden moderator changes are covered in Section 8.4.3. In all cases, reactivity additions of the magnitudes estimated for the foregoing cases, if occurring rapidly, will result in an abnormally short reactor period leading to a scram condition.

The 5 MW reactor instrumentation includes redundant safety interlock circuits on channels deemed to be crucial to reactor safety. Two independent electronic scram circuits and an electromechanical backup scram circuit are provided for reactor power, reactor period, D_2O temperature, D_2O flow, and D_2O level in the core tank. Even if the virtually impossible simultaneous failure of all circuits should occur, the inherent characteristics of negative void and temperature coefficients are expected to terminate the resultant excursion well below the melting point of the fuel.

In the event of the ultimate failure, fuel melting, a release of fission products will take place. The surrounding environs and populace are protected from exposure to high concentrations of radioactivity by the steel containment building enclosing the reactor. The leak-tightness of the containment building, as established by repeated testing, would prevent serious outleakage of volatile fission products. A 12-inch thick concrete shield around the periphery of the building would reduce the direct radiation hazard caused by the contained fission products. These items are discussed in Section 4.3 and Appendix C.

Reactor safety is further enhanced by the clear definition of responsibilities and of accident procedures for all persons at the Nuclear Research Center. Required classroom instruction and testing administered by the Health Physics group are described in Section 6.4. Areas of responsibility and procedures for experiment approval are unambiguously defined. (Section 6).

3. SITE CHARACTERISTICS

3.1 Geography and Demography

The Georgia Tech Research Reactor is located on the 330-acre campus of the Georgia Institute of Technology. The campus lies in a residential and commercial area just north of the center of downtown Atlanta. The location of the reactor is illustrated in Figures 3.1 - 3.3.

Over 13,000 students are enrolled at Georgia Tech. The Institute employs approximately 5,000 faculty and staff. The City of Atlanta, according to the 1990 census, has a population of 394,017 and covers 323.4 km^2. This represents a reduction in population of approximately 40,000 people since the reactor was constructed. In 1990, the 20-county Atlanta metropolitan area had a population of 2.8 million people and encompassed an area of 13,264.7 km^2. The metropolitan area is now in excess of 3.1 million people. The eastern boundary of the Georgia Tech campus coincides with the combined leg of I-75 and I-85, the major traffic arteries which run north to south through Atlanta (see Fig. 3.2).

The 1990 census tracts which include the campus and its immediate surroundings are shown in Figure 3.4. Most of the campus is encompassed by tract 10.95, see Figure 3.3. The populations and areas of each of these tracts are given in Table 3.1.

3.2 Hydrology and Geology

Atlanta is located in the foothills of the southern Appalachians in north central Georgia. The terrain is rolling to hilly and slopes downward to the east, west, and south so that the drainage of the major river systems is generally into the Gulf of Mexico from the western and southern sections of the city and to the Atlantic from the eastern portions of the city.

Atlanta is situated in the geographic province called the Piedmont Plateau. It has a moderately strong relief. In localized areas, the surface is rugged and extremely hilly. The Atlanta area is characterized by very "choppy" terrain with a fairly uniform elevation of hilltops ranging from 650 feet to 1050 feet elevation above

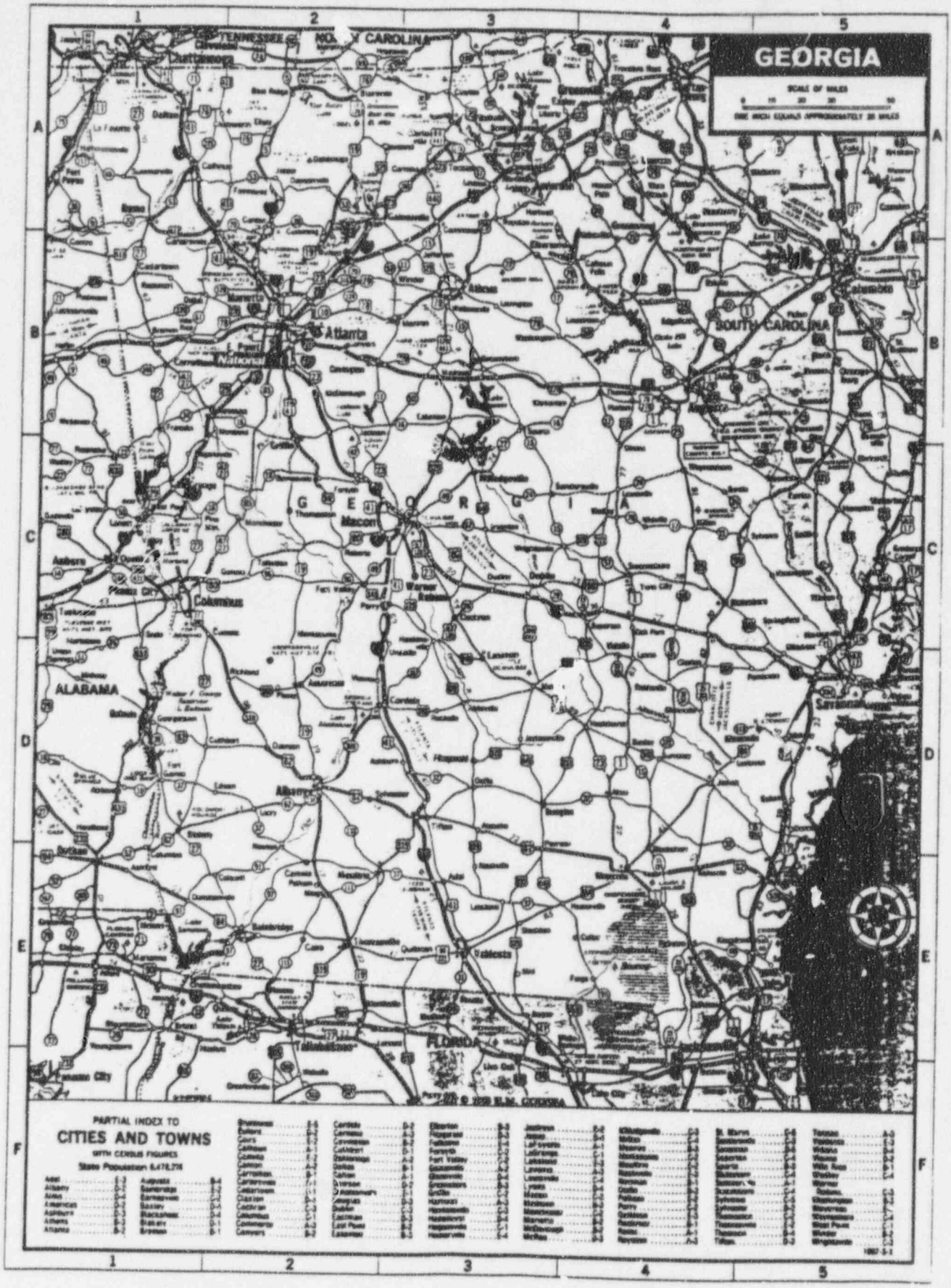


Figure 3.1 Map of the State of Georgia.

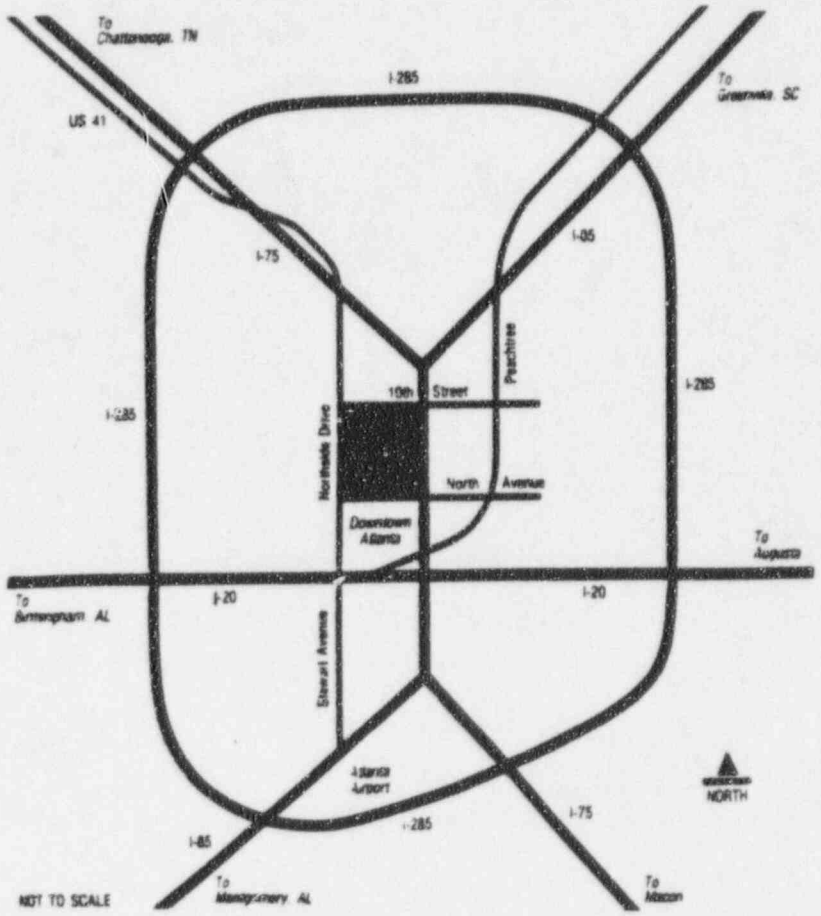


Figure 3.2 Map Indicating the Location of Georgia Tech Campus with Respect to Major Traffic Arteries in Atlanta.

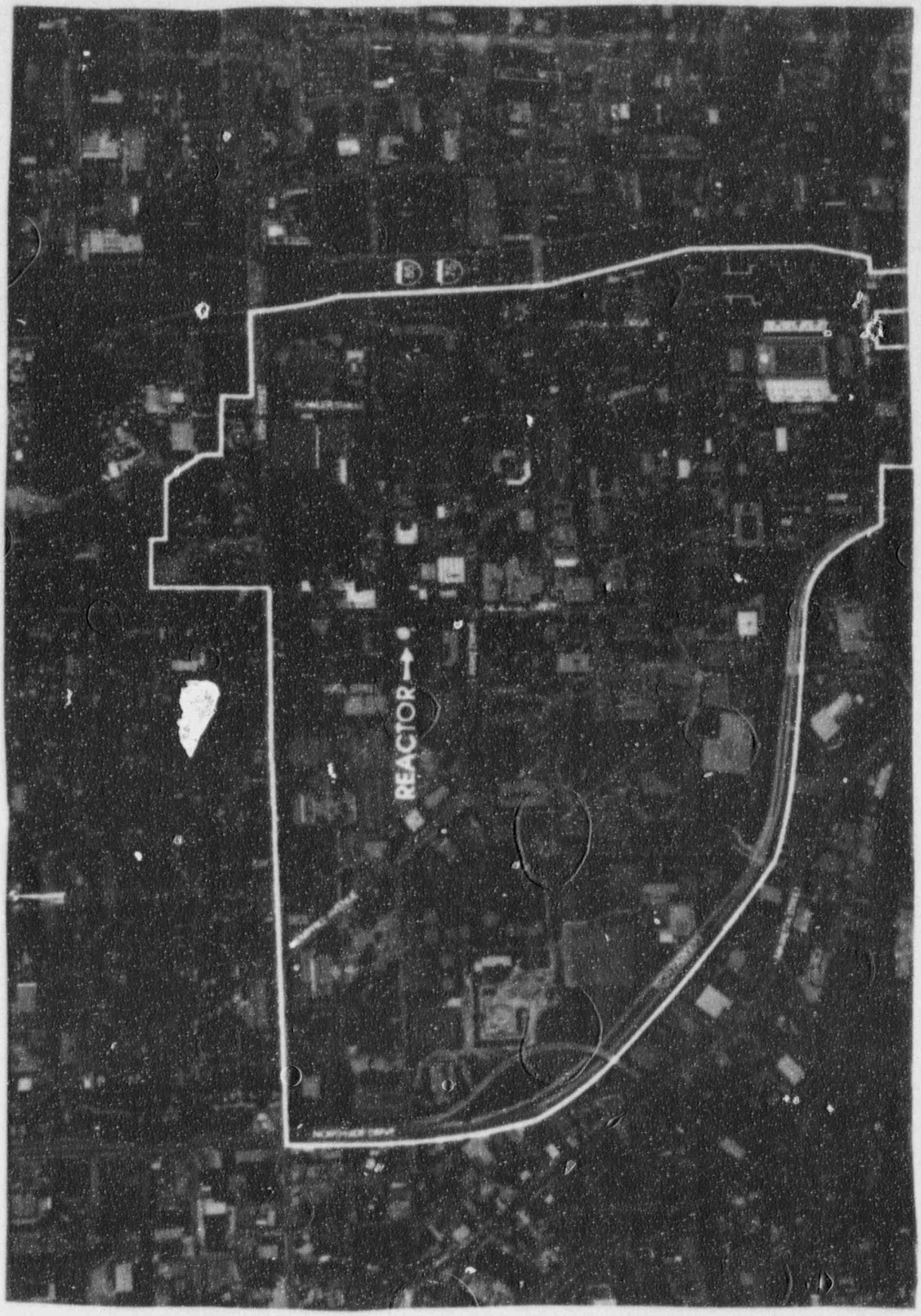


Figure 3.3 Aerial of the Reactor Facility and Vicinity.

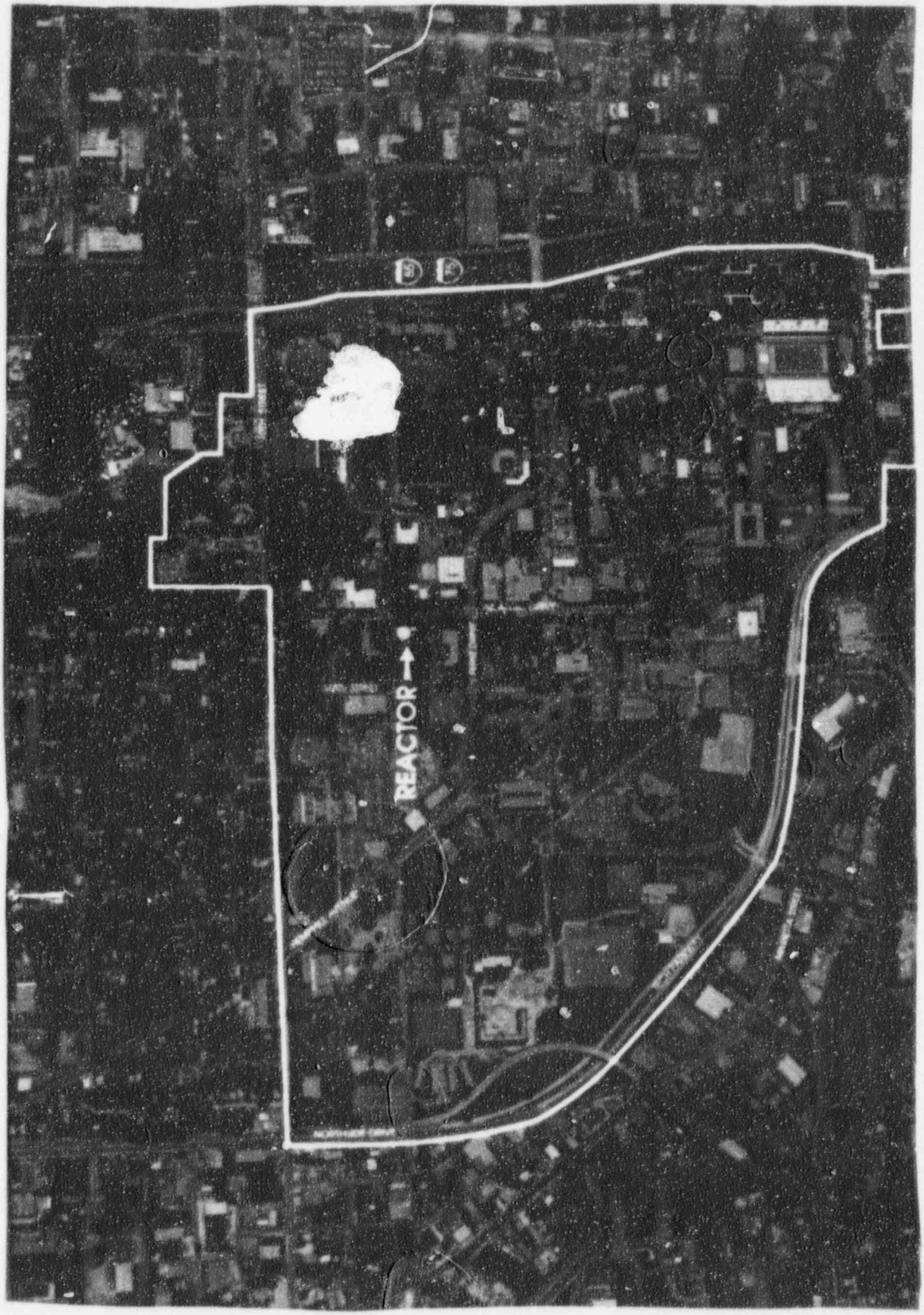


Figure 3.3 Aerial of the Reactor Facility and Vicinity.

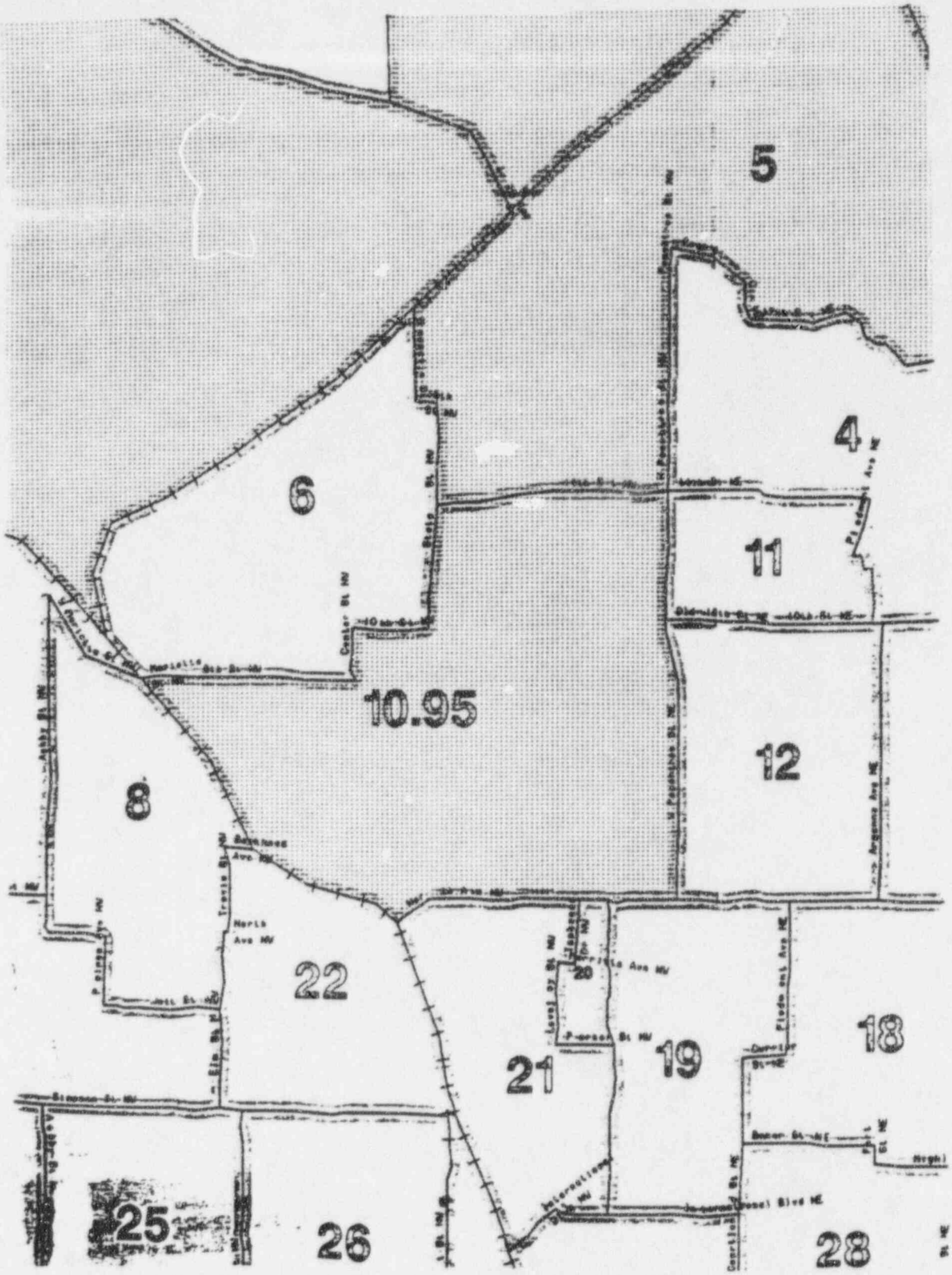


Figure 3.4 1990 Census Tracts in the Vicinity of the Georgia Tech Reactor

mean sea level. The hillsides are steep and meet to form sinuous draws or gullies. The lower and larger draws contain perennial stream channels.

The Chattahoochee River heads in the Appalachian Mountains in northeast Georgia and flows southwesterly in a narrow valley, passing 4.6 miles to the northwest of the reactor site. Its drainage area at Atlanta is approximately 1,450 mi^2. The flow of the Chattahoochee has been regulated by Lake Sidney Lanier since January 1956. The average discharge for the 57 years of recorded data is 2,547 cfs.

At the northwest corner of the reactor facility is a topographic depression. (See Figure 4.3 for detail). This depression is the upper end of a draw running southwest to northeast, which formerly drained in this area. The draw is now blocked off by Atlantic Drive. Surface drainage at the site concentrates in the depression west of Atlantic Drive. It is confined to runoff from the city block which contains the site, an area of 10 acres. The streets have low curbs which normally prevent inflow of surface runoff from outside this area. Catch basins in State Street and Atlantic Drive at the low points west and northeast of the facility, respectively, receive the runoff in the streets.

Underlying the gully and the topographic depressions is a 72-inch, concrete-pipe storm sewer . It receives runoff from the catch basins in Atlantic Drive and in the east depression. The sewer follows the natural drainage, running roughly 1200 feet east of Atlantic Drive to a trunk sewer.

The rocks underlying the Atlanta area are a crystalline complex of igneous and greatly altered metamorphic rocks. The latter are chiefly biotite gneiss and muscovite schist. The host rock is intricately intermingled with granite, chiefly, and also with hornblende gneiss, pyroxenite, and pegmatite. Igneous intrusions conform to the foliation of the older rocks.

Structural planes in the bedrock, along with openings capable of transmitting water which exist or are eventually developed, are due to folding, faulting, intrusions, and jointing and other fracturing. The first is the most prominent type of discontinuity in Georgia. They have a northeast trend and, southeast of the Chattahoochee River, a gentle, southeast dip. All rocks in the Atlanta area display some kind of fracturing.

Overlying the bedrock and weathered from it is a layer of mantle rock. Its profile is characteristic of the Piedmont region. The surface stratum consists of several feet of red clay which may include angular quartz particles, weathered mica, kaolinitic clays, and iron oxides. The second stratum consists of several feet of red clay which may include angular quartz particles, weathered mica, kaolinitic clays, and iron oxides. The second stratum is more granular soil which is graded from sandy silt at the top to silty sand in the lower part. The third stratum, which lies just above the bedrock, is a layer of extremely variable thickness and composition. It contains strong, hard rock which is fractured in irregular lenses. The variability of this layer is due largely to preferential weathering along the faults and joints. In the Atlanta area the mantle rock has a maximum thickness of 130 feet and an average thickness of about 55 feet.

Alluvium of sand and gravel is present only along the Chattahoochee River and its main tributaries. Its maximum thickness is 30 feet. Borings for soil exploration at the reactor site in the summer of 1958 revealed considerable variation in the elevation of the ground water table, 870 to 901 feet elevation above mean sea level with depths ranging from 11 to 40 feet. This was attributed to the presence of fill over an older ground surface, and to old drainage paths across the reactor site.

3.3 Meteorology and Climatology

The Gulf of Mexico and the Atlantic Ocean are approximately 250 miles south and southeast of Atlanta, respectively. Both the Appalachians and these two maritime bodies exert influence on the Atlanta climate. The temperatures are moderated throughout the year. Summer temperatures are moderated somewhat by elevation but are rather warm, but prolonged periods of hot weather are unusual. The mountains to the north tend to retard the southward movement of Polar air masses, resulting in rather mild winters. Cold spells are not unusual but are rather short-lived. Late March is the average date of the last temperature of 32°F and mid-November is the average date for the first temperature of 32°F. This results in an average growing season of 234 days.

Abundant precipitation fosters natural vegetation and the growth of crops in the Atlanta area. Minimum dry precipitation periods occur mainly during the late

summer and early fall, with maximum thunderstorm activity occurring during July. On the average 49.8 thunderstorms occur per year, with 10.4 thunderstorms on the average occurring in July. A snowfall of 4 inches or more is expected about once every 5 years. Ice storms, freezing rain or glaze, occur about twice every three years, impacting travel. Severe ice storms occur about once in ten years.

Atlanta's coldest month is January with an average daily high of 51.2°F and an average low of 32.6°F. The hottest month is July with an average daily high and low of 87.9°F and 69.2°F, respectively. The normal yearly precipitation is 48.61 inches while the mean wind speed is 9.1 mph with a prevailing wind direction from the Northwest. The surface wind rose data from the Air Protection Branch of the Georgia Department of Natural Resources is shown in Figure 3.5.

On the average there are 18 reported tornadoes per year in the state of Georgia. The highest occurrence of tornadoes is during the months of March and April when 50% of the total number occurs. On relatively few occasions, hurricanes hit the southeast Georgia coast. Since most do not reach the state or move inland into the state, only moderate winds and heavy rainfalls occur.

3.4 Seismology

The Appalachian Piedmont, in which Atlanta is located, has historically been an area of low earthquake activity. Much of the hazard is from distant, more active areas capable of producing larger earthquakes, such as eastern Tennessee, the Charleston (South Carolina) seismic zone, and the New Madrid (Missouri) fault zone. The largest estimated ground motions are from the February 7, 1812 New Madrid earthquake and the 1886 Charleston earthquake which produced Modified Mercalli Intensities (MMI) of V-VI and VIII, respectively. The intensity VIII shaking at Atlanta from the Charleston earthquake consisted of damage to unsupported masonry such as chimneys.

Only three historical earthquakes have occurred within 100 km of Georgia Tech. They were of intensities IV, VI and IV and occurred in 1913, 1914 and 1933 respectively. All three occurred at distances greater than 88 km from Georgia Tech.

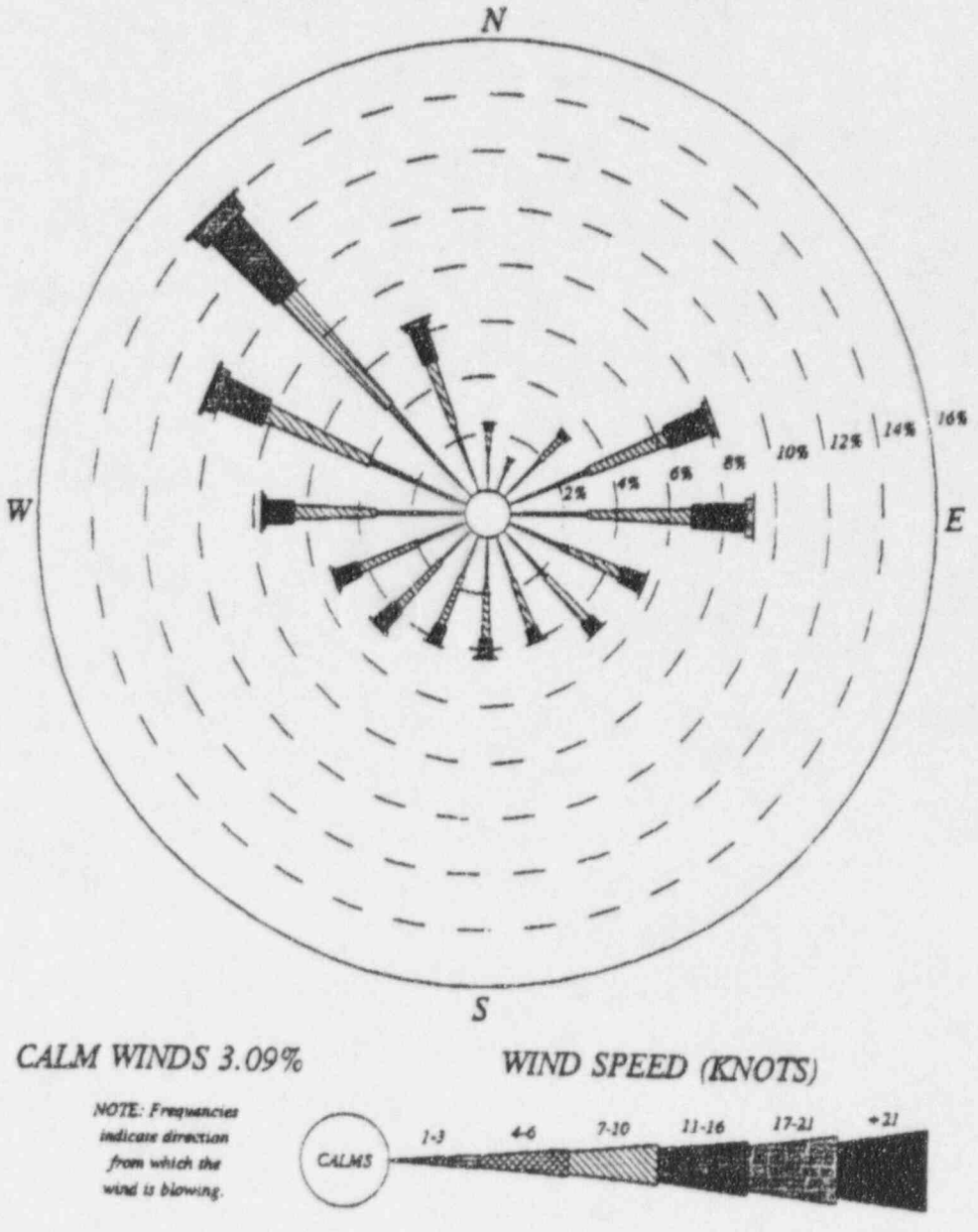


Figure 3.5 Annual Surface Wind Rose for Atlanta.

Table 3.1
Populations and Areas for 1990 Census Tracts in the Georgia Tech Campus Vicinity

Tract No.	Population	Area (Km^2)
10.95	6,460	2.7
6	1,335	1.4
8	1,516	1.0
22	1,835	0.8
21	2,706	0.8
20	1,212	0.1
19	615	0.9
12	3,137	1.1
11	1,427	0.5
4	1,545	1.5
5	2,564	3.3

Bibliography

Local Climatological Data. Atlanta. Georgia, National Oceanic and Atmospheric Administration, 1992, ISSN 0198-1560.

Water Resources Data for Georgia, U. S. Geological Survey, 1990.

Population and Housing Characteristics for Census Tracts and Block Numbering Areas, 1990 Census of Population and Housing, Atlanta, GA, MSA, Issued 7/93, U. S. Department of Commerce, Bureau of Census, 1990 CPH-375.

Herrick, S. M., and H. E. Legrand, "Geology and Ground-Water Resources of the Atlanta Area, Georgia," Georgia Geological Survey, Bulletin No. 55, Atlanta, 1949.

Walters, James V., "The Bearing Capacity of Drilled Piers on Partially Decomposed Rock, " M.S. Thesis, Georgia Institute of Technology, Atlanta, 1958.

"Report to the Georgia Institute of Technology on Soil Explorations at the Proposed Nuclear Reactor Site," Law Engineering Testing Company, July 1, 1958.

"Volume I. Summary Report Seismic Hazard Study: Georgia Institute of Technology Campus," Law Engineering, March 16, 1993.

J. A. Ruffner, Climates of the United States, 2nd edition, Volume I, Gale Research Company, Detroit, Michigan, 1980.

4. THE REACTOR FACILITY

4.1 Description of the Site

The immediate vicinity of the site is shown in Figure 4.1, a recent aerial photograph. The Electronic Building and its grounds are approximately 200 feet south and southwest of the reactor building. The Physics building is approximately 700 feet south of the reactor building. The Baker building is about 200 feet to the west and the physical plant is about 50 feet east and north east. The site topology is shown in Figure 4.2.

The east boundary of the Neely Nuclear Research Center in which the GTRR is located, is Atlantic Drive, a street that carries moderate local traffic. The main laboratory and parking lot entrance are from Atlantic Drive. More details of the two-acre site are shown in the plot plan, Figure 4.3. The land immediately adjacent to the reactor and the laboratory is surrounded by a personnel fence. The only opening in this fence is a truck gate at the rear of the laboratory, which is unlocked only for deliveries.

The point of closest public approach permitted by the fence is 45 feet from the containment building. The land slopes downward to the north and west around the laboratory building so that the elevation is at first floor height on the south and east, and ground floor level to the north and west. The location of the reactor cooling tower, waste storage tanks, and exhaust are shown in Figure 4.3.

4.2 Description of Laboratory Facilities

4 2.1 General Layout

Laboratory and office space and a variety of support facilities are housed in a two-story building adjoining the containment building. This building is approximately 90 feet by 130 feet and contains 24,000 square feet of floor space. Figure 4.4 is a plan of the first, or main floor. In the southeast corner of the building is a high bay area measuring 46 feet by 56 feet. This area contains the high level radiation facilities including the two hot cells, radiochemistry laboratory, decontamination room and storage pool; all are discussed in detail in Section 4.2.2

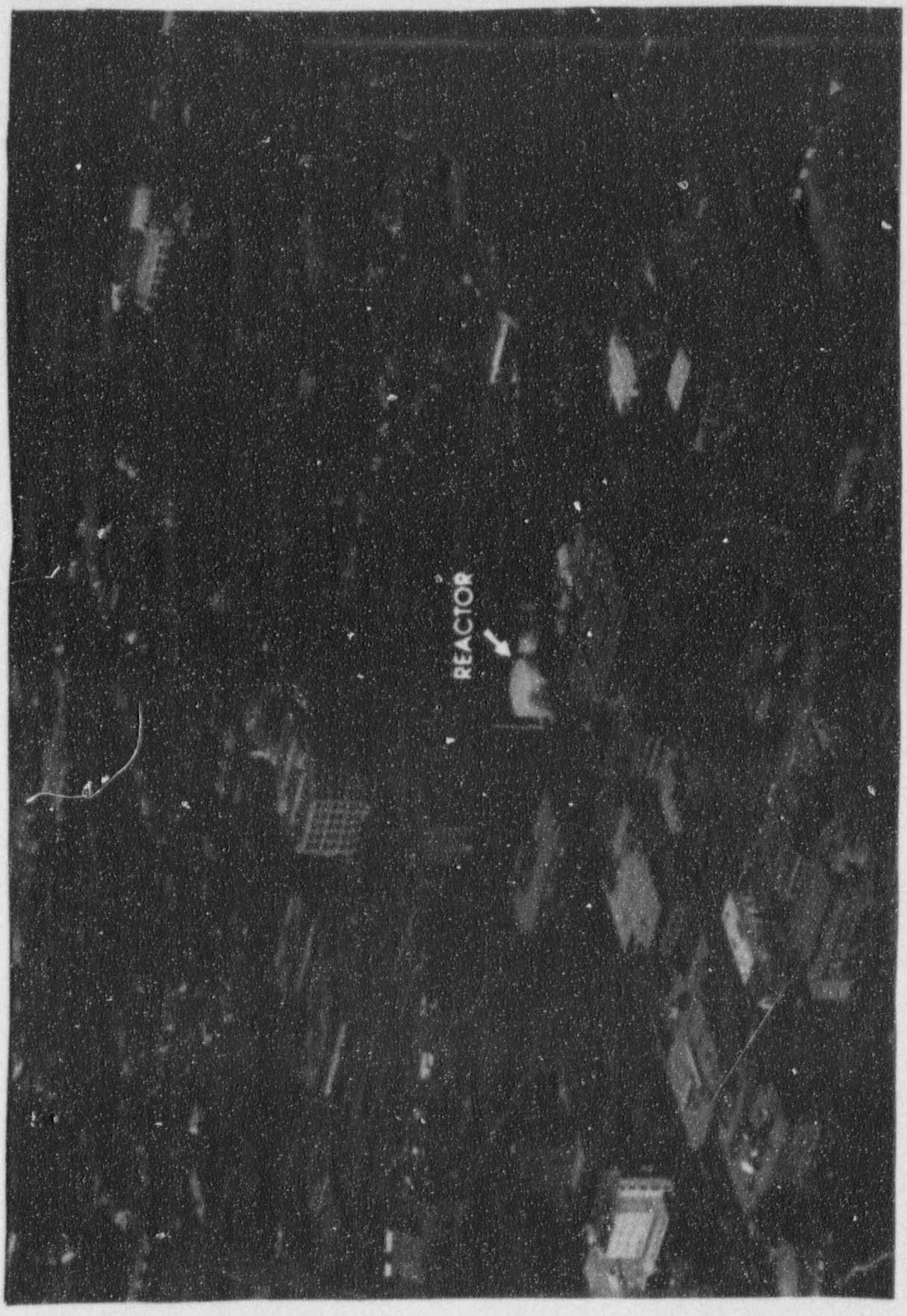


Figure 4.1. Aerial Photograph of Reactor Vicinity.

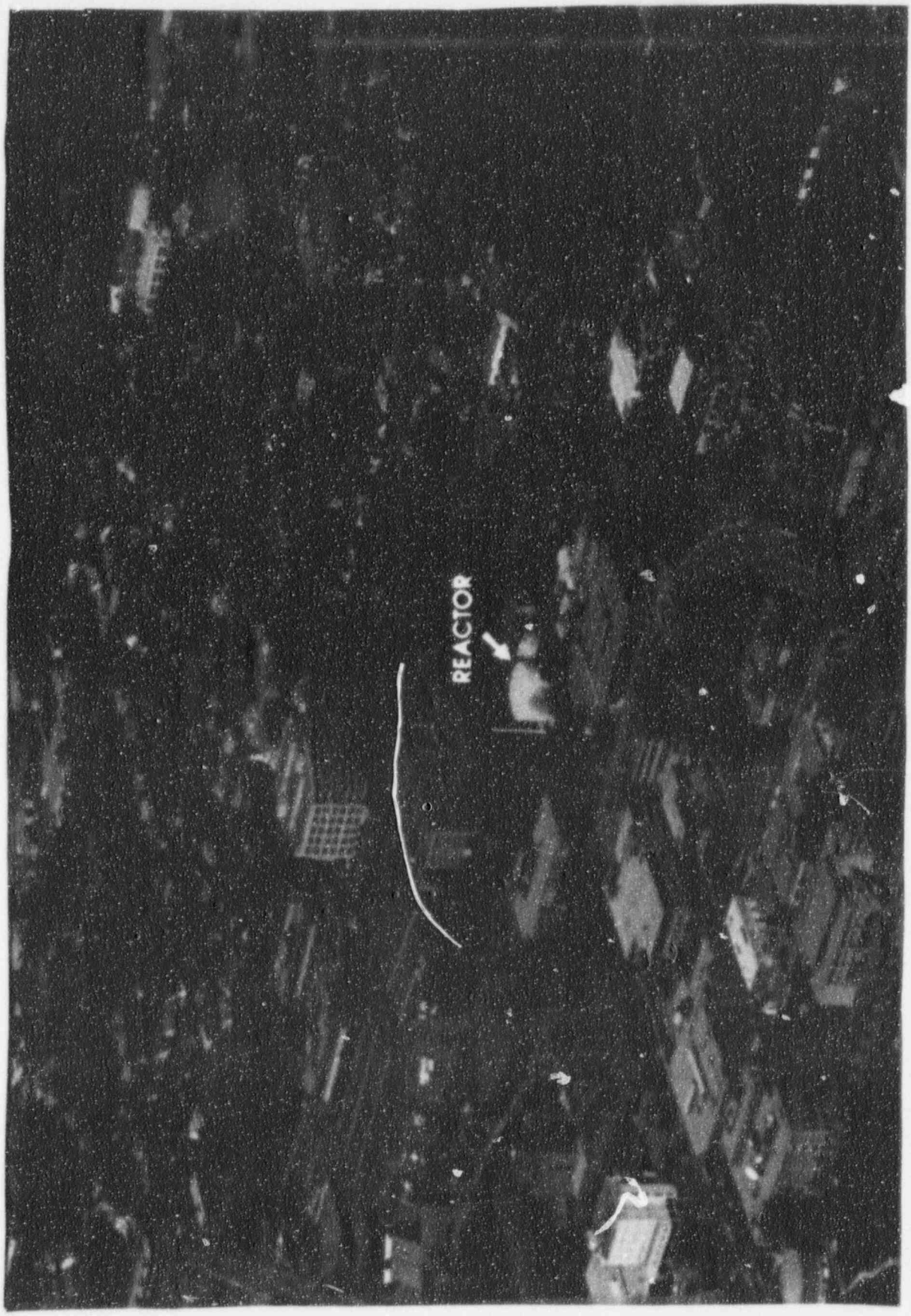


Figure 4.1. Aerial Photograph of Reactor Vicinity.

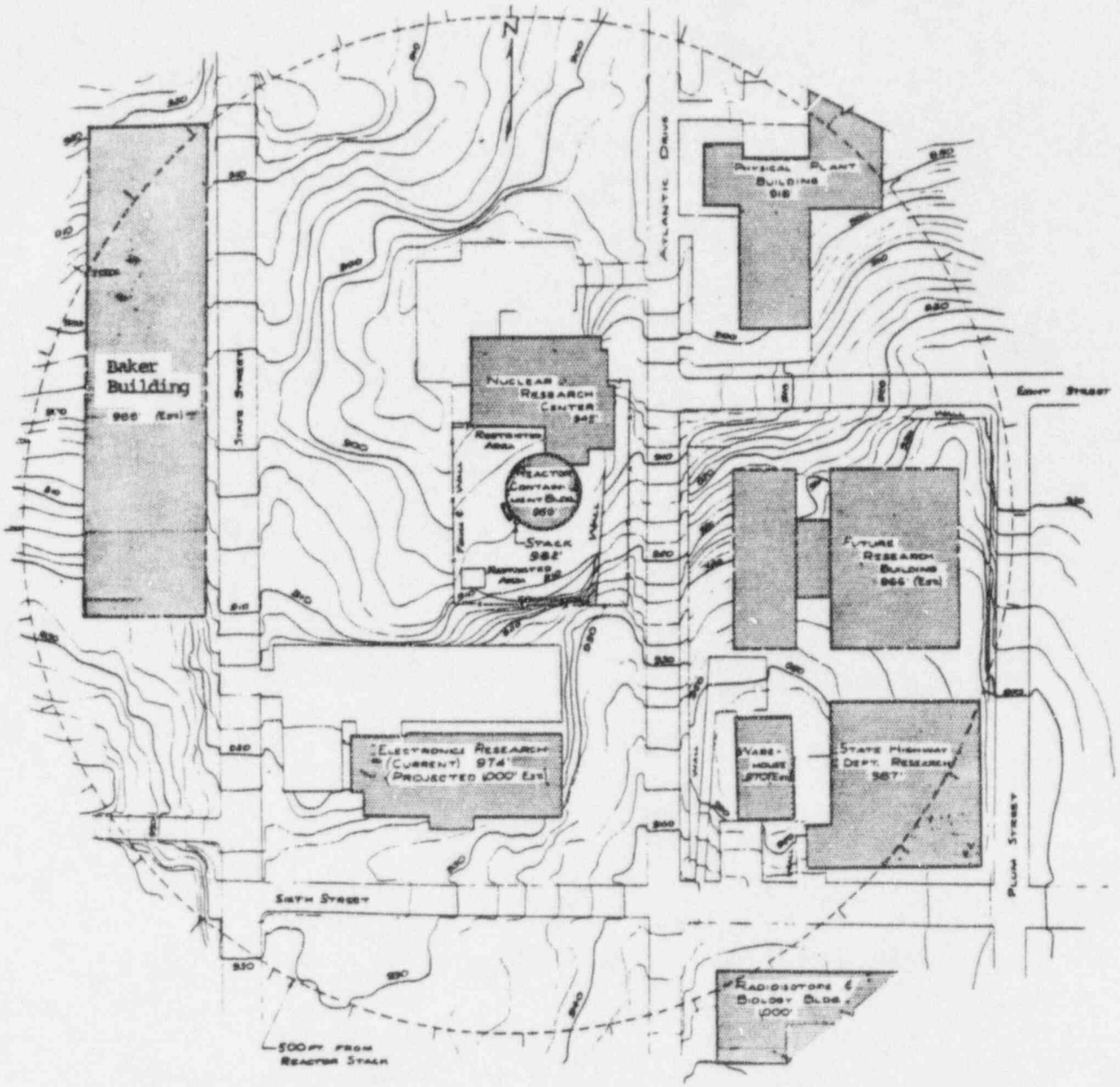


Figure 4.2. Topographic Map of Nuclear Research Center Site.

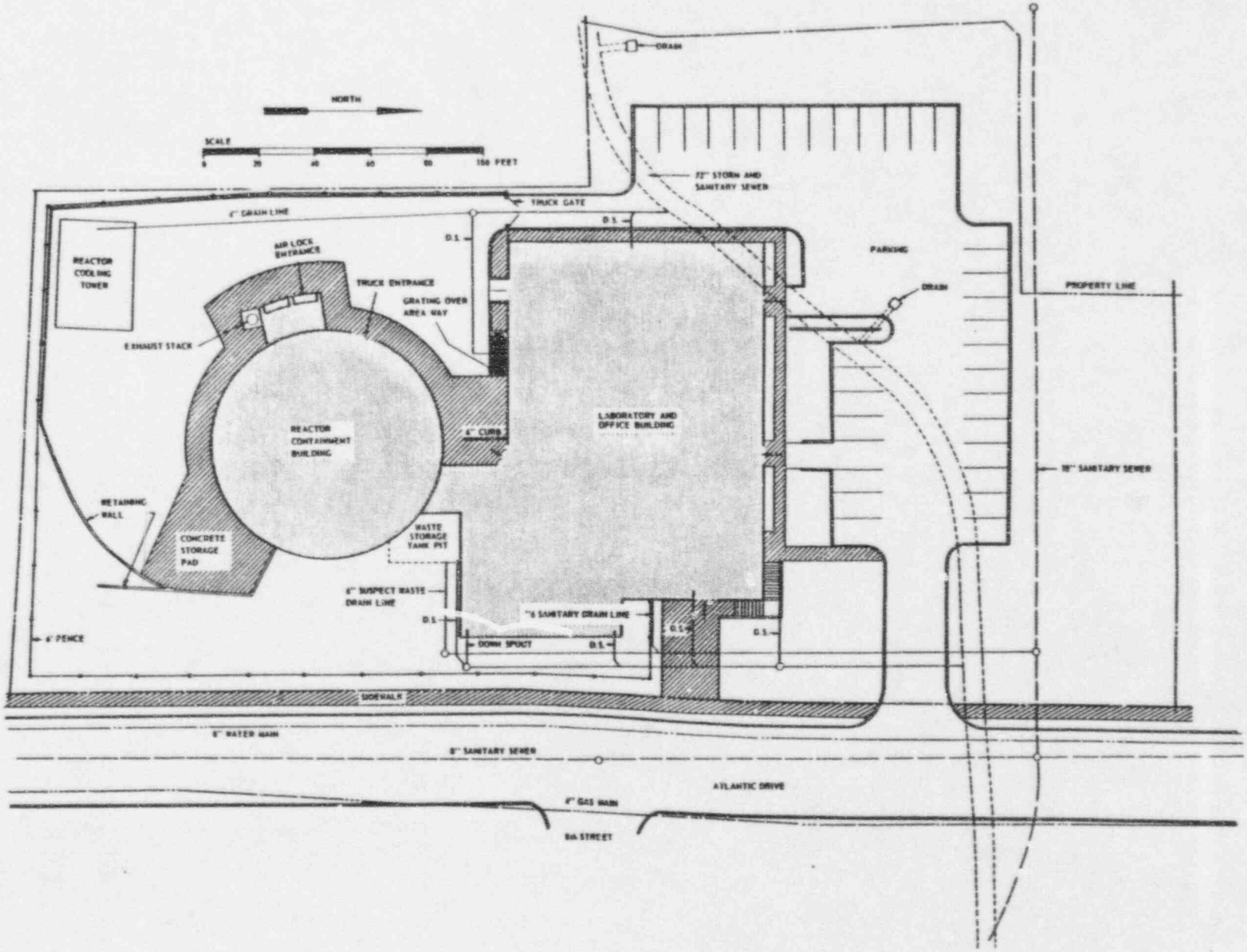


Figure 4.3 Nuclear Research Center Plot Plan.

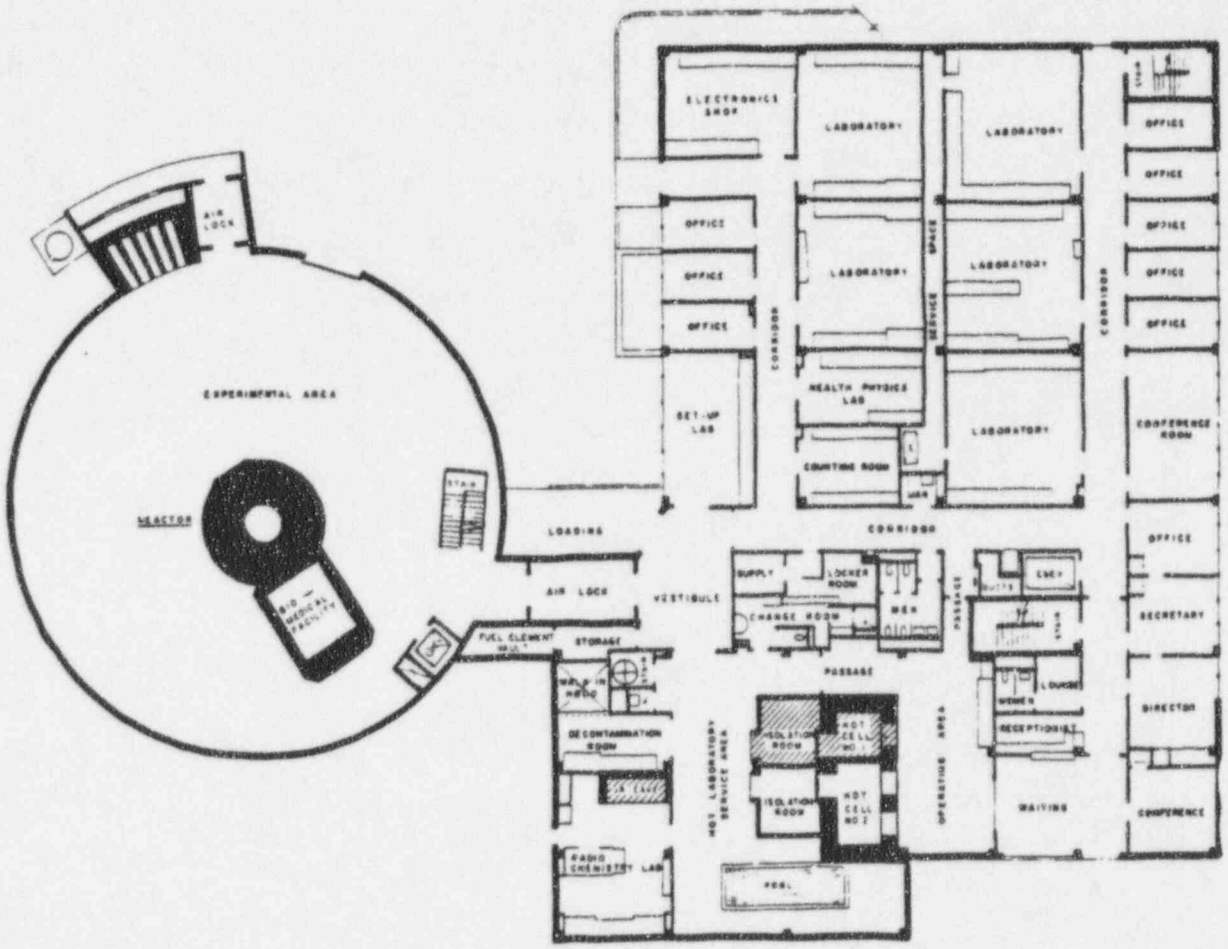


Figure 4.4 Nuclear Research Center First Floor Plan.

below. The area is normally closed off from the remainder of the building, with access restricted to the change room entrance. The handling of materials with potential contamination will be restricted to this area. Only sealed sources or very small quantities of radioactive material are handled in the low level laboratories.

The change room acts as a buffer between the security zone and the rest of the laboratories. It contains lockers for individuals having to change into protective clothing, storage space for protective clothing and equipment, a shower and washstand for personnel decontamination, and space for personnel monitoring equipment. A bench acts as a step-over divider between areas. An alternate path through the shower stall also connects these two areas. The change room has been located next to the air lock entrance leading into the reactor building. In case a serious contamination problem arises in the reactor area, traffic to that area can easily be restricted so as to make passage through the change room mandatory. Thus, the change room can also act as a buffer zone for the reactor.

In addition to possible traffic between the change room and reactor, the vestibule in front of the air lock entrance establishes several other routes. Doors to the high-level area, outside loading platform, fu[illegible]ment storage vault, Health Physics laboratory, and main laboratory building co[illegible]dors all open from this vestibule. This facilitates the transfer of new fuel elements from the vault to the reactor, of used elements and hot experiments from the reactor to the pool or hot cells, of materials between the loading platform and the reactor or high level area, or of experimental equipment from the main laboratories to the reactor or hot cells. All such transfer operations must pass stringent monitoring before leaving the controlled area.

Transfer of radioactive material between the hot cell area and the reactor is facilitated by a variety of shielded containers ranging through several sizes of hand carried pigs, steel-jacketed lead boxes on casters, and four-ton and seven-ton lead casks. The larger casks are handled by overhead cranes serving the reactor area, the high bay area, and the hot cell. Fork lift trucks transfer the casks through the air-lock between the two areas.

Reference to Figure 4.4 reveals the back-to-back arrangement of the low level laboratories. These laboratories are separated by a service chase which contains all utilities. The basic module for these laboratories is approximately 20 feet by 24 feet, with one of the six modules divided in half to form the counting room and water chemistry laboratory. Although the allocation of laboratory space depends upon the research programs in progress at any given time, the present assignments are typical of anticipated usage. Present designations include radiobiology, nuclear chemistry, biochemistry, metallurgy, physics, and electronics. Space along the north and south walls is devoted to offices and conference rooms.

The top or second floor, shown in Figure 4.5, is a gallery which permits visitors to view work in the high bay area behind the hot cells and within the containment building without entering either area. The gallery is effectively isolated by glass viewing panels, thus providing visitors with an excellent view of the facility while subjecting them to minimal possibility of contamination or radiation exposure. Moreover, distraction for operating and research personnel is greatly reduced.

Final grading established the outside ground level at the elevation of the first floor on the east (main entrance) and south (loading platform). The land slopes so that the lower or ground floor is at ground level on the west and north sides of the building. The employee entrance is from the parking lot to the north. The ground floor, shown in Figure 4.6 contains most of the mechanical and electrical equipment, a detector instrumentation laboratory, dark room, a complete machine and welding shop, animal quarters, a receiving and stockroom and, along the north wall, additional office space. The northeast corner was designed as a bio-medical suite which may be isolated from the remainder of the building. It contains offices, two laboratories and two small rooms. These latter two rooms may be used to accommodate human patients for periods of a few hours before or after treatment. This suite is located near the elevator leading to the first floor and bio-medical irradiation facility, and near the employee entrance which can be used for ambulance delivery. The space is now used as general offices and laboratories.

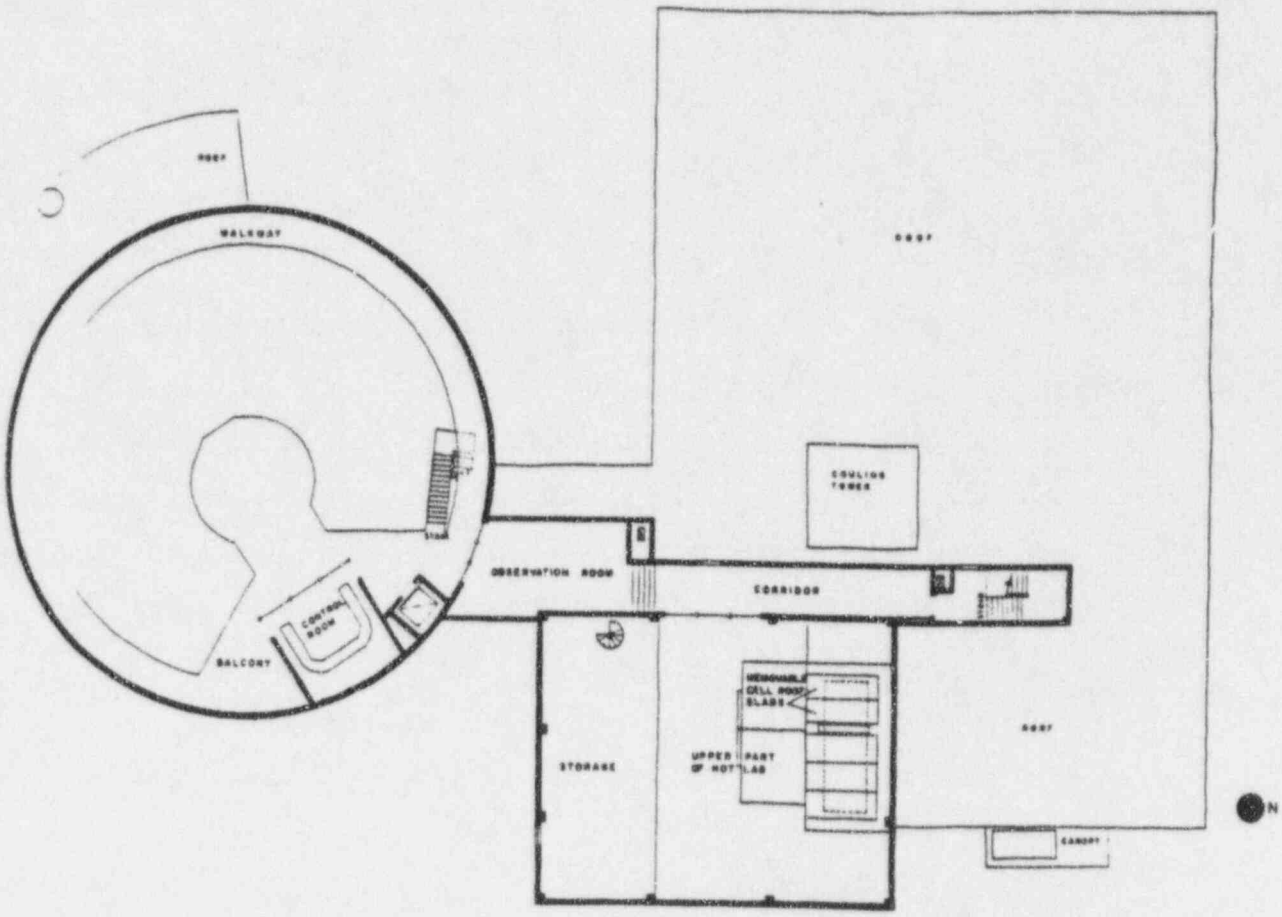


Figure 4.5. Nuclear Research Center Second Floor Plan.

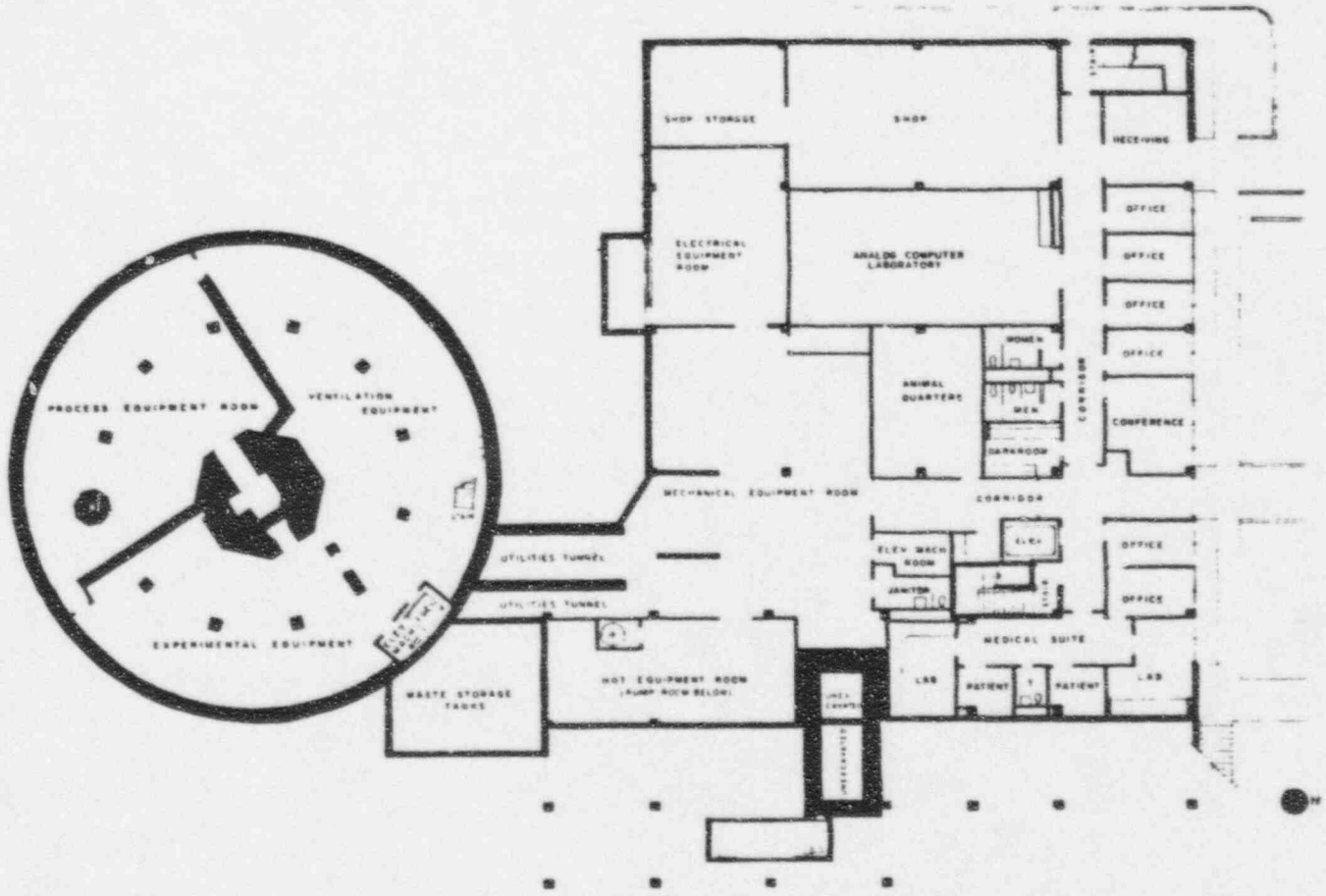


Figure 4.6 Nuclear Research Center Ground Floor Plan.

All city water for the entire Center is supplied through a single 4-inch line which enters the ground floor of the laboratory building. The water passes through a gate valve, strainer, backflow preventer, pressure reducing valve, and a second gate valve. Following this, the line branches for distribution throughout the facility. The backflow preventer is a reduced pressure type, Crane D-851, size 4, Model A. Backflow preventers of this general type have been tested at Oak Ridge National Laboratory to determine if backflow occurred under adverse conditions (water hammer, upstream vacuum, etc.). Under these test conditions there was no detectable backflow. These tests are described in report number ORNL 3380.

4.2.2 The Hot Laboratory

The hot laboratory includes the high bay area in the southeast corner of the building and the adjoining hot cell operating area (refer to Figure 4.4). Work involving greater than a few millicuries of loose radioactive material usually is conducted within the hot laboratory portion of the building. The hot cell operating area is free of radioactive contamination. There are routine restrictions on access to this area because of control to the security zone. The cell service area is treated as suspect and access to it will be carefully controlled. There are three possible entrances--a large truck door from the air lock vestibule, a personnel door from the operating area and a door from the change room. The first two are closed, with access restricted to the change room route. The hot cell isolation room, handling and storage pool, radiochemistry laboratory, decontamination room, cleaning supply, closet and circular stair to the equipment basement are all accessible from the cell service area.

The hot cells are constructed of dense concrete (minimum density 215 pounds per cubic foot) three feet thick. The larger cell is 7 feet by 13 feet inside, while the smaller is 7 feet by 7 feet. Both are 13 feet, 4 inches high inside with removable roof slabs of 2 feet, 6 inches thick normal concrete. The larger cell is equipped with two liquid-filled viewing windows which match the shielding capacity of the wall. Each of the two viewing stations is equipped with a pair of Model 8 and Model D

mechanical master-slave manipulators. Access to each cell from an isolation room is through a doorway at the rear. The cell doorway is closed by a door which slides horizontally, parallel to the rear cell wall. The 14 inch thick steel inter-cell is removed to form one cell 21 feet long.

Because of budgetary limitations, only the larger of the two cells has been completely outfitted and is now in service. The walls and ceiling of the smaller cell are complete, but the window and door openings are closed by stacked lead brick 8 inches thick.

Sample containers are introduced into the completed cell through the door or roof. A chute and built-in elevator mechanism connects the storage pool to the larger cell so that samples may be passed directly from one to the other. The front of each cell is provided with a number of removable stepped plugs to be used for the installation of hydraulic, mechanical, pneumatic or electrical connectors for remotely controlled equipment. A minimum of utilities is permanently installed in the cell. Instead, all the usual services are available immediately in front of the cells for insertion as required.

Immediately behind the larger hot cell is an isolation room approximately 10 feet by 10 feet. This room acts as a buffer zone or air lock and reduces the spread of contamination from the cell interior. The walls consist of 8 inches of solid concrete. The shielding thus provided is sufficient to permit temporary storage of contaminated equipment. Both the hot cell door and isolation room door are padlocked, and keys are maintained by the Health Physics Office. High radiation levels inside the cell are indicated by a buzzer and a warning light at each door.

Along the east side of the high bay area is a storage and handling pool filled to a depth of 18 feet with water. This 5 foot by 20 foot pool provides space for spent fuel element storage, gamma irradiation experiments using spent fuel elements or other multicurie sources, disassembly of large pieces of radioactive equipment such as in-pile loops or fuel assemblies and other work requiring remote operation on large objects with good visibility. The contamination level in the pool water is controlled by recirculating it through filters and ion exchange resins.

The radiochemistry laboratory may be reached from the service area. This 18 foot by 26 foot room contains two radioisotope hoods for high millicurie level work.

Next to the radiochemistry laboratory is a 9 foot by 18 foot decontamination room. Facilities in this room include a hooded sink and an 8 foot by 8 foot walk-in hood. The latter is used for scrubbing down objects which are too large to place under the standard size fume hood. This room is also used for any hot mechanical maintenance work which must be done on portable equipment, and for packaging of high level radioactive wastes prior to shipment. Health Physics personnel utilize this hood for interim storage of incoming radioisotope shipments.

The high bay area includes the hot cells, pool, and roof of the radiochemistry and decontamination rooms. This entire area is accessible to a 15 ton capacity bridge crane. The roof mentioned above provides storage space for equipment, shipping containers, etc. and, for this reason, crane access to it is desirable. Personnel access is provided by a circular stair.

The circular stair also leads down to a hot equipment basement which is shown in Figure 4.6. This portion of the ground floor is normally sealed off from the adjoining mechanical equipment room by closing and locking the connecting doors. In addition to the stair access, a large hatch makes the basement area accessible to the 15 ton crane. This permits removal of heavy objects such as contaminated ion exchange resins in shielded containers. The hot equipment basement houses the hot cell ventilating equipment, radioactive liquid waste storage and treatment system, pool water treatment system, and other mechanical facilities which present a potential contamination problem.

4.2.3 Laboratory and Hot Cell Ventilation

With exception of the high bay and building service areas on the ground floor, the laboratory building is air conditioned. The air handling system is designed to maintain a pressure gradient throughout the building in such a direction that the spread of airborne contamination is minimized. In general, air is supplied to the office and bio-medical suite, flows into the corridors and thence to the laboratories

through transoms above the laboratory doors. Each laboratory can be isolated from the corridor by closing manually-operated louvers.

Laboratories are designed to have a lower pressure than the corridors by the hood exhausts. The air flow in the laboratories is once-through with no re-circulation. The corridor air, however, is re-circulated to an extent which depends on the number of hoods in operation. When most hoods are in use, re-circulation is minimized and conditioned outside air is supplied to maintain the proper pressure gradients. In this way a contamination problem which might accidentally arise in a laboratory can be confined to that laboratory. The radiochemistry laboratory in the high bay area draws its supply air from the reactor air lock vestibule. It is maintained at a pressure below that in the high bay and vestibule by the hood exhausts.

All fume hoods are equipped with constant speed blowers. Radioisotope hoods have individual filters and manually-operated dampers to regulate face velocity and compensate for filter loading. An inclined tube gauge on the front of each hood indicates the pressure at the hood exit. Each gauge for hoods in use with radioactive materials has been calibrated in terms of actual flow rate at a given hood opening so that the operator can assure himself that the flow rate is adequate. Radioisotope hoods in the low level portion of the building are exhausted to the atmosphere through roughing and "absolute" filters installed in series. Filters and blowers are located on the building roof which is accessible from the stairwell.

Hoods in the radiochemistry laboratory and decontamination room within the high bay area are exhausted by individual blower through roughing and "absolute" filters. These blowers and filters are located on the roof above their respective rooms, but within the high bay. The only exception is the walk-in hood which has its "absolute" filter and blower located beneath it in the hot equipment room.

The exhaust system for the hot cells is very similar. Each cell is exhausted through roughing filters into a common plenum, then through a bank of several "absolute" filters and a 2000 cfm blower. A second 2000 cfm blower in parallel with

the first is turned on automatically if the pressure in either cell rises above a set point of the order of 0.1 inch of water below atmospheric pressure. This could occur if a shield door or plugged port were opened or if the primary blower failed.

All exhausts from the high bay area (hoods and hot cells) are ducted individually through the ceiling of the area and released at a common point within a cupola on the roof.

4.2.4 Liquid Waste Handling Systems

Liquid wastes from the laboratory buildings are collected by one of three systems. All sanitary wastes enter conventional cast iron drain lines which lead into a common 6 inch line before emerging from the laboratory building. The path followed by the sanitary drain after leaving the building is shown in Figure 4.3.

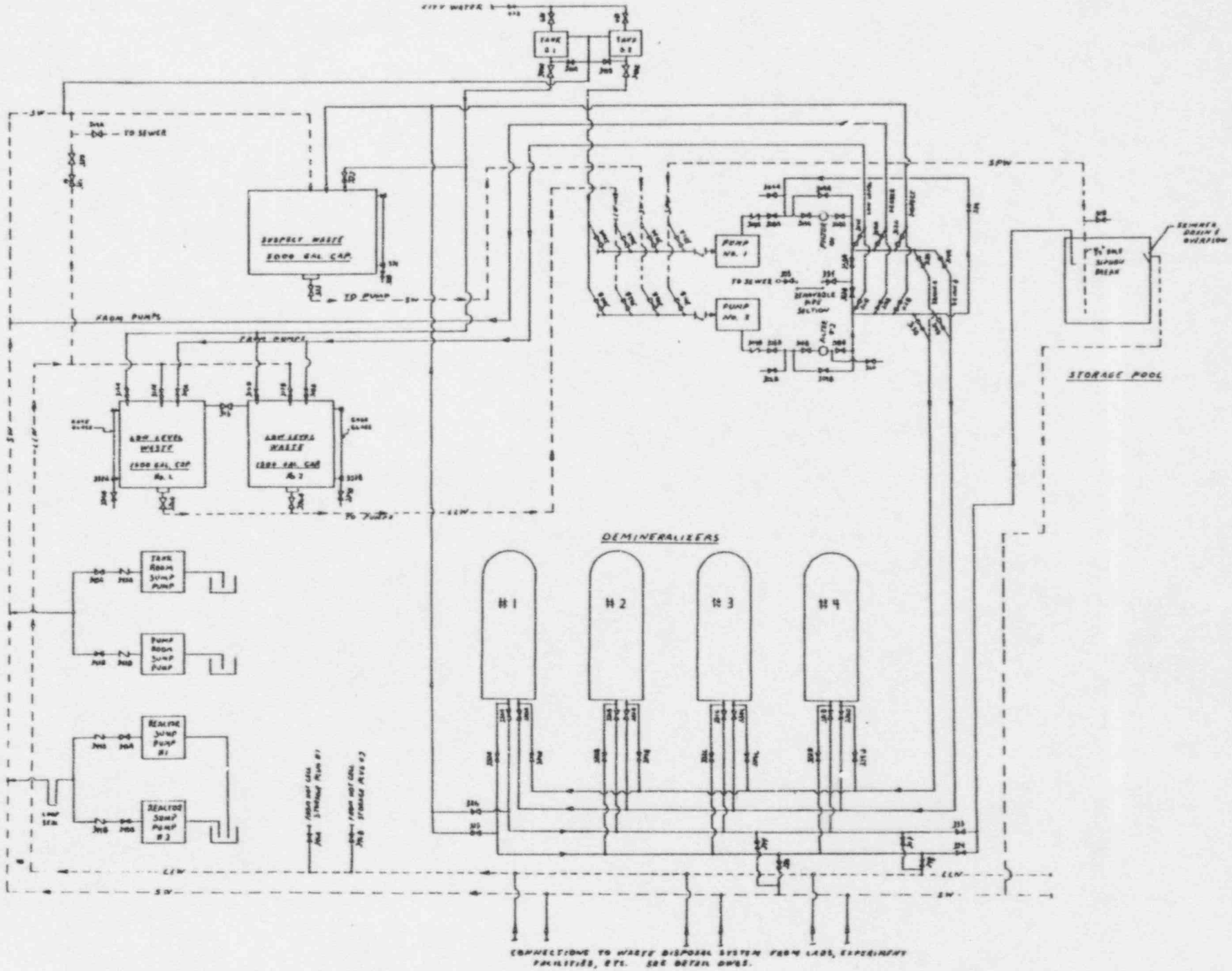
Laboratory wastes and storage pool water are handled by the two systems shown diagrammatically in Figure 4.7. The majority of laboratory sinks, cup sinks and floor drains empty into a suspect waste system. These lines all drain to a common point in the pump room beneath the hot equipment room. At this point the suspect waste is drained to the 5000 gallon suspect water tank. The suspect waste can be shunted to the low level retention tanks if desired, for example, during testing of suspect waste prior to pumping.

All drains from the high bay area of the laboratory building, and several from other areas lead to a low level waste collection system. These wastes empty into a 1500 gallon tank in the waste storage tank pit. The location of this tank pit is shown in Figure 4.6 The bottom of this pit is actually one level below the hot equipment room floor and at the same depth as the floor of the pump room. The tank pit contains a second 1500 gallon tank which is held in reserve to receive low level waste, and the 5000 gallon tank of the suspect system.

Two pumps and a valve manifold in the pump room permit the contents of the retention tanks to be transferred from one to another. The wastes can also be pumped through ion exchange columns and filters in the hot equipment room. Additive tanks in the hot equipment room permit flocculating agents, neutralizers

Figure 4.7

NNRC Liquid Waste Handling System



or other reagents to be added to any retention tank. Operation of the suspect and low level systems is discussed further in Section 7.1.2.

When any of the waste retention tanks nears a full condition, samples of the contents are drawn after thorough mixing, and submitted to Health Physics for radioactivity assay. Provided that the contents meet regulatory requirements, the tanks are emptied by pumping the contents to the municipal sanitary sewer system. If regulatory requirements are not met, the decontamination processes described above will be carried out.

One of the pumps and a deionizer is used to re-circulate the contents of the storage pool in order to maintain the water quality. Water is pumped from a sump in the pool floor and returned to the pool a few feet below the water surface. No drain is provided, thus requiring that the pool be pumped out in order to empty it. Emptying is permitted only after the pool water has been sampled and the radioactivity level proven to be below acceptable limits for discharge. The pump transfers the pool contents to the suspect waste line. The pool discharge line is equipped with a syphon break to prevent the pool from emptying if a pipe failure occurs in the pump room. Make-up water is provided by a conventional hose connection along the east wall of the hot equipment room. Skimming action is obtained by allowing water to overflow into a scum gutter installed across one of the narrow sides of the pool. The scum gutter drain leads to the suspect waste system.

Both the low level and suspect drain lines, and the storage pool piping are of PVC where exposed, and Duriron where buried or otherwise inaccessible. Retention tanks are carbon steel with a vinyl coated interior.

4.3 Description of Reactor Containment Building

4.3.1 General Layout

The reactor containment building is, basically, a cylindrical steel tank with a diameter of 82 feet. The steel bottom is flat, while the top, or roof, is a torispherical

dome rising to approximately 50 feet above ground level. This structure provides a relatively leaktight barrier to the escape of gas from its interior and, thereby, reduces the hazard associated with the release of fission products from the reactor core. The essential features of the building construction are shown by Figure 4.8.

The flat tank bottom rests upon concrete footings. The lower 7 foot - 8 inch section of the tank is filled with concrete. This thick slab serves two purposes: it provides ballast against the buoyant force of ground water, and it supports the remainder of the interior building structure, including the reactor. The top of this slab is the basement floor. Above this, supported by concrete columns and the reactor pedestal, is the first floor slab.

Inside the steel tank wall is a 12 inch thick layer of concrete which extends to approximately 34 feet above the outside ground level. This inner concrete wall has a dual function: it shields an observer outside the building from a radiation source within, and it supports a 20 ton capacity polar crane which services the reactor and first floor area.

The basement of the containment building houses all reactor process equipment, the exhaust ventilation equipment and the electrical load center for the building. The basement floor plan is shown in Figure 4.9. Although the basement is at the same elevation as the ground floor of the laboratory building, there is no direct access between the two areas. The basement can be reached only by elevator or stairway from the first floor of the containment building. The reactor process equipment is located within an area surrounded by two-foot-thick concrete walls. The "absolute" filters and blower for the containment building exhaust are adjacent to this shielded area. A large holdup volume through which this exhaust must pass is cast into the basement floor slab. Another large void in this concrete slab provides an expansion chamber which is connected to the helium/nitrogen space above the reactor core through a graphite rupture disk.

Much of the basement floor has been left unoccupied so that it may be used for setting up experimental equipment. To enable connections to be made between apparatus on the first floor and auxiliary equipment in the basement below, the first

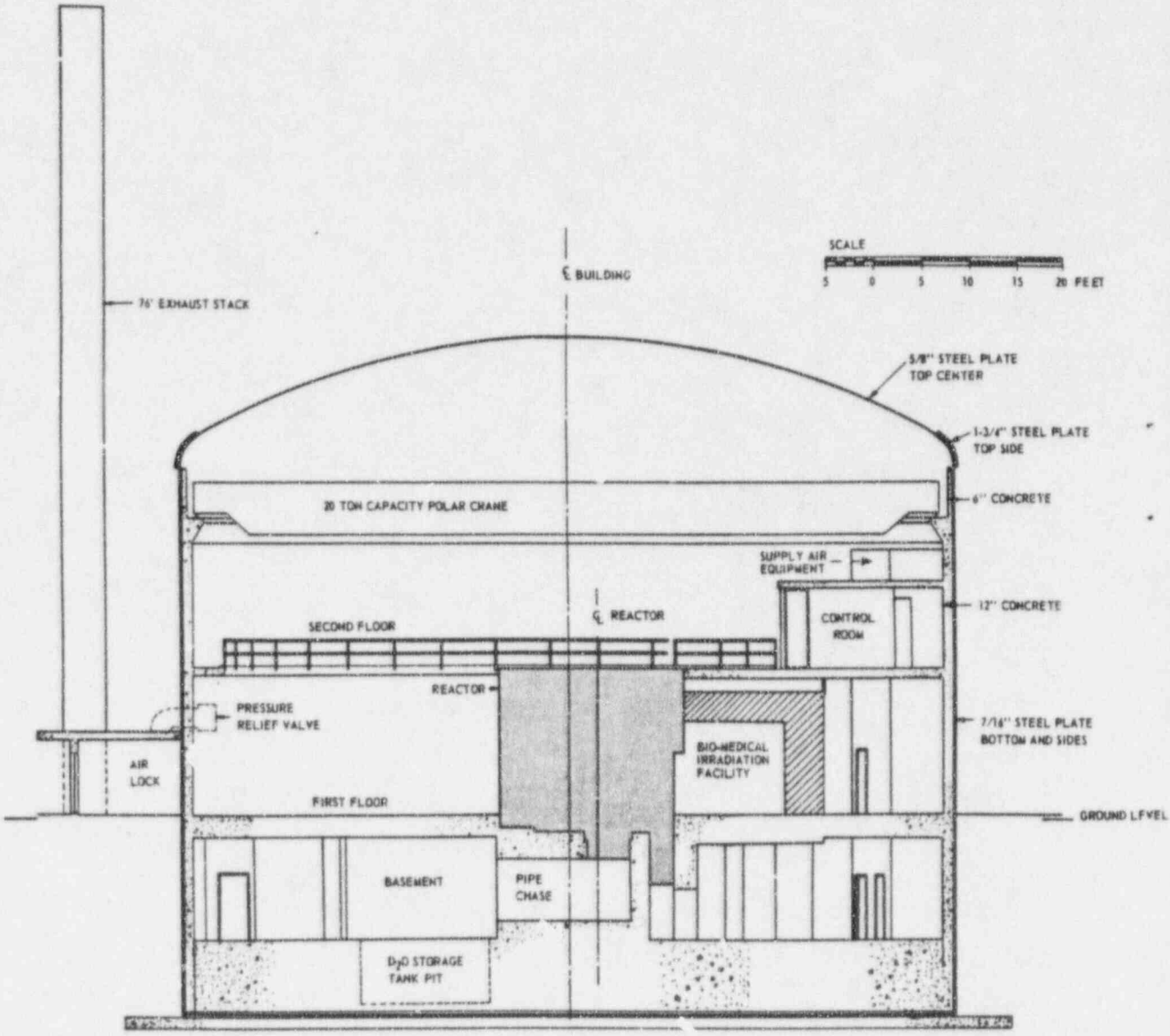


Figure 4.8 Reactor Containment Building Cross-Section A-A.

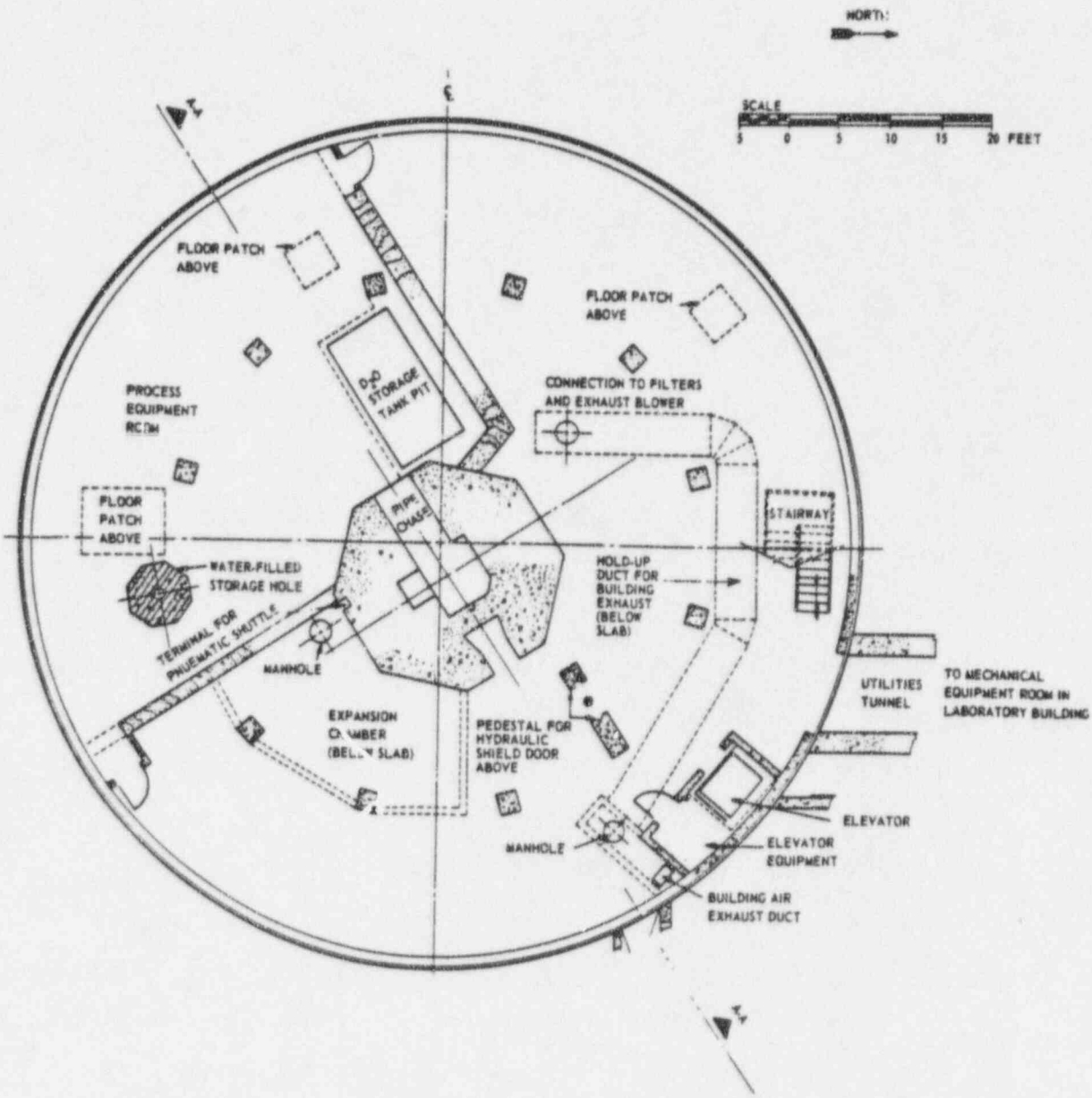


Figure 4.9 Reactor Containment Building Basement Floor Plan.

floor slab has many pipe penetrations. These are located around the circumference of the reactor shield and around the building periphery. The equipment and terminals for the two 1 1/2 inch diameter pneumatic shuttles are in the basement near the reactor pedestal. Samples may be introduced and discharged at this point or routed into the laboratory building. With the exception of the two water lines to the reactor cooling tower, all service penetrations of the containment building enter through the basement wall directly under the main air lock. At this point, the concrete inner wall has been omitted so that the steel shell is exposed. On the laboratory building side also the steel is exposed in the utilities tunnel which connects to the mechanical equipment room.

As Figure 4.10 shows, the first or main floor of the containment building is largely unoccupied except for the reactor near the center. The outer limits of the biological shield may be approximated in cross-section by a 20 foot diameter circle. The center of this circle is displaced horizontally about three feet from the center of the building. This permits more efficient use of the polar crane when engaged in operations above the reactor. The crane can reach almost every point on the first floor which is not covered by the balcony or walk-way. Several floor patches indicated in Figure 4.10 permit large and heavy objects to be moved between floors by the crane. A 12 inch I.D. fuel element storage hole is accessible from the first floor level by removing a cover plate in the floor.

The containment building is entered at the first floor level by one of three routes: the main air lock leading to the laboratory building, a smaller personnel air lock leading to the outside service court, and a truck door which may be unsealed and opened only when the reactor is shut down. A shielded bio-medical irradiation room is located on the first floor immediately adjacent to the reactor. More information concerning this installation is contained in Section 4.4.6.2.

The top of the reactor biological shield is 15 feet above the first floor. At this level, shown in Figure 4.11, there is a walk-way which runs completely around the circumference of the building. At one point, this walk-way is enlarged to form a second floor level which connects with the top of the reactor. The reactor control

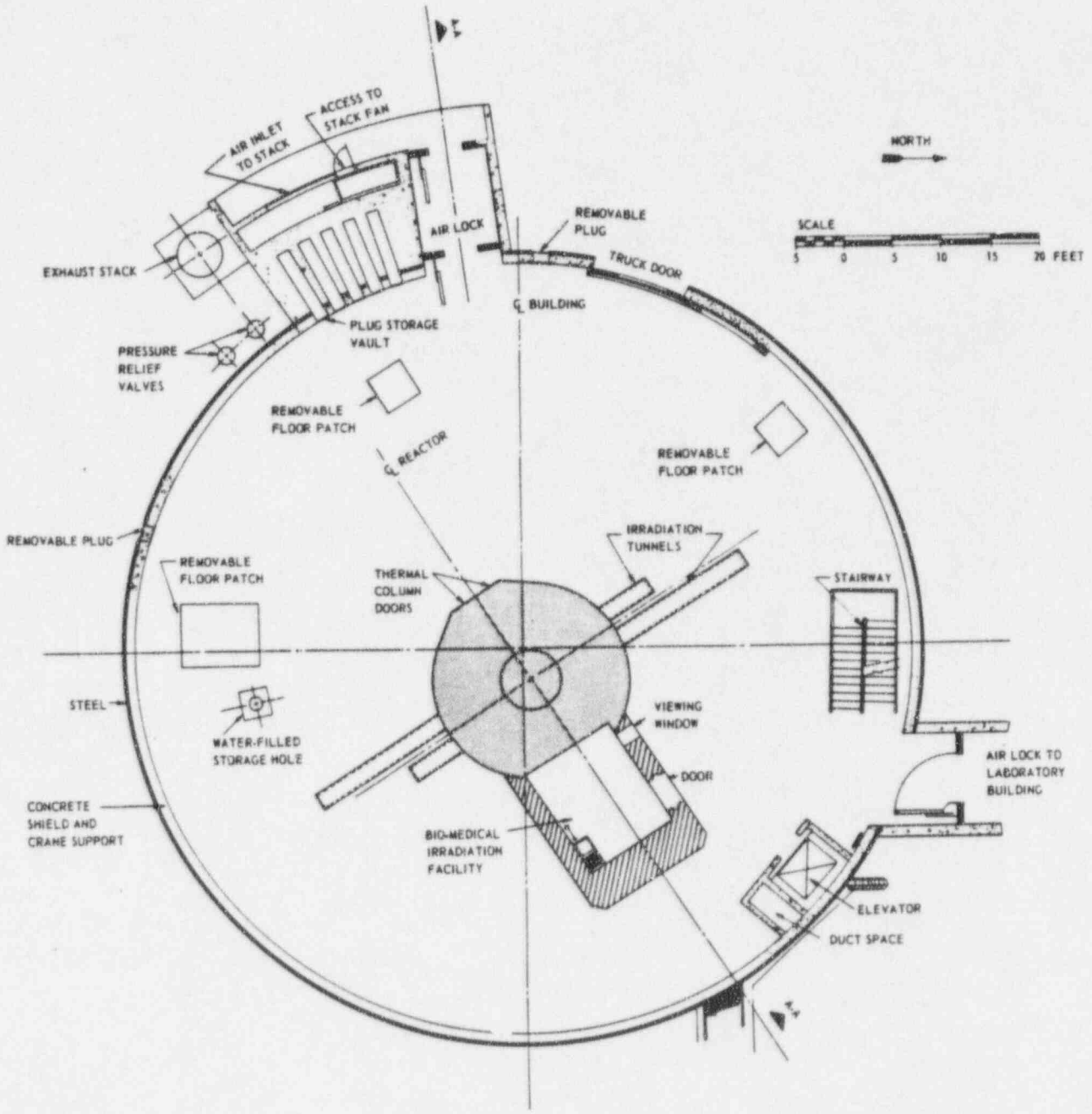


Figure 4.10 Reactor Containment Building First Floor Plan.

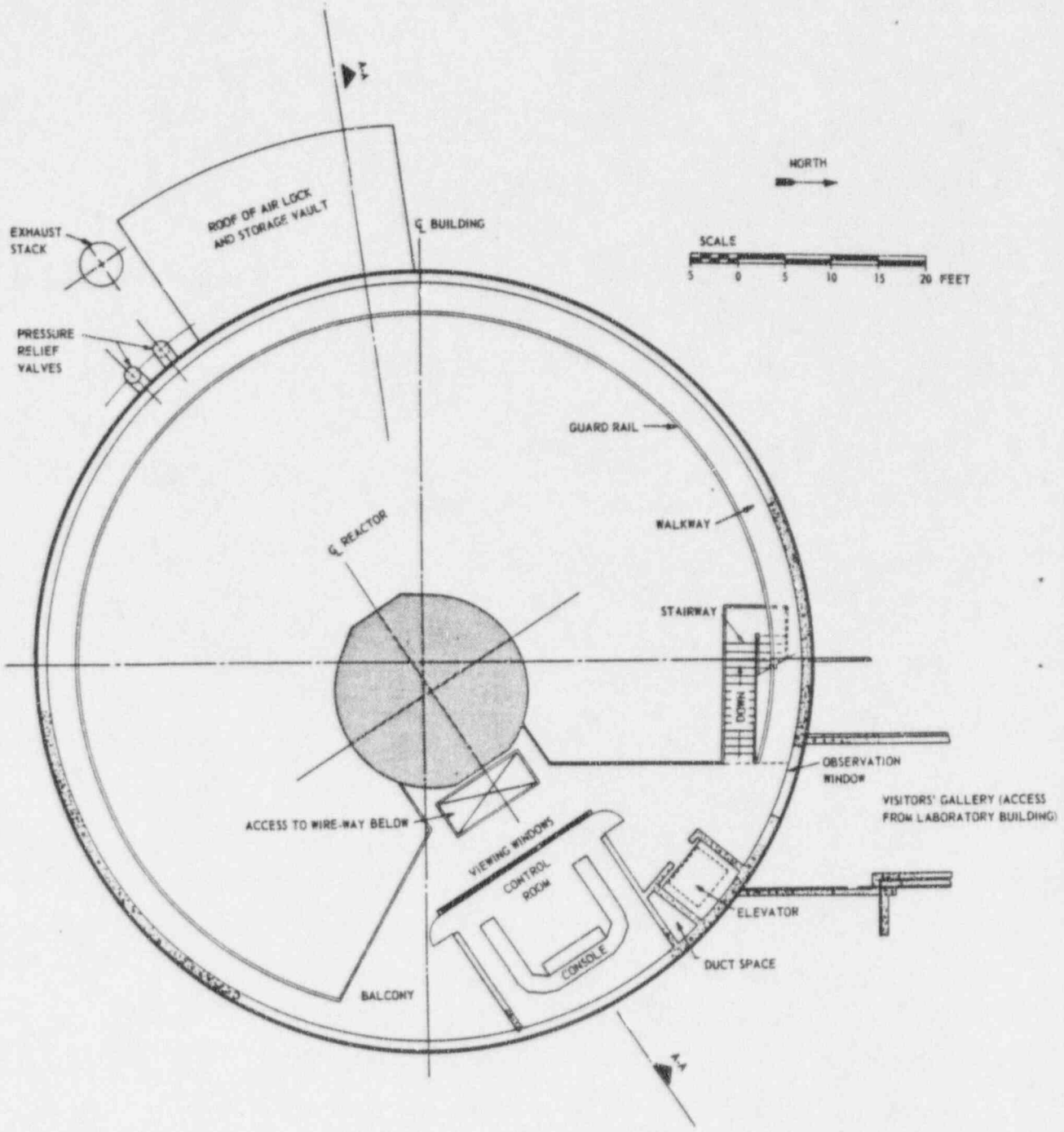


Figure 4.11 Reactor Containment Building Second Floor Plan.

room occupies much of this second floor. Glass panels in the front wall of the control room provide the reactor operators with a partial view of the main floor. The second floor may be reached by using the elevator or the stairs.

4.3.2 Provisions for Insuring Leak-Tightness

The most important aspect of leak-tightness of the containment building is the welded steel tank which completely encloses the building. During construction, welds were radiographically inspected at all intersections of two or more welds, in addition to inspection by radiographic testing of welds in accordance with ASME Code, Section VIII, Subsection B, paragraphs UW-51 and UW-52. All welds on the flat plate bottom of the tank, and any other welds required to be covered or encased during erection were given a vacuum-type soap bubble test. After erection of the containment shell was completed and all openings were closed, all welds not previously vacuum tested were soap bubble tested at 2 psig. The building was tested again after all penetrations were made and sealed, prior to acceptance by Georgia Tech, and annually since that time. The tests and preventive maintenance program in place provide assurance that there will be no degradation of the containment structure over the next 20 years. Leak rate measurements and the test procedure followed are discussed in section 7.2.2. All test results have been well within the design criterion that leakage will not exceed 1/2% of the building volume in 24 hours per 1 psig internal pressure. This corresponds to a total leakage rate of 1% in 24 hours at 2 psig, the figure estimated to be the maximum internal pressure which would be developed within the building as a result of a power excursion and an ensuing aluminum-heavy water reaction (see Appendix A). Although the maximum internal pressure considered credible is only 2 psig, structural requirements based on expected external pressures have resulted in a design which is believe capable withstanding an internal pressure of at least 7.5 psig.

Careful attention was given to all penetrations of the steel shell in order to meet the leakage rate criterion. A description of penetrations and the method of sealing each is summarized in Table 4.1.

The truck door seal is effected by bolting the door tightly against an inflatable rubber gasket. The airlocks (Henry Pratt Company Model PS-M) used for access to

Table 4.1

GTRR Containment Building Penetrations or Inserts

Penetration	Approximate Size	Number	Sealing Method
Truck Door	10' x 13'	1	Inflatable Gasket
Personnel Airlock	3'6" x 7'0"	1	Inflatable Gasket
Equipment (Main) Airlock	5'0" x 8'0"	1	Inflatable Gasket
Viewing Window	3'3" x 3'8"	3	Neoprene Gasket
Electrical Penetrations		22	Potted Condulets
Conduit Size	3"	1	
	2-1/2"	2	
	2"	4	
	1-1/4"	2	
	1"	6	
	3/4"	7	
Secondary Coolant System (Supply and Return)	8"	2	Liquid Loop
Chilled Water Loop (Supply and Return)	4"	2	Liquid Loop
Heating Water Loop (Supply and Return)	2"	2	Liquid Loop
Domestic Hot Water Loop (Supply and Return)	1"	2	Liquid Loop
Cold Water Supply	1"	1	Liquid Loop
Elevator Pit Drain	4"	1	Liquid Loop and IPS Cap
Building Sump Discharge	4"	1	Liquid Loop
ECCS Makeup Water	1"	1	Liquid Loop
Air Intake and Exhaust	24"	2	Automatic Valve
New Pneumatic Sample Handling System	2"	2	Automatic Valve
Old Pneumatic Sample Handling System	1-1/4"	4	IPS Cap
Pressure Test Nozzles	2"	2	IPS Cap
Vacuum Breakers	10"	2	Check Valve
Air Cross Connection	1"	1	Check Valve and Ball Valve
Beam Extension Ports	18"	2	Bolted Flange with Gasket

the containment building for personnel and small equipment are also sealed with an inflatable gasket and are designed for a pressure of 2 psig. The manually operated airlocks contain the usual interlocking feature to insure that one door is sealed at all times, as well as electro-mechanical detectors for low-air pressure. The systems contain check valves and reservoirs so that seals remain inflated even when line pressure is lost; the doors can be opened and resealed from the accumulator capacity.

The viewing window mentioned in Section 4.2.1 constitutes an insert in the steel shell. This window is glazed with two sheets of 3/8 inch thick tempered glass laminated with an 0.008 inch thick plasticized polyvinyl butyryl resin between, and is set in a neoprene gasket.

Nineteen electrical penetrations are made through the containment shell at the utilities tunnel on the laboratory side of the containment building. No penetrations smaller than 3/4 inch NPS were made. Seals were effected by welding around conduit couplings on both sides of the containment shell and filling the sealing condulets with sealing compounds on both sides. Three spare conduits were sealed with standard pipe caps.

Containment building pipe and duct penetrations are either sealed by "U" shaped loops of sufficient height of the fluid flowing to contain a pressure of 2 psig, or sealed by valves which are automatically closed by the building isolation safety circuit. The loops are also effective as expansion joints, and reduce stresses on the steel shell to which the pipes were welded directly. The six penetrations sealed by a building isolation signal are the building ventilation inlet and outlet ducts, four pneumatic lines for the remote terminals of the pneumatic ("rabbit") irradiation facility. In addition, an isolation signal shuts off the building exhaust fan. Building isolation is induced automatically by signals from either of three instruments: a gas monitoring system which samples the building exhaust upstream of the holdup duct, a Kanne chamber, or a moving filter monitor. The latter two instruments sample the building exhaust outside of the containment system before dilution and subsequent exhaust from the building stack. An isolation is also induced by a power failure.

The 24-inch building ventilation ducts are sealed by natural gum rubber-seated fast-acting butterfly valves designed for air service. Each duct has two

independent valves; a quick closing valve on the outboard which closes in 1.5 seconds, backed up by an identical valve but with a slower operator which closes in 2.1 seconds. The quick-closing valves are air-to-open and are closed by a mechanical spring. The air supply to the valve is controlled by solenoid operated valves. The valves are closed by interruption of the electrical supply to the solenoids (power failure or signal from the isolation circuitry) or by failure of the building air supply. The backup valves are closed by slower, completely pneumatic operators which require air to open or close. Each valve is supplied from a large reservoir with a demonstrated capacity to operate the valves about 30 times after interruption of the building air supply. Solenoid valves operated by the same circuitry used for the quick-closing valves are used to control the air to the pneumatic operators.

The four 2-inch pneumatic valves on the pneumatic sampling lines are air-driven ball valves also operated by the building isolation circuitry. The ball valves have a solenoid-valve-controlled air supply for opening, and are spring loaded for quick closing action. The operators are designed to close in 1/2 second, and are rated for continuous duty at up to 100 cycles/minute.

To prevent damage to the building from excessively high external pressure, the shell is equipped with two 10-inch vacuum breaker valves. These valves are set to open when atmospheric pressure exceeds air pressure within the building by 0.12 psi, and to pass a minimum of 2000 cfm at 0.20 psi.

Two 18-inch openings in the containment building wall have been provided for possible future extension of horizontal beam tubes. Each opening is closed by a concrete filled, steel plug which is sealed to the flanged, steel liner of the hole by means of a butyl rubber gasket and eight 5/16-inch bolts. To use the opening, the plug would be removed and the hole closed by means of a plate which would be gasketed and bolted to the liner flange. This would provide an air tight seal capable of withstanding an internal pressure of 2 psig.

All drain lines from the building flow into a common sump below the basement floor level. From here the liquid effluent is automatically transferred through a liquid loop seal to the laboratory building waste system. In addition, a closed header system collects effluent water used for cooling auxiliary equipment and some experiments (air compressor, oil cooler for hydraulic system on

biomedical facility, electromagnets on neutron diffraction apparatus, etc.). After analysis, the water passes through a liquid loop seal and is discharged to the city sanitary sewer system. This effluent is analyzed at the same frequency as the secondary coolant which is also blown down and overflows to the city sanitary sewer system.

4.4 Description of Reactor

4.4.1 Reactor Core Arrangement

The basic reactor consists of a 6 foot diameter aluminum vessel containing about 1100 gallons of heavy water in which the fuel assemblies and control elements are installed. The fully loaded core contains 19 fuel assemblies arranged in a triangular array on a 6-inch pitch. This core forms a vertical right cylinder approximately 2 feet in diameter and height. The core is located within the vessel so that the core centerline of the reactor vessel, and the core horizontal midplane is at a point 3 feet above the vessel bottom. Surrounding the core is a 2 foot thick layer of D_20 coolant which serves as a neutron reflector.

Additional neutron reflection is obtained from a 2 foot thick graphite region on the sides and beneath the reactor vessel. The vessel is suspended free of the graphite by mounting a shoulder of the vessel on the thermal shield steel support structure. The lower section of the top shield (lower top shield, Figure 4.12) is located within the upper portion of the vessel and rests upon the internal surface of the same vessel shoulder. Consequently, the load of the lower top shield plug is also carried by the support structure. The top shields are removable for access to the vessel. If necessary the vessel itself can be removed after removal of both top shield plugs.

Degradation of the D_20 by light water is minimized by maintaining an atmosphere above the D_20 that is essentially free of water vapor. This is accomplished by maintaining a helium or nitrogen gas blanket at about 6 inches of water pressure above atmospheric. The vessel closure is sealed by a neoprene gasket installed between the inner surface of the reactor vessel and the outer surface of the lower top shield (see Figure 4.12)

The volume between the outer reactor vessel wall and the inner thermal shield face is called the graphite region. It is filled with 4-inch x 4-inch stringers of grade AGOT graphite. During reactor operation, the thermal neutron flux will significantly activate the natural argon contained in air in this region. To minimize the external argon activity, all of the known accessible penetrations into this graphite region have been sealed. Facilities are installed for purging of the region with helium or other gas, but experience has shown that minimum argon-41 release is obtained with no gas purge.

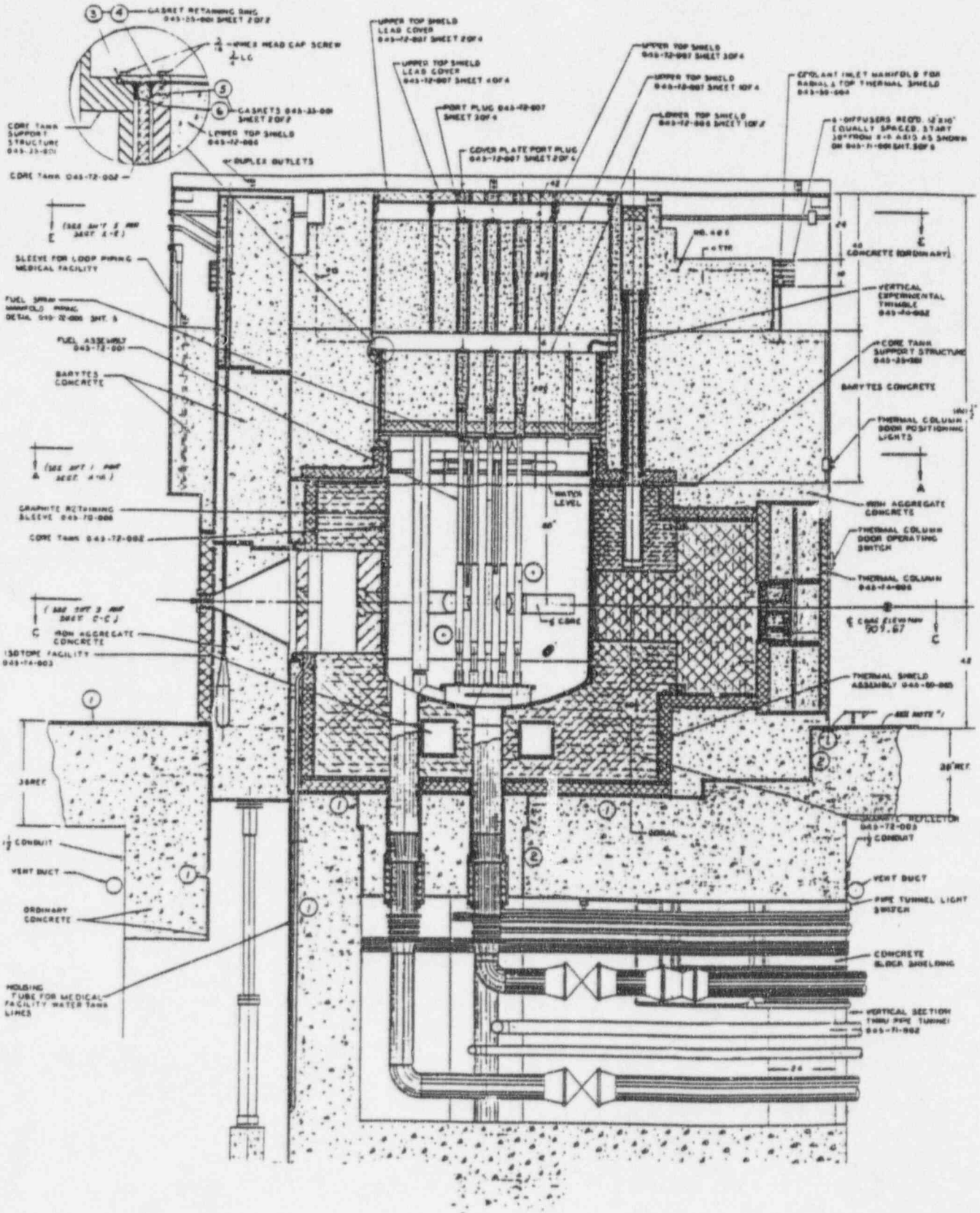
The annular biological shield, containing appropriate penetrations for the reactor experimental facilities, extends approximately 5 feet outward from the thermal shield and completes the reactor structure. The general arrangement of the reactor is shown in Figures 4.12, 4.13, 4.14, and 4.15.

4.4.2 Reactor Vessel

The reactor vessel is fabricated of type 1100 and 6061 aluminum alloys and is designed to withstand an internal pressure of 9 psig. The vessel is cylindrical in shape, 10 feet - 4 inches high with 3/8-inch thick walls. The upper/section is 6 feet - 6 inches in diameter, providing a mounting shoulder 2 feet - 6 inches down from the top edge, which supports the internal top shield and upon which the vessel is mounted.

The vessel walls contain 11 re-entrant beam hole nozzles and 2 through-tube nozzles to permit either the withdrawal of neutron beams to externally mounted apparatus or the installation of experiments adjacent to the reactor core. In addition, there are 4 re-entrant nozzles located just below the vessel shoulder.

Figure 4.12 Vertical Section D-D Through Reactor



SECTION D-D

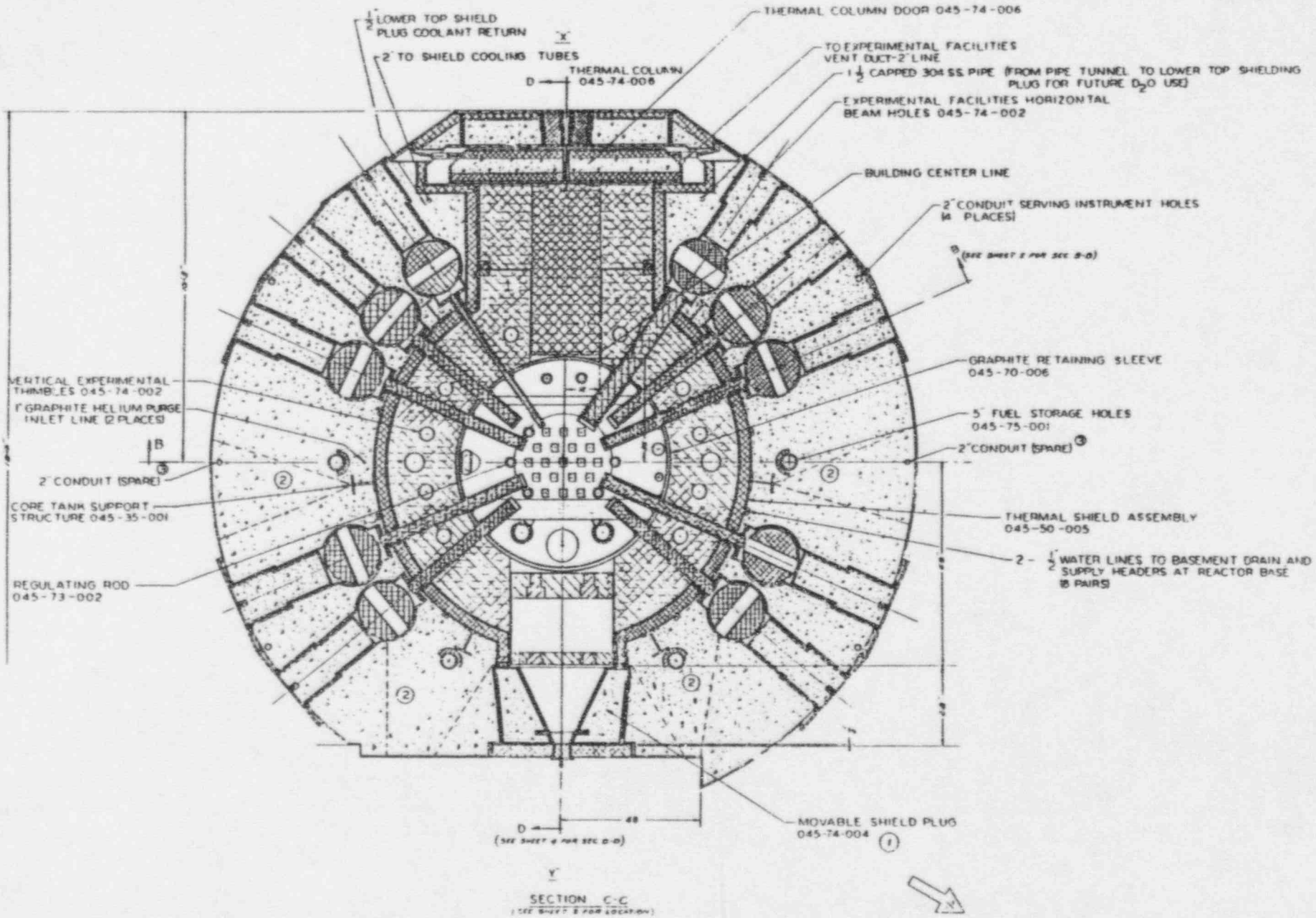
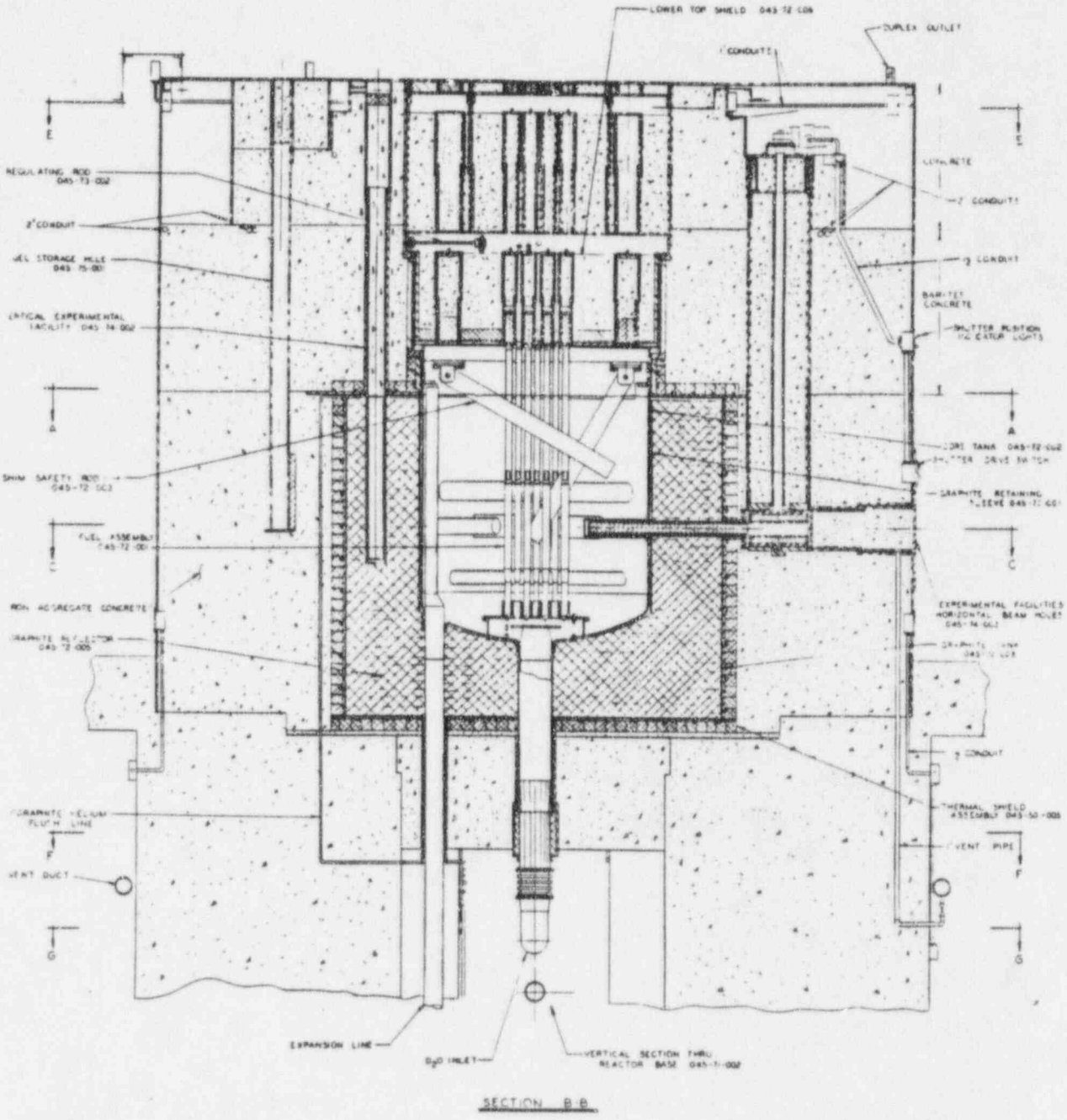


Figure 4.13 Horizontal Section C-C Through Reactor



SECTION B-B

Figure 4.14 Vertical Section B-B Thru Reactor.

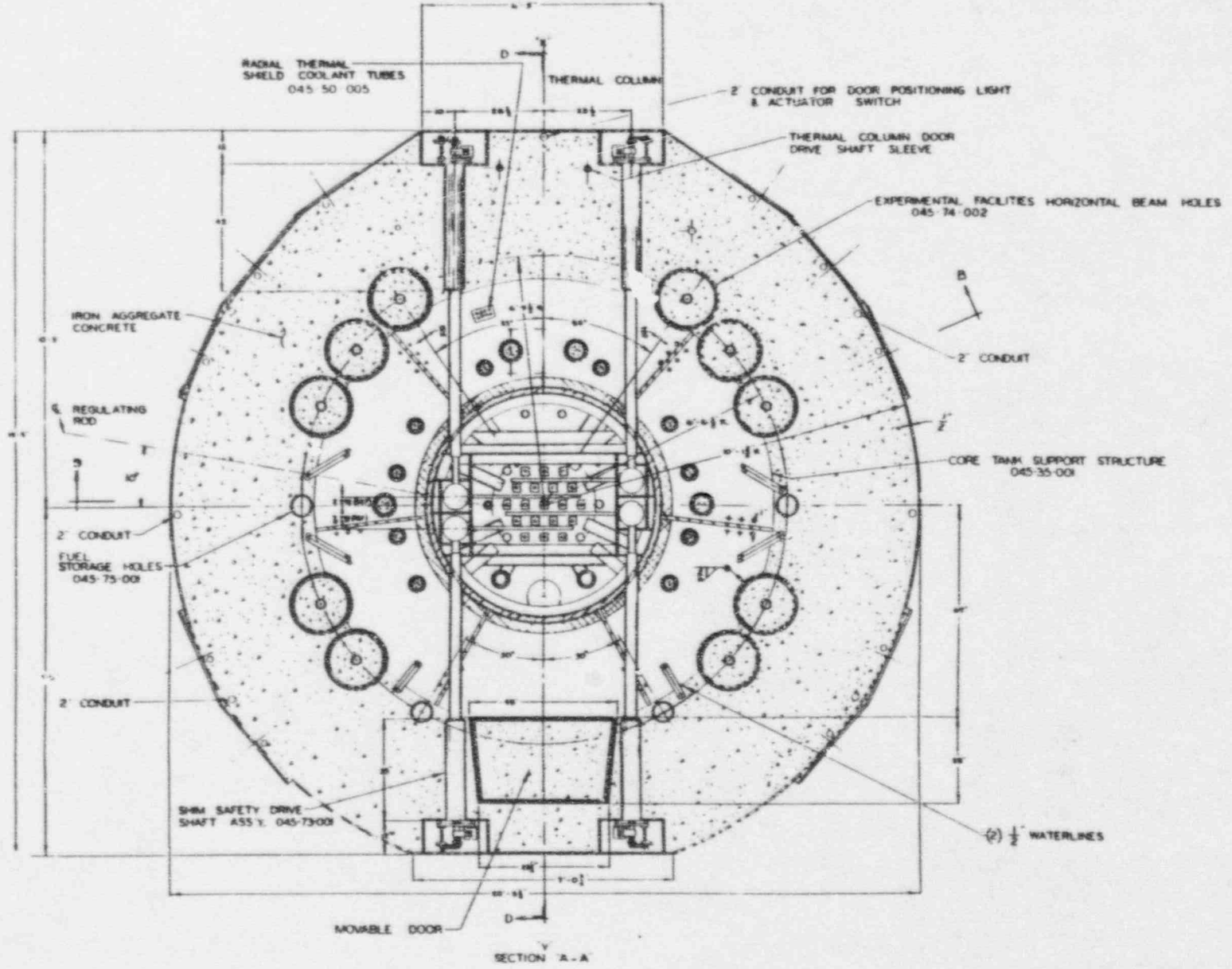


Figure 4.15 Horizontal Section A-A Thru Reactor.

These nozzles permit the passage of the shim-safety blade drive shafts to the internally mounted blades. The shim-safety blades are mounted on the welded aluminum plate structure which spans the reactor vessel immediately below the vessel shoulder.

The lower head of the vessel is dished and is penetrated by the coolant inlet and outlet pipes, moderator overflow and drain lines, and the over pressure relief duct. The lower head also contains the core support plate and guide tube assembly upon which the core is mounted. These general features are shown in the previously mentioned figures.

4.4.3 Fuel Elements

The standard HEU fuel element for the GTRR contains 16 individual curved aluminum-uranium alloy plates. The fuel matrix is 0.020 inches thick, 2 1/2 inches wide, and 23 1/2 inches long. Each plate is clad with type 1100 aluminum alloy 0.015 inches thick, 2.848 inches wide, 25 inches long and has a 5 1/2 inch radius of curvature. The cladding is applied by the "picture frame" method used in fabricating the MTR fuel. This process develops a metallurgical bond between the fuel alloy and cladding at all interfaces. Each plate will contain approximately 11.75 grams of U-235.

The standard LEU fuel element contains 18 individual curved aluminum U_3Si_2 plates. All the external and internal dimensions of LEU fuel elements are identical to those of HEU fuel elements except the spacing between the individual plates has been decreased to accommodate 18 instead of 16 plates. Each LEU plate contains 12.5 grams of U^{235}. The cladding for LEU plates is aluminum type 6061.

The maximum size core contains 19 assemblies. The fuel bearing section of the assembly is a completely enclosed box 2.959 inches by 2.772 inches by 27 1/2 inches long. Coolant flow passages, nominally 0.106 inches thick by 2.583 inches wide for HEU fuel and 0.089 inches wide for LEU fuel, are obtained by inserting the edges of the fuel plates into longitudinal slots machined in the side plates. The fuel plates are permanently fastened to the side plates.

The fuel section is equipped with a lower locating end-fitting and an upper box extension piece and mounting flange. These items are attached to the fuel

section by inert gas shielded, electric arc fusion welds. The mounting flange of the upper box extension piece is bolted to the underside of the lower top shield port plug which supports the assembly and provides top alignment. The lower locating end-fitting is inserted in a guide tube of the core support plate accomplishing bottom alignment. This method insures proper location of each assembly in the desired lattice position and, because of the weight of the supporting shield plug, provides positive hold down action against upward coolant flow forces. The standard GTRR fuel assembly is shown in Figure 4.16.

In addition to the standard element described above, two special removable plate elements are available. These are identical to the standard element except that the lower locating end fitting and lower fuel spacer (comb) can be removed by screws. This will permit the central 10 fuel plates (which are not permanently fastened) to slip out of their grooves in the side plates.

Additionally, two dummy elements are available for hydraulic testing. These are identical to the standard element except that the fuel plates contain no uranium.

4. 4. 4 Control Elements and Drives

The reactor is controlled by means of four shim-safety elements and one regulating element. The shim-safety elements are flat, hollow blades consisting of cadmium metal 0.040 inch thick, clad inside and out with type 6063-T6 aluminum alloy 0.083 inch thick. These blades weigh approximately 20 pounds and are 5.5 inch wide by 1 inch thick. The hollow center is filled with helium and sealed. The cadmium section has a length of 45.5 inches, and the length from pivot point to end is 60.75 inches. The regulating rod is a 24 inch long tube of cadmium metal, 1.380 inch I.D. and 1.420 inch O.D., jacketed inside and out with 0.040 inch of type 1100 aluminum alloy.

The shim-safety blades are mounted at the top of the reactor vessel, and swing through the core between adjacent rows of fuel assemblies. The blades are driven through their full travel of 55 degrees of arc by horizontal shafts which are engaged

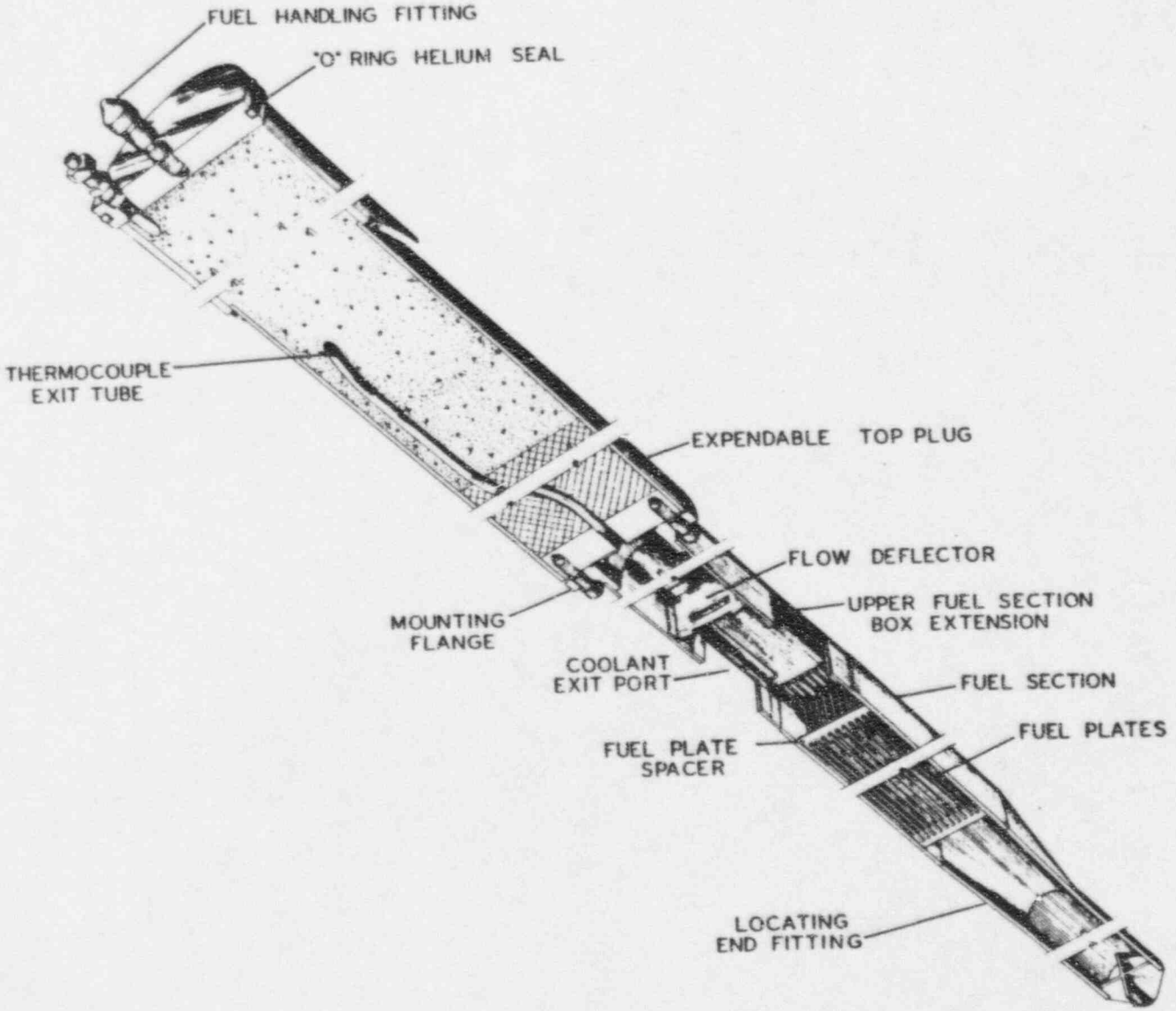


Figure 4.16 Perspective of Fuel Assembly and Plug.

through electromagnetic clutches to the drive motors. The drive assembly incorporates a spring which is compressed as the shim-safety blade is withdrawn, thus insuring rapid insertion of the blade when the clutch is de-energized by a scram signal. The position versus time of each shim-safety blade has been measured by electronic and high-speed photographic means; the results are represented in Figure 4.17. From this figure it is evident that the rods are largely inserted within 0.4 seconds after rod motion commences.

Electronic measurements have been made regularly of the interval between initiation of a manual scram and the time at which the blade position is reached which represents a 90% insertion of the total rod worth. These times are typically 450 to 500 milliseconds. A representative measured time between manual scram initiation and the actual commencement of blade motion is 47 milliseconds.

A relatively long amount of time is spent in the last several degrees of motion after the blade has engaged the dashpot and before it trips its down-limit microswitch. The shim-blade "down" position is 61 degrees below the horizontal. Position indication is obtained by selsyn pairs connected directly to the horizontal shim-safety blade shaft. The positions are displayed on dial indicators at the control console.

The purpose of the shim-safety blades is to control large amounts of reactivity. The worth of these blades range from (4% to 5.5%). Recent calibrations made with a 17 element HEU core are shown in Figure 4.18. While these blades may be inserted individually or as a group, withdrawal is limited to individual element movement at a maximum speed of 0.2 degree/second.

The regulating rod operation is in a vertical direction through the core and is intended for fine control only. It is immobilized by a reactor scram. The twelve-inch total motion results in the bottom of this rod moving from the core centerline in the "in" position to the top of the core in the "out" position. The rod is coupled directly to a ball nut which is driven by a lead screw. These components are located in a lower port plug of the top shield. The lead screw is driven through a right

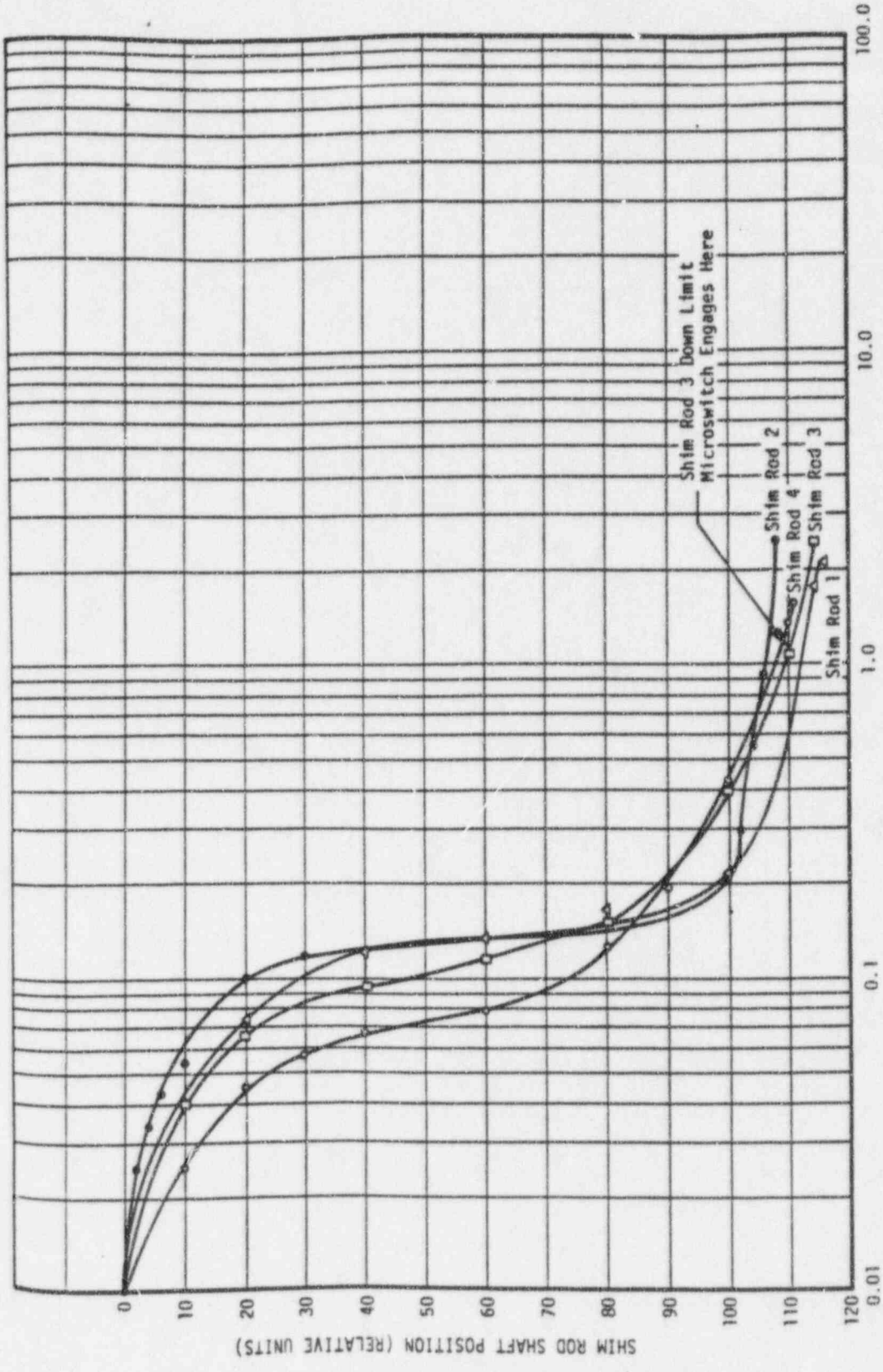


Figure 4.17. Shim-Safety Control Rod Position as a Function of Time Following a Drop from the Banked-Critical Position.

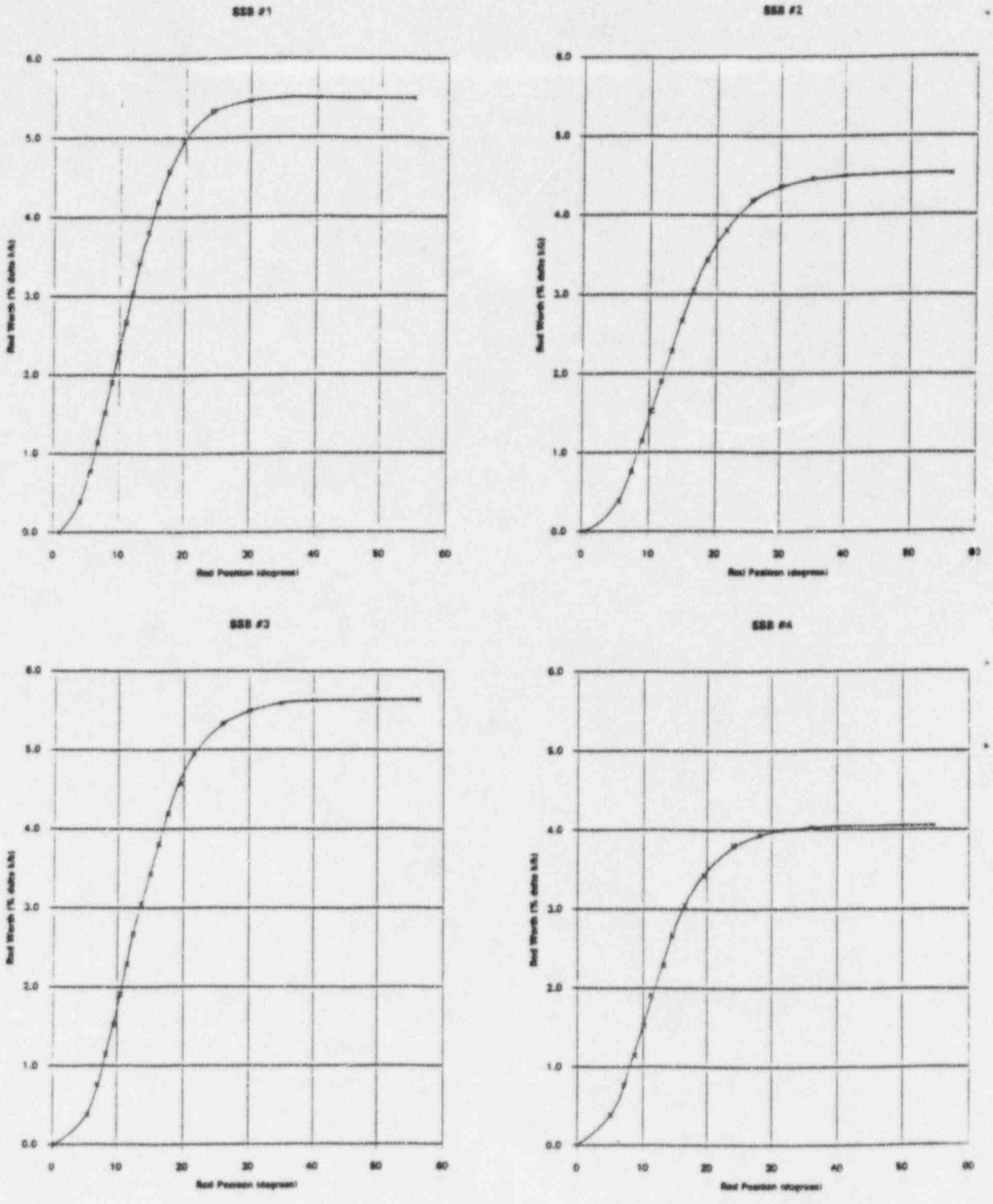


Figure 4.18 Shim-Safety Blade Calibration Curves for HEU Core

angle gear box located at the top of the lower port plug, by a horizontal shaft connected to the drive motor mounted at the external face of the biological shield. Position indication is provided by a selsyn pair connected to the horizontal shaft; this rod position is displayed on the control console.

The regulating rod is used for power level control and adjustment after a critical position has been attained by the shim-safety blades. Its reactivity worth for an HEU core is about 0.4%. The speed of this rod is fixed at 0.2 inch/second, which limits the maximum rate of reactivity charge to less than 0.01% per second. A regulating rod calibration curve is shown in Figure 4.19.

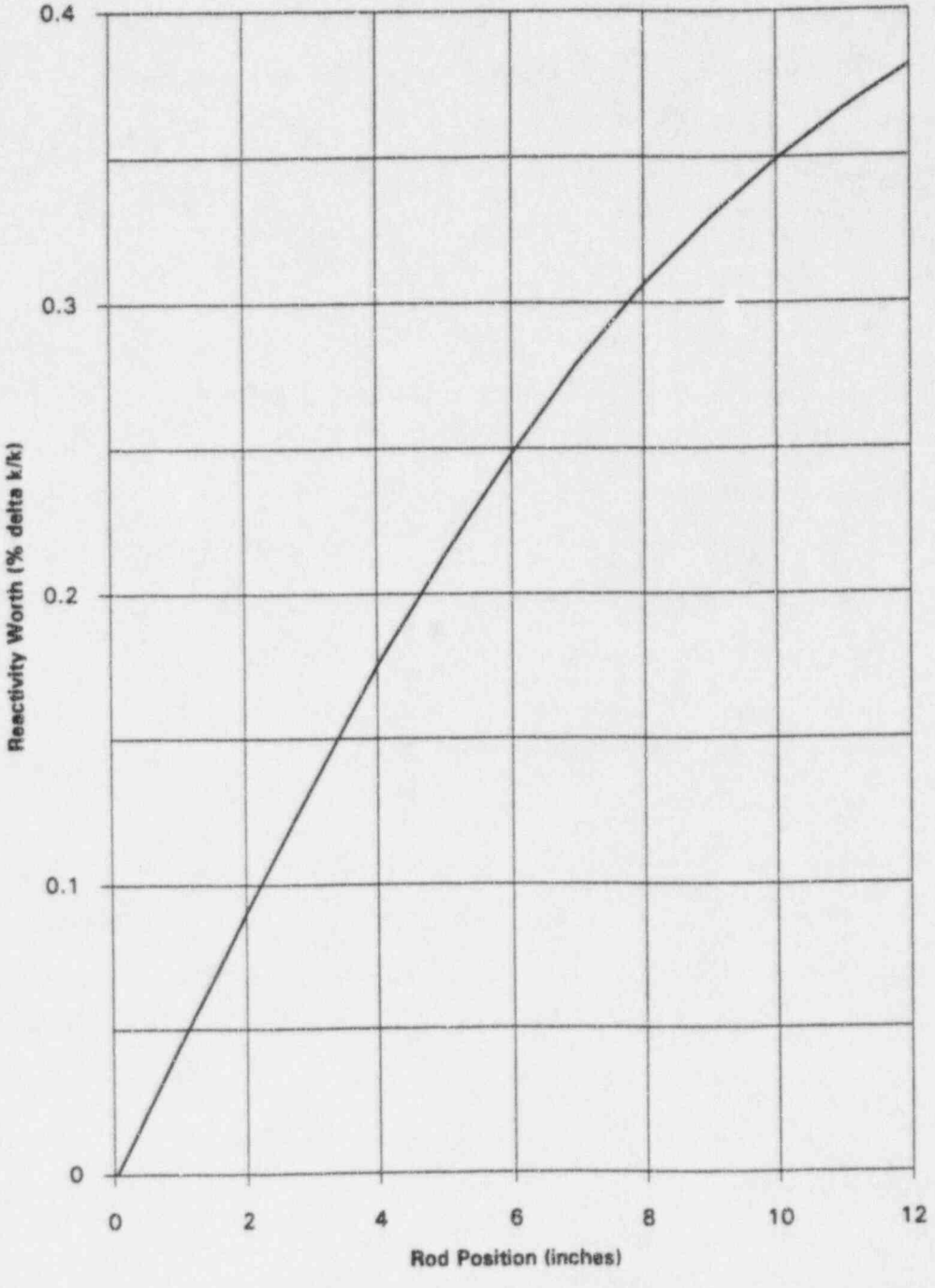
4. 4. 5 Biological Shield

The biological shield consists of layers of boral, steel, lead, and concrete surrounding the graphite reflector. The first layer is a 1/4 inch thick sheet of boral staked to the inside of the steel shield tank enclosing the graphite. A 3 1/2 inch thick layer of lead, containing the copper tubes of the shield cooling system, is cast in the annular space between the external surface of the steel shield tank and an outer steel retainer. By pouring molten lead into this region, an adequate thermal bond between the shield tank and the cooling tubes has been assured. This section of the shield is commonly termed the thermal shield. Its major function is to reduce heating in the concrete portions of the biological shield as a consequence of absorption of radiation from the core.

The outermost portion of the biological shield is a thick layer of concrete completely enveloping the reactor. In order to minimize the thickness of this layer, very dense concrete is used in preference to lighter, ordinary concrete in that portion of the shield immediately surrounding and adjacent to the core proper.

The concrete shield is composed of three cylindrical sections of concrete poured one on top of the other and sheathed with a 1/2 inch thick steel plate. The first is a 20 foot - 3-1/4 inch O.D. by 10 foot - 9-1/4 inch I.D. cylinder of very dense (270 pounds per cubic foot) iron punchings and limonite concrete. This section extends from the reactor floor level to the top of the shield tank. The second section

Figure 4.19 Regulating Rod Worth for the GTRR with HEU Core



is a 20 foot - 3-1/4 inch O.D. by 6 foot - 10-1/2 inch I.D. cylinder of barytes concrete (215 pounds per cubic foot) about 4 feet - 6 inches high. The top section is a stepped cylinder of ordinary concrete 16 feet - 9-1/4 inch O.D. by 7 feet - 2 inch I.D. at the base. The O.D. is reduced to 11 feet - 0 inches approximately 2 feet from the top, since the full shielding thickness is not needed at this elevation.

The shield contains a large number of both horizontal and vertical penetrations to accommodate the various experimental facilities. Additional penetrations are provided for the passage of control element drive shafts, coolant and service piping, service wiring, ventilation, and instrumentation. All penetrations through the shield are stepped and are equipped with permanent steel liners. Where these penetrations extend into the D_2O or graphite regions, the liners are seal-welded to the shield tank. Major openings in the shield, such as the thermal column, are equipped with movable shielding doors or shutters. All other experimental hole are provided with stepped shielding plugs to prevent hazardous radiation streaming.

The ordinary concrete reactor foundation provides adequate shielding for personnel working in the area, except in the pipe tunnel directly beneath the reactor. At this point there is 3 feet 2 inches of ordinary concrete and 3 1/2 inches of lead. Access to the pipe tunnel, therefore, is not permitted during reactor operation. Entry to the pipe tunnel is through the process equipment room only; the doors to this room are locked during reactor operation, and opening of either door activates an annunciator alarm in the control room.

The reactor top shield consists of two large diameter shielding plugs located directly above the reactor vessel. The first of these is the internally mounted lower shield discussed in the Section 4.4.1. The bottom plate of this shield plug is stainless steel containing 1% by weight of boron. Immediately above this plate is a lead thermal shield and cooling arrangement similar to that described previously. The balance of the 2 foot - 5 1/2 inch shielding depth is filled with iron punchings and limonite concrete.

The upper section of the top shield rests on an internal shoulder provided in the collar of the support structure, leaving a space of 6 inches above the top surface of the lower internal shield. This plug is 7 feet-1 inch O. D. by 4 feet-7 1/2 inches

deep and is essentially a flat-bottomed steel can containing punchings and limonite concrete to a depth of 3 feet-3 inches. A 4 1/2 inch thick cover is supported on the walls of the can, leaving a 6 inch deep wire-way between the underside of the cover and the top of the concrete.

Both shields contain individually plugged ports through which fuel assemblies and experiments may be inserted into the reactor core or reflector regions. The fuel port plugs contain provisions for the passage of thermocouple leads. All the port plugs and the ports are stepped to minimize radiation streaming.

4.4.6 Experimental Facilities

The reactor is equipped with numerous horizontal and vertical experimental facilities to be used for the extraction of beams of fast and slow neutrons and for the performance of irradiations within the facilities. Measured neutron fluxes in some of these facilities are given in Table 4.2 and Figure 4.20, for an HEU core. The expected usage of the different experimental openings is described in the following sections and their location is shown in Figure 4.21.

4.4.6.1 Vertical Experimental Facilities

The top of the reactor contains a total of 46 vertical penetrations of which 41 are for experimental use, including fuel element positions. Twenty-seven of these are located in the D_20 region within the reactor vessel. The remaining 14 are dispersed through the graphite reflector region. All penetrations other than fuel element positions are provided with double aluminum thimbles. The outer thimble supports and protects the inner sample thimble. Experimental thimbles located in the graphite reflector region are equipped with an "O" ring to effect a gas seal between the O.D. of the outer thimble and the I.D. of the penetration liner. Experimental penetrations in the D_20 region are sealed by an 'O" ring located in the lower top shield port plug.

Table 4.2 THERMAL NEUTRON FLUX IN THE GTRR, HEU CORE

SYMBOL	DESCRIPTION	SIZE	THERMAL NEUTRON FLUX (MAX)	
			1 Mw Measured	5 Mw Projected
H-1	Horizontal Beam Tube	6" ID	2.4×10^{13}	1.2×10^{14}
H-2 to H-9	Horizontal Beam Tube	4" ID	$1.5\text{-}2.4 \times 10^{13}$	$.75\text{-}1.2 \times 10^{14}$
H-10	Horizontal Beam Tube	2" x 6"	2.2×10^{13}	1.1×10^{14}
H-11, H-12	Horizontal thru-tube	6" ID	2.0×10^{13}	1.0×10^{14}
H-13, H-14	Horizontal thru-tunnel	12" x 12"	5×10^{12} *	2.5×10^{13}
H-15, H-16	Pneumatic Tube	1-½" ID	1.3×10^{13}	6.5×10^{13}
V-1 to V-19	Fuel Element Positions	3" x 3"	3×10^{13}*	1.5×10^{14}
V-20 to V-23	Vertical Thimble (core)	2-5/8" ID	2.3×10^{13}	1.1×10^{14}
V-24, V-25	Vertical Thimble (reflector)	3-1/2" ID	4×10^{12} *	2×10^{13}
V-27, V-28	Fast Flux Facility	4" ID **	3×10^{12}*	1.5×10^{13}
V-33 to V-42	Vertical Thimble (reflector)	4" ID	8.4×10^{11}	4.2×10^{12}
V-43 to V-46	Vertical Thimble (reflector)	6" ID	8.1×10^{11}	4.5×10^{12}
	Bio-Medical Facility	port 4" ID	1.0×10^{10} (at port face)	5.0×10^{10}
		room 10' x 12'	4×10^{9} (out 3")	2.0×10^{10}
	Thermal Column	5' x 5'	1.7×10^{12}	8.5×10^{12}

*calculated value
** without U-235 converter

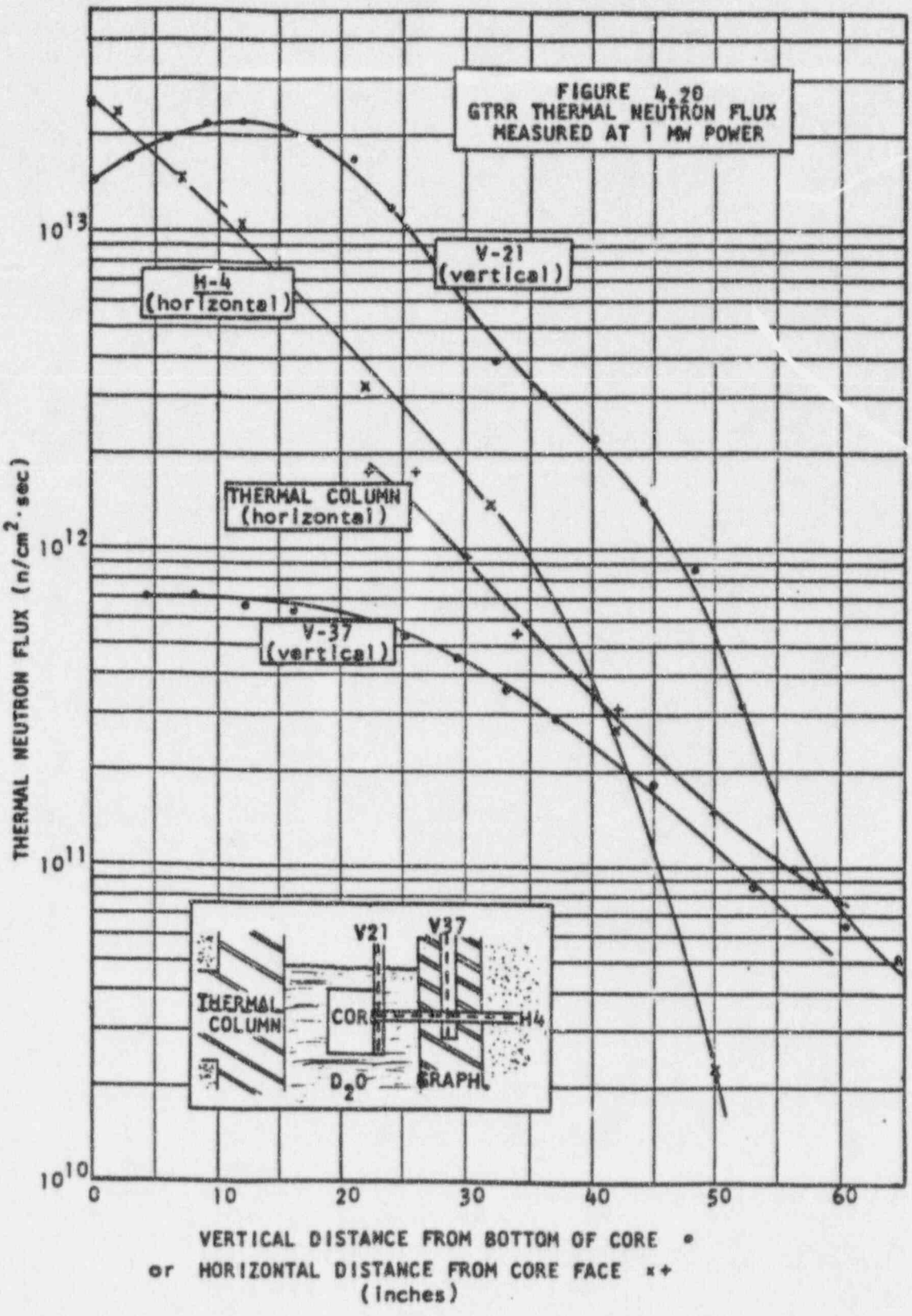


Figure 4.20 GTRR Thermal Neutron Flux Measured a 1 Mw Power in HEU Core

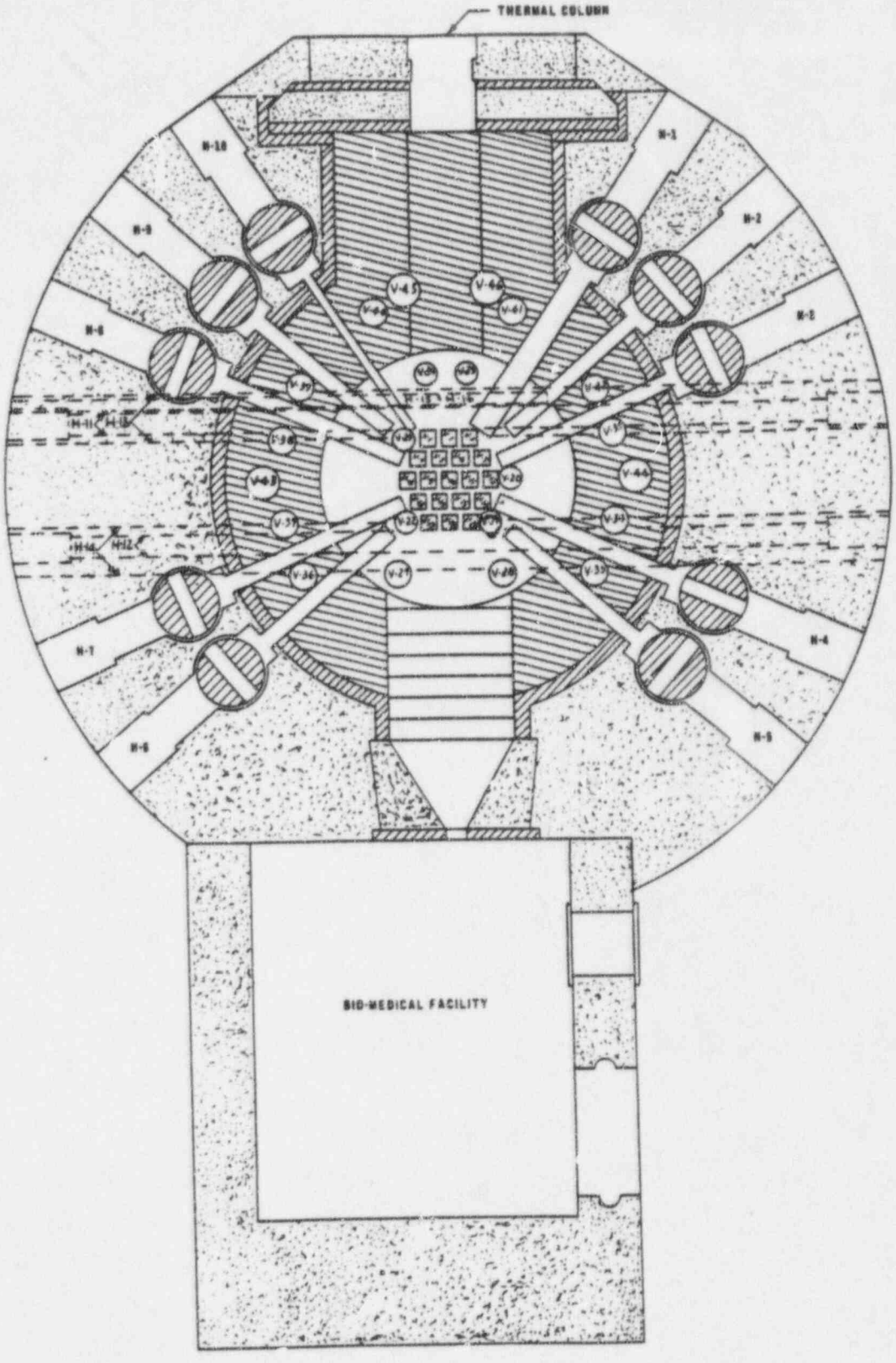


Figure 4.21. Horizontal Section of GTRR at the Core Midplane.

Nineteen of the D_20 region openings are fuel assembly positions, any of which could be used for irradiations. A second group of thimbles, designated V20 through V23, are extensions of the core lattice, but are intended primarily for sample irradiations. These thimbles, 3 1/4 inch diameter, are located peripherally about the lattice and extend down to the plenum chamber of the core support assembly. Stations V24 and V25 are similar to this group except that each position is approximately 28 inches from the center of the core and 8 inches inside the vessel wall. The thimbles extend to just below the core midplane.

Openings V27 and V28 are 4 inch I.D. fast flux facilities located just inside the D_20 region. These thimbles can be used for irradiation of specimens in a neutron flux which is predominately fast. The conversion of thermal neutrons to fast can be accomplished by building into the inner (sample) thimble a sheet of enriched uranium. When used in this manner the outer thimble is connected to the plenum chamber. This allows the D_20 coolant to flow upward between the outer and inner thimbles to provide cooling for the converter plate. There are no plans to procure a uranium converter in the immediate future. Until a definite need arises, V27 and V28 will be operated in the same manner as the other vertical facilities for irradiations requiring well thermalized neutrons.

Stations V33 through V42 are 4 inch I.D. vertical thimbles. All 10 are in the graphite region and approximately 8 inches outside the reactor vessel wall. They extend to a point just below the midplane of the core. V45 and V46 are 6 inch I.D., vertical thimbles which stop near the top of the thermal column approximately 30 inches above the core midplane. They are located in the graphite region approximately 12 inches outside the reactor vessel wall. Stations V43 and V44 are similar, except in depth, to V45 and V46. These thimbles extend downward to the midplane of the core.

4.4.6.2 Horizontal Experimental Facilities

The reactor contains 24 horizontal openings, a thermal column, and a bio-medical irradiation facility. Stations H1 through H10 are horizontal beam ports, all

of which lie in the horizontal plane passing through the center of the reactor. Stations H1, H3, H4, H7, H8, and H10 are so located that they look directly at fuel elements and thus give good fast or epithermal neutron beams. The locations of the horizontal openings and their sizes are as follows:

a. H1 is a 6 inch I.D. beam port which extends into the D_20 region to a point 16 inches from the core axis.
b. H3, H4, H7, and H8 are 4 inch I.D. beam ports similar to H1.
c. H2, H5, H6, and H9 are 4 inch I.D. beam ports which extend to a point approximately 20 inches from the core axis.
d. H10 is a rectangular beam port measuring 2 inches by 6 inches which extends to a point 15 inches from the core axis.

All 10 horizontal ports are provided with rotating shutters so that the beam intensity may be reduced to avoid the danger of overexposure to radiation while adjustments are made to equipment. The shutter assembly extends to the top of the reactor shielding and is entirely removable; however, only the lower portion revolves during opening and closing of the port. Each shutter is sealed at the top to prevent the leakage of argon-41 from the beam port thimbles into the ventilation system during reactor operation. These shutters are manually operated from the top of the reactor shield structure. Indicating lights showing open or closed positions are located on the face of the shield above the beam port opening and in the reactor control room. Provisions are included to supply utilities to these locations.

H11 and H12 are 6 inch tangent through-tubes which extend across the reactor. They pass through the D_20 region tangent to the core. H11 is located near the top of the active core and H12 on the opposite side near the bottom of the core. These holes are particularly useful for performing engineering-type experiments requiring the circulation of a coolant.

H13 and H14 are 11 inch square through-tubes. They pass through the graphite region below the reactor vessel and are intended primarily for sample irradiations.

H15 and H16 are twin 1 1/2 inch I.D. tubes, housed in a common reentrant nozzle, which pass through the D_20 reflector tangent to the core. They are intended

as pneumatic sample handling devices for experiments requiring irradiations of short duration.

H17 through H22B are the eight instrument positions. They contain the ion chambers and counters required for the operation and control of the reactor.

A thermal column, 5 feet square, is provided as an extension of the graphite reflector. It is fitted with a shutter and with heavy shielding located at the outer face. The shutter opens horizontally giving a port 4 inches by 4 inches or 16 inches by 16 inches. A number of removable graphite stringers which extend up to the reactor tank wall are provided in the thermal column.

A shielded room for bio-medical research is located on the side of the reactor opposite the thermal column. This facility is designed to allow accurate exposures of biological specimens to a wide angle beam of thermal neutrons with a relatively low background of fast neutrons and gamma rays. It is fitted with bismuth gamma shield, a collimator, shutter, and provisions for a converter plate system. The opening in the reactor is surrounded by the shielded room. The use of a converter plate will permit the fast flux to be increased to about 10^{10} n/cm^2/sec with a corresponding decrease in thermal flux and increase in gamma rays.

The bio-medical facility shutter is operated by means of a hydraulic cylinder which is capable of opening or closing it in 20 seconds or less. The system is fail-safe in that power or equipment failure closes the shutter, but keeps the hydraulically operated entrance door in place. The required operating equipment is located in the basement area. This same area houses the H_2O system which cools the bismuth shield. This system is diagrammed in Figure 4.22.

The shielded room is approximately 10 feet by 12 feet inside and is shielded with 2 feet of barytes concrete along the sides. The back wall, which is subject to beam impingement, consists of 4 feet of ordinary concrete covered by 1/4 inch of boral and 1/2 inch of lead. The roof is ordinary concrete 3 feet thick. Access to this area is through a vertically moving, hydraulically operated, shield door. Emergency access, in the event of door failure, is possible through a manhole in the ceiling of the room. This manhole may be removed by means of the building crane. A ladder is permanently installed on the wall below the manhole.

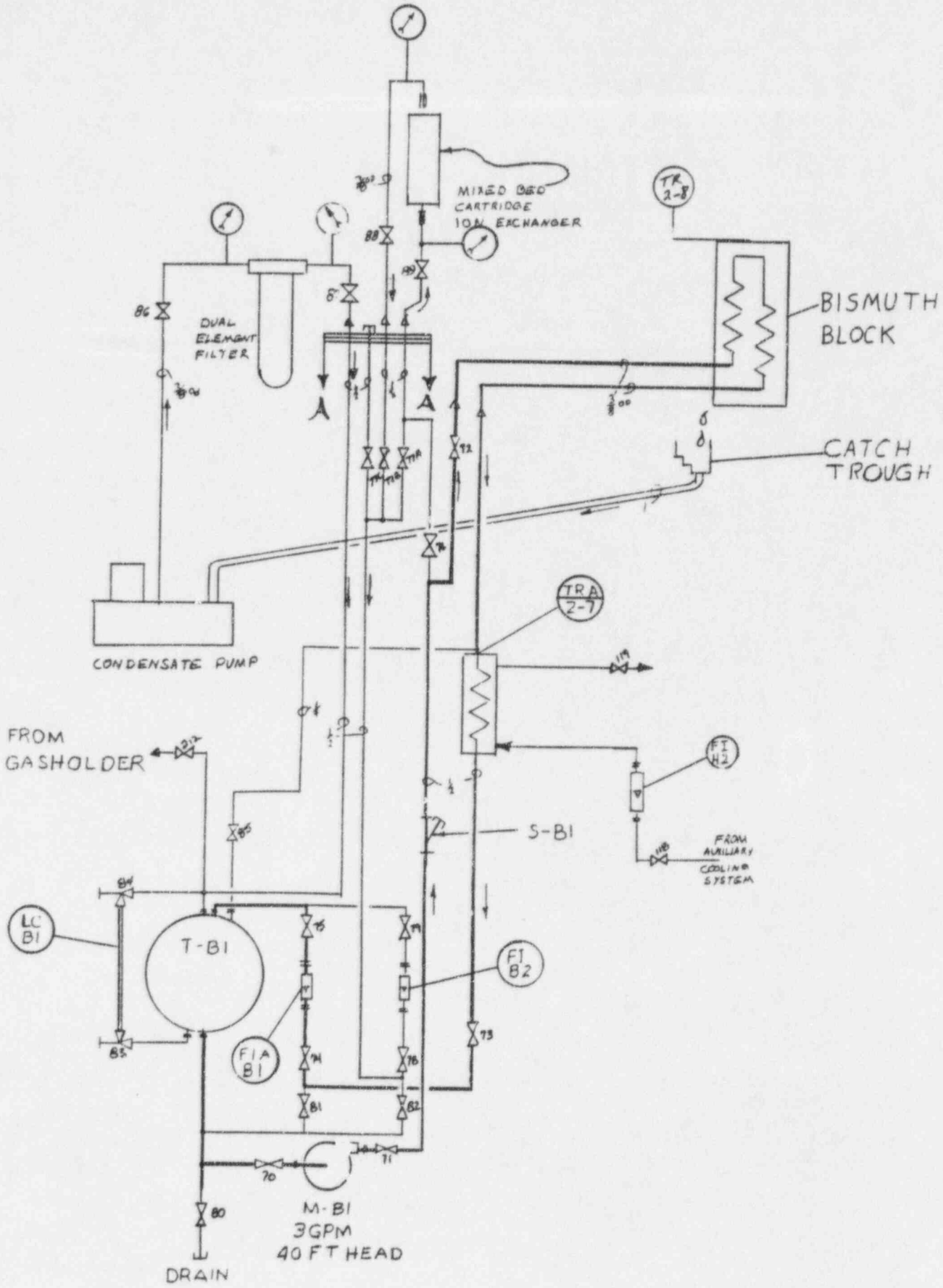


Figure 4.22 Biomedical Facility Cooling System Diagram

4.4.7 Reactor Instrumentation

Operating experience with the GTRR instrumentation since 1964 has demonstrated that it is a safe, practical system. Revisions of the instrumentation for 5 MW operation retained the desirable basic features of the previous system, while expanding it to meet the more stringent requirements of higher power operation.

Calibrating, testing and maintaining the GTRR instruments are continuous processes. Upgrading, under 10 CFR 50.59 requirements, has been implemented on several occasions. The basic instrumentation package of the GTRR will function as intended and safely for the next twenty years.

4.4.7.1 Nuclear Instrumentation

The instrumentation system contains seven permanently installed nuclear instrumentation channels: a count rate meter channel, two micro-micro ammeter channels, two log-N and period amplifier channels, and two power level trip amplifier channels. The operating range of the count rate meter is from 1×10^{-4} to 10 watts. At or near the critical condition, the micro-micro ammeter circuits begin indicating. These instruments cover a range of from less than one watt to the full power of 5 MW. Duplicate circuits are provided since this channel supplies the power level recorder, power level indicator, and the automatic power control system. At the one watt power level, the log-N and period amplifier instruments become operative. These channels also cover the remainder of the range to full power (5 MW). The last channels to become operative are the two power level trip circuits which cover the 1 kW to 5 MW range.

4.4.7.2 Reactor Safety Interlock System

The "electronic scram" via the transistor switch eliminates relay operation time, thus providing a more rapid scram of conceivable value for power and period scram conditions. Scram signals in the form of contact openings of relays and of mechanical switches are denoted as "electromechanical scrams".

An analysis of 5 MW operating conditions and malfunctions showed that fuel plate burnout was an intermediate step in the more serious of conceivable

accidents. All conceivable conditions leading to burnout were analyzed, and instrumentation required to detect each condition was specified. This analysis led to the definition of five "reactor safety" circuits which could detect and prevent burnout: reactor power, reactor period, D_20 temperature, D_20 flow, and D_20 level in the core tank. The instrumentation system was then redesigned to give independent, redundant electronic scram circuits with electromechanical backup for each of the five "reactor safety" parameters. The reactor safety circuits, shown in Figure 4.23, provide separate systems from the detector, through to the power supply for the scram switches, which are now operated in pairs rather than four in parallel. Redundancy is achieved through the electronic scram circuitry, and signal inputs to the Trip Logic Units are taken from a point as near the detector as possible to provide additional independence from display signal conditioning and electromechanical scram circuitry. Although each electronic scram circuit drops only two shim safety blades, the remaining two are released by the electromechanical circuit for a scram initiated by any instrument channel. In addition, electronic signals from each power and period instrument go to both Trip Logic Units, thus providing a "fast" scram of all four blades for power and period scram conditions.

Because more than one of the five safety parameters are affected by conditions leading to burnout, the simultaneous failure of four or more circuits would be required to prevent a desired scram. An electromechanical scram is also initiated by loss of power to any instrument in the scram circuitry, and by several malfunction and incorrect switch position interlocks. Details of these interlocks, and of the remainder of the safety interlock system are given in Table 4.3.

A safety circuit checkout procedure and instrument rack have also been devised to allow rapid, accurate measurement of the trip point of each of the reactor safety circuits, and verification of ability to scram through either electronic scram path or the electromechanical scram path. A safety system check will be part of the normal cold-startup procedure. The reactor will not be operated without two power level trip circuits, two Trip Logic Units, and two Trip Actuator Amplifiers functioning properly in the scram circuit.

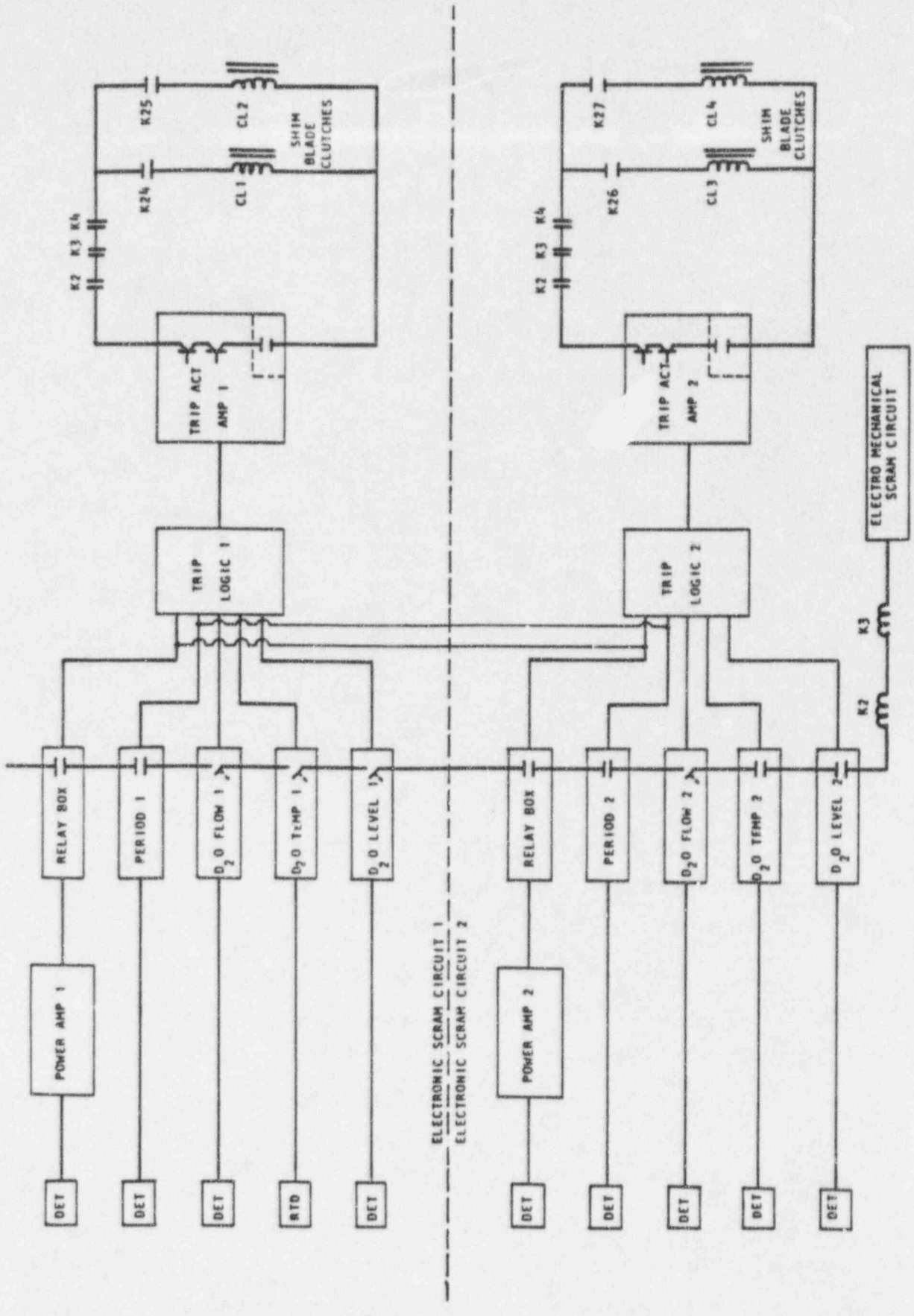


Figure 4.23. Schematic of GTRR Reactor Safety Scram Circuitry.

4.4.7.3 Automatic Reactor Power Control System

The automatic power control system is comprised of a three mode controller, a power set box, and a two-phase servo motor which drives the single regulating rod. The controller compares a set point signal to a signal proportional to actual reactor power in a bridge circuit, and amplifies and conditions the resulting error signal to provide proportioning, rate, and reset action.

Since only the regulating rod is driven by the automatic power control system, the maximum reactivity change due to the system is 0.01% per second (see Section 4.4.4). If the error voltage indicates a difference of more than 10% between the set point and actual reactor power, the power set box activates an annunciator alarm, and switches to the manual control mode, thus interrupting regulating rod motion.

TABLE 4.3 GTRR SAFETY INTERLOCK SYSTEM

Item No.	Circuit Designation	Action				Primary Sensing Element
		Scram	Delay Scram	Prevents Startup	Ann. Only	
1	Power Trip (2 circuits)	x		x		Uncompensated Ion Chamber
2	Period Trip (2 circuits)	x		x		Compensated Ion Chamber
3	Low D_2O Flow (2)	x		x		1. In-line turbine type flow tube 2. DP Switch
4	High D_2O Temperature (2)	x		x		RTD & Thermo-couple in Reactor D_2O outlet line
5	Low D_2O Level in Core Tank (2)	x		x		Fressure Trans-ducer & Differential Pressure Switch
6	Magnet Actuator Amplifier	x		x		Circuit in the actuator amplifier
7	Low Ion Chamber Voltage	x		x		a) Circuit in power supply chassis b) "Trouble" monitor in flux monitors
8	Calibrate Switches off Operate Position	x		x		Circuitry in period and flux monitor chassis
9	Reflector Drain Valves Open	x		x		Mechanically operated switch
10	No D_2O Overflow	x		x		In-line resistance probe
11	Containment Doors Open	x		x		Pneumatic and mechanically operated switches

TABLE 4.3 GTRR SAFETY INTERLOCK SYSTEM

Item No.	Circuit Designation	Action				Primary Sensing Element
		Scram	Delay Scram	Prevents Startup	Ann. Only	
11a	Reactor Isolation Valves not open	x		x		Mechanical Switches
12	High H_2O Temperature	x		x		RTD
13	Low H_2O Flow		x	x		Venturi type flow tube
14	Control Air Low Pressure		x	x		Pneumatically operated switch
15	Low Shield Coolant Flow		x	x		Variable area type flow tube
16	High Shield Coolant Temperature		x	x		In-line thermo-couple
17	High Bismuth Coolant Temperature		x	x		In-line thermo-couple
18	Low Bismuth Coolant Flow		x	x		Variable area type flow tube
19	Control Rods Off Down Limit			x	x	Mechanical limit Switches
20	High Building Radiation			x	x	Several Beta - Gamma monitors
21	Emergency Cooling Tank Low			x	x	Two series - connected switches
22	Low Neutron Count Rate			x	x	Source range monitor (LCRM)
23	Ventilating System Low Flow				x	Flow sensing switches in vent system ducts
24	Radiation High - Ventilation Duct				x	Beta - Gamma Geiger tube
25	High H_2O Coolant Activity				x	Beta - Gamma Geiger tube
26	D_2O Leak				x	Conductivity circuitry

TABLE 4.3 GTRR SAFETY INTERLOCK SYSTEM

Item No.	Circuit Designation	Action				Primary Sensing Element
		Scram	Delay Scram	Prevents Startup	Ann. Only	
27	Outside Automatic Controller Servo Range				x	Circuitry in automatic controller
28	Regulating Rod Low Limit				x	Mechanical limit switches
29	Regulating Rod High Limit				x	Mechanical limit switches
30	Low D_2O Temperature				x	RTD
31	High D_2O Conduc - tivity Before Ion Exchanger				x	In-line conduc - tivity cell
32	High D_2O Conduc - tivity After Ion Exchanger				x	In-line conduc - tivity cell
33	Low H_2O Temperature				x	RTD
34	Low Helium Flow				x	Variable area type flow tube
35	Low Helium Level				x	Mechanically operated switch
36	High Helium Level				x	Mechanically operated switch
37	Low Recombiner Temperature				x	In-line thermo-couple
38	Stack Exhaust High Activity				x	a) Ionization chamber(Kanne) b) Beta-Gamma tube (MAP-1)
39	Process Room Doors Open				x	Mechanical switch
40	CW Basin Low Level				x	Two series connected float switches

4.4.7.4 Emergency Power

Analysis of the probable events following loss of electrical power, or of disaster events accompanied by loss of power was directed at distinguishing between essential power for personnel and reactor safety, and desirable, but non-essential "convenience" power consumption. Items in the essential category are to be provided with battery power sources. Automatic battery-powered lighting is supplied at all locations in the containment building where corrective action may be required following loss of power. Exit paths in both the containment and laboratory building are lighted. Communications will be maintained by a battery-powered backup automatically upon loss of line power. "Walkie-talkies" are available to maintain two-way communications for personnel involved in corrective action following an incident.

Most of the routine portable radiation monitoring devices are now battery powered, and operating units are always available at designated locations in the containment and laboratory buildings.

The "convenience" emergency power will continue to be supplied by the 35.0 KVA natural gas engine-generator set. The system is checked on regular basis. Preventive maintenance is performed on a regular basis, therefore, the system will continue to be available for the next twenty years.

4.4.8 Reactor Heat Dissipation

The major portion of the heat generated by the reactor is removed by the primary D_2O system and transferred to the secondary H_2O system. The remaining heat is extracted by the shield cooling and reactor ventilating systems. The D_2O enters the plenum chamber of the core support assembly located at the bottom of the reactor vessel from which it is discharged to the individual fuel assemblies. Flow is upward through the fuel assemblies and into the vessel at the top of the core. The D_2O is discharged to the pump through a pipe located in the lower head of the vessel. It then passes through heat exchangers located in a shielded area of the containment building basement, is cooled by the ordinary water of the secondary system, and returns to the plenum chamber. The ordinary water of the secondary system flows to a cooling tower, located externally to the reactor containment building, where the heat is released to the atmosphere. Schematic flow diagrams of

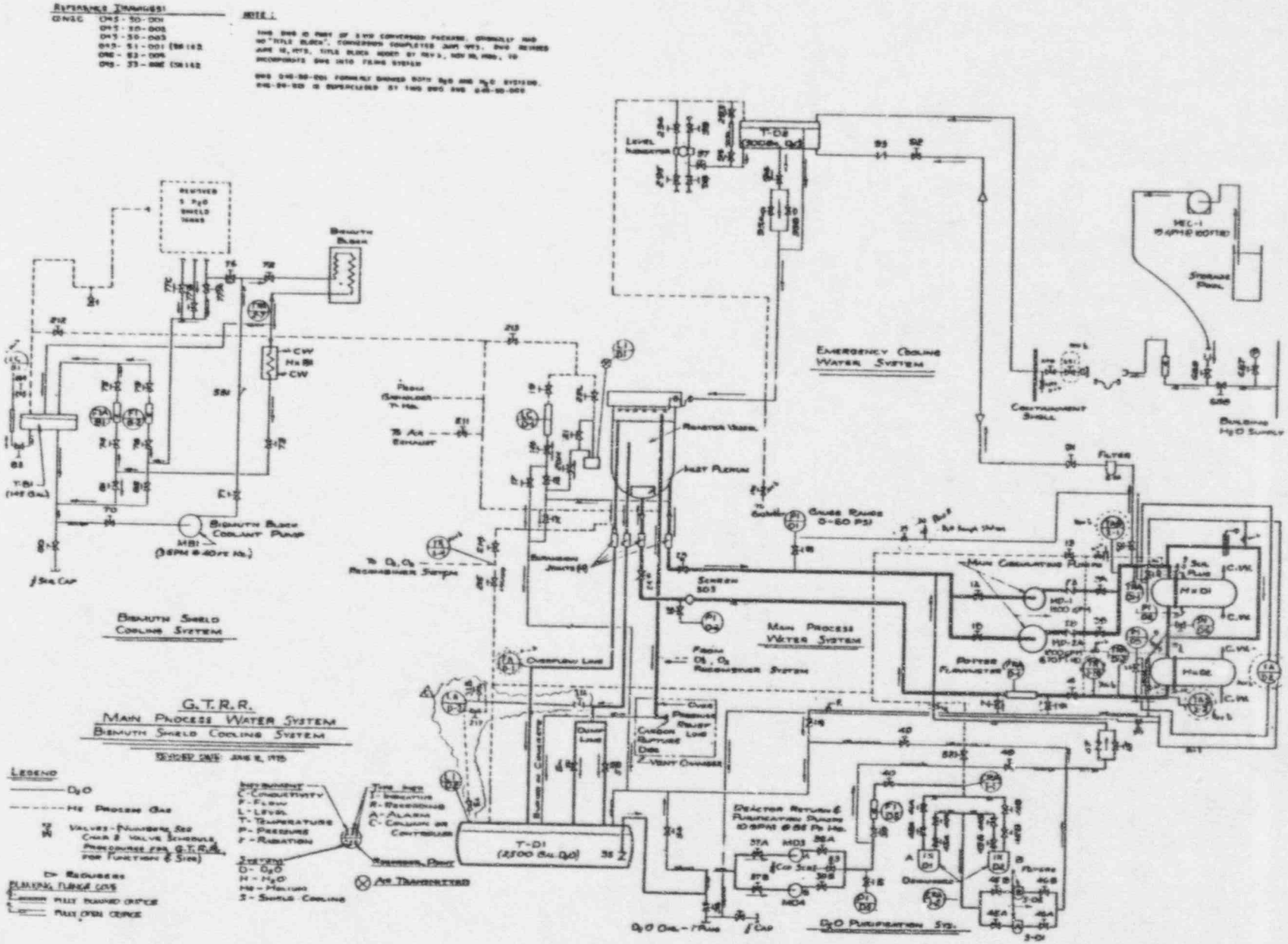


Figure 4.24 GTRR Main Process Water System

the D_2O primary and H_2O secondary cooling systems are given in Figures 4.24 and 4.25, respectively. A brief description follows of the primary and secondary systems as modified for 5 MW operation.

4.4.8.1 Primary D_2O System

The main components of the D_2O system are the reactor vessel, the storage tank (TDl), the main and standby D_2O circulating pumps, and the two heat exchangers. The D_2O inventory is approximately 2350 gallons (21,700 pounds) distributed as follows: 1100 gallons in the reactor vessel, 350 gallons in the Emergency Cooling system, and the remaining 900 gallons in the heat exchangers and piping.

The storage tank is constructed of type 304 stainless steel and has a capacity of 2500 gallons. It is approximately 6 feet in diameter and 13 feet long. It is located in a pit in the floor of the process equipment room.

The main D_2O circulating pump is a centrifugal type capable of pumping 1800 gallons per minute with a discharge pressure of approximately 50 psig at a water temperature of 135°F. The standby pump will provide a reduced flow and may be used with the reactor operating in mode 1 operation.

Two heat exchangers connected in series are used to remove the heat from the reactor. Both exchangers are shell and "U" tube construction using type 304 stainless steel tubes. Both units have double tube sheets to eliminate the possibility of cross-leakage via a faulty tube-to-tube-sheet joint. The system is arranged such that the pressure in the D_2O or tube side of each exchanger is always greater than the corresponding H_2O shell side pressure. Tubing leaks therefore will result in D_2O leaking into the H_2O system.

A constant level of D_2O is maintained within the reactor vessel by the installation of an overflow pipe installed inside the reactor tank. This provides about 29 inches of top reflector D_2O above the fuel elements.

A small amount of D_2O is withdrawn from the storage tank TD1, purified, filtered, and returned to the pump suction leg. This water overflows the reactor vessel via the above noted overflow line and provides a constant D_2O level within

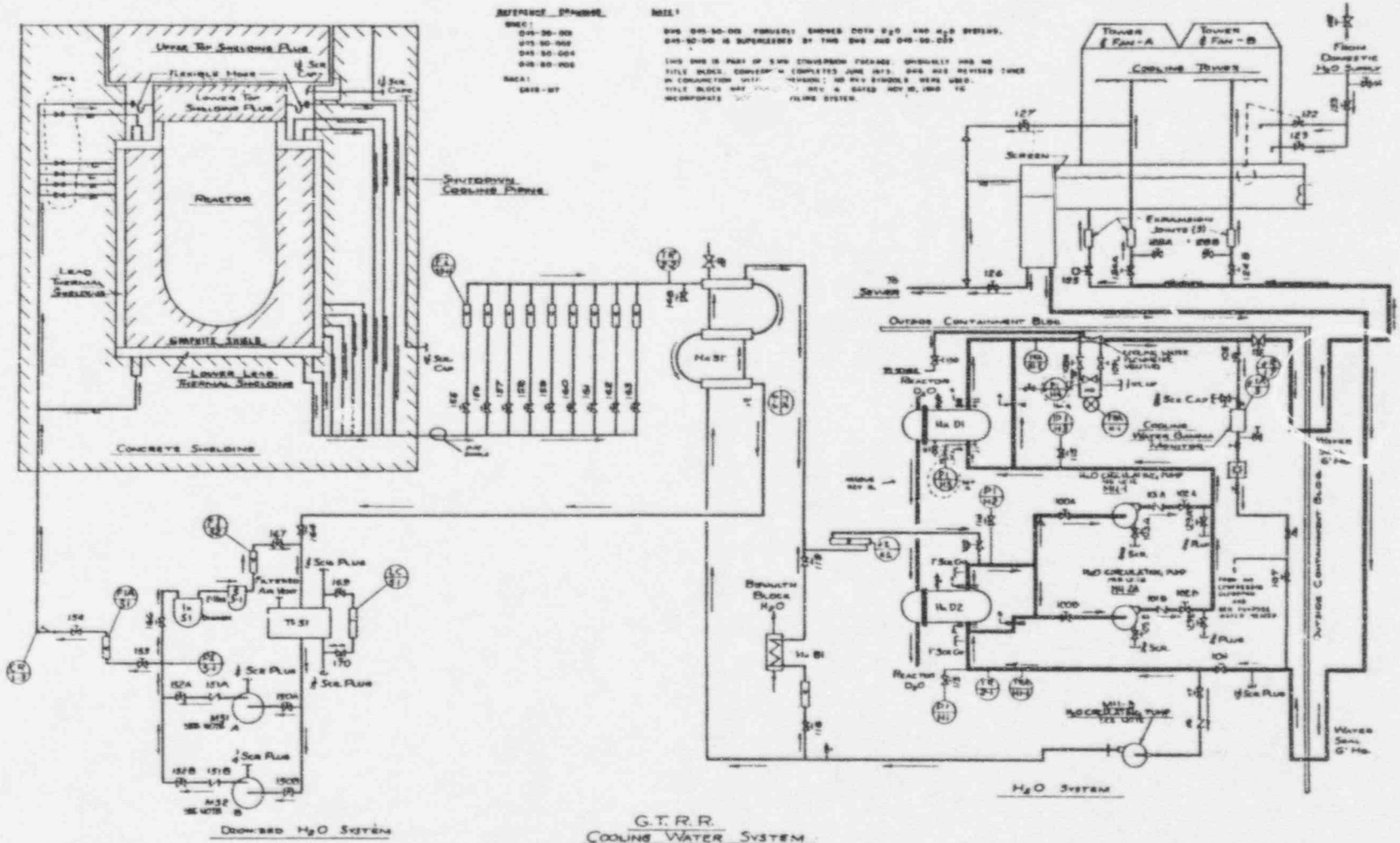


Figure 4.25 GTRR Secondary Water System

the vessel. A second fixed pipe installed within the vessel provides the means for draining the top reflector of the reactor and, at the same time, insuring that no fuel is uncovered. The top of this pipe is about one inch above the fuel region of the core. The reflector is drained by opening two fail-safe, spring-loaded valves installed in parallel. Two valves are used so that the water can be drained even though one valve should fail. All of the piping and valves in contact with the D_2O are fabricated of stainless steel. Welded construction is used when possible to minimize D_2O leakage. Most of the possible leakage points in the system are monitored by a conductance type leak detection system tied into the control room alarm circuitry.

4.4.8.2 Secondary H_2O System

The main components in the secondary H_2O system are the main and standby circulating pumps and the cooling tower. The main pump will circulate H_2O at 1200 gallons per minute through the shell side of the heat exchangers and to the cooling tower outside the containment building. The standby pump can be used for mode 1 operation. The cooling tower is a two-section unit of the cross-flow induced draft type. There is a single speed fan in each unit and a water bypass loop to provide for variation in cooling capacity as a result of seasonal temperature changes. The tower is designed with a 79° F wet bulb temperature and an 8°F approach.

4.4.8.3 Emergency Cooling System

The objective of the emergency cooling system provided as a part of the 5 MW design modification is to ensure that sufficient time is available to take the necessary steps to sustain fuel element cooling in the event of loss of D_2O from the reactor vessel. Specifically, this system will be capable of providing (for operation at power levels above 1 MW) 8 gpm total flow to the fuel elements for a period of at least 30 minutes. The complete vaporization of this amount of coolant would provide a heat removal rate of 36,200 BTU/minute, due to heat of vaporization alone.

During the first minute after reactor shutdown, the fraction of reactor power falls from .059 at 1 second to .040 at 1 minute [4.1] (See Figure 4.26) for an infinite previous operating period. Thus, the total reactor after heat is 11,400 BTU/minute at 1 minute after reactor shutdown. If one assumes a flux peaking factor of 1.5 times the average fission product inventory, and an additional overall factor of 1.17 times the average heat production to account for gamma ray heating due to both internally originated gamma rays and gammas from other elements[4.2] the maximum heat generated in the hottest element of a 14-element (minimum) core 1 minute after shutdown is 1430 BTU/minute. With the emergency coolant flow rate of 8 gpm equally divided among 19 fuel element positions, the latent heat of vaporization alone (no credit taken for sensible heat) provides an element with 1900 BTU/minute of heat removal capability. While the emergency coolant continues for 30 minutes, the heat source becomes smaller, providing an even larger margin of conservatism. In addition, these calculations have not made any allowance for loss of heat by conduction into the fuel element side plates, and then into other structural members. Nor has any convective heat loss been credited.

The emergency cooling system is shown schematically in Figure 4.24. A 300-gallon D_2O tank is located in the containment vessel at an elevation above the reactor tank such that an emergency cooling flow of D_2O at 8 gpm can be supplied for at least 30 minutes following a loss of coolant in the reactor tank. While this is a static storage tank, initiation of flow will be automatic. Flow will be started by any of the following four conditions:

1. Low D_2O reactor tank level from level indicator #1
2. Low D_2O tank level from level indicator #2
3. Loss of electrical power
4. Loss of air

The independent level indicators will open the parallel stop valves in the gravity flow line between the storage tank and the distribution manifold in the reactor. Since the valves are of the normally open type, a loss of power will also cause them to open. The flow can also be initiated manually by operator action. A locked-open manual stop valve will permit operator intervention for the purpose of

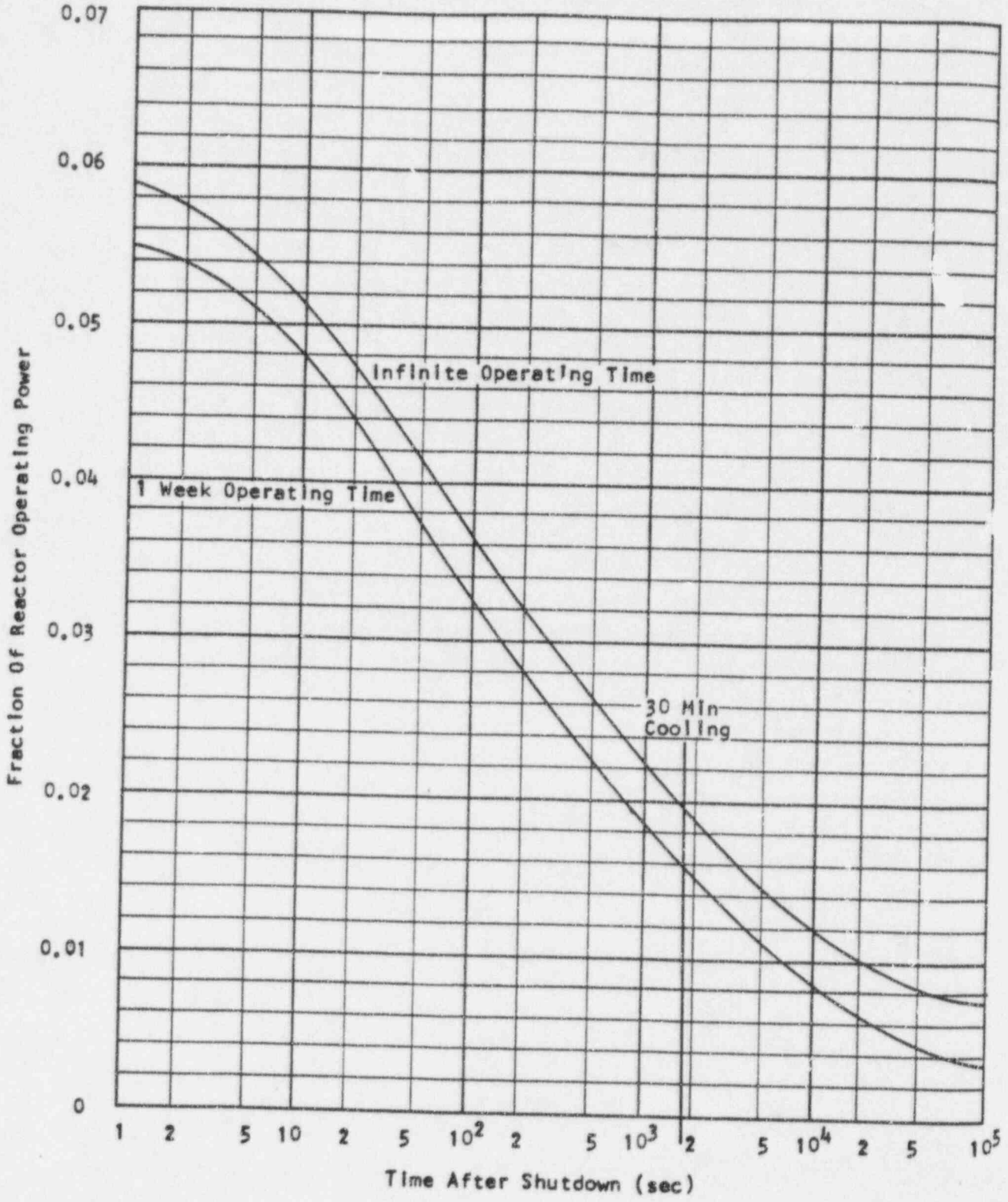


Figure 4.26. Rate of Production of Decay Heat After Shutdown (from Reference 4.1).

conserving D_2O emergency cooling storage tank supply in the event that no need for emergency cooling exists simultaneous with a power failure. The distribution manifold has been positioned and its delivery calibrated during low power testing. Its satisfactory operation is monitored in two ways. The flow from manifold to emergency coolant ports provided in each fuel element is visually observed by a borescope. In addition, each fuel element is provided with one thermocouple which is located in the coolant exit area just beneath the emergency coolant distribution plate (see Section 4.4.3). The thermocouple is used to sense the presence of emergency coolant at a temperature significantly different from ambient. Lower temperature emergency coolant D_2O is supplied during a test with the reactor shut down.

A secondary emergency coolant supply is city water, which will be supplied to the emergency cooling storage tank through a quick-connect spool-piece in the laboratory building pipe tunnel.

4.4.9 Reactor Auxiliaries

4.4.9.1 Shield Cooling System

The neutron and gamma ray absorption occurring in the graphite reflector and the first layer of the biological shield generates heat. Approximately 10 kW of heat per thermal MW of reactor power is removed by the shield cooling system. This system removes the heat while maintaining a temperature in the biological shield of less than 150 F.

Shielding heat is removed by one of two sets of parallel copper tubes cast in the lead thermal shielding layer enveloping the sides and bottom of the steel shield tank. A similar set of tubes is contained in the lead of the lower top shield. Heat flows from the graphite and the thermal shielding into the light water circulating through one set of the cooling tubes. The second set is held in reserve for use in the event of cooling tube failure. The heat is carried to a heat exchanger where it is transferred to the secondary coolant system for dissipation to the atmosphere through the cooling tower.

The primary circuit of the shield cooling system is a closed loop containing an

ion exchange column, a filter, two circulating pumps and the necessary valves and control instrumentation. Each pump delivers 35 gpm at 55 psi. The ion exchange column, located in a by-pass loop, operates at 4 gpm. The primary shield cooling circuit contains flow and temperature instrumentation wired into the reactor alarm circuitry.

4.4.9.2 D_2O Purification System

Contamination of the D_2O can affect the dissociation rate, corrosion rate, and coolant loop radioactivity. Consequently, the moderator coolant must be maintained at a high degree of purity. This is accomplished in the GTRR by withdrawing 7 gpm from the D_2O coolant system for circulation through a filter and ion exchange bed. Two such beds are provided so that the radioactivity in one may be allowed to decay and the resin may be replaced while the other is in use. This system maintains the specific resistance of the D_2O above $10^6 ohm/cm^3$.

The purification system consists of two parallel piping loops containing mixed resin beds and after-filters. Constant monitoring of water purity is obtained through the use of conductivity cells located at the inlet and outlet of the resin columns. The purification system is connected to the primary coolant piping through the D_2O overflow return system. This arrangement utilizes the head developed by the overflow return pumps as a driving force for the purification system flow.

As a portion of the conversion to 5 MW operation a sampling system for the heavy water primary coolant-moderator was installed to permit the drawing of samples in a location external to the process equipment room. This avoids potential sampling inconveniences stemming from the increasing radiation levels in the process equipment room.

Except for the resin column tanks, the material of construction for the entire system is stainless steel. The tanks are acrylic coated carbon steel and contain 2 to 3 cubic feet of high grade, mixed bed resin. Provisions have been made for deuterization of the resins. After use, the resin is drained and flushed of all D_2O. The connections are closed and, if necessary, the entire column placed in a shielded

container for shipping to a burial site. Since the life of each bed is expected to approach one year, a relatively leisurely decay and shipment schedule is indicated.

4.4.9.3 Top Reflector Control System

The top 28 inches of D_2O moderator serves as a neutron reflector for the core. Loss of this reflector results in a reduction in reactivity and, therefore, may be used as a backup for the normal shutdown and scram methods. The reflector is drained through a 4 inch pipe which connects the reactor vessel to the storage tank of the primary D_2O system. The inlet to this drain pipe is located about one inch above the top of the core to prevent inadvertent exposure of the fuel. Pneumatically-operated, full-opening, butterfly valves located in this line permit the top reflector to be drained away in about 60 seconds. The valves used are of the normally open type to provide the desired fail-safe feature in the event of a loss of power or pneumatic pressure. Two identical valves are employed in parallel to reduce the possibility of system failure because of valve malfunction.

4.4.9.4 Recombiner System

A dissociation rate of 0.01 liters of liquid D_2O per MW-hr is expected. In order to recombine the dissociated D_2 and O_2, a catalytic recombiner loop is provided.

As mentioned previously, a helium or nitrogen blanket covers the moderator. Fifteen cubic feet per minute of this gas carrying D_2 and O_2 is withdrawn, dried by heating, and passed over a palladium-on-alumina catalyst bed where the D_2 and O_2 recombine as D_2O. The helium or nitrogen and D_2O vapor are returned to the reactor system. A schematic drawing of the system is shown in Figure 4.27.

The catalyst bed consists primarily of a 6 inch I.D. cylinder, 18 inches long, filled with 1/8 inch diameter alumina pellets upon which palladium has been deposited. This bed volume gives a space velocity of approximately 30 min^{-1}. Experimental data[4.4] indicate that this velocity should allow only a few ppm of dissociated gas to return to the reactor, thereby limiting the maximum equilibrium D_2 concentration to less than 0.1%. This quantity represents less than 2.5% of the

minimum explosive concentration in air. Circuit interlocks cause an alarm if the helium or nitrogen flow rate falls below an adequate value.

4.4.9.5 Reactor Ventilation System

Fresh air for the containment building is brought in through dual isolation valves located above the control room. This air is mixed with recirculating air, heated or cooled as required, and discharged to the main volume of the building. The exhaust air is drawn primarily from a series of ports around the upper reactor face. This air is ducted to the basement of the containment building and combined with the exhaust from several experimental facilities. The air is monitored and then passed to a hold-up duct cast into the basement floor. The duct provides about 12 seconds of delay at a flow rate of 4000 cfm. From the duct the air stream passes through a filter bank of roughing and high efficiency filters. The air then enters the exhaust blower and exits the containment building through two exhaust butterfly valves. At the base of the exhaust stack, the reactor effluent air is mixed with 30,000 cfm of fresh air and the total (~34,000 cfm) discharged from the 76 foot stack. A schematic drawing of the system is shown in Figure 4.27.

The exhaust air is drawn from the vicinity of the reactor in an effort to sweep away any gaseous and particulate radioactivity. The majority of this activity is argon-41 created by neutron absorption in the natural argon contained in air. In some cases it has been necessary to provide appropriate seals to restrict the diffusion of argon-41 from regions of high thermal neutron flux to the areas surrounding the reactor.

4.4.9.6 Overpressure Relief System

One of the major inherent safety characteristics of the GTRR is its ability to reduce reactivity by the formation of steam voids within the coolant channels of the fuel assemblies. This steam formation can result in excessive pressurization of the reactor vessel if the steam generation is rapid or persists for long duration. Since complete voiding of all fuel channels can account for more than the maximum

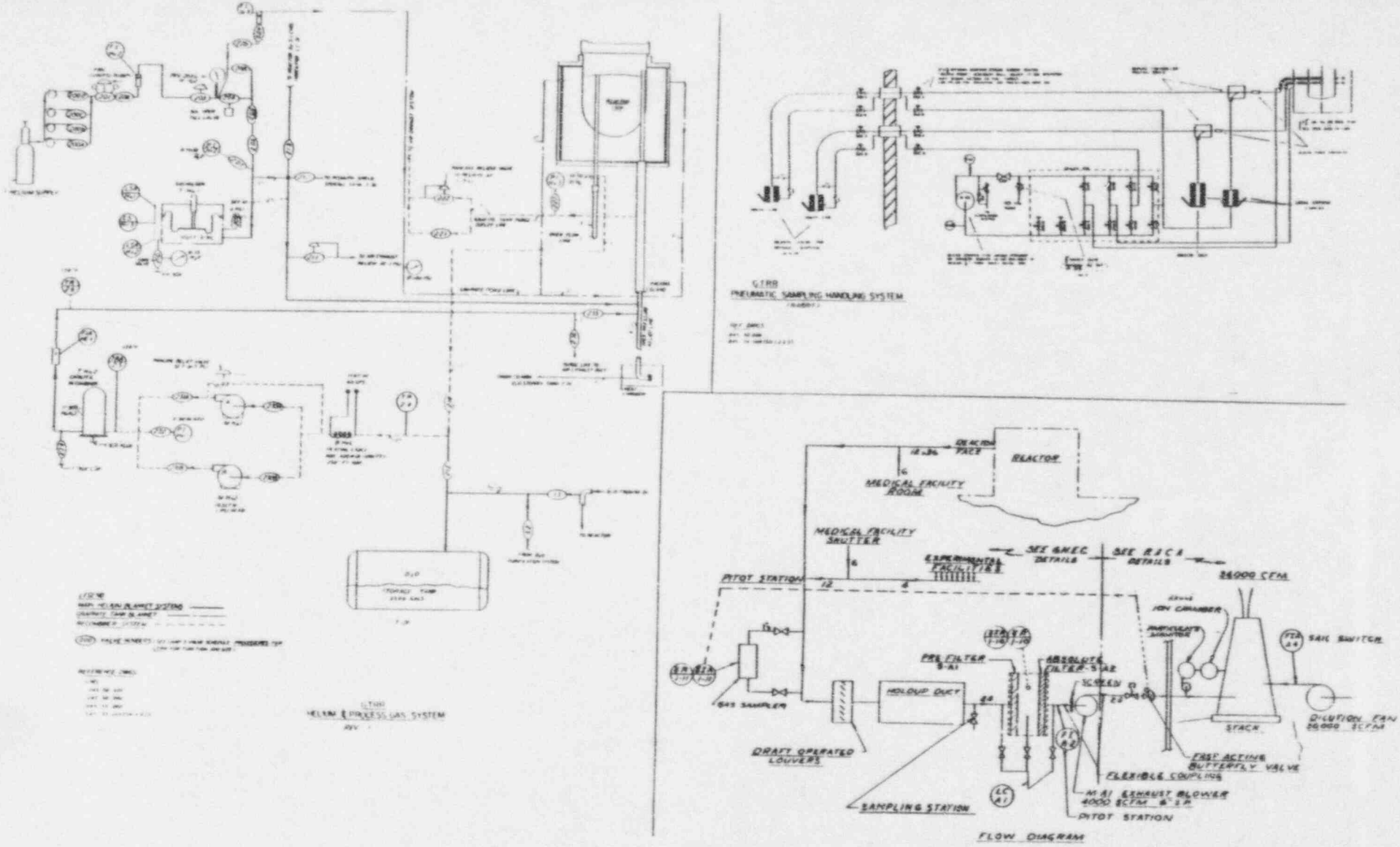


Figure 4.27. GTRR Gas, Pneumatic Handling, and Exhaust Systems.

available reactivity, steam generation should be of short duration.

As a preventive measure against pressurization of the reactor vessel subsequent to steam generation, a 6 inch pipe connects the gas volume in the upper part of the reactor tank to a 1500 cubic foot expansion chamber provided beneath the basement floor near the center of the containment building. This pipe is closed at its lower end with a graphite diaphragm which will rupture at a pressure of approximately one psig. Rupture of the diaphragm will permit the steam to expand into the chamber relieving the pressure within the vessel.

4.4.9.7 Fuel Handling Systems

Following 5 MW operation, fuel assen.blies are not removed from the core until at least 12 hours after shutdown. This cooling and decay period insures that fuel plate temperature in the element with the greatest fission product inventory will not exceed 450° C during the transfer in a dry coffin. This conclusion is based on the experimental results obtained at Harwell[4.5] with instrumented fuel elements of a design quite similar to the GTRR element, but under more stringent experimental conditions. Figure 4.28 summarizes these results.

Fuel assemblies are removed from the reactor by means of a coffin which is put into position by the building crane. The coffin contains an integral handling tool designed to preclude dropping the attached assembly. The replacement of a fuel assembly requires the removal of the upper top shield port plug and thermocouple wiring. The coffin is then positioned over the port, and the entire fuel assembly and shield section, shown in Figure 4.16, is raised up into the coffin through the bottom door. This door is then closed and the coffin removed from the reactor building to the fuel storage area provided in the adjacent building.

The top section of the fuel assembly is removed in the storage area and retained for radioactive decay and possible reuse. The lower section of the assembly containing the fuel may be used as a source of gamma rays for experimental purposes. When it is no longer of value for this use it will be returned to DOE.

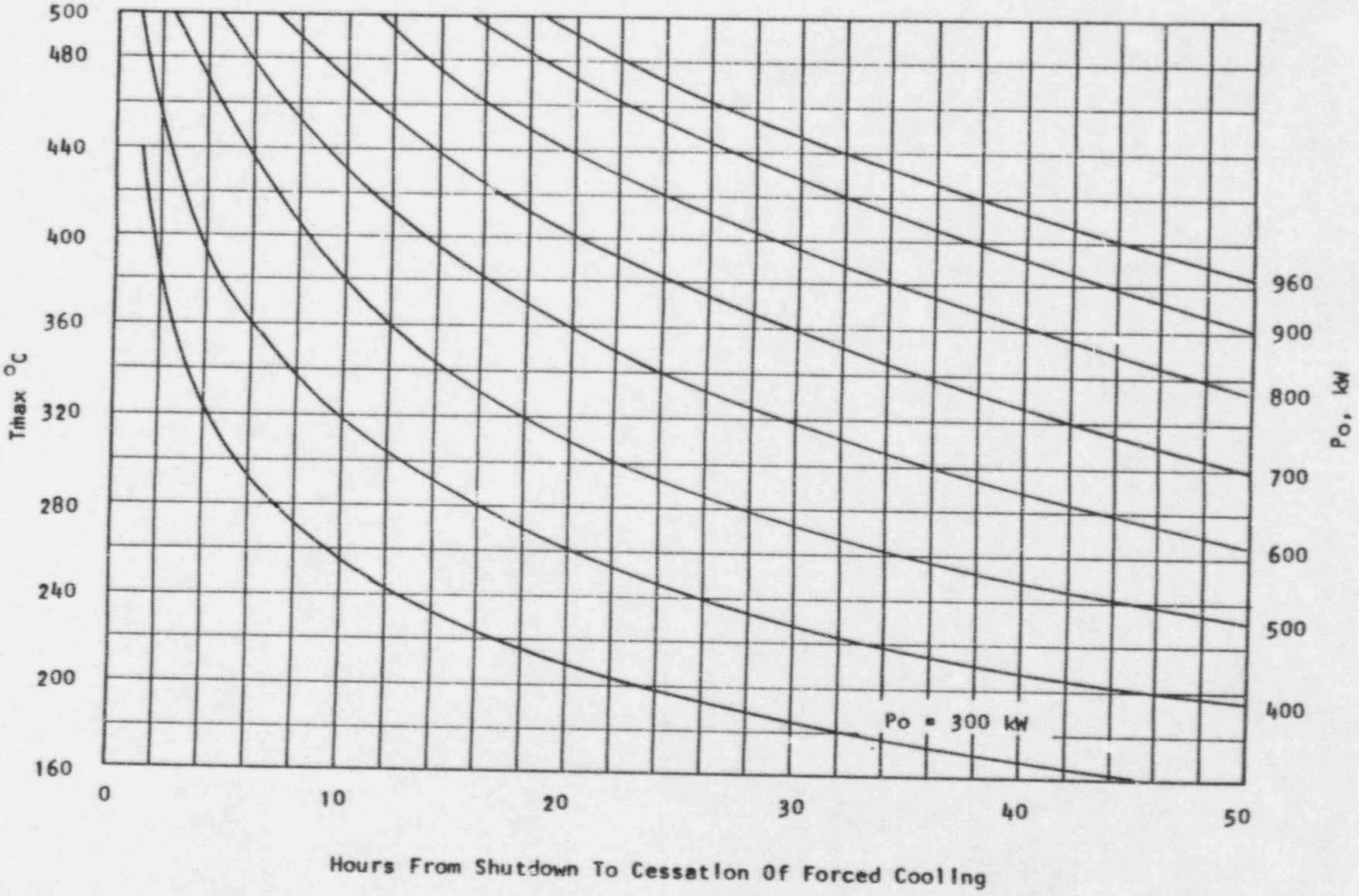


Figure 4.28. Temperature vs. Time from Shutdown for Dido Element.

The 5 MW reactor is not operated with less than 14 elements. Using a factor of 1.5 as the ratio of power generated in the hottest element to the average power generated corresponds, in the 14-element core, to a maximum power of (1.5 x 5000 kw/14 elements) = 535 kW. From Figure 4.28 it is seen that a thermocouple at the top of the fuel element box would not be expected to exceed 450°C if forced cooling were suspended six hours after shutdown. Eight hours of forced cooling will be maintained after shutdown. An additional minimum wait-time of 4 hours will be provided before the fuel element is transferred from the reactor vessel to the transfer coffin, thus providing a total elapsed time from shutdown to transfer of 12 hours.

4.5 Radiation Monitoring

4.5.1 Facility Monitoring

All major reactor systems are monitored continuously. In most cases, the activity levels are indicated or recorded in the reactor control room. If measured activity exceeds pre-set levels, both aural and visual alarms are given. An indication of excessive gaseous activity in the exhaust system produces an automatic isolation of the containment building. The isolation circuits automatically close both the air supply and exhaust butterfly valves. In addition, isolation valves on the pneumatic transfer system are closed.

The secondary H_2O coolant is monitored for the presence of gross radioactivity by using a detector in a by-pass loop downstream from the heat exchanger. Leakage of D_2O across the heat exchanger to the H_2O side will produce an increase in the observed activity in the light water due to the presence of nitrogen-16 in the D_2O. The alarm is set at as low a point as is practical.

The exhaust from the reactor experimental facilities, biomedical facility, and reactor face may contain a variety of activated gaseous and particulate products. In order to keep the total activity generated by activation of air to a minimum, special precautions have been taken to reduce the amount of air exposed to a high neutron flux. These precautions are discussed in Sections 4.4.9.5 and 8.5.4.

All air which is exhausted from the containment building (except for a small

air exhaust line from the pump room) flows into a common duct in the basement. At this point a sample of the air is passed through a gas monitoring system. The air then enters a holdup volume which is designed to allow time for the GM-type gas monitor to respond before the air passes from the containment building. Air is pulled from the holdup volume by a blower, and passes through a roughing and a high efficiency filter. The previously mentioned exhaust from the pump room enters the system just before the air passes through the filters. The air then passes through two fast-acting automatic butterfly valves. These valves automatically close if a gas monitor indicates excessive activity. Air which passes these butterfly valves is monitored for gaseous and particulate radioactivity just as it leaves the containment building. As the air leaves the containment building at a flow rate of 4,000 cfm (maximum), it enters a plenum in the base of the stack where it is diluted with 30,000 cfm of air drawn from the outside. The air is finally released from the stack at a height of 76 feet above ground. The reactor exhaust normally will be the only source of gaseous wastes to be released to the environment. If any other operations which could release significant amounts of radioactivity are to be performed, special arrangements will be made to monitor or collect and dispose of the activity to insure that the appropriate limits are observed.

All liquid effluent from the containment building is collected in the waste storage system which is located in a pit below the high bay area. This system is shown diagrammatically in Figure 4.6. Although these liquids do not normally contain significant amounts of radioactivity, the contents of the waste storage tanks are not discharged to the City of Atlanta sanitary sewer system until the tank has been isolated, the contents properly agitated, and a sample analyzed to determine that the waste is within limits specified in 10CFR20. Records of all discharges of liquid wastes are maintained by Health Physics.

Radiation monitoring packets are installed at selected locations throughout the reactor facility. For continuous monitoring of radiation levels, the containment building is equipped with ten external gamma radiation detectors. Each of the ten monitors relays an indication of the radiation level to the reactor control room.

Each station has an adjustable, preset alarm level. If the radiation level exceeds this amount, the reactor operator in the control room and anyone in the vicinity of the detector is alerted. Five of these area monitors are located so as to monitor the main floor and the control room level of the reactor building. The other five monitors are located at appropriate points in the basement. Continuous air sampling in the containment building proper is done for both particulate and gaseous radioactivity with preset visual and audible alarms.

Independent area monitors are located in the hot laboratory area. These monitors provide personnel in their immediate vicinity with a continuous indication of the gamma dose rate and with an audio and visual alarm if the preset radiation level is exceeded.

Two criticality monitors are permanently installed. One is located near the spent fuel element storage pool in the high bay area. The second is adjacent to the cold fuel element storage vault near the containment building. In addition to dose rate indication and alarm at the monitor itself, each unit is connected to a remote dose rate recorder. Alarm signals which may be clearly heard by persons in the vicinity indicate the presence of excessive levels of radiation.

4.5.2 Personnel Monitoring

Personnel monitoring complies with the provisions of 10CFR, Part 20, as a minimum standard. The Manager, Officer of Radiation Safety is responsible for formulating detailed personnel monitoring procedures.

Persons who enter the containment building are required to wear personnel monitoring devices. Visitors usually are accommodated by allowing them access to the viewing gallery, offices, and laboratories outside the radiation control zones. Personnel monitoring devices are not required for such persons unless there is a possibility that they will be exposed to radiation in excess of the minimum limits specified in Part 20 for personnel monitoring. Visitors who are allowed to enter the containment building or high bay area are required to register before entering, are provided with appropriate personnel monitoring devices, and are escorted by a

Georgia Tech employee, except where special arrangements are made and approved by the Manager, Officer of Radiation Safety (MORS).

Persons who are permitted to work with significant radioactivity or to enter the containment building or high bay area without escort must meet standards of training established by MORS.

Instruments are provided to allow persons leaving the controlled area to monitor their hands and shoes for radioactivity. All persons who are potentially exposed to loose radioactive materials are required to monitor themselves at appropriate times.

Permanent records of personnel monitoring results are maintained by the Office of Radiological Safety. Individuals are permitted to examine their own personal monitoring file upon request.

4.5.3 Area Monitoring

The Office of Radiological Safety conducts a program of routine and special area monitoring of the reactor and laboratory buildings. External radiation levels, airborne activity, and surface contamination are measured. Schedules for routine surveys are determined on the basis of degree of utilization and levels of radioactivity being handled in various areas. Special surveys are performed whenever a non-routine activity takes place involving possible significant exposure to radioactivity. Under certain conditions, established by MORS, persons other than members of the Office of Radiological Safety may be authorized to perform radiation surveys. Each new installation is carefully surveyed when it is first put into operation and, if the potential hazard warrants, it will be added to the routine survey program. The Office of Radiological Safety maintains records of all surveys, and reports significant results to the appropriate persons.

4.5.4 Environmental Monitoring

An environmental monitoring program has been carried on with the cooperation of the Radiological Health Section of the Georgia Department of Public

Health since initial reactor startup. Thermoluminesent dosimeters are placed at 50 locations outside the perimeter of the reactor facility and are changed on a quarterly basis. No statistically valid indication of an increase in environmental radioactivity levels has been observed through analysis of the data produced in this program.

All evidence, both theoretical and empirical, indicates that the only radioactive material emitted from the stack in measurable quantities is argon-41. Every attempt is being made to isolate the sources of the argon-41 production and minimize its release. It is highly unlikely that the radioactivity released under reactor operating conditions at 5 MW will cause any person continuously residing or working in the neighborhood to be exposed to more than a small fraction of the total effective dose equivalent permitted in 10CRF 20.1301. However, the environmental monitoring program will continue to demonstrate the validity of this assumption by direct measurement rather than by theoretical analysis. Equipment for continuous, automatic measurement and recording of wind speed and direction has been installed as an aid to the selection of monitoring points and the analysis of the resultant data.

The environmental monitoring program includes the following elements:

A. The State Radiological Health Section will continue their program of thermoluminescent dosimeter (TLD) monitoring which has been in effect since initial reactor startup.

B. Georgia Tech began a supplementary thermoluminescent dosimeter monitoring program outside of the reactor perimeter fence in December 1966. Currently thirty TLD's are placed in locations which current meteorological conditions indicate will be the most likely to receive the maximum dose from argon-41. These badges are being changed every three months.

C. The program of monitoring the radiation dose at the reactor perimeter fence will continue.

D. Special determinations of radiation doses using ionization chambers will be made under specific meteorological conditions which indicate

that the plume of radioargon may maximize dosage in a specific area. These data will be compared with background readings obtained in the same areas when the reactor is not operating.

The environmental monitoring program described will be continued indefinitely, subject to any improvements in method which may become apparent. If at any time the data should indicate that a person occupying any location in the environment might receive an annual dose in excess of 10CRF20 limits as a result of the operation of the Georgia Tech Research Reactor, steps will be taken to analyze, control, and further limit the release of radioactivity and to guarantee compliance with governmental regulations.

References

4.1 Glasstone, S and A. Sesonske, Nuclear Reactor Engineering, D. Van Nostrand Company, Inc., Princeton (1993) p. 101

4.2 Technical Specifications for the MIT Research Reactor, MIT-NE-62, (1965)

4.3 Private Communication, Reactor Operations Division, Argonne National Laboratory, to G. M. Brown., General Nuclear Engineering Corporation, January, 1959

4.4 Costikyan, T.W. et. al., "The Catalytic Recombination of Hydrogen and Oxygen", Report No. AECD-3969, June 2, 1952

4.5 Merrett, D. J. and Taylor, D. J., "Fission-Product Heating of Uncooled Fuel Element after Unloaded from DIDO", AERE-MI317 (1964)

4.6 McGoff, D. J., "FORM-A Fourier Transform Fast Spectrum Code for the IBM-709", NAA-SR-Memo-5766, September. 1960

4.7 Bohl, H., et.al., "MUFT-4, Fast Neutron Spectrom Code for the IBM-704", WAPD-TM-72, July, 1957

4.8 W. W. Graham III, et.al, "Kinetics Dorameters of a Highly Enriched Heavy-Water Reactor, Final Report", TID-23037, April, 1966

5.0 REACTOR PHYSICS AND THERMAL HYDRAULICS ANALYSES FOR HEU AND LEU CORES

BY

J. E. MATOS, S. C. MO AND W. L. WOODRUFF
ARGONNE NATIONAL LABORATORY

SEPTEMBER 22, 1992

SUMMARY

This report contains the results of design and safety analyses performed by the RERTR Program at the Argonne National Laboratory (ANL) for conversion of the Georgia Tech Research Reactor (GTRR) from the use of HEU fuel to the use of LEU fuel. The objectives of this study were to: (1) maintain or improve upon the present reactor performance and margins of safety, (2) maintain as closely as possible the technical specifications and operating procedures of the present HEU core, and (3) utilize a proven fuel assembly design that is economical to manufacture. Extensive collaboration with Dr. R. Karam, Director of the Neely Nuclear Research Center at Georgia Tech, took place on all aspects of this work.

The LEU fuel assembly has the same overall design as the present HEU fuel assembly, except that it contains 18 fueled plates with LEU U_3Si_2-Al fuel instead of 16 fueled plates with HEU U-Al alloy fuel. This LEU silicide fuel has been approved by the Nuclear Regulatory Commission for use in non-power reactors.

Documents that were reviewed by ANL as bases for the design and safety evaluations were the GTRR Safety Analysis Reports, the GTRR Technical Specifications, and responses by the reactor organization to AEC questions in licensing the reactor for 5 MW operation.

The methods and codes that were utilized have been qualified using comparisons of calculations and measurements of LEU demonstration cores in the Ford Nuclear Reactor at the University of Michigan and in the Oak Ridge Research Reactor at the Oak Ridge National Laboratory. Additional qualification has been obtained via international benchmark comparisons sponsored by the IAEA for heavy water research reactors.

Only those reactor parameters and safety analyses which could change as a result of replacing the HEU fuel in the core with LEU fuel are addressed. The attached summary table provides a comparison of the key design features of the HEU and LEU fuel assemblies and a comparison of the key reactor and safety parameters that were calculated for each core. The results show that all of the objectives of this study were fully realized and that the GTRR reactor facility can be operated as safely with the new LEU fuel assemblies as with the present HEU fuel assemblies.

SUMMARY TABLE

HEU and LEU Design Data, Core Physics, and Safety Parameters for Conversion of the Georgia Tech Research Reactor

DESIGN DATA	HEU Core	LEU Core
Minimum Number of Fuel Assemblies	14	14
Maximum Number of Fuel Assemblies	19	19
Fuel Type	U-Al Alloy	U_3Si_2-Al
Enrichment, %	93	19.75
Uranium Density, g/cm^3	0.65	3.5
Number of Fueled Plates per Assembly	16	18
Number of Non-Fueled Plates per Assembly	2	2
^{235}U per Fuel Plate, g	11.75	12.5
^{235}U per Fuel Assembly, g	188	225
Fuel Meat Thickness, mm	0.51	0.51
Cladding Thickness, mm	0.38	0.38
Cladding Material	1100 Al	6061 Al

REACTOR PARAMETERS	HEU Core	LEU Core	Number of Assemblies
Cold Clean Excess Reactivity, % $\Delta k/k$	11.7 ± 0.4	9.4 ± 0.4	17
Coolant Temperature Coefficient, % $\Delta k/k$ Δ°C	-0.0076	-0.0067	14
Doppler Coefficient, % $\Delta k/k$/°C	-0.0	-0.0017	14
Whole Reactor Isothermal Temp. Coeff., % $\Delta k/k$/°C	-0.0224	-0.0232	14
Coolant Void Coefficient, % $\Delta k/k$/% Void	-0.0383	-0.0333	14
Limiting Power Peaking Factor	1.54	1.58	14
Prompt Neutron Lifetime, μs	780	745	14
Effective Delayed Neutron Fraction	0.00755	0.0075-0.0076	14
Shutdown Margin, % $\Delta k/k$ (Max. Worth Shim Blade and Reg. Rod Stuck Out)	-7.1 ± 0.2	-8.8 ± 0.2	17
Top D_2O Reflector Worth, % ~k/k (For D_2O 2" Above Fuel Meat)	-2.1 ± 0.3	-2.4 ± 0.3	17
Reactor Power Limits -1625 gpm Flow Rate			
Based on Departure from Nucleate Boiling, MW	11.5	10.8	14
Based on Flow Instability Criterion, MW	10.6	10.6	14
Limiting Reactor Inlet Temperature, °F	172	170	14
Limiting Reactor Outlet Temperature, °F	188	187	14
Limiting Safety System Settings - Forced Convection			
Reactor Power, MW	5.5	5.6	14
Coolant Flow Rate, gpm	1625	<1625	14
Reactor Outlet Temperature, °F	139	145	14
Margin to D_2O Saturation Temperature, °F	8	11	14
Max. Fuel Plate Temp. for LOCA after 8 Hours Cooling, °C	425	400	14
Maximum Positive Reactivity Insertion, % $\Delta k/k$	>2.2	>2.2	14

5.0 ANALYSES FOR CONVERSION OF THE GEORGIA TECH RESEARCH REACTOR FROM HEU TO LEU FUEL

J. E. Matos, S. C. Mo, and W. L. Woodruff
RERTR Program
Argonne National Laboratory
Argonne, IL 60439
September 1992

5.1. INTRODUCTION

This report contains the results of design and safety analyses performed by the RERTR Program at the Argonne National Laboratory (ANL) for conversion of the Georgia Tech Research Reactor (GTRR) from the use of HEU fuel to the use of LEU fuel. The objectives of this study were to: (1) maintain or improve upon the present reactor performance and margins of safety, (2) maintain as closely as possible the technical specifications and operating procedures of the present HEU core, and (3) utilize a proven fuel assembly design that is economical to manufacture.

The design and safety analyses in this report provide comparisons of reactor parameters and safety margins for the GTRR HEU and LEU cores. Only those parameters which could change as a result of replacing the HEU fuel in the core with LEU fuel are addressed. Documents that were reviewed by ANL as bases for the design and safety evaluations were the GTRR Safety Analysis Reports,[1] the GTRR Technical Specifications[2], and responses[3,4] by the reactor organization to AEC questions in licensing the reactor for 5 MW operation.

The LEU fuel assembly has the same overall design as the present HEU fuel assembly, except that it contains 18 fueled plates with LEU U_3Si_2-Al fuel and two nonfueled plates instead of 16 fueled plates with HEU U-Al alloy fuel and 2 non-fueled plates. A detailed safety evaluation of LEU U_3Si_2-Al fuel can be found in Reference 5.

The methods and codes that were utilized by ANL have been qualified using comparisons of calculations and measurements of LEU demonstration cores[6-10] in the Ford Nuclear Reactor at the University of Michigan and in the Oak Ridge Research Reactor (ORR) at the Oak Ridge National Laboratory. Additional

qualification has been obtained via international benchmark comparisons[11,12] sponsored by the IAEA.

5.2. Reactor Description

The GTRR is a heterogeneous, heavy-water moderated and cooled, tank-type reactor fueled with 93% enriched MTR-type U-Al alloy fuel. Horizontal and vertical sections through the reactor are shown in Figs. 1 and 2, respectively. Provision is made for up to 19 fuel assemblies spaced 6 inches apart in a triangular array. The current core consists of 17 fuel assemblies. Each assembly consists of 16 fueled and two non-fueled plates with a fissile loading of about 188 g ^{235}U. The total fissile loading of a fresh 17 assembly core would be about 3.2 kg ^{235}U.

The fuel is centrally located in a six foot diameter aluminum reactor vessel which provides a two foot thick D_2O reflector completely surrounding the core. The reactor vessel is mounted on a steel support structure and is suspended within a thick-walled graphite cup. The graphite provides an additional two feet of reflector both radially and beneath the vessel. The core and reflector system is completely enclosed by the lead and concrete biological shield.

The reactor is controlled by means of four cadmium shim-safety blades and one cadmium regulating rod. The four shim-safety blades are mounted at the top of the reactor vessel and swing downward through the core between adjacent rows of fuel assemblies. The regulating rod is supported on the reactor top shield and extends downward into the radial D_2O reflector region. This rod moves vertically between the horizontal midplane and the top of the core.

The heat removal system is composed of a primary heavy-water system and a secondary light-water system. The heavy-water system includes the reactor vessel, the primary D_2O coolant pumps, the D_2O makeup pump, the heat exchangers, and the associated valves and piping. The light-water secondary system is composed of the circulating water pumps, the cooling tower, and associated valves and piping.

The LEU reference core used in this analysis consists of 17 fuel assemblies with the same arrangement as the present HEU core. Each fuel assembly contains 18 fueled plates with 225 g ^{235}U when fresh. The LEU core will use the same control system, heat removal system, and auxiliary systems as the current HEU core.

Fig. 1. Horizontal Section of GTRR at the Core Midplane.

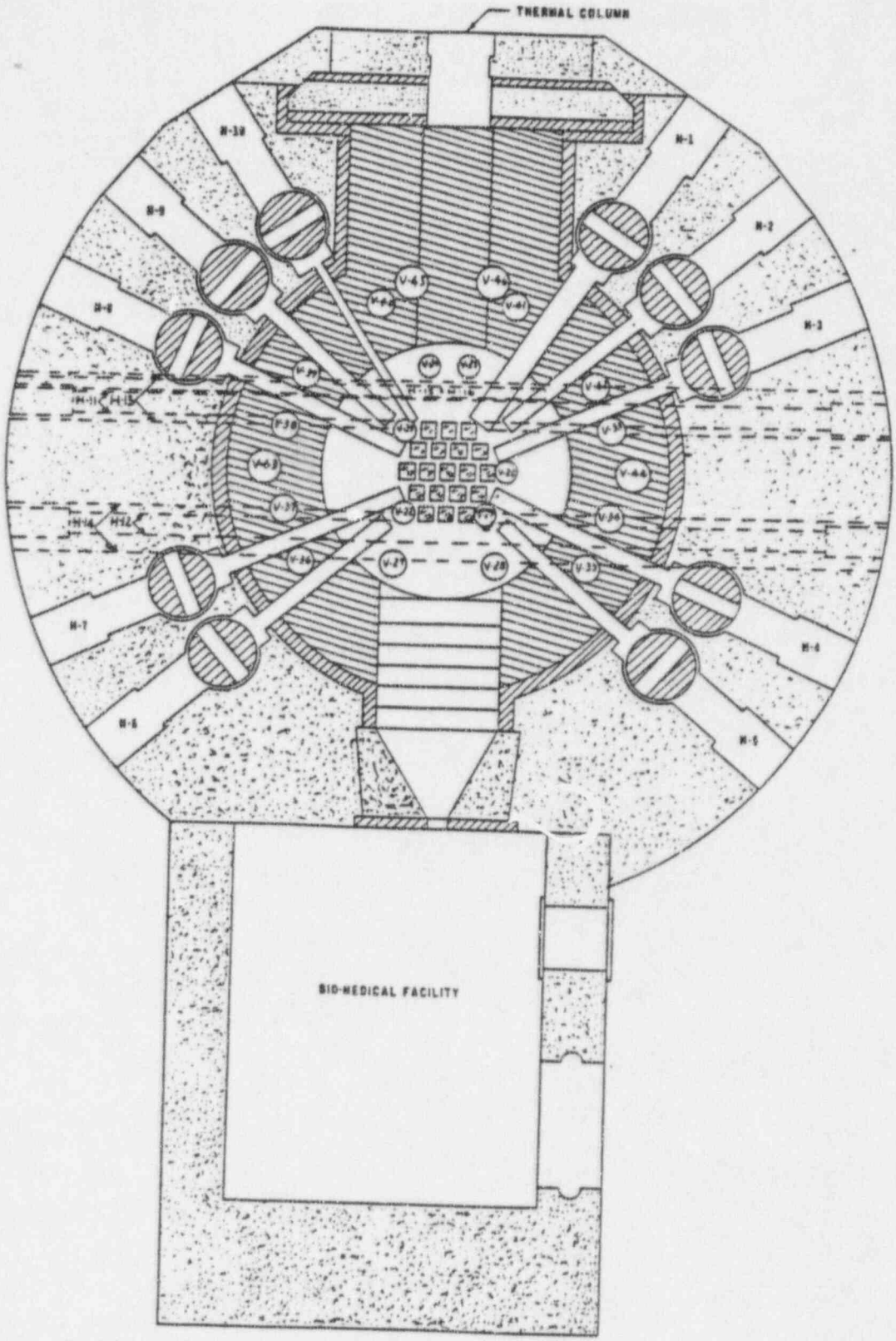
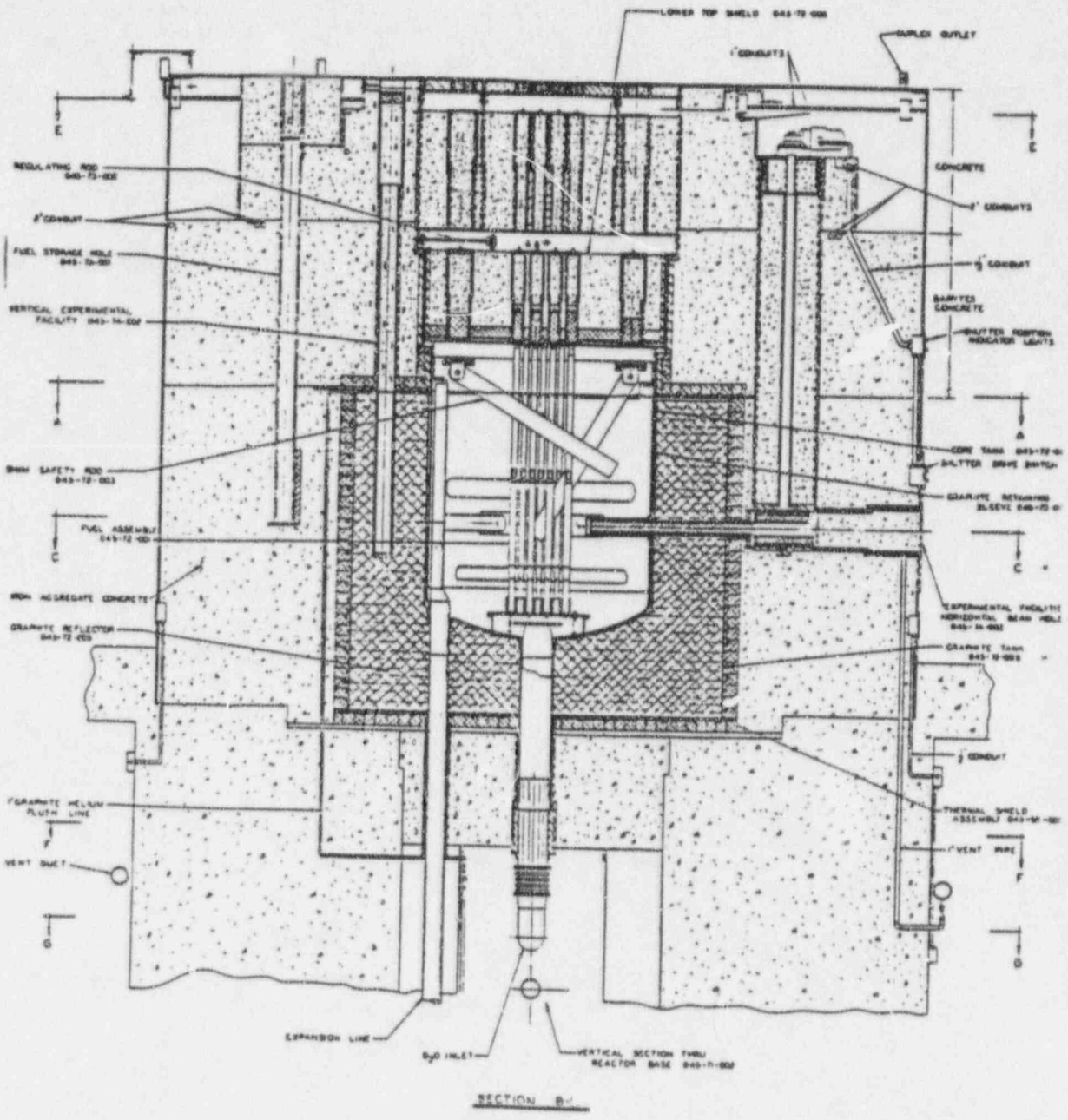


Fig. 2. Vertical Section Through Reactor.



SECTION B-

5.3. Fuel Assembly Descriptions

The geometries, materials and fissile loadings of the current HEU fuel assemblies and the replacement LEU fuel assemblies are described in Table 1. A schematic diagram of the HEU fuel assembly is shown in Fig. 3. The LEU fuel plate is the standard DOE plate containing U_3Si_2-Al fuel with ~3.5 g U/cm^3 and 12.5 g^{235}U. The external dimensions and structural materials of both assemblies are identical, except that the LEU assemblies utilize 6061 Al instead of 1100 Al.

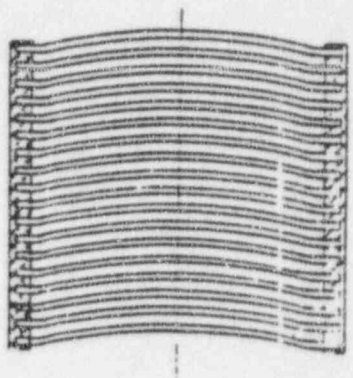
Table 1. Descriptions of the HEU and LEU Fuel Assemblies

		HEU	LEU
Number of Fueled Plates/Assembly		16	18
Number ol Non-Fueled Plates/Assembly		2	2
Fissile Loading/Plate, 9 ^{235}U		11.75	12.5
Fissile Loading/Assembly, 9 ^{235}U		188	225
Fuel Meat Composition		U-Al Alloy	U_3Si_2-Al
Cladding Material		1100 Al[1]	6061 Al[2]
Fuel Meat Dimensions			
Thickness,	mm	0.51	0.51
Width,	mm	63.5	58.9 - 62.8
Length,	mm	584 - 610	572 - 610
Cladding Thickness,	mm	0.38	0.38

[1] 10 ppm natural boron was added to the composition of the cladding and all fuel assembly structural materials to represent the alloying materials, boron impurity, and other impurities in the 1100 Al of the HEU assemblies.

[2] 20 ppm natural boron was added to the composition of the cladding and structural materials of the LEU assemblies to represent the alloying materials, boron impurity, and other impurities in 6061 Al. Aluminum with no boron or other impurities was used in the fuel meat of both the HEU and LEU assemblies.

Fig. 3. HEU Fuel Assembly Schematic



5.4. CALCULATIONAL MODELS

5.4.1 Nuclear Cross Sections for Diffusion Theory Models

Microscopic cross sections in seven energy groups (Table 2) were prepared at 23°C using the EPRI-CELL code[13] for the HEU and LEU fuel assembly geometries and fissile loadings. The integral transport calculations in EPRI-CELL were performed for 69 fast groups and 35 thermal groups (<1.855 eV), which were then collapsed to seven broad energy groups for use in diffusion theory calculations.

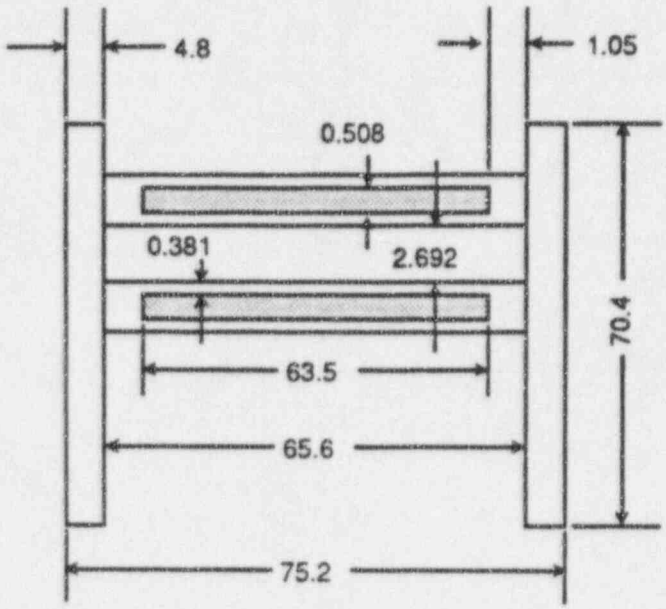
Table 2. Seven Group Energy Group Boundaries

Group No.	Upper Energy	Lower Energy	Group No.	Upper Energy	Lower Energy
1	10.0 MeV	0.821 MeV	5	0.625 eV	0.251 eV
2	0.821 MeV	5.531 keV	6	0.251 eV	0.057 eV
3	5.531 keV	1.855 eV	7	0.057 eV	2.53×10^{-4} eV
4	1.855 eV	0.625 eV			

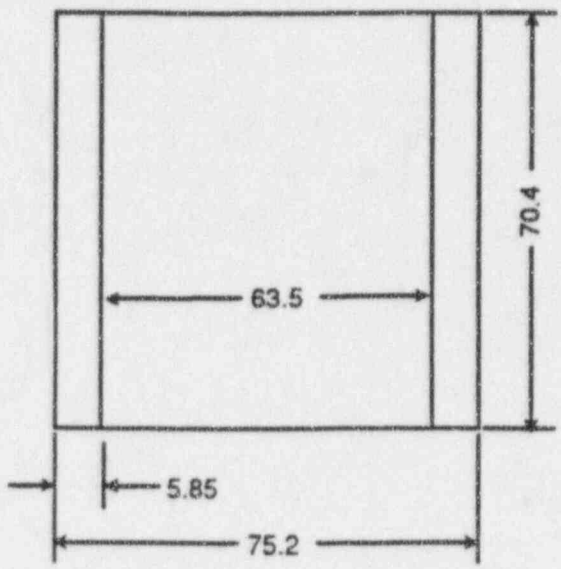
Figure 4 shows the dimensions of the HEU and LEU fuel assemblies and the fuel assembly models that were used in the diffusion theory calculations for the reactor. The fueled and non-fueled regions were modeled separately. A non-fueled region consists of a sideplate and the fuel plate aluminum (plus associated water) between the fuel meat and the sideplate.

Figure 5 shows the unit cell geometry and dimensions that were used in EPRI-CELL to generate microscopic cross sections for the fueled and non-fueled regions of the HEU and LEU assemblies. The non-fueled region inside the assembly is represented by the "extra region 1" containing calculated volume fractions of aluminum and heavy water associated with each fuel plate. "Extra region 2" was modeled to represent the heavy water outside the assembly that is associated with each fuel plate. Its thickness was chosen to preserve the water volume fraction in the physical unit cell of each fuel assembly. All cell calculations were done using a fixed buckling of 0.00373 cm^{-2}, which corresponds with the anticipated axial extrapolation length of about 21 cm in each fuel assembly in the reactor diffusion theory calculations.

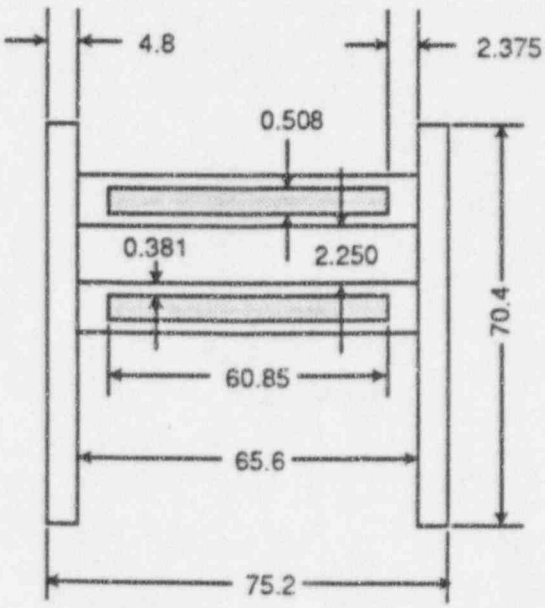
Fig. 4 Models for HEU and LEU Fuel Elements
(Dimensions in mm)



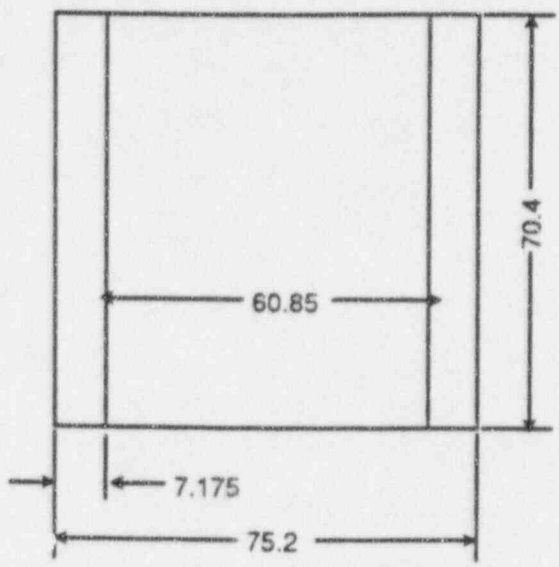
HEU Fuel Element
16 Fueled Plates
2 Non-Fueled Plates



DIF3D Model for
HEU Fuel Element



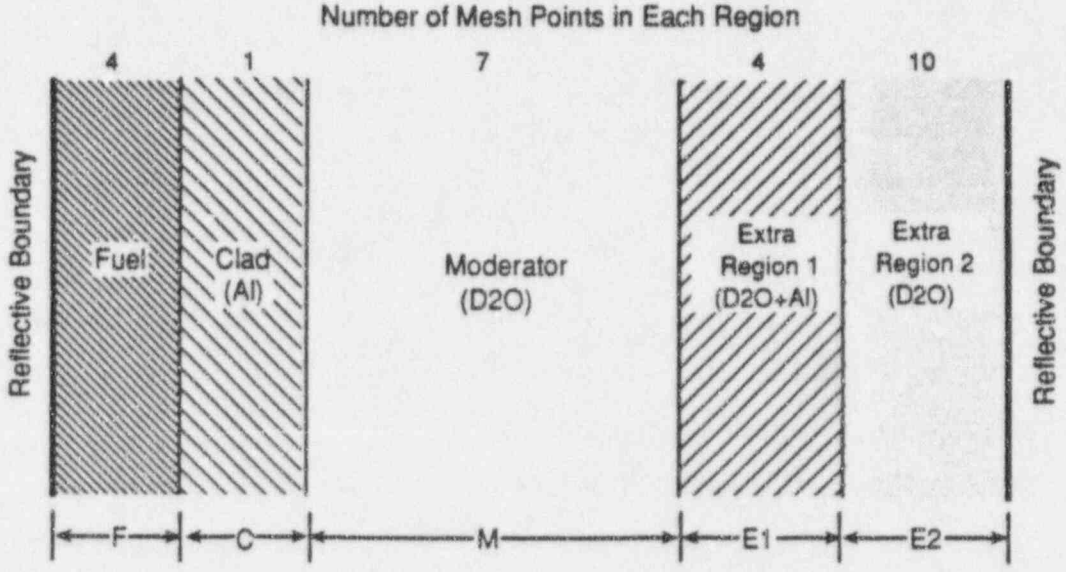
LEU Fuel Element
18 Fueled Plates
2 Non-Fueled Plates



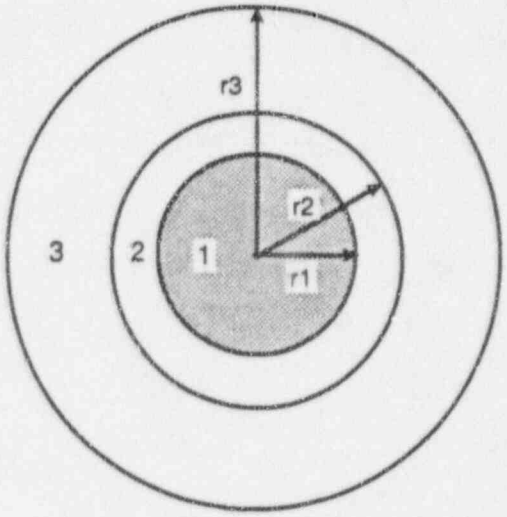
DIF3D Model for
LEU Fuel Element

Fig. 5. EPRI-CELL Model for Generating Fuel Element Cross Sections
(Dimensions in mm)

Core	Fuel Plates /Elem.	Half Fuel Region (F)	Clad Region (C)	Half Moderator Region (M)	Half Extra Region1 (E1)	Extra Region2 (E2)	Extra Region1 Al/D2O Volume Fraction
HEU	16	0.254	0.381	1.346	0.6244	7.2934	0.6499/0.3501
LEU	18	0.254	0.381	1.125	0.6567	6.7654	0.6611/0.3389



Unit-Cell Specifications for Fueled and Non-Fueled Portions of Fuel Element
(Fuel Region Cross Sections: Collapse using Fluxes over F, C, and M;
Non-Fuel Cross Sections: Collapse using Fluxes over E1 Only)



Region 1: Homogenized Fuel
r1 = 35.8 mm
Region 2: Mixture of Al + D2O
r2 = 41.1 mm
Region 3: D2O
r3 = 80.0 mm

Unit-Cell Specifications for D2O Between Fuel Elements
(Collapse Cross Sections using Fluxes over Region 3 Only.)

Each EPRI-CELL case was run three times using the local fine-group spectra over the fueled region and the two extra regions to collapse the fine group cross sections into 7 broad groups. This procedure was performed because the fueled region, the non-fueled region inside the fuel assembly and the water outside each fuel assembly were modeled as separate regions in the diffusion theory model of the reactor. Cross sections for the heavy water and graphite reflectors and for the fuel assembly end fittings were calculated using a unit cell model consisting of a pure ^{235}U fission spectrum on a 10 cm thick slab of water.

5.4.2 Reactor Models

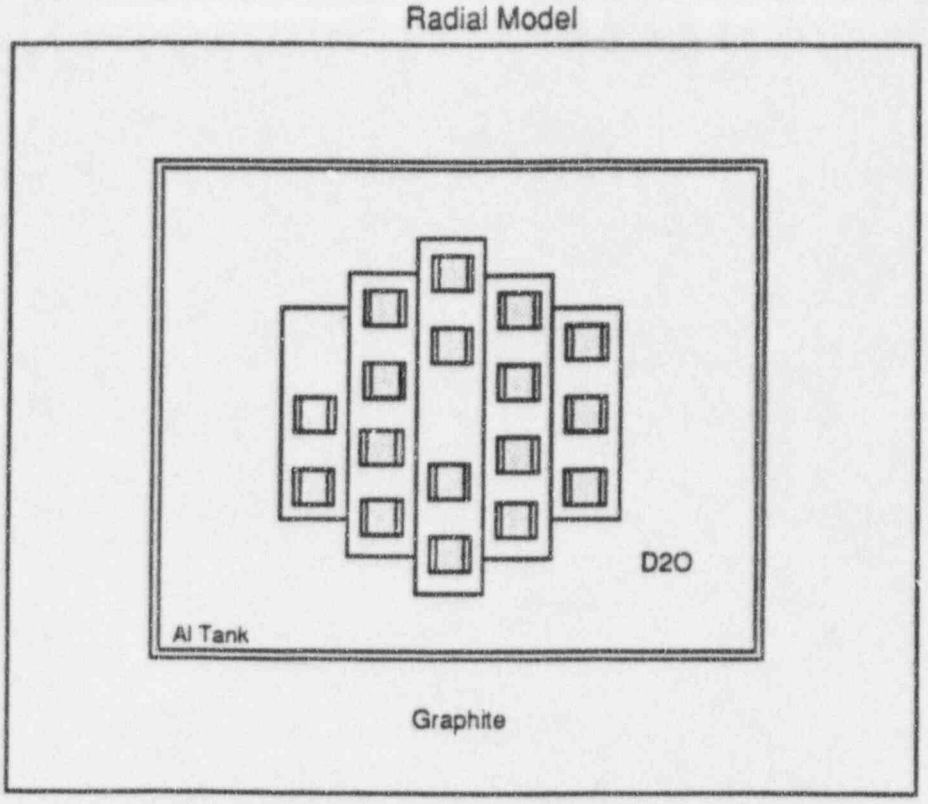
Reactor calculations were performed in three dimensions using the VIM continuous energy Monte Carlo code 14,15 and the DIF3D diffusion theory code[16].

A detailed Monte Carlo model of the reactor was constructed including all fuel assemblies, the shim-safety rods, the regulating rod, beam tubes and experiment penetrations, the bio-medical facility, and the top and bottom reflector regions in order to obtain absolute excess reactivities and shutdown margins for comparison with limits specified in the Technical Specifications. Nuclear cross sections were based on ENDF/BV data. The experiment facilities that were modeled are shown in Table 3.

In diffusion theory, the reactor was modeled in rectangular geometry with a heterogeneous representation of the fueled and non-fueled portions of the fuel assemblies and the water between fuel assemblies (see Fig. 6). The four shim-safety rods (control arms) that swing between the fuel assemblies, the regulating rod, and the various reactor penetrations (reactivity worth ~4.5% $\Delta k/k$)were not included in diffusion theory model. The bottom axial reflector and the radial reflector were also simplified. The LEU model is identical with the HEU model except for the fuel assembly materials.

A simplified Monte Carlo model corresponding with the diffusion model was also constructed in order to verify that the diffusion theory model was correct.

Fig. 6. Radial and Axial Models for Diffusion Theory Calculations



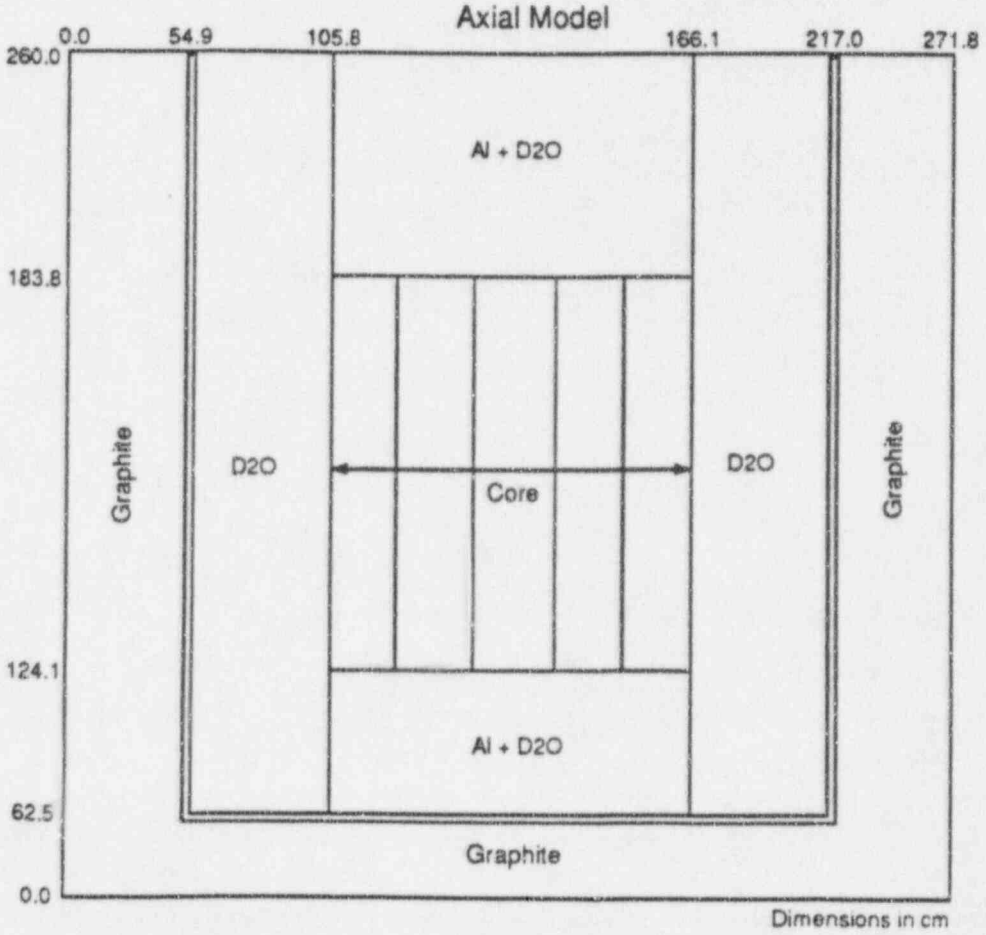


Table 3. Experimental Facilities Included in the Detailed Monte Carlo Model.

8	vertical experiment tubes filled with air in the D_2O reflector
2	vertical experiment tubes filled with air in the graphite reflector
12	vertical experiment tubes filled with graphite in the graphite reflector
14	horizontal beam tubes filled with air and penetrating the D_2O and graphite reflectors
8	horizontal beam tubes filled with graphite and penetrating both reflectors
2	horizontal beam tubes filled with 12" graphite, remainder air and penetrating both reflectors

Biomedical Facility: A portion of the graphite reflector between the vessel and the biomedical facility consists of a bismuth shield and air (see Fig. 1).

Thermal Column

5.5. NEUTRONIC PARAMETERS

5.5.1 Critical Experiment for HEU Core

In 1974, a critical experiment was built using 9 fresh HEU fuel assemblies. The core was made critical at different shim-safety blade positions[17] with the regulating rod nearly fully-withdrawn and nearly fully-inserted. The k_{eff}'s calculated for these critical configurations using the detailed Monte Carlo model were 0.991 $\pm$ 0.002 and 0.988 + 0.002. The corresponding reactivity values were -0.91 $\pm$ 0.20% $\Delta k/k$ and -1.22 $\pm$ 0.22% $\Delta k/k$, respectively. The reactivity bias of about -1.0 $\pm$ 0.3% $\Delta k/k$ in the calculations is attributed to uncertainties in the nuclear cross sections and uncertainties in the reactor materials.

5.5.2 Cold Clean Excess Reactivities

Calculated excess reactivities (including reactivity bias) for the reference HEU and LEU cores with 17 fresh fuel assemblies are shown in Table 4. The Technical Specifications limit the excess reactivity to a maximum of 11.9% $\Delta k/k$. The LEU core is expected to satisfy this requirement.

Table 4. Excess Reactivities of HEU and LEU Cores with 17 Fuel Assemblies

	Calculated Excess React.[1], %Δk/k± 1σ	
	Fresh HEU Core	Fresh LEU Core
Detailed Monte Carlo Model	11.7 ± 0.4	9.4 ± 0.4
Simplified Monte Carlo Model[2]	16.8 ± 0.4	14.3 ± 0.4
Diffusion Theory Model[2]	16.6	14.6

[1]The reactivity bias of -1.0 ± 0.3% Δk/k was added to calculated values.
[2] Without experiment penetrations, shim-safety blades, and regulating rod.

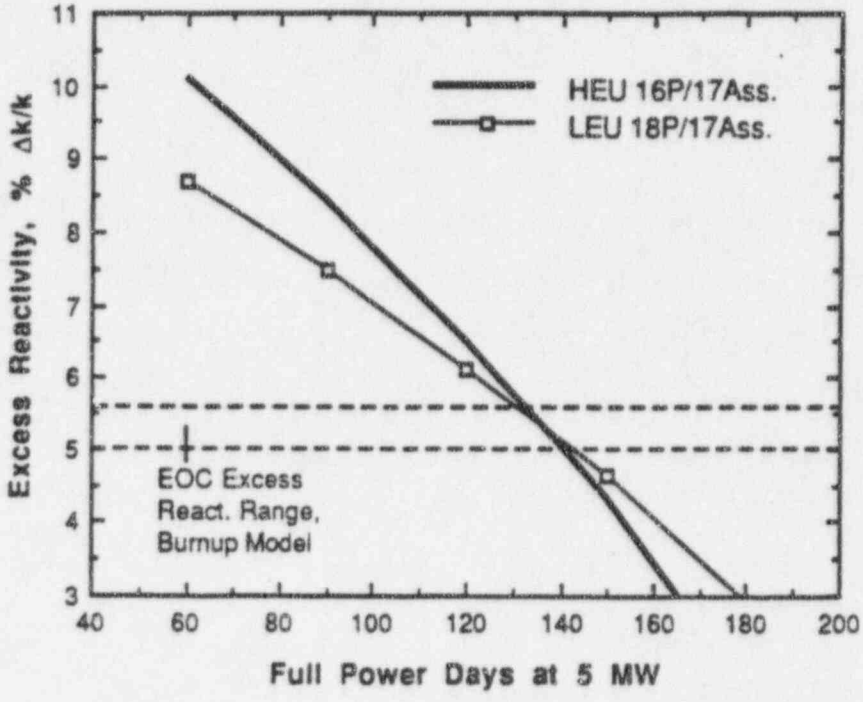
Differences between the detailed and simplified Monte Carlo models were described in Section 4.2. The reactivity effect of (1) replacing all vertical and horizontal experiment facilities inside the heavy water vessel with D_2O, (2) replacing all air-filled experiment facilities in the graphite reflector with graphite, and (3) replacing the bismuth shield and air in front of the biomedical facility with graphite was calculated[18] to be 4.5 ± 0.3% Δk/k. The worth of replacing the control absorbers in their fully-withdrawn position with D_20 was calculated to be 0.1 ± 0.3% Δk/k, a value consistent with zero worth. Thus, the simplified Monte Carlo model and the diffusion theory model are reasonable representations of the reactor if the reactivity worth of the experiment facilities is taken into account.

5.5.3 Burnup Calculations

Burnup calculations were run using the REBUS code[19] for HEU and LEU cores with 17 fuel assemblies to estimate fuel lifetimes. Reactivity profiles (including the 1%Δk/k reactivity bias) are shown in Fig. 7 over a limited burnup range. Excess reactivity values for fresh cores computed using the diffusion theory model are shown in Table 4. The dashed lines show the end-of-cycle excess reactivity range that accounts for reactivity losses due to experiment facilities (4.5 ± 0.3% Δk/k), cold-to-hot swing (~0.3% Δk/k), and control provision (~0.5% Δk/k) that are not included in the diffusion theory burnup model. Reactivity losses due to

equilibrium Xe and Sm are included in the curves. We conclude that the lifetime of the LEU core will be about the same as that of the HEU core when absolute errors in the calculations are taken into account.

Fig. 7. Burnup Reactivity Profiles



5.5.4 Power Distributions and Power Peaking Factors

Power distributions and nuclear power peaking factors were calculated using the diffusion theory model for HEU and LEU cores with 14 and 17 fuel assemblies. As stated previously, the shim-safety rods, regulating rod, and experiment penetrations were not represented. The results are shown in Fig. 8 for the 14 element cores and in Fig. 9 for the 17 element cores. The reason for calculating cores with 14 fuel assemblies is that this is the minimum GTRR core size and cores with 14 assemblies will be used to compute the thermal-hydraulic safety margins.

From the point of view of thermal-hydraulic safety margins, the most important neutronic parameter is the total 3D power peaking factor (the absolute peak power density in a fuel assembly divided by the average power density in the core). The total power peaking factor is defined here as the product of two components: (1) a radial factor defined as the average power density in each assembly divided by the average power density in the core and (2) an assembly factor defined as the peak power density in each assembly divided by the average power density in that assembly. The assembly factor is a pointwise factor computed at the mesh interval edge and includes both planar and axial power peaking.

The data in Figs. 8 and 9 show that the power distributions and power peaking factors are nearly the same in fresh HEU and LEU cores with 14 fuel assemblies and in fresh HEU and LEU cores with 17 fuel assemblies. The percentages of reactor power shown in Figs. 8 and 9 do not add to 100 % because ~2.5% of the energy is deposited outside the fuel assembles.

5.5.5 Reactivity Coefficients and Kinetics Parameters

Reactivity coefficients were computed for HEU and LEU cores with 14 and 17 fresh fuel assemblies as functions of temperature and void fraction using the 3D diffusion theory model. Also computed were the whole-core void coefficient, the reactor isothermal temperature coefficient, and the prompt neutron lifetime. Fresh cores were calculated because they are limiting cores. As fuel burnup increases, the neutron spectrum becomes softer and the reactivity coefficients become more negative.

Fig. 8. Power Distributions and Power Peaking Factors
HEU and LEU Cores with 14 Fuel Assemblies

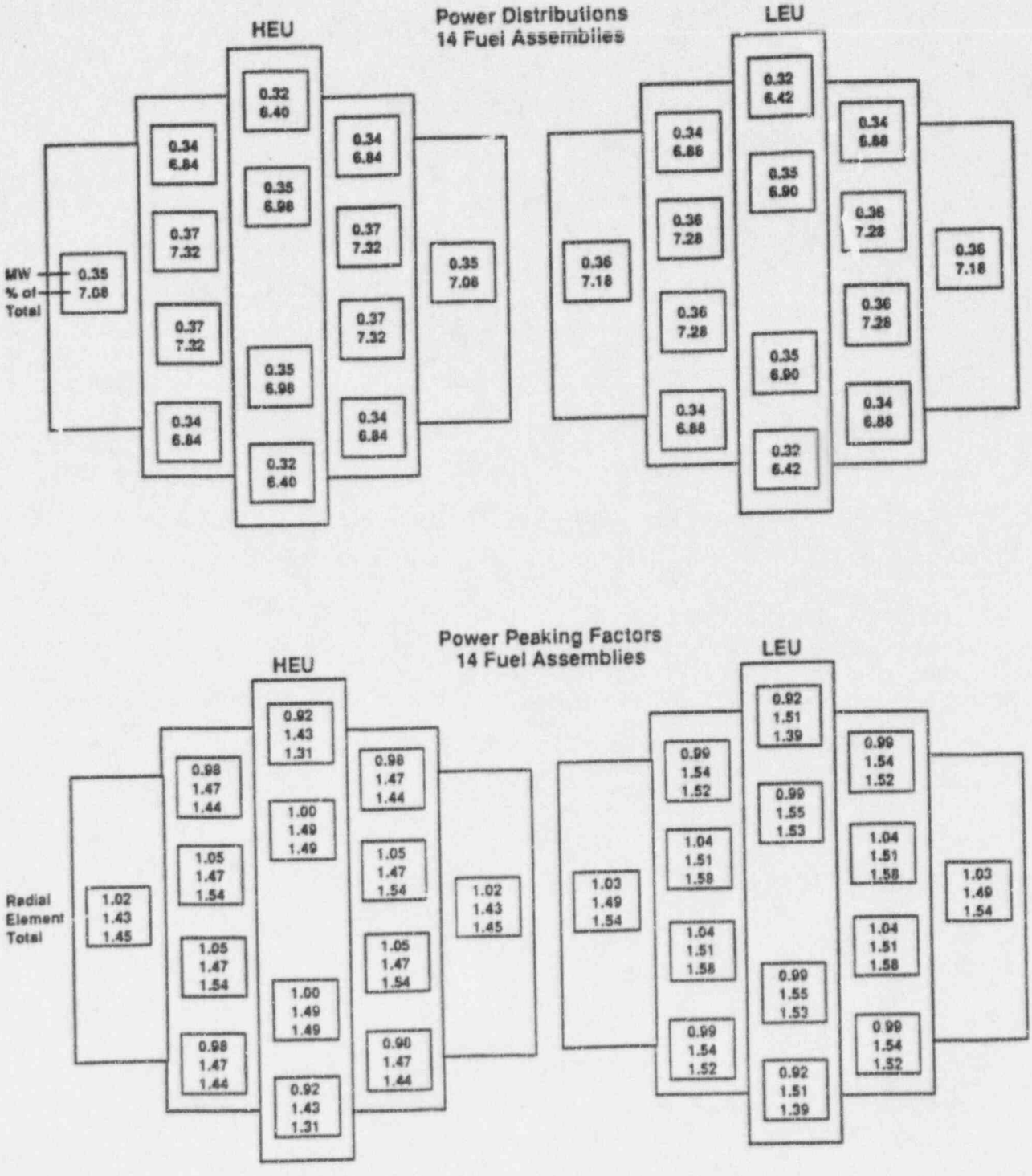
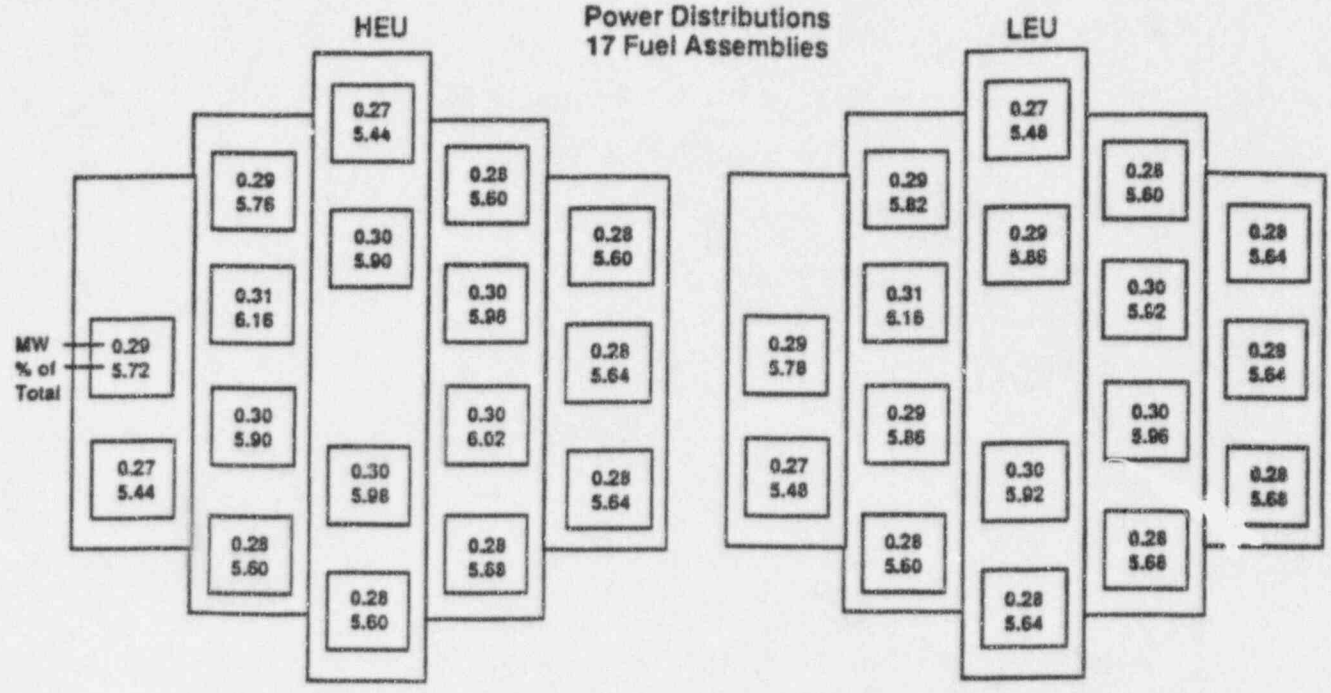
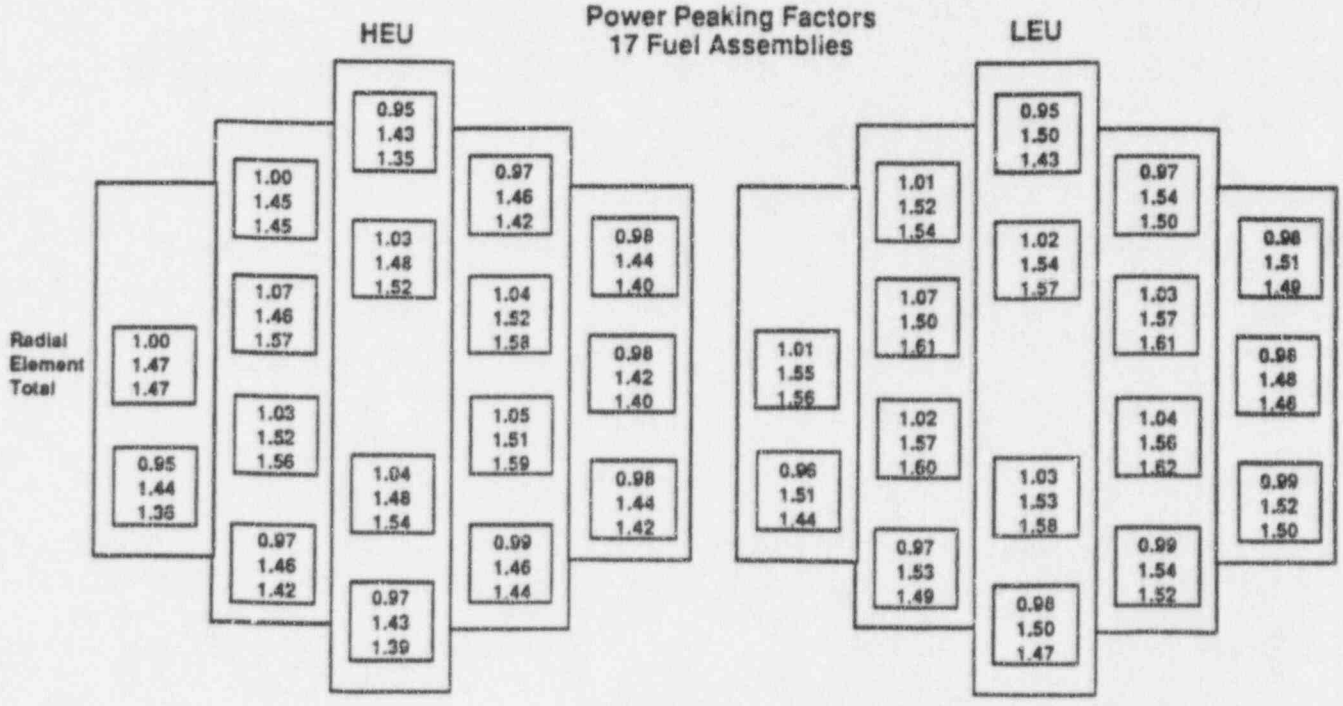


Fig. 9. Power Distributions and Power Peaking Factors
HEU and LEU Cores with 17 Fuel Assemblies





Reactivity changes were calculated separately for changes in coolant temperature, coolant density, and fuel temperature while holding all heavy water outside the fuel assemblies at 23°C. Slopes of the reactivity feedback components at 45°C are shown in Table 5 along with the void coefficient for a uniform 1% change in the coolant density in all fuel assemblies. These reactivity feedback coefficients will be used in the transient analyses in Sections 9 and 10 because the transients considered involve heating of the fuel and coolant. Heating of the heavy water outside the fuel assemblies would have only a small effect because of the time constants involved in the transients.

Table 5. Reactivity Coefficients (% Δk/k/°C at 45°C) and Kinetics Parameters

	HEU		LEU	
	14 Ass.	17 Ass.	14 Ass.	17 Ass.
Coolant Temperature	-0.0062	-0.0055	-0.005	-0.0053
Coolant Density	-0.0014	-0.0014	-0.0012	-0.0013
Fuel Doppler	~0.0	~0.0	-0.0017	-0.0020
Sum	-0.0076	-0.0069	-0.0084	-0.0086
Whole Reactor Isothermal[1]	-0.0224	-0.0201	-0.0232	-0.0215
Void Coefficient[2]	-0.0383	-0.0392	-0.0333	-0.0350
l_p[3],μs	780	704	745	695
β_{eff}	0.007554	0.007554	0.0075 - 0.00765	

[1] Includes fuel, coolant, inter-assembly water, and reflector.
[2] % Δk/k/% Void. Uniform voiding of coolant in all fuel assemblies.
[3] Calculated prompt neutron lifetime.
[4] Measured effective delayed neutron fraction.
[5] Estimated value.

The sum of the coolant and fuel Doppler reactivity coefficients in Table 5 are slightly more negative in the LEU cores than in the HEU cores. The Doppler coefficient actually has a larger weight than shown in Table 5 because the fuel temperature normally increases more rapidly than the coolant temperature. The coolant void coefficient for all fuel assemblies in the core is slightly more negative in the HEU cores than in the LEU cores.

The reactor isothermal temperature coefficient for the 5 MW clean core with 16 HEU fuel assemblies was calculated in the GTRR Safety Analysis Report (Ref 1, p. 98) to be -0.0232% $\Delta k/k/°C$ at 45°C. The reactor isothermal temperature coefficients shown in Table 5 for clean cores with 14 and 17 HEU assemblies are in good agreement with this value. The corresponding reactor isothermal temperature coefficients for LEU cores with 14 and 17 assemblies are slightly more negative than those for the HEU cores. A breakdown of calculated isothermal reactivity feedback components for the coolant, interassembly water, and reflector of an HEU core with 17 fresh fuel assemblies is shown in Attachment 1.

In April 1992, the whole-reactor isothermal temperature coefficient was measured to be - 0.0338 $\Delta k/k/°C$ in a 17 assembly HEU core with about 10,000 MW-hr burnup over the period 1974-1992 (R. Karam, GTRR; private communication). Although these measured and calculated data cannot be compared directly (temperature coefficients normally become more negative with increasing burnup), it does indicate that measured temperature coefficients in the GTRR may be more negative than calculated values.

The calculated prompt neutron lifetimes shown in Table 5 for the LEU cores with 14 fuel assemblies and with 17 fuel assemblies are slightly smaller than those in the corresponding HEU cores because the LEU cores have a slightly harder neutron spectrum.

The fission component of the delayed neutron fraction in both the HEU and LEU cores was calculated to be 0.0071. The difference between this value and the β_{eff} of 0.00755 measured in the HEU core is attributed to delayed neutrons resulting from dissociation of heavy water by neutrons and gamma rays. The latter component of β_{eff} has not been computed here. Since the fission components of β_{eff} were computed to be the same in the HEU and LEU cores, we expect that the heavy water components of β_{eff} and thus the total effective delayed neutron fractions will be very similar as well.

5.6. SHUTDOWN MARGINS

The Technical Specifications require that the reactor have a shutdown margin of at least 1% Δk/k with the most reactive shim-safety blade and the regulating rod fully withdrawn. Measured reactivity worths[20] of the shim-safety blades in the present HEU core are shown in Table 6. The blade with the highest reactivity worth is blade #3.

Table 6. Measured Reactivity Worths of Shim-Safety Blades in HEU Core (9/26/90)

Shim-Safety Blades	Reactivity Worth %Δk/k
Blade #1	5.55
Blade #2	4.66
Blade #3	6.21
Blade #4	4.41

Table 7 compares shutdown margins calculated using the detailed Monte Carlo model for HEU and LEU cores with 17 fresh fuel assemblies. The regulating rod and shim-safety blade #3 were fully-withdrawn and the other three shim-safety blades were fully-inserted. The results show that both cores satisfy the 1% Δk/k shutdown margin requirement of the Technical Specifications.

Table 7. Calculated Shutdown Margins for HEU and LEU Cores with 17 Fresh Fuel Assemblies.

Core	Shutdown Margin, % Δk/k
HEU	-7.14 ± 0.25
LEU	-8.84 ± 0.21

In addition to the automatic protective systems, manual scram and reflector drain provide backup methods to shut the reactor down by operator action. The top of the core is covered by 29.75 inches of D_2O, measured from the top of the fuel meat. The top 28 inches of D_2O can be drained through a 4 inch pipe which connects the reactor vessel to the storage tank of the primary D_2O system. The reactivity worth of

the top 28 inches of reflector was measured[21] to be 2.75% Δk/k in an HEU core composed of 15 fuel assemblies with 142 g ^{235}U per assembly.

Monte Carlo calculations using the detailed Monte Carlo model were done to compare reactivity worths of the top D_2O reflector in HEU and LEU cores with 17 fresh fuel assemblies (188 g ^{235}U HEU, 225 g ^{235}U LEU). Several calculations were first done for each core to determine shim-safety blade positions that would bring the reactor near critical. Results in Table 8 for cases with 1" and 2" of D_2O reflector above the top of the fuel meat show that the top reflector worths of the HEU and LEU cores are very similar. Thus, the shutdown capability of reflector drain in the LEU core will be very similar to that in the present HEU core.

Table 8. Calculated Top Reflector Worths (% Δk/k) of HEU and LEU Cores with 17 Fuel Assemblies and Control Blades near Critical Positions

Top D_2O Reflector	HEU Core	LEU Core
D_2O 1" Above Fuel Meat	- 2.58 ± 0.29 (1~)	- 2.73 ± 0.31 (1~)
D_2O 2" Above Fuel Meat	- 2.05 ± 0.28	- 2.42 ± 0.30

5.7. Thermal-Hydraulic Safety Parameters

Thermal-hydraulic safety limits and safety margins calculated using the PLTEMP code[22] for the LEU core with 14 fuel assemblies (see Fig. 8) were compared with the thermal-hydraulic safety parameters used as bases for the current Technical Specifications. The analyses by ANL for the LEU core used a combined multiplicative and statistical treatment of a revised set of engineering uncertainty factors. Attachment 2 lists the engineering uncertainty factors used by Georgia Tech for analyses[23] of the HEU core and discusses the factors used by ANL, the rationale for their choice, and the method used to combine them. Results for the HEU core obtained using ANL's statistical treatment of the engineering uncertainty factors agree well with the analyses performed by Georgia Tech.

5.7.1 Safety Limits In the Forced Convection Mode

The current Technical Specifications utilize departure from nucleate boiling (DNB) as a basis for establishing safety limits on reactor power, coolant flow, and coolant inlet (or outlet) temperature. This report evaluates these limits based on flow instability as well as DNB criteria. The modified Wheatherhead correlation[23,24] was used for DNB and the Forgan-Whittle correlation[25,26] was used for flow instability.

Calculated reactor power limits based on DNB and flow instability are shown in Table 9 for 14-assembly HEU and LEU cores with the minimum coolant flow of 1625 gpm and with the coolant low flow limit of 760 gpm. A maximum inlet temperature of 123°F was used in all cases. Power limits based on the flow instability criterion are smaller than those based on DNB, but are still adequate to ensure the safety of the facility. The main reason for the difference in reactor power limits in the HEU and LEU cores is that the manufacturing specifications for LEU silicide dispersion fuel plates contain a factor of 1.2 for homogeneity of the fuel distribution while the HEU alloy fuel has a corresponding factor of 1.03.

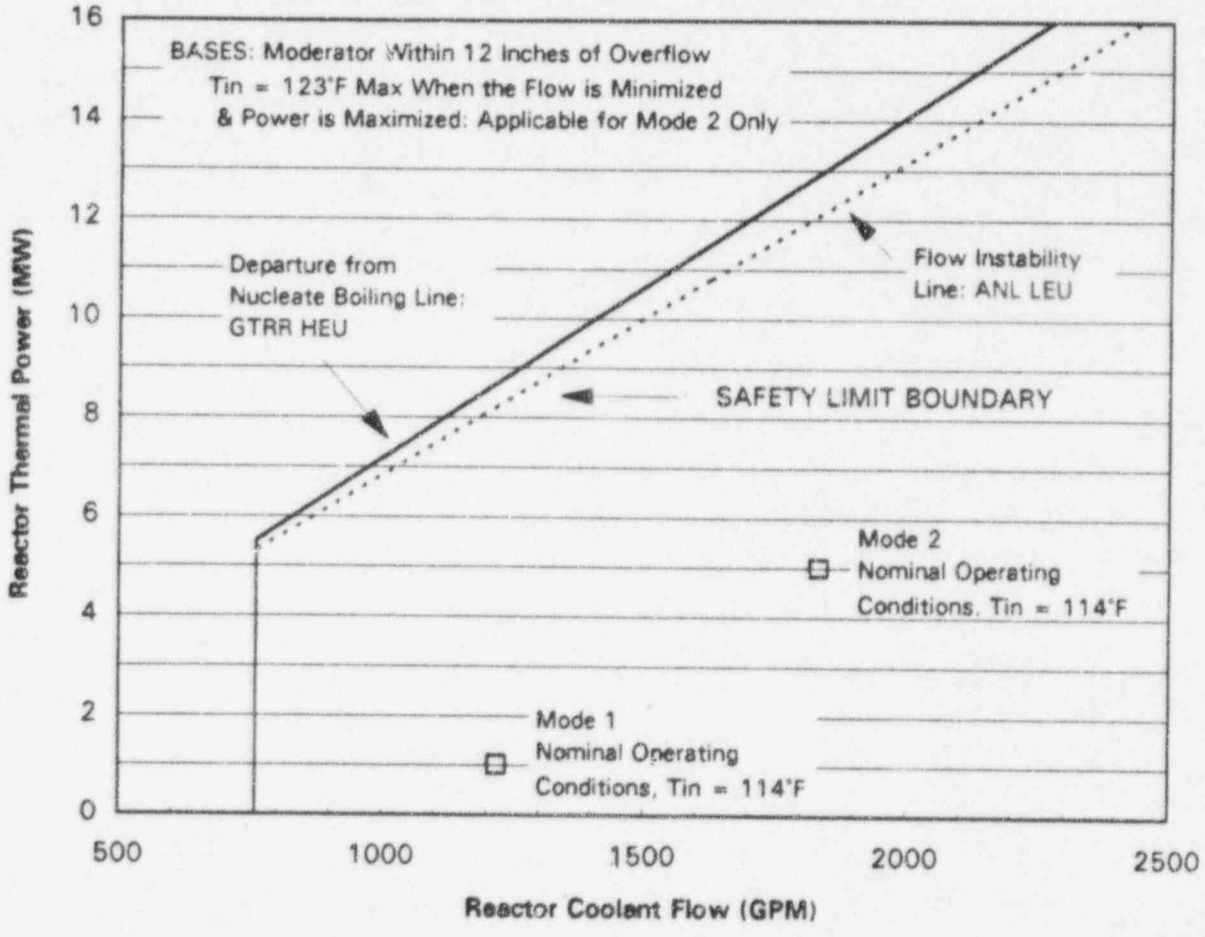
Table 9. Reactor Power Limits in 14 Assembly Cores for a Maximum Inlet Temperature of 123°F Based on Departure from Nucleate Boiling and Flow Instability.

Reactor Coolant Flowrate, gpm	GTRR HEU	ANL-LEU 14	ANL-LEU 17
	Reactor Power Level (MW) for DNB		
760	5.5	5.3	4.9
1625	11.5	10.8	10.7
	Reactor Power Level (MW) for FI		
760	5.3	5.0	4.7
1625	10.6	10.6	10.4

[1] Calculated by ANL using GTRR engineering uncertainty factors in Ref. 23.
[2] Calculated by ANL using revised engineering uncertainty factors (see Attach. 2).

Figure 10 shows the calculated reactor power limits as functions of reactor coolant flow based on DNB for the HEU core and on flow instability for the LEU core. In the LEU core, we recommend a power limit of 10.6 MW based on the flow instability criterion for the minimum coolant flow of 1625 gpm and the maximum inlet temperature of 123°F..

Fig. 10 GTRR Safety Limit for Forced Convection



More detailed data for the minimum coolant flow rate of 1625 gpm and the maximum inlet temperature of 123°F are shown in Table 10. The LEU fuel assembly has reduced power per plate, a smaller flow area, a higher coolant velocity, and a larger pressure drop due to friction. The peak cladding surface temperature is larger by about 5°F and the margins to DNB and flow instability are adequate.

Table 10. Thermal-Hydraulic Data for 14 and 17-Assembly Cores with the Minimum Coolant Flow of 1625 GPM and the Maximum Inlet Temperature of 123°F.

	GTRR HEU	ANL-LEU 14	ANL-LEU 17
Coolant Velocity, m/s	2.44	2.61	2.15
Friction Pressure Drop[3], kPa	10.9	15.0	10.5
Power/Plate[4], kW	21.2	18.8	15.5
Outlet Temperature of Hottest Channel, °F	157	156	157
Peak Clad Surface Temperature, °F	219	224	224
Minimum DNBR[5]	2.29	2.17	2.14
Limiting Power Based on Min. DNBR, MW	11.5	10.8	10.7
Flow Instability Ratio (FIR)[6]	2.12	2.11	2.07
Limiting Power Based on FIR, MW	10.6	10.6	10.4

[1] Calculated by ANL using engineering uncertainty factors used in Ref. 23.
[2] Calculated by ANL using revised engineering uncertainty factors (see Attachment 2).
[3] Pressure drop across active fuel only.
[4] Assuming 95% of power deposited in fuel.
[5] Using modified Weatherhead Correlation[23,24] for DNB.
[6] Using Forgan-Whittle Correlation[25,26] with $\eta = 25$.

Safety limits for the reactor inlet temperature were calculated at the maximum reactor power of 5.5 MW and the minimum coolant flow of 1625 gpm. The results are shown in Table 11. Data for the GTRR-HEU core are based on DNB. ANL results for the LEU core are based on both DNB and flow instability criteria. A safety limit for the reactor outlet temperature was then established by adding the average temperature rise across the core to the limiting inlet temperature. These results show that the HEU and LEU cores have nearly identical safety limits on the reactor inlet and outlet temperatures.

Table 11. Safety Limits on Reactor Inlet and Outlet Temperatures.

	GTRR-HEU[1]	ANL-LEU[2]	
Parameter	DNB	DNB	Flow Instability
Limiting Reactor Inlet Temp., °F	172	171	170
Ave. Coolant Temp. Rise across Core, °F	16	17	17
Limiting Reactor Outlet Temp., °F	188	188	187

[1]Data from Ref. 23 based on DNB criterion.
[2] Calculated using ANL engineering uncertainty factors in Attachment 2.

5.7.2 Safety Limits In the Natural Convection Mode

The current Technical Specifications state that the reactor thermal power shall not exceed two (2) kW in the natural convection mode. This specification is based on GTRR experience showing that no damage to the core and no boiling occurs without forced convection coolant flow at power levels up to 2 kW. We expect that this specification will also hold in the LEU core because the average power per fuel plate will be lower in the LEU core. Each LEU fuel assembly will contain 18 fuel plates while each HEU assembly contains 16 fuel plates.

5.7.3 Limiting Safety System Settings In the Forced Convection Mode

The safety system trip setting in the current GTRR Technical specifications for power levels >1 MW and for power levels ≤ 1 MW are shown in Table 12.

Table 12. Safety System Trip Settings

Parameter	Reactor Power Level >1 MW	Reactor Power Level <1 MW
Thermal Power	5.5 MW	1.25 MW
Reactor Coolant Flow	1625 GPM	1000 GPM
Reactor Outlet Temperature	139°F	125°F

These safety system trip settings are based on a criterion[3] that there shall be no incipient boiling during normal operation. The criterion is applied by ensuring that the surface temperature at any point on a fuel assembly does not exceed the coolant

saturation temperature at that point. This criterion is conservative because there is an additional margin of ~26°F between the D_2O saturation temperature and the temperature at which onset of nucleate boiling occurs.

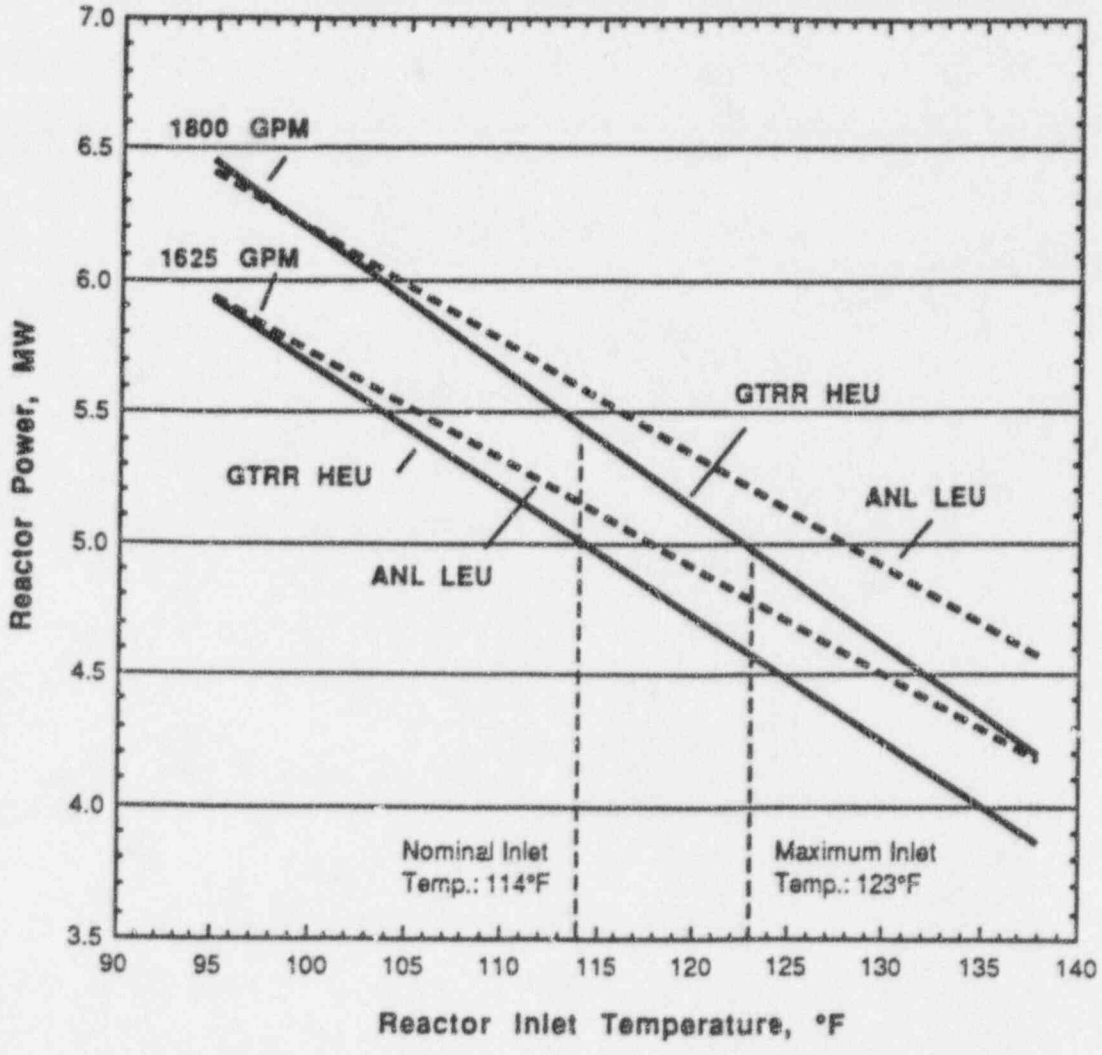
Figure 11 shows the combinations of reactor power, coolant flow rate, and reactor inlet temperature that were calculated to have zero subcooling (fuel surface temperature = coolant saturation temperature) for HEU and LEU cores with 14 fuel assemblies. Data for the HEU core were reproduced from Fig. 1 of Ref. 3. Table 13 provides the parameter combinations which correspond with the safety system trip settings shown in Table 12. The trip setting of 139°F on reactor outlet temperature was obtained by adding the 16°F temperature rise across the core to the maximum inlet temperature of 123°F. Similar considerations based on operation during the period 1964 to 1973 were applied to determine the safety system trip settings for power levels equal to or less than 1 MW.

Parameter combinations that have zero subcooling in the LEU core are shown in Table 13 and in Fig. 11. Since the values for the LEU core are more conservative than those for the HEU core, the current safety system trip settings for the HEU core can also be used for the LEU core.

Table 13. Parameter Combinations for Zero Subcooling with 14-Assembly HEU and LEU Cores

	GTRR HEU			ANL LEU		
Reactor Power, MW	5.5	5.0	5.0	5.6	5.0	5.0
Coolant Flow Rate, gpm	1800	1625	1800	1800	<1625	1800
Reactor Inlet Temp., °F	114	114	123	114	114	128
Temp. Rise Across Core, °F	16	16	16	17	17	17
Reactor outlet Temp., °F	130	130	139	131	131	145

Fig. 11. Thermal-Hydraulic Limits Based on Zero Subcooling For Operation at Power Levels ≤ 5 MW.



The results in Table 14 show that the degree of subcooling (ΔT_{sub}) at the hottest spot of the limiting fuel assembly under normal operating conditions is expected to be 11°F in the LEU core and 8°F in the HEU core. Another criterion that is often used in research reactors is that the margin to onset of nucleate boiling (ONB) should be equal to or greater than 1.2. ONB occurs at a temperature of about 246°F, which is ~26°F above the D_2O saturation temperature of 220°F. The margin to ONB in the LEU core was computed by increasing the reactor power until ONB occurred and dividing by the nominal reactor power of 5 MW. These margins are adequate to ensure that the LEU core can be operated safely at a power level of 5 MW.

Table 14. Margins to D_2O Saturation Temperature and ONB for 14-Assembly Cores

Parameter	GTRR-HEU[1]	ANL-LEU[2]
Thermal Power, MW	5.0	5.0
Reactor Coolant Flow, gpm	1800	1800
Reactor Inlet Temp., °F	114	114
ΔT_{sub}, °F	8	11
Margin to ONB[3]	-	1.44
Limiting Power Based on ONB, MW	-	7.2

[1] Data from Refs. 3 and 23.
[2] Calculated using ANL engineering uncertainty factors in Attachment 2.
[3] Using the Bergles and Rohsenow correlation[27].

Calculations were also done to examine the adequacy of the current safety system trip settings shown in Table 12 for operation at power levels equal to or less than 1 MW. Since data from analyses of the HEU core by Georgia Tech were not available, calculations were done using the GTRR-HEU and the ANL-LEU engineering uncertainty factors shown in Attachment 2, a thermal power of 1.25 MW, a reactor coolant flow of 1000 gpm, and an inlet temperature of 123°F. The results shown in Table 15 for the degree of local subcooling (ΔT_{sub}) and the flow instability ratio indicate that the current trip settings on reactor power and coolant flow are conservative and are adequate to ensure the safety of the facility for operation at power levels that are ≤ 1 MW.

Table 15. Selected Thermal-Hydraulic Safety Margins with 14-Assembly Cores and Power ≤ 1 MW.

Parameter	GTRR-HEU[1]	ANL-LEU[2]
Thermal Power, MW	1.25	1.25
Reactor Coolant Flow, gpm	1000	1000
Reactor Inlet Temp., °F	123	123
Peak Surface Clad Temp., °F	162	164
ΔT_{sub}, °F	58	56
Flow Instability Ratio	5.4	5.3

[1]Calculated using GTRR HEU engineering uncertainty factors in Attachment
[2]Calculated using ANL LEU engineering uncertainty factors in Attachment 2.

5.7.4 Limiting Safety System Settings In the Natural Convection Mode

The Technical Specifications state that the reactor thermal power safety system setting shall not exceed 1.1 kW when operating in the natural convection mode. This specification is based on GTRR experience showing that the reactor can be operated at one kW indefinitely without exceeding a bulk reactor temperature of 123°F. We expect that this safety system trip setting will also be adequate for the LEU core.

5.8. COOLING TIME REQUIREMENTS

The Technical Specifications for the HEU core state that containment integrity shall be maintained when the reactor has been shutdown from a power level greater than 1 MW for less than eight hours. In addition, a minimum cool down time of twelve hours is required before fuel assemblies are transferred out of the reactor.

Fuel melting and subsequent release of fission products could result from a loss-of-coolant accident following reactor shutdown if sufficient decay heat is present. Containment integrity is therefore required until the decay heat generation rate is less than that required to melt the fuel plates. A limit of 450°C was set in the Technical Specifications as the upper value for a fuel plate temperature to preclude melting of the plates. The decay time needed to ensure that this temperature would not be reached was calculated in Ref. 23.

The analysis method and input parameters described in Ref. 23 were used to reproduce the results for the HEU core. The same methodology was then used for the LEU core, with modification of the input parameters appropriate for the LEU fuel assembly design. A standard 3-week operating history consisting of 4.33 days at full power of 5 MW and 2.67 days shutdown was used for 14 assembly cores with HEU and LEU fuel. The analysis in both cases was applied to a fuel assembly which has been subjected to a power peaking factor of 1.5 (see Fig. 8). As in Ref. 23, the peak power was increased by 17% to account for the incremental heat contribution due to additional gamma heating from surrounding fuel assemblies in the core and was decreased by 15% to take credit for an improved convection condition in the reactor vessel.

Three input parameters that were used for the HEU fuel assembly in Ref. 23 were modified for the LEU fuel assembly design: (1) the parameter hA_{CTRR} was reduced from 3.03×10^{-4} kW/°C for an HEU plate to 2.88×10^{-4} kW/°C for an LEU plate based on the heat transfer areas of the HEU and LEU fuel meat shown in Table 1, (2) the mass of aluminum associated with one fuel plate was reduced from 0.418 lbm for an HEU plate to 0.377 lbm for an LEU plate, mainly because U_3Si_2 fuel particles occupy approximately 31% of the fuel meat volume in an LEU plate; no credit was taken for the specific heat of the U_3Si_2 particles, and (3) most importantly, the maximum power per fuel plate in the LEU assembly was reduced by a factor of 16/18 since an HEU assembly contains 16 fueled plates and an LEU assembly contains 18 fueled plates.

The results for loss-of-coolant from the reactor vessel after eight hours of cooling showed a maximum plate temperature of 425°C in the HEU core and 400°C in the LEU core. The maximum temperature occurred 45 minutes after loss-of-coolant in the HEU core and 50 minutes after loss-of-coolant in the LEU core. For the more confined heat transfer situation, without gamma rays from other fuel assemblies, but with a restricted heat transfer volume, the maximum fuel plate temperature after a twelve hour cool down was calculated to be 361°C for an HEU plate and 340°C for an LEU plate. The maximum temperature occurred 60 minutes after removal from the HEU core and 50 minutes after removal from the LEU core.

We conclude that the current Technical Specification requirements on cooling times are more conservative for the LEU core than for the HEU core. The most important factor is the reduced power per plate in the LEU core. However, any reduction of technical specification cooling time requirements for the LEU core should be based on measurements in the GTRR.

5.9. LIMITATIONS OF EXPERIMENTS

The Technical Specifications contain three limitations of experiments that could be affected by changing the fuel in the core from HEU to LEU:

a) The magnitude of the potential reactivity worth of each unsecured experiment is limited to 0.004 $\Delta k/k$.

b) The potential reactivity worth of each secured removable experiment is limited to 0.015 $\Delta k/k$.

c) The sum of the magnitudes of the static reactivity worths of all unsecured experiments which coexist is limited to 0.015 $\Delta k/k$.

The objective of these specifications is to prevent damage to the reactor and to limit radiation dose to personnel and the public in event of experiment failure. Qualification of the PARET code that was used for the transient analysis is discussed first, followed by the calculated results.

5.9.1 Comparison of Calculations with SPERT-II Experiments

The PARET code[28] was originally developed at the Idaho National Engineering Laboratory for analysis of the SPERT-III experiments, which included both pin-type and plate-type cores and pressures and temperatures in the range typical of power reactors. The code was modified by the RERTR Program at ANL to include a selection of flow instability, departure from nucleate boiling, single- and two-phase heat transfer correlations, and properties libraries for light water and heavy water that are applicable to the low pressures, temperatures, and flow rates encountered in research reactors.

To validate the PARET code for use with heavy water reactors, calculated and measured data were compared[29] for the SPERT-II BD-22/24 HEU core[30] (24 MTR-type fuel elements with 22 plates per element). This core is similar to the GTRR in design. The tests performed in the BD-22/24 core included only nondestructive transients. Calculated transient parameters shown in Ref. 29 are in very good agreement with the measured data and validate the PARET code for use in calculating transients in heavy water research reactors.

5.9.2 Inadvertent Reactivity Insertions Due to Experiment Failure

The consequences of inadvertent step reactivity insertion of 0.4% $\Delta k/k$ and 1.5% $\Delta k/k$ in HEU and LEU cores with 14 fuel assemblies were evaluated. The model and methods that were used for analysis of the SPERT-II BD-22/24 HEU cores were also used to analyze the HEU and LEU cores of the GTRR.

Inputs to the code for analysis of the GTRR included the prompt neutron lifetime, effective delayed neutron fraction, temperature coefficients of reactivity, and power distributions discussed in Sections 5.4 and 5.5. Temperature coefficients included contributions from only the coolant and the fuel. Axial power distributions for the average channel of the HEU and LEU cores were represented by chopped cosine shapes having peak-to-average power densities of 1.19. In the hot channel, these axial shapes were scaled to produce peak power densities in the limiting fuel assemblies of the HEU and the LEU cores that are consistent with the power distributions shown in Fig. 8.

Calculations were performed for step reactivity insertions of 0.4% and 1.5% $\Delta k/k$ with the reactor at nominal operating conditions of 5 MW thermal power, a coolant flow rate of 1800 gpm, and a reactor inlet temperature of 114°F. A scram signal was initiated when the reactor power reached the safety system overpower trip setting of 5.5 MW. A time delay of 100 ms was assumed between introduction of the scram signal and release of the shim-safety blades. The results of these calculations are shown in Table 16.

Table 16. Results of Assumed Step Reactivity Insertions Due to Experiment Failure

Parameter	HEU Core		LEU Core	
Step Reactivity Insertion, % Δk/k	0.4	1.5	0.4	1.5
Asymptotic Period, s	0.18	0.05	0.18	0.05
Peak Power, MW	7.4	27.5	7.4	27.2
Peak Surface Cladding Temp., °F	184	277	179	267
Peak Coolant Outlet Temp., °F	135	-	135	-

A positive step reactivity change less than 0.4% Δk/k caused by the ejection or insertion of experiments would result in transient behavior that would not exceed the safety limits for the HEU or LEU cores that were discussed in Section 7.1. The peak power of 7.4 MW in both cores is well below the safety limits of 11.5 MW in the HEU core and 10.6 MW in the LEU core. Similarly, the peak coolant outlet temperatures are well below the limiting reactor outlet temperature of 188°F.

Step reactivity insertions of 1.5% Δk/k would result in peak surface cladding temperatures that are far below the solidus temperature of 1220°F (660°C) in the 1100 Al cladding of the HEU core and far below the solidus temperature of 1080°F (582°C) in the 6061 Al cladding of the LEU core. Thus, no damage to the fuel and no release of fission products is expected.

5.10. ACCIDENT ANALYSES

A spectrum of accident scenarios was evaluated by Georgia Tech in its safety documentation[1,3,4] for 5 MW operation. These scenarios included (1) failure of electrical power, (2) failure of various reactor components, (3) a startup accident in which one shim blade and the regulating rod were withdrawn simultaneously, (4) reactivity effects resulting from the melting of fuel plates, (5) assumed maximum positive reactivity insertion, and (6) the Design Basis Accident. A review of these scenarios concluded that only scenarios (3) - (6) could be affected by changing the fuel assemblies from HEU to LEU, and only these scenarios are addressed here.

5.10.1 Startup Accident

The worst case for a possible startup accident in the current HEU core was determined[3] to result from the simultaneous withdrawal of one shim blade and the regulating rod. An experiment was done in the GTRR to simulate reactor behavior when reactivity was added at rate of approximately 0.005 Δk/k per second starting from a power level of 5 kW. Within 3 seconds, the reactor was automatically scrammed by a positive period trip. The power level at the scram point was 6.5 kW. On this basis, it was concluded[3] that if the reactor were operating at 5 MW, the reactor would be scrammed by the overpower trip at 5.5 MW or the log-N period systems would scram the reactor at a power level of no more than 7 MW. Since this is well below the 11.5 MW burnout power level of the GTRR, no fuel plate melting would be expected.

Calculations were done here using the PARET code for the HEU and LEU cores with 14 fuel assemblies in which reactivity was added at a rate of 0.005 Δk/k per second starting from a power level of 5 MW. Except for the reactivity addition rate, inputs to the code were the same as those described in paragraphs 2 and 3 of Section 9.2. Both the HEU and LEU cores were scrammed by the overpower trip at 5.5 MW. A time delay of 100 ms was assumed between introduction of the scram signal and release of the shim-safety blades. Both cores reached a peak power of 5.9 MW at a time of 0.335 s after the transient was initiated. Peak surface cladding temperatures of 177°F and 172°F were reached in the limiting fuel assembly of the HEU and LEU cores, respectively. The peak power is well below the safety limits of 11.5 MW in the HEU core and 10.6 MW in the LEU core. The peak surface cladding temperatures are far below the solidus temperature of 1220°F in the 1100 Al cladding of the HEU core and far below the solidus temperature of 1080°F in the 6061 Al cladding of the LEU core. Thus, no damage to the fuel and no release of fission products is expected.

5.10.2 Reactivity Effects of Fuel Plate Melting

The reactivity effect of melting individual fuel plates within an assembly due to the blockage of individual flow channels was analyzed[4] for the current GTRR HEU core by estimating the reactivity change caused by removing the two central

fuel plates in a fuel assembly at the core center. It was concluded that the loss of one or more fuel plates would result in a negative reactivity effect.

Calculations were done for HEU and LEU cores with 14 and 17 fresh fuel assemblies using the reactor diffusion theory model described in Section 4.2 and in Figs. 8 and 9. The results in Table 17 show that the reactivity effect of removing one or two fuel plates from a fuel assembly near the center of the HEU and LEU cores and replacing the fuel plate volume with D_20 is expected to be negative.

Table 17. Calculated Reactivity Effect of Removing Fuel Plates from a Fuel Assembly Near the Center of the HEU and LEU Cores.

	Reactivity Change, % Δk/k			
	14 Assembly Cores		17 Assembly Cores	
	HEU	LEU	HEU	LEU
1 Fuel Plate Removed	-0.060	-0.037	-0.043	-0.028
2 Fuel Plates Removed	-0.127	-0.078	-0.090	-0.060

5.10.3 Fuel Loading Accident

During refueling operations, all control blades are required to be fully inserted and the top D_2O reflector drained to storage. Calculations in Section 6 indicated that the shutdown margin with the blade of maximum worth stuck out of the core is expected to be - 7.1 ± 0.3% Δk/k in the HEU core and - 8.8 ± 0.2% Δk/k in the LEU core. The shutdown margins will be more negative with all shim safety blades inserted. In addition, the reactivity worth of the top reflector is at least 2% Δk/k.

The current GTRR safety analysis report analyzed a hypothetical fuel loading accident scenario assuming, in violation of established startup procedures, that the shim safety blades are withdrawn so that the reactor is just sub-critical and that the D_2O is at the normal operating level. A fresh fuel assembly was then assumed to be dropped into the center core position, resulting in a sudden reactivity insertion of 2.5% Δk/k. We consider this postulated scenario to be incredible and no analysis of this scenario is presented in this report. The maximum positive reactivity insertion is addressed in Section 5.10.4.

5.10.4 Maximum Positive Reactivity Insertion

The Technical Specifications limit the potential reactivity worth of each secured removable experiment to 1.5%Δk/k and the sum of the magnitudes of the static reactivity worths of all unsecured experiments which coexist to 1.5% Δk/k. The purpose of this analysis is to show that there is a sufficient margin between the maximum allowable reactivity worth of a single experiment and the maximum step reactivity insertion that can be tolerated without fuel damage, assuming failure of reactor scram systems.

Analysis[1] for the current HEU core used SPERT-II experimental data[30] as a basis for estimating the step reactivity insertion that would result in the onset of steam blanketing in the GTRR. In the present analysis, the PARET code was used to compute the step reactivity insertion required to initiate steam blanketing (film boiling) in both the SPERT-II B22/24 core and 14-assembly GTRR cores with HEU and LEU fuel. Some of the kinetics parameters and key PARET results are provided in Table 18. Power peaking factors are similar in the SPERT-II and GTRR cores. The inverse period corresponding to the onset of steam blanketing as determined from the SPERT experimental data[1,30] is about 13 s^{-1}. The PARET code predicts the onset of film boiling for a step insertion of $2.0 (1.5% Δk/k) with an inverse period of 12 s^{-1}, in good agreement with experiment.

The same methodology was used to compute GTRR cores with 14 fuel assemblies. These cores have smaller coolant void coefficients than the SPERT-II B22/24 core, but the step insertions needed to initiate film boiling (~$2.0) and the peak surface cladding temperatures (250-260°C) at the onset of steam blanketing are nearly the same. At the time of peak power, the energy deposited per plate is about the same in the SPERT and GTRR cores. The peak surface cladding temperature at the time of peak power is about 220°C in the GTRR cores and about 204°C in the SPERT core.

The SPERT-II B22/24 tests[30] indicate that even more extensive film boiling (or steam blanketing) does not result in temperatures that exceed the solidus temperature of the cladding. The most extreme case in the test series with a reactivity insertion of $2.95 (2.2% Δk/k) resulted in a peak surface cladding

temperature of 337°C, a temperature far below the solidus temperature of 582°C for 6061 Al cladding. The GTRR SARI also notes that the maximum temperature for large insertions is primarily limited by the energy deposited in the plate with very little effect from the boiling heat transfer.

Since the behavior of the SPERT-II B22124 and GTRR 14-assembly cores is very similar, a step reactivity insertion greater than 2.2% Δk/k would be required to initiate melting of the GTRR LEU core. The margin of at least 0.7% Δk/k above the maximum allowed reactivity worth of 1.5% Δk/k for a single experiment is sufficient to ensure that the facility is safe in the unlikely event that the maximum allowed reactivity were inserted in a step and the reactor scram system failed to function.

Table 18. Comparison of Kinetics Parameters and Onset of Steam Blanketing Results

	SPERT II	14 Assembly GTRR	
	B-22/24	HEU	LEU
Prompt Neutron Generation Time, μs	660	780	745
Beta Effective	0.0075	0.00755	0.00755
Coolant Temperature Coeff., $/°C	-0.00867	-0.00874	-0.0689
Void Coefficient $/% Void	-0.0729	-0.509	-0.0442
Doppler Coefficient, $/°C	~0.0	~0.0	-0.00096
Operating Pressure, kPa	122	127	127
Step Reactivity Insertion, $(% Δk/k)	2.00	1.99	1.95
Inverse Period, s^{-1}	12	19	19
Energy/Plate at t_m, kWs	31.8	31.2	32.0
Peak Cladding Temperature at t_m, °C	204	218	225
Peak Cladding Temperature at Onset of Steam Blanketing, °C	252	257	257

where t_m is the time of peak power.

5.10.5 Design Basis Accident

The Design Basis Accident for the HEU core was determined[4] to be the melting and release of the fission products from one fuel assembly into the containment atmosphere. This accident was assumed to occur during a fuel transfer

operation in which an irradiated fuel assembly was being moved from the core to the fuel storage area using a shielded transfer cask. Fuel assemblies are not normally discharged from the reactor until at least 12 hours after reactor shutdown. This ensures that sufficient fission product decay heat has been removed from the assembly and that the surface temperature of the fuel plates will not reach 450°C when the assembly is moved into the cask.

In spite of administrative controls, it is conceivable that a fuel assembly could be withdrawn from the reactor prior to a 12 hour cooldown period. Some or all of the fuel plates within the assembly could then melt and release some of their fission products into the containment atmosphere.

The source term for evaluating the radiological consequences of this accident was obtained[4] by assuming that an HEU fuel assembly with equilibrium burnup was removed from the core before the 12 hour cooldown period. All of the plates in the fuel assembly melt and the isotopes of iodine, krypton, and xenon were released to the containment. The methodology for the dose calculations and the results are shown in Ref. 4. The limiting dose is the thyroid dose from the iodine isotopes.

Since the HEU and LEU cores operate at 5 MW, neutron flux levels and equilibrium concentrations of iodine, xenon, and krypton will be about the same in the two cores. Burnup calculation results shown in Section 5.3 concluded that the lifetime of the LEU core will be comparable to but probably less than that of the HEU core. As a result, concentrations of the other fission products in LEU fuel assemblies will be the same or less than those in HEU fuel assemblies. The exception is that the LEU assembly will contain larger concentrations of plutonium isotopes. Reference 31 contains a detailed analysis comparing the radiological consequences of a hypothetical accident in a generic 10 MW reactor using HEU and LEU fuels. This analysis concluded that the buildup of plutonium in discharge fuel assemblies with ^{235}U burnup of over 50% does not significantly increase the radiological consequences over those of HEU fuel. Because fission product concentrations in the GTRR HEU and LEU cores are expected to be comparable, the thyroid dose shown in Ref. 4 will be the limiting dose for both cores.

5.11. FUEL HANDLING AND STORAGE

Three Technical Specifications apply to the handling and storage of fuel assemblies. The objective of these specifications is to prevent inadvertent criticality outside of the reactor vessel and to prevent overheating of irradiated fuel assemblies.

Irradiated fuel assemblies are stored in aluminum racks fastened to the side walls of a light water pool. There is one rack along each of the two walls and each rack can accommodate up to 20 assemblies in a linear array. The center-to-center spacing of the assemblies is six inches and the separation between assemblies is about three inches.

A systematic nuclear criticality assessment[32] has been done for infinite-by-infinite arrays of fresh LEU fuel assemblies with ^{235}U contents between 225 and 621 grams using the ORR fuel storage rack spacing specifications[33] of 0.7 inch assembly separation and 6.8 inch row separation. An assembly similar to the GTRR LEU assembly with a ^{235}U content of 225 grams gave a k_{eff} of 0.72, well below the maximum k_{eff} of 0.85 needed to ensure an adequate margin below criticality for storage of irradiated fuel assemblies. The GTRR storage configuration discussed above will have k_{eff} less than 0.72.

Calculations[1] with HEU fuel assemblies have shown that four unirradiated fuel assemblies cannot achieve criticality. Calculations of HEU and LEU cores shown in Section 5.2 indicate that a grouping of four LEU assemblies will be less reactive than the same configuration of HEU assemblies. Thus, the current specification that no more than four unirradiated fuel assemblies shall be together in any one room outside the reactor, shipping container, or fuel storage racks will also hold for the LEU assemblies.

REFERENCES

1. Safety Analysis Report for the 5 MW Geo:gia Tech Research Reactor, December 1967. Final Safeguards Report for the Georgia Tech Research Reactor, February 1963.

2. Appendix A tc Facility License No. R-97: Technical Specifications for the Georgia Tech Research Reactor, Docket No. 50-160, June 6, 1974.

3. Letter, R. S. Kirkland, GTRR Reactor Supervisor, to USAEC, October 22, 1971.

4. Letter, R. S. Kirkland, GTRR Reactor Supervisor, to USAEC, June 23, 1972.

5. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors", NUREG-1313, July 1988.

6. M. M. Bretscher and J. L. Snelgrove, "Comparison of Calculated Quantities with Measured Quantities for the LEU-Fueled Ford Nuclear Reactor," Proc. International Meeting on Research and Test Reactor Core Conversions from HEU to LEU Fuel, Argonne National Laboratory, Argonne, IL, November 8-10, 1982, ANL/RERTR/M-4, CONF-821155, pp.397-425 (1983).

7. M. M. Bretscher, "Analytical Support for the Whole-Core Demonstration at the ORR," Proc. 1986 International Meeting on Reduced Enrichment for Research and Test Reactors, Gatlinburg, TN, November 3-6, 1986, ANL/RERTR/TM-9, CONF-861185, pp.287-301 (1988).

8. R. J. Cornella and M. M. Bretscher, "Comparison of Calculated and Experimental Wire Activations," Proc. 1986 International Meeting on Reduced Enrichment for Research and Test Reactors, Gatlinburg, TN, November 3-6, 1986, ANL/RERTR/TM-9, CONF-861185, pp.302-309 (1988).

9. M. M. Bretscher, J. L Snelgrove, and R. W. Hobbs, "The ORR Whole-Core LEU Fuel Demonstration", Trans. Am. Nucl. Soc. 56, 579-581 (1988).

10. M. M. Bretscher and J. L. Snelgrove, "The Whole-Core U_3Si_2 Fuel Demonstration in the 30-MW Oak Ridge Research Reactor", ANL/RERTR/TM-14, July 1991.

11. IAEA Guidebook on Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels, Addendum on Heavy Water Moderated Reactors, IAEA-TECDOC-324, Appendix F, pp. 182-250 (1985).

12. J. E. Matos, E. M. Pennington, K. E. Freese, and W. L. Woodruff, "Safety-Related Benchmark Calculations for MTR-Type Reactors with HEU, MEU, and LEU Fuels," paper included in IAEA Safety and Licensing Guidebook on Research Reactor Core Conversions from HEU to LEU Fuel, Volume 2, Analytical Verification, Draft #7, June 1985.

13. B.A. Zolotar et al., "EPRI-CELL Code Description," Advanced Recycle Methodology Program System Documentation, Part II, Chapter 5 (Oct. 1975).

14. E. M. Gelbard and R. E. Prael, "Monte Carlo Work at Argonne National Laboratory," in Proc. NEACRP Mtg. Monte Carlo Study Group, July 1-3, 1974, Argonne, Illinois, AN-75-2 (NEA-CRPL-118). Argonne National Laboratory, p. 201 (1975).

15. R. Blomquist, "VIM - A Continuous Energy Neutronics and Photon Transport Code", Proc. Topl. Mtg. Advances in Reactor Computations, Salt Lake City, Utah, March 28-31, 1983, p.222, American Nuclear Society (1983).

16. K. L. Derstine, "DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite Difference Diffusion Theory Problems," ANL-82-64, April 1984.

17. GTRR Operation Log Book, June 10 and June 11, 1974.

18. ANL Internal Memorandum, K.E. Freese to J.E. Matos, "Monte Carlo Calculations for the GTRR Reference Core", March 12, 1991.

19. R. P. Hosteny, "REBUS-2, Fuel Cycle Analysis Capability," ANL,7721, October 1978.

20. "Measured Reactivity Worths of Shim-Safety Blades and Regulating Rod", GTRR Work Order #90317, Record No. 188, 26 September 1990.

21. GTRR Operation Log Book, "Top Reflector Worth", May 17, 1966.

22. K. Mishima and T. Shibata, "Thermal-hydraulic Calculations for KUHFR with Reduced Enrichment Uranium Fuel," KURRI-TR-223 (1982), and K. Mishima, K. Kanda, and T. Shibata, ~Thermal-hydraulic Analysis for Core Conversion to Use of Low-enriched Uranium Fuels in the KUR," KURRI-TR-258 (1984).

23. Appendix A to Facility License No. R-97: Draft Technical Specifications for the Georgia Tech Research Reactor, Docket No. 50-160, May 26, 1972.

24. R. J. Weatherhead, "Nucleate Boiling Characteristics and the Critical Heat Flux Occurrence in Subcooled Axial-Flow Water Systems," ANL-6674, 1963.

25. R. H. Whittle and R. Forgan, "A Correlation for the Minima in the Pressure Drop Versus Flow-Rate Curves for Subcooled Water Flowing in Narrow Heated Channels," Nuclear Engineering and Design, Vol. 6, (1967) pp. 89-99.

26. "Guidebook on Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels", IAEA-TECDOC-233 (1980) pp. 99-106.

27. A. E. Bergles and W. M. Rohsenow, "The Determination of Forced-Convection Surface-Boiling Heat Transfers, " Transactions of the ASME 86 (Series C - Journal of Heat Transfer), pp. 365-371 (August 1964).

28. C. F. Obenchain, "PARET - A Program for the Analysis of Reactor Transients," IDO-17282, Idaho National Engineering Laboratory (1969).

29. W. L. Woodruff, "Additional Capabilities and Benchmarking with the SPERT Transients for Heavy Water Applications of the PARET Code, "Proc. XIIth International Meeting on Reduced Enrichment for Research and Test Reactors, Berlin, 10.-14. September 1989., Konferenzen des Forschungs-zentrums Julich(1991).

30. J. E. Grund, "Self-Limiting Excursion Tests of a Highly Enriched Plate-Type D20-Moderated Reactor, Part I. Initial Test Series", USAEC Report IDO-16891, Phillips Petroleum Co., July 12, 1963.

31. W. L. Woodruff, D. K. Warinner, and J. E. Matos, "A Radiological Consequence Analysis with HEU and LEU Fuels," Proc. 1984 International Meeting on Reduced Enrichment for Research and Test Reactors, Argonne National Laboratory, Argonne, IL, October 15-18, 1984, ANL/RERTR/TM-6, CONF-8410173, pp. 472-490 (July 1985).

32. R. B. Pond and J. E. Matos, "Nuclear Criticality Assessment of LEU and HEU Fuel Element Storage," Proc. 1983 International Meeting on Reduced Enrichment for Research and Test Reactors, Japan Atomic Energy Research Institute, Tokai, Japan, October 24-27, 1983, JAERI-M 84-073, pp.416-425 (May 1984).

33. J. T. Thomas, "Nuclear Criticality Assessment of Oak Ridge Research Reactor Fuel Element Storage," ORNL/CSD/TM-58, Oak Ridge National Laboratory (1978).

34. W. L. Woodruff, "Evaluation and Selection of Hot Channel (Peaking) Factors for Research Reactor Applications," Proc. X International Meeting on Reduced Enrichment for Research and Test Reactors, Buenos Aires, Argentina, September 28 - October 1,1987, pp.443-452.

ATTACHMENT 1

ISOTHERMAL REACTIVITY CHANGE COMPONENTS FOR AN HEU CORE WITH 17 FRESH FUEL ASSEMBLIES

The purpose of this attachment is to analyze the components of the reactor isothermal temperature coefficient for the heavy water in various regions of the reactor tank. The calculations were done for an HEU core with 17 fresh fuel assemblies. Reactivity component values for heavy water outside of the fuel assemblies are expected to be very similar in LEU cores. Reactivity coefficients for the fuel and coolant shown in Table 5 of Section 5.5 are also very similar in HEU and LEU cores.

The reactor was divided into three regions: (1) the heavy water inside the fuel assemblies, (2) the heavy water between fuel assemblies, and (3) the heavy water reflector. On the outer edges of the core, a heavy water thickness equal to one-half the water thickness between fuel assemblies was included as part of the inter-assembly water. The remaining heavy water in the tank is referred to as the reflector. Calculations were performed by separately changing the water temperature and density in each region while holding the water in the other two regions at 23°C. Least-squares fits were then done to obtain reactivity values at intermediate temperatures.

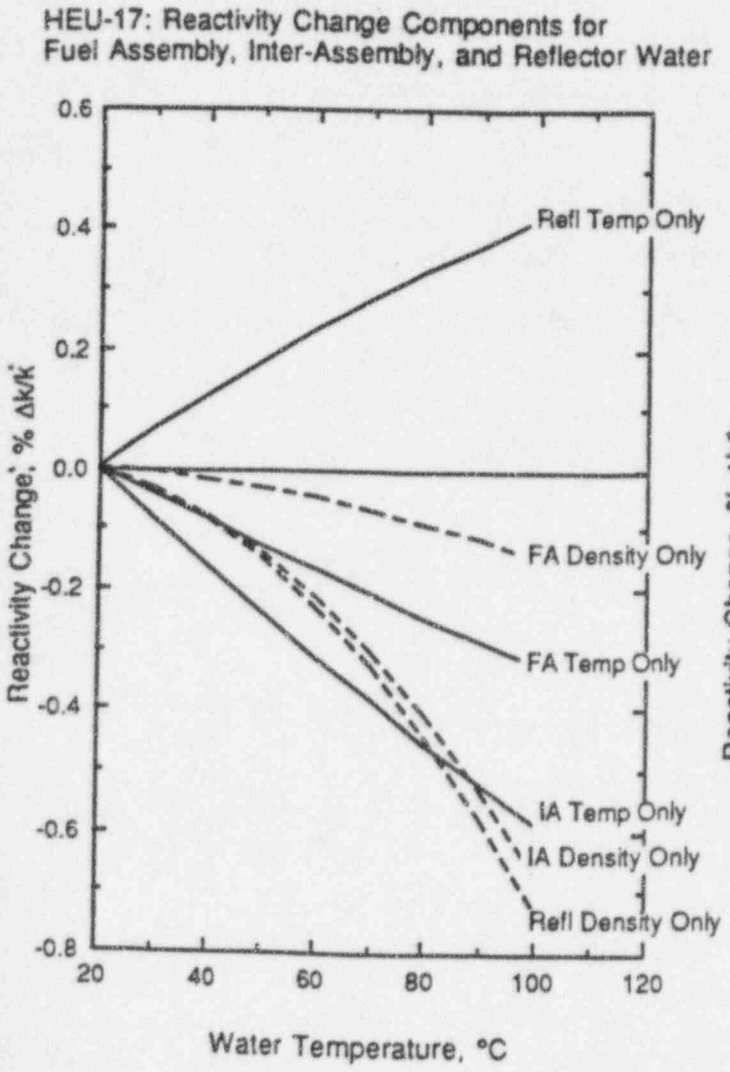
Reactivity changes relative to 20°C for water temperature and density changes in each region are shown in the attached figure. Increasing the heavy water temperature and decreasing its density in the fuel assemblies and between fuel assemblies results in negative reactivity changes for both the temperature and density components. In the reflector, the water density component is negative, but the water temperature component is positive. Combined temperature and density effects for each heavy water region show that reactivity changes with increasing water temperature are negative for the fuel assembly and inter-assembly water. In

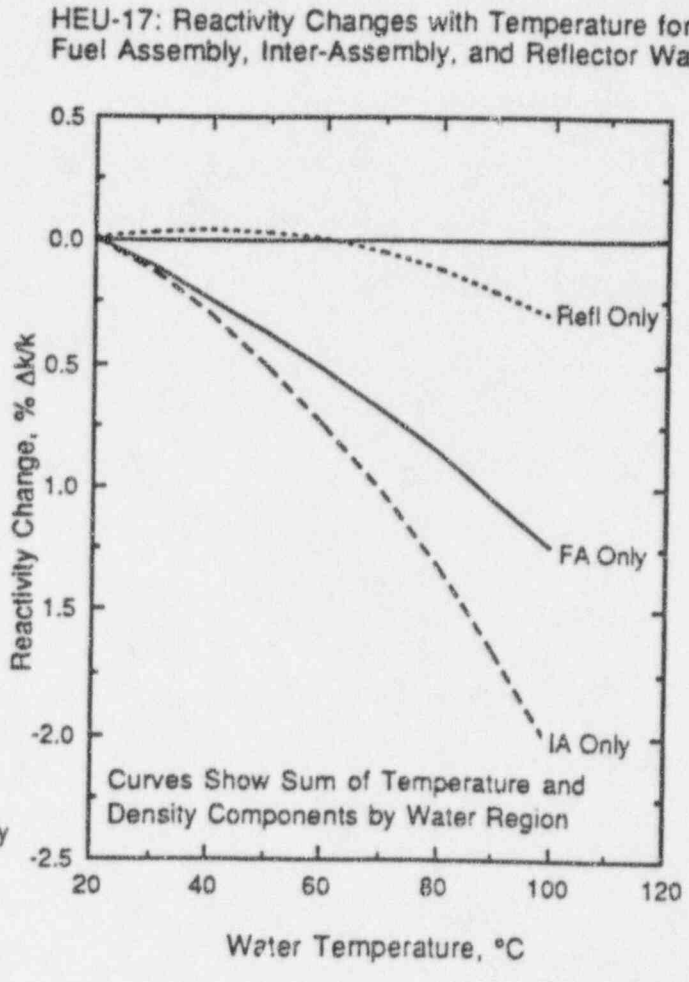
the reflector, net reactivity changes are slightly positive for heavy water temperatures up to about 60°C and then become negative with further increases in temperature.

The sum of the temperature and density components over the three heavy water regions is negative for the entire temperature range between 20°C and 100°C. A direct calculation of the isothermal temperature coefficient in which all changes were made simultaneously gave results which are in good agreement with those obtained by summing the various components.

Figure 1-1. Calculated Reactivity Changes (in % Δk/k) with Temperature for a Fresh GTRR HEU Core with 17 Fresh Fuel Assemblies

FA = Fuel Assembly Water; IA = Inter-Assembly Water; Refl = Reflector Water





ATTACHMENT 2

ENGINEERING UNCERTAINTY FACTORS

This attachment addresses the engineering uncertainty factors (or hot channel factors) that were used to compute the thermal-hydraulic safety limits, safety margins, and safety system trip settings in HEU and LEU cores with 14 fuel assemblies. The rationale for choosing these factors and the method used to combine them are outlined along with a summary of results for the HEU and LEU cores.

The PLTEMP code[22] used in the ANL analyses allows for introduction of three separate engineering hot channel factors as they apply to the uncertainty in the various parameters (as opposed to a single lumped factor). The three hot channel factors are:

F_q for uncertainties that influence the heat flux q
F_b for uncertainties in the temperature rise or enthalpy change in the coolant
F_h for uncertainties in the heat transfer coefficient h.

The code also allows introduction of nuclear peaking factors for the radial, F_r and axial, F_z, distributions of the heat flux.

While there is no generally accepted method for the selection of hot channel factors, these factors are normally a composite of sub-factors, and the sub-factors can be combined either multiplicatively, statistically [$F_b = 1 + \sqrt{\sum (1 - f_{bi})^2}$], or as a mixture of the two. A detailed description of methods for calculating hot channel factors is contained in Ref. 34. The multiplicative method of combining the

sub-factors is very conservative and somewhat unrealistic. The statistical method recognizes that all of these conditions do not occur at the same time and location.

The engineering uncertainty factors that were combined multiplicatively and used by Georgia Tech in analyses[3,23] of the HEU core are shown in Table 2-1. The factors that were combined statistically and used by ANL for calculations of the HEU and LEU cores are shown in Table 2-2.

Key thermal-hydraulic safety limits and safety margins for the HEU and LEU cores computed using the Georgia Tech factors and the ANL factors are compared in Table 2-3. Results for the HEU core obtained using ANL's statistical treatment of the engineering uncertainty factors agree well with the analyses performed by Georgia Tech. Except for the reactor power limit, data for the LEU core are comparable to or more conservative than those for the HEU core. An LEU core power limit of 10.6 MW based on the flow instability criterion is considered to be adequate.

Table 2-1. GTRR-HEU Engineering Uncertainty Factors[23]

Uncertainty	F_q	F_b	F_h
Equivalent Diameter	-	1.09	-
Fuel Distribution	1.03	1.03	-
Axial Flux Peaking	1.19	-	-
Power Level Measurement	1.03	1.03	-
Flow Distribution - Plenum	-	1.07	-
Flow Distribution - Channel	-	1.10	-
Multiplicative Combination	1.26	1.36	1.0

Table 2-2. ANL-HEU and ANL-LEU Engineering Uncertainty Factors

Uncertainty	ANL-HEU Factors			ANL-LEU Factors		
	F_q	F_b	F_h	F_q	F_b	F_h
Fuel Meat Thickness [a]	1.04	-	-	1.04	-	-
^{235}U Loading	1.03 [b]	1.03 [b]	-	1.03 [c]	1.03 [c]	-
^{235}U Homogeneity	1.03 [d]	1.03 [d]	-	1.20 [e]	1.10 [e]	-
Coolant Channel Spacing	-	1.17 [f]	1.03 [f]	-	1.22 [g]	1.04 [g]
Power Level Measurement [d]	1.03	1.03	-	1.03	1.03	-
Calculated Power Density [h]	1.10	1.10	-	1.10	1.10	-
Coolant Flow Rate [h]	-	1.10	1.08	-	1.10	1.08
Heat Transfer Coefficient [h]	-	-	1.20	-	-	1.20
Statistical Combination	1.12	1.23	1.30	1.23	1.28	1.31
Multiplicative Combination	1.26	1.55	1.33	1.41	1.72	1.35

a Derived from fuel plate thickness specification of 50 ± 2 mils.
b Assumed to be the same as for the LEU plate.
c From LEU fuel plate loading specification of 12.5 ± 0.35 g ^{235}U.
d GTRR-HEU value from Table 2-1.
e From LEU plate fuel homogeneity specification.
f Computed based on coolant channel spacing of 106 ± 10 mils and fuel plate thickness specification of 50 ± 2 mils in HEU assembly (see Ref. 34 for calculation method).
g Computed based on coolant channel spacing of 89 ± 10 mils and fuel plate thickness specification of 50 ± 2 mils in LEU assembly (see Ref. 34 for calculation method).
h Assumed values.

The ANL factors for F_q and F_b were combined statistically using the relation $F = 1 + \sqrt{\sum (1 - f_i)^2}$. The corresponding factor for F_h was obtained by statistically combining the factors for the coolant channel spacing and the coolant flow rate and multiplying the result by the factor for the heat transfer coefficient.

Table 2-3. Comparison of Key Thermal-Hydraulic Safety Parameters for HEU and LEU Cores wtn 14 Fuel Assemblies.

Reactor Power Limits for a Maximum Inlet Temperature of 123°F

Reactor Coolant Flow, gpm	GTRR-HEU	ANL-HEU	ANL-LEU
	Reactor Power Level (MW) for DNB[23,24]		
760	5.5	5.7	5.3
1625	11.5	11.9	10.8
	Reactor Power Level (MW) for Flow Instability[25,26]		
760	5.3	5.1	5.0
1625	10.6	11.0	10.6

Thermal-Hydraulic Data with Min. Coolant Flow of 1625 GPM and Max. Inlet Temp. of 123°F.

	GTRR-HEU	ANL-HEU	ANL-LEU
Coolant Velocity, m/s	2.44	2.44	2.61
Friction Pressure Drop[1], kPa	10.9	11.0	15.0
Power/Plate[2], kW	21.2	21.2	18.8
Outlet Temperature of Hottest Channel, °F	157	154	156
Peak Clad Surface Temperature, °F	219	229	224
Minimum DNBR[3]	2.29	2.37	2.17
Limiting Power Based on Min. DNBR, MW	11.5	11.9	10.8
Flow Instability Ratio (FIR)[4]	2.12	2.19	2.11
Limiting Power Based on FIR, MW	10.6	11.0	10.6

[1] Pressure drop across active fuel only.
[2] Assuming 95% of power deposited in fuel.
[3] Using modified Weatherhead Correlation[23,24] for DNB.
[4] Using Whittle-Forgan Correlation[25,26] with $\eta = 25$.

Safety Limits on Reactor Inlet and Outlet Temperatures.

Parameter	GTRR-HEU	ANL-HEU		ANL-LEU	
	DNB	DNB	Flow Inst.	DNB	Flow Inst.
Limiting Reactor Inlet Temp., °F	172	175	172	171	170
Ave. Coolant Temp. Rise across Core, °F	16	17	17	17	17
Limiting Reactor Outlet Temp., °F	188	192	189	188	187

Margins to D_2O Saturation Temperature and ONB

Parameter	GTRR-HEU	ANL-HEU	ANL-LEU
Thermal Power, MW	5.0	5.0	5.0
Reactor Coolant Flow, gpm	1800	1800	1800
Reactor Inlet Temp., °F	114	114	114
ΔT_{sub}, °F	8	5	11
Margin to ONB[1]	-	1.34	1.44
Limiting Power Based on ONB, MW	-	6.7	7.2

[1] Using the Bergles and Rohsenow correlation[27].

Power Levels and Inlet Temperatures for Zero Subcooling at a Coolant Flow of 1800 GPM

Parameter	GTRR-HEU	ANL-HEU	ANL-LEU
Thermal Power, MW	5.45	5.35	5.6
Reactor Inlet Temp., °F	114	114	114
Thermal Power, MW	5.0	5.0	5.0
Reactor Inlet Temp., °F	123	122	128

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

The organization for the management and operation of the reactor is indicated in Figure 6.1. The Director of the Nuclear Research Center has over all responsibility for direction and operation of the reactor facility, including safeguarding the general public and facility personnel from radiation exposure and adhering to all requirements of the operating license and Technical Specifications.

The Manager, Office of Radiation Safety, advises the Director, Nuclear Research Center in matters pertaining to radiological safety. She/he has access to the Vice President, Interdisciplinary Programs and/or the President of the Institute as needed.

The minimum qualifications with regard to education and experience backgrounds of key supervisory personnel in the Reactor Operations group are as follows:

(1) Reactor Supervisor

The Reactor Supervisor must have a college degree or equivalent in specialized training and applicable experience, and at least five years experience in a responsible position in reactor operations or related fields including at least one year experience in reactor facility management or supervision. He must hold a Senior Reactor Operator's license for the GTRR.

(2) Reactor Engineer

The Reactor Engineer must have a combined total of at least seven years of college level education and/or nuclear reactor experience with at least three years experience in reactor operations or related fields. He shall be qualified to hold a Senior Reactor Operator's license.

Whenever the reactor is not secured, the minimum crew complement at the facility is two persons, including at least one senior operator licensed pursuant to 10 CFR 55.

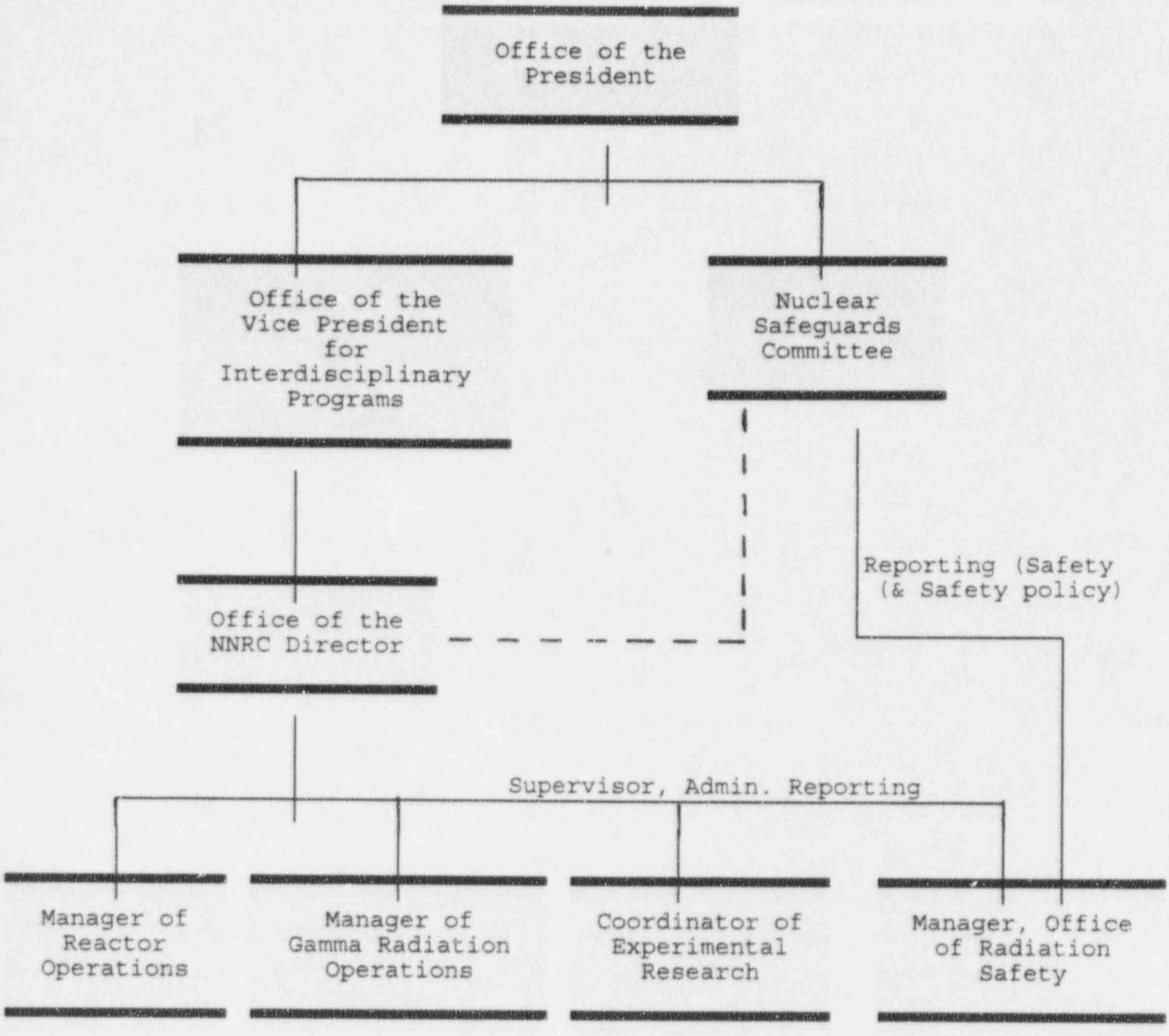


Figure-6.1 Georgia Tech Organization for Management and Operation of GTRR.

An operator or senior operator licensed pursuant to 10 CFR 55 must be present at the controls unless the reactor is shutdown as defined in The Technical Specifications.

6.2 NUCLEAR SAFEGUARDS COMMITTEE

A Nuclear Safeguards Committee established by the President of the Institute is responsible for maintaining health and safety standards associated with operation of the reactor and its associated facilities.

The Committee is composed of five or more senior technical personnel who collectively provide experience in reactor engineering, reactor operations, chemistry and radiochemistry, instrumentation and control systems, radiological safety, radiation protection, and mechanical and electrical systems. A minority of the Committee members are selected from the GTRR staff.

The Committee meets quarterly and as circumstances warrant. Written records of the proceedings, including any recommendations or occurrences, are distributed to all Committee members and the President's Office.

A quorum consist of not less than a majority of the Committee membership which includes the chairman or his designated alternate. The operating staff may not constitute a majority of those present.

The Committee:

(1) Reviews and approves determinations that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question pursuant to 10 CFR 50.59 (a).

(2) Reviews reportable occurrences.

(3) Reviews and approves proposed operating procedures and proposed changes to operating procedures. Minor modifications to operating procedures which do not change the original intent of the operating procedure may be approved by the Director of NNRC on a temporary basis. The Committee will consider such minor modifications at the next scheduled meeting.

(4) Reviews and approves proposed changes to Technical Specifications and license excluding organizational structure. The responsibility and authority for organizational structure resides with the President of the Institute.

(5) Reviews and approves proposed experiments and tests utilizing the reactor facility which are significantly different from tests and experiments previously performed at the GTRR.

(6) Reviews and approves proposed changes to the facility made pursuant to 10 CFR 50.59(c).

(7) Reviews violations of Technical Specifications, license, or internal procedures or instructions having safety significance.

(8) Reviews operating abnormalities having safety significance.

(9) Reviews audit reports.

(10) Audits reactor operations and reactor operation records for compliance with internal rules, procedures, and regulations and with licensed provisions, including Technical Specifications at least once per calendar year (interval between audits must not exceed 15 months).

(11) Audits the retraining and requalification program for the operating staff, at least once every other calendar year (interval between audits must not exceed 30 months).

(12) Audits the results of action taken to correct those deficiencies that may occur in the reactor facility equipments systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits must not exceed 15 months).

6.3 ADMINISTRATIVE CONTROLS OF EXPERIMENTS

6.3.1 Evaluation by Safety Review Group

No experiment is performed without review and approval by the Nuclear Safeguards Committee. Repetitive experiments with common safety

considerations may be reviewed and approved as a class. Criteria for review of an experiment or class of experiments include (a) applicable regulatory positions including those in 10 CFR Part 20 and the technical specifications and (b) in-house safety criteria and rules which have been established for facility operations, including those which govern requirements for encapsulation, venting, filtration, shielding, and similar experiment design considerations. as well as those which govern the quality assurance program required under 50.34. Records are kept of the Nuclear Safeguards Committee's review and authorization for each experiment or class of experiments.

6.3.2 Operations Approval

Every experiment must have the prior explicit written approval of the Licensed Senior Operator in charge of reactor operations. Every person who is to carry out an experiment is certified by the Licensed Senior Operator in charge of reactor operations as to the sufficiency of his knowledge and training in procedures required for the safe conduct of the experiment.

6.3.3 Procedures for Active Conduct of Experiments

Detailed written procedures are provided for the operation of each experimental facility. The Licensed Operator at the console must be notified just prior to moving any experiment within the reactor area and must authorize such movement. Each experiment removed from the reactor or reactor system is subject to a radiation monitoring procedure which anticipates exposure rates greater than those predicted. The results of such monitoring is documented.

6.3.4 Procedures Relating to Personnel Access to Experiments

There must a documented procedure for the control of visitor access to the reactor area to minimize the likelihood of unnecessary exposure to radiation

as a result of experimental activities and to minimize the possibility of intentional or unintentional obstruction of safety. There must be written training procedure for the purpose of qualifying experimenters in the reactor and safety related aspects or their activities, including their expected responses to alarms.

6.3.5 Quality Assurance Program

There is a quality Assurance Program covering the design, fabrication, and testing of experiments, including procedures for verification of kinds and amounts of their material contents to assure compliance with the technical specifications.

6.4 Administrative Procedures

Procedures and major changes thereto are reviewed and approved by the Nuclear Safeguards Committee prior to being effective. Changes which do not alter the original intent of a procedure may be approved by the director of the facility. Such changes are recorded and submitted periodically to the Nuclear Safeguards Committee for routine review.

Written procedures are provided and utilized for the following:

(1) Normal startup, operation and shutdown of the reactor and of all systems and components involving nuclear safety of the system.

(2) Installation and removal of fuel elements, control blades, experiments and experimental facilities.

(3) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

(4) Emergency conditions involving potential or actual release of radioactivity.

(5) Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.

(6) Radiation and radioactive contamination control.

(7) Surveillance and testing requirements.

(8) A site emergency plan delineating the action to be taken in the event of emergency conditions and accidents which result in or could lead to the release of radioactive materials in quantities that could endanger the health and safety of employees or the public. Periodic evacuation drills for facility personnel shall be conducted to assure that facility personnel are familiar with the emergency plan.

(9) Physical security of the facility and associated special nuclear material.

6.5 Operating Records

The following records and logs are prepared and retained at the facility for least five years:

(1) Normal facility operation and maintenance.

(2) Reportable occurrences.

(3) Tests, checks, and measurements documenting compliance with surveillance requirements.

(4) Records of experiments performed.

The following records and logs are prepared and retained at the facility for the life of the facility:

(1) Gaseous and liquid waste released to the environs.

(2) Offsite environmental monitoring surveys.

(3) Radiation exposures for all GTRR personnel.

(4) Fuel inventories and transfers.

(5) Facility radiation and contamination surveys.

(6) Updated, corrected, and as-built facility drawings.

(7) Minutes of nuclear Safeguards Committee meetings.

(8) Records of radioactive shipments.

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

In the event of a reportable occurrence, as defined in the Technical Specifications, the following action is taken:

1. Reactor conditions are returned to normal or the reactor shall be shutdown. If it is necessary to shut the reactor down to correct the occurrence, operations are not to be resumed unless authorized by the director of the facility.
2. All reportable occurrences must be promptly reported to the reactor supervisor and the director of the facility.
3. All reportable occurrences must be reported to the Nuclear Regulatory Commission in accordance with Section 6.7.
4. All reportable occurrences are to be reviewed by the Nuclear Safeguards Committee.

6.7 Reporting Requirements

The following information is to be submitted to the U.S.N.R.C. in addition to the reports required by Title 10, Code of Federal Regulations.

6.7.1 Annual Operating Reports

A report covering the previous year is submitted to the office of the Regional Administrator, Region II, with a copy to the Director, Office of Nuclear Reactor Regulation, by March 1 of each year. It includes the following:

(l) Operations Summary

A summary of operating experience occurring during the reporting period including:

(a) changes in facility design,

(b) performance characteristics (e.g., equipment and fuel performance),

(c) changes in operating procedures which relate to the safety of facility operations,

(d) results of surveillance tests and inspections required by these technical specifications,

(e) a brief summary of these changes, tests, and experiments which required authorization from the Commission pursuant to 10 CFR 50.59(a), and

(f) changes in the plant operating staff serving in the following positions:

1. Director, Nuclear Research Center
2. Reactor Supervisor
3. Reactor Engineer
4. Manager, Office of Radiation Safety
5. Nuclear Safeguards Committee members

(2) Power Generation

A tabulation of the thermal output of the facility during the reporting period.

(3) Shutdowns

A listing of unscheduled shutdowns which have occurred during the reporting period, tabulated according to cause, and a brief discussion of the preventive actions taken to prevent recurrence.

(4) Maintenance

A discussion of corrective maintenance (excluding preventative maintenance) performed during the reporting period on safety related systems and components.

(5) Changes, Tests and Experiments

A brief description and a summary of the safety evaluation for those changes, tests, and experiments which were carried out without prior Commission approval, pursuant to the requirements of 10 CFR Part 50.59(b).

(6) Radioactive Effluent Releases

A statement of the quantities of radioactive effluents released from the plant, with data summarized following the general format of USNRC Regulatory Guide 1.21:

(a) Gaseous Effluents

1. Gross Radioactivity Releases

a. Total gross radioactivity (in Curies), primarily noble and activation gases.

b. Average concentration of gaseous effluents released during normal steady state operation. (Averaged over the period of reactor operation.)

c. Maximum instantaneous concentration of gaseous radionuclides released during special operations, tests, or experiments, such as beam tube experiments, or pneumatic tube operation.

d. Percent of technical specification limit.

2. Iodine Releases

(Required if iodine is identified in primary coolant samples, isotopic analysis required in (a)l. above or if fuel experiments are conducted at the facility.)

a. Total iodine radioactivity (in Curies) by nuclide released, based on representative isotopic analyses performed.

b. Percent of technical specification limit.

3. Particulate Releases

a. Total gross radioactivity (β, γ) released (in Curies) excluding background radioactivity.

b. Gross alpha radioactivity released (in Curies) excluding background radioactivity. (Required if the operational or experimental program could result in the release of alpha emitters.)

c. Total gross radioactivity (in Curies) of nuclides with half-lives greater than eight days.

d. Percent of efficient concentration for particulate radioactivity with half-lives greater than eight days.

(b) Liquid Effluents

1. Total gross radioactivity (β, γ) released (in Curies) excluding tritium and average concentration released to the unrestricted area or sanitary sewer (averaged over period of release).

2. The maximum concentration of gross radioactivity (β, γ) released to the unrestricted area.

3. Total alpha radioactivity (in Curies) released and average concentration released to the unrestricted area (averaged over the period of release).

4. Total volume (in ml) of liquid waste released.

5. Total volume (in ml) of water used to dilute the liquid waste during the period of release prior to release from the restricted area.

6. Total radioactivity (in Curies), and concentration

(averaged over the period of release) by nuclide released, based on representative isotopic analyses performed for any release which exceed 1×10^{-7} μCi/ml.

7. Percent of technical specification limit for total radioactivity from the site.

(7) Environmental Monitoring

For each medium sampled, e.g., air, surface water,soil, fish, vegetation, include:

(a) Number of sampling locations and a description of their location relative to the reactor.

(b) Total number of samples.

(c) Number of locations at which levels are found to be significantly above local backgrounds.

(d) Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and the location of that point with respect to the site.

(e) The maximum cumulative radiation dose which could have been received by an individual continuously present in an unrestricted area during reactor operation from:

1. direct radiation and gaseous effluent, and
2. liquid effluent.

When levels of radioactive materials in environmental media, as determined by an environmental monitoring program, indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, 10 CFR Part 20, estimates of

the likely resultant exposure to individuals and to population groups and assumptions upon which estimates are based will be provided.

(8) Occupational Personnel Radiation Exposure

A summary of radiation exposures greater than 500 mrem (50 mrem for persons under 18 years of age) received during the reporting period by facility personnel (faculty, students, or experiments).

6.7.2 Non-Routine Reports

(1) Reportable Occurrence Reports

Notification will be made within 24 hours by telephone and telegraph to the Office of the Regional Administrator, Region II, with a copy to the Director, office of Nuclear Reactor Regulations followed by a written report within 10 days to the Office of the Regional Administrator, Region II, with a copy to the Director, office of Nuclear Reactor Regulations in the event of the reportable occurrences as defined Technical Specifications. The written report on these reportable occurrences, and to the extent possible, the preliminary telephone and telegraph notification must:

(a) describe, analyze, and evaluate safety implications,

(b) outline the measures taken to assure that the cause of the condition is determined,

(c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems, and

(d) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

(2) Unusual Events

A written report must be forwarded within 30 days to the Office of the Regional Administrator, Region II, with a copy to the Director, Office of Nuclear Reactor Regulations in the event of:

(a) Discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the Technical Specifications.

(b) Discovery of any substantial variance from performance specifications contained in the Technical Specifications or in the Safety Analysis Report.

(c) Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

7.0 WASTE DISPOSAL AND FACILITIES TESTING

7.1 Waste Disposal and Control

7.1.1 General Policy Regarding Waste Disposal

The collecting, packaging, storing, shipping and release to the environment of radioactive wastes generated within the facilities of the Nuclear Research Center will be conducted by the Health Physics group. It is the responsibility of the Manager, Office of Radiation Safety, to insure that waste handling and disposal activities do not endanger the health of personnel at Georgia Tech or the general public. Such activities are closely monitored at all times by the Health Physics group. They must confirm the safety of the operation or suspend it until necessary corrections are made.

The release of radioactive waste to the environment will be in conformity with the Technical Specifications for the Nuclear Research Center and within the limits of 10 CFR part 20. The policy in effect at the Center is to reduce the release of radioactive materials to the minimum consistent with the efficient operation of the facilities and, at all times, to comply with existing governmental regulations.

It is the responsibility of every individual who uses the facilities of the Nuclear Research Center to insure that the radioactive waste from operations under his direction is properly segregated, contained and labeled. It will be his additional responsibility to keep the volume of waste generated to the absolute minimum by use of proper techniques, choice of materials, and any other means available. All personnel using the research facilities receive instructions in radioactive waste management from the Health Physics staff. The formal training program (Section 6) given by the Health Physics staff is mandatory for all employees who use the facilities.

7.1.2 Liquid Waste

The sanitary waste system collects liquids only from operations wherein no entry of radioactive materials could be reasonably anticipated, such as toilets, lavatories, water fountains, kitchenette, floor drains in

equipment areas, animal quarters, shop areas, and other areas of a similar nature. All sanitary wastes enter conventional cast iron drain lines which lead into the common 6-inch line emerging from the laboratory building. All suspect wastes from the containment building are directed into the liquid waste handling facility.

A manhole in the outfall of the waste handling facility is located outside of the fence surrounding the reactor and laboratory building. This is provided to facilitate sampling of the effluent from the waste handling facility immediately prior to its discharge into the sanitary line serving the laboratory building, An isolating valve which permits the outflow to be closed off from the sanitary line is provided at this point.

All liquid waste other than that described above as part of the sanitary waste system is considered to be in one of the following three categories:

a. Suspect Waste--includes liquid wastes which are expected to contain little or no radioactivity and, therefore, may be discharged to the city sewerage system after monitoring.
b. Low Level Waste--those wastes which contain quantities of radioactivity which might be undesirable to release to the city sewerage system and, therefore, require further analysis,
c. High Level Waste--small wastes containing quantities of radioactivity which are too large to permit discharge directly to the city sewerage system.

All wastes which are defined as suspect or low level are passed directly to the waste retention tanks. Liquid in these tanks must be analyzed and, if necessary, treated to remove excess radioactivity before being released to the city sewer. The approval of Health Physics is required for each release. The previously mentioned system of demineralizers and filters is available to treat low level waste also.

All high level wastes will be collected in appropriate containers in the laboratory as they are generated. Whenever necessary, these containers will be picked up by personnel of the laboratory operations group and stored in a designated control area. If significant reduction in activity cannot be achieved by radioactive decay, the liquid will be solidified and handled as a solid waste,

All liquid suspect wastes generated within the containment building will be collected in a sump located in the basement of the reactor building. These liquids will be pumped from the sump into the suspect waste system.

7.1.3 Solid Waste

Most solid wastes such as laboratory glassware, blotting paper, small pieces of reactor equipment, etc, will be quite low in radioactivity. These wastes will be collected and stored in a segregated and properly labelled area. At appropriate times these wastes will be packaged according to DOT specifications and shipped to an authorized receiver of radioactive wastes. Solid wastes containing a high level of radioactivity will be stored as may be necessary to prevent excessive exposure of personnel. In most cases the material will become low level waste if the radioactivity is allowed to decay a sufficient time. Some highly radioactive materials such as spent fuel elements will require special handling and containers. Such operations will be planned in accordance with NRC and DOT regulations.

7.2 Scheduled Facilities Testing

7.2.1 Testing of Emergency Cooling System

Verification that emergency coolant water is actually being supplied to each fuel element is vital to the safe operation of the reactor. This verification is obtained in two different ways.

At intervals of approximately 6 months, a visual inspection is made beneath the lower top shield using a borescope. This instrument permits actual observation of emergency coolant water entering all fuel elements (or fuel element mockups). The second method utilizes the thermocouples installed in each fuel assembly to monitor the exit cooling water. Following a reactor shutdown and after allowing sufficient time for decay heat removal, the top reflector will be drained to the storage tank. This will expose the fuel element thermocouples to the helium gas blanket. If the bulk D_2O is allowed to remain at 90-100°F, then the vessel and the helium gas will be at approximately this same temperature. Emergency cooling will then be initiated and the fuel element temperatures monitored. Because the

emergency coolant water will be ~ 75°F, there will be a sharp temperature decrease on all functioning thermocouples when emergency coolant water reaches the distribution plate just above the fuel in each element. This test is run at monthly intervals.

7.2.2 Containment Building Testing

The reactor containment building is pressure tested periodically to verify that leakage from the building is within acceptable limits. The procedure used to conduct the tests is the reference tank method as described in "Leakage Rate Testing of Containment Structures for Nuclear Reactors", proposed standard ANS 7. 60 published by the American Nuclear Society Standards Committee. Four interconnected gas-tight tanks are located throughout the building to which the pressure in the containment vessel is compared. Each tank is sized and positioned so that the ratio of the tank volume to the building volume in which that tank is immersed is approximately the same for all tanks. The reference tank system is pressurized and observed for leaks for a 24-hour period prior to and subsequent to the building test.

A table summarizing previous test results on the GTRR containment building is shown below.

Summary of Containment Building Test Results

Date	Leakage from Containment Vessel/24 period @2psig	Test Conducted By
2-11,12-63	0.6% of building volume	Chicago Bridge and Iron [7.2]
2-12,13-63	0.4% of building volume	Chicago Bridge and Iron
12-20~21-64	0.7% of building volume	GTRR personnel [7.3]
12-21,22-64	0.5% of building volume	GTRR personnel
2-12,13-66	0.4% of building volume	GTRR personnel [7.4]
2-17,18-67	0.4% of building volume	GTRR personnel [7.5]
3-15,91	0.70% of building volume	GTRR Staff
4-24,92	0.63% of building volume	GTRR Staff
3-24,93	0.63% of building volume	GTRR Staff

These rates are well within the general NRC specification of leakage for containment buildings of 1/2 percent of the building air volume in 24 hours per pound of over-pressure.

Prior to the building pressure test, the vacuum relief devices are checked by first pressurizing them independently and determining the leak rate, and then checking the operation of the values by application of a vacuum.

REFERENCES

7.1 Fox, J. K., and Gilley. L. W., "Critical Experiments with Arrays of ORR and BSF Fuel Elements, Report No. ORNL CF-58-9-40.

7.2 C. B. & I, et. al., Final Pneumatic Test and Leakage Rate Determination, Containment Vessel, Georgia Tech Reactor Project," Chicago Bridge and Iron Company, Chicago, Illinois, Final acceptance test report, 18 pp. (April, 1963).

7.3 Roberts, C. J., et. al., "GTRR Containment Building Leakage Rate Measuremen December, 1964," Nuclear Research Center Engineering Experiment Station, Georgia Institute of Technology, Atlanta, Georgia, Memorandum, 7 pp. (January 15, 1965).

7.4 Apple, F. C., et. al., "GTRR Containment Building Leakage Rate Measurement--February 1966, "Nuclear Research Center Engineering Experiment Station, Georgia Institute of Technology, Atlanta, Georgia, Memorandum, 4 pp. (April 25, 1966).

7.5 Apple, F. C., et. al., "GTRR Containment Building Leakage Rate Measurement--February 1967, "Nuclear Research Center Engineering Experiment Station, Georgia Institute of Technology, Atlanta, Georgia, Memorandum, 3 pp. (March 28, 1967).

8. REACTOR HAZARDS EVALUATION

8.1 General Safety Considerations

Heavy-water moderated and cooled reactors are among the safest which can be constructed. This safety is predicated, basically, upon the demonstrated ability of the D_20-moderated reactors to absorb reactivity additions internally by moderator changes. This ability is manifest in the negative void and temperature coefficients characteristic of these types. A heavy water reactor exhibits much of the same power-limiting ability as the light water moderated and cooled reactors. Moreover, the low neutron absorption, long neutron lifetime and consequent sensitivity to neutron leakage inherent in heavy water moderated machines enhances these characteristics.

The Borax and Spert programs have repeatedly demonstrated the shutdown mechanisms by which water reactors protect themselves. Although these mechanisms are not always quantitatively understood, certain conclusions may be drawn with regard to their actions. For example, it is possible to predict the performance of a boiling reactor by the extrapolation of experimental data obtained from the Borax I and II experiments. This method, used successfully in connection with Borax III and IV, EBWR, VBWR, and others, is considered to be a valid approach. Since all water reactors possess the ability to function as boilers, these same methods can be used to analyze the behavior of normally non-boiling reactors when subjected to unusual operating conditions or power levels. Consequently, this approach has been used to investigate the effect of large reactivity additions to the GTRR using the data of the boiling experiments.

Although the ultimate inherent safety of the GTRR can be adequately shown, recourse to this capability should be limited to those situations where other control provisions have failed. Consequently, an extensive system of interlocks is provided to protect the reactor against equipment and instrumentation malfunctions and operator errors. The safety interlocks include short period trips, over-power trips, high coolant temperature trips, and low coolant flow and temperature trips. All equipment is designed to fail safe in the event of improper operation or loss of power.

8.2 Operating Hazards

A nuclear reactor is comprised of interrelated electrical, mechanical, hydraulic and pneumatic systems. As a result of this interrelationship, the malfunction of any of these systems can affect the safety of the reactor. The following sections describe possible failures of these systems, their possible consequences, and the protection provided against their occurrence in the GTRR design.

8.2.1 Power Failures

The loss of the electrical power supply results in the disruption of all the reactor systems. The electrical circuits are designed for fail-safe operation in the event of a power loss. The systems so affected, and their operation, are described in the succeeding paragraphs.

The operation of any reactor trip circuit will result in the immediate insertion of the shim-safety blades. Any two of the four blades control sufficient reactivity to shut down the reactor independently. The shim-safety blade drive shafts are connected to the drive motors through electromagnetic clutches supplied in pairs from two power sources. Any interruption of the power supply will result in disengagement of the clutches and the insertion of all four shim-safety blades. The regulating element, which is not designed for scram duty, will become inoperable and will remain fixed at the position occupied at the moment of power loss. The drain valves of the moderator drain system are of the normally open type; therefore, loss of electrical power will result in the opening of these valves causing the top D_2O reflector to be drained to the storage tank.

If electrical power fails, the containment interlocks effect the complete closure of the building. This requires that the stop valves, located in the ventilating inlet and outlet air ducts be closed. Since these are normally closed solenoid valves, loss of power provides automatic closure. Stoppage of the supply fan at the base of the stack for any reason, including a power failure, will cause a damper to close the outside air inlet to the stack.

A power disruption will also cause the pumps to stop and coolant flow to cease. Both the fuel and the thermal shield are capable of dissipating the decay heat

satisfactorily following shutdown without benefit of coolant flow. A detailed treatment of this situation is contained in Section 8.2.2.

A power failure will cause automatic initiation of emergency cooling since the parallel coolant stop valves are normally-open, pneumatic, solenoid-controlled valves. After a power failure, the reactor operator will determine if a requirement for emergency cooling actually exists. A locked open manual valve is provided which can be manually closed by the operator to conserve the emergency cooling D_2O supply in the event a loss of coolant has not occurred simultaneous with the power failure. Emergency cooling is discussed in detail in Section 8.3.

8.2.2 Pump Failures

The loss of the primary D_2O pump or the secondary cooling water pump can result in undesirable reactor operating conditions. These systems are therefore provided with high temperature and low flow interlocks with the reactor scram circuitry. Of the two pump failures, the loss of the D_2O pump is the more serious. Two independent low D_2O flow scram interlocks, and loss of electrical power interlocks have been provided in the reactor safety instrumentation. It is therefore reasonable to assume that the reactor will scram because of low flow shortly after an electrical power failure or the more serious case of pump shaft seizure. There will be a short period of coast-down flow and then a no-D_2O flow condition will exist.

The heat transfer mechanisms that occur are very difficult to evaluate theoretically. Some boiling may initially take place in the fuel coolant channels. The ability of water to flow into the element against escaping steam vapor cannot be estimated well. The lattice spacing of the GTRR is, however, quite large. Experimental evidence indicates that external cooling of the fuel by the large mass of water surrounding each element would be considerable. At ORR, tests were run in which the coolant pumps were shut off and the low flow signal allowed to scram the reactor. At power levels up to 15 MW in the closely spaced ORR elements, there was no observed boiling. Based on the much more severe ORR case, it is reasonable to expect that complete loss of D_2O flow after shutdown of the GTRR at 5 MW does not constitute a hazardous condition.

8.2.3 Instrument Failure

The redesign of the instrumentation and reactor safety circuits to a redundant system with backup provides a maximum of operational safety. Because of the interdependence of parameters measured by the 10 "reactor safety" circuits, the very improbable simultaneous failure of four or more circuit functions would be required to block automatic scram action.

For example, ten different paths leading to safety blade insertion may be identified for a reactor overpower condition. Two specific instrument failures in each of the two independent reactor power trip circuits must be postulated to prevent an automatic scram by the power trip circuit; the redundant D_2O temperature or period circuits would still detect the overpower condition and provide the desired scram. Assuming the virtually impossible simultaneous failure of eight independent instruments noted above, the two independent manual scram circuits and the self-limiting negative reactivity with increased temperature and void introduction at the onset of boiling, provide additional maximum accident limits.

This same redundancy and overlapping function concept applies to the other reactor safety instrumentation. One must in each case postulate multiple instrument failures to block scram action.

The reactor safety circuitry is designed to fail-safe where possible. Failure of the flux (power) amplifiers to complete an internal electronic self check produces a scram signal. All scram initiating instruments are independently fail-safe on loss of power. The use of duplicate channels and overlapping of ranges of the nuclear instrumentation also provides a means of detecting instrument malfunctions.

Failure of certain process instrumentation could conceivably lead to an "economic" accident, but in no case would permit an undetected personnel safety hazard. The five system parameters designated "reactor safety" parameters were carefully chosen to detect malfunctions resulting from faulty process instrumentation to preclude hazardous situations following process instrument failure. For example, failure of the H_2O low flow or H_2O high temperature instruments, coupled with an abnormal condition in the H_2O system, would be

detected by the redundant D_2O temperature circuits. Low H_2O temperature conditions, if undetected, could result in freezing and rupture of H_2O piping. Such an accident would require repair and inconvenient reactor downtime, but would not constitute a hazard.

Failure of the radiation monitoring equipment (Section 4.5.1) could hardly go undetected because of the variety of independent instrumentation in operation. A gas release within the containment building, if undetected by the unlikely failure of the ten permanently mounted monitors, would be quickly picked up by one or more of the three building exhaust gas monitors.

8.2.4 Safety Element Failure

Since any two of the four shim-safety blades will shut down the reactor with the maximum excess reactivity present, a hazardous situation due to failure to scram is extremely unlikely. The fail-safe design of the drive system and the regular tests of scram performance enhance reliability of the system. However, should some unforeseen event necessitate it, shutdown could be independently accomplished by operation of the moderator drain system to remove the top D_2O reflector, which has a total worth of 2.75%.

Should the mechanism of safety element failure be in the drive system in such a manner that the blade being withdrawn at its maximum rate cannot be stopped, the period or overpower trips, or both, should scram the reactor. The operator can, of course, stop the blade motion by turning off the power to the rod drives.

8.2.5 Automatic Regulating Element Failure

The regulating or fine control element is of low reactivity worth (~0.4%) and operates at relatively slow speeds (0.2 in/sec). The maximum rate of reactivity addition obtainable with this rod is only .01%/sec. Consequently, the addition of the total 0.4% will produce an asymptotic period of ~ 5 seconds after about 1 minute. Because of the separation of the controller and the power set box, malfunction of the controller does not affect the automatic switching of the system to the manual control mode.

Failure of the automatic regulating system such that withdrawal of this rod cannot be stopped will result in operation of either the period trip or over-power trip scram circuits. Failure of these circuits will cause a reactor scram by the D_2O high temperature interlock. Failure of all scrams would result in an increase in reactor power until the reactivity is absorbed by the void or temperature coefficients as described in Section 8.4. The reactor will then continue stable operation at this new power level until shut down by some external means. In no case does a hazard exist.

8.3 Loss of Coolant, The Maximum Credible Accident

Two types of loss of coolant accidents can be envisioned. The first is the complete loss of all the moderator/coolant from the vessel. This represents only a problem in decay heat removal, since the loss of enough moderator to uncover the fuel will cause a loss of reactivity sufficient to shut down the reactor. The second type is caused by failure of the primary D_2O coolant pumps. This case has been previously treated in Section 8.2.2.

Complete loss of coolant is not credible; however as a consequence of a reactor vessel rupture, a re-entrant nozzle rupture, or a break in the coolant piping an accident may be postulated. A rupture of either the reactor vessel or a re-entrant nozzle will permit the moderator to flow into the graphite reflector region. Since the graphite is contained in a tank and the beam hole ports are sealed, a slow loss of moderator will result. A break in the coolant piping, however, could result in the complete and rapid loss of the coolant.

In the event of such a complete coolant loss, the shutdown reactor would require emergency cooling in order to remove fission product decay heat. The GTRR emergency cooling system, described in Section 4.4.8.3, would commence fuel element cooling automatically from a D_2O gravity head storage tank. During the 30 minutes that cooling is supplied from this source, an additional long-term supply would be placed on-line as detailed in Section 4.4.8 3.

It is not credible that the process of connecting additional long term water supplies might somehow be prevented or interrupted. The storage pool water and

city water are two independent sources of cooling water to remove decay heat. Never the less the dose rates outside the containment vessel following a complete core meltdown and attendant fission product release are presented in Appendices B and C. These analyses show that such an accident results in a thyroid dose of 300 rem in two hours at a distance of 164 feet under least favorable meteorological conditions and an external dose of 25 rem in two hours at 70 feet.

8.4 Effect of Assumed Reactivity Additions

The Borax and Spert reactors have repeatedly demonstrated that water cooled, water moderated reactors of suitable design may have a very substantial self-protection against the effects of reactivity accidents, even in the absence of corrective action by the reactor control system. This self-protection is provided by the negative steam-void coefficient of reactivity and the negative temperature coefficient of reactivity, both of which can result in important reactivity reductions as the reactor power rises. The GTRR has been designed with a high degree of self protection of this type. In this section, estimates are made of the behavior of the reactor under hypothetical conditions of rapid reactivity addition.

8.4.1 Step Reactivity Addition

In evaluating the capability of the reactor to protect itself by the negative reactivity feedback mechanism, the major concern is that the maximum temperature reached by the fuel plates during a transient not exceed the melting point. The maximum positive step reactivity insertion is associated with the sudden insertion of 1.5% Δ k/k with the core critical. An accident of this type is highly improbable since the fuel loading procedure requires all shim safety control elements to be fully inserted. Thus, such an accident could only result from a clear violation of a well-established procedure. For further details see Section 9.2 of Chapter 5.

8.4.2 Fuel Loading Accidents

During refueling operations, all control elements are required to be fully inserted and the top D_2O reflector drained to storage. Following the refueling

operation, the reactor startup will be accomplished in accordance with standard practice. Under these conditions, a sudden introduction of reactivity is impossible.

8.4.3 Moderator Changes

In almost all cases, changes to the moderator resulting from reactor operations will be in the direction of lowered density or operating level. Both these conditions decrease the reactivity. The exceptions to this are the inadvertent admission of abnormally cold coolant to the reactor and the filling of an experimental thimble with D_2O.

The negative temperature coefficient of reactivity exhibited by the GTRR results in an increase of reactivity as the water temperature decreases. This situation gives rise to the so-called "cold coolant slug" accident. Such an accident is prevented by normal startup procedures which require that the primary and secondary D_2O and H_2O coolant pumps be placed in operation and the proper flow rates and temperature conditions obtained before the control elements are withdrawn. A low temperature interlock produces an alarm if the low coolant temperature limit of 50° F is reached.

If it is assumed that the system interlocks have been bypassed or are inoperative, the following improper procedure could result in the maximum cold coolant injection accident:

a. The reactor has been down for a time period sufficient for all coolant to reach ambient conditions.
b. Outside air temperature is 5° C or less.
c. The reactor is taken from the secured condition to the 5 MW level and brought up to a temperature of 100° C with all pumps inoperative.
d. Both the D_2O and H_2O coolant pumps are started at full flow when the reactor operating temperature has reached 100°C.

The primary D_2O coolant pumps can supply 1800 gpm to the reactor vessel which will produce a 7.8 ft/sec coolant velocity through the fuel.

If one arbitrarily assumes that natural convection will produce circulation of both secondary and primary coolant through the heat exchanger and all of the D_2O

in the external circuit (9750 lb) approaches a temperature of 5°C a "worst case" situation results. These assumptions are, however, extremely pessimistic. At full pump flow, 9750 lb is about a 36 second supply of coolant to the core. Since the coolant volume is 14.7% of the total moderator volume and the temperature coefficient of the reactor is 0.015% per °C, a 95°C drop in coolant temperature represents about a 0.21% reactivity addition. If this 0.21% were added instantaneously, the reactor power would rise on a period greater than 10 seconds and a scram would be initiated by the over-power trip. If this circuit fails, the reactor will rise in power and local boiling will begin. Since the 0.21% can be accounted for by about 2.8% voids in the coolant, local boiling will compensate for the excess reactivity.

After about 36 seconds, the supply of cold water will be exhausted and warmer water will be available at the reactor inlet. Boiling will cease as the temperature coefficient takes effect and the power will gradually decrease. If one very pessimistically assumes that the inlet water is not heated as it flows through the fuel elements, the total reactivity that could be added before the supply of 5°C water is exhausted is approximately 1.3%. It was demonstrated above that an addition of more than this magnitude will not cause fuel melting even if accomplished instantaneously. In the assumed accident 0.21% would be added in one-quarter second and the remaining 1.1% in the next 36 seconds.

A second moderator change causing an addition of reactivity can occur as a result of a re-entrant nozzle failure. The subsequent filling of the port has the effect of removing reflector voids from the reactor and increases the reactivity. To minimize this hazard, all facilities contain sealed thimbles within the nozzle. A nozzle rupture will, therefore, allow D_2O to fill only the 3/16 inch annulus between the thimble O.D. and nozzle I.D. The removal of this small amount of reflector void will have a negligible effect upon reactivity. If, however both the nozzle and the thimble were to rupture, all of the reflector void associated with this beam port could be replaced with D_2O.

Normally, all the facilities are filled with graphite plugs when not in use. During this time a rupture of both nozzle and thimble would be little worse than rupture of the nozzle alone. However, it is expected that experimental

requirements will necessitate the removal of these plugs from time to time. A rupture of both the nozzle and thimble in this situation could produce a substantial reactivity effect. The replacement of void by D_2O has been measured and is 0.77% for a 6 inch diameter tangent-tube and about 0.34% for a 6 inch diameter beam port. Positive reactivity insertions of this magnitude are quite capable of being compensated for in an orderly way through temperature and void feedback mechanisms should period and overpower trips fail. They are well below the values for which fuel melting becomes a consideration.

8.5 Release of Radioactivity to Surrounding Area

Since relatively large amounts of fission products are associated with an operating reactor, the possibility of escape of these fission products must be considered. The succeeding sections discuss circumstances which could possibly result in release of fission products, and the precautions which have been taken to eliminate or alleviate this hazard.

8.5.1 Cladding Failure

The fuel element used in the GTRR is an aluminum-uranium alloy plate, clad with aluminum. Aluminum is highly resistant to corrosion by low-temperature, high-purity water. Consequently, the clad will, under normal circumstances, prevent any corrosion of the fuel bearing part of the plate. However, mechanical damage or corrosion of the cladding can result in exposure of the Al-U alloy with subsequent fuel corrosion and some fission product release to the D_2O. Experience has shown that with fuel of this type, the extent of the corrosion and release will be largely a function of the amount of fuel bearing plate surface which is exposed. The system contamination resulting from a cladding failure will be small and will present little impediment to reactor operations or hazard to the reactor and environs.

8.5.2 Melting of Fuel Plates

The preceding discussions of possible accidents have shown that the chance of fuel melting, while remote, is conceivable. It is arbitrarily postulated here that

fuel melting occurs with the attendant release of fission products and that these must be contained within the building housing the reactor. Additionally, should some nuclear excursion of unspecified cause and a metal-water reaction occur, the safe absorption of the released chemical energy and the protection of the surrounding populace from over-exposure to radiation define the criteria for the containment building design.

The energy which must be contained is contributed by three processes. The first is the energy which was absorbed by the coolant and moderator when raised from ambient conditions to the operating temperature of the reactor. The second is that resulting from the excursion which caused the fuel melt-down, and the third is the chemical energy liberated by the rapid oxidation of aluminum and subsequent oxidation of the D_2 released by the Al-D_2O reaction.

A calculation of the internal pressure in a 260,000 cubic foot building caused by this total energy liberation is given in Appendix A. The calculation is based upon the following assumptions:

a. Aluminum reacted (grams)	12,500
b. D_2O available for release (pounds)	10,000
c. D_2O temperature (°F)	134
d. Building air temperature (°F)	70
e. Relative humidity of building air (%)	50
f. Building air pressure (psia)	14.7
g. Volume of air in building (ft^3)	260,000
h. Reactor power level (MW)	5

It is further assumed that the reactor vessel ruptures, D_2O escapes to the building space, a deuterium explosion does not occur, and the nuclear excursion contribution is equivalent to 135 MW-sec. Although an excursion energy of this magnitude is not foreseeable from any accident postulated herein, this value has been arbitrarily chosen as a factor of conservatism in the building design. The calculations indicate the maximum internal pressure will be 2.11 psig at 109° F.

It is assumed that the melting of the fuel plates will release fission products to

the reactor vessel. It is further assumed that the excursion and water-metal reaction will rupture the reactor vessel and disperse these fission products throughout the interior of the building. The severity of the external radiation hazard is a function of the quantity and character of the released fission products.

MTR fuel plates have been slowly melted experimentally, and the fission product release determined. Under these circumstances 10% of the rare gases and 2% of the iodine were released. A negligible release of particulate products was observed. Fission product release studies have indicated the need for assuming greater release fractions (100% for Kr and Xe release, 50% for I release) and these values were used in calculating the source terms of Appendices B and C.

8.5 3 Reactor Containment

The GTRR is housed within a steel containment shell to provide maximum protection for the surrounding area against atmospheric radioactive contamination. Inside the steel containment vessel is a cylindrical concrete shadow shield 12 inches thick. As shown in Appendix C, this shield reduces the external radiation dose caused by the maximum possible quantity of contained fission products.

The containment shell is designed to withstand an internal pressure of 2 psig with a safety factor of three. The building leakage rate at 2 psig above atmospheric pressure has been shown to be less than 1/2% of the building volume per day at 2 psig. Continued leak-tightness tests are being made periodically.

In the event of a nuclear excursion, an observer downwind from a slow leak in the containment building would be exposed to radiation from an airborne fission cloud. Also, radioactivity could be inhaled. Those isotopes which were retained in the body would irradiate the tissue for an extended time. An estimate of the exposure from a fission cloud is presented in Appendix B.

8.5.4 Discharge of Gaseous Effluent

The major source of gaseous radioactive plant effluent is the air contained in the experimental facilities and other void spaces near the high flux region. The effluent system is designed to allow short half-lived activation products to decay

before being released to the environment and to filter out particulate matter as small as 0.3 microns in diameter with near 100% efficiency. Actual measurements have shown the design to be effective in eliminating short half-life isotopes and particulates from the effluent. Therefore, the only readily detectable radioactive constituent in the stack effluent is argon-41 and routine releases are always below 10CFR20 limits.

REFERENCES

8.1 Cole, T. E., and Cox, J. A., "Design and Operation of the ORR," Vol. 10, 86, Second Geneva Conference Report (1958).

8.2 Grund,J. E., Davis, T. H., and Johnson, R. L., Nuclear Start-Up of the Spert II Reactor with Heavy-Water Moderator, USAEC Report IDO-16762, Phillips Petroleum Co., April 20, 1962.

8.3 Graham W. W. III, et al., "Kinetics Parameters of a Highly Enriched Heavy-Water Reactor, Final Report", TID-23037, April, 1966.

8.4 Docket 50-160, USAEC June 30, 1966.

8.5 Grund, J E., Self-Limiting Excursion Tests of a Highly Enriched Plate-Type D_2O-Moderated Reactor. Part I. Initial Test Series, USAEC Report IDO-16891, Phillips Petroleum Co., July 12, 1963.

8.6 Dietrich, J. R. and Layman, D. C., Transient and Steady State Characteristics of a Boiling Reactor. The Borax Experiment 1953, USAEC Report AECD-3840, February, 1954.

8.7 Spano, A. H., and Miller, R. W., Spert I Destructive Test Program Safety Analysis Report, USAEC Report IDO-16790, Phillips Petroleum Co., June 15, 1962.

APPENDIX A

CALCULATION OF THE PRESSURE EFFECTS OF AN ALUMINUM-HEAVY WATER REACTION

A.1 Introduction

The possibilities and effects of water-metal reactions in conjunction with nuclear accidents have been extensively investigated. The results of these investigations have formed the basis for the assessment of the dangers associated with this phenomenon in a number of previously published reactor hazards summaries. Concurrently, semi-standardized calculational procedures have been developed to determine the pressure conditions for which containment buildings must be designed. The following calculation of the effects of an aluminum-heavy water reaction in the GTRR are predicated upon these same data, assumptions and mathematical procedures. It should be noted that this calculation is not related to any specific credible accident, but deals with an extreme limiting set of simultaneous conditions.

A.2 Definition of the Accident

It is postulated that the GTRR is operating at a power of 5 megawatts within a steel containment shell. An accident then occurs, which results in a reactor excursion and fuel melt-down. Subsequently, 25% of the aluminum in the fuel assemblies reacts with the D_2O. During the course of this accident the reactor vessel ruptures. The total energy available from the excursion, from stored energy in the reactor system and from the water-metal reaction contributes to pressurization of the containment building.

A.3 Assumptions

The reactor containment building becomes pressurized from warming of the existing air and the addition of D_2O vapor. The energy for these two effects is supplied by:

(1) The nuclear excursion whose energy release is assumed to be 135 megawatt-seconds.

(2) Chemical reaction of aluminum and D_2O.

(3) Chemical reaction of D_2 and O_2.

(4) Decrease in total sensible heat content of the liquid D_2O coolant.

The pressurization actually occurs very slowly because the major energy contribution is by (4) above and the D_2O is initially well below the saturation temperature. During this slow change some energy will definitely be lost through building walls to the atmosphere however, for simplicity and in order to maximize the end results, it is assumed that no energy leaves the building nor is any absorbed into structural members. The internal energy of the air and D_2O systems combined remains constant except for the addition of nuclear and chemical energy. A D_2 - O_2 explosion cannot occur because the total D_2 released is not sufficient to bring the building concentration to the minimum explosive limit.

The following initial conditions and data are assumed for the calculation:

a. Building air volume (ft^3)	260,000
b. Building air temperature (°F)	70
c. Building air relative humidity (1%)	50
d. Building pressure (psia)	14.7
e. D_2O available for release (lbs)	15,000
f. D_2O temperature (°F)	134
g. Aluminum in core (lbs)	110
h. Aluminum oxidized (lbs)	27.5
i. Excursion energy (MW-sec)	135

A.4 Calculations

A.4.1 Excursion Energy

The 135 MW-sec excursion produces 1.28×10^5BTU (Δh_1).

A.4.2 Al-D_2O Reaction

The aluminum and deuterium oxide combine chemically with 25% of the aluminum being consumed. The 25% value is based on Borax experiments. The

resultant production of 232 gm-moles of Al_2O_3 releases 1.73×10^5 BTU (Δh_2) to the D_2O.

A.4.3 D_2 - O2 Reaction

Three moles of D_2 are formed for each mole of Al_2O_3. The resultant 696 gm-moles of D_2 react slowly with building air, thereby producing 1.94×10^5 BTU (Δh_3).

A.4.4 Energy (Sensible Heat) Released from D_2O

As mentioned previously, the D_2O is well below the saturation temperature. However, the D_2O system was contained prior to the postulated accident and, therefore, was not in temperature equilibrium with the building air. After rupture of the piping or the core vessel the bulk D_2O will cool off, losing heat to the surroundings until equilibrium is established. The amount of energy released from the D_2O is not known until the equilibrium equation is solved by trial and error to yield the equilibrium temperature.

A.4.5 Equilibrium Pressure and Temperature

The equilibrium equation is:

$$\Delta h_1 + \Delta h_2 + \Delta h_3 + WC_p (T_o - T_q) + \frac{V}{V_h} E_h =$$
$$\frac{V}{V_a} C_a (T_q - T) + \frac{V}{V_h} C_h (T_q - T) + \frac{V}{V_d} E_d$$

where

C_a = Specific heat of air (constant volume), BTU/lb°F.

C_h = Specific heat of H_2O vapor (constant volume), BTU/lb°F.

C_p = Specific heat of liquid D_2O, BTU/lb°F

E_d = Latent heat of vaporization of D_2O at T_q, BTU/lb.

E_h = Latent heat of vaporization of H_2O at T_q, BTU/lb.

P = Initial total pressure, psi.

P_a = Initial partial pressure of air, psi.

P_d = Partial pressure of D_2O vapor at T_q, psi.

P_h = Initial partial pressure of H_2O, psi.

T = Initial temperature of building air, °F.

T_o = Initial D_2O temperature, °F.

T_q = Final equilibrium temperature, °F.

V = Building free volume ft^3

V_a = Specific volume of air at T and P_a, ft^3/lb.

V_h = Specific volume of H_2O vapor at T and P_h, ft^3/lb.

V_d = Specific volume of D_2O vapor at T_q and P_d, ft^3/lb.

W = Weight of D_2O released, lb.

In accordance with the vapor-liquid relationships for the H_2O - HDO - D_2O system, the D/H atom ratio in the vapor will approach (within about five percent) the D/H atom ratio in the liquid. The total H_2O available in the air is small compared with the D_2O released; therefore, practically all of the H_2O must condense in order to attain equilibrium. In the development of the equilibrium equation, the H_2O vapor initially present is assumed to condense in the liquid D_2O.

Thus, in the energy balance the energy released consists of the excursion energy Δh_1, the Al-D_2O reaction energy Δh_2, the D_2-O_2 reaction energy Δh_3, the sensible heat given up by the liquid D_2O while cooling from T_o to T_q, and the latent heat given up by the condensing H_2O vapor at T_q. This total released energy is absorbed by increasing the initial air and H_2O vapor temperatures from T to T_q and by vaporizing liquid D_2O at T_q in an amount sufficient to establish the equilibrium vapor pressure.

The equilibrium temperature T_q was computed to be 109°F. At this temperature, the final pressure is estimated as follows:

$$\text{partial pressure of air} = P_a \times \frac{460+T_q}{460+T} = (14.52)\left(\frac{569}{530}\right) = 15.565 \text{ psia}$$

$$\text{partial pressure of } D_2O \text{ at } T_q = 1.10 \text{ psia}$$

The total pressure is 16.665 psia or 1.965 psig.

In further considering the maximum pressure, it is entirely possible that the rate of approach to equilibrium may be different for different portions of the overall process. In order to allow for the fact that the H_2O may condense slowly, but the D_2O may evaporate quickly, it is assumed that the H_2O may exert a large percentage of its initial partial pressure during the pressure transient. For present purposes, 75% is arbitrarily selected as a reasonable allowance for H_2O vapor.

Thus,
$$0.75P_h \times \frac{460+T_q}{460+T} = (0.75)(0.18)(\frac{569}{530}) = 0.145 \text{ psia}$$

The maximum pressure is now estimated to be:

$$1.965 + 0.145 = 2.11 \text{ psig}$$

Upon arriving at this estimate of pressure, it is used as a conservative value for purposes of establishing design criteria. Energy absorbed by the building or contents will obviously reduce the equilibrium temperature T_q and, in turn, the maximum pressure.

APPENDIX B

CALCULATION OF RADIATION DOSES RESULTING FROM THE RELEASE OF FISSION PRODUCTS INTO THE ATMOSPHERE

B.1 General

Estimates have been made of the radiation dosages which would be received by persons outside the reactor containment building should there be a release of fission products into the building and leakage of the building air to the outside. The radiation exposures considered here are those which would result from passage of the airborne cloud of radioactive contaminants over the ground. These include the external beta and gamma radiation exposures and the internal exposure of critical body organs; the iodine dose to the thyroid and the strontium dose to the bones.[B9] Values as defined in Title 10, Code of Federal Regulation, Part 100, for an exclusion area, a low population zone, and a population center distance are included.

The radiation exposure received by a person standing at a given distance from the reactor building depends on such factors as: inventory of fission products stored within the core at the time of release, fraction of the core fission products escaping into the building air, building out leakage rate, and atmospheric dispersion properties. Hence, in the analysis, certain basic assumptions are required concerning the release of the fission products, the atmospheric conditions, and the tightness of the building at the time of release. The results obtained are based on assumptions which, except for the arbitrary one that a release has occurred, are considered conservative for the reactor and building design. The calculations are described in sufficient detail to permit additional computations based on other assumptions to be made.

The material contained herein is divided into two sections. The first section describes the model for the release and spread of radioactivity and gives the necessary references and formulas used in calculating the radiation dose. The second section presents the results obtained with the assumed model.

B.2 Method and Assumptions Used in Dose Calculations

Although such an event is not considered likely because of the limitations on available reactivity and because of the inherent self limiting characteristics of the reactor, it is postulated that an accident has occurred in which melting of the fuel plates has occurred. The reactor is assumed to have been operating continuously at the five megawatt power level long enough to have attained equilibrium concentrations of the fission products, i.e., the iodine, bromine, xenon, and krypton isotopes. Previous tests involving slow melting of MTR fuel plates have shown that ten percent of the xenon and krypton isotopes and two percent of the iodine isotopes were released.[B.1] However, later values,[B.2] comfortably conservative for the relatively low power GTRR have been used including a value of 50 percent iodine release to the containment building, with 25 percent of the total iodine inventory (shown later to be the crucial component) being released to the outside environment. It is further assumed that the released isotopes escape the reactor biological shield and are mixed with the air in the containment building. The equilibrium activities of the various isotopes released into the reactor containment building were obtained from the literature.[B.3,B.4]

For the sake of arriving at an upper limit to the dose, the equilibrium activity is considered to be constant for two hours after release. Because of this assumption, any volatile fission products with less than a one minute half-life were omitted from the source term. Included in the source term are daughter products of the volatile fission products, notably Sr-89 and Sr-90 which are produced and reach their maximum activity outside the reactor.

The concentration of fission product activity in the atmosphere outside the reactor building and the resultant radiation exposure will depend on the wind direction and velocity, degree of atmospheric turbulence, and the building leakage rate. The highest dose is obtained when the person exposed is directly downwind from the leak. The method of computation is based on O. G. Sutton's formula and utilizes the standard equations and curves.[B.5] For continuous ground level release of fission product, the formula for the centerline concentration reduces to the

following: $$\zeta = \frac{2Q}{3600\ \pi\ C^2\ \bar{\mu}\ x^2} \qquad (1)$$

where

ζ = Concentration of activity, curies per cubic meter of air

Q = Continuous source strength, i.e , building out leakage in curies per hour

x = Distance downwind from source, meters

$\bar{\mu}$ = Mean wind speed, meters per second

C = Generalized diffusion coefficient, meters $^{n/2}$

n = Dimensionless parameter associated with atmospheric stability

The following previously estimated values of the diffusion parameters for two different atmospheric conditions are used to calculate the concentration of activity, ζ , for a specified leakage rate at various distances, x, from the leakage source.

Atmospheric Conditions	n	C^2	$\bar{\mu}$
Severe Inversion	0.50	0.008	1
Mild Lapse	0.25	0.024	3

The external beta dose resulting from immersion in the radioactive fission product cloud is obtained from the following equation:

$$D_\beta = (5.26 \times 10^2)\ \zeta\ E \qquad (2)$$

Where ζ is the concentration of activity in curies per cubic meter of air obtained from Equation (1), D_β is the external beta dose in rads received in two hours after the fission product release, and E is the effective beta energy in MeV per disintegration. The derivation of Equation (2) follows that in Reference B.5, page B-7. It is based on the assumption that, in an infinite medium in equilibrium, as much energy is absorbed in each unit volume as is released in it. Because of the limited range of beta particles in tissue, D_β should be considered a surface or skin dose. The depth dose would be considerably less. The estimate of D_β is quite conservative since the value of ζ is based on the equilibrium fission product inventory at the moment of release. This inventory is considered constant during the two hour duration of exposure while, in fact, it would be decreasing continually.

The internal dosage to critical organs is calculated in the following manner. The activity, A, deposited in the critical organs is given by:

$$A = J F \zeta t \qquad (3)$$

where

A = Activity deposited in organ, curies

J = Inhalation rate, cubic meters per hour

F = Fraction of inhaled activity deposited in critical organ

t = Duration of exposure, hours

The corresponding initial dose rate to the critical organ for a person who has been immersed in the fission product cloud is given by the expression:

$$D = (5.1 \times 10^4) \frac{AE}{W} \qquad (4)$$

where

D = Initial dose rate, rad per day

W = Weight of critical organ, kilograms

E = Effective energy of radiation, MeV per disintegration

The total integrated dose to the critical organ is related to the initial dose rate by the equation:

$$TID = 1.44\ D\ T \qquad (5)$$

TID = Total integrated dose, rads

T = Effective half-life of the radioisotope, days

The values of J, F, E, W, and T appearing in Equations (3), (4), and (5) may be obtained from Reference B. 6 for the various radioisotopes and critical organs involved. References B. 2 and 3. 7 gives additional information on the various iodine isotopes.

It is possible to calculate the gamma dose resulting from immersion in a radioactive cloud by a method which is analogous to that used to estimate the beta dose. However, another method is available in the form of the J. Z. Holland monogram given in Reference B.5, page B-7. The monogram gives the gamma

dosage resulting from sudden discharge into the atmosphere of the contents of a nuclear reactor which has been operating at a steady power level. The dosage read from the monogram must first be corrected to account for the fact that not all of the fission products are released. Also, since the activity is not immediately released into the atmosphere outside the building, but leaks out of the building at a finite rate, the dosage obtained by use of the monogram must be corrected accordingly. The resulting equation for determination of the dose is as follows:

$$D = D_n \; R \, t \, M \qquad (6)$$

where

D_n = Dose read from nomogram, roentgens

R = Fraction of gross gamma activity released into containment building

t = Duration of exposure, hours

M = Leakage rate of building, fraction of building volume per hour

B.3 Results of Radiation Exposure Calculations

The results for a two hour exposure immediately after fission product release are given below in Table B.I. The estimated beta and gamma doses are received only while the subject is immersed in the cloud. The estimated thyroid and bone doses are accumulated over the subject's lifetime following the two hour immersion. Data for four different distances downwind of the point of release and under two different atmospheric conditions are presented. The building leakage rate for all calculations was assumed to be 0.5 percent of the building volume per day. This is a reasonable number since tests have shown actual leak rate at 2 psi to be less than 0.5 percent per day.

Table B.1

Radiation Dose from a Two Hour Exposure to a Fission Product Cloud at Various Distances Downwind from Reactor Building

Severe Inversion

Distance, x (meters)	External Beta Dose (rads)	External Gamma Dose (rads)	Lifetime Thyroid Dose (rads)
50	8.13	0.25	285.71
100	0.30	0.13	10.36
200	0.11	--	3.57

Mild Lapse

Distance, x (meters)	External Beta Dose (rads)	External Gamma Dose (rads)	Lifetime Thyroid Dose (rads)
50	0.10	--	42.50
100	0.04	--	12.50
200	--	--	3.75

From the above table it can be seen that the controlling dose from an accidental release of fission products to the atmosphere is from the inhalation of iodine to the thyroid. It is also evident that the worst case for release would be during a stable or inversion meteorological condition. The direct and scattered radiation dose as calculated in Appendix C is less controlling than the thyroid dose for determining an exclusion area. If the meteorological parameters Cy, Cz, $\overline{u}$, and n that were used in TID-14844 are assumed to be representative for this facility during stable conditions, and a building leakage rate of 0.5 percent per day rather than 0.1 percent/day is used, the following results can be calculated:[B.2]

	Radius from GTRR Reactor Building		Dose
	Meters	Feet	
An exclusion area [B.8]	50	164	300 rem to thyroid in 2 hours
A low population zone	500	1,640	300 rem to thyroid for infinite time
A population center distance	665	2,182	1 1/3 times the distance from reactor to outer boundary of low population zone

REFERENCES

B.1 Parker, G. W., and Creek, G. E., "Experiments on the Release of Fission Products from Molten Fuels," in Reactor Safety Conference held at New York City, October 31, 1957, Report No. TID-7549 (Pt. 2).

B.2 TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," Div. of Licensing and Regulations, AEC, March 23, 1962.

B.3 Foller, I. L., Chapman, T. S., and West, J. M., "Calculations on U-235 Fission Product Decay Chains," Argonne National Laboratory, Report No. ANL-4807, May, 1952.

B.4 Blomeke, J. O., "Nuclear Properties of U-235 Fission Products," Oak Ridge National Laboratory, Report No. ORNL-1783, September 30,

B.5 "Meteorology and Atomic Energy," U. S. Weather Bureau, July, 1955.

B.6 "Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water," National Bureau of Standards, Handbook 52, March 20, 1953.

B.7 Dunning, G. M., "Thyroid Dose from Radioiodine in Fallout,"Nucleonics, Vol. 14, February, 1956, p. 40.

B.8 Reacto Site Criteria, Rules and Regulations, Title 10, Part 100, AEC.

B.9 W. L. Woodruff, D. K. Warinner, and J. E. Motos, "A Radiological Consequence Analysis with HEU and LEU Fuels," proc. 1984 International Meeting on Reduced Enrichment for Research and Test Reactors, Argonne National Laboratory. Argonne, IL. Oct. 15-18, 1984, ANL/RERTR/TM -6, Conf. 8410173, pp. 472-490 (July 1985).

APPENDIX C

CALCULATION OF RADIATION DOSES FROM GTRR CONTAINMENT BUILDING FITTED WITH VOLATILE FISSION PRODUCTS

C.1 Description of Postulated Accident

The postulated accident involves a total loss of coolant with no credit from the emergency coolant supply. Under this condition, it is conceivable that melting of some, or all, of the reactor fuel elements would take place, resulting in the release of fission product gases into the containment vessel. In the first two hours following the postulated accident, persons in the vicinity of the reactor building would receive a radiation dose due to fission product gases trapped inside the containment vessel. Analyses of this type of accident have shown that the four gases of major importance are iodine, bromine, krypton, and xenon .[C.1] A conservative estimate of the release fractions, 50% halogen release and 100% release of the noble gases, was used in the calculations. The release would consist of six iodine isotopes of importance, two isotopes of bromine, five isotopes of krypton, and seven isotopes of xenon (See Table C.1). These isotopes have half-lives ranging from several seconds to several days and emit gammas with energies up to 4 MeV (Table C.2).

Table C.1

Assumed Release of Isotopes Following Core Burnout

$$S_{av} = S_0 \left(\frac{T_{1/2}}{1.385}\right) \left(1 - e^{-\frac{1.385}{T_{1/2}}}\right)$$

S_0 = The initial source strength in disintegration per second of the particular isotope

S_{av} = The average source strength of the particular isotope over a two hour period following the accident.

Isotope	S_0 (dps)	Half Life (hrs)	S_{av} (dps)
I 131	$2.31*10^{15}$	193.0	$2.3*10^{15}$
132	3.52	2.4	2.67
133	5.2	20.8	5.05
134	6.09	0.875	3.05
135	4.37	6.68	3.96
136	2.42	0.024	0.042
Kr 83m	$0.86*10^{15}$	1.9	$0.61*10^{15}$
85m	1.55	4.36	1.33
87	4.18	1.3	2.57
88	5.9	2.77	4.64
89	5.8	0.053	0.22
Br 83	$0.38*10^{15}$	2.3	$0.285*10^{15}$
84	0.56	0.5	0.19
Xe 131m	$0.0451*10^{15}$	288	$0.0451*10^{15}$
133m	0.256	55	0.25
133	10.3	127	10.2
135m	2.47	0.26	0.462
135	9.25	9.13	8.6
137	9.54	0.065	0.448
138	10.1	0.283	2.04

Table C.2

Important Gamma Emission from I, Kr, Br, and Xe

Where, S(E) = Average source strngth over a two hour period for gammas of energy E

Isotope S_{av} (dps)	Gamma Energy (Mev)	Relative Intensity (%)	S(E) (dps)
I 131 $2.3*10^{15}$	0.08	2.2	$50.8*10^{12}$
	0.163	5.3	122
	0.284	5.3	122
	0.364	80	1850
	0.637	9	208
	0.722	3	69
I 132 $2.67*10^{15}$	0.53	25	667
	0.62	6	160
	0.67	94	2510
	0.78	80	2135
	0.96	20	534
	1.16	8	213
	1.40	11	294
	1.96	5	133
	2.20	2	53
I 133 $5050*10^{12}$	0.53	94	4750
	0.85	5	253
	1.40	1	51
I 134 $3050*10^{12}$	0.418	11	336
	0.534	16	488
	0.62	25	763
	0.84	100	3050
	0.894	84	2560
	1.07	27	824
	1.14	21	640
	1.25	14	427
	1.43	6	183
	1.52	4	122
	1.61	8	244
	1.78	9	274
I 135 $3960*10^{12}$	0.42	6.9	273
	0.86	10.7	424
	1.04	9.1	360
	1.14	37	1465
	1.28	34	1345
	1.46	11.5	455
	1.72	19	752
	1.80	11.4	452

Table C.2 (continued)

Important Gamma Emission from I, Kr, Br, and Xe

Isotope S_{av} (dps)	Gamma Energy (Mev)	Relative Intensity (%)	S(E) (dps)
I 136 $42 * 10^{12}$	0.20	13	$5.5 * 10^{12}$
	0.27	19	8
	0.39	20	8.4
	0.46	4	1.7
	0.75	3	1.3
	1.00	6	2.5
	1.32	100	42
	1.55	4	1.7
	1.72	2	0.8
	1.89	5	2.1
	2.25	7	2.9
	2.40	13	5.5
	2.63	11	4.6
	2.84	8	3.4
	3.18	5	2.1
Br 83 $285 * 10^{12}$	0.046	20	57
Br 84 $190 * 10^{12}$	0.879	42	80
	1.01	8.2	16
	1.90	15	29
	2.47	7.4	14
	3.03	5	10
	3.28	2.4	5
	3.93	10.3	19
Xe 131^m $45 * 10^{12}$	0.163	5	2
Xe 133^m $250 * 10^{12}$	0.2328	16	40
Xe 133 $10200 * 10^{12}$	0.081	36	3670
Xe 135^m $462 * 10^{12}$	0.53	86	397
Xe 135 $8600 * 10^{12}$	0.25	97	8340
	0.61	3	258
Xe 137 $448 * 10^{12}$	0.90	3.3	15
	1.80	66	296
Xe 138 $2040 * 10^{12}$	0.42	100	$2040 * 10^{12}$
	0.51	20	408
	2.01	52	1060

Table C.2 (continued)

Important Gamma Emission from I, Kr, Br, and Xe

Isotope S_{av} (dps)	Gamma Energy (Mev)	Relative Intensity (%)	S(E) (dps)
Kr 83m $610*10^{12}$	0.009	9	55
Kr 85m $1330*10^{12}$	0.1495 0.3050	77 15.7	1020 209
Kr 87 $2570*10^{12}$	0.403 0.85 2.05 2.57	92 8 3.1 22	2370 206 80 565
Kr 88 $4640*10^{12}$	0.163 0.191 0.363 0.85 1.20 1.55 1.85 2.19 2.40	7 35 5 23 4 14 14 18 35	325 1620 232 1065 186 650 650 835 1620
Kr 89 $220*10^{12}$	1.70 3.70	65 35	143 77

C.2 Shielding by the Reactor Containment Building

The reactor containment building is basically a cylindrical, 7/16-inch thick steel tank with an outside diameter of 82 feet. The steel bottom is flat, while the top is a spherical dome rising to approximately 50 feet above the ground level. Inside the steel tank wall is a 12-inch thick layer of concrete which extends to about 34 feet above the outside ground level. A 6-inch layer of concrete extends upward another 6 feet. In the event of the postulated accident, the concrete wall would serve as a shadow shield to protect persons' outside the containment building from the radiation within. The roof of the building, which consists primarily of a 5/8-inch thick steel plate, would provide very little shielding.

C.3 Calculation of the Dose

The total dose received by an observer outside the building at ground level would result from direct beam radiation and air-scattered radiation. Because of the anisotropic nature of the shield provided by the containment building, the air-scattered dose is a significant part of the total dose. The dose resulting from the air-scattered radiation and the dose resulting from the radiation which traversed the shield were calculated separately. These calculations, briefly described here, are discussed in detail in Reference C.3.

The air-scattered dose was calculated using single scattering theory. The source is assumed to lie on the vertical building centerline, 25 feet above the first floor level. This is slightly above the center of the building volume. For purposes of calculating the air-scattered dose, the containment vessel was considered to be a cylinder with an open roof. The dose was evaluated using the 20 gamma energy groups shown in Table C.3. The resulting calculation showed that the air-scattered dose in the first two hours after the accident would be about four roentgens at the site boundary and would gradually decrease as one moves away from the containment vessel.

Table C.3

Total Average Gamma Spectrum From I, Br, Kr, Xe

Energy (Mev)	Range (Mev)	S(E) ($*10^{12}$ Gamma/Sec)	S(E) ($*10^{12}$ Mev/sec)
0.1	0.04 - 0.15	4853	485
0.2	0.15 - 0.25	2115	423
0.3	0.25 - 0.35	8679	2604
0.4	0.35 - 0.45	7101	2840
0.5	0.45 - 0.55	6712	3356
0.6	0.55 - 0.65	1389	833
0.7	0.65 - 0.75	2579	1805
0.8	0.75 - 0.85	5186	4149
0.9	0.85 - 0.95	4603	4143
1.0	0.95 - 1.05	913	913
1.2	1.05 - 1.30	5100	6120
1.4	1.30 - 1.50	1025	1435
1.6	1.50 - 1.70	1018	1629
1.8	1.70 - 1.90	2570	4626
2.0	1.90 - 2.10	1302	2604
2.2	2.10 - 2.30	891	1960
2.5	2.30 - 2.75	2210	5525
3.0	2.75 - 3.25	16	48
3.5	3.25 - 3.75	82	287
4.0	3.93	19	76

The direct dose was calculated assuming a homogeneous mixture of fission products inside the containment vessel, except that no diffusion into the basement level was assumed. Since the basement is a large, well shielded volume with an open stairwell entrance, a factor of conservatism is introduced by this assumption. The direct dose at any point outside the containment vessel can then be considered to consist of two components. The first component would be the dose resulting from the radiation which traversed the one-foot concrete shield and the 7/16-inch steel plate. The second component would result from the radiation from the fission products near the top of the containment vessel, which needed only to travel through the 5/8-inch thick steel roof of the containment vessel in order to reach the given point outside.

The fraction of the total radiation considered to traverse the nearly unshielded top was equal to the fraction of the total volume "seen" as unshielded. The volume "seen" as unshielded, of course, depended on the position of the point outside. When one is standing close to the containment building at ground level, the top cannot be seen, so the component of the radiation from the top cannot be seen, so the component of the radiation from the top would be zero, whereas at a point 200 feet away and 90 feet high, a large portion of the radiation which one would receive would be coming through the nearly unshielded top. Since the contribution to the dose from the radiation near the top is only important at significant distances from the containment building, the point source approximation was used for this component. For the component of the radiation passing through the cylindrical side of the containment of the radiation passing through the cylindrical side of the containment building, a distributed source was used made up of 1156 point sources, uniformly distributed throughout the volume of the cylinder. At a given point outside the containment building, then, the dose of this component was evaluated for each source point using 20 gamma energy groups. The computer results are summarized in Figure C.1, and the worst-case curves are shown in Figure C.2.

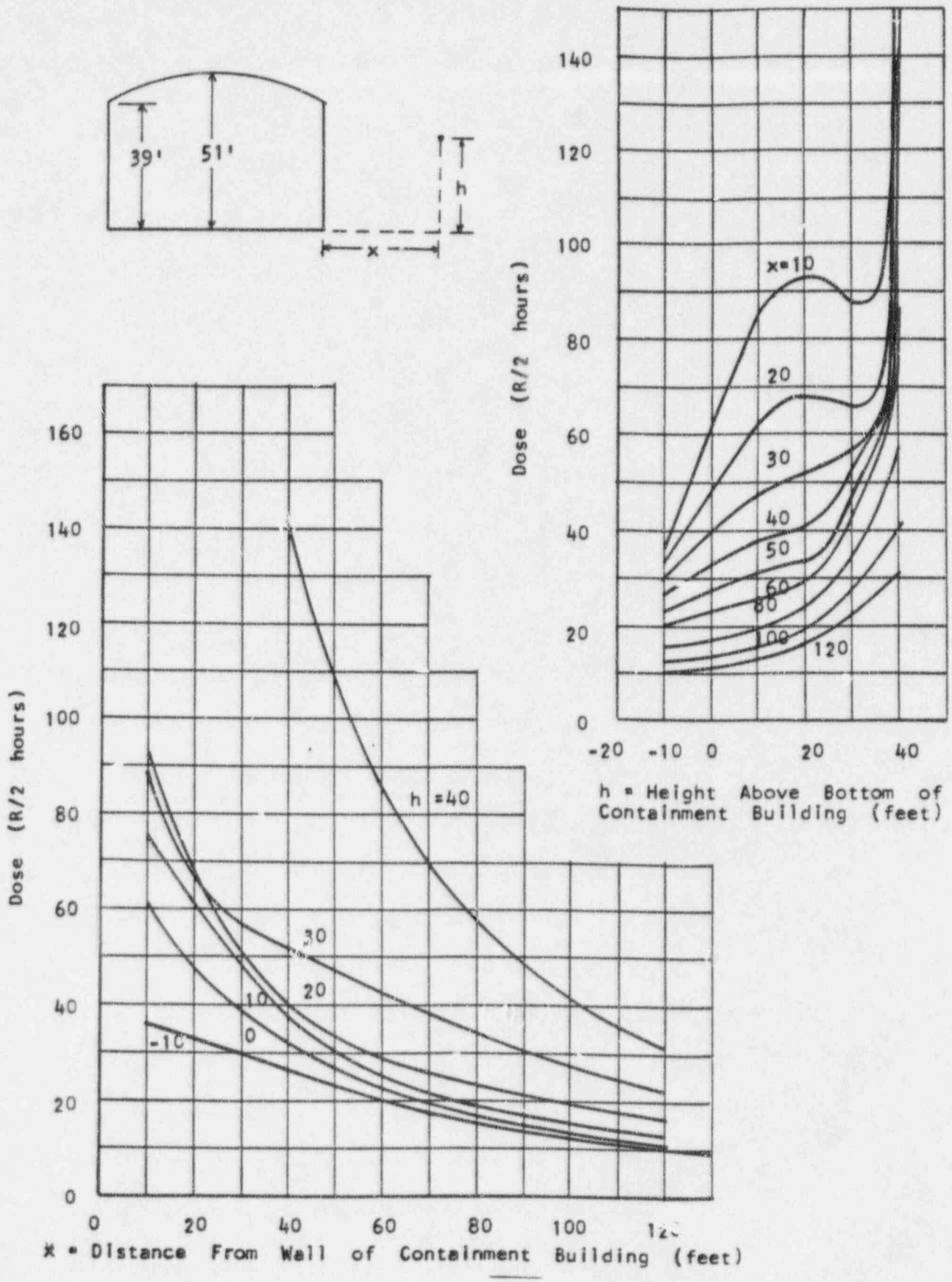
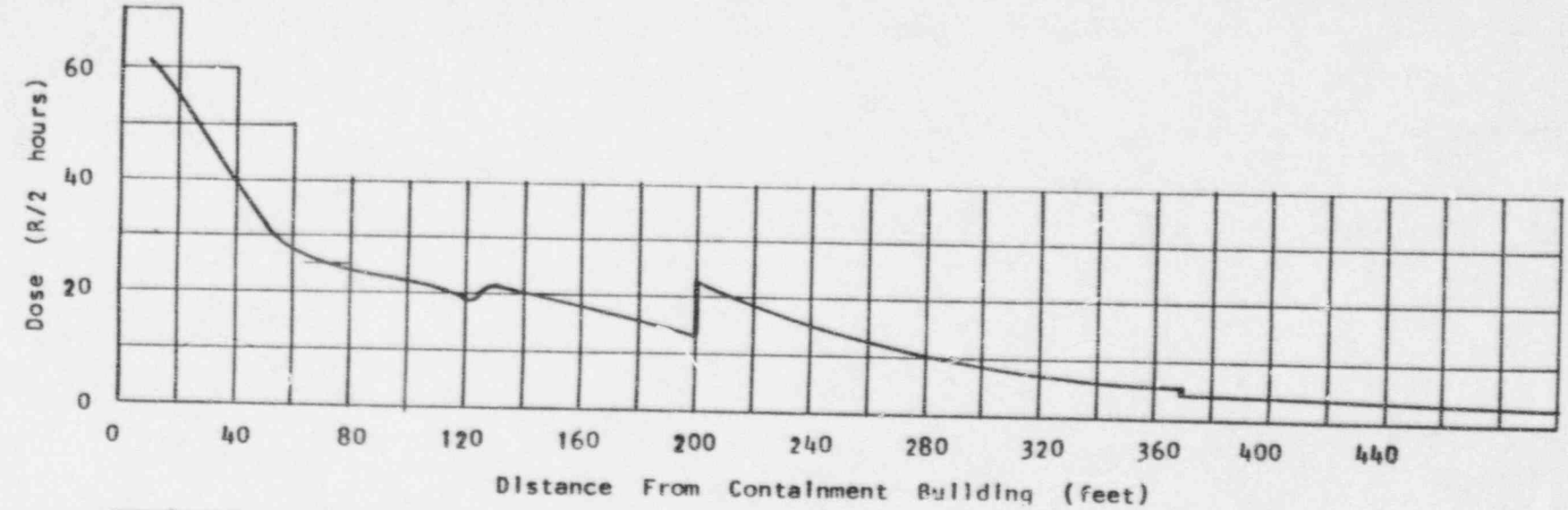


Figure C.1. 2-Hour Dose Near GTRR Containment Building after Accident.



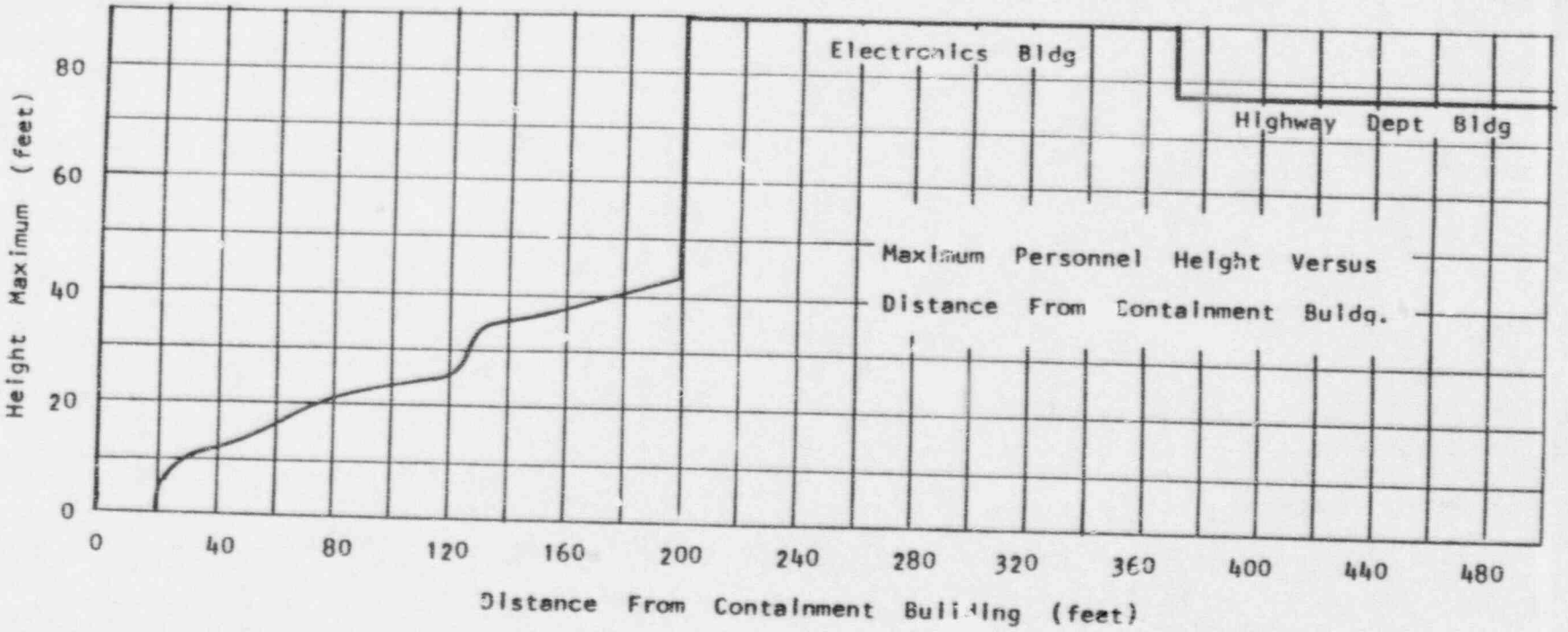


Figure C.2. Worst-Case 2-Hour Dose vs. Distance from Containment Building.

C.4. Conclusions

The worst-case curves show that the minimum exclusion area boundary is determined, not by the accident postulated in this appendix, but rather by the fission product release accident discussed Appendix B. Thus, the existing shielding appears to be adequate for 5 MW operation particularly in view of the conservative assumptions used.

REFERENCES

C.1 DiNunno, J. J.; Anderson, F. D ; Baker, R. E.; Waterfield, R. L., "Calculation of Distance Factors for Power and Test Reactor Sites", Division of Licensing and Regulation, USAEC, March 23, 1962 (TID-14844).

C.2 Creek, G. E.; Martin, W. J.; Parker, G. W., "Experiments on the Release of Fission Products from Molten Reactor Fuels," Oak Ridge National Laboratory, July 22, 1959 (ORNL-2616).

C.3 Williams, J. R., "Calculation of the Dose in the Vicinity of the Georgia Tech Research Reactor Facility Following a Postulated Maximum Credible Accident," Georgia Tech Nuclear Engineering Series, Technical Report No. GT-NE-8, November, 1967.